Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards APR1400 Subcommittee Meeting

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Tuesday, June 5, 2018

Work Order No.: NRC-3760 Pages 1-457

NEAL R. GROSS AND CO., INC.
Court Reporters and Transcribers
1323 Rhode Island Avenue, N.W.
Washington, D.C. 20005
(202) 234-4433

1
2
3
4 DISCLAIMER
5
6
7 UNITED STATES NUCLEAR REGULATORY COMMISSION'S
8 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
9
10
11 The contents of this transcript of the
12 proceeding of the United States Nuclear Regulatory
13 Commission Advisory Committee on Reactor Safeguards,
14 as reported herein, is a record of the discussions
15 recorded at the meeting.
16
17 This transcript has not been reviewed,
18 corrected, and edited, and it may contain
19 inaccuracies.
20
21
22
23

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

+ + + + +

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

APR1400 SUBCOMMITTEE

+ + + + +

TUESDAY

JUNE 5, 2018

+ + + + +

ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

RONALD G. BALLINGER, Chairman

DENNIS C. BLEY, Member

WALTER L. KIRCHNER, Member

JOSE MARCH-LEUBA, Member

JOY L. REMPE, Member

PETER C. RICCARDELLA, Member

GORDON R. SKILLMAN, Member

MATTHEW SUNSERI, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIALS:

CHRISTOPHER BROWN

CHRISTIANA LUI

ALSO PRESENT:

TONY AHN, KHNP

ROSS ANDERSON, ENERCON

AARON ARMSTRONG, NRO

AYO AYEGBUSI, NRO

KEVIN COYNE, NRO

GARY DOUGLAS, Westinghouse

ANNE-MARIE GRADY, NRO

MICHELLE HAYES, NRO

YONG IN, KHNP

TERRY JACKSON, NRO

KERRI KAVANAGH, NRO

BYUNG JO KIM, KEPCO E&C

JAEGAB KIM, KEPCO E&C

JUNGHO KIM, KHNP

YOUNGKI KIM, KEPCO E&C

YUN HO KIM, KHNP

JEFFREY LEARY, ENERCON

DONGWON LEE, KHNP

DAEHAWN LIM, KEPCO E&C

KWANGIL LIM, KEPCO E&C

MARK LINTZ, NRO

JILL MONAHAN, Westinghouse

HO RIM MOON, KHNP

TONY NAKANISHI, NRO

JIYONG (ANDY) OH, KHNP

CHAN Y. PAIK, FAI

HANH PHAN, NRO

MARIE POHIDA, NRO

ROBERT ROCHE-RIVERA, NRO

GREG ROZGA, ENERCON

ROB SISK, Westinghouse

JAMES STECKEL, NRO

COURTNEY ST. PETERS, NRR

ROBERT SWEENEY, KHNP/AECOM

BOYCE TRAVIS, NRO

ANDREA D. VEIL, Executive Director, ACRS

HANRY WAGAGE, NRO

DAVE WAGNER, KHNP

WILLIAM WARD, NRO

AGENDA

Opening Remarks and Objectives 5

Opening Remarks by NRC Staff and KHNP 8

Chapter 19.1 PRA 9

Chapter 17 - Quality Assurance and

Reliability Assurance 209

Chapter 19.2 - Severe Accident Evaluation . . . 240

Public Comments 278

Member Comments 278

Closing Remarks and Adjourn 281

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

(8:30 a.m.)

CHAIR BALLINGER: The meeting will now come to order. This is a meeting of the APR1400 Subcommittee of the Advisory Committee on Reactor Safeguards.

I'm Ron Ballinger, Chairman, Co-Chairman, of the APR1400 Subcommittee. ACRS Members in attendance are Dennis Bley, Dick Skillman, Matt Sunseri, Pete Riccardella, Jose March-Leuba.

And I, I don't know. We also have our consultant, Steve Schultz, with us, thank you very much.

Chris Brown, who is down at the end, he's the best dressed guy in here, and Christiana Lui --

(Off-microphone comment.)

CHAIRMAN BALLINGER: -- oh, hiding back there, are the designated federal officials for this meeting.

The purpose of today's meeting, and ultimate meeting I might add, is for the Subcommittee to receive information briefing from the Korean Electric Power Corporation, Korea Hydro & Nuclear Power, Westinghouse, ENERCON and other KHNP contractors.

And NRC Staff regarding the design certification application for the APR1400. Mainly the PRA DCD Chapter and SAR for Chapters 19.1 and 19.2, along with DCD Chapter and SAR for Chapter 17 QA and Reliability Assurance.

The ACRS was established by statute and is govern by the Federal Advisory Committee Act, FACA. That means that the Committee can only speak through its published letter reports.

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

It is a little bit difficult this time because the FRN for this meeting was published this morning. So, you got to be quick.

The said, we also set aside ten minutes for spur of the moment comments from members of the public attending or listening to our meetings. Written comments are also welcome.

The ACRS section of the U.S. NRC public website provides our charter, bylaws, letter reports and full transcripts of all full and subcommittee meetings. Including slides presented at the meetings.

The rules for participation in today's meeting were announced in the federal register on Tuesday June 5th, which would be today. The meeting was announced as an open/closed to public meeting, although the agenda is all open except at the very end, unless we need it.

This meant that the Chairman can close the meeting as needed to protect proprietary information to KHNP or its vendors. No request for making a statement to this Subcommittee has been received from the public.

A transcript of the meeting is being kept and will be made available as stated in the federal register notice. Therefore, we request that participants in this meeting use the microphones located throughout the meeting room when addressing the Subcommittee.

And we all have to say this, there is a little green light on the top of your microphone, if you want to talk, make sure that little green light is shining, it's on. Which means you push a button at the base of that little thing.

Participants should first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

We have a bridge line established for interested members of the public to listen in. To minimize disturbance, the public line will be kept in a listen only mode. The public will have an opportunity to make statements or provide comments at a designated time towards the end of the meeting.

I request that the meeting attendees and participants silence their cell phones and other electronic devices please.

We now invite Bill Ward, NRO senior project manager for this project, to introduce the presenters and start the briefing. Bill?

MR. WARD: Thank you. We're very happy to be here again. I know a lot of work has gone into preparing and updating the PRA on KHNP side and executing the review on the NRC side.

And I'd like to say in all caps that we're really happy to be finally in this meeting.

CHAIRMAN BALLINGER: Okay, so the floor is --

(Off-microphone comment.)

MR. SISK: Thank you, Chairman. This is Rob Sisk, Westinghouse. We just wanted to thank the Committee for the opportunity to present Chapter 17 and 19.

As I think was noted, we have a lot to cover today, so without any further ado, I'll introduce Mr. Yong In to lead us through Chapter 19.

CHAIRMAN BALLINGER: I mean, I'll chime in as well. This is, I think, the penultimate meeting for this Subcommittee. Our intention is to have a meeting in July at which point we would write the final letter.

So there's a lot of work that's been gone into this so we appreciate it. Thank you.

MR. IN: My name is Yong In. I'm the project manager for the PRA Chapter 19.1. And I will be introducing the Chapter 19.1 presentation this morning.

Next slide. There are six presenters covering the operation and topics as shown in this slide. I won't introduce each one but there are three presenters will be presenting their material this session and then the second session will be presented by three other gentlemen.

Next one. Basically, for the overview of Chapter 19.1, there is two main documents. The first document is DCD Chapter 19.1. Which includes basically the PRA information.

And then there are some inputs in Tier 1

document, but it wasn't really made part of the PRA documentation. The PRA documentation is in 19.1.

Here is a brief update history of the APR1400 PRA. Initially we had the PRA Revision 1 and the, Revision 0 and the Revision 1.

Revision 0 was initially completed in September of 2013 and it was updated in, for the low power shutdown portion. It was updated September 2014.

And then we had the, what we call Revision 0A, which was basically conversion to CAFTA software. And then Revision 1 was completed in the September 2017, last year. And actually that was completed, PRA based SMA was completed March of this year.

All PRA documentation were completed to support the DCD Rev. 3. Which is pending our software.

And with that, I'll hand it over to Mr. Greg Rozga. He will present the internal events Level 1 PRA.

MR. ROZGA: Good morning, everyone, my name is Greg Rozga, I work for ENERCON in support of KHNP and KEPCO doing the Level 1 PRA. And today we're going to be discussing the methodology used. And later on we'll go over, later on in the agenda we'll

talk about the results of the PRA update.

The methodology for the initiating event analysis, again, since we are in design certification space, we have no plant-specific operating experience therefore we rely on generic data.

We did look at other similar plants to identify potential initiating events. And we also did a high-level system failure modes and effects analysis to identify any unique initiating events.

Those initiating events are then grouped based on their impacts to the core protection functions and plants responses. And our final group, we compared them to generic sources. For example, NUREG-5750.

The calculation of the initiating event frequencies, again, was performed using a generic data. And we have an assumed criticality factor of .95.

For the support, for our support system initiating events, there were several support system initiating events identified. And initially we created support system initiating event fault trees to calculate those frequencies. And those numbers were compared to industry generic data.

And for the updated PRA model, based on

that comparison and given the fact that portions of some of these systems are in the conceptual design phase, and since our numbers were relatively close to the generic initiating event, we went with the generic initiating event data from NUREG-6928. And that's for partial and total loss of CCW and ESW and for the loss of instrument air.

MEMBER BLEY: Greg, at what point would you expect those to be revised based on a final design? At the COL stage or just pre-fuel load should that ever come to pass?

MR. ROZGA: Yong, could you answer that because I don't know about the COL?

MR. IN: Yes, there is a COL, it will be a concealed stage.

MEMBER BLEY: It will be a concealed stage.

MR. IN: Yes.

MEMBER BLEY: Okay.

MR. ROZGA: The accident sequence analysis is performed by reviewing the key safety functions and how each initiating event impacts those key safety functions and what systems have to operate during those accidents to support the key safety functions.

And then we develop event trees looking at

key functional requirements to reach a safe and stable state and prevent core damage. And when we do that, we identify any systems that need to operate, any operator actions that have to take place.

We define the accident sequences in a manner consistent with the plant system design. And the operating procedures, as they are, in the design certification stage.

And the expected plant response. And we get that expected plant response from thermal hydraulic analyses.

We also account for the initiator dependency on the systems. So, if the initiator has an impact on the operating of any of the support systems, if it could operate any operator actions, those are accounted for in the model. And we end up with a small event tree, large fault tree, model.

For the success criteria, the definition of core damage is consistent with the ASME standard supporting requirement SC-A2, where we use a peak node temperature of 2200 Fahrenheit and we're using a detailed core modeling program. In our case we use RELAP.

Or 1800 for codes that have a simplified core modeling, such as MAAP.

MEMBER MARCH-LEUBA: Just curious, is that one node or there has to be more than one node excessive temperature? Sometimes we allow for a few, a couple of pins to exceed the temperature.

MR. ROZGA: Jeff, do you have any --

MR. LEARY: I don't.

MEMBER MARCH-LEUBA: It's not important.

MR. ROZGA: Okay. All right. Again, since we don't have any plant specific data, all of our reliability, availability and common cause failure data is based in generic sources.

Most of that comes from NUREG-1928, the online update. There was some vendor specific data on the POSRVs that we use. We also use alternate industry sources when there was equipment that's not in 6928.

And then for special events data, we used the, for the LOOP non-recovery probabilities, again, we used a generic data from NUREG-6890.

The human reliability analysis, there are three types of events. Type A events are pre-initiator events.

And those are events that can cause systems to be misaligned, and there were approximately 60 pre-initiators identified. And they're based on

review of test or maintenance procedures from the reference plant.

Again, this will need to be verified during the COL state.

Type B initiating events are initiating events related to an initiating event. And since we used the generic data for this model, there are no Type B initiating events.

Type C initiating events are after, I'm sorry, Type C operator actions are those post-initiating event. And there is about 70 operator actions that are modeled.

And the dependency among those operator actions were evaluated. And methodologically, how that's done is, initially we re-quantify the model with the HRA value set to a very high number near one.

And the reason we do that is to make sure that we don't truncate out any potential risk significant HRA combinations. And then we analyze the HRA combinations for dependency.

And we use the dependency module in the HRA calculator, which looks at various inputs such as time delays, the queues, sequential timing, the location of the action, whether it's in the control room or outside the control room.

And based on an evaluation of subsequent operator actions, a dependency level was assigned for each action within that group.

MEMBER SKILLMAN: Greg, please go back one slide.

MR. ROZGA: Yes, sir.

MEMBER SKILLMAN: At the first set of bullets, you identify 60 pre-initiators based on test and maintenance procedures from the referenced plants. How do we have confidence that the reference plant, or plants, are large Ps versus Bs, and how are we confident that the unique differences of the APR1400, for instance, the safety injection tanks and no real low pressure injection, are factored into the HRA analysis?

MR. ROZGA: The reference plants that we used for this are SKN 3 and 4. The Shin-Kori 3 and 4. And those plants are very similar to the APR1400.

Some of the differences were the addition of the two diesel generators, the ultimate heat sink is site, ultimately that's going to be site dependent.

But most of your internal systems, like your SI system, your aux feedwater system, the in plant portion of the ultimate heat sink, most of that is very similar, if not the same.

MEMBER SKILLMAN: Thank you.

MR. ROZGA: You're welcome.

MEMBER SUNSERI: So, just so I understand what you're doing on Slide 11 when you say you re-quantify the, so the I'm interpreting that is you run the model with everybody forms perfectly and you start the combination --

MR. ROZGA: No, with everybody almost guaranteed to fail.

MEMBER SUNSERI: Okay. So it's just the opposite of what I'm thinking.

MR. ROZGA: Yes. Yes. Because we want to identify, let's say you had four operator actions and each one normally is about e to the minus 3 and if you have an initiating event and then you have those four, it's going to, our truncation is at e to the minus 13, that would fall off and we would never identify that.

So what we do is we set all those HEPs close to one so we get, we make sure that we capture all those. That's just done for the identification.

Later on when we do the final quantification, we incorporate the actual, actually, when we do the final quantification, we start the final quantification with the elevated HEPs and then, so we have a very, very large set of cut sets. And

then at that point we apply the dependencies.

And then after we do that, then we stick in the actual HEPs and then truncate. That way we make sure that we don't prematurely truncate and risk significant operator action combinations.

MEMBER SUNSERI: Okay, that's helpful. I was trying to understand how that worked, but that clarifies. Thanks.

MEMBER SKILLMAN: One more question.

MR. ROZGA: Yes, sir.

MEMBER SKILLMAN: Please go back to Slide 5.

MR. ROZGA: Okay. Oh.

MEMBER SKILLMAN: In the context of the third bullet set, what is the meaning of criticality factor .95 assumed?

MR. ROZGA: We assume that 95 percent of the time we are in critical. And five percent of an average year goes to this model then low power shutdown modes.

MEMBER SKILLMAN: Oh, now I understand. Thank you. Back to 11, thank you.

CHAIRMAN BALLINGER: Speaking of 11, always is not a number. So, you said you set the value, well, it's actually valves.

MR. ROZGA: Oh, I'm sorry, that should be values. Yes.

CHAIRMAN BALLINGER: To a value near one.

MR. ROZGA: Yes.

CHAIRMAN BALLINGER: Near is also not a number. So, I'm curious as to what you mean by that, and is it a variable for sequences?

MR. ROZGA: It is a number. And we usually use .09.

And the only reason we do that is, sometimes when you use one, the quantification engine will assume that it's a logical true which is different than the number one. And so we just pick a very high number.

CHAIRMAN BALLINGER: So, it's just an engineering judgement?

MR. ROZGA: Yes.

CHAIRMAN BALLINGER: And that's being --

MR. ROZGA: All we want to do is make sure that we capture all of our risk significant combinations.

CHAIRMAN BALLINGER: Yes, okay.

MR. ROZGA: And then the last bullet here is very important. We also ensure that we capture the dependency between the Level 1 model and any operator

actions in the Level 2 analysis.

MEMBER BLEY: Greg?

MR. SCHULTZ: Greg?

MR. ROZGA: Yes, sir.

MEMBER BLEY: Oh, go ahead, Steve.

MR. SCHULTZ: Oh. A couple of questions on this section. On the previous slide you indicated, the third bullet on the first set of descriptions, that there is going to be a verification when the detailed test and maintenance procedures become available.

Isn't there an opportunity to, given all of the work that's gone into the PRA, to influence what those detailed tests and procedures are going to be in an effort to assure that either you've achieved what you intended to do with the maintenance and tests procedures, given what you've learned from the PRA?

MR. ROZGA: So, you're referring to, if one of these pre-initiator HRA events are very risk significant, that you could alter the procedure to make sure that you have verification of the system alignment. That is possible --

MR. SCHULTZ: Possible, but it's not, there's a lot of instructions that are being provided to the COL applicant --

MR. ROZGA: Yes.

MR. SCHULTZ: -- that are intended to connect what has been learned from the PRA to the operation of the plant. The final operation of the plant.

I'm looking both for you and to the Staff to verify today, that that is well defined so that we, you could take advantage, and the Applicant take advantage, of what has been learned from the PRA.

MEMBER BLEY: I mean, related to that you have, you talked about how the PRA was used to affect the design in certain aspects. Using the PRA to affect the policies and operations is a reasonable extension. I think that's where Steve --

MR. SCHULTZ: And maybe it's in the way it's phrased here, but in the interactions that you had with the Staff recently, in documenting the, what the responsibilities of the applicant will be, the COL applicant will be given, what's gone into the PRA.

What I was seeing there is a stronger nexus between the results of the PRA and what would be intended to be accomplished at that turnover.

MR. ROZGA: Yes. I can say that we do evaluate and we provide risk ranking lists for all the operator actions. I can't speak to what that next

step, whether the COL applicant --

MR. SCHULTZ: So, that sounds like perhaps more thinking in that area would be useful, to translate what you just described into some useful actions that the Applicant could take to make sure that transition is done as effectively as possible.

MR. ROZGA: I do.

MR. SCHULTZ: One more question, and backing up a bit to the success criteria. And just a comment, maybe you can clarify.

The peak node temperature exceeding is selected for RELAP5 --

MR. ROZGA: 2200.

MR. SCHULTZ: -- 2200. And then later below you say based on best estimate analyses.

Usually a best estimate analyses, using RELAP5, would include uncertainty in determining whether you've exceeded the peak node temperature. I'm just curious as to whether there is some, whether that in fact makes sense to use a best estimate evaluation and then use a upper bound, what I would consider to be an upper bound criteria for core damage. Or cladding damage.

(Off-microphone comment.)

MR. SCHULTZ: Maybe --

MR. ROZGA: Yes.

MR. SCHULTZ: Perhaps it comes in, in a different way in the analysis, but the way it's presented here is that suggesting that you're perhaps assuming more margin than might be available if you include uncertainty in the evaluation.

MR. IN: Yes, I don't believe we used the upper bound. It wasn't basically based on the best estimate for the RELAP.

MR. SCHULTZ: Okay. Thank you.

CHAIRMAN BALLINGER: Back to the previous line of questioning that Steve had. The Shin-Kori plants are operating, or at least one of them is, and presumably there is a PRA for that plant and presumably, again, that in the development of that PRA, that development feed back into the design of the Shin-Kori plant.

So the PRA that you can, I guess compare with if you will, is a PRA that has incorporated, presumably, that iterative process. So, when you use Shin-Kori PRA as the reference plant, that should have some things built into it already, that have affected the HRA issues there, is that correct?

Maybe I'm not phrasing the question right, maybe Steve can get a better handle on the language.

MR. ROZGA: Yes, I'm not quite sure I understand.

CHAIRMAN BALLINGER: The reference plant is Shin-Kori.

MR. ROZGA: Correct.

CHAIRMAN BALLINGER: In Korea, I'm assuming that there is a full-blown PRA that's been done for Shin-Kori.

MR. ROZGA: Yes.

CHAIRMAN BALLINGER: As part of that PRA development, the iterative process of using the PRA to influence the design has taken place, I think, for Shin-Kori.

So the reference plant that you're using has had the PRA developed along those lines. So you're comparing the PRA for this plant.

I don't know, maybe comparing is the wrong word, but you're using this plant and referencing using the reference plant at Shin-Kori. Which has that stuff already built into it.

MR. ROZGA: Right. Those were the, test and maintenance procedures were from Shin-Kori 3 and 4. I do not know whether any input from APR1400 was feedback to Shin-Kori.

CHAIRMAN BALLINGER: I mean, I thought it

would be the other direction?

MR. ROZGA: Yes. Yes, well, it's feed in this direction based on our use of those procedures and the assumption that the APR1400 procedures will be very similar to those procedures, based on the similarity of the systems.

CHAIRMAN BALLINGER: Okay.

MR. ROZGA: I don't know if they have enough operating experience yet to have, I mean, they were just recently critical, so.

MR. IN: I'm Yong. And to provide backup information, there is a COL item, 19.1, which says that the COL is to review as design as both information including the custom, the operator procedures to make sure that the HRA analysis results are reviewed. So, there is a general COL item that makes sure that the results are reviewed through the development of the procedures.

MR. SCHULTZ: Thank you. I know that's important. And we'll talk more about that I'm sure today, thank you.

MEMBER BLEY: That kind of begs the question, many of the things that aren't in place now, and your peer review pointed out some of this, for doing the HRA, including specifics of the location of

equipment details and procedures, aren't there now, but some of this won't, at least the procedures probably won't be available at the time of a COL, but they will be available before fuel load.

Is there an ITAAC or anything else other than a COL item that raises the issue that the PRA needs to be fixed up in those areas?

Because I don't think you can do it as the COL. Nobody has been able to do it yet at the COL stage.

MR. ROZGA: I would have to check on that whether there is an ITAAC for that.

MR. IN: Yes --

MR. ROZGA: That I don't know.

MR. IN: -- I'm not sure if there is an ITAAC item, specific item, related to the PRA update. But the COL holder is to make sure that the use of this design certification PRA is applicable to the specific COL applicant.

MEMBER BLEY: Yes, but they're going to be in the spot of saying the same thing. At least most everybody has.

By the time, it will be after they get their license but before they startup that they'll actually have all of that in place. So, somehow that

needs to be flagged for later.

We'll probably flag that at some point before the end of this month though.

MR. SCHULTZ: Thank you.

MEMBER SKILLMAN: May I ask this please? And particularly to you, Yong.

As Shin-Kori is proceeding through its early year or years of operation, how is the operating experience from that plant being recorded so that, that experience can be taken advantage of with this design?

Is there a reporting system such as INPO where if there is an event, the operators report to a central --

MR. IN: Here Shin-Kori maintains a operating experience in the database, similar to the INPO. The Korean Industry, there is only one utility so there is no equivalent INPO organization.

So here it's simply, it's often maintained in the operating experience. They have a database, it's something similar to the EPIX, maintained by INPO.

MEMBER SKILLMAN: So how do the designers of this design, this APR1400, find their way into important information from Sin-Kori that might

influence the design of the --

MR. IN: Designers have access to the database.

MEMBER SKILLMAN: Do they use it?

MR. IN: Yes. Because if there are, actually, the designers who are supporting the APR1400, this design certification project, are the similar people. The same people who are also supporting this 3.3 INPO.

MEMBER SKILLMAN: Okay, thank you.

MR. IN: Yes.

MEMBER SKILLMAN: Thank you.

MR. ROZGA: All right. Well, once we have our model and all our data, we go onto the quantification process.

We use industry standards codes to do that. We use EPRIs CAFTA and we use FTREX as the quantification engine.

We have a CDF truncation at 10 to the minus 13. And we did a truncation analysis to demonstrate that that is a viable truncation limit.

MEMBER BLEY: Can you tell us a little bit about how you did that?

MR. ROZGA: Well, what we do is we start quantification at a low level, maybe e to the minus 8,

e to the minus 9, and then you continuously quantify by lowering the truncation limit, basically until the computer can't handle it anymore, and then you review those results and you try to see that your CDF is asymptoting.

And the ASME standard gives some recommendations where they say, like when you start seeing a decade-to-decade, when that increase is less than about five percent. But they also want to make sure that the choice of your truncation does not limit the inclusion of any potential risk significant systems, trains or operator actions.

MEMBER BLEY: I'm more interested in, not the guidance, but in your process.

MR. ROZGA: Okay. Yes.

MEMBER BLEY: Did it approach stable or did you get peaks along --

MR. ROZGA: There is, it's a nice curve, it's starting to asymptote. It's around five percent.

MEMBER BLEY: Okay.

MR. ROZGA: But to me, to me, the convergence isn't as important as when it's the detail review of the look of the results at either e to the minus 12, e to the minus 13, e to the minus 14. And then you see, am I adding, or by using e to the minus

13, am I not including any potential risk significant operator action, am I not including any potential risk significant equipment.

And to me, that's more important for the use. But yes, the asymptote was near e to the minus 5 for internal, I'm sorry, was near about five percent for internal events.

For some models, like the low power shutdown model, it converges at e to the minus 10 or e to the minus 11. However, when you do that, you don't get very many cut sets. And then when you do the additional quantifications at lower and lower truncation limits, you start seeing additional equipment show up on those lists.

So for those, we did not use e to the minus 11 we went further down. And then we settled on e to the minus 13 as the one that captures all that.

But it also allows you to compare apples-to-apples between your different models. They're all truncated at the same value.

So that analysis was done for all nodes, all models.

MEMBER BLEY: I guess the one thing one worries about sometimes there is, are you truncating things with a high RAW, such that if something goes

wrong with a particular component, it could pop up on the list way up high.

MR. ROZGA: Yes.

MEMBER BLEY: Did you look for that?

MR. ROZGA: Yes. Well, yes, that, when we were -- when we looked at the further truncations on it, e to the minus 14, some models we were able to go down to e to the minus 15. We had determined that we had captured all the risk significant systems.

Now, sometimes there might be a common cause event that has a very high RAW, but in that case, we had already determined that based on some other failure mode of that component, that it was already risk significant, so therefore it wouldn't have impacted that component being put in, so for instance the RAP list.

MEMBER BLEY: Okay, thanks.

MR. ROZGA: You're welcome. All PRAs end up with some circular logic, and the circular logic was broken at the DC batteries.

And we do use a cut set recovery process, so after the initial quantification we do a post-processing on the cut set, and that's where we include offside power recoveries, the HRA dependencies rules, which are multipliers. And then we minimize the cut

sets. All that's pretty boilerplate.

Okay. And at this time, I'd like to introduce Jeff Leary. And he's going to be talking about the Level 2.

MR. LEARY: Yes, good morning. My name is Jeff Leary, I'm with ENERCON supporting KHNP and --

MEMBER BLEY: I want to interrupt you, I didn't realize we were jumping to Level 2 quite so fast.

MR. ROZGA: I'm sorry, I should have asked if there were any additional questions.

MR. LEARY: Started to get away with that one.

MEMBER BLEY: Let me find my note. Somewhere in here, are you going to deal anymore with the reactor coolant pump seals?

MR. ROZGA: Yes. Yes.

MEMBER BLEY: Okay. That's going to come later?

MR. ROZGA: Yes, that will be in the PRA update section after the break.

MEMBER BLEY: Okay.

MR. ROZGA: In the SER, the Staff points out two or three places where they thought you had cut sets that included neutrally exclusive events. And

apparently, the resolution in those cases was, you agreed to make those COL items for the future, to examine them.

Did you agree that they were strictly neutrally exclusive?

MR. ROZGA: Yes. Yes.

MEMBER BLEY: One of them had to do with essentially violating practice by tagging out --

MR. ROZGA: Multiple trains of --

MEMBER BLEY: -- multiple trains of equipment. We have a lot of things in the PRA that model combinations of events that have never occurred.

MR. ROZGA: Yes.

MEMBER BLEY: And we're worried about those so we model them, we look for them. Are there any cases in your memory in operating plants where people have violated this tagging out multiple trains?

MR. ROZGA: I believe in the operating experience you could find some. I'm sure --

MEMBER BLEY: If you're unaware of them, if you --

MR. ROZGA: Yes.

MEMBER BLEY: -- go back and look at TMI --

MR. ROZGA: Yes.

MEMBER BLEY: -- one of the very first events, when they condensate and feed, was all the emergency feedwater pumps were tagged out. Nobody ever quite figured out why that happened at least, and anything was written down.

During a Salem grass intrusion event some years ago, one of the operators in the control room managed to put all of the feed pumps in total lock, all of them, and the shift sup didn't know about it.

So we got things that have really happened in the real world and we're saying, they can't happen. I'm uncomfortable with that, and I'm going to ask the Staff about this when it comes around to their turn.

But you seem to agree with them, so I'm uncomfortable with your position here too.

MR. ROZGA: Yes, I mean --

MEMBER BLEY: These can happen.

MR. ROZGA: Yes. Yes. The individual system --

MEMBER BLEY: I mean, on the electrical side we don't live with that, we put in interlocks such that people can't do such things and connect multiple pieces together that shouldn't be connected because you know administrative requirements don't always work. But go ahead.

MR. ROZGA: Yes. The individual system unavailabilities are based on, well, for our plant we're using generic data but they're based on industry experience where they are done properly.

Now, there is a possibility that there could be some overlap. Some of the operators do make errors.

However, that would most likely be a small fraction. Maybe one percent of all that unavailability time. So to say that --

MEMBER BLEY: Two pretty bad events.

MR. ROZGA: Yes.

MEMBER BLEY: One sort of bad and really, really bad happened that way.

MR. ROZGA: Yes.

MEMBER BLEY: And that's what we're looking for, is the risk.

MR. ROZGA: Yes.

MEMBER BLEY: So they're not likely, but when they happen they can be pretty bad.

MR. ROZGA: You're right.

MEMBER BLEY: This one bothers me. The first of those I couldn't quite tell, and I didn't go back and chase it far enough to see I agree they're mutually exclusive but that one sounded like it might

have been.

MR. ROZGA: Yes.

MEMBER BLEY: This one, if you don't model it this way, it belongs somehow in a HRA. You know, do you somehow violate these things? Go ahead.

MR. ROZGA: Okay, any other questions before, okay.

MR. LEARY: The Level 2 analysis on the high-level methodology. The Level 1 sequences were grouped into plant damage states based on similar characteristics for evaluation in a Level 2.

Containment of entries were developed to evaluate the phenomenological occurrences for the severe action. And decomposition of entries were used for detailed analysis of each of the nodes within the containment of entries.

The model was originally created in the, using the SAREX code and utilized plant damage state bridge trees, also called extended event trees, to capture all of the inter-system and intra-system dependencies between Level 1 and Level 2 systems.

When the code, or when the model was converted to CAFTA, the CAFTA code has the ability to full link Level 1 and Level 2. But the same methodology was retained for consistency. So the

CAFTA model looks the same using the same approach.

Computer codes used, the MAAP 4.0.8 code was utilized for analyzing severe accident progression and source term releases. And again, CAFTA was the, ultimately used for the quantification.

These next two slides present some of the design features of the ARP1400 containment and terms of dealing with severe accidents.

The first is the containment building itself. It's a large dry containment, and the large design is, by the large nature is, able to deal with many, many events, many pressure events that the smaller containments can't handle. So the large containment takes care a lot of issues.

The reactor cavity is designed to minimize challenges posed by direct containment heating and ex-vessel phenomena. Specifically, there is a convoluted floor path out of the reactor cavity, out in the upper containment. Which greatly limits, in a high-pressure melt ejection greatly eliminates how much debris can come out and get it thrown out in containment.

The floor area of the cavity is very large to allow for spreading of a molten core debris. The more its spread out the less the depth at which it can accumulate, which increase coolability.

There is a cavity flooding system, which is another way to allow getting water down into the reactor cavity. Getting water into the cavity if the molten debris comes out of the bottom of the vessel, enhances the coolability of the debris, which cuts down on the, on molten core concrete interaction.

Which is good for stopping the progression down through the bottom of the containment and also to minimize the generation of combustible gases.

And there's a hydrogen mitigation system to limit the concentration of which hydrogen can accumulate within containment. And the hydrogen system includes both the ignitors and passive autocatalytic recombiners.

The plant damage state generation is, starts with the Level 1 core damage event tree sequences and adds to them questions of containment systems to develop the bridge trees or extended event trees.

Some of the examples of questions that are added to after core damage has occurred is status of containment sprays, that was not already asked for a containment heat removal in a Level 1, the status of heat generators, again, if not already asked in Level 1. Containment isolation is another example.

After extending event trees and asking those questions, the sequences are then grouped based on similar characteristics as far as how they'll progress through a severe accident. And group, PDS grouping parameter, some of them include whether or not the containment was bypassed, status of containment isolation, whether it was a LOCA or a transient can affect the RCS pressure at the time of vessel failure.

And after all of the binning, as a result, where there was 108 plant damage dates to find, that were then transferred into the containment event tree analysis.

The containment event tree is developed to, for evaluation of the phenomenological accident, different phenomena of the accident progression.

The detailed analysis of each of the nodes of the different phenomena are --

MEMBER BLEY: Sorry, Jeff, I want to back it up just a little.

MR. LEARY: Sure.

MEMBER BLEY: When you did your grouping, you came up with 108, which is a lot --

MR. LEARY: It is a lot.

MEMBER BLEY: -- plant damage state, did

you have Level 2 analysis, thermal hydraulic analysis, that really showed you needed that many?

MR. LEARY: They were --

MEMBER BLEY: This is an in-principle kind of --

MR. LEARY: No, there was a, there was a iteration between the plant damage state and containment event tree performing some cases to see what was really necessary. Out of the 108, I forget the numbers, some of them turned out to not be logically possible.

MEMBER BLEY: Yes, not populate. Yes.

MR. LEARY: And there was a number of them which have zero frequencies. But yes, it was, I think the big one at the end was the addition of, the question of steam generator status.

All of them, almost, maybe not double, but greatly increase the number of sequences to evaluate consequential tube rupture.

MEMBER BLEY: Okay, go ahead.

MR. LEARY: The decomposition of entries, again, are used for detailed analysis of the different event, containment event tree modes. Which each one represents a different phenomena.

Some of the events that are considered in

the containment of entry include whether or not the containment was bypassed, containment isolation status, evaluates the potential for an induced tube rupture, evaluates high-pressure melt ejection effects, stream explosions, both in vessel and ex-vessel, effects of hydrogen phenomena, potential for a late containment over pressurization and then ultimately a question, if everything else stays intact, whether or not the basemat stays intact.

The end points of the containment of entry are then grouped together into source term categories, also called release categories. Again, based on similarities of realistic characteristics in terms of the release magnitudes and timing.

The source term calculations are performed using the MAAP 408 code. And the criterion used for whether the release was binned as large or small is consistent with one of the definitions in NUREG-6595. Which was, I believe was considered large if it was greater than two and a half percent of the semi-volatile and volatile fission products were released.

And the question of early versus later was based on whether or not there was enough time for successful evacuation of the surrounding public.

Are there any questions on Level 2? All

right. Okay, then next we'll turn it over to Mr. Lee for --

MR. LEE: Seismic assessment. Okay, good morning everyone. My name is Dongwon Lee from KEPCO E&C. I'm in charge of seismic PRA team.

I'm going to present to you a PRA based estimate for at-power and low power shutdown. I shall the read the update PRA based estimate for at-power and low power shutdown to respond to RAI 19-10, 19-85.

For us to begin, we did a PRA-based seismic SMA review. Actually, approached it for a risk evaluation of a seismic includes PRA-based SMA. Therefore, we used this methodology for the APR1400.

MEMBER BLEY: Let me ask you a question. At one point earlier, Rev. 0 or whatever, you had a seismic PRA. Or at least the beginnings of one.

Is that you did here for the margins analysis essentially re-framing what you had done in that earlier work or was this new work?

MR. LEE: It's the same as earlier work.

MEMBER BLEY: Okay.

MR. LEE: Yes.

MEMBER BLEY: So you just kind of re-framed how you present the results.

MR. LEE: Yes, yes.

MEMBER BLEY: Okay, go ahead.

MR. LEE: Okay. There are three similar guidance for the PRA-based estimate. NUREG-0800 and SECY-93-087 and DC/COL-ISG-020.

They provide guidance for implementation of PRA-based estimate. And MIN-MAX method determined the HCLPF capacity of an accident sequence.

And also, we used the system modeling enhanced by NRC. Such as in NUREG/CR-4334, NUREG-5076 and IPEEE.

Next. Now I'm going to introduce the seismic input motion. Actual APR1400 didn't have the seismic header, that's why we used the certified seismic motion response spectra, which is based on the Reg. Guide 1.60. And it enhanced in high frequency and anchored to a peak ground acceleration of 0.3g.

If you look at the right figures, the blue is referred to the certified seismic design response spectrum. And that is defined at free field ground surface.

And then for the frequency, seismic margin earthquake is equal to 1.67 times the CSDRS. If you look at the right figures, the red one is the seismic margin earthquake.

Okay, next. To develop seismic equipment

list, we considered the following. First, define the seismic initiating and the consequential events.

We could have select SSCs related to the initiating events. And also, determine the safety functions needed for response, such as the following. And also, we identify systems needed to fulfill safety functions based on the internal events PRAs and powered by onsite emergency AC courses.

But there are exceptions, because of the lack of the detailed information. The SSCs which are fed power by offsite power and the distribution system which is the known as the ruggedness, such as piping, HVAC duct, cable trays, et cetera.

Next. Okay, this is SSCs with design-specific capacities. We only have the specific HCLPFs such as the following. First, building structures and reactor coolant system components.

Next. This is, I'm going to introduce the approach for HCLPF capacity evaluation. And this is kind of a procedure or the methodology.

First, identify the critical failure modes. To do this job, we reviewed APR1400 design-specific reports, calculations and drawings.

And also, we identified potential failure modes by comparing design seismic demand to the design

capacity, relative to design margin. Based on the information, we selected a governing failure mode for HCLPF capacity evaluation.

And also, we have to determine the seismic demand. As I mentioned before, we don't have the specific, site-specific seismic hazard and the response spectrum, therefore we use the APR1400 design-specific seismic demands.

And also, we used the static capacity equations based on the ACI 349 and ASME Section III Service Level D allowable or EPRI NP-6041.

Next. And also we, to consider the ductile failure modes, we calculate the inelastic energy absorption capabilities.

And to evaluate the HCLPF capacities of SSCs, we use the CDFM method in the EPRI NP-6041. And HCLPF is equal to or greater than seismic margin earthquake, that 1.67 times the CSDRS.

Next. This is critical capacity of other SEL components. First, inherently ruggedness items, for example, distribution system, check valve, small instrument, et cetera, was assumed to be rugged.

Because, based on the DC/COL-ISG-020, so called the super element, could have masked the seismic risk result. That's why we needed these kinds

of assumptions.

And also, we have another assumptions for the very small LOCA. Which is assumed to be 0.3g. In accordance with EPRI SPRA implementation guide, we have the, we use the option to, which is, the very small LOCA is equivalent to 0.3g SSC level.

And also, critical offsite power is assumed at 0.09. Based on this handbook. And also, for the, consider the characteristics of the seismic event, the component of the same type in the system conservatively assumed to be 100 percent correlated.

Next. To develop seismic login model, we performed the failure model effective analysis for SSCs in the SEL. And based on these, SSCs in the SEL were modeled individually.

If you look at the first thing, SEIS, is meaning, mean to be seismic. And the second one is equipment ID, and the last thing is HCLPF number.

If you look at the basic event name, you can easily find out how much have the, how much equipment have the HCLPF number. For example, SEIS-SIPP02, that's 0.5g. That means the pump have the 0.5g.

And also, HEPs are modeled to figure out the important operator action or human error. And

also random failure event from internal PRA model were adopted in the PRA-based SMA model.

Next. This is on at-power Level 1 and Level 2 initiating event. We have the ten different initiating event.

And based on the SMA, we could have the consequence of the initiating event. And if you look at the third column, there are a kind of containment failure shift, and the failure of a containment isolation and the failure of a containment isolation or a rate of a pressure failure.

That is based on the failure mode effective analysis.

Next. As a result of accident sequence analysis at-power plant level HCLPF is equal to 1.67 times CSDRS. It met the SECY-93-087.

Now I'm going to introduce the insights for the power. This is the review piece by seismic event.

First thing is equipment related to secondary heat removal such as motor-driven aux feedwater and the turbine-driven aux feedwater pump.

And the second one is equipment related to long-term containment heat removal. Such as containment spray and shutdown cooling systems. And

lastly, equipment related to essential power, such as EDG and 4.16k volt electrical panel.

Next. This is the important operators' actions based on the accident sequence analysis. We have the four different operator actions. The first thing is the operator action to perform aggressive secondary cool down after small LOCA.

And operation action to initiate a hot leg action and operation to open POSRVs in early phase of feed and bleed operation and operator action to cool IRWST water using shutdown cooling system pump.

And this is important, random failure. We have the EDG is figure out one of the important random failure.

Next. Here is the end of the PRA-based SMA for at-power, now I am going to introduce the, briefly, for the low power shutdown. The methodology is the same as at-power and the SER is based on at-power. But added some equipment to consider the low power shutdown condition.

Seismic fragility is also same as at-power, but additional equipments have to be considered, such as containment hatch and the polar crane, et cetera.

Plant operating condition was considered

and modified from POS 3B to 12A considered. And we, plant condition is grouped into two things, whether RCS is intact or not intact.

Plant response analysis is similar at-power, but LPSD conditions are considered.

This is kind of a result, plant level HCLPF is equal to 1.67 times CSDRS, which is from MIN-MAX method.

And also, we have the insight for the vulnerability by seismic event and important random failure and the important operator actions. Total information is in the DCD Revision 2.

Next. Okay, now I'm going to summarize what I did for the PRA-based SMA. Methodology is SECY-93-087 and DC/COL-ISG-020.

Plant operation condition is considered for full power and the low power shutdown. And the seismic fragility, based on CDFM in EPRI NP-6041.

HCLPF for structure and NSSS component is higher than 0.5g. And the other SSCs in the SEL is equal to or higher than 0.5g. These tree events are based on the assumption that I mentioned before.

And accident sequence is based on the seismic PRA model, such as the event tree and fault tree. Based on this accident sequence analysis we

have the conclusion, plant level HCLPF is equal to 1.67 times CSDRS for at-power and LPSD post.

And also, we have the insight for at-power and low power shutdown. We have the seismic vulnerabilities and the important operator actions and the important random failure vent.

That is what I have prepared for this meeting. If you have a question, feel free to ask.

MR. SCHULTZ: I have one question. Associated with the expectations of the COL Applicant in performing verification and validation of what has been done here with regard to the values as well as the site-specific construction, facility construction and so forth, that's captured, some of that is captured in the COL items 19.18 and 19.19.

And when one reads through all of those expectations, it seems as if the expectation is that the COL applicant will essentially re-perform the seismic evaluation in PRA. Is that the right picture framework?

MR. LEE: Actually, as I mentioned in my presentation, we have the, a link to the information, that's why we couldn't do a specific PRA-based --

MR. SCHULTZ: Yes.

MR. LEE: -- that's why, just the 0.5g

level meets our expectation. And if we have the site-specific information, then we will do a re-evaluation of these things to make this kind of call.

MR. SCHULTZ: But there are also, based on the evaluation that is done then, the intent is to assure that the values associated with structures and components, for the very most part, are going to meet the HCLPF value. Not only for the plant as designed, but also the structures that are supporting, you can call them support but they're really just the onsite structures that have been designed by the Applicant, by the COL applicant, to match that up.

Then to move forward and capture the quantification of all of that and the seismic PRA results.

MR. LEE: That's correct.

MR. SCHULTZ: So that that information is available to move forward with a complete external events and internal events PRA picture. Do I have that right?

MR. LEE: That's right.

MR. SCHULTZ: Good. Thank you.

MR. LEE: You're welcome.

MR. ROZGA: If there is no other questions, then we can move on to fire. Everyone

fine? Okay.

MR. IN: Yes. We have 11 slides for the fire PRA topic, so I think we can just go ahead and move on to the fire PRA.

(Off-microphone comment.)

MR. IN: I'm sorry. We have 11 slides for the fire topic so I think we can move on.

MR. ROZGA: Again, this is Greg Rozga from ENERCON and I'll be discussing the internal fire PRA for at-power and low power shutdown.

And like Mr. Lee, what I'll do is I'll describe what was done for the at-power and then I'll say what was done a little bit different for low power shutdown. Because there is so many similarities.

First task in the PRA, in the fire PRA, is to partition the plant into individual physical analysis units. And for the APR1400 there is 390 physical analysis units.

And they're all based on fire areas with rated barriers. And of these, 279 are in the auxiliary building. And that turned out to be very important.

The aux building is highly compartmentalized with almost every single room has three-hour rated barriers. And each inside room there

is a smaller amount of equipment than you would have in a traditional plant.

And all the physical analysis units, they are mainly single rooms but they may contain two or more rooms. We did not take credit for any interior rooms, or any interior partitions.

And the low power shutdown model uses the same physical analysis units. We do understand that there are removal barriers, however, those removal barriers are mainly just used for large equipment replacement, which happens very rarely.

And we made an assumption. And there is a COL item that this will be done during de-fuel. And if it's not, then they're going to have to re-analyze the impact of removing those barriers.

MEMBER SUNSERI: I supposed that there are tech specs on those barriers to keep them in place like that. Most operating plants I'm familiar with there are --

MR. ROZGA: Yes.

MEMBER SUNSERI: -- requirements for that. Is that the case here?

MR. ROZGA: That I do not know, but I know that they are part of fire barriers. And some of them may be part of flood barriers so I suspect there is

some.

MEMBER SUNSERI: I mean, there would be ways to degrade the barrier and put a compensatory fire watch or something in place but I don't know what the practice here is. Okay.

MR. ROZGA: Yes, we actually, for the PRA, we do evaluate the risk impact of each of the individual barriers. Not just the removable barriers, but normal fire doors that can be opened as well. So all those are evaluated as part of the fire PRA for both at-power and low power shutdown.

Tasks 2 and 3 are the equipment and cable selection. And for both the at-power and low power shutdown internal events PRA, that equipment list is based on the equipment that's in the internal events PRA with equipment added specifically for fire.

For the at-power model, there were some non-modeled spurious operations that were incorporated into the model for their potential impact for a fire induced spurious operation. And there was additional equipment added in the low power shutdown model for the fire-induced LOCAs that could happen during low power shutdown.

Cables for all of the internal events equipment were routed based on the reference plant.

And the number and type of penetrations for the multi-compartment analyses are also from the reference plant.

You notice these tasks kind of go out of order. These task numbers are from NUREG-6850 but I am presenting them in the order how you build the model.

So, after you have identified all your cables, the next thing you do is you do your detailed circuit failure analysis or your circuit failure model likelihood analysis, which essentially is a long phrase for saying calculation of the spurious operation probabilities.

For the APR1400, all cables were assumed to lead to worse case failure mode. If it was a power cable it was considered loss of function, if it was a control failure, it would be the worst case of either failure to operate or spurious operation.

And then for fiber optic cables, there was a loss of function or failure to operate. There was no spurious operation assumed for fiberoptic cable damage.

And then for the circuit failure mode likelihood analysis, all spurious operations were assumed to occur with a conditional probability of

one. So, if the cable, if the control cable was damaged and the worst thing that could happen to that component was to have a spurious operation, it was assumed that that spurious operation occurred.

We took no credit for clearing the hot shorts and no credit was taken for recovery of spuriously operated valves.

MEMBER BLEY: Let me ask you a question and try to understand how you dealt with it. Worst case spurious operation has to be a scenario specific thing. And some potential fire scenarios, a valve operating, opening, might be the worst thing and other ones not opening might be the worst thing.

Did you assume one worst case across the board or did you go on a cut set by cut set basis or a scenario-by-scenario basis?

MR. ROZGA: Worst case across the board. Generally if there was a valve that could lead to a flow diversion, it was assumed that there could be a potential for spurious opening of that valve.

If there was something in that line that was in the flow line, we would assume that, and it's a normally opened valve, we would assume that there would be a normally closed.

And, probably in 99 percent of those

cases, the worst case spurious operation is the same for most sequences. I don't know if there were --

MEMBER BLEY: But we worry about things a lot worse than 10 to the minus 2 scenario so, I don't have a specific example but I bet there are somewhere in a particular cut set it would be worse if the thing opened rather than failing close. And vice versa. And you didn't look for those?

MR. ROZGA: No, sir.

MEMBER BLEY: Okay. I think that's something, it's probably okay for now, given the way the Staff is focused this on capability category one for design cert stage, but later stages it becomes important to look at those details I think.

MR. ROZGA: Yes. As part of the spurious operation evaluation, all power operated valves and pumps were looked at, on a line-by-line basis. And it was discussed, what would be the impact of spurious operation.

MEMBER BLEY: I don't know what line-by-line means to you.

MR. ROZGA: Well, all the valves --

MEMBER BLEY: To me --

MR. ROZGA: All the valves --

MEMBER BLEY: -- it might mean --

MR. ROZGA: All the valves in a --

MEMBER BLEY: -- scenario-by-scenario --

MR. ROZGA: Yes.

MEMBER BLEY: -- in which case, that's not the same thing.

MR. ROZGA: Yes. Correct. Task 5 is development of the fire induced risk model. And, the at-power and the low power shutdown fire induced risk models are both based directly on the respective internal events models.

However, as previously stated, the basic events for the equipment that was identified in Task 2 were added into the fire model. And, again, those were valves for spurious operation, et cetera.

And for at-power, the fires in each physical analysis unit are resumed to result in either a general transient, loss condenser vacuum, loss of DC Train A or B, loss of feedwater, loss of offsite power, a partial loss of CCW or a small LOCA based on equipment damage. There were no other fire induced initiators identified.

And if there was a room, if there was a physical analysis unit where there was no PRA equipment or cables, so none of that was damaged, we still assume that you'll have at least a plant trip,

and we modeled that as a general transient. Whether any equipment is damaged or not, the assumptions made to the operator would trip the plant.

And all fire induced failures are assumed to be non-recoverable, including offsite power. So, in the internal events model we model offsite power recovery on a loss offsite power.

For fire, the assumption is that you can't do that within your 24 hour mission time because of the fire damage, so we don't take credit for it.

For the low power shut down --

MEMBER BLEY: Just to --

MR. ROZGA: Yes, sir.

MEMBER BLEY: I'm trying to remember. It comes in a separate issue, but you didn't look at the response to the Fukushima kind of thing so, FLEX, that sort of thing?

MR. ROZGA: We did not take credit for any FLEX equipment.

MEMBER BLEY: Okay. But here is a case where one could? Okay.

CHAIRMAN BALLINGER: But back to the, for the uneducated metallurgist in me, back to spurious operation. What does that mean?

MR. ROZGA: That means that there is a

fire --

CHAIRMAN BALLINGER: Okay, I understand.

MR. ROZGA: -- which damages a cable and then you have a phase-to-phase, short and --

CHAIRMAN BALLINGER: Okay. So it's a terminal event is what you're saying, not a case where you get operation then it goes back to normal and then fails, then goes back. So that's not, you're not calling that spurious?

MR. ROZGA: Yes. With respect to -- correct. We are saying that it goes in the undesirable position --

CHAIRMAN BALLINGER: And stays there.

MR. ROZGA: And stays there, yes.

CHAIRMAN BALLINGER: Okay.

MEMBER BLEY: There were --

CHAIRMAN BALLINGER: That's what I --

MEMBER BLEY: -- Ron, there were a whole series of tests done out at Sandia --

MR. ROZGA: Yes, I was there for that.

MEMBER BLEY: Oh, were you?

MR. ROZGA: Some of them. Some of them, yes.

MEMBER BLEY: Okay.

MR. ROZGA: Very interesting.

MEMBER BLEY: Yes, it was good stuff. And the industry ran some as well. And sometimes there were cases where you had a hot short and then the hot short cleared later.

In fact, usually if it was a hot short, not usually, often it would later clear. But, if you have already driven a valve to its position, then clearing it, unless it's a spring-loaded kind of thing, won't get you out of the woods.

But, there was lots of evidence of these kinds of things happening. And there were cases where it went both directions, but they were fewer. But there has been a lot of testing, it's not a made-up worry.

MR. ROZGA: Yes. And the fire methodology, the way it's performed is, is you assume that the hot short happens, it doesn't clear. If it turned out that that was a risk significant event, then we would have gone back and evaluated it.

But usually what happens, especially with the way that this plant is segregated, if you're damaging a cable on the SI system, you're probably damaging another component on the SI system which might, you know, so end up, even if that hot short were to clear, it may or may not resolve your problem.

And so there were none that were identified as risk significant. So we never did go back and do any additional re-analysis.

For low power shutdown, the fire induced risk model screened POS 7, 8, 9 and 12. And we use the same bases as the internal events. Eight, you're de-fueled, 7 and 9, your water level is so high, by the time you boil it's well beyond 24 hours. And the same thing with 12b.

And the unscreened low power shutdown, physical analysis unit fires are assumed to result in an event considered in the low power shutdown model, based on equipment damage in the scenario and the POS.

The APR1400 model has an assumed outage schedule where certain equipment is in test and maintenance and therefore the other trains are operating. So therefore, if you have the alpha train of shutdown cooling as your main cooling, then damage to the other division would not stop your normal cooling and you would never have a specific train.

So therefore it depends on the POS that you're in and the equipment damage identifies what the initiating event is, in each physical analysis unit.

Task 6 is calculation of the initiating event frequency. The NUREG-6850 and Supplement 1 was

used for the at-power counting and apportioning methodology.

And the EPRI 1016735 is used for the generic frequencies. The low power shutdown ignition frequencies were for NUREG/CR-7114.

For the HRA analysis, all current human failure events calculated using the HRA methodologies within the HRA calculator. So we did it using the same methodologies that we did for the internal events.

However, all of those events can be impacted by the fires themselves. So there is, you have different stress levels, there are documents, EPRI documents, for calculating the HRA which suggests that you increase the time delay by a certain amount, et cetera.

And then the other thing is, all fires are assumed to at least partially impact one division of instrumentation. So we always assumed that the operators had some spurious indications in the control room.

And then the dependency was reevaluated using the same methodology used in the internal events, but including the fire HE impacts.

MEMBER BLEY: Well, since you mentioned

some spurious indications in the control room, did you look at scenarios where loss of control reached the point that you have to abandon the control room?

MR. ROZGA: We only did control room abandonment for control room fires. There was a --

MEMBER BLEY: That's the habitability concern?

MR. ROZGA: Correct. Correct. There was an assumption made that because of the divisional separation, you would always have some instruments that should be giving you the proper reading and that the operators would be trained that once they know the location of the fire, that if they had differing instrumentation, they would rely on the instrumentation from the undamaged division.

MEMBER BLEY: For future consideration, there is a recent NRC EPRI report on control room abandonment that considers more things.

MR. ROZGA: I just saw that.

MEMBER BLEY: Yes.

MR. ROZGA: I just saw that.

MEMBER BLEY: It's worthy of a look should you ever have to do a COL or a pre-fuel load PRA. Fire PRA.

MR. ROZGA: I appreciate that.

CHAIRMAN BALLINGER: For reference purposes, we've been joined by Member Walt Kirchner.

MR. ROZGA: Welcome. Task 7, 8 have to do with, Task 8 is quantitative screen and Task 8 is where we start doing some high-level fire modeling.

However, for the APR1400, we have very little knowledge of the ignition source target locational relationships. So, to try to do any reasonably accurate fire modeling is very difficult.

So, initially we did full room burnout scenarios in all the physical analysis units for the quantitative screening. And then there was some initial high CDF areas based on that full room burnout, included the main control room, the containment and the turbine building. And then there were 35 other single compartment PAUs that were above the screening criteria.

And for those important PAUs, the only type of detailed fire analysis, and I say that in quotes, is we take credit for the automatic suppression systems and we take credit for manual suppression, only in cases where we know somebody will be there. So therefore we do it in the main control room and for hot work fires.

There should be a fire watch in there. At

least be the person doing the hot watch, or the hot work.

There was actually no fire modeling performed. So if the suppression fails, and those are based on the non-suppression probabilities in 6850, and Supplement 1. And if the suppression fails, then we just assume that you have the full room burnout scenario.

We also don't have any knowledge at this time of any intervening combustibles. So there was no screening of even the smallest ignition source. It was assumed that if there was an ignition sources in the PAU, it will eventually consume the entire room.

And one of the main impacts of that is that all unscreened multi-compartment scenarios are assumed to be possible. And by unscreened I'm referring to the screen criteria in 6850 where if the room that the fire would go to doesn't have any more PRA equipment than is already damaged, then it's not necessary to evaluate that fire.

So then, for the detailed analysis that we did do, we did look at main control room abandonment scenarios. We had to look at containment for, because there's the potential for a small LOCA and you consider the containment as one big giant fire area,

you will impact all four trains of instrumentation. So therefore we had to look in the containment building at individual sources.

And the turbine building was important because there is so much equipment in there it ends up having a very high initiating event frequency and there is offsite power cables in there and it just turns out that you had a high CDF if you assumed everything failed for every fire.

And then the 35 other single compartments were just analyzed because of the high initial CDF that didn't meet that screening criteria we talked about in the prior slide. And the last detailed analysis is the multi-compartment analysis.

The seismic fire interaction analysis is just a qualitative assessment, and it was based on design documents. Generally, you would follow that design document review with a walk down to verify all the assumptions you made, but obviously at this point we were unable to do that.

MEMBER SKILLMAN: Greg, let me ask this. For these simple compartments, and if you will, compartments where there is a moderate or large mechanical component, where there might be lubricating oil in flasks or bottles or tanks that are associated

with that device, how does the PRA account for that?

Is there some link to transient combustible program or to tech specs, such that the btu or kilowatt potential burden becomes excluded or included. How is that handled?

MR. ROZGA: Yes. The exact size of those fires is not evaluated because we assume any fire in an area would end up failing the entire compartment.

However, there are, what's termed in 6850, transient influencing factors. So you take, you identified areas where a lot of people are doing a lot of work and maintenance and you basically a portion your transient frequency and put the ones where if you have flammable liquid cabinets in those areas or machine shops or areas where a lot of maintenance is done, high population areas. So you skew the frequency more towards those rooms.

MEMBER SKILLMAN: Thank you.

MEMBER BLEY: Greg, two things. I'm a little behind on my reading here.

MR. ROZGA: Okay.

MEMBER BLEY: Does this plant have a consolidated remote shutdown panel if you have to abandon the control room or is it a distributed system?

MR. ROZGA: It's a consolidated.

MEMBER BLEY: Did you look at pathways between the main control room and that location for the specific fires, well, you're only looking at control room abandonment if it's a fire in the control room.

MR. ROZGA: It's the fire in the control room, correct.

MEMBER BLEY: Yes. So you're assuming there is no other fire elsewhere, so there is no problem getting to them?

MR. ROZGA: Yes, there is generally two paths. And for the, I don't want to say something that I don't know is correct or not. I don't know if this, if what I am thinking of was part of a conceptual design or whether it's actually part of the design, so I just won't say it.

But for, let me just say the main remote shutdown room, it's consolidating and there are at least two pathways that you can get to it from the main control room.

MEMBER BLEY: Okay. And the other thing, I apologize, I had to slip out at the end of the seismic piece, you talked some about human reliability. Here you mentioned a combination of fire

and seismic.

In the seismic area you did HRA? Human reliability analysis. You modeled human performance in the seismic?

MR. ROZGA: Yes.

MEMBER BLEY: How did you account for the mental effects of the seismic event?

(Off-microphone comment.)

MR. ROZGA: There's a lot of anecdotal evidence that people respond in some odd ways to having been through the kind of seismic event that could upset the plant. Not just the room shakes a little bit but you get knocked off your feet and the earth turns to jelly kind of thing.

People end up with kind of blackout periods for short-term to even an hour or two. They do routine tasks pretty well.

Anecdotal information. Again, they might get in a car and drive home, no memory of having done that.

Do you have any thought to how the seismic event would affect the people or did you use the standard HRA that you use for the Level 1 PRA?

MR. LEE: Which area, for the PRA-based SMA they didn't do that specifically because the PRA-

based SMA just figure out the important operations and --

MEMBER BLEY: And didn't quantify them?

MR. LEE: Yes.

MEMBER BLEY: Okay.

MR. LEE: We just tried to figure which --

MEMBER BLEY: What people would need to do?

MR. LEE: Yes.

MEMBER BLEY: Okay. And that was the margins analysis, that's okay. Okay, thanks.

MR. SCHULTZ: Greg, one question on your last bullet.

MR. ROZGA: Sure.

MR. SCHULTZ: The interaction analysis seismic fire, or seismic fire, you've indicated it's qualitative because it doesn't of course include the walk down that would be required to make it a more robust evaluation. Perhaps you can help me by providing some comparison or results associated with what that qualitative analysis showed.

The reason I ask is that, in the COL direction, the COL application is there to perform a walk down to confirm a qualitative seismic fire interaction assessment. So, is there enough

information that's provided in both preparation for that walk down and then how that walk down would be determined to perform?

Everything is still stated as qualitative, so --

MR. ROZGA: Right. Well, let me --

MR. SCHULTZ: -- how do you get to the final results and what does that look like --

MR. ROZGA: Yes.

MR. SCHULTZ: -- compared to the other risks that the plant is subjected to?

MR. ROZGA: Let me first clarify that. The seismic fire interaction analysis is always a qualitative analysis. There is no quantitative analysis done, even in existing plants.

And you look at things like, do you have, are your flammable liquids or flammable gas tanks, are they adequately mounted, you look at your fire brigade paths and whether the cable trays or whether equipment that's in those pathways, whether they're adequately mounted and whether that would impede.

You look at the diversity of communication equipment. And what I did was, I went through the DCD and went through all of the individual sections that talked about those particular aspects of the fire,

seismic fire interaction analysis.

And basically my analysis, for example said, based on the DCD, there are these four different diverse communication systems so therefore if the earthquake impacted one you would still have walkie-talkies or cell or other.

And I discussed the mounting of tanks and such based on the information that's in the DCD. At this point it would be up to the, well, I don't even know if it will be the COL applicant or it would actually be at, when the plant is actually built, somebody at that point needs to go and they have to verify that those cable trays are adequately seismically rugged and they would not impact the flow path of the fire brigade. And they would have to verify all those assumptions.

MR. SCHULTZ: That really what I was looking for because, hopefully in that later to be designed equipment and configuration, what you've just described you would look for in a walk down would have already been determined to be put in its proper place if you will.

That those things would have been taken into consideration so that when an external event expert team walks through, they wouldn't find anything

of that nature.

MR. ROZGA: Correct.

MR. SCHULTZ: That you've described.

MR. ROZGA: And that ultimately --

MR. SCHULTZ: So, hopefully there's enough information --

MR. ROZGA: -- that was the conclusion.

MR. SCHULTZ: -- that's provided from the qualitative evaluation and analysis to provide that type of guidance so that there's nothing that is founded at the last moment, or later on in plant operations that needs to be adjusted, amended, at great expense.

MR. ROZGA: Correct.

MR. SCHULTZ: Okay.

MR. ROZGA: Things such as the fire brigade flow path for each individual fire, nobody is going to know that at this point, and probably not even in the COL stage. So there may be, I can envision instances where there might be something where people did not realize that this might be a flow path.

But at that time, then they, if it's been identified, then they can go and upgrade the securing down of the panel or the cable tray.

MR. SCHULTZ: And there's no mention in what's been discussed today about detections, fire detection systems?

MR. ROZGA: That's discussed in the seismic fire analysis as well. And so we talk about seismic event. You can have dust and how that could impact the ion, the ionic fire detectors. And so, all that is part of that analysis, but it's done in a qualitative basis.

MR. SCHULTZ: And it's been part of the evaluation within the fire PRA?

The detection system capability or is that just assumed not to be --

MR. ROZGA: No.

MEMBER BLEY: Without the seismic event.

MR. ROZGA: Yes. Yes, that's a seismic event. But even in the normal fire PRA, we really don't take credit for, well, we do for automatic suppression if, say for instance you need to have a detector go off and then that starts the actuation, we have the failure probabilities of those components.

MR. SCHULTZ: Okay, thank you.

MEMBER BLEY: So, I hadn't thought about this before, but, in your qualitative seismic fire analysis, if in fact you have this strong seismic

event and you create dust all over the plant, you could have all of those things go off and, how does the fire main pressure hold up if you have all the sprinklers turn on at one time, I have no idea.

MR. ROZGA: Yes.

MEMBER BLEY: Did you look?

MR. ROZGA: Well, I mean, depending on the type you would have to melt the fusible link. So if there is no local fire in the area.

So, yes. So, for the seismic fire interaction analysis --

MEMBER BLEY: Yes, even if it's charged you don't get it in unless you cut the fire.

MR. ROZGA: Yes. You have to have the seismic event, and that seismic event has to effect equipment in an area that could cause a fire. So that limits the areas that you have to look at.

Okay. Okay, so the results of the internal fire PRA at-power, there's total of 480 single compartment analyses. And if you recall, there were only 390 physical analysis units. But when we did the detailed analysis, we broke those down into several scenarios.

And there is 1,054 multiple compartment scenarios. And due to the highly compartmentalized

nature of the APR1400, the CDF is generally very well distributed. There is only 24 scenarios with a CDF higher than one percent.

Fifty percent of the CDF is in top eight scenarios. And if you look at the last bullet, we have a conservative assumption for the main control room.

Since we don't have the control abandonment procedures at the time the analysis was done, there was just, if you met the abandonment conditions, the CCDP was assumed to be .1. And that took credit for the operator actions, equipment failures that would happen.

And based on that, main control room abandonment scenarios make up about one-third of the CDF.

MEMBER SKILLMAN: Greg, would you go back please?

MR. ROZGA: Yes, sir.

MEMBER SKILLMAN: Just one slide. My question is asking your opinion. It sounds dandy that the highly compartmentalized nature prevents there from being large contagion within the containment for some event. Whether it's fire or something else.

So that sounds like a really good thing.

At the same time, the highly compartmentalized containment is like a rat's nest. You got to know where you're going and why, because there is so many permutations and combinations, you got to figure which is the right compartment to get to and when.

So, I'm asking your opinion because it's clear you have been dealing with this for a long time. Not only on this design but on other designs.

Is a highly compartmentalized nature a good thing or a bad thing?

MR. ROZGA: I believe that it is. And --

MEMBER SKILLMAN: It is what?

MR. ROZGA: It is good.

MEMBER SKILLMAN: Okay.

MR. ROZGA: It is good. And the design of the APR1400, it's not like you go from room to room to room to get, there is, on each level, there is a main corridor. And from that main corridor you get to go into each of the individual rooms.

And so I think for an operator it would be relatively easily to find those rooms, you don't have to go into a room to get to another room. I'm not saying that there aren't any cases like that, but they are very few and far between. Almost every single room can be reached from the main corridor.

And each quadrant there is stairwell and elevators. So, when we were doing our fire induced impact on the manual actions, we were always able to identify at least two paths where the operator could get to any area.

And because of the compartmentalization, that kind of kept the fire in a certain area. Even assuming the multi-compartment analysis. So overall, yes, I believe that it is a good design feature.

MEMBER SKILLMAN: Okay, thank you. Thanks.

MR. ROZGA: You're welcome.

CHAIRMAN BALLINGER: Related to that, this statement has been articulated many times about the advantage of a highly compartmentalized nature. How different is that from other, let's say, CE plants that we know about?

Is there a big different between say Palo Verde and APR1400? I don't know the answer to it, I'm just curious as to whether there is a big difference.

MR. IN: Yes, Palo Verde --

(Off-microphone comment.)

CHAIRMAN BALLINGER: Yong.

MR. IN: Oh, sorry. Palo Verde is basically a two-division train plant design. This is

basically, we got four independent trains.

CHAIRMAN BALLINGER: Well I understand that part --

MR. IN: Yes.

CHAIRMAN BALLINGER: -- but it's the compartmentalized nature. Meaning, rooms are different.

Is there a big difference between --

MR. IN: Yes, the basic philosophy of the compartment design is very similar. Even the Palo Verde design, nobody had that new concept.

MEMBER BLEY: One thing we've seen in some plants is the increased potential for internal flooding affecting the equipment in one compartment if your drainage doesn't work the way you expect or the flooding doesn't come from where you expect it to come from.

MR. ROZGA: Right. It gets --

MEMBER BLEY: If it comes from the other side or something else. But, I can't speak to any details on this one.

CHAIRMAN BALLINGER: Because I'm looking at a risk poster from another plant, okay. And 65 percent of the risk CDF is due to fire. But none of that is related to what I would consider

compartmentalized nature of the rooms, it's control room fires.

Switch gear room, battery room, main turbine room. So, it's very subjective but it's just, I'm just curious, that's why I asked the question.

MEMBER BLEY: And with no details behind it.

CHAIRMAN BALLINGER: And with no details behind it, right.

MR. ROZGA: And we have, we kind of have the same thing. We have the main control room, the EDG, the electrical equipment rooms. We are hitting some of those same, some of those same areas.

The other differences based on experience is the separation of cables and the older vintage of plants isn't as good as there's lots of fires where you end up hitting multiple divisions.

CHAIRMAN BALLINGER: Okay, I get it.

MR. ROZGA: Yes.

CHAIRMAN BALLINGER: Thanks.

MR. ROZGA: For low power shutdown, again, there is many, many scenarios. Nine hundred-eight single compartment over 6,000 multi-compartment.

And, like in the, what I had mentioned earlier, the fire scenario is defined by both the

initiating event and the POS. And that's why you have so many more scenarios that you evaluate.

And like the at-power event, because we're really compartmentalized, the CDF is generally well distributed. Only 22 scenarios are higher than one percent, 50 percent of the CDF is in the top ten scenarios, 90 percent in the top 59.

And the CDF is concentrated in the EDG rooms, the electrical equipment rooms and the ultimate heat sink building.

MEMBER KIRCHNER: Greg, as a result of this exercise, did you identify any design deficiencies that you would change or pass on to the COLA applicant to make physical improvements?

MR. ROZGA: We made recommendations with respect to either the evaluation of the fire damage impact to specific cables or to protect those cables or move them in certain areas.

And the reason I state it that way is because, if you recall, we didn't do any detailed circuit analysis, we assumed that the worst case cable damage.

So, if there is an important cable that's in a room, the first step that the COL applicant would do, I would assume, is they would just do the detail

circuit analysis on the cable and see if it really results in the impact that we assume, and if it is, then they would consider moving that cable elsewhere or otherwise protecting it with some other fire control measure.

And that was the extent of the feedback from the fire analysis.

MEMBER KIRCHNER: Thank you.

CHAIRMAN BALLINGER: Okay, we're at a convenient breaking point, both in terms of the assigned scheduled and presentation, so, I'd like to take a recess until, I don't know, somewhere between the eight and the seven.

(Laughter.)

(Whereupon, the above-entitled matter went off the record at 10:21 a.m. and resumed at 10:40 a.m.)

CHAIRMAN BALLINGER: So we're back in session. Go ahead.

MR. IN: Next topic is internal flooding PRA, and it will be covered by Mr. Ray Dremel.

MR. DREMEL: Good morning, I'm Ray Dremel, I'm with ENERCON working with KHNP and KEPCO on the internal flooding PRA analysis.

So, characteristics of the APR1400 that

have a significant impact on the internal flooding PRA. First of all, the auxiliary building has a four quadrant design.

The quadrants are very well separated, especially down low where all the water goes. So, a flood in one quadrant generally does not get into another quadrant.

They have emergency overflow lines between upper levels and lower levels. So any water that is released up above flows very easily down the lower levels.

And within rooms within a quadrant so that you don't get a significant accumulation within one room, it then goes up and gets things up above.

Also, in the auxiliary building most things are, there is very limited flood sources. Most things are cooled by component cooling water, which is a finite volume.

Relatively small volume compared to the auxiliary building volume. And, about the only major flood source would be fire protection --

MEMBER BLEY: Well, it has make-up capability.

MR. DREMEL: A hundred gallons a minute. You know, measure it in terms of 100 gallons.

MEMBER BLEY: Is the maximum --

MR. DREMEL: Yes. Yes.

MEMBER BLEY: Of course, if you run out that you got other troubles.

MR. DREMEL: Right. Right. Yes, if you lose CCW then it's like 80,000 gallons, roughly, in the whole system, make-up with 100 gallons.

The auxiliary building is designed to hold 600,000 gallons in the lower quadrant, lower volume of any one quadrant. So that's nothing.

Next slide.

MEMBER SKILLMAN: Ray, when you say that's nothing, on the one hand I might agree with you, unless the compartment area and the elevation of a critical component are such that even a minor component cooling water leak has the capability to flood that device.

MR. DREMEL: Yes, and we do consider that.

MEMBER SKILLMAN: Thank you.

MR. DREMEL: So, if we have a flood in this room, or a room the size of this desk, 80,000 gallons is significant to that, but it's not significant to going across to a different quadrant and failing another division of equipment.

MEMBER SKILLMAN: Yes. Okay, thank you.

MR. DREMEL: So, a flood induced initiating event is any uncontrolled release of any fluid that also fails PRA equipment, whether its water, steam, oil, lube oil, fuel oil, anything that releases.

And it also, we need to consider the flood, the equipment is failed by the flood has to cause a reactor trip immediately or result in a tech spec required shutdown within 24 hours.

MEMBER BLEY: I got a question for you.

MR. DREMEL: Yes.

MEMBER BLEY: Did you look, the drains from one floor to the floor below, can one actually see through them or is there an ITAAC to make sure they actually pass water?

MR. DREMEL: There are emergency overflow lines and then there are drains. The emergency overflow lines are 12 inch, I think. On the order of 12 inch diameter that just is a short stub of pipe.

I'm sure they would have some sort of OSHA required so you don't drop something down on somebody's head. So, the emergency overflow lines are pretty, it's hard to mess those up.

The drains we credited were only credited for, well, they were also credited in the design basis

flood analysis. The d over 2, t over 2. Those are a much smaller volume. Those are only typically credited with like 100 or so gallons a minute.

And they would have to be maintained for the design basis flood analysis. If the drains are capable of performing their design flood analysis, putting a little more water in them, they would still work.

So, I did not confirm there was an ITAAC item for them, but I assumed that was part of the design.

MEMBER BLEY: I've seen pieces of pipe that ought to pass water that didn't after construction.

MR. DREMEL: Yes. As I said, where we credited drains as opposed to the emergency overflow lines, they are also credited with the design basis flooding analysis. So there would have to be something to maintain the design basis as well the PRA.

MEMBER BLEY: Go ahead.

MR. DREMEL: Because the plant is not built, we can't go in and look around at it. When we are analyzing flood propagation, we would assume that a design flood barrier would prevent propagation up to

its design level, but not beyond that. And that's pretty conservative.

There were some rooms where you had a concrete wall that is a flood barrier up to six inches in height. On one of the upper elevations.

It's also a three-hour rated fire barrier. One would expect that if you went in and actually looked at the wall, but that concrete wall, because the fire barrier has no holes in it, but there would be nothing to prevent someone from drilling a hole through the, six inches off the floor, filling it with some sort of silicone foam. Wouldn't expect to see that, but we just assume those would only be, prevent propagation up to their design flood height.

If we had an actual plant, would expect it to be much higher than that.

Non-water tight doors, because we don't know what they're made of, how they're anchored right now, we assume they fail when a water level of one foot gets on one side of the door, if it makes the accident worse.

So there are some rooms where you have a door on either end, one door goes out to, say a stairwell, the other door goes to, say a switchgear room or a pump room, they're both non-water tight

doors, they're both the same door. We assume the one that fails goes into the switchgear room or the pump room. We don't take credit for the failure of the door down to the stairwell that would relief flow.

So, any barrier failures that make the event less we don't consider. We do consider flow through drains, emergency overflow lines and other open pathways.

The accident sequence analysis, based on the internal events, we have over 130 flooding events that were explicitly evaluated. Most of these events result in an accident analysis because they result in either a manual shutdown within 24 hours or an assumed reactor trip because something needed to maintain the plant operation gets wet.

For high energy line breaks, there are steam lines or other high temperature lines in the aux building. The ASME standards says you have to consider to treat them conservatively, so what we said was, the high energy line break barriers remain intact up to their design, the design size of the high energy line break.

If you have a held bigger than design, then we assume the barrier fails and everything in that rooms fails and the next barrier. So one door,

the next door fails.

If there is four doors to four different rooms all four of those doors fail. We fail everything in the next room as well as actually fire protection if there is any in that room.

So you get the high energy line break effect as well as the flooding, the water from the fire protection.

The insights is, there is no one significant event. Most of the events that are important to the flooding PRA are beyond design basis flooding events. The design basis considers a d over 2, t over 2 over breaks. We go up the double ended breaks of the piping.

I would expect the internal flooding risk would be much less if we could go in and look at the as-built plant because then we could actually look at the door failure potential, and also crediting barriers, there may not be flood barriers, but solid concrete walls that are only credited for six inches of flood height.

Most of the breaks do not require isolation because CCW is a closed-loop system, you can't isolate. Or the time available to isolate is long as opposed to several hours for after the

flooding event the operators have to turn off like CCW make-up.

And overall the flooding risk is pretty low compared to current generation plant.

MEMBER BLEY: So, the source is usually come to the cooling water, there is no service water comes in these rooms?

MR. DREMEL: There is no service water in the --

MEMBER BLEY: Fire main?

MR. DREMEL: There is fire water in the auxiliary building.

MEMBER BLEY: You looked at fire floods?

MR. DREMEL: We looked at fire water and that's essentially an unlimited source.

MEMBER BLEY: Yes. Okay.

MR. DREMEL: Yes, the IRWST is the other big source. And that's only down low. And the quadrants are designed to take the entire release of that without going over to the next quadrant. That's it for flooding.

MEMBER KIRCHNER: May I ask the same question I asked Greg?

So, based on all this analysis, is there any design improvements that you would recommend for

the COLA applicant since you don't have an as-built plant that would, are there any significant vulnerabilities?

You say there is no one significant event, but just based on what you know of the conceptual physical layout, is there any changes, any vulnerabilities that you would address as a result of this analysis?

MR. DREMEL: No, because I would expect if we actually looked at the as-built plant, the risk would be lower than it is now. The risk is already very low for flooding because of the compartmentalization.

There's always things you can do to make something better, but I don't think it would be worth the time and money to do anything. The flooding is, typically flooding is on order of the internal events. This is what an order of magnitude less than the internal events PRA. And it's treated very conservatively.

MEMBER KIRCHNER: Thank you.

MR. KIM: My name is Jaegab Kim from KEPCO E&C, I work KHNP.

I'll present low power shutdown analysis, Level 1 and Level 2. This slide, low power shutdown

analysis strategy.

Major tasks include line items. Most of the major tasks is to perform plant operating state development. The same at-power. So a detailed analysis have been performed of all plant operating state development.

Next slide. This slide, analysis philosophy of POSs. APR1400 plant operating state has been developed based on NUREG/CR-6144 and the reference plant operating experience.

Total number over POS is 18. Each POS has a specific feature. For example, primary system water level, primary system pressure and temperature and the test mode.

All right, this slide is initiating event selection. Low power --

MEMBER BLEY: I'm sorry, can you back up to the last slide? This is the full set of plant operating states for a refueling outage, did you also look at maintenance outages that don't open up the reactor coolant, the reactor itself?

MR. KIM: Pressure remains open.

MEMBER BLEY: I'm sorry, I couldn't hear?

(Off-microphone comment.)

MR. KIM: Yes. The plants operating state

has considered pressurized remain open.

MEMBER BLEY: I'm either missing your answer or I'm not communicating my question.

MR. DREMEL: The answer is, the plant operating states would occur during a maintenance outage. A forced outage mid-cycle would be the same as here.

So, pressurizer manway open, pressurizer manway can either stay hot in mode --

MEMBER BLEY: It's a subset of these?

MR. IN: Yes --

(Off-microphone comment.)

MEMBER BLEY: And my question was, did you look at the subset and the frequency of those maintenance outages compared to the frequency of the refueling outage or you just looked at refueling outages?

MR. IN: Yes, we did not look at the forced outages specifically, we only looked at the normal refueling outages.

MEMBER BLEY: Okay.

MR. KIM: All right. Low power shutdown initiating event based on industry documents in NUREG/CR-6144 and EPRI TR-1003113. Which has specifically low power shutdown initiating event.

And APR1400 initiating event were also compared to publicly available information on current and advance evolutionary PWRs.

This slide, accident sequence analysis. Accident sequence analysis models for combination of system response and operator actions that could occur during the event based on mitigation functions/systems available during the POS.

Event tree analysis is used to delineate these combinations to present these events. This table is one example for POSC.

This slide, success criteria. Test is submitted to summarize the analysis has been performed using MAAP 4 and RELAP5. Which is considering to be at-power.

Considering the initiating event, limiting plant conditions for each POS and equipment availability specified for each accident sequence.

MEMBER BLEY: I'm sorry, I want to interrupt you again. The event tree you showed is for the loss of shutdown cooling, which I assume means loss of flow in the shutdown cooling system. Or does it also mean draindown of the --

MR. KIM: No.

MEMBER BLEY: No. Did you look at

draindown events, pumped out events?

MR. KIM: Yes.

MEMBER BLEY: Okay. That isn't clear from the slides, but, okay, go ahead.

MR. KIM: Draindown event considered other PRAs.

(Off-microphone comment.)

MR. KIM: Yes. Draindown event also considered.

And the core damage is defined based on the ASME PRA standard and NRC Inspection Manual Chapter 0609. Which is defined peak temperature for fire department is 1340 Fahrenheit.

This slide, LPSD Level 2 analysis. For POS with RCS and containment intact, Level 2 conservatively estimated in using the full power conditional probability of large release.

For POSs with RCS intact but containment hatch open, are failure to close the hatch assumed to be large release. A successful closure of hatch before boiling evaluated using full power conditional probability of large release.

For POSs with RCS head removed, detailed Level 2 PRA are developed. For portions of the analysis, the full power Level 2 models are

conservatively used as bounding estimates.

LPSD Level 2 fire modeling are same as internal events.

LPSD insights. Results are dominated by failure of operator actions. Closure and some POSs is not available for steam generator path.

Results indicate that the draindown and reduced inventory POSs are highly risk significant.

MEMBER BLEY: The one thing I want to make sure I understand. For the conditions with the RCS intact, you really did no analysis, you just assumed that everything that can happen at full power happens here, is that right?

MR. KIM: Yes.

MEMBER BLEY: Okay.

MR. DREMEL: There's RCS intact in Mode 5 and there's RCS Mode 3, hot standby. We assumed everything could happen. Everything that happens at-power happens in shutdown.

But RCS, I just to want to make sure we didn't mislead you, when you said RCS intake, you can be at 200 degrees and the RCS is still intake, that is considered in the shutdown, separately.

MEMBER BLEY: But you didn't analyze that condition?

Your slide said, just intact, it didn't make the distinction you're making now, but when you're in shutdown, your only, I don't know what that slide meant. Go back two slides or three.

MR. DREMEL: Is this the one here?

MEMBER BLEY: Yes. First bullet. The way I read that first bullet is, for the cases and the time period during which the RCS is intake and you're not at-power, you assumed that the results from the at-power PRA apply during that same time period. That's how I read that bullet. Is that the way I should read that bullet or does it mean something else?

MR. DREMEL: Jeff?

MR. LEARY: Yes, Jeff Leary with ENERCON. Yes, that's correct.

MEMBER BLEY: Okay.

MR. LEARY: Your interpretation is correct.

MEMBER BLEY: And your belief is that's pessimistic? Okay.

MR. ROZGA: Any other questions on this section?

MEMBER BLEY: Can you give me a --

MR. ROZGA: Sure.

MEMBER BLEY: -- a brief statement of what, oh no, I've got that. Never mind, I don't need that.

MR. ROZGA: All right. I believe this is the final section in 19.1, is that true, Yong?

MR. IN: It is, Jeff.

MEMBER BLEY: Yes.

MR. ROZGA: This is on the 2017 PRA update. And, again, this is Greg Rozga from ENERCON.

For the 2017 update, all of our data was updated so the initiating event frequency was updated. Based on the NRC website, other NUREGs, vendor data and a specific calculation for ISLOCA, based on the identified paths and the updated data for the failure rates, for the components leading to an ISLOCA.

The data updates, including the CCF data, is based on availability, reliability and common cause failure on the NRC website. The HRA dependency analysis, well, the HRA analysis, including the dependency analysis was revised.

There were some design changes that were incorporated into the model, and there were other model changes that were made based on responses to RAIs. And just other model error corrections that could have come from the peer review or just other

internal review, self-identified errors.

The HRA, so now, all independent HEPs for all modes and all models are within the HRA calculator. And there's been a full dependency analysis for each model, including the dependency between Level 1 and level 2 human failure events.

We also applied a minimum joint HEP of 1E-06. So when there are multiple HEPs, even if the dependency analysis that there combination would be 1E-7, we backstopped that and said it can't be any lower than 1E-06.

MEMBER BLEY: Within the HRA calculator, which isn't a model of itself, which models did you use?

MR. ROZGA: Which?

MEMBER BLEY: Which models that are present in the HRA calculator did you use for which kind of, human events?

MR. ROZGA: Depending on the event, there was the CBDTM, there is ASEP. And then there were some HEPs that were just based on a screening value. And I think those were the three main methods.

MEMBER BLEY: That you used, okay.

MR. ROZGA: Yes. And it's, the HRA analyst determines which one is appropriate.

MEMBER BLEY: Okay, that's good --

MR. ROZGA: Okay.

MEMBER BLEY: -- for now.

MR. ROZGA: The ATWS model was changed in the original REV. 0 PRA. The original ATWS model was based on a single data point and didn't consider the impacts from turbine trip or the number of POSRVs available to relieve RCS pressure.

Since that time, that analysis has been done, we've incorporated those results. The unfavorable exposure time percentages are now incorporated in the model and they are impacted by whether or not you have successful turbine trip and the number of available POSRVs for pressure relief.

The RCP Seal LOCA model has been updated. And what's in the PRA is based on WCAP-18067-P Rev. 0. And that's for the KSB Type F, RCP seal failure model.

The potential for RCP seal LOCA is included in station blackout, consequential station blackout and both partial and total losses of CCW or ESW.

It does consider the reestablishment of seal injection via the auxiliary charging pump, that has to be done within 20 minutes. The auxiliary charging pump is independent of any CC cooling. So if

you lose all seal cooling or if you lose all CCW, you can still start the aux charging pump to reinitiate seal injection.

MEMBER BLEY: As I recall, that 20 minutes is based on potential damage to the thrust bearing, is that right?

MR. ROZGA: I, Ray, you know more about seals, is this about --

MR. DREMEL: I would have to refer to --

MEMBER BLEY: At one point you talked about, in the past, or the report did, that you can live without the seal cooling, component cooling water, except for the thrust bearing. And I thought that was the basis for the 20 minutes.

MR. DREMEL: I would have to defer to someone who knows about the seal model better than I do.

MR. ROZGA: We don't have those details up here. There is someone here, I think, that can answer that question. But it's neither Greg nor I.

MEMBER BLEY: Okay, Rob.

(Off-microphone comment.)

MEMBER BLEY: Okay. We can do it later.

MR. ROZGA: Okay, we can come back to that then. All right. The model also says that the RCPs

must be tripped within 20 minutes to prevent seal LOCA and non-SBO sequences.

And even if the RCPs are successfully tripped, there is a possibility of seal LOCA. And that conditional seal failure probability is based on the model that's developed in the WCAP.

And there are some issues that impact that probability. And those include whether or not the control bleed-off has been isolated within 20 minutes, depends on the amount of RCS subcooling and it depends on the duration of the event.

So the PRA, we assumed a 24 hour mission time. For the RCS subcooling, we assumed the worst case, which was, I think it was less than 50 degrees Fahrenheit. I would have to verify that in the WCAP. And then we evaluated cases where both the CBO was isolated within 20 minutes and when it wasn't.

MEMBER BLEY: It was also discussed earlier, I think at the earlier meeting, that there is essentially no operating experience with these seals, right?

MR. ROZGA: Again, that's a question for the RCP seal --

MEMBER BLEY: With another new seal design that seemed to only make improvements, it turned out

that wasn't true when we had the real seals out in the operating world for a few years.

MR. ROZGA: Yes. And I think that this next slide, we know that those conditional seal failure probabilities are based on a model. I personally don't know the operating, any operating experience behind them.

However, we understood that this is an area where we're should do a sensitivity analysis. And so we increased the conditional seal failure probabilities.

And we did it for both the case where CBO was isolated and where it's not. We just set the conditional seal failure probability to 1.1 and .1. And we really don't see a significant increase until maybe they get to a value of one and ten.

And the seal failure probabilities, depending on all the conditions, is somewhere in the 2 to 3E-4 per pump range. So they would have to increase by a couple orders of magnitude before we would start seeing a significant increase here.

And so therefore comfortable using the value that's in the NUREG. And we understand that even if that value was off by a factor of two or a factor of five or a factor of ten, it would not

significantly impact the results of the PRA.

MEMBER BLEY: That's slightly comforting. At the last meeting on these seals, John Stetkar asked you a question, which I'm going to repeat --

MR. ROZGA: Okay.

MEMBER BLEY: -- a little later on more general results. But he said, why didn't you account, you or Westinghouse, whoever did these calculations, account for the state of knowledge correlation, also known as data dependency on these events.

And he did a calc to convince himself you did not do that yet. Do you have an answer now or have you thought about that anymore?

MR. IN: This Yong. I believe those were discussed during the closed session.

MEMBER BLEY: Do you want to save that for closed session?

MR. IN: Yes.

MEMBER BLEY: Okay.

MR. IN: Except, we weren't part of that discussion.

MR. DOUGLAS: Well, I can, I mean, this is Gary Douglas, Westinghouse PRA engineer. I worked on the seal model.

MEMBER BLEY: Okay.

MR. DOUGLAS: We did an update to the model --

MEMBER BLEY: Since our last meeting?

MR. DOUGLAS: Right. Since our last meeting.

MEMBER BLEY: Okay.

MR. DOUGLAS: State analysis correlation. And made some other adjustments to the model based on the comments from the last meeting as well.

MEMBER BLEY: Oh, okay.

MR. DOUGLAS: So, in the end, a slight increase in the numbers, but, the overall conclusion is unchanged.

MEMBER BLEY: Okay. I'm not sure, did we get that change?

(Off-microphone comment.)

MEMBER BLEY: Okay. Maybe we'll bring it up again if we have a closed session. Thank you. I'm glad you did that.

MR. ROZGA: Another change that was made to the model is, we included software common cause failures into the digital I&C models. The hardware failures in the plant protection and the ESF component control system is still from the reference plan. However, we've incorporated PPS software CCF events.

They're added directly to the model.

There is an application software common cause failure for the bistable LCLs and the group and loop controllers. And those individual application softwares each fail all of their component types.

So it's a true common cause. So the bistable module application software failure fails all of the bistables.

We also incorporated a PPS operating system common cause failure, which I liken it to Microsoft Windows. You have all of your individual applications that they could fail. The operating system would fail all of them.

And we did a similar modeling for the diverse protection system. The hardware failures are still a hardware failures from the reference plant, however, DPS software, CCF events were added to the existing model.

And the PPS and DPS software were assumed to be independent based on DCD Chapter 7, which states that the PPS and DPS are designed using different hardware and software to address postulated common cause, software common cause failures.

So therefore, although we have, we could have common cause failures which fail PPS and we could

have common cause failures which could fail DPS, we did not assume that there was a common cause that would fail both of them. They have different software architecture and other things that prevent that.

The operating system common cause failure, the value used was 1.2E-06 per demand. And for each application software it was 1.2E-05 per demand. And those were based on vendor information.

And again, these are --

MEMBER BLEY: I've got to ask, what kind of basis is there for that vendor information? Do you have any idea or do they just tell you these numbers?

MR. ROZGA: No, there is a, is it in WCAP? There is a WCAP where there is a risk analysis which goes through all the analysis to come up with what those results are. And we understand that this is a analysis that was done and therefore we, again, felt it was prudent to do sensitivity analyses on those numbers just so we understand the impact of those numbers on the PRA model.

MEMBER BLEY: Can you tell me which WCAP that is that works to give us the common cause contribution? Did you have that in one of the --

(Off-microphone comment.)

MEMBER BLEY: Thank you. Was that one

that was submitted with the application?

MR. IN: I'm not sure, but we'll get you that WCAP model.

MEMBER BLEY: Okay. Number and if it was submitted.

MR. IN: Yes. All right.

MEMBER BLEY: Thank you.

MR. ROZGA: So we did several sensitivity cases and we performed those sensitivity cases by increasing either the operating system or the application system or several of those, by factors of ten and factor of 100. And we just wanted to understand how sensitive the model was to the numbers that were used.

And, the case where we increase everything by a factor of ten, we had a CDF increase of about 39 percent. And that's getting to be relatively significant.

We're still at 1.5E-3. That's case two. Which still meets the NRC goals for reactor safety. We start seeing much bigger increases when we go by a factor of 100.

When we look at the individual application softwares, what we find is the most important are the loop controllers. Above the loop controllers there

are several places where you have manual operator actions to recover for a group controller failure.

For in a reactor trip, there's manual reactor trip. There's things that, of course, the loop controllers don't impact reactor trip, but if you're looking at just bistables and the LCL modules, you still have manual reactor trip and you have the diverse protection system.

Over, when you get down to the loop controller, then your recovery probabilities get limited. So, that's kind of an expected result. We see a similar result when we fail the operating system software.

And, in Case 11 we see the importance of the diverse protection system. If we did not have the diverse protection system, the CDF increase about 400 percent and the LRF increased over 500 percent. So, the DPS is a very important system to the APR1400.

MEMBER MARCH-LEUBA: So in these software failures, you're modeling failure to scram?

I mean, you said probability per demand, so --

MR. ROZGA: Correct.

MEMBER MARCH-LEUBA: -- is that most right?

MR. ROZGA: Yes. Yes.

MEMBER MARCH-LEUBA: Because, software failures tend to build up with a break in time. I mean, if you keep your computer up for two months something bad is going to happen.

And they can start doing something similar while you fire it. It can start opening valves where they're not supposed to be open. And I assume you didn't assume that.

MR. ROZGA: No, we looked at --

MEMBER MARCH-LEUBA: Misfiring. Yes, you just assumed failure to scram, you did not --

MR. ROZGA: Correct.

MEMBER MARCH-LEUBA: -- they're misfiring --

MR. ROZGA: Correct.

MEMBER MARCH-LEUBA: -- which, would be probably more likely than failure to scram.

MR. ROZGA: Understand.

MEMBER BLEY: In the SER, I want to read you a sentence that the Staff said because you didn't go through this. Sensitivity studies on the Digital I&C, they also evaluated the importance of operator action to overcome common cause failure. Can you tell us about that? I don't think you have a slide on

that.

MR. ROZGA: No. No, these sensitivity cases are just with respect to the failure of the individual common causes, but inherently, since operator actions can overcome failure of, for instance, group controller application software.

If there is a common cause failure in the bistables, that can be overcome by manual action to trip the reactor or to initiate SI.

MEMBER BLEY: And you did an analysis of that or you just qualitatively looked at those?

MR. ROZGA: Those operator actions are in the PRA.

MEMBER BLEY: Oh, they are? Okay.

MR. ROZGA: Yes. Correct.

MEMBER BLEY: I missed those. Okay, thank you.

MR. ROZGA: Yes. There was some other system changes. For the aux power system, the main change was that the auxiliary AC power source, which is a gas turbine, can only power trains Alpha or Bravo.

In the original model, it was assumed that it could support Alpha and Charlie at the same time, or Bravo and Delta. So that change was made.

There were also AC power changes to the service water cooling tower fans and valves due to design change, which we'll talk a little bit about later. There were power supple changes to various equipment.

And because of those changes, there were some assumed cable routing that we had to do for the fire PRA.

For the instrument air system, it changes from a one train three compressor system to two complete trains. The instrument air also no longer supports service air, so you no longer have that potential flow diversion path. Service air now has its own compressor and it's not even incorporated in the model.

MEMBER MARCH-LEUBA: So, these changes were incorporated, are they assigned based on the PRA results?

MR. ROZGA: No, these were, well --

MEMBER MARCH-LEUBA: I mean, at least they were informed by the PRA results? You don't know?

MR. ROZGA: Yes, I don't know the flow path, exactly when they happened.

MEMBER MARCH-LEUBA: Are these changes good?

MR. ROZGA: Yes. Yes.

MEMBER MARCH-LEUBA: So they were informed by PRA, right?

MR. IN: This is Yong. Yes. Some were, some changes went by the PRA results. From the Rev. 0. And the, I believe the design change was associated with the SX, was not from the PRA, it was from something else.

When they went from MOV to HOV. That wasn't where they powered the PRA.

MEMBER MARCH-LEUBA: Yes, but I'd like to emphasize that out of all these studies, those changes improve the safety of the reactor. They make the reactor better. All the other calculations, they just calculate the number. So these are good.

MR. ROZGA: Yes.

MEMBER MARCH-LEUBA: This is a good outcome of this study.

MR. ROZGA: So, for the change in instrument air the updated model decreased the instrument air failure probability by almost a factor of five.

However, the overall impact on the CDF actually was relatively small, and that's because instrument air has a small role in the Level 1 PRA.

So the increase in reliability was insignificant with respect to CDF.

I did not evaluate it with respect to LRF, I was just looking at CDF at the time.

The essential service water system, there were changes. And again, this whole part of the SX system, which is the cooling tower, this is all conceptual design. This part is going to end up depending on site-specific, whether you're going to draw water from a lake or whether you're going to have cooling towers.

But the change included changes to the number of fans in each cooling tower, changes from MOVs to hydraulic operating valves for the cooling tower inlet and bypass valves. And, again, it required some assumed routing of cables for the fire PRA.

And the last main system change was the auxiliary building, HVAC for the radiologically controlled area.

The original PRA, I assume that there were only two, there was an Alpha and a Bravo, and they supported both divisions, all four trains, of pumps in a containment spray, CV, SC and SI. The new design, they actually have two separated divisions and there

is two trains in each division.

The one other difference was that the CC pump rooms are now only cooled by the cubicle coolers, they don't have a redundant supply from the VK exhaust.

MEMBER SKILLMAN: Greg, before you change. At the last meeting, our colleague John Stetkar raised the question about review of the calculations for the cooling of the compartments.

MR. ROZGA: Yes.

MEMBER SKILLMAN: And in my review, I did not see any new information. What review has been provided to confirm that the compartment cooling capability is as it needs to be?

MR. ROZGA: There was a calculation, the room heating calculation. That's a, was that a KHNP or a KEPCO?

MR. IN: Yes, there was a room heat-up calculation done for each pump room and the electrical equipment room.

MEMBER SKILLMAN: Have those calculations been submitted on the record? Are these internal calcs or are these calcs that have come over for review by the staff?

MR. IN: I believe it was, it was not

submitted as a, I don't think it was in document, however, your staff auditor --

MEMBER SKILLMAN: Oh, then I'll save this question for the Staff then.

MR. IN: Yes.

MEMBER SKILLMAN: Thank you.

MR. ROZGA: Other model changes and corrections that had some impact on the model was, there was an NRC review comment about modeling of the consequential loop. We have now corrected that.

There was a self-identified ISLOCA path that we somehow missed in the original analysis during our subsequent reviews. We identified it.

However, it turned out that ISLOCA frequency did not change because there were other paths that were still dominating. However, this path has now been added.

And we also added an evaluation of fire and floods on local manual actions to unlock the POSRVs for feed and bleed operation. They're normally, they have the breakers racked out to prevent fire induced spurious operation.

And I believe that one came from an NRC, from an RAI as well. It was identified and so we went back and incorporated that information.

Now, the results. There were numerous competing changes to all the models. There was a decreased in the impact from the RCP seal LOCAs in the ATWS because prior to having those analyses reports done, we just had single point estimates at what we thought was a conservative value. When we put the actual numbers in, it actually resulted in decreases.

We had a general overall increase due to a more detailed HRA dependency, especially going from Level 1 to Level 2, to make sure that we captured all those dependencies.

Some of the data increased and some of the data decreased from the last update. Again, our data is based on generic sources and so we had some increases in CDF because of that.

And then there were also some increases and decreases due to some of those system changes that we just talked about.

On the next slide I'll be presenting some, or no. There were some major changes made to the flooding model due to correction of some self-identified errors.

There were some input files that had wrong information in them, and we identified it and corrected them. However, the result is that both of

the flooding impact for both at-power and low power shutdown, they're still very small even though we identified these and made the correction.

And then for the at-power Level 2, performed a full HEP dependency analysis between Level 1 and Level 2 and the Level 2 and Level 2 actions. The Level 2 interface with Level 1 was updated to accommodate the new Level 1 event tree changes.

For example, in the RCP seal LOCA we now have new event trees.

And Level 1 changes carry through to Level 2. So notable impacts to the LRF are, the data updates, especially the conditional probability of loop after a steam generator tube rupture.

Initially, that conditional probability of loop, we misread it and we segregated it between LOCAs and transients, however, it should have been separated between those initiators that result in an SI and those that don't result in SI.

And there's a lower value if you don't get an SI. So initially, the tube rupture wasn't considered a LOCA and therefore it was binned in the lower conditional probability category. It's now in the upper conditional probability category. So that resulted in an increase in the Level 2.

MEMBER MARCH-LEUBA: So, you made me curious. What use is the loss of offsite power because you break a tube? I mean, I what's the --

MR. ROZGA: Well, the thought is, is that when you have a trip you can perturbate the grid and that can result --

MEMBER MARCH-LEUBA: Yes, but that would be with or without breaking the tubes.

MR. ROZGA: Correct. There is two values that are used. One of them is, if you just have a normal trip and that value is about 2E-3. And then the other is, if you get a trip within SI and the --

MEMBER MARCH-LEUBA: What was an SI?

MR. ROZGA: With a safety injection signal.

MEMBER MARCH-LEUBA: Okay.

MR. ROZGA: Actuation. And the thought being is that, in addition to the perturbation of taking the plant offline, you're also putting all this other equipment, you're loading it on because of the SI signal and therefore the value --

MEMBER MARCH-LEUBA: The SI is 100 percent passive. You just open the valve and the tides go in.

MR. ROZGA: No. At APR1400 the SI pumps are motor-driven.

MEMBER MARCH-LEUBA: Don't we have SI tanks? I'm confused. The one --

MR. ROZGA: Yes. With actuation of a safety injection signal. So when you --

MEMBER MARCH-LEUBA: Oh, safety injection pumps.

MR. ROZGA: -- get an SFAS signal.

MEMBER MARCH-LEUBA: Okay.

MR. ROZGA: Right. So when you get your SFAS signal, all your SI pumps start your, all your pumps start. So, in addition to the perturbation on the grid just because of the plant going offline, there is this other equipment that's being loaded on it.

There not really any strong basis for the number, it's one of those things where we think it's going to be something higher and so they increased it by a factor of ten. And so that's the number that, yes.

The other Level 1 changes that carry over through to Level 2 that had a noticeable impact was the RCP seal LOCA modeling. And then the addition of the software CCF on digital I&C.

For the low power shutdown Level 2 changes, again, there was a full HEP dependency

analysis made the emergency containment spray backup system consistent with the at-power modeling.

The previous modeling had some non-conservatism. I believe it was just a point estimate it wasn't fully flushed out.

The low power shutdown fire Level 2, there were a couple of flag files which we identified some incorrect settings. We corrected those. One resulted in the reduction in LRF and the other resulted in an increase in LRF. And I think those approximately offset each other.

Now here is that final slide. The point being that the changes were from a variety of sources. There were some design changes that were incorporated. There were some self-identified errors and there were some errors identified by the NRC during the RAI process, and those were incorporated.

When we look at the results, you can see Rev. 0 which is documented in Rev. 1 of the DCD. And PRA Rev. 1 is what's in DCD Rev. 2.

And you can see that the results are, they're not huge on a percentage-wise. If you look at the values. The ones where you'll see the large increases are in the internal floods and those were those two errors that I identified earlier.

However, even with those large increases there is still a much smaller portion of the CDF, of the overall CDF and LRF.

MEMBER BLEY: Greg, I've been waiting for this slide because --

MR. ROZGA: Okay.

MEMBER BLEY: -- because I didn't know anywhere else to raise this issue.

In the Staff's SER they quote your results from Rev. 0 I guess, and the mean value for internal events is shown here, 1.3 minus 6. But the range, from the fifth to the 95 the is only a factor of ten. It's a pretty tight distribution.

Two questions about the tightness of that distribution. The first is, the same issue that John brought up on the pumps, the state of knowledge correlation, especially when you have fourfold redundancy so now you're getting X to the fourth kind of terms, that makes a big difference, especially you get threefold and twelvefold redundancy.

Did you account for that in the quantification?

MR. ROZGA: Yes.

MEMBER BLEY: You did?

MR. ROZGA: Yes. In the CAFTA, the way

that CAFTA is setup --

MEMBER BLEY: It ran the full distributions?

MR. ROZGA: Yes.

MEMBER BLEY: Okay. That then isn't one, although usually that gives you a broader distribution.

The other is, you did a group of sensitivity analyses, some you've gone through here, I think there was some other ones before that. In a final PRA, and this is a mix of final and preliminary, so I'm not pushing real hard on this one, but the one before fuel load, should there ever be one, the main function, to me, of doing the sensitivity analyses is to reflect that back in to the uncertainty analysis for the PRA.

And those areas where we don't have solid data where we might be off because we don't have a long history with this kind of equipment, I don't think you did that. I think the sensitivity analyses are separate from the uncertainty.

It looks to me like the uncertainty distribution is a parametric uncertainty calculation.

MR. ROZGA: Correct.

MEMBER BLEY: That's correct, right?

MR. ROZGA: Correct. Yes.

MEMBER BLEY: If later there is a COL in a real plant, those things ought to be reflected into the uncertainty distribution for the PRA not just sitting there as sensitivity analyses. My opinion any way.

MR. ROZGA: Understand. The risk insights from the PRA update, the overall results are similar to the previous results. The overall CDF decreased about 12 percent but the LRF decreased about ten percent.

And then when we went back and reviewed what the design changes were and the data updates, it was consistent with what we would have expected to see. Based on the PRA update, the risk contribution from RCP seal LOCA is low and the risk contribution from the software CCFs of digital I&C is low.

And that last one is mainly because of the diverse protection system. If you remember on that slide where we failed the diverse protection system, the CDF went up about 400 percent and the LRF went up over 500 percent.

MEMBER KIRCHNER: Let me ask Dennis' question again. So what uncertainty is on these numbers?

You know, you've made these many changes to update the PRA, but what kind of uncertainty would you put on this, is the uncertainty greater than the changes?

I didn't say that well in a statistical space but do you see where I'm getting at?

MR. ROZGA: The, I'm not sure, but let me answer then --

MEMBER KIRCHNER: Well, let me just take --

MR. ROZGA: -- and then you can tell me if I'm answering --

MEMBER KIRCHNER: Well, let me take the numbers are face value.

MR. ROZGA: Okay.

MEMBER KIRCHNER: The CDF went down 12 percent but the LRF went up ten percent. Well, as far as the public is concerned, it's the latter one that's an issue and therefore you did all this and it went up ten percent.

I don't want to say the risk went up ten percent, that's a misinterpretation. So, what I'm getting at is, okay, you have done this, how do you feel about it. And so what's the uncertainty on these numbers?

MEMBER MARCH-LEUBA: Yes. And --

MEMBER KIRCHNER: And should we be concerned that, because, this is fairly, I'm thinking ahead to, not what you're presenting today but other applications of PRA for designs that will be much more in the paper state than mature technology, evolutionary technology like this.

So, you've done all this analysis, you found model, places where the models could be improved or there were mistakes in physically representing the plant, et cetera, et cetera, and then you come up with these numbers. So, what's the takeaway from this, should I be concerned or should I be saying, hmm, you made these "improvements" and now I see a ten percent increase in the LRF?

MR. ROZGA: Yes. Yes, that ten percent increase in the LRF, and Jeff can verify this as the Level 2 analysis, but it's my understanding a lot of that came from the changing of the conditional offsite power probability for tube rupture where initially it was grouped in that 2E-3 and now it's up in order of magnitude. And I think that was the significant impact.

So that really wasn't, that wasn't a change in the plant, there was no design change that

resulted in that, that was just an initial error in our analysis that's been corrected.

MEMBER KIRCHNER: Okay.

MR. ROZGA: So, to say that the LRF went up ten percent, the LRF went up ten percent, not because of any of the design changes we made, but because of an analysis error that's been corrected.

MEMBER KIRCHNER: Okay, so then what confidence do I take in this that you found some modeling errors? This is a very model obviously so --

MR. ROZGA: Yes. It is, it's a very complicated model. It's gone through several reviews. Portions of the model, including the internal event, Level 1 and Level 2, went through an independent peer review and from that independent peer review there were issues identified, which have been addressed.

For those parts of the model that were not done when that independent review was done, we've done internal, we call them gap analyses. But basically, we did the same thing, we took the ANS/ASME standard and the low power shutdown draft standard, in fact I believe that is still a draft, and we evaluated to each one of those points.

Overall, I am relatively confident in the numbers. I understand, as a PRA analyst, that there

is tens of thousands of input that go into these.

MEMBER KIRCHNER: Yes.

MR. ROZGA: And a more mature PRA is generally always better because there is more time that you've used it and you have identified these, but I believe we have gone through sufficient reviews that I feel very confident that the numbers here are representative.

And in fact, in some instances, I would say that they are, I feel comfortable saying that they're conservative. For example, in the fire PRA where we took no credit for fire modeling, we assume that every fire, no matter how small, resulted in full room burnouts of all the rooms, we assumed all the cables failed in the worst --

MEMBER KIRCHNER: Okay.

MR. ROZGA: -- there could be a cable for a switchgear room that really just went to a chart recorder, but we knew that it was a cable for that switchgear and so we assume that that switchgear failed. So, I feel comfortable and confident that the numbers we have are, show that the risk is low.

MEMBER KIRCHNER: Okay.

MEMBER MARCH-LEUBA: I'd like to pose the same question a different way. When you say the ten

percent difference in the LRF is statistically significant, by that I mean, assuming you have infinite amount of money on time and you get ten different group, teams of people.

You get a Korean group that makes their own PRA, a U.S. group that makes their own PRA, a French group that makes, you get then of them, will the results be within ten percent of each other?

MR. ROZGA: They'll probably be within the 5/95 percent. I don't know if they will be within ten percent.

MEMBER MARCH-LEUBA: I will say a factor of two if you get lucky. So, I mean, ten percent is not significant. You're even going to say it went up.

MR. ROZGA: Yes. And that's why we bring in the independent peer review because, yes, you get two PRA analysts in a room and ask them what the right answer is, you got a 50/50 chance of getting it right.

PARTICIPANT: No you don't.

(Laughter.)

PARTICIPANT: Only 50?

MR. ROZGA: No, it's not that great.

(Laughter.)

(Off-microphone comment.)

MEMBER MARCH-LEUBA: Well, he means he's

got two people. I would assume there will be three opinions.

MR. ROZGA: Yes. Any other questions?

MEMBER BLEY: Well, I just want to make a comment following up on Jose's. In fact, the confidence in the numbers, that ten percent is a small thing, but given the kinds of changes they had, those were all changes, the predominance of them pushed in the upper direction so it almost surely went up.

Maybe it didn't go from where it was to where it is now, it might have gone from some other number to a little higher number, but it was in the up direction.

MR. ROZGA: Correct. All right.

CHAIRMAN BALLINGER: I have a question, which I'm going to ask the staff as well. On the current status slide, on my thing it's slide Number 76, I seem to have missed it somewhere.

MR. ROZGA: Oh, I didn't get there yet.

CHAIRMAN BALLINGER: Oh, okay.

MR. ROZGA: Yes, I'm coming.

CHAIRMAN BALLINGER: Oh, it's coming?

MR. ROZGA: Yes.

CHAIRMAN BALLINGER: Ah, I'll wait.

MR. IN: Yes, if there is no, this

concludes all the technical presentation that was presented by the seven presenters. I mean, the six presenters on the seven different topics.

And the last slide is the overview of Chapter 19.1. And basically I have one slide, which is the current status.

And the Chapter 19 is complete and we are in the process of going, DCD Rev. 2 was issued and now we're preparing for the Rev. 3.

And there was 14 open items from Phase 3 review and these have been all resolved with sufficient discussions with the Staff.

And most of the mark-ups were incorporated into the DCD Rev. 2, except the mark-ups associated with the RAI 19.1-10 and 19.1-85. And these are the, these are related to the PRA based SMA mark-ups. And that will go into the Revision 3.

MEMBER BLEY: And we'll --

CHAIRMAN BALLINGER: Okay, so now I might ask a question.

MEMBER BLEY: -- have that by next month, right?

CHAIRMAN BALLINGER: Well, that's my question. What DCD revision are we writing the letter to? Because it makes a big difference.

MS. HAYES: I would say, we wrote the SER to Rev. 1 plus the mark-up, the confirmatory items as marked up in those two RAI responses.

CHAIRMAN BALLINGER: So, we're writing a letter to DCD Rev. 1?

MS. HAYES: Plus the proposed changes in the RAI responses that are identified as confirmatory items in the SER.

CHAIRMAN BALLINGER: Which they just said that all the confirmatory items are in Rev. 2.

MEMBER BLEY: Except these two.

MS. HAYES: No, those --

CHAIRMAN BALLINGER: Except these two.

MEMBER BLEY: So, if we write to Rev. 1 plus what we hear today --

CHAIRMAN BALLINGER: Is that what you're saying?

MEMBER BLEY: -- that should be equivalent --

CHAIRMAN BALLINGER: That's what I'm trying to get at.

MEMBER BLEY: -- to Rev. 2.

MS. HAYES: Rev. 1 plus the DCD proposals and the RAI responses. I think you have these RAI responses.

CHAIRMAN BALLINGER: I'm sure we do, I just need to be sure that we know exactly what we're, our universe is.

MS. HAYES: Hanh, did you want to clarify something?

MR. PHAN: Hello, my name is Hanh, I'm the lead reviewers for PRA for the accident. Please write your letters using the Revision 2, because that's the Staff, information that the Staff used for our accepted evaluation without open items.

We received that version just about a month before we finalized our SER. But that reflects in our final SER.

CHAIRMAN BALLINGER: Okay, so it's Rev. 2 minus this RAI.

(Off-microphone comment.)

MR. BROWN: Hey, Ron, we meet informally with Bill tomorrow, we can handle this offline.

CHAIRMAN BALLINGER: Yes, yes. Okay, I'm still confused.

MR. BROWN: Yes, we'll get it right.

CHAIRMAN BALLINGER: Okay, this is, unless there are other questions --

MR. SISK: This is Bob Sisk, Westinghouse. Before we wrap-up this section, we indicated we were

going to get back to Dennis on the RCP seal question --

CHAIRMAN BALLINGER: But you also --

MR. SISK: -- I'd like to circle back around on that if we have --

CHAIRMAN BALLINGER: Somebody also said that it might be proprietary?

MR. SISK: I think --

CHAIRMAN BALLINGER: It might be closed.

MR. SISK: -- after looking at it I think we can, depending on where ACRS members go, I think we can address this in an open session.

CHAIRMAN BALLINGER: Okay. Okay.

(Off-microphone comment.)

MR. DOUGLAS: Gary Douglas again, Westinghouse, PRA engineer.

Just addressing the 20 minute question assumption made in the PRA modeling has a basis that's bored out in operating experience. The assumptions are that if you continue to operate reactor coolant pumps without cooling, when we have loss of seal cooling, in general practice you'd have more than 20 minutes but we choose 20 minutes conservatively.

In some of the analysis and testing and operating experience that has occurred, that number is

further out. Thirty minutes, 45 minutes.

There have been cases and no damage has occurred once they ended up securing the pump. And then investigating and inspecting.

MEMBER BLEY: But you have to secure the pump.

MR. DOUGLAS: So the assumption is we have seal damage at the 20 minute mark, and that's the number that we --

MEMBER BLEY: But there is some point, is it earlier, when you have to secure the pump to avoid damage to other bearings? Like the upper radial bearing and the thrust bearing. It's got no cooling water.

MR. DOUGLAS: No, no.

MEMBER BLEY: The oil has got to be heating up.

MR. DOUGLAS: The oil, it is heating up yes. That's more of a motor question. And that's also is --

MEMBER BLEY: Well, it might be motor question, but if you wipe the bearing it's a pump question.

MR. DOUGLAS: Yes. Right. And the assumption is still have 20 minutes till it holds, for

that case.

MEMBER BLEY: Okay.

MR. DOUGLAS: Yes.

MEMBER SKILLMAN: When you said no cooling, does that imply concurrent loss of component cooling water and seal injection or only loss of component cooling water?

MR. DOUGLAS: This is for the loss of seal cooling which is both the total loss of cooling.

MEMBER SKILLMAN: So, it's no seal injection plus no component cooling water?

MR. DOUGLAS: Correct. We can operate indefinitely with loss of either.

MEMBER SKILLMAN: Got you. Okay.

MR. DOUGLAS: That's all. That's all, yes.

MEMBER SKILLMAN: I'm just making sure that's clear.

MR. DOUGLAS: That's correct.

MEMBER SKILLMAN: Okay, thank you.

MR. SISK: I've got one other comment I'll make just to your question. There was a question also on the state of knowledge correlation, and that information was incorporated into internal reports. We did do an evaluation based on the comments from

Committee Member Stetkar back on the December 6th meeting.

He had a couple of items that he raised and we took those questions back. I can show you a quick note but we basically incorporated that internally.

It was determined that, specifically through the state of knowledge correlation as indicated by Gary, that the change, slight, had no impact on the conclusions of the body of the report, so we captured that. There is no change or conclusion at Korea, as we speak. For basically records to be completed.

CHAIRMAN BALLINGER: Other questions? Okay, then we will recess until the big hand is on the 12 and the little hand is on the 1, otherwise known as 1 o'clock.

PARTICIPANT: On that clock.

CHAIRMAN BALLINGER: On that clock.

(Whereupon, the above-entitled matter went off the record at 11:50 a.m. and resumed at 1:03 p.m.)

CHAIR BALLINGER: Okay, we're back in session. And the floor is yours.

MR. STECKEL: Thank you very much. My name is Jim Steckel, I'm the project manager for this

1 particular chapter, PRA chapter of the KHNP design
2 control document. And we're here to talk about 19.1
3 today.
4 And to my right is Han Phan and to his
5 right Courtney St. Peters, and to her right is Odunayo
6 Ayegbusi, otherwise known as Ayo. And we'll have
7 others who will come in and out as we go through the
8 different subsections of this section of the chapter.
9 It's been a pleasure working with
10 everyone, including the client. It was a lot of work.
11 We got through quite a few things, and they met us
12 every step of the way. So we're ready to move on, and
13 I'm going to turn over the discussion to Mr. Hanh
14 Phan.
15 MR. PHAN: Thank you, Jim. My name is
16 Hanh Phan, I am the lead reviewer of PRA for APR1400,
17 PRA and severe accident evaluations. First, on behalf
18 of the technical reviewers and our PRA and severe and
19 severe accident, Michelle Hayes, allows me to obtain
20 our appreciation for your comments in Phase 3.
21 Today we are back, delighted to be shared
22 to you our evaluation report without open items. At
23 this point I would like to have one clarification,
24 that on Page 1 of the SER we clearly say that the
25 report reflects the information in Revision 0,

Revision 1, and the Revision 2.

So only six confirmatory items on Page 1 to 5 that have not been incorporated into the DCD yet. So just for a clarification. Next page, please.

The public in today's presentation are selective, and we can see there are important technical areas in Phase 4 and in support our conclusions. In this presentation we will give you a breakdown on the Staff's review and we will look into the modeling of digital I&C, RCP seal LOCA, PRA software conversions, the PRA update in the internal events at-power PRA.

And then we get into the Level 2 PRA including the severe accident induced steam generator tube rupture. Next, the low power and shutdown internal events level one and level two including the operator's correlations.

Next, the summit at-powers and and during low power and shutdowns. We also give you a status update on the internal events, internal fire PRA and internal flooding PRA, and then finally the Staff findings and conclusion. Next page.

By the end of Phase 2, the Staff identified 13 open items on PRA. The key activities in Phase 4 including reviewing updated PRA, resolving

and closing the open items, validating RAI's responses, assessing impacts on PRA from the design changes, verifies revised DCD information against the DCD and DCD other chapter and the information in the electronic submittal.

And we also reevaluate the PRA in all the chapters of the DCD including the physical security in Chapters 13.6, the ITAAC, the RAP, the human factor engineering and the environmental report.

Per SRP guidance, the Staff focused on the qualities, completeness, and consistencies of the information in the DCD to ensure that the results, the conclusion, the insights obtained of the PRA are valid. Next, please.

Of those, the DCD Chapter 19 does not provide the entire PRA information. However, the PRA audit that we initiated in Phase 1 and stays open at this point allows the Staff to access to the PRA notebooks and other PRA information.

During the audit, the Staff said was supporting documents relevant to the level one and level two internal events PRA, internal fire PRA, internal flooding PRA, low power and shutdown PRA, PRA-based seismic margin assessment, and external events assessment.

Throughout this review, the Staff issued 21 RAIs with 103 questions, and we had over 200 audit questions. At this point, all RAIs and audit questions are resolved. Only six left as confirmatory items.

I will stop here if you have any questions on the high levels of our information before we get to the next.

MR. SCHULTZ: Just one quick question. As you've described the sampling of documents and so forth, obviously your review has gone on for several years.

MR. PHAN: Yes.

MR. SCHULTZ: And so can you describe the timeframe of the sampling of the documents in all of these areas? Has it been continuous, have you been performing frequent audits related to the calculations as things have developed over the years?

MR. PHAN: Yes.

MR. SCHULTZ: How would you characterize that aspect of the review?

MR. PHAN: In March 2015 the Staff sent the application. We had a face-to-face meeting a month after that, and then the applicant posted the PRA information in the electronic reading room. And

the Staff accessed the information in there including the PRA notebook.

Since then, they kept the ERR open and continuously hosting update information. The Staff not reading every single page, but we frequently review the new information in the ERR.

MR. SCHULTZ: You had the current information available to you for review at any time you had question?

MR. PHAN: Yes.

MR. SCHULTZ: Excellent. Thank you.

MR. PHAN: Yes. Yes.

MEMBER SUNSERI: Does your synopsis of the 21 RAIs, 103 questions include the last two RAIs that were on the Applicant's last slide of their presentation?

MR. PHAN: Yes.

MEMBER SUNSERI: Okay, thank you.

MR. PHAN: That was all the questions. So with that, I would like to move to the internal events PRA and to be presented by Courtney St. Peters.

MS. ST. PETERS: Hello. I'm Courtney St. Peters. I helped review the level one internal events PRA. Just wanted to apologize, I'm feeling a little under the weather, so I'm coughing quite a bit.

So I'll try not to cough during my presentation. I only have two slides, so hopefully I can get through them, but I did just want to let you guys know.

My first slide is on digital IMC, it's a follow-up from our last presentation. I wanted to close the loop on this. This RAI has been closed. Since we met last and presented to you guys, the Applicant had addressed all of the questions we asked in the RAI.

We held multiple meetings and discussions between the probabilistic staff, our Chapter 7 folks, the deterministic people, and also the applicant to work through and make sure that there were consistencies through both chapters and verified that the probabilities were acceptable at this stage.

At the end of it, the operating system software common cause failure probability was evaluated to be $1.2E^{-6}$. The application software common cause failure probability was $1.2E^{-5}$. And then the applicant also evaluated 12 different sensitivity cases to better understand the impact of software reliability values.

And then as a follow-up from earlier I know there was a question related to the operator.

There was an operator action sensitivity case as well. That's addressed more in the RAI response for Question 1915.

MEMBER KIRCHNER: The genesis of the $1.2E^{-6}$ and $1.2E^{-5}$ is what? That was a someone mentioned at Westinghouse reference earlier in the Applicant's presentation? I'm just curious where those numbers came from and why they look suspicious to me. Why not just $1x10^{-6}$?

Where did the accuracy come from and why the convenient order of magnitude to the same order of accuracy, at least in terms of significant figures. I'm just, this is not my field but it just, it sticks out at me.

MR. PHAN: The numbers here from the applicant --

PARTICIPANT: Mic.

MR. PHAN: Oh. The numbers shared here, they are from the Applicant's estimation. We caught that. Even those they document that in their technicals report. The Staff, at this point we don't have any technical basis to justify these numbers. At this point, these are just a placeholder. We use the numbers to understand the importance of the software common cause failures and the hardware of common cause

failures.

MEMBER MARCH-LEUBA: But is your expectation that a COL applicant would have to revisit those numbers?

MR. PHAN: Yes, sir.

MEMBER MARCH-LEUBA: And is that clear on the CR? Is that made clear on the CR?

MR. PHAN: Yes.

MS. ST. PETERS: There's a COL item in Chapter 7 related to that.

MEMBER MARCH-LEUBA: And it goes, that 10^{-6} number is grossly inconsistent with my experience with the software I write which obviously should not be allowed within ten miles of a power plant. But it's still very low.

And it should be design-specific, especially the operating system should have built-in and the watchdogs and the process that they use to ensure that the operating system continues to operate.

So the danger once we approve a number like this people will use this number and say that's the one NRC blessed. So we have to make sure that we're not blessing it, we're just accepting it or something.

MS. ST. PETERS: In the RAI response,

there is a COL item that's been added. It says the COL applicant is to provide justifiable software reliability data for software used in the digital INC systems. There's that.

MEMBER MARCH-LEUBA: Yes. It's on the record that it's okay for based on the sensitivity studies that it's not really a crucial number. It's okay to use this, but it should not set a precedent because especially the operating system is so much system-dependent, variation-dependent.

MR. PHAN: The Staff understand that. To come up with the numbers, additional studies to be needed and we don't have that information like the development or the testing of the software. At this point, we cannot justify. We just use that to understand the modelings. Thank you for your feedback and comments.

MS. ST. PETERS: Next slide, please. And then my last slide is on the RCP seal LOCA. So after the last meeting, the RCP seal LOCA testing results was submitted on the docket. In that report there was a statement about the associated post-accident operator actions were identified but not further evaluated because they are plant specific.

The testing report, the testing results

supported a conditional seal failure probability that's listed in the DCD Revision 2 Table 19.1-16. The applicant also performed multiple sensitivity cases to determine the impact on CDF related to the conditional seal failure probability.

Any questions on that? I'm so sorry.

MEMBER BLEY: It's okay, you made it through your speech.

MS. ST. PETERS: I know. I just, I couldn't miss the opportunity to present. So you know, I had to be here for this.

CHAIR BALLINGER: The paramedics are outside.

(Laughter)

MS. ST. PETERS: Those are my slides, so I'm going to turn this over to Ayo.

MEMBER BLEY: Well, before you go ahead, I was looking through all the slides and I didn't see another place to raise this issue, so I'm going to raise it here, and I have a few questions. It's not related to your presentation at all.

On Pages 1937 and 1938 of the SER under significant sequences and cut sets you identify two places where you thought, well you told the applicant that their model was wrong, and one I kind of agree

with you which was the case where the recirc issue.

And when they went back and looked, they didn't really need that, so they pushed this off to COL to fix up the PRA. But you implied in the review that a single failure event is just unacceptable no matter what the frequency.

And that's kind of silly because we all know there are single failure events, so I didn't quite get why you got so spun up about the possibility of a single failure event as long as the frequency is very low.

The second one of those, however, has to do with a case where you said it was mutually exclusive events, that they would have all trains of a system out of service for maintenance. And you said well, it's not expected to occur.

I agree with you, it's not expected to occur however we have at least two that pop to my memory and there may be more cases where this is really happened in operating plants. So I'm kind of concerned that we tell people they don't have to consider things that have already happened when we're doing lots of analysis on things that have never happened.

What's the justification for that? And if

you don't know where these have happened, I mentioned them earlier, if you haven't read your own report on the Three Mile Island accident, go read it. There's a lot of good information there.

It happened there and it also happened at Salem at one point in time during the grass intrusion event where contrary to policy, practice, and everything else, all trains of a system were either tagged out of service or were put in disabled condition.

So why are these mutually exclusive and why are they not credible. And what do you mean by not credible? And I don't know which of you wants to address that.

MR. AYEGBUSI: Well, I think I'm better off addressing it because I wrote that.

MEMBER BLEY: You wrote that. Okay, well let's start with not credible. What the heck does that mean?

MR. AYEGBUSI: All right, so my name is Ayo Ayegbusi. I'm a reliability and risk analyst, and I did internal events along with Courtney. So I don't remember exactly what words I used in there. The idea was to pass, well so --

MEMBER BLEY: The words were they're

mutually exclusive, they can't happen, they're not expected to happen and that's true because of administrative policies, and they're not credible, which really bothers me because they've happened at least twice and probably more in the history of our operating plants.

MR. AYEGBUSI: Okay, so as far as credible, I think that's a word we probably should look at revising.

MEMBER BLEY: Almost anywhere you use that it's a good idea to look at it and consider it.

MR. AYEGBUSI: Right. But the caveat there is using credible in the sense of how the PRA should be -- in the sense of how the design of the plant should be modeled, right, and the PRA.

MEMBER BLEY: At least to me the design ought to be modeled consistent with the operating history we know.

MR. AYEGBUSI: Well, all our policy and all our guidance, right, says that the applicant should model the design of the plant as built and as it will be operated. So that's where our focus is on.

Like I said, you know, I think the word, you know, the fact that you read it and you've taken it a different way, I think we need to re-look at that

work. But the intent there was, you know, the PRA standard and all our guidance, like I said, was focused on as designed in this case as will be operated.

The intent there is to say that, you know, it's not to say that it's not possible. Obviously you can defeat tech specs or audit requirements. The intent there is to say that, you know, they didn't identify this, right, and they didn't address this.

So if they had in their write-up come out and said we're doing this because this has occurred before, right, we've acknowledged it. We've identified this, we've acknowledged it, you know, that this is in our results and it's occurred before and we want to retain it there, that would have been a different conversation.

And then that way it would be put within, it would have been set up within that context, right? It would have been put in context.

MEMBER BLEY: The way I read your words is you went back to them and said your model's wrong because this cannot happen. And that bothers me. You know, if they had argued that they had covered this sort of thing in their analysis of common cause failures, I might buy that if they really included,

made sure those kind of things showed up in the database for common cause failures.

But you direct them to get rid of it because it can't happen, and we know it can, it has. I find that kind of odd.

MR. AYEGBUSI: So I never went back to them and said it was wrong. I never went back to them and directed them to correct it. We asked a question as far as they said they've met the requirements of the PRA standard, one of which is, and I was going to cover this two slides on, one of which is supporting requirements under quantification B7 and B8 that has you address mutually exclusive events.

MEMBER BLEY: And mutually exclusive I understand. I don't understand why these are mutually exclusive. Mutually exclusive means these two things can't happen together. If you put this one in this position, you can't put the other one in that position for some quizzical reason.

I just don't get it, and I'll probably say something about it somewhere. I don't think this changes their results. In fact, their results don't include it which brings me to the last piece of this string.

Somewhere along the line between the

earliest design certs I was involved in which started with the SPWR and some others, we had lots of things refer to the COL. Here we're referring things to the COL.

But I remember one or two of these ago, and it could have been this one earlier, the Staff told us their internal approach has changed and the designers shouldn't be referring things to the COL to do that, you know, maybe that's your job as a staff.

And I'm curious about why you're happy that they've referred these and other things to the COL, and did I misunderstand where I thought the Staff was pushing in this area. And that's more of a management question, I guess. Anybody want to attack that?

MR. PHAN: May I go back to your comment on the task maintenance that continues. When we have the determined not acceptable, it goes right on the plans operatings procedures. Operators should not put more than one train out of service.

Of course they can do that, but at least you know certification. But for normal operation, they should not have two. Of course, if one fails they can recover. But for the normal operations, they should not schedule more than one train.

From that perspective, the scenarios are not a normal operation of the plant. We understand that there is a possibility of the operators unintentionally tag or put out of the trains pumps or systems, and that not reflect in the PRA.

Those scenarios, too many combinations and I don't think the applicants addressed that in their PRA.

MEMBER BLEY: I'll stand with what I said before. Now can somebody address the putting things off to the COL issue? Did I misunderstand where the Staff was trying to have design certs go?

MS. HAYES: No. This is Michelle Hayes, I'm the acting branch chief. The COLA items, we've been given guidance from OGC that you can't push something off to the COLA item that's necessary to make a finding on the adequacy of the design cert.

In this case, the COLA item, we've demonstrated that it didn't impact the results that much and so we thought it was acceptable.

MEMBER BLEY: So it's okay. Okay, I was just asking because I didn't know how you were being driven in that area. Fair enough, thanks.

MEMBER REMPE: Pull that string because I want to bring it up later but more. Shouldn't there

be guidance for the Staff that's reviewing particular items so it's clear to the COL applicant what must be done to satisfy what's been pushed off, or is it just okay to say oh, we'll let them decide that later.

It seems like then you're going to open it up where the regulator in one region of the country might say it's acceptable versus what a regulator in another region of the country might deem acceptable.

MS. HAYES: So items that are put off to the COLA applicant we review as part of the COLA and they would include how they address those items in the COLA and we would review that in the COLA as they are.

MEMBER REMPE: Okay. So again, it will be an open -- in the case of external reactor vessel cooling is where I'm confused. There's not much guidance on what's needed to be acceptable.

And I was going to wait until 19.2, but since we're talking about pushing things off, I just, I'm wondering how something that's a big item like that and right now there's the United Arab Emirates, they didn't use it. Shin-Kori did.

And yes, the plant can be safe, but what is an applicant going to have to do to satisfy the regulator on something that's kind of a big item that way? And shouldn't there be some guidance on

something like that or is there some sort of policy on what's needed to satisfy something that way?

MS. HAYES: I would have to look at that COLA item specifically, but if the applicant chose to use it as part of their design, they would have to submit their evaluation as part of their COLA and then we would review that when we reviewed the COLA and we would evaluate it to the same guidance we would have used when we were doing the DC had they included it in the DC.

But in this case it's not required to make a finding because they're not taking any credit for it.

MEMBER REMPE: Right, they're not taking credit for it but then you don't want adverse effects. I mean, what's required to do it?

MS. HAYES: So the COLA applicant will have to do that and they'll have to submit it to us and we'll review it. I'm missing something.

MR. WAGAGE: This is Hanry Wagage from the staff. I believe that CBX analyses and specifically that item, that COL item, if the Committee would like to wait until I make my presentation then I can help with that item. Thank you.

MEMBER REMPE: Thank you.

MR. AYEGBUSI: Okay, so I'm going to cover two slides. Each slide covers one big item. The first one is model conversion. The Applicant earlier presented the fact that they converted the model from SAREX to CAFTA.

So we evaluated that conversion as documented in the conversion report. We found that the conversion was acceptable mainly because the results from the previous model to the new model were pretty similar. And we also looked at cut sets sequences and risk significant SSCs and we also saw that there was similarity there.

And in the areas where they didn't get the same results, that would explain the differences there whether it was because, you know, they went out and fixed some issues that they already had in the model, and we went through extensive conversation about that.

So in conclusion, you know, we found the model conversion was properly done and, you know, the results and insights were very similar and we didn't have any reason to ask further questions as long as it's actually adequately captured in a DCD.

So, are there any questions on model conversion?

MEMBER KIRCHNER: Well, just because the

answers are similar doesn't necessarily, it could be fortuitous. What is the major difference between the SAREX, not being a practitioner in this area, SAREX versus the CAFTA models?

And besides that the results were very close, from a fundamental standpoint what are the major differences in the software models, and didn't you examine specifically what impact that might have on the results rather than just saying the gross answer is 10^{-6} so it's okay, or whatever the answer was.

MR. AYEGBUSI: Okay, so this model conversion, part of why this came up was because the applicant had used SAREX, right, and we were trying to look into the V&V, the verification and validation of the software.

And I'm not saying that that's why they converted, I'm just saying that when we wanted to do that in that period of time, the applicant said they were going to switch to CAFTA which CAFTA is a platform that has been, you know, has that established based on how it was developed.

And that's, we're familiar with CAFTA, right? So speaking to specific to SAREX, you know, I couldn't get into the details of that because the

conversion came pretty early in the application stage.

MEMBER KIRCHNER: I'm just reacting to the way, maybe it's just the wording of the slides but if I were doing an analysis with two different physics models or two different thermal hydraulic models and I got similar results, that wouldn't be, that's good but that's not enough.

I'm presuming you went through and looked at validation and verification, or verification and validation for the new software vis a vie the old software or something like that, right? Not just that the answers are close because we're down in 10^{-6} space with significant figures of three decimal points and such.

It's just the conversion is acceptable because it was very close to what was previously estimated, that just wouldn't pass muster in general going from one software package to another. Do you see my point?

MR. PHAN: Yes, I do. CAFTA, in PRA space CAFTA is one of the most popular software being used for many operating plants to PRA. SAREX, that is software actually develop by the Applicant. In Phase 1 we asked the Applicant on this particular SAREX software and the Applicant told us that they're going

to change from SAREX to CAFTA.

The reason for doing that, unclear to the Staff. We don't have the justification that up to the applicants the software that they want to use to do their PRA. And here we're just talking about the software, we're not talking about the models of the PRA.

Yes, thank you. The statement of acceptable based on the numbers is not clear. But what we tried to say there is not just the numbers, we looked at the outcome including the quantification, the cut sets, and the components or events included in the cut set.

We had a face-to-face discussion with the Applicant and we asked for any difference between those two outcomes. And the Applicant clearly explained to us why there are differences and justifies any number changes.

And that's why we say a newer quantification be acceptable, but not just based on a single number or any others beside the outcome from the quantification.

MEMBER KIRCHNER: Okay, thank you.

MR. AYEGBUSI: All right, so if I can add, I mean, I think to your point, I think what we did

that we didn't put on a slide was ensuring that the event tree stayed the same, or sampling the event trees, right, that there were no changes there, fault trees, there are no significant changes there.

They're basically taking the inputs from one package and putting them in another package. But again, to that detail, we didn't have that information. So we could have probably captured that on this slide.

MEMBER KIRCHNER: Thank you.

MR. AYEGBUSI: Are there any other questions on the model conversion? Okay, so getting to the hot topic, once again, over the course of the application and application review, the Applicant decided to update the PRA and update the results and risk insights and the DCD when necessary.

So we had to take a look at this to the updated PRA and the results to ensure that given all the changes that have been made, whether it's design changes or corrections that they, then the current PRA reflects the APR1400 design.

And that's, I think the key word there is it reflects the current design. The key word is reflect. During that audit period, we identified two I would say high level issues while reviewing cut sets

from the top 50 percent of cut sets that make up the internal events CDF.

The one issue which was kind of raised earlier has to do with we identified two cut sets where in response to an initiating event you would have only one component, that if it failed it would result in core damage.

So when we saw that, and I should probably just talk about the specifics that happen in this case. If you had a large break LOCA in the hot leg, right, and a failing of one of the station junction pumps, it would get you to core damage.

And so when we saw this, we saw this as, you know, an indication, and I specify the word indication, an indication of a potential single failure vulnerability.

So after identifying this issue we had subsequent conversations with, internally with our reactor systems folks and, you know, the feedback that we got was that this particular case was not a case that is typically analyzed for because based on how the vessel is set up and how the plant is designed.

And then when we had conversations with Applicant, they were able to show that additional analysis that they performed that safety injection

pump was not required during the hot leg break. Or, I take that back. Hot leg injection which is done by safety injection pump is not required if you had a large LOCA in one of the hot legs. So --

MEMBER MARCH-LEUBA: So basically that cut set became a zero element cut set? You have a single failure?

MR. AYEGBUSI: Yes, it had a single failure. So basically --

MEMBER MARCH-LEUBA: Because it's not near so it is not core damage.

MR. AYEGBUSI: Correct, correct.

MEMBER MARCH-LEUBA: All right.

MR. AYEGBUSI: Correct, yes. So that was the one issue.

MEMBER MARCH-LEUBA: So how was it resolved? I mean, you let cut set 55 and 73 are still in the model?

MR. AYEGBUSI: Are they still in the model?

MEMBER MARCH-LEUBA: Yes.

MR. AYEGBUSI: I can't make statements about the model because I don't get to see the model. I only get to see the results, so they are in the DCD. There are mark-ups.

When this issue was discussed with Applicant, the Applicant agreed to make clarifications in the DCD, as Michelle spoke to earlier, to clarify that, you know, this is not something that's expected. I forget the specific words, but this is not something that's expected for this design.

MEMBER BLEY: In your SER you said that they put it off as a COL action item to re-look at this.

MR. AYEGBUSI: Correct. Yes, so the two items there is, you know, making that clarification and putting in an identifying a COL action item for the COL to address this.

MEMBER MARCH-LEUBA: Yes, because we're talking about the proverbial meteorite the size of a city. It would destroy the whole plant, but it has a very local -- but this one would have a significant probability of failure because they made the 100.

So a single failure point with some contribution to the CDF you can put the cutoff at one percent or ten percent, then it becomes relevant. And the COL will confirm this statement, that the injection is not required?

MR. AYEGBUSI: That's correct.

MEMBER MARCH-LEUBA: And what calculation

will they do that they haven't done already?

MR. AYEGBUSI: So --

MEMBER REMPE: They find out it is reqd.

MEMBER KIRCHNER: Then why wouldn't you just go look at Chapter 15 and LOCA analyses and determine that this is, acceptable isn't the right word, but that this particular event is probably not going to lead to core damage?

MEMBER MARCH-LEUBA: Right, it's an acceptable conservatism in the analysis.

MR. AYEGBUSI: Well, so the word conservatism, we didn't want to use that word at all in this case because this is, when we looked at Chapter 15 and talked to our Chapter 15 folks, you know, they clearly said to us this is not something that is analyzed, it's not needed to be analyzed. And frankly --

MEMBER KIRCHNER: Well wait a minute, wait a minute, wait a minute. They may not have analyzed it in detail, but they have to go through a spectrum of the breaks and locations to find out what is limiting for Chapter 15.

So by default, what they've done is demonstrated to your colleagues in Chapter 15 that this hot leg injection is not the limiting large break

LOCA. Right?

MR. AYEGBUSI: Because in the modeling, they made an assumption. They made an assumption in the modeling of the plant. So you're right, in Chapter 15 that analysis, it's not part of that analysis for Chapter 15.

But in the modeling of the plant in Chapter 19, they made an assumption that a hot leg injection will be required if you had a large break LOCA in a hot leg.

So it's really a question of a Chapter 19 assumption for the PRA model. And so to answer your question about there's nothing that needs to be done in addition to show that, or to be able to address this issue, right, it's a matter of having to revise the DCD and going through that lengthy process of doing that, and as Michelle spoke to earlier.

So that's why we found it acceptable to just --

MEMBER MARCH-LEUBA: My concern, I mean, it's more philosophy than anything else. If it can be resolved now, why dump it to the COL? I don't think they're going to do any calculation other than -- when the COL comes back and say well, we don't have calculations on it, NRC told us it was okay. So why

even dump it in there?

MR. AYEGBUSI: I agree with you. It's a matter of time. Like I said, you know, the Applicant would have to go back and take it out of the model and run the models and run the results and --

MEMBER MARCH-LEUBA: It's okay to leave it in the PRA model if you know it's a conservatism.

MR. AYEGBUSI: But so as a former inspector, right, when you refer to the DCD or the FSAR, whatever you have in there, you know, there's an indication that that's acceptable.

And we don't want that to misconstrue, we don't want anyone to misconstrue that as seeing that and saying it's acceptable which goes back to, which goes to the next issue of the test of maintenance is we don't want anyone to come up and say well the PRA says it's acceptable, hence we can put it in this condition.

MEMBER MARCH-LEUBA: See, but right here I'm thinking whenever somebody wants to build this plant, they'll have to go with a COL. They'll have to go through these two issues. And what they will do is we'll re-run analysis and we know it's okay. But they're going to spend 100-some days trying to do the paperwork to get it down. We're complicating

everybody's life.

MR. AYEGBUSI: I 100 percent agree with you. It's the constraints that we have, I guess.

MEMBER MARCH-LEUBA: It's often the paperwork, not the actual work.

MR. AYEGBUSI: Is there any other questions on that one issue? All right, so the second issue, we identified three cut sets where they had the same components on different trains, similar components on different trains in testing and maintenance in the same cut set.

And as we discussed earlier, you know, there are a number of issues there. One is our expectations of PRA is the model is built based on as designed and as will be operated. And two, the Applicant has stated that they meet the requirements in the PRA standard.

However, this is one case where they did not meet that, the requirement when you apply the layer of as designed and as will be operated on top of that. So similar to the previous issue, clarification has been put into the DCD about this, about this issue and as a COL action item for the COL applicant to when they perform, develop their PRA model for decisive cert of PRA to look at these items and take action,

you know, whether it be deleting, which is what the PRA standard will call for, deleting it or explaining it if they want to keep it in there. And then we would have to review that again.

MEMBER MARCH-LEUBA: This one is, and I don't understand the last bullet, the details of the technical. But is this what Dennis was saying, that they really are not mutually exclusive events? Is this the one you were talking about?

MR. AYEGBUSI: This is the issue we were talking about, correct.

MEMBER MARCH-LEUBA: All right. So if this is an agreement in your conclusion, maybe we should say something. I'm pretty sure Dennis will.

MR. AYEGBUSI: Okay.

MEMBER MARCH-LEUBA: You have to come back and address it.

MR. AYEGBUSI: Okay.

MEMBER BLEY: I suspect if I get my way it will be something saying that you ought to consider this in the pre-fuel load PRA, but not come back and fix it now. That's where I'm leaning.

MR. AYEGBUSI: Okay.

MEMBER BLEY: But that's just me. I don't write the letter.

MEMBER SKILLMAN: I was curious whether or not an item that might communicate this level of angst around ACRS or in the staff might not have a COL item with an asterisk that says this better be resolved in tech specs. It would seem to me that that's where that would get taken care of.

It would seem to me your tech specs would say under these conditions you must have or you may not have. But the operators would know with precision what may be in service and out of service concurrently.

MEMBER BLEY: They already have that, Dick. It's my point is even when you have that, sometimes people screw up.

MEMBER SKILLMAN: I got that.

MEMBER MARCH-LEUBA: And I have an example.

MEMBER SKILLMAN: I mean, I came from the TMI culture so I kind of understand this as the years progressed after March 20th of 1979. We got, at least I think we got smarter and we changed our paperwork and we got to a place in TMI1 where I'm sure we could have tricked the system, but by and large there were checks and balances to make sure that we didn't find ourselves in the same situation that we unfortunately

found ourselves in on March 20th, 1079.

MEMBER MARCH-LEUBA: We're talking about the PRA analysis. Are they going to come back and say that this has a probability of one in a million years.

MEMBER SKILLMAN: All I'm saying is when a finding like this comes out of a design cert action, and that's what we're in, a design cert action, maybe this one gets an asterisk that says hey, this one deserves special attention either in your tech specs or whatever guidance document is in the control room because that's where the rubber really meets the road. Enough said. Thank you. Off my platform.

MR. PHAN: Dr. Bley and other members, we appreciate your comments and we clearly understand that the condition of the operator's errors hurt more than one train or systems out of service, that not in the PRA yet, at least any DC PRAs.

For operating PRA, I don't be aware any modeling of that yet, and I believe that it fit that the PRA should somehow reflect those scenarios, and that not modeling with any PRA as I know of up to this point. Thank you for your feedback.

And that the end of our presentation on the internal events PRA. Up next I would like Hanry Wagage to present to you the Level 2 internal events

PRA.

MR. WAGAGE: My name is Hanry Wagage. The Committee had a concern on consequential steam generator property probabilities used by the Applicant. The concern was that Applicant used NUREG-1570 probabilities. NUREG-1570 was for Westinghouse PWR, Westinghouse steam generators.

And later, they stop, they put NUREG-2195 had shown that combustion engineering, steam generators similar to what Applicant used, have more failure probabilities than Westinghouse steam generator tube ruptures.

So the Applicant, this is how, why we accept those accepted values. The Applicant, although it used NUREG-1570 steam generator tube rupture probability, it had the benefit of a document by EPRI which had the same conclusion that steam generator ruptures will have higher probability than Westinghouse, that based on that in addition to using NUREG-1570 values, the Applicant increased those values using the experience from that EPRI report.

NUREG-2195 estimate probability was .22 and .31. .22 is for aux feedwater operating and .31 is when the aux feedwater was not operating.

Still these values that what Applicant

used is similar to these values. Still it's lower. That's a proprietary value. We can talk about that during a closed session if the Committee would like to go to a closed session.

In addition to that, the Staff had the benefit of SOARCA study which showed that steam generator properties would not result in large releases. The two reasons for this conclusion was in SOARCA study was that aerosol deposition in steam generators and subsequent severe accident induced hot leg failure which releases isotope in the containment and the depositor.

Based on the fact that Applicant increased the values in 1570 and SOARCA study showed that steam generator tube ruptures would not cause large releases. Staff accepted the Applicant probabilities.

MEMBER KIRCHNER: So how do you explore this in PRA space? I mean, what you're citing here are detailed deterministic models. And this material's been presented to the Committee. There's a large amount of uncertainty in those kinds of calculations as to whether the hot leg goes and ruptures before the steam generator tubes.

And yes, there's differences between the CE and the Westinghouse types. Do you propagate some

uncertainty and look at either/or, or are you just going to your last bullet, which seems to suggest that it only goes and releases into the containment?

MR. WAGAGE: Okay, Applicant did not assume that hot leg tube rupture occurs, Applicant did not go to the next one. But if your question is related to how the SOARCA study came up because numbers we have were --

(Simultaneous speaking.)

MEMBER KIRCHNER: No, no. I've had that presentation. We've had that presentation. I'm not asking you to redo that. I'm saying that there's a large amount of uncertainty and such a complicated complication beginning with the assumption of what's going on in the core and then how hot the gasses coming out are, steam and gas, whatever.

So I would expect that in PRA space you would propagate some uncertainty and allow for the fact, well it looks like from those detailed analyses the hot leg will rupture first, but maybe not. So can't you propagate in PWR space either or and wait one more, wait perhaps the hot leg rupture being more probable that the steam generator tube rupture.

But having sat through all those detailed analyses, I would still say there's a lot of

uncertainty in those scenarios and which materials are going to fail first. So I would expect in PRA space you would give some conditional probability to one and then a different to the other and then propagate that through your analyses and see what you come up with.

MR. SCHAPEROW: This is Jason Schaperow from the staff. I would like to address that. What we've, back in the NUREG-1570 days, the question was what would fail first, the steam generator tube or the hot leg.

But as we progressed and did the SOARCA analysis with an integrated modeling, we found that there really wasn't an either/or question. We found out we always got a hot leg rupture because rupturing a single tube was not enough to depressurize the RCS.

The hot leg, if it was postulated that the hot tube ruptured first, we found the hot leg would rupture within a few minutes. And we actually did sensitivity studies with both MELCOR and SCDAP/RELAP, and we also varied the number of tubes that failed for the rupture.

MEMBER REMPE: So let me ask it a different way. Looking at John's notes a long time ago, they did a decomposition tree where they assumed different situations. If the loop seal occurred they

assumed the thermal induced failure.

But they had other probabilities that were, like, .498 and a pressure induced steam generator tube failure of .037. Does the Staff go through and look at those numbers and say yes, I think that's right, and how do you evaluate the values selected by the Applicant in this case? And I think that's kind of the way to frame the question of what you're trying to ask.

MEMBER KIRCHNER: That's kind of where I was going, yes.

MEMBER REMPE: And I don't know if that's a question for you or you.

MR. SCHAPEROW: Yes, I can certainly address the severe accident and any other pressure induced and the spontaneous, I guess the more spontaneous tube rupture. They have numbers in their PRA for those too, I imagine.

MEMBER REMPE: Yes, so did you go through and look at the numbers? I mean, you have the analyses from the NUREGs and all that, but did you look and see what the Applicant selected in their PRA and look at those numbers and say yes, that seems reasonable and how do you make that judgement?

MR. WAGAGE: Some numbers I looked at and

found reasonable that this particular one that the steam generator rupture or because they did not go to that hot leg creep rupture.

MEMBER MARCH-LEUBA: Just to put it in perspective, this severe accident scenario, how does it continue to the final LRF? Is it one percent, ten percent, fifty percent? I mean, if it's .1 percent of the final contribution then you can have a large error and who cares. If it's 50 percent, then oh.

MEMBER REMPE: Aren't bypass failures pretty important for this plant?

MEMBER MARCH-LEUBA: I think so.

MEMBER REMPE: I missed this morning's session but I thought there was some backup slides that they had showing it was important.

MEMBER MARCH-LEUBA: I'm asking do you have, is it ten percent of the final contribution?

MR. WAGAGE: I don't remember the number right now.

MEMBER MARCH-LEUBA: But this is important? You don't remember? They are trying to find out.

MR. PHAN: We don't have the numbers.

MEMBER MARCH-LEUBA: Okay, that's fine.

MR. PHAN: But can we ask the Applicant to

come up with the importance ranking?

MR. SISK: This is Bob Sisk. Is it possible to bring up our slide, the backup slide that is being referred to? I think we have the people here to answer the question.

MEMBER MARCH-LEUBA: Do you know what number it is? We need to get --

MR. LEARY: If I remember right, it's slide number 92 in the back up.

MEMBER MARCH-LEUBA: Go first to Page 92.

(Off-microphone comments.)

MR. LEARY: May I speak to this?

MEMBER MARCH-LEUBA: Yes.

MR. LEARY: Okay. This is Jeff Leary. So the NUREG-2195, when you're talking about consequential tube rupture, there are really two things that are of interest in terms of what's important or not important in terms of APR1400 LRF.

The first kind of consequential tube rupture is the kind that you've been talking about in the last ten minutes or so which is during the severe accident progression, so it could be, you know, a loss of feedwater initiating event and injection fails. You've got a core damage at high pressure.

During that progression the tubes heat up

and consequential to rupture occurs. The second item that was calculated in NUREG-2195 was a specifically for ATWS and steam line break initiating events and a consequential tube rupture that happens to it because of the high differential pressure across the tubes.

In the first of those two, the severe accident progression, it knows the -- we did a sensitivity calculation for APR-1400 and increased all of the numbers in the APR-1400 Level 2 by 20 percent except for the ones that were already, you know, I didn't put it over 1.0. If it was already 1.0 we didn't increase it.

But all the other numbers increased by 20 percent, and the total LRF increased by only two percent. So it shows not a real high contribution.

The second, if you go to the next slide, please. The second part of that is the ATWS and steam line break induced tube ruptures. These are what are important in the APR-1400 Level 2. And in these, it turns out that the NUREG-2195 numbers are actually lower than what was used in the APR-1400 Level 2.

So as you can see up there, the APR-1400 Level 2, the consequential probability of an induced tube rupture for both ATWS and steam line break was 2.7 E-2. In the NUREG-2195, those are separated into

two different probabilities. For the ATWS it was 80-3 and for the steam line break it was 40-3.

So those are less than what was in the APR-1400. Doing a sensitivity in which the APR-1400 changed those numbers from 2.70-2 to 80-3, so it's using the ATWS number there for both the ATWS and the steam line break.

That decreased the LRF by 31 percent. So it shows that the real significance in the APR-1400 Level 2 is in this part, in the ATWS and steam line induced tube ruptures and compared to the NUREG, the new numbers in the NUREG, it's actually lower. So if those numbers were used in the APR-1400, the overall LRF would be expected to go down. If that addresses the question.

MEMBER MARCH-LEUBA: But this, as the first one it says it is significant. It's worth spending some time reviewing it. And you did.

MR. LEARY: This part is significant, but the ATWS and steam line induced tube ruptures, or the other part which is the severe accident induced tube ruptures, there's not a high significance for the APR-1400.

MEMBER MARCH-LEUBA: It's not?

MR. LEARY: No.

MEMBER MARCH-LEUBA: Okay.

MEMBER REMPE: Is that because you assumed 100 percent basically thought, well if a seal failure occurs you just went ahead and assumed that you would have a thermal induced failure?

MR. LEARY: No, ma'am. No, no. The only place where there was an assumed 100 percent failure was for the severe accident cases in which the loop seal cleared.

MEMBER REMPE: Right.

MR. LEARY: And it's a relatively small, that and in addition the APR-1400 has the design capability for depressurizing the reactor to get one of the SAM-G actions as you get towards the progression of core damage.

That's a relatively small frequency that ends up in that category with the loop seal cleared. When the loop seal doesn't clear, that's when the conditional probabilities are in the some of the numbers you quoted, some of the 1.0 numbers you quoted.

MEMBER REMPE: Right, okay.

MEMBER MARCH-LEUBA: And to the risk of wasting some time with physics, for this calculation did they not incorporate the lessons learned from the

Staff calculation that the hot leg is going to break anyway and there's not going to be as much leakage through the steam generator, but most of the iodine and all that bad stuff is going to stay in containment through the hot leg?

MR. LEARY: The APR-1400 analysis did not --

MEMBER MARCH-LEUBA: All the inventory goes through the steam generator?

MR. LEARY: There's a probability in there for hot leg failing first. But the --

MEMBER MARCH-LEUBA: What I've heard, and it makes sense to me, that the hot leg fails also. They both fail at the same time and therefore there is a bigger path for radionuclides inside containment than outside.

MR. LEARY: Right. So in that case none of it goes to LRF.

MEMBER MARCH-LEUBA: Well, some of it goes.

MR. LEARY: Some of it goes out.

MEMBER MARCH-LEUBA: But this is a smaller amount.

MR. LEARY: But it's not a large release --

MEMBER REMPE: So to waste more time on physics, usually they do the calculation and if they get a temperature such that the hot leg would fail, they'll depressurize the reactor so there's no concerns anymore about the steam generator tube failing, and that's true with MELCOR or MAAP.

And you might have different numbers. And so what I was trying to get the Staff to say earlier is yes, well we looked at our MELCOR analyses or our prior analyses and we thought those numbers of .45 or whatever the number is are reasonable based on experience, and I didn't quite hear the Staff respond back that way.

But that's kind of, you wouldn't have both because you would depressurize, and so you reduce the load usually is the way this calculation --

MEMBER MARCH-LEUBA: What I heard is that both happen because there is not sufficient depressurization through a few tubes in the steam generator.

MEMBER REMPE: That's true, yes. But if the hot leg fails first.

MEMBER MARCH-LEUBA: If the hot leg fails first, then the steam generator will be fine. Okay. So it's important and you guys have addressed it

somehow.

MR. LEARY: Yes, sir.

MS. POHIDA: May I continue?

MEMBER REMPE: Yes.

MS. POHIDA: Okay. I'm Marie Pohida and I'm going to briefly summarize my --

MEMBER MARCH-LEUBA: John left me in charge of microphones. Your microphone needs to be on. John left me, he left me in charge.

MS. POHIDA: Okay, is that better? Okay. All right, so I'm going to briefly summarize the low power shut down PRA review that was done for Level 1 and Level 2. Just to clarify, the POS definitions, the plant, you know, operation states, they were evaluated for completeness from hot shutdown to RCS heat up.

The time to boiling and core uncovery was calculated for each plant operational state using RELAP by the Applicant. I evaluated the definition of pauses by the status of RCS penetrations --

MEMBER BLEY: For this analysis, Marie, how did they define uncovery? Top of core or two thirds down or what? I mean, there was a lot of work done 20 years ago that kind of said for these conditions at shutdown you have to get the level down

well below half before you really start having any damage.

MS. POHIDA: Okay, I may have to defer to the Applicant on this. As I recall, the time to core damage was estimated as peat cladding temperature around 1,300 degrees Fahrenheit but I like to defer to the Applicant to make sure that my memory's correct.

(Off-microphone comments.)

MS. POHIDA: I beg your pardon?

MR. OH: This is Andy. Your memory is correct.

MS. POHIDA: Oh, thank you.

MEMBER BLEY: Well, wait. That still didn't answer it for me. Instead of defining core uncovery you define the time until you reach a specific temperature, is that what you just said?

MS. POHIDA: No. The plant operational states were defined in a DCD table, Table 19.1-92. And so for each plant operational state from hot shutdown to RCS heat up, the time to boiling was calculated. This was for the Level 1.

MEMBER BLEY: That's easy.

MS. POHIDA: Yes, time to boiling.

MEMBER BLEY: The time to core uncovery.

MS. POHIDA: And time to core uncovery.

MEMBER BLEY: My question was does that mean a particular temperature or does that mean a particular level, top of core, mid core? What does core uncovery mean? What was calculated?

MS. POHIDA: If I could defer to the Applicant, I believe it was top of active fuel. Okay, thank you. Okay, so --

MEMBER BLEY: I didn't hear any answer.

MS. POHIDA: -- top of active fuel.

MEMBER BLEY: Okay.

MS. POHIDA: My memory is being challenged here.

MEMBER MARCH-LEUBA: What I would say, the record will not show head bobs. You have to say.

MS. POHIDA: Okay. So what we did is I evaluated the POS definitions for the status of all the RCS penetrations, if the pressurizer manway is open, closed, the vessel head is installed or it's removed. You know, RCS level, are they at the plans, are they at mid-loop, is there level in the pressurized inventory, and decay heat if it's before or after refueling.

Event trees were developed for each plant operational state except for the refueling cavity flooded condition, and that was evaluated

qualitatively.

For when the refueling cavity was flooded, we did look at the time to boiling and the time to core uncovery with and without installed reactor internals. And we also reviewed the potential for drain paths from the refueling cavity and the availability of level instrumentation. Next slide, please.

There were two open items that were evaluated or actually resolved in other chapters. The first one was a DCD discrepancy between the Level 2 PRA and the shutdown containment closure tech specs identified in LCO 3.6.7.

The tech specs initially required containment closure below, three feet below the reactor vessel flange which is defined as reduced inventory. That was incongruent with the Level 2 PRA that said any time the RCS is open, be it pressurized or manway, the containment will be closed.

So the LCO for 3.6.7 was updated to require containment closure mode five with the RCS loops not filled and mode six with water level less than 23 feet above the top of the reactor vessel flange.

The second issue that was resolved in

Chapter 5 had to do with the potential for air entrainment and vortexing of the shutdown cooling pump during mid loop conditions, and that impacts the RCS overdrain frequency.

So an ITAC was added to DCD Revision 1, Tier 1, Table 2.4 that states that shutdown cooling will not be impaired by gas entrainment during mid loop while operating its maximum allowable flow rate and operating at its lowest allowable hot leg level for decay heat removal.

May I go to the next slide, please? Oh, I'm sorry.

MEMBER BLEY: Yes, run that one by me again and explain it a little bit.

MS. POHIDA: Okay.

MEMBER BLEY: So the ITAC is they're going to fill this thing up to some point and run the pumps and see what happens?

MS. POHIDA: As I understand there's going to be a test or report, something documenting, there's going to be an item documenting that if you run this thing at the highest anticipated flow rate at the lowest operational hot leg level, that the pumps themselves will not be impaired by air.

MEMBER BLEY: All right. But all of the

events in which we've stuffed air into these things have happened when we were operating where we ought to be and all of a sudden the water started going away.

So this doesn't really help us for the real events that have led to vortexing in the past.

MS. POHIDA: I need to think. I'm thinking --

MEMBER BLEY: I don't remember a case where they got into vortexing problems when they were not losing water.

MS. POHIDA: Yes, yes. They're all due from losing water. All right, I have --

MEMBER BLEY: Which can still happen?

MS. POHIDA: Yes. And that was analyzed in the PRA. I'm thinking of events where people entered into mid loop operations at a higher decay heat load than anticipated.

MEMBER BLEY: Of course they're getting there pretty fast.

MS. POHIDA: They're getting there faster. And they had to increase the flow rate to keep up with decay heat. And they ran into air entrainment problems there.

In this design a note was added to tech specs stating that they're not going into mid loop

operation before 48 hours. Does that help you with your question?

MEMBER BLEY: Yes, yes.

MS. POHIDA: Okay. May I continue?

MEMBER KIRCHNER: So the 48 hours then implies some cooldown or something, right?

MS. POHIDA: Yes, 48 hours post shutdown. Yes.

MEMBER KIRCHNER: Does that really answer the question or the concern you raised, or is it pump performance of function of temperature and head? And the water, ambient water temperature and head.

MS. POHIDA: Well, the --

MEMBER KIRCHNER: See where I'm going with that? You know, you just think --

MS. POHIDA: Could you please restate that?

MEMBER KIRCHNER: You just feel that after 48 hours that this problem would not occur?

MEMBER BLEY: Let me try something on Walt's question, make sure I don't screw things up. There's kind of an implication because of this issue of decay heat. And it's clear, temperature's the key issue. Are they going to run this ITAC cold or are they going to have it heated up somehow? And if they

have it heated up, how far are they going to have to heat it up. Are they just going to do that with our HR bumps? It seems unlikely.

MS. POHIDA: May I get back to you on how --

MEMBER BLEY: Sure.

MS. POHIDA: -- that ITAC is stated? Okay. Is that --

MEMBER BLEY: Next time we get together is all right for me, but somebody might, if you get it sooner I think we would be good.

MS. POHIDA: Okay. May I continue with the next slide? Okay. Regarding the operator dependence between Level 1 and Level 2, you know, as common in most shutdown PRA areas, particularly for operating plants, if you look at the core damage cut sets and the LRF cut sets, there tends to be a slew of operator actions in the same sequence.

And so the concern was the dependency between operator actions between the Level 1 and the Level 2 analysis. I asked that in an RAI. And so in response to that, the Applicant performed a sensitivity study using a minimum joint HEP of 1E-5.

And that's what was suggested in NUREG-192, the good practices for implementation of HRA.

And as a result of this sensitivity study, there was just a small increase in LRF.

As was explained earlier this morning, the Applicant did revise their dependency analysis during the PRA update.

That concludes my discussion on shutdown Level 1 and Level 2 risk. Is there any additional questions? Thank you very much.

MR. PHAN: Thank you, Marie.

MS. POHIDA: Oh, you're welcome.

MR. PHAN: We are exactly at the half hour presentation. Please let us know how much time we have left for the PRA portion.

PARTICIPANT: Five minutes.

CHAIR BALLINGER: The question is how much time do you have?

MR. PHAN: Yes.

CHAIR BALLINGER: 6 o'clock.

MR. PHAN: Yes. For the PRA portion.

MEMBER REMPE: They've got five minutes left of the allocated time is the answer.

CHAIR BALLINGER: Okay, when the big hand is on -- no, about five minutes. Go ahead.

MR. PHAN: The next topic on the seismic, per SECY-93-087 and the related SRM, it's acceptable

for the Applicants to perform PRA based SMA seismic margin assessment to access the seismic event.

As you all are aware that currently there is no single document provides step-by-step how to complete one. However, the guidance to perform one are spread amongst the guidance we listed on this slide. Both of the Applicant and the Staff develop and reviews the APR-1400 PRA based SMAs using the guidance in these documents. Next page.

The objective of the PRA based SMA is to evaluate the plant HCLPF and identify significant seismic vulnerabilities and insights to demonstrate that the design's robustness.

So with that, the Staff focused on the framework for assessing the potential significant failures and the basis for developing the seismic equipment list, and then the seismic fragility evaluation that driving the sequence level and plant level HCLPF, and last, the seismic related risk insights went from the assessment.

In March 2015 when we accept the Applicant, the Staff found that the information in DCD Revision 1 not complete relevant to the seismic evaluations. Therefore, we issue many questions under seismic.

This year, the Applicant submitted the revised PRA based SMA at power and also during low power and shutdown. And those two questions, or II questions are confirmatory items and there's 19-85 and 19-10. Next, please.

In general, there are six major tasks to complete the PRA based SMA. One by one we will update you the status of each. Next, please.

Task one, identify seismic hazards input. The Staff presented to you a review of this pass in Phase 3. There are no changes from Phase 2 review.

Task two, develop seismic equipment list. The staff finds that the SEL is reasonable and comprehensive. The list includes the SSCs that are not models in the internal event PRA. For example, piping, cable trays, tanks, ventilation, and so on. The list also includes the buildings that house the PRA and passive components. Those are listed here.

Task three, evaluate the seismic fragility. There are no changes seen, we present to you in Phase 3. Next, please.

MEMBER SKILLMAN: On that slide under Task two in the third bullet, are the cranes included in that --

MR. PHAN: Yes, sir.

MEMBER SKILLMAN: -- list? Yes? Thank you.

MR. PHAN: Task four, assess systems and sequences, and accident sequences. There are five steps in this task. Step A, seismic induced initiated events. The staff finds the Applicants have followed the guidance and rationally conduct this step. The Applicant assessed multiple concurrent seismic induced initiators, for example LOOP and LOCA or LOOP and loss of component cooling waters.

In modeling, the Applicant includes plant transients, LOOP loss of 4kv AC bus, SBO, loss of instruments and control, main steam line break, loss of component cooling waters, and ATWS. And that those identifies in the guidance.

The Applicant considers all LOCAS, small, medium, large, and also analyze supportings failures such as failures of the engineer's safety features. Next, please.

Step B, evaluation of the safety functions. The Staff finds Step B acceptable since the Applicants evaluate all key safety functions. The Applicants identified to finalize and the support systems those required to meet the safety functions and took no credit for no safety and no seismic task

1 systems.

Step C, plant's response and mitigation systems. The Staff finds Step C logical and acceptable. The applicant develop plant respond logic based on the at powers internal events PRA.

The Applicants included random failures and human error failures in the internal events PRA, and also develop event tree and fault tree to generate core damage and large release sequences initiated by a seismic event including random failures and human errors. Next, please.

Structure failures. The Staff find this step acceptable. The Applicants include a seismic Cat I buildings that contains SSC in the internal event PRA and also Cat II structures that could impact Cat I and any other equipment in the PRA.

Conservatively, the Applicants have said that failures of these buildings would lead directly to core damage. And in this assessment, there are eight buildings that the Applicants analyzed. Next, please.

Step E, correlations between seismic failures. As you are aware, this step cannot fully perform at this stage because the correlation heavily relies upon the locations and the direction of the

equipment since the information is not fully developed and the plant walkdown cannot be performed at this point.

Therefore, using boundings correlation factors is acceptable for APR-1400's PRA based SMA. The correlation factors between components was assigned by either zero, fully dependencies, or 1.0. Oh, sorry, fully independencies and 1.0, fully dependent.

Seeing that component location in the same building and same elevations are treated at fully correlations, Table 19.1-42 of the DCD show the correlation factors for all the SSCs in the SEL. The Staff found the correlation acceptable.

Task 5, calculations of the sequence level and plant level HCLPF. Per interframe Staff guide 20, it's acceptable, and the Applicants use the min/mad method for computing the sequence level and the plant level HCLPF.

The randoms and the humans failure probabilities were assessed in the combination with the HCLPF for each action in sequence. The Staff finds the seismic accident sequences as consistent with the fragility evaluation, the plant's response, and the key assumptions, and the Staff verified that

no accident sequence has a HCLPF lower than 0.5g. Next, please.

Based on these reviews, the Staff finds that if all of the assumptions remain valid, and all seismic related COLs information items are fulfilled, then the designs can withstand the review level earthquake of 1.67 times CSDRS and meets the .5g HCLPF capacity as discussed in the SECY-93-087 and the related SRM, although the APRIL-1400 PRA based SMA does not support the determination of the important contributors to the seismic in the probabilistic sense.

However, it allows identification of the plant features that are important to the design's capacity. The Staff concludes that the APR-1400 design satisfies the expectation of SECY-93-87 and the related SRM regarding seismic evaluation and capacity.

Next, please. In compliance with the regulations and per COL information items, the PRA based SMA will continue to be updated, COL Applicant will update the PRA based SMA to include size, features, design departures, site-specific SSC, and soil effects.

Also, COL's Applicants will ensure that the results from the PRA based SMA remain valid for

their plant. In addition, COLs holders will develop a seismic PRA in accordance with the requirements of 10 CFR 51.71(h).

That the presentation on the seismic high power. I will stop here if you have any questions before we get to the low power and shutdown.

I just have one slide that focuses on the low-power shutdown period based estimating. For this analysis, the same methodology was used for the at-power PRA based estimating. The low-power shutdown seismic equipment list was developed for the full power list. But in our review, we searched for equipment that was unique to shutdowns, for example, having an open equipment hatch and transporting the reactor vessel head for removal and reinstallation, which you know necessitates the polar crane and also the jib crane was added to the seismic equipment list.

Okay. Seismic-induced shutdown initiators were evaluated for bounding POSs. This is a margins analysis not a seismic PRA. And these POSs were selected to represent the RCS intact and the RCS not intact. So the two bounding POSs which were selected were POS 5 at midloop with the RCS vented via pressurizer manway and the reactor vessel head installed.

The second POS was POS 3B, which is cold shutdown with the RCS intact and the steam generators available.

Seismic-induced initiators were selected from the shutdown PRA and they, of course, include loops, small LOCAs and seismic-induced rupture of the CVCS drain line.

All shutdown sequences had HPCLFs of 0.5g or higher and the plant-level HPCLF is also 0.5g. And we checked to make sure that the seismic cutsets they reflect the significance of operator actions.

And that concludes my discussion, if there are any questions.

Thank you.

MR. PHAN: Up next, the staff will update you with the internal fire PRA and internal flooding PRA.

MR. NAKANISHI: Good afternoon. My name is Tony Nakanishi and I was responsible for the review of internal fires and floods. I fully expected the applicant to kind of go through the details related to the methods. So what I'd like to do is focus more on the staff perspectives at a high level.

So our view is that the applicant provided fairly detailed models for fires and floods that

allows them to support identification of risk contributors from a fire and flooding perspectives. I think it reflects the APR1400 design configuration very well.

Now having said that, there are certain design aspects that are still unknown, like the specific routing of the cables and things like that where you have to make some assumptions. And we found that, in general, the applicant used assumptions that are biased appropriately in the conservative direction.

And you know the other aspects, there are some unknowns at the design stage and so the identification of those key assumptions is important. And there's appropriate mechanisms by the COL items to ensure that those assumptions will come true for the as-built and as-to-be-operated condition.

We also found that the applicant appropriately used the guidance, to the extent possible and also to the extent necessary at the design certification stage.

So based on the adequacy of the models, we found that the fires and floods, the risks associated with them compare favorably with respect to the Commission goals and that appropriate risk insights

were identified to support the design certification application.

So if you could, go to the next slide, Jim.

Just a quick update on some of the open items that were identified and a couple of them have to do more with the documentation of risk insights in the DCD, which the applicant have addressed.

One particular model uncertainty they have addressed is the control room abandonment assumption. They had a crude assumption, relatively, where if there is a fire in the control room and if you're not able to suppress the fire in sufficient time, then you would lead to core damage. So that type of an assumption could affect the risk insights that may come out of the assessment. So we found that the applicant did some sensitivity study to evaluate that assumption.

And you know when you vary the probability of control room abandonment, the important sequences change. In this particular case, the turbine building-induced or fire-induced loss of offsite power became more important.

The other open item that we focused on was the ex-control room operator action ability to

implement the feed and bleed operation. And the thought here was if there was a fire or a flood in certain areas of the plant, what would be the effect on the operator to affect that operator action. And we found that the applicant had properly addressed that in the PRA.

And finally, we already talked about conversion but that -- you know the staff could not finalize our findings until the conversion was complete, which it now is.

So at this point, all the open items are closed and the staff finds the applicant's analyses acceptable.

MR. SCHULTZ: Tony, and for these open time the documentation is also complete?

MR. NAKANISHI: So there is, at the very least, it's a confirmatory item. I'd have to look at --

MR. SCHULTZ: That's what I was asking about.

MR. NAKANISHI: Yes, I'll have to look at that CR but certainly the applicant has proposed acceptable DCD markups. And it may be that we just need to verify that that gets implemented in the DCD.

MR. SCHULTZ: Thank you.

MR. NAKANISHI: So that concludes my portion.

MR. SCHULTZ: Before we go into the conclusions, I think that's next, I just had one question, administrative, perhaps.

KHNP presented results, their results of the PRA results, updated, and so forth. And I'm finding they're different than the table in the SE, safety evaluation by the staff.

I was wondering if you have a simple answer to why the results are different.

MR. PHAN: The information reflects in this presentation from the Revision DCD. It is unclear to me why there are differences.

MR. SCHULTZ: But you noticed it, too?

MR. PHAN: Not until this morning.

MR. SCHULTZ: Okay. That's when I noticed it also. If we could get a clarification on that, I would appreciate it.

MR. PHAN: Yes.

MR. SCHULTZ: Thank you.

The results aren't significantly different but the relationships are a little bit different between the CDF and the LRF. So it would be useful to identify why the -- to straighten out the difference.

Thank you.

MR. PHAN: May I ask the applicants to give us an explanation to the staff because this new. We did not see -- we have not seen these numbers until this morning. If you have the answer handy, otherwise, the staff will come back with the explanation.

MS. HAYES: So I think the numbers they presented were Rev. 0 and Rev. 1 and we're looking at Rev. 2. That might be the answer. We can confirm that.

MR. SCHULTZ: That's Rev. 2 of the PRA or --

MS. HAYES: The DCD.

MR. SCHULTZ: Oh, okay, then it's labeled as Rev. 2.

CHAIR BALLINGER: Can you supply that to Chris?

MEMBER BLEY: It's in multiple places, too. Just for example, on 19 -- I don't have my glasses on -- 19.35, where you give the uncertainty analysis but you showed the mean value for the level 1. That's the value on their table from Rev. 0 and not the value from the later one.

So I don't know how prolific that is. I

didn't chase all those numbers.

MEMBER REMPE: So even though we're not into Section 19.2, the staff SE, the most recent one on page 16 out of 55 and 31 out of 55 gives peak radiation environments of 4.4 times 10 to the minus 5th gray and even elaborates how that they compared with values from prior DCDs.

And if you look at pages 1800 out of 1912 of the DCD, the most recent version we were given, as well as the severe accident topical report page 517 out of 693, you'll see values of 4.4 times 10 to the plus 5 gray, if my division is correct. I tried to blow it up to make sure. But those values seem a little significantly different.

And I assume things will be corrected before it's issued, right?

MR. PHAN: Thank you. With that, the staff would like to get to the conclusions.

On the next two slides, 28 and 29, those are the staff findings. And you could see them in our safety evaluation report, should we run out of time.

May I jump to the conclusion? And we will answer any questions on the staff findings.

But page 30, the conclusions. The staff concludes that the applicant has adequately addressed

the Commission's objectives regarding the preparation and use of its PRA in conformance with the SRMs on SECY-89-102, Implementation of the Safety Goals, and the Commission policy statement under Use of the PRA Methods in Nuclear Activities. Hence, the staff expects that the APR1400 PRA will continue to be updated, improved, and used throughout its life from design certification to COL's application until the end of the APR1400 operation, as required by 10 CFR 52.79(d) and 10 CFR 50.71(h).

And that takes us to the end of our presentation. We are happy to answer any additional questions.

MR. SISK: This is Rob Sisk, Westinghouse. We were just talking to try to reconcile the differences in the table. I just wanted to confirm with the team the values that are in the SER are mean values, which is what it states in APR means to the LER evaluation guidance.

The values that were presented this morning by Kate and Pete were an analysis not a point value -- a point value, not the mean value. So there will be some differences between the mean value and the analytical value that was presented.

So I just hope that that helps for

clarification purposes.

CHAIR BALLINGER: Thank you.

Okay, we're at -- we're late but we're at a convenient break point, where we can shift to Chapter 17. So let's take a -- have a 15-minute recess and come back at 5 minutes after 3:00. That would be when the big hand is on the one and the little hand is close to the three.

(Whereupon, the above-entitled matter went off the record at 2:50 p.m. and resumed at 3:05 p.m.)

CHAIR BALLINGER: Okay, we're back in session.

You have the floor, Chapter 17.

MR. OH: Yes, my name is Jiyong Oh, Andy Oh, KHNP. I will present for Chapter 17 on Quality Assurance and Reliability Assurance.

Chapter 17 consists of the quality assurance program and reliability assurance program. I will introduce quality assurance program and Ross Anderson will present for the reliability assurance program.

First is the quality assurance program. Basic policy of the quality assurance program for APR1400 design certification includes the following:

top management policy to enforce safety culture, and training and qualification, and organizational freedom and independence from cost and schedule, and also it has some authority to holding the work stop if there is something wrong. And in addition to that, direct reporting to top management applying the feedback is included for the quality assurance program.

Next one is approval and implementation, and improvement of the QA, quality assurance program, for APR1400. Quality assurance program was established to comply with the following requirements: 10 CFR 50 Appendix B and 10 CFR 21, Reporting of Defects and Non-Compliance, and ASME NQA-1-2008 and 2009 addenda.

In addition to that, as the quality assurance program for APR1400 was accepted by NRC that QAPD Topical Report is submitted -- Revision 5 is submitted 2016 May. And the SER with no open items was issued 2018, May 8.

QA program for APR1400 is implemented and improved with the corrective action and KHNP reviewed and approved every safety-related document and also implemented to annual QA auditing for all organizations.

MEMBER MARCH-LEUBA: I'm just curious. Is

that auditing by NRC to you, or you to your vendors, or --

MR. OH: We to do our vendor. So this is --

MEMBER MARCH-LEUBA: KHNP?

MR. OH: Right, QA program is for our QA program.

MEMBER MARCH-LEUBA: Okay.

MR. OH: NRC inspection is --

MEMBER MARCH-LEUBA: So you are committed -- KHNP has committed to generally perform an audit of all the vendors.

MR. OH: Right. Correct, sir.

Next one is organization and oversight for the implementation of the QA program. KHNP worked with the pre-qualified design suppliers. We have four suppliers. KEPCO E&C Architecture Engineers and System Designs they are responsible for the architecture engineering and system design. And the fuel and core design is KEPCO NF is responsible for that design. And DOOSAN has provided a major component design.

KHNP performed the following quality functions: KHNP evaluated and selecting, and qualifying the suppliers before contract, and also

reviewed and approved design product provided by suppliers, and control RAI responses and related document changes. And if there is some design change, design change here and we held a Design Review Committee and also the control to interface information through the design suppliers. Finally, KHNP implement annual QA audits and checking the corrective action that help in every design suppliers.

MEMBER SKILLMAN: Andy, before you change, Criterion 3 of Appendix B to 10 CFR 50 is the criterion that applies to design control.

MR. OH: Criterion 3.

MEMBER SKILLMAN: Criterion 3 of 10 CFR 50 Appendix B. That's your first bullet number one.

MR. OH: You mean the --

MEMBER SKILLMAN: I haven't asked my question yet. I'm just trying to outline my question.

So Criterion 3 of Appendix B is design control.

MR. OH: Design control.

MEMBER SKILLMAN: Yes. Design control, for many entities, simply means controlling the design, meaning it controls those individuals that use CAD or whatever the design program is to design a component, or a structure, or whatever. But

underlying that is a calculation code that is used to identify the stress, the strain, the vending, the ultimate allowable pressure, those types of things.

What controls does KHNP impose on the underlying codes and changes to the codes? For instance, there are codes for analyzing the structure, a code for analyzing a pressure vessel, a code for analyzing momentum or thermal hydraulic conditions. How does KHNP ensure that the analytical codes are controlled? If there's a revision to the code from Revision 1 to a Revision 2, how does KHNP assure that the product that they're buying is consistent with the code of record?

MR. OH: So my understanding for the Member Skillman's question is in case in our APR1400 design DOOSAN applies some code and they implement some design. Right after that, they apply another code. They change their design. How does KHNP control that changes?

MEMBER SKILLMAN: That's correct.

MR. OH: I think for -- if that -- basically, we have the QAPD, the quality assurance program for the project. That's the big one. And then based on that, the QAPD is the contractor have their own QA program. However, in that case is if

there is some substantial changes, they have to report to the KHNP main control person. No notification for such big changes and we are not allowed to do that because that change can affect -- change can create some interface data. Not only is the change influencing their company, that change can influence to other chapter, other areas.

MEMBER SKILLMAN: That's correct.

MR. OH: So the KHNP is kind of the central head center. So if there is some change that happens, they are responsible for reporting it to the KHNP. We control and also we share the information with other contractors. If that change is indirectly influenced to other vendors, other contractors.

So that is in our QA program. That works in our procedure.

MEMBER SKILLMAN: I hear you say that it's covered in your QAPD --

MR. OH: I think so.

MEMBER SKILLMAN: -- and that KHNP is the central clearinghouse to maintain configuration control for what your contractors are doing.

MR. OH: Major loophole we have, as I said before, we have Design Review Committee. If there is some changes required in some design, we have some

Design Review Committee. That's not only for KHNP. Other design companies, they all review the design and also they will share the information with other contractors.

So maybe very editorial or simple changes is not very significant. However, if they think as a design have to be changes, they have to go through the Design Review Committee.

And also if the design change is decided in our Design Review Committee, then they also have to apply their QA procedure, depending on their company.

MEMBER SKILLMAN: I would suggest that you wisely point out that very small changes probably don't need control. I would suggest to you that every change must be evaluated and the ones that are of no consequence and known to be of no consequence can be dismissed but there must be some form of an evaluation.

If I go back in the last 20 or 30 years of my background on this exact issue, I'm always surprised at how a very subtle change that was originally identified as being of no consequence really had a consequence but it was not recognized because no one took the time to sit down and think about what that small change might do.

MR. OH: Yes.

MR. KIM: Yes, this is Jungho Kim. I am KHNP APR1400 deputy project manager.

So as you just said that actually we have every minor is controlled by our KHNP QA process. So every change is controlled by KHNP QA process.

So we control everything, including all the major design change also.

Also we just every -- every year we audit each design company. And so we just check if the original quality change and we have a chapter called application and the validation, and verification, that kind of quality we are checking/auditing.

MEMBER SKILLMAN: Thank you.

CHAIR BALLINGER: So your audit converts it from a one-way path to a two-way path, which is good.

MR. OH: At least is comprised of DCD. DCD is our basic document. Even though there is a simple change, KHNP controls the DCD changes. So we don't allow some, even if it is editorial, the design company cannot change their DCD. Owner of DCD is KHNP so they need some approval from KHNP.

But calculation document, actually the owner is the design company. So it is -- that is very

substantially important things that have to report to us. On the calculation there is some editorial, they have their own QAP program. So they can correct it.

But the majority of things KHNP is controlling for the quality assurance program for all the contractors.

MR. SCHULTZ: The important thing is it's all configuration control and the audit process of the vendors are going to identify whether there's been any changes that are significant to affect the DCD.

MR. OH: Correct, sir, yes. So every --

MR. SCHULTZ: But your expectation is that it's going to be reported to you directly.

MR. OH: Right. And also for in case of the audit, the idea of actual did audit every year and I just audit -- during audit, we have some authority to still review the calculation for the vendors. So we saw the calculation, everything, and compare it for the previously what changes made between last year to this year.

So if there is some change that happens, then we can ask the vendors why these have changes. This is significant.

MEMBER SKILLMAN: Of course. Thank you.

MR. OH: Just a regular update or

something.

MEMBER SKILLMAN: What I wanted to communicate also is changes that do not seem to be significant may be. For example, the change from the word may to must or must to may can change an outcome completely or the change from will to shall can create that same result. And unless the inspectors have a magnifying glass that's thick enough to catch that, then by very small increments, the integrity of the design can be diminished and it can happen while we're sleeping. And that's my point.

MR. OH: Okay, thank you.

MEMBER SKILLMAN: Thank you.

MR. OH: Thank you for your comments. So KHNP will take into consideration your valuable comments to improve our QA program. Thank you.

So next is for reliability assurance program. That will be covered by Russ Anderson.

MR. ANDERSON: Thank you, sir. Can you hear me okay?

CHAIR BALLINGER: Yes. The light is on.

MR. ANDERSON: Yes. I'm going to talk about the efforts that we've made to revise and update the reliability assurance program in response to NRC questions and feedback.

I need to apologize. Two weeks ago I couldn't speak and my throat is a little raw. I'm going to try to muddle through. We've got about eight slides here. So it should not take long.

Following NRC feedback on initial, a couple of our RAIs, we went in and we began by revising the reliability assurance program, the RAP procedures entirely. We gutted them and overhauled them considerably. I made extensive use of my own background. I sat on the Expert Panels for the Maintenance Rule at North Anna and Surry for years. And the RAP has multiple functions but one of the big ones is to feed into an eventual Maintenance Rule, Maintenance Rule-equivalent program, once the site is up and running.

So I've seen what works and what doesn't and what it needs to run a good program and revised the primary three procedures here, 9, 10, and 11 -- implementation, Expert Panel roles, responsibilities, and the actual process itself for identifying which components should be designated as risk significant.

Once that was done, then we went ahead and the Expert Panel itself convened in order to review the final PRA model results and update the RAP list. It had been through a couple of revisions, at least,

previously. We've now got Rev. 3 to the RAP Notebook that has been issued and it's based upon recent meetings of the Expert Panel using these revised procedures.

We took a look at all the available PRA analyses, full power, shutdown, internal events, fire, flooding, Level 1 and Level 2 in order to establish a criteria for whether to evaluate components as potentially risk-significant. We utilized the Reg Guide 1.200 criteria, RAW greater than 2 for individual component or Fussell-Vesely greater than half a percent for an individual component.

There is also a common cause criterion. We'll mention that later.

We had several meetings of the Expert Panel. The procedures were revised in the middle of 2017 and we had several meetings of the Expert Panel in the fall of last year.

The first meeting was a face-to-face with the full panel in Korea but the models had not been fully revised so we weren't ready to do the formal Expert Panel. However, what we did was a dry run of the procedures. We wanted to make sure that they were doing what we need to have done, that is to put together a good panel, a good criteria, good

procedural steps, and went through and had a good dry run with the procedures.

Later, we had an actual formal meeting, once the models were completed, once we had the PRA results, the statistics for the importance measures. We went ahead at that point and went through all the analyses and combined them, collated them, put them together, developed PRA recommendations for everything that exceeded the criteria or were close, for that matter.

We even took a look at a few things that weren't modeled but merited some potential qualitative assessment.

The second meeting took a full week, very detailed, very extensive discussion, reviewed the full scope of everything. We took a look at possible changes, all possible additions, deletions, and went through a rigorous, a very thorough, and in the end, I found it to be a reassuring process for establishing the latest and the appropriate scope for the reliability assurance program.

Subsequently, we did have a couple of electronic meetings as well. The procedures allow us to have some short discussions limited scope, simple enough that they can be handled basically by an

electronic or even an email solicitation. So we had a couple of those during our scope as well. But the majority of the work was done in the face-to-face meeting the week-long.

The process itself during the face-to-face meeting was very rigorous. We had a full team in there, representatives from all appropriate disciplines -- PRA, operations, maintenance, QA, I believe there are some others as well in there. We had at least one of several people from each of those disciplines met together. The discussions were very good, very thorough.

I compared that to the things that I saw in my prior experience with operating plants in the United States and the discussions were just as rigorous, just as thorough, as anything that I have seen stateside.

We had a high level of experience among the members and good participation throughout. Nobody came in and sat and watched. There was good discussion everywhere, everybody that needed to fit on in and contribute from their experience, be it ops, maintenance, whatever, and sharing was ready to do so and did so.

We also made extensive use of subject

matter experts and had a fairly steady stream of SMEs in there throughout the entire process.

My observation, I thought it was consistent with practices in Maintenance Rule programs for currently operating plants with mature programs in the United States.

We did have a couple of open items on the process. The NRC had asked us to revise the DC Chapter 17.4 partly because when it was first drafted the Standard Review Plan version that was Rev. 0 and a Rev. 1 was subsequently issued. We needed to update that. We needed to do some cleanup as well and we did that.

Question 17.04-2 was a request for additional COL items. These issues have been resolved and they presently are in a confirmatory status.

Let's see, discussion of the RAIs. I touched on that already so I won't review that again.

Status right now. Andy, may I ask you to address Chapter 17?

MR. OH: Yes. I would like to mention about the current status of Chapter 17. Chapter 17 has been completed and KHNP continues to monitor Chapter 17 to assure any conforming changes are addressed.

One open item that has been made in Phase 3 that was identified in Phases 2 and 3 has been resolved following extensive discussion with the staff.

And changes in Chapter 17.04 as reviewed and marked in our response will be incorporated in DCD Revision 3.

So currently in Revision 2, the DCD has not incorporated latest version of this response.

MR. ANDERSON: Okay. Can we answer any questions?

MEMBER KIRCHNER: Yes, I'd like to ask, as a result of going through all the PRA materials that you reviewed, did you change any risk-significant -- did you change your list of RAP SSCs as a result of this exercise?

MR. ANDERSON: Yes, we did. Many --

MEMBER KIRCHNER: Are you going to highlight what you changed? Just I don't --

MR. ANDERSON: There are too many to highlight but I can give you some examples.

MEMBER KIRCHNER: Many? Okay, if you would.

MR. ANDERSON: The prior table had rows with several hundred different entries. The current

table has rows with several hundred different entries. And these are grouped. All the trains show up in a single row. So we're looking, oh, it could be a thousand or so components, easily, in the list. The majority of them didn't change -- diesel generators, aux feed pumps, SI pumps.

MEMBER KIRCHNER: All right.

MR. ANDERSON: They were there. They're still there. But we still went through and took a look and we said okay, these are risk-significant last time around. The PRA results still say that they should be risk-significant. We put it to the committee and the committee concurred with their retention.

We added in a number of components. I actually looked at an example just the other day. There were a couple of check -- not check valves but manual valves on a split header that were modeled and just barely risk-significant for a manual spurious closure but there was also a common feeder that wasn't modeled, which would be as risk-significant as failures of both of those at the same time. So we added that one on in on a deterministic basis.

We took a look and made sure we have train consistency. If Train A is risk-significant or

recommended as risk-significant by the PRA results, Train B gets grouped in with it. We don't separate trains that way. They're all together. It was be confusing in an application to have anything different.

There were a few that were no longer recommended as risk-significant by the PRA results. The results dropped below for RAW 2, Fussell-Vesely 0.5 percent. In that case, we highlighted it to the panel and we point it out. Here's this component. Here's what it does. It was previously risk-significant but now the PRA results suggest that it should be low risk.

We supplemented with whatever discussion was pertinent for those components, put it to the committee, they made the call.

We also took a look at some qualitative issues. I gave the example of the manual valve. We took a look at a few others as well but there were enough changes that probably would not be worth the trouble to try to go over that list.

I hope that is an acceptable answer.

MEMBER KIRCHNER: Yes, thank you.

MR. ANDERSON: Okay. Anybody else?

MR. SCHULTZ: Ross, there's a -- in the

safety evaluation, I expect there's a lot more detail in the CDC -- DCD. The expectation for the COL is the applicant is to describe the uses of PRA in support of a licensee program such as the reactor oversight process.

Based on your experience, how do you see that transition occurring? How difficult a process do you think it will be based on all the work that has been done by KHNP? How would you see the process going for the COL applicant?

MR. ANDERSON: I think we've staged it pretty well. We have a good list with a good basis. It is well-documented. They'll need to make sure they transition on in because right now a RAP program is, by design, a pretty good match with Maintenance Rule, which is one of the things that it feeds into but it's not perfect. There are some tweaks, some subtle differences with the way common cause is handled, for example, and they'll need to take a look at that.

But by and large, the majority of it ought to be a pretty good fit. We will be available to support them when that time comes that they need their support.

MR. SCHULTZ: Would you expect that there would be a need for Expert Panel advice in that

transition and subsequently?

MR. ANDERSON: Possibly. I could see getting some questions why was this risk-significant. Why was this not risk-significant?

By and large, I would not expect a lot of such questions. Really I think the basis is pretty good. We've listed only the components in the RAP notebook but we've listed the risk statistics. And for everything that's not modeled with a PRA basis for at least the initial recommendation is risk-significant. Everything that is qualitative has a measure of discussion with it.

MR. SCHULTZ: Okay. So even though there will be some decisionmaking, important decisionmaking associated with balance of plant that not now included in the PRA, that there will be some seismic evaluations that need to be done. The Expert Panel work that you've done so far in terms of this, the RAP program, should be satisfactory either to carry on through or to serve as a sufficient starting point.

MR. ANDERSON: It would be presumptuous to think that they'll just take everything as is and not question it.

MR. SCHULTZ: Of course.

MR. ANDERSON: They can and they should

take a look at it. They may wind up doing some tweaking, to be honest, but I expect that to be minimal.

MR. SCHULTZ: And it sounds as if your guidance is documented sufficiently to provide for that transition and support.

MR. ANDERSON: It's almost 300 pages, yes.

MR. SCHULTZ: That sounds good. Thank you.

MR. ANDERSON: It's a rigorous document.

MR. SCHULTZ: Thank you.

MR. ANDERSON: Thank you, sir.

Anybody else?

Okay, thank you very much.

CHAIR BALLINGER: Staff ready to go?

MR. LINTZ: Good afternoon. I'm Mark Lintz, Chapter PM for Chapter 17, Quality Assurance and Reliability Assurance.

I am aided by my reviewers, Aaron Armstrong, who did the quality assurance, and Ayo Ayegbusi, who did the reliability -- the RAP.

I'll turn it over to Aaron.

MR. ARMSTRONG: Is this on?

MR. LINTZ: Yes.

MR. ARMSTRONG: Okay. My name is Aaron

Armstrong. I performed the review for Chapter 17.1, .2, .3, and .5. The regulatory requirements were for Appendix B to Part 50, 10 CFR 50, and they have been satisfied.

The QAPD Topical Report was approved and provided the QA requirements for Sections 17.1, .2, .3, and .15.

We also performed a QA implementation inspection in July 2017 and it's publicly available under Inspection Report 05200046/2016-201. All the identified items, which I believe were four, have been closed and the closure and the information that KHNP submitted to us addressed our concerns from the inspection activity.

So with that, we didn't have a lot of questions last time but I would like to entertain any questions you guy have about 17.1, .2, .3, and .5.

MEMBER SKILLMAN: Aaron, let me ask the same question to you that I asked the applicant. It has to do with subordinate analytical codes under Design Control Criterion 3, Appendix B, 10 CFR 50.

What I'm really thinking about is San Onofre. A very slight change, use of codes, everybody thought everything was fine. We end up with a plant out of service. But that's just an example of what

we've seen for years in the industry where a subordinate code has changed, a very slight change that has a major impact.

So my question is here's a design cert just coming out of the box. What review or what attention was given to assuring that subordinate design codes have the control that they need so that the product that is described in the DCD is what it is advertised to be?

MR. ARMSTRONG: Okay, during our inspection activity and I guess as a QA background, when things are procured as Appendix B or safety-related in accordance with 50.2, as you know Part 21 would be invoked.

MEMBER SKILLMAN: Only -- Part 21 is when there's a deficiency that's been identified.

MR. ARMSTRONG: When a deviation is identified, that is correct. But there is oversight during the QA part, which is the audit activities that are performed from the supplier to sub-supplier because they are procuring safety-related requirement or safety-related components.

So in that avenue I think that the major part of the Appendix B stuff, if the program is filed correctly and implemented, things would be identified

and evaluated. And that's what we observed when we did our inspection.

MEMBER SKILLMAN: Fair enough. Okay, thank you. Thanks. Okay.

MR. ARMSTRONG: If there's no more questions, I'll pass it over to him and he can do the RAP section.

MR. AYEGBUSI: All right. Once again, my name is Ayo Ayegbusi from NRO.

All right, so as the applicant mentioned a few minutes ago, the initial submittal/application was written in accordance with Rev. 0 of the SRP. However, per the application documentation guidance, we had a Rev. 1 of the SRP available more than six months prior to when the application was submitted.

So the first RAI we wrote after reviewing this section was for the applicant to either provide a reliability assurance program that followed the guidance in Rev. 1 of the SRP or their, as with all SRPs, their own program and then describe how it deviates from the SRP guidance.

So subsequent to the RAI and a lot of extensive discussions with the applicant, the applicant did end up using the guidance in SRP 17.4 Revision 1, disseminated that for our evaluation. And

because of the PRA, really the PRA update, they also updated the D-RAP list. And so we evaluated that.

As far as the program goes, they, in essence, utilized the guidance in SRP 17.4. There was no particular aspect that they deviated from. And for the D-RAP list, what we did the first time around -- well, early on what we did was because we already knew that they had to revise the program, which had a different set of requirements on how you develop your D-RAP list and how you present your list, but also early on we already knew that the applicant already planned revisions and updates to the PRA. We did not review the initial D-RAP list.

So our review of this, of the revised D-RAP list was the first review of the list. Reviewing the list, we had significant interface with applicant about items on the list but the most or probably the only real significant one was the in-containment refueling water storage water tank sump. That, part of what we do in our review is we look at the risk-significant SSCs that are provided in Chapter 19. And this IRWST sump was on that list as quite a bit risk-significant. However, we didn't see it in the D-RAP list, as expected.

So we had a discussion with the applicant

and they performed an Expert Panel. And we subsequently added the IRWST sump to the D-RAP list.

And you know so part of that review of the D-RAP list looked extensively at the work that the Expert Panel did. Part of that was also looking at the makeup of the panel, the experience of the panel, the diversity of that experience, and the discussions that we had ensuring that it wasn't just on PRA but also included operating experience and all the deterministic evaluation and things like that.

And so we were able to conclude that the Expert Panel was thorough, it was adequate, and that the D-RAP list was developed using the methodology that they had proposed in DCD 17.4 and that the list was comprehensive.

That's pretty much it.

MEMBER SKILLMAN: Let me ask this. The IRWST in the APR1400 is very -- fulfills the same function and performance requirements as a BWST or an RWST standalone on a P or B in our current licensing arena.

So with the IRWST not being considered originally, did that lead the NRC staff to ask the question what's the extent of condition; if this was forgotten, what else was forgotten?

MR. AYEGBUSI: That is correct. So like I said, what we typically do is compare the Chapter 19, Risk-Significant SSCs, to what's on a D-RAP list. And whatever we don't see that was -- whatever we see that wasn't included on that list we ask questions about that.

As the applicant had mentioned earlier, they had some manual valves that we saw that didn't end up on the list. And so in those cases, we look to see what the discussions were had and how that was documented.

So yes, it's -- the extent of condition was done both on our side and on their side because that was a question that we asked to see why it was missed.

But as you've seen, there are a lot of moving parts as we reviewed the application.

MEMBER SKILLMAN: Thank you. I yield. Thank you.

MR. AYEGBUSI: You're welcome.

MR. SCHULTZ: But you were satisfied with what you learned in terms of the cause of the event, if you want to call it that, that certain items were missed in the review that had been done to date. And that was before the Expert Panel was convened?

MR. AYEGBUSI: No, no. So the Expert Panel was convened, the list was submitted as part of the revision of the -- or the revised write-up to that, to Chapter 17.4.

So in essence, we can say that the IRWST sump was the only item that was missed that was not addressed by the Expert Panel.

MR. SCHULTZ: Okay.

MR. AYEGBUSI: So it wasn't -- in essence, it wasn't provided to the Expert Panel for consideration as part of the risk-significant SSCs. All the items were provided to the Expert Panel.

MR. SCHULTZ: But you then explored why that happened?

MR. AYEGBUSI: Why?

MR. SCHULTZ: Why the sump was not provided to the Expert Panel.

MR. AYEGBUSI: Yes, we raised it. I don't remember what the issue was.

MR. SCHULTZ: Okay.

MR. AYEGBUSI: But like I said, I think it went along the lines of what revision, what was available when.

But actually on this particular topic there is -- the -- now it's coming back to me.

There's the IRWST sump and then there's the strainer or the trash holdup --

PARTICIPANT: Trash rack.

MR. AYEGBUSI: It's a trash rack but I forget the specific name. So, in essence, it wasn't broken down. It wasn't initially broken down and it was subsequently broken down in the modeling. But because the list was based on the initial list, they had the trash rack, for lack of a better word, but didn't have a sump pump.

MR. SCHULTZ: Did not have the detail that you would have liked to have seen.

MR. AYEGBUSI: Well, it didn't have the -- it didn't include the results from the PRA update. So when the PRA was revised --

MR. SCHULTZ: Okay.

MR. AYEGBUSI: -- they broke it into two components.

MR. SCHULTZ: I understand.

MR. ANDERSON: May I comment on it to try to assist a little bit here?

It's not the IRWST but the sump itself and the strainers that were the issue that Ayo identified. The IRWST's suction source for a side is a risk-significant, no question.

What happened is that when we took a look at the model results, we had a single term in there identified with a description as the sub-strainers RAW full power internal events, about 3,000, no question. They went to the panel for recommendations risk-significant. Many of the other analyses, fire, flood, shutdown, showed up a lot of places highly risk-significant.

What happened during the Expert Panel discussion was that the panel noted that we were talking about the filtration function. The strainers are the second line of filtration. They are not the first line. The first line are what we call trash racks. They are 1.5-inch mesh and they stop the larger debris up to an inch and a half. The strainers are 0.09-inch and they are in an entirely separate location.

And the panel said no, it's the first line of filtration that should be most risk-significant. So they were added to the RAP list as highly risk-significant. It looks like they're in the odd place in the RAP list because they don't have an SI tag number. They have an IW for in-core water tag number. So if you look under SI, you won't find them. If you look under IW, India whiskey, you will find them and

they are in there as trash racks because that's the first line of filtration and the most important.

The NRC staff identified correctly does that mean that the strainers are not important. It was a good question. It was a good catch. So we had a subsequent solicitation of the Expert Panel, emailed electronic, do you believe that the strainers should be added on in. They concurred. We added them on in.

CHAIR BALLINGER: But the trash rack is not part of the IRWST. It's part of this HV -- it begins with HV and I'm forgetting the last letter.

So that's where the trash rack is.

MR. ANDERSON: It is the first line, the first of two stages of filtration for recirculation of water back into the IRWST.

CHAIR BALLINGER: Yes.

MR. ANDERSON: The first function, the most important one; the second one, we decided to add that in as well. That was the question that the NRC staff correctly identified. It was a good catch and we added it back in. So, additional background.

MR. AYEGBUSI: If I could just add I think we missed a word here. It should be IRWST sump strainer or IRWST strainer. So it's incomplete, just FYI.

CHAIR BALLINGER: Any questions?

Okay, we change topics again. Can we get the KHNP folks up with 19.2?

Okay, ready to go?

MR. SISK: Rob Sisk, Westinghouse. I will introduce Mr. Byung Jo Kim to lead us through 19.2.

MR. KIM: Okay, thank you. Good afternoon, ladies and gentlemen. My name is Byung Jo Kim. I'm working for KEPCO Engineering and Consulting, KEPCO E&C, senior analyst.

I would like to introduce my colleague Dr. Chan Young Paik, Vice President of Fauske and Associates from Illinois. Thank you. He has supplemented the Severe Accident Evaluation by using MAAP code in this process.

So this morning we have been discussing the DCD Chapter 19.1, the probabilistic risk analysis. At this time, I am going to talk about DCD Chapter 19.2, Severe Accident Evaluation, in determining staff's point of view.

Firstly, I am going to talk about what contents included in 19.2, what key documents were submitted to the staff, and what major changes of DCD were made during the Phase 2, 3, and 4, and the list of open items.

And open items are summarized briefly and current status of Chapter 19.2 is discussed in more detail.

The first section of 19.2 is what the regulations are concerned in the severe accident evaluation, such as SECY-93-087 and Reg Guide 1.216.

The second section of 19.2 described severe accident prevention design of APR1400 and how APR1400 design is able to prevent the severe accident from anticipated initiating events, such as ATWS, mid-loop operation, station blackout, fire, and ISLOCA.

The third section illustrates severe accident mitigation features of APR1400 and the deterministic evaluation of these features. Overview of containment design is given here in this section, and severe accident progression of APR1400 is described. The design characteristics of severe accident mitigation features are addressed in detail.

The methodology analysis we did of these mitigation features are also given.

The first section is called containment performance capability. The goal and conclusion of containment performance analysis is addressed in this section.

And the following sections presents

accident management, consideration of potential design improvement, and combined license information.

The main document for 19.2 is obviously design control document, Tier 1 and Tier 2. Besides the DCD, the technical report, severe accident analysis report were provided.

And during the Phase 2 and Phase 3, 22 RAI items were asked by the staff and we had five open items from the Phase 3.

As a response to these RAIs, a number of DCD changes were made and confirmed by staff. What you see in this table is the important ones among the changes.

Subsection 3.2.1 covers in-vessel melt progression. Initial DCD did not contain the vessel overhead failure mechanism. The DCD Revision 2 incorporates the detailed vessel failure mechanism considered in MAAP4.08.

Subsection 3.2.2 details with ex-vessel melt progression. Initial DCD did not have how we take into account the uncertainties relating to the ex-vessel melt progression. So we revised the DCD such that the risk of considerable uncertainties concerning the calculation and conservative approach to account for the uncertainties.

Subsection 3.3.3 denotes molten core-concrete interaction. Because DCD Revision 0 had only brief description for linking CORQUENCH code and MAAP4 code. So we revised the DCD to incorporate the following: initial conditions and conservative approach used in CORQUENCH calculation, and modeling of heat transfer across upper crust. And missed figure for MAAP result was also corrected in the DCD.

Subsection 3.3.5.2.1 contains fuel-coolant interaction. Initial DCD did not provide specific analysis results of in-vessel explosion. Therefore, DCD was revised to incorporate plant-specific dynamic load due to the in-vessel steam explosion and lower head integrity assessment against the predicated load is also incorporated.

Subsection 4.1 concerns containment performance goal. DCD was revised by incorporating the goal addressed in Reg Guide 1.216 Regulatory Position 3.

Subsection 4.2.1 handles combustible gas control inside containment. DCD Revision 0 had no assessment results against the hydrogen combustion load. So DCD was revised to account for containment integrity under the hydrogen combustion in terms of Reg Guide 1.216, Regulatory Position 2.

Subsection 4.2.2 represents the containment pressurization results. In the initial DCD, there were only brief description for containment performance results. So we have revised the DCD to incorporate the following: how we select the accident scenarios based on the PRA and deterministic viewpoint, how we determine bounding pressure and temperature expected from the selected sequences, and how and what we modeled for containment performance study using finite element approach.

CHAIR BALLINGER: Now I hate to keep asking this question but I see Rev. 0. I see Rev. 2. I don't see Rev. 1. And we'll settle that with the staff, I guess, but we're operating with Rev. 1 in our case.

MEMBER MARCH-LEUBA: I believe this morning you told us that the DCD is against Rev. 2.

CHAIR BALLINGER: No, I thought he said it's against Rev. 1. Two? Okay. All right, cleared up.

MR. KIM: Okay. And specific result of train from the finite element study is also incorporated in Revision 2. And what the conclusion for the containment integrity on the basis of Reg Guide 1.216, Regulatory Position 3 is also

incorporated into the DCD Revision 2.

By Phase 3, we have five open items in Chapter 19.2. The first two items are steam explosion issues and the next two items were containment performance analysis items. These four items all have been introduced in the DCD changes listed so I don't have to tell you.

And the last one item, 19-53 is the MAAP confirmatory calculation items, which was incorporated in DCD Chapter 19.1 PRA section.

First open item was in-vessel steam explosion, question number 19-63. The staff asked for the justification of numerical code and the triggering time used in the analysis.

So KHNP responded that the numerical and the TEXAS-V code and the trigger in time can create a conservative result and, accordingly, revised the technical report in 2017. Consequently, this item was resolved and closed.

The second open item was ex-vessel steam explosion, question number 19-64. Staff asked similar questions with in-vessel steam explosion for the first two questions. And plus, the staff asked the details of loading -- pressure loading influenced on the cavity walls and the possible concrete damage.

KHNP provided the conservative load distribution, along the cavity walls, as it is kept fluid and would take into account uncertainty of underwater explosion phenomena.

And concrete and line plate damage details and the conclusions were also provided in the response and incorporated in the technical report and DCD January of 2018.

So as a result, this item was confirmed and closed.

Question number 19-70 asked finite element model of containment such as the global and local nonlinear finite element models, such as -- and clarifying accident temperature and updating DCD.

As a response, KHNP goes into detail and provided the details of finite element models such as used -- what code is used in analysis and model elements.

KHNP also provided the conservative bounding temperature based on the severe accident analysis. The degradation of material properties by the elevated temperature was also provided.

The response was incorporated into DCD Revision 2 and this item was also closed.

The last open item was severe accident

sequences contributing to the core damage. The Question 19-71 asked identify the more likely severe accident sequences and they apply the loads on the containment liner.

KHNP responded that the selection of accident corresponded to the higher than 90 percent of core damage frequency from the PRA response study. And plus, the conservative accident sequences for the dominant initiating event were also taken into calculation.

So accident selection complies with Reg Guide 1.216, Regulatory Position 3. The bounding pressure and temperature was decided from the containment response analysis of those selected sequences. And then we applied this bounding pressure in temperature to the containment RSS as integral to the containment liner.

The DCD was revised accordingly and this item was resolved and closed.

Now let me tell you the current status of DCD Section 19.2. KHNP continues to monitor Chapter 19.2 to assure any conforming changes are addressed. By Phase 3, there were four open items. They were items identified in Phase 2 and 3 have been resolved with adequate and sufficient discussion with the

staff.

So changes in Chapter 19.2 as reviewed and marked up in response to the RAIs were incorporated into the DCD Revision 2, which all of you see it.

So as official, Chapter 19.2 is concluded with no open items right now.

Abbreviations shown in this talk are listed here.

And lastly, we have three COL items in Chapter 19.2 associated with equipment survivability and performance of rod penetrations, such as equipment hatch and personnel airlocks, and the development of an accident management plan.

So this is the end of my presentation. Thank you for your time.

MEMBER REMPE: So could we ask you to go through the backup slides? I think it would allow us to keep the discussion in an open meeting versus having a closed meeting with the staff's slides later, especially, I believe, starting like on slide 15.

MR. KIM: This one?

MEMBER REMPE: Yes, and just go through all of them, if you don't mind. I might have a few questions and interrupt you but I would appreciate it.

MR. KIM: Regarding the in-vessel

retention and external reactor vessel cooling, the first item, and the updated PRA does not rely on the presence of this system. And APR1400 Chapter 19.1 and Chapter 19.2 analysis did not take credit for this system.

MEMBER REMPE: So you've got two plants up and running. One of them is taking credit for external reactor vessel cooling, Shin-Kori, right?

MR. KIM: No, no.

MEMBER REMPE: It does not take credit for external reactor --

MR. KIM: Yes, yes.

MEMBER REMPE: It does take credit for --

MR. KIM: No, no. No credit in Shin-Kori Unit 3 and 4, and in our project.

MEMBER REMPE: Not in the NRC project but in Korea you do implement external reactor vessel cooling?

MR. KIM: Yes, implement it but we do not take --

MEMBER REMPE: Credit.

MR. KIM: We do not take the credit.

MEMBER REMPE: So you don't have any accident management option at Shin-Kori in Korea for using it.

MR. KIM: The Shin-Kori Unit 3 has IBRW system as one of the SAMG options.

MEMBER REMPE: Right, that's what I was trying to get to.

MR. KIM: Right.

MEMBER REMPE: So you do have it there.

MR. KIM: Right.

MEMBER REMPE: In the United Emirates, the plan is to not have it in a severe accident management.

MR. KIM: Yes, not yet.

MEMBER REMPE: Is that the only difference? Do you have the same structural integrity? In response, before you get to severe accident, you implement and start up the cavity flooding system. It keeps flooding up. You don't stop it at a certain level. You have in-vessel retention, even if you don't take credit for it. You don't have any changes. You just keep flooding it up and you know that water is going to be going in the cavity, no matter what. There is no difference whether you use it or not or whether you take credit for it or not.

MR. KIM: According to the SAMG implement code at Shin-Kori Unit 3, the operator can choose one

of two options, if he wants. One is the cavity flooding system, which means that the final level does not leach into the reactor vessel lower heads. It's a kind of IVR-ERV system. So if the operator can use this system because this system relies on many external system, so external power supply, and operator manual actions, and so on.

And people may choose the IVR-ERV system as one of the SAMG systems. The operator initiates the shutdown cooling pump or open gates to make pump to flood the reactor cavity up to the platform to somewhat make it better.

So operator only need to choose one option between the two.

MEMBER REMPE: So if he chooses the option to have external reactor vessel cooling, more water has to go there and it's not available for whatever else you might want it in the plant.

And your SAMGs have to account for the more complex situation where the operator has two options to select.

MR. KIM: Right.

MEMBER REMPE: So you have that situation that would differ.

But as far as how the plant is built,

there is no difference in what you're going to recommend at a place where it says I don't want to deal with that. You don't put more -- like in the APR1000, they reinforced the insulation around the vessel. They made sure there were flow paths for water to come in and steam to go out. But you go ahead and have that same reinforced structural integrity in your plant, whether they use it or not, right?

Okay, that helps. Then, the only thing, I guess, and I'll save my question for the staff is I hope the staff will know how to -- I guess they don't review SAMGs, so it doesn't matter. But it just seems like how the COL applicant implement it and whether it's adequate, and the structural analyses and things like that that has to be done, the assessment for the structural loading, the staff -- the guidance on what's acceptable or not is what's concerning me, that they're just leaving this to the COL applicant as I tried to convey earlier.

Is there something you can tell me that says oh, KHNP will know how to help this COL applicant and the staff so that they're not going to just be starting from scratch?

DR. PAIK: This is Chan Paik from Fauske

and Associates.

I think in the APR1400 in the SAMG they are going to use an in-vessel retention with ex-vessel coolant plus in-vessel injection. So the way they accommodate is typically using the shutdown cooling pump. It may take 30 minutes to raise the water level all the way to the holdup level.

Then after 30 minutes, they hope the POSRV depressurizes the RCS. So whatever the power and shutdown cooling, the pump available, they can start injecting inside the vessel. Okay?

So when you have an ex-vessel cooling plus in-vessel injection, not every case can be saved, especially if you start the in-vessel injection after the most of core be located to low plenum, it may not be possible to cool with the injection alone, even with the ex-vessel cooling.

So that kind of case can be scary, resulting in a vessel failure, even if you have ex-vessel cooling and in-vessel injection. But as long as the in-vessel injection goes in before a lot of a mass relocate to low plenum, with the ex-vessel cooling, with the in-vessel injection, then you could prevent a massive failure.

MEMBER REMPE: So, again, SAMGs are going

to have to have identified what signals the operators should be looking for decide how things are working, if it's effective or not. But of course, again, staff doesn't review the SAMGs.

But I just am kind of thinking about what -- there's a lot more uncertainty that I wish wasn't there.

MR. KIM: Yes. And so on the basis of SAMG develop for Shin-Kori Unit 3, of course there are so many preconditions for operator to help him to decide IBRW can operate or not or should I try to zip it. And one of the preconditions, for example, of course the RCS pressure and the availability of it based on electric power and so on.

So SAMG developed for Shin-Kori Unit 3 basically tried to initiate in the IVR-ERV dual purpose of higher priority as IVR-ERV. So operator try to find out IVR-ERVC is available right now, then he goes onto this way. But he find out it's not available, then he will go to the flooding system flow paths in the SAMG.

MEMBER REMPE: Okay, thank you. Go ahead and finish the next few slides, too. I'd like to hear your discussion about HDR.

DR. PAIK: This HDR is a very

compartmentalized containment. Now is they remove the HDR so HDR containment is no longer available. But it's a 20-meter diameter and about 50-meter height. So from the bottom to top is about 60 meters. From ground level, it is about 50 meters.

So if you look at the picture on the left side, you can see some of these spiral paths, stairwell. And the other side is the equipment hatch is open. So those are two of the major flow paths. Between them, there was no flow paths. So that means the flow has to come down below that stairwell and cross the other side and go up over all those things.

And another key thing it has an external spray like the AP1000. It has actually a steel shell containment and they spray on the outside of the steel shell to cool the containment.

So if we look at the HDR 11-2, light gas injected mixes well the gas above but below the injection point, they did not see much impact of these injection. So whatever the steam or hydrogen injected above it, it has it but below it, it doesn't see the high temperature steam or the gas.

But if you look at the HDR result, especially in the dome hydrogen concentration, actually they use the helium in HDR, concentration

goes up very high and then comes down. What happens is when external sprays are turned on, essentially you have a rapid cooling at the top. So you are sucking the steam and air to the dome. So what's left is sort of a hydrogen kind of thing. So hydrogen concentration goes up.

And then after a certain point, because the gas is continuously being cooled by external spray, the dome temperature is kind of coming down and eventually the density may get heavier than the lower part because the temperature is going down. And then we have this kind of a rapid turnover. It means whatever the highly concentrated gas comes down into the lower part and mixes.

CHAIR BALLINGER: Excuse me but a couple of us are a little bit ignorant on this.

DR. PAIK: Yes.

CHAIR BALLINGER: What is HDR 11-2?

MEMBER REMPE: That -- real quick. Remember how I complain sometimes about they had the sub-note little model in MAAP and they don't in MELCOR? And so in one of their comments in the report was that they got better matching with the HDR tests in Germany because of the --

CHAIR BALLINGER: Okay, so it was a test

in Germany.

MEMBER REMPE: Yes.

CHAIR BALLINGER: All right.

(Simultaneous speaking.)

CHAIR BALLINGER: Okay.

DR. PAIK: This is a non-pressurized containment and HDR is the German superheated reactor. But the reactor had an instability. So after three months of starting, they shut it down and they used it for the experimental, the starting for the thing.

11-2 was --

CHAIR BALLINGER: I'm trying to make the connection. Okay.

DR. PAIK: Next. When we're looking at the APR1400 containment nodalization, we have a fine nodalization near the source location to capture local high gas concentrations and also reactor cavity and steam generator compartment.

And we have a little bit of coarse nodalization away and above the source location, such as the reactor dome where the gas is expected to mix well.

So we like to have sufficient nodalization to represent a density-driven global natural circulation flow path.

So if we look at some of the natural tendency toward the mixing in the large dry containment like APR1400, this formation of a high concentration stratified layer away from the source is very unlikely in a large dry containment because when the gas is coming on with the hydrogen kind of low density, then it will entrain the air, surrounding air so it goes up. So the gas is deleted rapidly, due to all these entrainment.

And the concentration of the stratified layer increases only after that layer boundary reaches the source. So essentially, there is no more entrainment. Whatever the hydrogen is going to the north kind of things.

So the mixing within the stratified layer is enhanced due to the density-driven circulation plus the heat transfer to all these containment structures, especially if steam is released with gas.

So in APR1400 the source location is relatively low and the gas concentration effect due to stratification is kind of moderate.

So especially in APR1400, unlike U.S. larger dry PWR containment, the APR1400 has the number of PARS and also igniters. So the way APR1400, the procedure to selecting the PAR location is they are

looking at the scenario selection. So they look at all kinds of different scenarios. And they're looking at analyzing the hydrogen mixing and distribution with no PARs and see where it is the high concentration.

So for example, they look at the pre-analysis using the lumped code. So auditors, they are looking at the MAAP and they use a contain to try to see the containment, the hydrogen stratification.

They also did a supplementary study using a three-dimensional code in a certain location at how the hydrogen behaves and looking at the characteristics of hydrogen generation and release. And then they initiate selected draft location and the capacity of PARs.

And then they repeat those kinds of analysis with PARs and try to assess the impact of the PARs and looking at the hydrogen concentration.

So they are looking at the comprehensive analysis using the lumped code with the PARs and they also look at whatever the condition. They look at the analysis of the flame acceleration, when the flame acceleration with the core or not. And also they are looking at the potential for the DDT for all these different kinds of conditions.

So the final decision on the PARs location

is they are looking at the corrective action according to the engineering judgment. I mean they are looking at all these based on analysis but they still have engineering judgment because the lumped parameter code sometimes may not give a detail. So they are looking at the, let's say, source location compared to where could be the best location to put the PAR.

So if you look at the distribution of the PARs inside of APR1400 containment, one of the fundamental principles that KHNP employed is first they place where the large convective flow is existing. And also the recombination capacity is proportional to the free volume of the compartment. So essentially like near the crane and dome they have like eight different PARs located on those things.

And also they install above hydrogen release point because it doesn't mean a whole lot if you put the PAR in below the source.

And another thing is they are also looking at the easy to implement and also easy to maintain and control. Because during the outage, you want to check where the PARs, whether any kind of deposits or those kind of things.

So those locations are looked at and also they are looking at locations protected from the

external damage. For example, they don't want to put it in where there can be a large LOCA or some kind of a blowdown effect kind of condition.

And because this is a largely open, dry containment and we believe because of the PAR location in the distribution, we can say it is very unlikely to develop a stratified layer rich in flammable gas in APR1400.

MEMBER REMPE: Thank you.

So earlier today I mentioned that there is a ten-order magnitude difference in your equipment survivability assessment for the values in gray that you have in your -- the severe accident report, as well as your DCD versus what the staff has in their SE. I tend to believe, because there is some other work that I did, that your numbers are correct. But do you have any changes that you want to report here?

DR. PAIK: For the ES, the temperature curve?

MEMBER REMPE: No, the radiation levels.

DR. PAIK: Oh, radiation level?

MEMBER REMPE: Yes, you had something that was like ten to the fifth -- 4.4 times ten to the fifth gray and they have ten to the minus fifth gray. And I think your values are probably correct, unless

you say here that you're off.

DR. PAIK: I cannot comment whether it is correct or not.

MEMBER REMPE: It needs to be resolved.

DR. PAIK: But the way we did it is we run the MAAP and calculate the source term for the efficient follow-up and then we run the dose code to evaluate the dose for the particular equipment.

MEMBER REMPE: Okay.

MS. GRADY: Anne-Marie Grady from the Containment Systems Branch.

And Dr. Rempe, you are absolutely correct that in the SE there is a table that lists the radiation dose in rads and it's ten to the plus seventh. And then in the same SE in at least two other places, it should be ten to the plus fifth and instead it says ten to the minus fifth. And that will be corrected.

MEMBER REMPE: Sounds good. Thank you. It would be a very strict requirement if you didn't.

CHAIR BALLINGER: Well we get ten to the minus fifth on an airplane trip -- I think, and then some.

I think -- is the staff ready to go?

MR. STECKEL: Good afternoon. This is Jim

Steckel and we are going to complete our review -- our presentation on our review of 19 -- Chapter 19 with our presentation on Section 19.2, Severe Accident Evaluation.

And Hanh, would you lead us in our discussion?

MR. PHAN: Thank you, Jim. Good afternoon, again.

Before getting into the questions, I'd like to clarify some items relevant to the applicant's presentation this morning and the values the staff reported in our safety evaluation.

Per SRP page 20, which states that the staff will review quantitative results surpassed mean core damage frequency and mean weakness frequency. And that's why in our report the staff evaluated and discussed the mean frequencies. In the applicant's presentation this morning, they report the point estimate from the PRA quantification and confirms that all the numbers are consistent with the DCD Revision 2 dated March 2018. And that clarifies one of the members' concerns regarding the frequency.

Okay, by the end of Phase 2, the staff identified five open items on severe accident evaluations. Next page, please. As of today, all

five are resolved. Only one confirmatory item, 19.2-53 is being tracked as a confirmatory item.

In this presentation, the staff will discuss the closing of those open items, including the MELCOR confirmatory analysis and the steam explosions.

All those that you have seen the slides, however, we received the talking points from Dr. Rempe. The staff is prepared to answer your questions.

With that, I would like to turn it over to Mr. Jason Schaperow.

MR. SCHAPEROW: Thank you. I'm Jason Schaperow of the PRA and Severe Accidents Branch. Next slide, please.

I will talk about the MELCOR confirmatory analysis. Just by way of review, the objective of these analyses that the staff does, these are independent calculations the staff does with its own MELCOR input deck that we developed. The objective is to confirm the applicant's use of MAAP for PRA and severe accident analysis.

The approach we use for these analysis is to perform our in-house independent analysis for select scenarios using MELCOR and then we compare them with the MAAP results.

Now the next bullet is something that we discussed at the previous ACRS meeting, which I think was roughly a year ago. We talked about our Phase 2 conclusion which was for the at-power accidents, we did see some differences between MELCOR and MAAP. We concluded that those differences were not likely to impact how the applicant used MAAP. In particular, their assessment of the LRF, their assessment of SAMDA, as well as the quantification of the containment event tree.

For shutdown accidents, as a result of our confirmatory analysis we did, we did identify -- we and the applicant identified some issues with the MAAP calculations.

Next slide, please.

So after we met with you, which was Phase 3, the applicant was working on their MAAP analysis for shutdown accidents to identify the issues that both we and they had uncovered. And they redid the analysis and they updated the FSAR. In particular, they addressed the modeling of the RCS and the containment configuration. They also addressed the code bug.

And the staff concludes that the revised MAAP analysis does address the issues that were

raised.

MEMBER REMPE: So there were two MELCOR models generated in this part of this design certification, right? There was one the ERI did for the hydrogen evaluations based on the MAAP input deck, right? And then you had the one that you did in-house with the MELCOR model. Is that a true statement that there really are two MELCOR models for APR1400?

MR. SCHAPEROW: So the staff relied on ERI, an agency contractor, to prepare the input deck. It's a very time-consuming effort and we did rely on them to do that for us.

We used Revision 4 of that deck, if I remember correctly, for this work.

MEMBER REMPE: So it's one deck. Okay.

MR. SCHAPEROW: I think the deck was subsequently revised a bit as a result of the hydrogen work. And they may have relied on a slightly later revision. That could be Revision 5 or 6 but it's the same deck.

MEMBER REMPE: It's one deck.

MR. SCHAPEROW: Yes.

MEMBER REMPE: Okay because I thought --

MR. SCHAPEROW: The calculations for this effort here, the non-hydrogen work were done in-house.

I did them.

The calculations for the hydrogen review, which is the 10 CFR 44 -- I don't remember -- that was done by the contractor is my understanding.

MEMBER REMPE: Okay because, again, this has been a while since we first looked at this and I know that they -- I looked back through that ERI report and they had said they basically used what KHNP gave them for a MAAP input deck to generate the plant deck.

Whereas, I seem to recall that you had said you started with another similar CE plant, basically. And that's why I guess I got confused that it was the same deck.

MR. SCHAPEROW: Yes, the deck that was used for my analysis as well as the hydrogen analysis MELCOR deck, the containment nodalization looks a lot like the MAAP containment nodalization. We borrowed heavily on the containment nodalization from MAAP.

MEMBER REMPE: Okay but the RCS was actually always based on another existing plan initially and then you fine-tuned it with the MAAP deck?

MR. SCHAPEROW: My recollection is that was built from scratch, based on the geometry that was

provided before the DCD came in. Our ERI, our NRC contractor built that deck from scratch. And again, it was a time-consuming effort.

MEMBER REMPE: Okay.

MR. SCHULTZ: Jason, the work that was done in Phase 4 with regard to the MAAP evaluation, a lot of more realistic modeling and the code bug, they affected the shutdown accident work only. It didn't help with the differences between MELCOR and MAAP for the at-power event analysis?

MR. SCHAPEROW: The code bug -- according to the applicant, the code but that was uncovered in this shutdown analysis was actually corrected for the at-power analysis. So I never saw those with the bug in it. They fixed it before they sent their application in. But it wasn't fixed in the MAAP analysis for shutdown accidents. So I saw basically the same bug they had found previously.

MR. SCHULTZ: Okay. So the comment that you have on the previous slide still holds. There wasn't anything that was done here that would have affected the differences between MELCOR and MAAP --

MR. SCHAPEROW: No.

MR. SCHULTZ: -- for at-power but you identified the differences weren't considered to be

significant.

MR. SCHAPEROW: No, the differences were not significant.

MR. SCHULTZ: Okay, thank you.

MR. WAGAGE: My name is Hanry Wagage. When we came in front of the Commission last time, we had open items in steam explosions because review was ongoing. Applicant was responding to our RAIs.

One of the open items was in in-vessel steam explosion. The concern was how the applicant determined what size of the cross-sectional area to be used for the calculation because TEXAS-V is one-dimensional code, that cross-sectional area is determined by the use of.

The applicant showed to our satisfaction that cross-sectional area used was the optimal value that gives the highest steam explosion pressures.

In ex-vessel steam explosions we had the same issue. Again, I did it the same way. On ex-vessel steam explosion, the staff found that the way the applicant calculated the impulse loading on the cavity, once the steam explosion loading is calculated in the calculation region, it has to be transmitted to the cavity, away from the calculation region.

Staff found that the equation used by the

applicant had errors. Applicant accepted the staff concern and correct the equation and calculated the correct loading on the walls.

Structural evaluation changed because the way the applicant used the cross-sectional area for the calculation region and also that how the impulse loading was calculated in waters.

The staff reviewed the applicant's structural evaluation and accepted it.

The last item is on in-vessel retention and external reactor vessel cooling operation, how this impacts steam explosion loading. Staff presented this as in-vessel retention and external reactor vessel cooling as a SAMD item. Our concern here was that not how well the system would work because this has to be -- staff will review it during COL phase. Our concern here was that if the system is used how the design would react -- respond to that.

So that's why we were asking the applicant to ensure that in-vessel retention and external reactor cooling has to be considered under SAMD if the operator decides to use this system. But for APR1400 design, they did not take any credit for the PRA.

MEMBER REMPE: So if I come in and want to build this plant and I try and say okay, I'm going to

assume a certain amount of melt those ex-vessels for my analysis of the steam explosion or whatever, can the staff two years down the line say that particular amount of mass is adequate but then, if another applicant comes in down the road, will the staff say oh, no, I think you ought to have twice that much melt?

I mean there's a lot of how you do this analysis and it could change. And that's why I keep saying that's something to put off to the COL applicant that could lead to regulatory uncertainty. And I just am curious about how this is going to progress.

But I guess --

MR. WAGAGE: It's not the amount of mass of melt coming down into the cavity. The concern was that if the in-vessel retention system works and the cavity is flooded to a much higher level, then that affects steam explosion loading because there is a larger depth of melt falling into the cavity.

MEMBER REMPE: Wouldn't the mass of material that goes into the water also affect the loading onto the structures because of how much the steam explosion is?

MR. WAGAGE: I would expect that if in-

vessel retention system had been working and the cavity is flooded, then there will be lesser mass coming then without the system because they are cooling from outside. In that case, I would expect lesser amount of mass than what the applicant assumed for this calculation.

MEMBER REMPE: So there is a specific -- I mean are the analysis parameters well-defined of what they need to evaluate? Do you understand what I'm trying to get to, for what scenario and how much melt comes down, I mean a global rupture or just the metal going out? I mean is the scenario well-defined that the regulator can't change the parameters on what has to be evaluated for that steam explosion?

Is the question clear?

MR. KIM: Yes, this is Byung Jo Kim from KEPCO E&C.

According to Dr. Rempe's concerns, we have to think about the two aspects. One is the condition in the ex-vessel. The other one is condition in the in-vessel at the time of reactor vessel failure under the IVR/ERVC operation. Okay?

MEMBER REMPE: Uh-huh.

MR. KIM: So if the operator chooses the cavity from the system, according to the SAMG, so

excessive water does not touch the reactor vessel lower head. But in case of the IVR/ERVC, when the water goes up to the higher, such as the holdup tower.

So these two different conditions may create different pressure from the ex-vessel steam explosion because in case of the CFS main jet clearly has some certain free fall distance and then the jet contact to the water. But in case of IVR/EVRC, the main jet contact the water at very close to the failure location.

And in case when you think of the in-vessel condition, the melt mass or the melt energy such as the support temperature property should be different from the CFS condition and the IVR/EVRC condition.

So when we got the Shin-Kori unit operating license from the Korean regulatory, the KINS, comprehensively leave you our safety analysis report going to IVR/EVRC case.

In conclusion, under the very conservative approaches, such as this, we stand up the initial condition of this vessel steam explosion from the hypothesis accident sequences to create the right pressure in the course of in-vessel and ex-vessel condition.

So under this very conservative and hypothetical conditions, the steam explosion load, of course, it goes to increase the certain level compared to the CFS condition but the cavity wall structure and the reactor vessel supporter and penetrations along the main steam line weren't any penetration, was not affected from the steam explosion under the failure of IVR-ERVC situation.

MEMBER REMPE: So what I think you're telling me is that -- and I'm just in the review but the DCD document clearly says what the load will be for evaluating, which must come from sort of severe accident where you assume a certain failure size and how much melt, whether it's acetic or metal goes out for these evaluations.

And although I didn't see it clearly in the staff SE, you have accepted those conditions for the analysis of this.

MR. KIM: Yes, during the RAI discussion, I provide for the Reference Branch, on RSS review, such as the Shin-Kori Unit 3 and he reviewed the Reference Branch data and caused the very highest similarity of APR1400 and Shin-Kori Unit 3.

Even though we did not carry out IVR-ERVC performance in the safety analysis we can say that if

operator chooses IVR-ERVC option and under whatever the reason in case failure the containment integrity and basement melt through does not care.

MEMBER REMPE: Okay, thank you.

MR. KIM: Okay. You are welcome.

MR. WAGAGE: Our reason for not mentioning that calculation provided for IVR operation by applicant was that there were some changes made during our review that we did not believe that IVR operation was -- that was outside our scope.

MEMBER REMPE: Well, okay. It sounds like they tried to fix the parameters for the analysis but you said no, I don't want to even say that those are the appropriate parameters for the analysis because it's outside the scope.

MR. WAGAGE: No, I just mean that example was what crosses an area for the calculation to be used. That's one. The second one was that how the impact is to be calculated on the cavity wall.

So those details we did not review that part of the calculation how that -- our conclusion did not depend on that because we did not get there because of these concerns for the in-vessel, for the other steam explosion loading that without using the IVR when it will be really from the bottom of the

vessel. And for that, the applicant provided results and we accepted those results.

But for when the applicant uses IVR-ERVC, then the failure would be in a different location, side failure. So we did not review that because the reason that these corrections have been made in that calculation is beyond our scope of review.

MEMBER REMPE: So again, if they assume side failure, they typically assume just the metal layer comes out, not ceramic like if you had a bottom failure. But right now it's kind of open field because you've said I don't want to review that because if they use the ERVC, it's something we'll review with the COL applicant.

So that kind of leaves it open, which I guess I'm kind of hearing a disconnect because I thought the gentleman from KHNP said no, we gave them what we believe are the appropriate parameters for both cases.

So again, I'm just kind of wondering if there's some regulatory uncertainty. And where there's regulatory uncertainty, there's a chance where an applicant might try and do something less safe or they might get stuck having to do something they think is overwhelming and not necessary. That's why I kind

of keep raising this, that it's a pretty big gap.

If there parameters for the analysis were well-defined, I wouldn't be as concerned is where I'm kind of going with it for both cases.

MR. WAGAGE: When you say parameters for the analysis, what does it mean? Does it mean that loading on the cavity walls or steam explosion pressures? So we did not review that part because these two corrections had to be made. That last item in the first two bullets, cross-section area has to be calculated, corrected. And calculation of impulse has to be corrected.

After these two corrections are done, we don't know how that analysis results presented would change. We did not review the analysis for external reactor vessel.

MEMBER REMPE: Okay, I think I understand the situation. Thank you.

MR. WAGAGE: You're welcome.

Based on its review, we found that the APR1400 design provides features for the prevention and mitigation of severe accidents and is in compliance with 10 CFR 52.47(a)(8) and (23) and in conformance with SECY-90-016 and SECY-93-087, and SRP 19.0.

MR. PHAN: That is the end of today's presentation. So if you have any questions to the staff on any topic, thank you.

CHAIR BALLINGER: Thank you. I think now we need to find out if there are public comments.

Is there anybody in the room that would like to make a comment or statement?

Not hearing a stampede to the microphone, we'll see if the -- get the public line open. You think it's open? It just cracked? Oh, okay I wanted to make sure it was open. The line is open.

Are there any people out on the public line that would like to make a comment?

Is there anybody out on the public line? If you're there, please identify yourself so that we can tell that for sure the public line is active.

Having zero for both requests, we can close the public line.

In that case, let's see where -- we should go around the table to see members have comments.

Joy?

MEMBER REMPE: No, I don't have any additional comments.

CHAIR BALLINGER: Walt?

MEMBER KIRCHNER: No. Thanks to the

presenters. Thank you.

CHAIR BALLINGER: Jose?

MEMBER MARCH-LEUBA: No, no additional comments.

CHAIR BALLINGER: Pete?

MEMBER RICCARDELLA: No comments.

CHAIR BALLINGER: Wow. Matt?

MEMBER SUNSERI: I don't have any additional comments. Thanks for the thorough presentations today, both the staff and applicant.

CHAIR BALLINGER: Dick?

MEMBER SKILLMAN: No comments. Thank you to the staff and the applicant. Thank you.

CHAIR BALLINGER: Steve?

MR. SCHULTZ: I would echo the thanks to the staff and to the applicant. The presentations clearly demonstrate the amount of work that's gone into this project over the years.

And my only comment would be some of what we've discussed today that I think what's important, based on the experience that we've had in the industry with regard to PRA development and application over the last several decades, that what seems very important is this next phase, which also represents a lot of work, and that's the challenge that will be

given to the COL applicant with regard to the next stage of implementation, in terms of balance of plant design and then the movement into operation.

And it's very clear from the experience in the industry just how important the PRA, the probabilistic risk assessment and severe accident analysis, is in the COL applicant's design and then following into plant operation in areas that we've discussed today, reliability assurance, quality assurance, training, both in terms of operations and engineering, engineering modification, operational design and decisionmaking.

I think the designer, as well as the staff will want to be sure that that transition is very smooth and that the process and the documentation that will allow that to happen effectively is available. So that's the next stage and I know it's a big challenge to everyone but as we move from the design to that aspect of application, it's going to be very important.

And I'm very encouraged to hear how much work has gone into the reliability assurance program as an example of how that documentation has been developed. That should make it tractable. And clearly the documentation that has gone into all of

the activities, both on the side of the designer and applicant, as well as the NRC is going to be very helpful here.

Thank you.

CHAIR BALLINGER: And I'd like to thank you for the rest of the committee. I feel like over this process we're at the end, almost -- almost. I feel like I've heard from -- we have heard from not only everybody on the staff in the building but everybody from KHNP. And I feel like I have spent more time with you guys than I have with my wife.

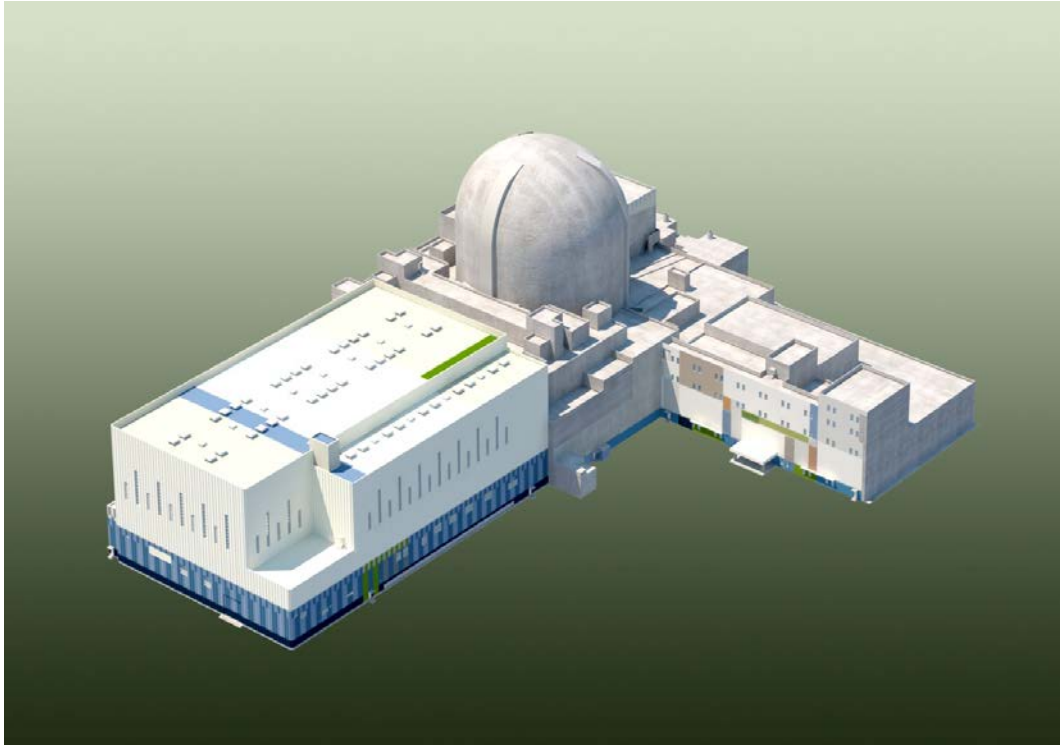
So it's an enormous amount of work. I'm being a little facetious but you guys have done a tremendous -- everybody, KHNP and the staff, have done a tremendous amount of work in actually a record short time, I think.

So thank you very, very much for the committee and we are adjourned.

(Whereupon, the above-entitled matter went off the record at 4:58 p.m.)

NON-PROPRIETARY

APR1400 DCA
Chapter 17: Quality Assurance and Reliability Assurance



KEPCO/KHNP

June 5, 2018

ACRS SC Meeting (June 5, 2018)

Contents

- **Overview of Chapter 17: Quality Assurance Program**
  - Basic Policy of QA
  - Approval, Implementation, and Improvement
  - Organization and Oversight for Implementation
- **Reliability Assurance Program**
  - RAP SSCs Update
  - Summary of Open Items
  - Current Status
- **Acronyms**

Chapter 17: Quality Assurance Program

❖ Basic Policy of QA

- Top management policy to enforce safety culture
- Training and qualification
- Organizational freedom and independence from cost & schedule
- Holding work stop authority
- Direct reporting to top management and applying feedback

Chapter 17: Quality Assurance Program

❖ Approval, Implementation, and Improvement

- QA program was established to comply with the following requirements.
  1. 10CFR50 Appendix B, QA Criteria
  2. 10CFR21, Reporting of Defects and Non-Compliance
  3. ASME NQA-1-2008 and 1a-2009 Addenda, QA Requirements
- QA program was accepted by the NRC.
  1. May 2, 2016: Submittal of Revision 5
  2. October 6, 2016: Approval of Revision 5
  3. March 13, 2018: Submittal of Approved Revision 2 based on Revision 5
  4. May 8, 2018: SER with no open items
- QA program is implemented and improved with the corrective actions.
  1. Reviewing and approving every safety-related document
  2. Annual QA auditing for all organizations

Chapter 17: Quality Assurance Program

❖ Organization and Oversight for Implementation

- KHNP worked with the pre-qualified design suppliers:
  1. Architectural Engineering and System Design: KEPCO E&C AE, SD
  2. Fuel and Core Design: KEPCO NF
  3. Component Design: DOOSAN
- KHNP performed these quality functions:
  1. Evaluating, selecting, and qualifying the suppliers before contracts
  2. Reviewing and approving design products prepared by the suppliers
  3. Controlling RAI responses and related document changes
  4. Design Review Committee for design change and interface
  5. Implementing annual QA audits and checking corrective actions

17. 4 Reliability Assurance Program

PRA Update and Review

- **Revised RAP procedures**
  - DC-DG-03-09, Implementation of the Reliability Assurance Program (RAP)
  - DC-DG-03-10, Expert Panel Roles and Responsibilities
  - DC-DG-03-11, Risk Significance Determination of RAP SSCs

- **Expert Panel reviewed final PRA model results and updated the RAP list**
  - Reviewed results for all available full power and shutdown, internal and external events PRAs; CDF and LRF
  - Utilized criteria recommended by RG 1.200 to identify candidates for risk significant classification

Expert Panel

- **Meetings by Expert Panel in 2017**
  - First meeting was a test run with the revised procedures only
  - Second meeting reviewed the updated PRA results
    - Weeklong meeting
    - Full RAP scope was reviewed
    - Accepted or rejected all PRA recommendations
  - Electronic sessions finalized RAP list for DCD certification

Procedure Number	DC-DG-03-10	Revision Number	03
Title	Expert Panel Roles and Responsibilities		

Observations

- **Process**
  - Rigorous discussion of many topics
  - Conservative approach when little or no PRA information was available
  - High level of experience among members, from diverse backgrounds; active & insightful participation by all
  - Process is consistent with observations of current Maintenance Rule programs for several US plants

Procedure Number	DC-DG-03-11	Revision Number	03
Title	Risk Significance Determination of RAP SSCs		

Summary of Open Items from Phase 3

❖ List of Open Items

No.	Related RAI	Topic	ADAMS Accession #
1	RAI 316-8305 (Q 17.04-1,2)	Selection of APR1400 DC RAP SSCs	ML15321A167

❖ Open item has been resolved and is in confirmatory status

Summary of Open Items from Phase 3

❖ Selection of APR1400 DC RAP SSCs

- Related RAIs : RAI 316-8305 (Q 17.04-1)
- Description of issue
  - Staff requested RAP program details that follow the guidance in SRP Chapter 17.4, Revision 1 or an alternative to the SRP acceptance criteria.
- Resolution
  - KHNP provided a revised RAP program description, including the RAP procedure revisions for the design certification.
  - APR1400 RAP SSC list was revised using updated PRA results and insights, in accordance with the revised RAP program and procedures.

Current Status

❖ **Chapter 17 has been completed.**

- KHNP continues to monitor Chapter 17 to assure any conforming changes are addressed.
- 1 open item that was identified in Phases 2 and 3 has been resolved following extensive discussions with the staff.

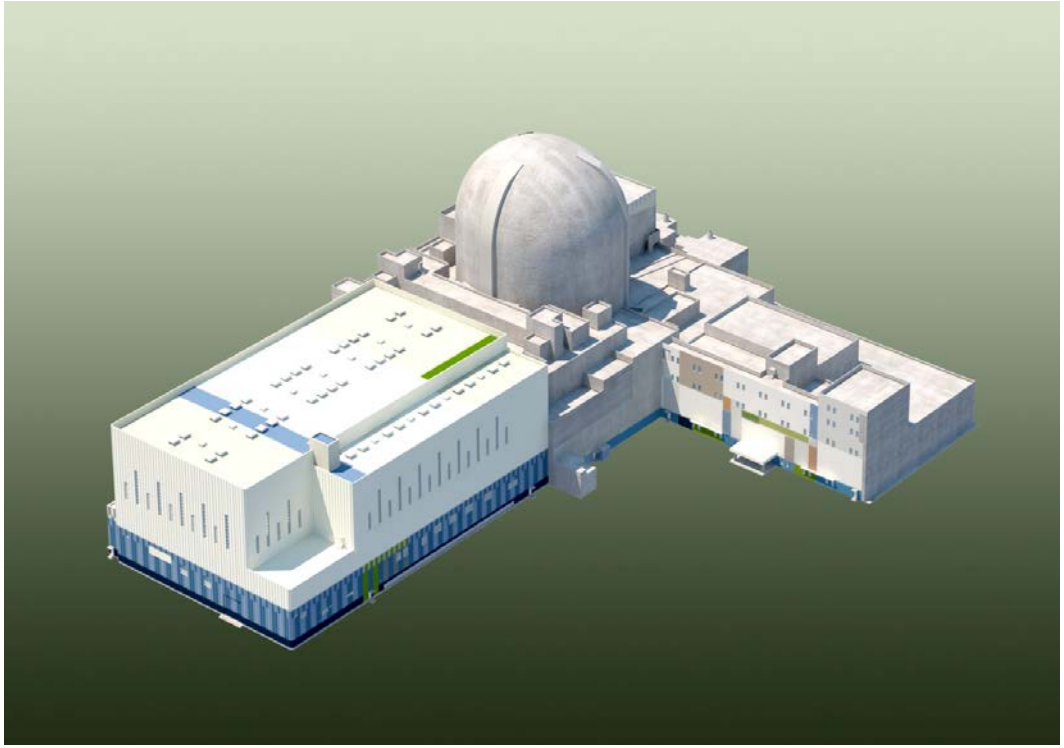
❖ **Changes in Chapter 17.4, as reviewed and marked-up in response to the RAIs, will be incorporated into the DCD Rev. 3.**

Acronyms

- QA: Quality Assurance
- RAP: Reliability Assurance Program
- SSC: Systems, Structures and Components

NON-PROPRIETARY

APR1400 DCA
Chapter 19.1: Probabilistic Risk Assessment



KEPCO/KHNP

June 5, 2018

ACRS SC Meeting (June 5, 2018)

Contents

Section	Title	Presenter
-	Overview of Chapter 19.1 and PRA Update History	Young In
19.1	Probabilistic Risk Assessment	-
19.1.4	Safety Insights from the Internal Events PRA for Operations at Power	-
19.1.4.1	Level 1 Internal Events PRA for Operations at Power	Greg Rozga
19.1.4.2	Level 2 Internal Events PRA for Operations at Power	Jeff Leary
19.1.5	Safety Insights from the External Events PRA for Operations at Power	-
19.1.5.1	Seismic Risk Evaluation	Dong-Won Lee
19.1.5.2	Internal Fire Risk Evaluation	Greg Rozga
19.1.5.3	Internal Flooding Risk Evaluation	Ray Dremel
19.1.6	Safety Insights from the PRA for Other Modes of Operation	-
19.1.6.1	Level 1 Internal Events PRA for Low Power and Shutdown Operations	Jaegab Kim
19.1.6.2	Level 2 Internal Events PRA for Low Power and Shutdown Operations	Jeff Leary
19.1.6.3	Internal Fire PRA for Low Power and Shutdown Operations	Greg Rozga
-	PRA Update (2017)	Greg Rozga
-	Summary of RAIs	Young In

Overview of Chapter 19.1

❖ List of Submitted Documents for Chapter 19.1

Document No.	Title	Revision	Type
APR1400-K-X-FS-14002 -P & NP	APR1400 Design Control Document Tier 2: Chapter 19 Probabilistic Risk Assessment and Severe Accident Evaluation	2	DCD
APR1400-K-X-IT-14001 -P & NP	APR1400 Design Control Document Tier 1	2	DCD

PRA Update History

- PRA Revision 0: At-power internal events Level 1 and 2, at-power internal fire Level 1 and 2, internal flooding Level 1 and 2, at-power PRA-based SMA, other external events – September 2013 (information cutoff date of December 2012)
- PRA Revision 0 – LPSD internal events Level 2, internal fire level 1 and 2, internal flooding Level 1 and 2 – September 2014 (information cutoff date of December 2013)
- PRA Revision 0A: Conversion to CAFTA software
- PRA Revision 1: Update to all PRA models, based on design changes, data update, and RAI responses – September 2017 (information cutoff date of November 2016)
  - PRA-based SMAs completed in March 2018
  - All PRA documentation completed to support DCD Rev. 3

19.1.4.1 Internal Event Level 1 PRA

Initiating Event Analysis

- **Identification of Potential Initiating Events**
  - Generic industry information sources
    - NUREG/CR-6928, 5750, 3485, GL 88-20, NUREG-1335, EPRI NP-2230
  - Information from similar plants
  - Plant-specific operating experience: N/A
  - Systematic review of the APR1400 design - a high level FMEA
- **Grouping of Initiating Events**
  - Impacts of initiating events on core protection functions and plant responses
  - Group of initiators expected to have a common core damage accident progression and success criteria
  - Comparison with Generic Source such as NUREG/CR-5750
- **Calculation of Initiating Event Frequencies**
  - Use generic industry data in NUREG/CR-6928 and Initiating Event Data Sheets - Update 2010
  - Criticality factor 0.95 assumed

Support System Initiating Events

- The support system initiating events (SSIEs) fault tree models were created to calculate their initiating event frequencies in the APR1400 PRA model (Rev. 0)
  - PLOCCW, TLOCCW, PLOESW, TLOESW, LOIA
  - The SSIE frequency was assigned as the more conservative value after comparing generic data source with the calculated frequency by using plant specific fault tree model.
- 2017 PRA updated model (Rev. 1) SSIE frequencies use the generic data sources:
  - The SSIE frequencies are representative based on comparisons between the design-specific frequencies and the generic frequencies
  - Portions of current ESW system are based on the conceptual design information (CDI), which can affect ESW SSIEs frequencies
  - Use of generic data sources for SSIEs for APR1400 DC PRA is reasonable, since risk applications for DC phase (e.g., RAP, SAMDA) are not impacted as in some risk applications for operational phase (e.g., online maintenance, MSPI, etc.)

Accident Sequence Analysis

- **Key Safety Functions**
  - Reactivity, RCS Inventory, RCS Pressure, Decay Heat Removal, Containment Heat Removal
- **Develop Event Trees**
  - Define key functional requirements
    - To reach a safe, stable state and prevent core damage, as well as identify the systems and operator actions for accident mitigation
  - Define accident sequences
    - In a manner consistent with plant system design, operating procedures, plant response, etc.
    - Apply T/H analyses to determine the accident progression parameters
  - Identify impacts of initiators to mitigating systems (dependency)
    - Success/failure of preceding systems, functions, human actions
    - System alignments, time-phased, phenomenological conditions
  - Develop Fault Trees: Small Event Tree/Large Fault Tree, Fault Trees are linked to Event Trees

Success Criteria

- **Definition of Core Damage**
  - Consistent with SR SC-A2 of ASME/ANS PRA Standard
  - Peak node temperature exceeds
    - 1204°C (2200°F) for a code with detailed core modeling (RELAP)

      or
    - 982°C (1800°F) for a code with simplified core modeling (MAAP)
- **Based on Best Estimate Analyses**
  - Utilize MAAP 4.0.8 and RELAP5/mod3
  - Review DCD Chapter 15 for consistency

Data and CCF Analysis

- **Data Source used in internal level 1 full power PRA**
  - **Component Unreliability Data**
    - **NUREG/CR-6928 Online Update**
    - **Vendor specific data for POSRV**
    - **Alternate industry sources**
  - **Component Unavailability Data**
    - **NUREG/CR-6928 Online Update**
    - **NUREG/CR-5500, Vol. 2, 3, 10, and 11**
    - **Surrogate data when no generic data available**
  - **Common Cause Failure Data**
    - **NUREG/CR-5497 Online Update**
  - **Special Events Data**
    - **NUREG/CR-6890 for LOOP non-recovery probabilities**

Human Reliability Analysis (1/2)

- **The HRA model**
  - **Type A: Pre-initiating event human interactions (errors that can occur during test and maintenance)**
    - **~ 60 pre-initiators are modeled**
    - **Based on the test and maintenance procedures from the reference plants**
    - **Will need to be verified when the detailed test and maintenance procedures become available during COL stage**
  - **Type B: Initiating event related human interaction (if not completed correctly may cause an initiating event)**
    - **Not explicitly modeled**
    - **Assumed implicit in the IEFs obtained from Operating Experience**
  - **Type C: Post-initiating event human interaction (evaluated to determine the likelihood of error)**
    - **~ 70 operator actions are modeled**
    - **Dependencies among the operator actions were evaluated**

Human Reliability Analysis (2/2)

- **Dependency Analysis Methodology**
  - Re-quantify model with HRA valves set to a value near 1.0
  - Analyze the HRA combinations for dependency
  - Dependency Level Evaluation
    - The EPRI HRA Calculator provides the tools to perform the dependency analysis (used for 2017 PRA update for all models/all modes)
    - Dependency between L1 and L2 operator actions is captured

Quantification

- **Quantification Process**
  - Quantification performed using CAFTA and FTREX
    - 10^{-13} is applied as cutoff value for CDF quantification
  - Logical loop treatment
    - Circular logic in the supporting system were broken at DC batteries.
  - Cutset recovery process using recovery file
    - Offsite Power recoveries, HRA dependency rules and minimize cutsets

19.1.4.2 Internal Event Level 2 PRA

Level 2 PRA Methodology

- Plant Damage State, Containment Event Tree and Decomposition Event Tree (CET/DET) methodology is used.
- Model originally created in SAREX and utilized PDS bridge trees to capture all inter-system and intra-system dependencies.
- Computer Codes
  - MAAP 4.0.8 utilized for analyzing severe accident progression and source term release
  - CAFTA for developing Level 2 PRA model

Severe Accident Mitigation Features (1/2)

- **Containment Building**
  - Pre-stressed concrete containment with a steel liner plate, Large Dry Containment
  - Containment Net Free Volume : 3.1×10^6 ft^3
- **Reactor Cavity Design**
  - Minimize challenges posed by DCH, FCI, MCCI
  - Convoluted Flow Path to decrease the amount of ejected core debris that reaches the upper containment
  - Large cavity floor area for corium debris spreading and coolability

Severe Accident Mitigation Features (2/2)

- **Cavity Flooding System (CFS)**
  - Minimize or eliminate corium-concrete attack due to MCCI
  - Minimize or eliminate the generation of combustible gases due to MCCI
- **Hydrogen Mitigation System (HMS)**
  - HMS limits hydrogen concentration in containment, generated from a 100-percent fuel clad-coolant reaction less than 10 v/o
  - HMS consists of 30 PARs and 8 Igniters

Plant Damage States (PDSs)

- Level 1 event tree sequences are extended to include additional questions important for Level 2 PRA. For example:
  - Containment sprays? (if not asked for heat removal in Level 1)
  - Steam generator status? (if not asked in Level 1)
  - Bridge tree (i.e., Extended Level 1 ET or PDS ET) sequences are grouped into PDSs isolation?
  - Containment based on similarities in the accident progression by PDS grouping parameters (Bypass, Containment Isolation, LOCA or Transients, etc.)
- As a result of PDS binning, 108 PDSs were defined and quantified to capture all the Level 1 - Level 2 dependencies.

Containment Event Tree

- The various containment failure mode and the major severe accident phenomena are represented as top events of the CET. Detailed evaluation of phenomena for each top event of CET is treated in Decomposition Event Trees (DETs).
- The Level 2 CET considered following containment challenges.
  - Direct Bypass (SGTR and ISLOCA)
  - Containment Isolation System Failures
  - Induced SGTR during the severe accident
  - HMPE/DCH or Blowdown Forces (rocket mode failure)
  - Steam Explosions (In-vessel and Ex-vessel)
  - Hydrogen Phenomena (slow combustion/detonation)
  - Late Containment Over-Pressurization
  - Molten Core-Concrete Interaction, including Basemat melt-through

Source Term Evaluations

- CET end points are grouped into the source term categories (or release categories) based on similarity of release characteristics (magnitude and timing).
- Source term release calculations are performed using the MAAP 4.0.8 code.
- Definition of a Large Release is ≥ 2.5% of volatile/semi-volatile (Iodine, Cesium, Tellurium) fission products (NUREG/CR-6595).
- Definition of early is release occurs before effective evacuation of the surrounding public after the general emergency declaration.

19.1.5.1 PRA-based SMA for At-Power

19.1.6.5 PRA-based SMA for LPSD

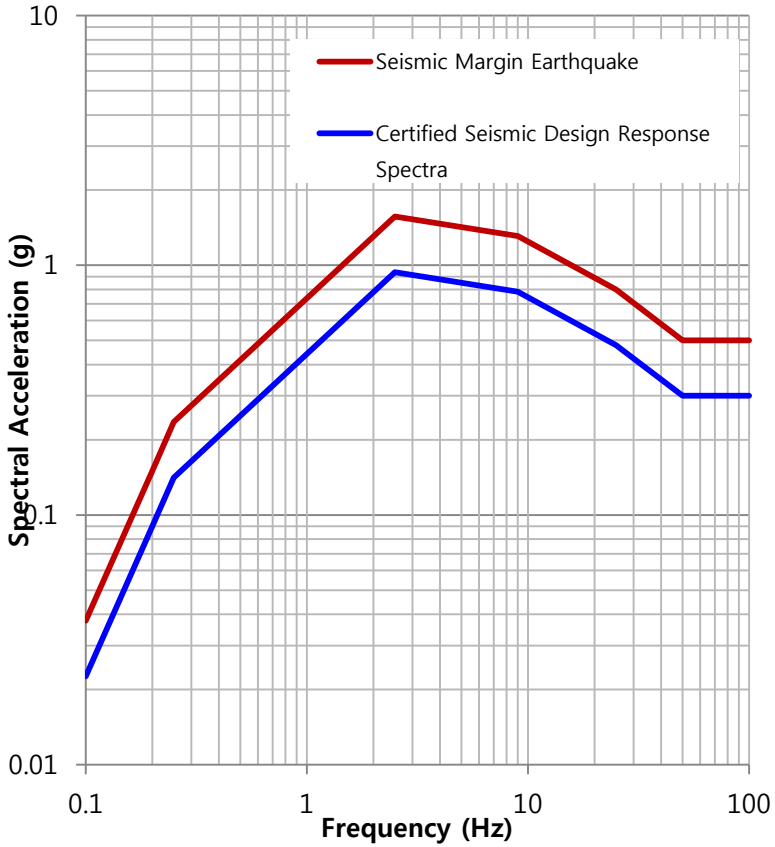
PRA-Based SMA Methodology

- Approach for risk evaluation of seismic events includes a PRA-based Seismic Margins Assessment (SMA)
- NUREG-0800, PRA and Severe Accident Evaluation for New Reactor, Rev02
- Staff Requirements Memorandum for SECY-93-087
- DC/COL-ISG-020
  - Provides guidance for implementation of PRA-based SMA
  - MIN-MAX Method
    - Determine the HCLPF capacity of an Accident Sequence
  - Systems modeling enhanced over original SMA guidance
    - NUREG/CR-4334 and NUREG/CR-5076
    - NRC's IPEEE guidance (in NUREG-1407)

Seismic Input Motion

- **Certified seismic design response spectra**
  - **Spectral shape of Reg. Guide 1.60 enhanced in high frequency**
  - **anchored to a peak ground acceleration of 0.3g**
  - **defined at free field ground surface**

- **Seismic margin earthquake is equal to 1.67 times CSDRS**



Seismic Equipment List

- **Define Seismic Initiating and Consequential Events**
  - SSCs are related to the IEs
- **Determine Safety Functions Needed for Response**
  - Reactivity control
  - Reactor coolant system pressure control
  - Reactor coolant system inventory control
  - Decay heat removal
  - Containment isolation and integrity
- **Identify Systems Needed to Fulfill Safety Functions**
  - Based on internal events PRA
  - Powered by onsite emergency AC sources
- **Exceptions**
  - SSCs which are fed power by offsite power
  - Distribution systems which is known as the ruggedness such as piping, HVAC duct, cable trays

SSCs with design-specific HCLPF capacities

- **Building structures**
  - Reactor containment building
  - Concrete internal structure
  - Auxiliary building
  - Emergency Diesel Generator (EDG) building
  - Diesel Fuel Oil Tank (DFOT) room building
- **Reactor Coolant System (RCS) components**
  - Reactor vessel & support
  - Reactor internals
  - Control element drive mechanism
  - Steam generators
  - Pressurizer
  - Reactor coolant pumps
  - RCS piping

Approach for HCLPF capacity evaluation

- **Identification of critical failure modes**
  - **Reviewed APR1400 design-specific reports, calculations, and drawings**
  - **Identified potential failure modes by comparing design seismic demand to design capacity, i.e., design margin**
  - **Selected governing failure modes for HCLPF capacity evaluation**

- **Seismic demands**
  - **Used APR1400 design-specific seismic demands**
    - CSDRS applied at plant finished grade in the free field for 8 generic soil sites and a fixed base case

- **Static capacity equations**
  - **Used code capacities per ACI 349, ASME Section III Service Level D allowable, or EPRI NP-6041-SL Rev.1**

Approach for HCLPF capacity evaluation (cont'd)

- **Considered inelastic energy absorption capabilities associated with ductile failure modes**

- **Evaluation of HCLPF capacities of SSCs**
  - **CDFM method in EPRI NP-6041-SL Rev. 1 applied to demonstrate HCLPF**
  - **HCLPF is equal to or greater than Seismic Margin Earthquake, i.e., 1.67 times CSDRS**

HCLPF capacities of other SEL components

- Inherently ruggedness items (e.g. distribution systems, check valve, small instrumentation, etc.)
  - Assumed to be rugged
  - DC/COL-ISG-020 : To avoid that so called "super element" mask the seismic risk result
- HCLPF of Very Small LOCA is assumed 0.3g
  - Based on EPRI-3002000709, Option 2 (SSE Level)
  - "SPRA implementation guide", seismic induced breaks in one or a very few small impulse lines connected to the primary circuits cannot be precluded, given the large number of such lines and their unusual configurations"
- HCLPF of Offsite Power is assumed 0.09g – Risk Assessment of Operational Events Handbook External Events, R.1.01, USNRC
- The component of the same type in the system conservatively assumed to be 100% correlated

Seismic Logic Model

- Performed FMEA for SSCs in SEL
- SSCs in the SEL were modeled individually
  - Naming Rule : SEIS-(Equipment ID)-FAIL-HCLPF

    (e.g. SEIS-SIPP02-FAIL-0.5G)
- HEPs are modeled to figure out the important operator action or human error
- Random failure event from internal PRA model were adopted in the PRA-based SMA model

Seismic Logic Model

- **At-power Level 1 & 2 Initiating Event**

IE (Current)	Description	Consequence
S-DCD	Direct Core Damage	CD leads to CF
S-SBO	SBO	CD leads to Failure of CTMT Isolation
S-IC	Loss of All I&C	
S-MSLB	Main Steam Line Break	
S-ATWS	Anticipated Transient Without Scram	CD may leads to Failure of CTMT Isolation or Late over pressure failure
S-TLOCCW	Total LOCCW	
S-LLOCA	Large LOCA	
S-MLOCA	Medium LOCA	
S-SLOCA	Small LOCA	
S-LOOP	LOOP	

Accident Sequence Analysis

- **At-Power Plant Level HCLPF**
  - **1.67 times CSDRS (0.5g HCLPF) meet the SECY-93-087**

- **At-power Insights**
  - **Seismic Event**
    - Equipment related to secondary heat removal (MD-AFWP/ TD-AFWP)
    - Equipment related to long term containment heat removal (CS and SC system)
    - Equipment related to Essential Power (EDG, 4.16kV etc.)

Accident Sequence Analysis (Cont'd)

- **At-power Insights**
  - **Operator Action**
    - MSOPH-S-ASC-SLOCA : Operator action to Perform Aggressive Secondary Cooldown After SLOCA
    - SIOPH-S-HLI : Operator action to Initiate Hot Leg Injection
    - RCOPH-S-SDSE-FW : Operator action to Open POSRVs in Early Phase of F&B Operation
    - SIOPH-S-IRWSTCOOL : Operator action to Cool IRWST Water Using SCS Pump
  - **Random Failure**
    - DGDGR-C-DGC, DGDGR-D-DGD : Random failure of EDG

PRA-based SMA during LPSD

Contents	Descriptions	DCD Rev. 2
Methodology	Same as At-power	19.1.6.5.1
SEL	Based on At-power and add the equipment to consider the LPSD condition	19.1.6.5.2
Seismic Fragility	• Methodology is same as At-power • Additional equipment were done such as containment Hatch, Polar crane, etc.	19.1.6.5.5 19.1.6.5.6.2
Plant Operating Condition	• POS 3B to 12A considered • Plant condition is grouped into whether RCS is intact or not	19.1.6.5.3
Plant Response Analysis	Similar to At-power but LPSD conditions are considered	19.1.6.5.4
Plant Level HCLPF	• “min-max” method are used for sequence-level and plant level HCLPF • 1.67 times CSDRS (HCLPF 0.5g)	19.1.6.5.6.3
Insight	• Vulnerabilities by seismic event, • Important Random failure • Important operator action	19.1.6.5.7

Summary for PRA-Based SMA

- **Methodology : SECY-93-087 and DC/COL-ISG-020**
- **Plant Operation condition : Full power and LPSD (POS 3B to 12A)**
- **Seismic Fragility : CDFM in EPRI NP-6041-SL Rev.1**
  - **Structure and NSSS > 0.5g (Specific Calculation)**
  - **Other SSCs in SEL ≥ 0.5g (Assumption)**
- **Accident Sequence : Seismic PRA model used (ET/FT)**
  - **Plant level HCLPF is equal to 1.67 times CSDRS (0.5g) At-power and LPSD**
  - **Insights for At-power and LPSD**
    - Seismic Vulnerabilities
    - Important Operator Actions
    - Important Random Failure Event

19.1.5.2 Internal Fire PRA for At-Power

19.1.6.3 Internal Fire PRA for LPSD

Internal Fire PRA – Task 1

- **390 Physical Analysis Units (PAUs) identified**
  - **PAUs are based on Fire Areas with rated barriers**
  - **~279 PAUs in Auxiliary Building – highly compartmentalized**
  - **PAUs are mainly single rooms, but may contain 2 or more rooms**
  - **LPSD Model uses same PAUs**
    - **Removable barriers assumed only used during LPSD POS 8 (defueled), so no impact on LPSD CDF or LRF**
      - **Used for large equipment replacement which is rare**

Internal Fire PRA – Tasks 2 & 3

- At-power FPRA equipment list based on At-power internal events PRA, and all LPSD FPRA equipment list based on LPSD internal events PRA
  - At-power equipment identified for non-modeled spurious operations
  - Additional equipment for LPSD "LOCAs" were identified
- Cables for all internal events equipment are routed (based on reference plant)
- Number and type of penetrations for multi-compartment analysis (MCA) are from reference plant

Internal Fire PRA – Tasks 9 & 10

- **No Task 9 Detailed Circuit Failure Analysis**
  - All cables assumed to lead to worst case failure modes
    - Power cable – loss of function
    - Control cable – worst of failure to operate or spurious operation
    - Fiber-Optic Cable – loss of function/failure to operate (e.g., no spurious operation)
- **No Task 10 Circuit Failure Mode Likelihood Analysis**
  - All spurious operations assumed to occur with a conditional probability of 1.0
  - No credit taken for clearing of hot shorts
  - No credit taken for recovery of spuriously operated valves

Internal Fire PRA – Task 5 (1/2)

- At-power and LPSD FIRMs both based directly on their respective internal events models
  - Basic events added for spurious operations which were screened in the internal events model
- At-power Fires in each PAU assumed to result in either GTRN, LOCV, LODCA, LODCB, LOFW, LOOP, PLOCCW or SLOCA based on equipment damage
  - No other fire-induced initiators were identified
  - If no PRA equipment/cables are damaged it is assumed to result in at least a trip which is modeled as a general transient
  - All fire induced failures are assumed non-recoverable including OSP

Internal Fire PRA – Task 5 (2/2)

- LPSD FIRM screens POS 7, 8, 9 and 12b (same bases as LPSD internal events model)
- Unscreened LPSD PAU fires assumed to result in an event considered in the LPSD model based on equipment damage in the scenario AND the POS

Internal Fire PRA – Task 6

- At-Power Ignition Frequencies based on NUREG/CR-6850 (and Supplement 1) counting and apportioning methodology and EPRI 1016735, "Fire PRA Methods Enhancements: Additions, Clarifications, and Refinements to EPRI 1011989" for frequencies
- LPSD Ignition Frequencies from NUREG/CR-7114

Internal Fire PRA – Task 12

- All current HFEs calculated using detailed HRA methodologies within the HRA Calculator
- All fire HFEs based on analogous internal events HFEs with fire impacts suggested in NUREG-1921 (e.g., increase in T_{delay}, higher stress, etc.)
- All fires assumed to at least partially impact one division of instrumentation
- Dependency re-evaluated using the same methodology used in the internal events, but including the fire HFEs impacts based on "fire-induced" changes

Internal Fire PRA – Tasks 7 & 8

- **No/limited knowledge of the ignition source – target locational relationships**
  - **No way to perform "reasonably accurate" fire modeling**
  - **Full room burnout scenarios for PAUs for Quantitative Screening**
  - **Initial high CDF areas include the MCR, Containment, Turbine Building and 35 other single compartment PAUs**
  - **For "important" PAUs, the only detailed fire analysis involved:**
    - **Credit for automatic suppression systems**
    - **Credit for manual suppression in MCR and for hotwork fires**
    - **No fire modeling performed – if suppression fails, then back to full room burnout scenario**
- **No knowledge of intervening combustibles**
  - **No screening of even the smallest ignition sources**
  - **Main impact is that all unscreened MCA scenarios are assumed to be possible**

Internal Fire PRA – Task 11 & 13

- Task 11 - Detailed Analysis
  - Main Control Room
    - Abandonment scenarios
  - Containment
    - SLOCA potential and all 4 trains of instrumentation
  - Turbine Bldg.
    - Very Large IEF and OSP cables
  - 35 Other Single Compartments
    - Based on high initial CDF
  - Multi-Compartment Analysis
- Task 13 – Seismic-Fire Interaction Analysis
  - All qualitative based on design documents

Internal Fire PRA – Results (At-power)

- At Power total of 480 SCA and 1054 MCA Scenarios
- Due to highly compartmentalized nature of the APR1400, CDF is generally well distributed
  - Only 24 scenarios are higher than 1% CDF
  - 50% of CDF in top 8 scenarios, 90% of CDF in top 45 scenarios
  - CDF concentrated in MCR, EDG and Electrical Equipment Rooms
  - MCR is ~1/3 of CDF due to conservative analysis
    - No alternative shutdown (ASD) procedure (CCDP assumed 0.1)

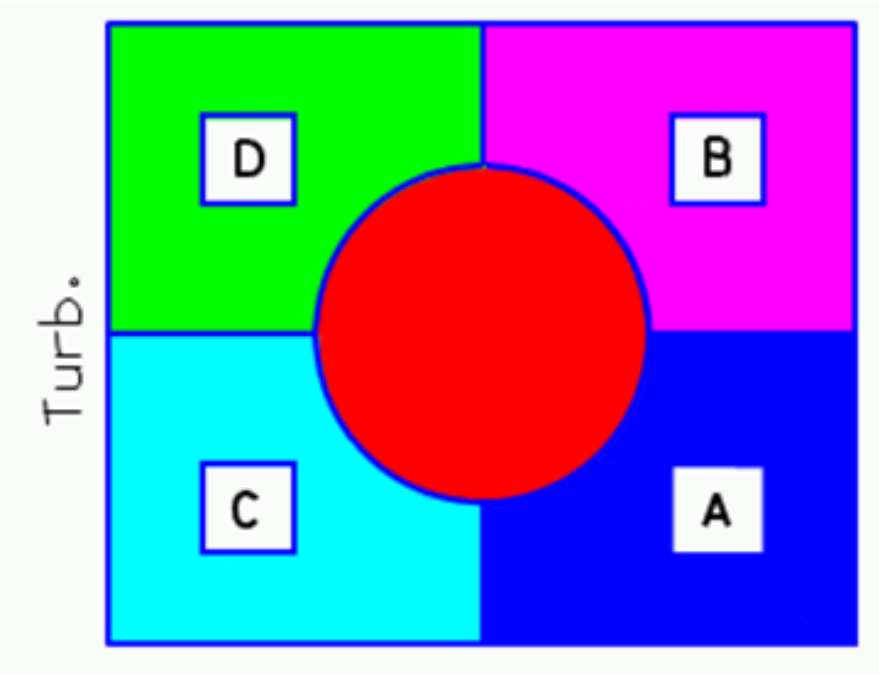
Internal Fire PRA – Results (LPSD)

- **LPSD total of 918 SCA and 6071 MCA Scenarios**
  - **LPSD Fire Scenario defined by both IE and POS**
- **Like the at-power results, due to highly compartmentalized nature of the APR1400, CDF is generally well distributed**
  - **Only 22 scenarios are higher than 1% CDF**
  - **50% of CDF in top 10 scenarios, 90% of CDF in top 59 scenarios**
  - **CDF concentrated in EDG and Electrical Equipment Rooms, and Ultimate Heat Sink (UHS) Building**

19.1.5.3 Internal Flooding PRA

APR1400 Characteristics

- **Auxiliary Building**
  - **Quadrant Design**
  - **Emergency overflow lines limit accumulation on upper elevations.**
  - **Limited flood sources**



APR1400 Internal Flooding Analysis (1/2)

- **Flood-induced initiating event**
  - Uncontrolled release of fluid that also fails PRA-related equipment.
  - Considers failures that result in reactor trip or a Technical Specification required shutdown within 24 hours.
- **Flood propagation**
  - Design flood barriers prevent propagation up to design level
  - Non-watertight doors fail at level of 1-foot if failure exacerbates accident, otherwise, doors remain intact (in general)
  - Barrier failures or other propagation pathways that ameliorate event are not considered.
  - Flow through drains, EOLs, other open pathways considered

APR1400 Internal Flooding Analysis (2/2)

- **Accident Sequence Analysis**
  - **Based on Internal Events Sequence**
    - **Over 130 events explicitly evaluated**
    - **Most events related to manual shutdown or assumed reactor trip**
  - **High Energy Line Breaks**
    - **Conservative treatment**
    - **HELB barriers intact up to design HELB**
    - **Break greater than design fails first barrier and actuates fire protection**

Flooding Insights

- No one significant event
- Most events are related to beyond-design-basis fire protection breaks.
- Propagation potential expected to be less if as-built plant could be analyzed
  - Door failure potential
  - Design flood barrier height
- Most breaks do not require isolation or time available is long

19.1.6 Low Power and Shutdown PRA

LPSD Analysis Strategy

- **Major Tasks**
  - Plant Operating State Development
  - Initiating Events Analysis
  - Accident Sequence Analysis
  - Success Criteria Analysis
  - Systems Analysis
  - Data Analysis
  - Human Reliability Analysis
  - Analysis of Large Release
  - Quantification
- **Same as at-power, plus POS development**

Analysis philosophy of POSs

APR1400 Plant Operating States

POS	Description	Primary System Water Level(1)	Primary System Pressure & Temperature	TS Mode
1	Reactor trip and Subcritical operation	In Pressurizer	2250 psia, 548-585˚F	1, 2
2	Cooldown with Steam Generators to 350˚F		2250-450 psia, 548–350˚F	3
3A	Cooldown with Shutdown Cooling System to 212˚F		450-15 psia, 350–212˚F	4
3B	Cooldown with Shutdown Cooling System to 140˚F		450-15 psia, 212–140˚F	5
4A	Reactor Coolant System drain-down (pressurizer manway closed)	Below Reactor Flange	Slight positive pressure or depressurized; <140˚F	5
4B	Reactor Coolant System drain-down (manway open)		Depressurized; <140˚F	5
5	Reduced Inventory operation and nozzle dam installation			5
6	Fill for refueling			6
7	Offload	Cavity flooded		6
8	Defueled	N/A	N/A	Defueled
9	Onload	Cavity flooded	Depressurized or slight vacuum during refill; <140˚F	6
10	Reactor Coolant System drain-down to Reduced Inventory after refueling	Below Reactor Flange		6
11	Reduced Inventory operation with steam generator manway closure			5
12A	Refill Reactor Coolant System (pressurizer manway open)			5
12B	Refill Reactor Coolant System (manway closed)		Depressurized, or at a slightly elevated pressure; <140˚F	5
13	Reactor Coolant System heat-up with Shutdown Cooling System isolation at 350˚F	In Pressurizer	15-450 psia, 140–350˚F	4
14	Reactor Coolant System heat-up with steam generators		450-2250 psia, 350–548˚F	3
15	Reactor startup		2250 psia, 548-585˚F	2, 1

(1) When level changes during a POS, the minimum level is listed.

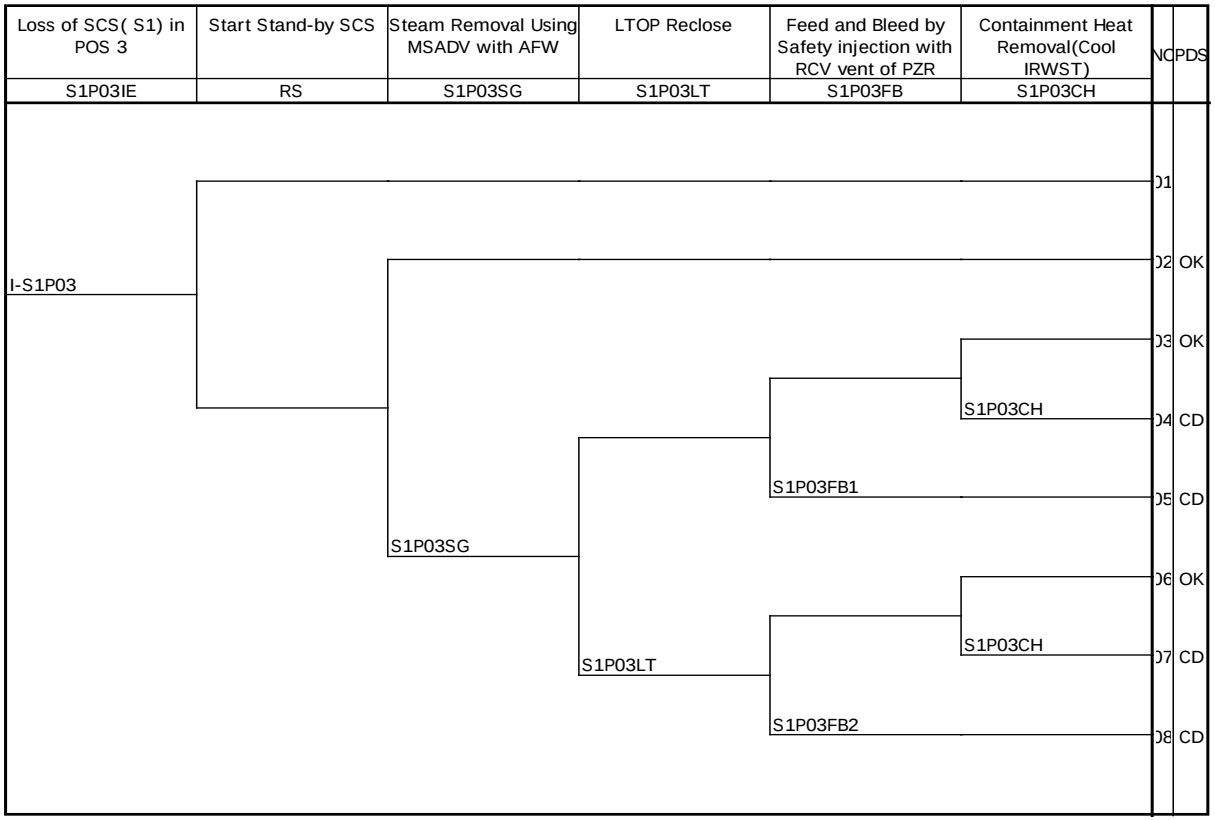
Initiating Event Selection

- **LPSD Initiating Events based on Industry Documents**
  - **NUREG/CR-6144**
  - **EPRI TR-1003113**

- **APR1400 initiating events were also compared to publicly available information on current and advanced evolutionary PWRs**

Accident Sequence Analysis

- **The AS analysis models the combinations of system responses and operator actions that could occur during the event based on mitigation functions/systems available during the POS**
- **Event Tree analysis is used to delineate these combinations to present these events**



Success Criteria

- Using MAAP 4.0.8 and RELAP5/mod3
- Considering the initiating event, limiting plant conditions for each POS, and equipment availability specified for each accident sequence
- Core Damage is defined based on the ASME PRA Standard and NRC Inspection Manual Chapter 0609

LPSD Level 2 Analysis

- For POSs with RCS and containment intact, Level 2 conservatively estimated using the full power conditional probability of large release (CPLR)
- For POSs with RCS intact but containment hatch open, failure to close hatch assumed to be large release. Successful closure of hatch before boiling evaluated using full power CPLR
- For POSs with RCS head removed, detailed Level 2 PRA developed. For portions of the analysis, the full power Level 2 models are conservatively used as bounding estimates.
- LPSD Level 2 Fire modeling same as internal events

LPSD Insights

- Results are dominated by failure of operator actions
- Results indicate that the draindown and reduced inventory POSs are highly risk significant

APR1400 DC PRA

PRA Update (2017)

PRA Update 2017

- Initiating event frequency update
  - Based on IEF data on NRC website, other NUREGs, Vendor Data, and calculation (for ISLOCA)
- Data updates, including CCF data
  - Based on availability, reliability and common cause failure data on the NRC websites
- HRA dependency analysis revision
- Design changes, including changes to technical bases
- Model changes from RAI responses
- Other model error corrections

Human Reliability Analysis (HRA)

- All independent HEPs for all modes / all models within HRA Calculator
- Full dependency analysis for each model including dependency between Level 1 and Level 2 HFEs
- Minimum joint HEP of 1E-6 was applied for multiple HFEs

Anticipated Transient Without SCRAM (ATWS)

- Original ATWS model based on one data point (16%), and did not consider turbine trip, or the number of POSRVs available to relieve RCS pressure
- Updated ATWS data provides Unfavorable Exposure Time (UET) percentages based on turbine trip, and number of available POSRVs

RCP Seal LOCA Model

- Based on WCAP-18067-P, Rev. 0, KSB RCP Seal (Type F) Failure Model
- Included in SBO, consequential SBO, PLOCCW, PLOESW, TLOCCW & TLOESW ETs
- Considers re-establishment of seal injection via Aux. Charging Pump within 20 minutes
- RCP must be tripped within 20 minutes to prevent seal LOCA in non-SBO sequences
- Even if RCPs are successfully tripped, there is a possibility of seal LOCA, this conditional seal failure probability (CSFP) is based on:
  - Controlled Bleed-Off (CBO) isolation (either isolated within 20 minutes, or isolation is assumed failed)
  - RCS subcooling
  - Event duration (assumed 24 hour mission time)

RCP Seal LOCA Model

- **Sensitivity Analysis increased the value of CSFP**
  - **Base Case (Total AP-IE) CDF = 1.1E-06/ryr**
    - CSFP = 0.01 resulted in CDF increase of ~3% (1.12E-06)
    - CSFP = 0.1 resulted in CDF increase of ~32% (1.44E-06)
    - CSFP = 1 resulted in CDF increase of ~342% (4.84E-06)
    - Total RCP Seal LOCA CDF 1.9E-09 (Base), 3.75E-06 (CSFP = 1)
  - **Base case LRF = 1.00E-7/ryr**
    - CSFP = 0.01 resulted in LRF increase of 1% (1.01E-7)
    - CSFP = 0.1 resulted in LRF increase of 5% (1.05E-7)
    - CSFP = 1 resulted in LRF increase of 66% (1.66E-7)

Software Failures in Digital I&C Model

- **Hardware failures in PPS/ESF-CCS still from reference plant**
  - **PPS Software CCF events added to existing model**
    - Bistable module application software failure fails all BPMs
    - LCL module application software failure fails all LCLs
    - Group Controller application software failure fails all GCs
    - Loop Controller application software failure fails all LCs
    - PPS Operating System CCF fails all BPMs, LCLs, GCs and LCs
- **Hardware failures in DPS still from reference plant**
  - **DPS Software CCF events added to existing model**
- **PPS and DPS software assumed independent based on DCD Chapter 7 (7.2.1.9) which states that PPS and DPS "are designed using different hardware and software to address postulated software common-cause failures"**
- **Operating system (1.2E-06/demand) and Application software (1.2E-05/ demand) CCF estimated from vendor information**

Digital I&C Sensitivity Cases

Case	Case Description	CDF	% Increase	LRF	%Increase
Base CDF	No changes	1.10E-06	N/A	1.00E-7	N/A
Case #1	OS CCF x 10	1.12E-06	3%	1.04E-7	4%
Case #2	All x 10	1.53E-06	39%	1.54E-7	54%
Case #3	All x 100	6.13E-06	459%	6.92E-7	592%
Case #4	DPSO-AP-DPS x 100	1.10E-06	<1%	1.00E-7	<1%
Case #5	DPSO-OS-DPS x 100	1.10E-06	<1%	1.00E-7	<1%
Case #6	PPSO-AP-BPM x 100	1.21E-06	10%	1.15E-7	15%
Case #7	PPSO-AP-GC x 100	2.10E-06	92%	2.22E-7	122%
Case #8	PPSO-AP-LC x 100	4.26E-06	289%	5.05E-7	405%
Case #9	PPSO-AP-LCL x 100	1.21E-06	10%	1.15E-7	15%
Case #10	PPSO-OS-PPS x 100	1.42E-06	30%	1.38E-7	38%
Case #11	No DPS	5.43E-06	396%	6.43E-7	543%
Case #12	All = 0	1.05E-06	-4.4%	9.43E-8	-5.7%

Other System Changes (1/2)

- **Aux. Power (AP)**
  - Main change was AAC can only power A or B train vs. A & C or B & D trains
  - AC Power changes to SX CT fans and valves due to SX CT design changes
  - Power supply changes to various equipment
  - Required assumed routing cables for Fire PRA
- **Instrument Air (IA)**
  - Change from a one train 3 compressor system to a 2 train system
  - IA no longer supports Service Air (SA), which has own compressor trains which are not in the model
  - Updated model decrease IA failure probability by a factor of ~4.5
  - Overall impact on CDF is insignificant. IA only has small role in the Level 1 PRA, so the increase in reliability is insignificant.

Other System Changes (2/2)

- **Essential Service Water (SX)**
  - Changes included number of fans in each CT, change from MOV to HOV for the CT inlet and bypass valves
  - Required assumed routing cables for Fire PRA
- **Aux. Building HVAC (VK) – Radiologically Controlled Area**
  - Original PRA model assumes AU01A and AU01B supports all CS, CV, SC and SI pumps based on old design. New design uses AU01A and AU01C for the Div. I pumps and AU01B and AU01D for the Div. II pumps.
  - Original design had CC pump rooms cooled by both cubicle cooler, and the emergency VK exhaust. In the new design, the CC pump rooms are only cooled via cubicle coolers.

Other Model Changes/Corrections

- Consequential LOOP model corrected per NRC review comments
  - Consequential LOOP used to be separated for LOCAs and transients
  - Model now assigns consequential LOOP events for initiators based on whether they generate an SI signal
- Self-identified missing ISLOCA pathways
  - Both Hot Leg injection lines screened from original analysis even though screening criteria were not fully met
  - Resulted in insignificant change to ISLOCA frequency (dominated by other ISLOCA pathways)
- Added evaluation of fire and floods on local manual action to "unlock" POSRVs for feed and bleed operation

PRA Update Results

- **Numerous competing changes to all models**
  - **Decrease in impact from RCP Seal LOCAs and ATWS**
  - **General increase due to HRA dependency**
  - **Some increase and some decrease due to data**
  - **Some increase and some decrease due to system changes**
- **Major changes in flooding mainly due to correction of self-identified errors**
  - **Original at-power flooding model did not fully account for damage after failure of operators to isolate a break**
    - AP flooding still very small < 5E-07
  - **Original LPSD flooding model did not account for UAT failures (erroneously assigned to main transformer which is offline)**
    - Major impact from this is in POS10 when the SATS are still offline for maintenance resulting in a LOOP. Some scenarios also fail 1 or more EDGs.
    - LPSD flooding still very small < 1E-07

At Power Level 2 Changes

- Performed full HEP dependency analysis between Level 1/Level 2 and Level 2/Level 2 actions

- Level 2 interface with Level 1 updated to accommodate Level 1 event tree changes (RCP Seal LOCA modeling)

- Level 1 changes carry through to Level 2. Notable impacts to LRF are:
  - Data updates (especially conditional probability of LOOP after SGTR and adverse MTC fraction in ATWS)
  - RCP seal LOCA
  - Addition of software CCF of digital I&C

LPSD Level 2 Changes

- Full HEP Dependency Analysis
- Made emergency containment spray backup system (ECSBS) modeling consistent with at-power ECSBS (previous modeling had some non-conservatism)
- LPSD Fire Level 2 – correction to flag file that had incorrectly failed both trains of containment isolation (e.g., %F078-A04C) => reduced LRF
- LPSD Fire Level 2 – correction to multi-compartment flag file that had incorrectly not failed both trains of CISO =>increased LRF

PRA Update Results

Operational Modes	Hazards	PRA Rev. 0 (DCD Rev. 1)		PRA Rev. 1 (DCD Rev. 2)		% Increase	
		CDF (/ryr)	LRF (/ryr)	CDF (/ryr)	LRF (/ryr)	CDF	LRF
At-Power	Internal Events	1.30E-06	1.10E-07	1.10E-06	1.00E-07	-15%	-9%
	Internal Fire	1.90E-06	1.60E-07	2.20E-06	1.90E-07	16%	16%
	Internal Flood	2.20E-07	1.80E-08	4.00E-07	2.50E-08	82%	41%
LPSD	Internal Events	2.60E-06	1.20E-07	1.80E-06	1.20E-07	-31%	0%
	Internal Fire	1.70E-06	1.30E-07	1.30E-06	9.60E-08	-26%	-26%
	Internal Flood	1.80E-08	---	8.00E-08	---	346%	---
Total		7.70E-06	5.56E-07(1)	6.80E-06	6.11E-07(1)	-12%	10%

(1) Includes the LPSD Internal Flood CDF bounding value.

PRA Update Conclusion

- **Risk insights from PRA update**
  - **Overall results are similar to the previous results**
    - Overall CDF decreased about 12%, but LRF increased about 10%
    - Given the design changes, and data update these differences are not unreasonable
  - **The risk contribution from RCP Seal LOCA is low**
  - **The risk contribution from the software CCFs of digital I&C is low**

Overview of Chapter 19.1

Current Status

- **Chapter 19 is completed.**
  - KHNP continues to monitor Chapter 19 to assure any conforming changes are addressed.
  - 14 open items that were identified in Phase 3 have been resolved with adequate and sufficient discussion with the staff.

- **Changes in Chapter 19.1 as reviewed and marked-up in response to the RAIs were incorporated into the DCD Rev. 2.**
- **The marked-up responses to RAI 19.1-10 and 19.1-85 will be incorporated into the DCD Rev. 3.**

Attachment: Acronyms & Abbreviations

AAC	Alternate Alternating Current
AFAS	Auxiliary Feedwater Actuation Signal
AFW	Auxiliary Feedwater
ATWS	Anticipated Transient Without Scram
BP	Bistable Processor
CBO	Controlled Breed-Off
CC	Component Cooling Water System
CCF	Common Cause Failure
CCW	Component Cooling Water
CDF	Core Damage Frequency
CDI	Conceptual Design Information
CISO	Containment Isolation
CS	Containment Spray System
CSFP	Conditional Seal Failure Probability
CT	Condensate Storage and Transfer System
CV	Chemical Volume Control System
DPS	Diverse Protection System
ECSBS	Emergency Containment Spray Backup System
EDG	Emergency Diesel Generator
ESF-CCS	Engineered Safety Feature – Component Control System
ESW	Essential Service Water
ET	Event Tree
GC	Group Controller
HEP	Human Error Probability
HFE	Human Failure Event
HOV	Hydraulic Operated Valve
HP Turbine	High Pressure Turbine
HRA	Human Reliability Analysis
I&C	Instrumentation and Control

Attachment: Acronyms & Abbreviations

IEC	International Electronical Committee
IEF	Initiating Event Frequency
IORV	Inadvertent Opening Relief Valve
IRWST	In-Containment Refueling Water Storage Tank
ISLOCA	Interfacing Systems Loss of Coolant Accident
KEPCO E&C	KEPCO Engineering & Construction Company
KHNP	Korea Hydro & Nuclear Power Company
LC	Loop Controller
LCL	Local Coincidence Logic
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPSD	Low Power and Shutdown
LRF	Large Release Frequency
MOV	Motor Operated Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSPI	Mitigating Systems Performance Index
MTC	Moderator Temperature Coefficient
NRC	U.S. Nuclear Regulatory Commission
PI SGTR	Pressure Induced Steam Generator Tube Rupture
PLOCCW	Partial Loss of Component Cooling Water
PLOESW	Partial Loss of Essential Service Water
POS	Plant Operational State(s)
POSRV	Pilot Operated Safety Relief Valve
PPS	Plant Protection System
PRA	Probabilistic Risk Assessment
RAP	Reliability Assurance Program

Attachment: Acronyms & Abbreviations

RCP	Reactor Coolant Pump
RCRY	Reactor Critical Year
RCS	Reactor Coolant System
RCW	Reactor Coolant Water
RCY	Reactor Calendar Year
RDT	Reactor Drain Tank
RVR	Reactor Vessel
SA	Service Air System
SAMDA	Severe Accident Mitigation Design Alternative
SAT	Standby Auxiliary Transformer
SBO	Station Blackout
SC	Shutdown Cooling
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SKN 3&4	Shin-Kori 3&4
SLOCA	Small Break Loss of Coolant Accident
SMA	Seismic Margin Analysis
SSIE	Supporting System Initiating Event
SX	Essential Service Water System
TLOCCW	Total Loss of Component Cooling Water
TLOESW	Total Loss of Essential Service Water
UAT	Unit Auxiliary Transformer
UET	Unfavorable Exposure Time
VCT	Volume Control Tank
VSLOCA	Very Small Break Loss of Coolant Accident
WO	Chilled Water System

Backup 19.1

HCLPF capacities of building structures

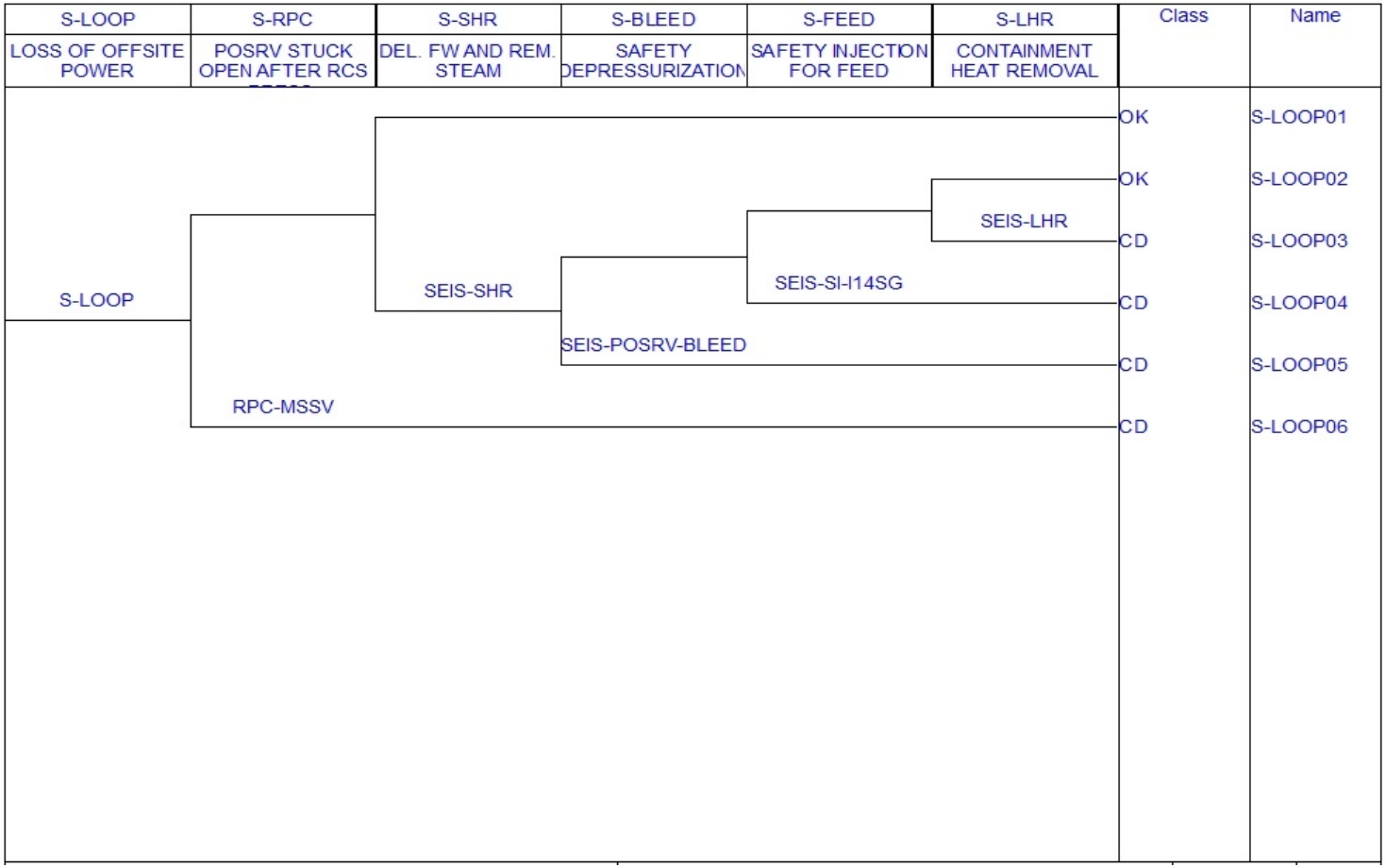
Building Structure	HCLPF (pga)	Critical Failure Mode
Reactor Containment	0.94g	Tangential shear failure near the base
Reactor Containment Concrete Internal	1.09g	Tangential shear failure of secondary shield wall near the base
Auxiliary Building	0.51g	Shear failure of Wall 15 at the basemat
Emergency Diesel Generator Building	0.87g	Shear failure of Wall 26 at the basemat
Diesel Fuel Oil Tank Room Building	0.73g	Shear failure of Wall 26.1 at the basemat
Stability of NI Structure	0.52g	Sliding toward the turbine building

HCLPF capacities of RCS components

RCS Component	HCLPF (pga)	Critical Failure Mode
Reactor Vessel	0.92g	Column support due to axial compression and biaxial bending
Reactor Internals	0.51g	Core support barrel lower flange due to primary membrane stress
Control Element Drive Mechanism	0.64g	Binding of control extension shaft with the upper pressure housing
Pressurizer	0.63g	Skirt support
	0.51g	Pressurizer spray nozzle
Steam Generator	0.60g	Snubber lever support assembly
	0.54g	Steam generator economizer nozzle
Reactor Coolant Pump	1.31g	Upper horizontal column support
RCS Piping	0.55g	Large loss of coolant at surge line nozzle Small loss of coolant at spray nozzle

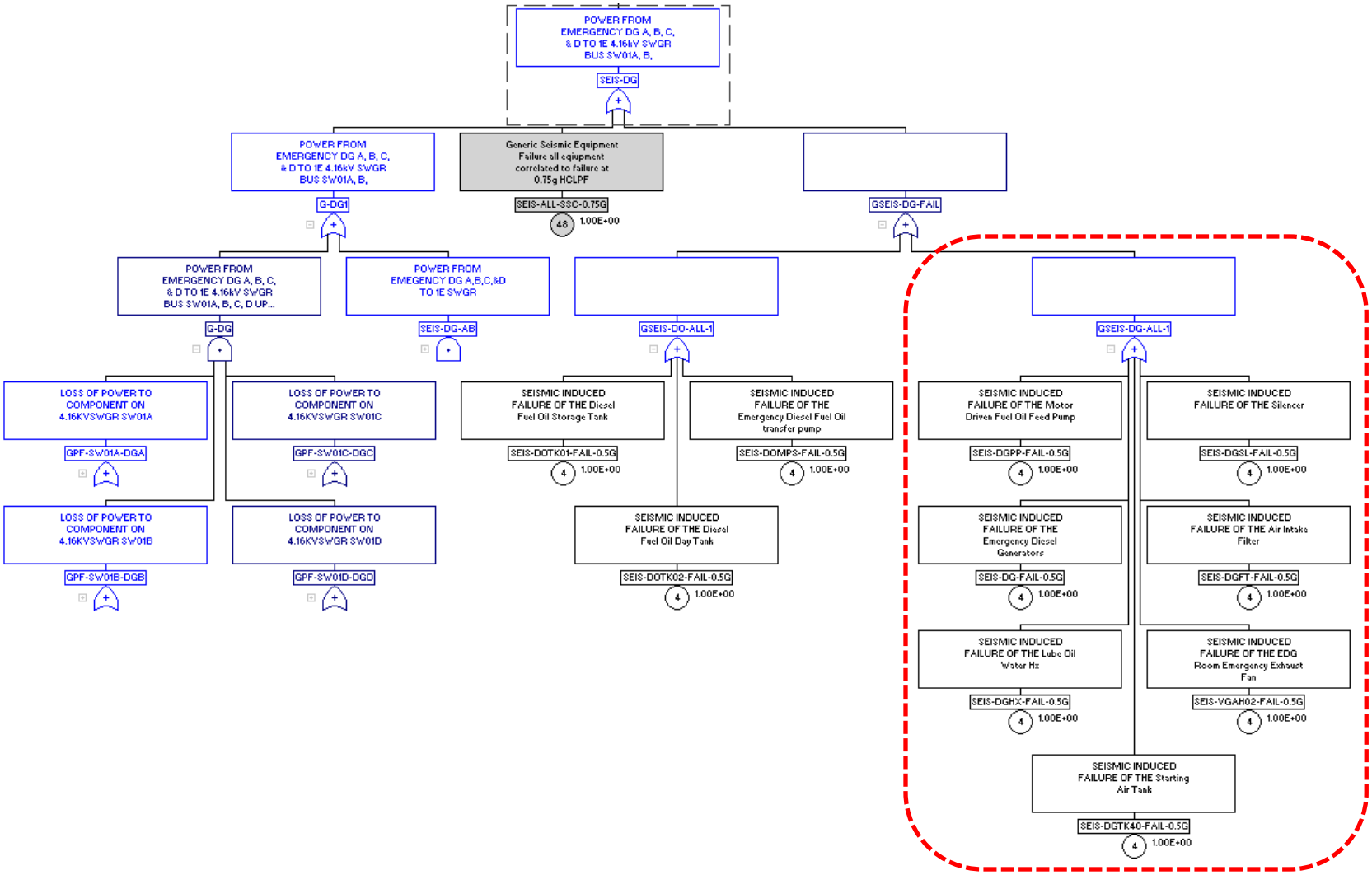
Seismic Logic Model (Cont'd)

- **Example of At-power PRA-based SMA ET (S-SLOOP)**



Seismic Logic Model (Cont'd)

- **Example of FT model (EDG and its auxiliaries)**



Internal Fire PRA – Task 4

- **No qualitative screening for at-power FPRA**
  - **Minimum general transient assumed for any fire in any PAU**
    - **Lack of knowledge in the design stage makes it difficult to support claim that a fire in the plant will not result in at least a trip**
- **LPSD Qualitative Screening**
  - **POS 1, 2, 14 and 15 screened in Task 5 (subset of at-power fires with CDF ~1% of at-power fire CDF due to short POS durations)**
  - **Since plant is already tripped in POS 3A - 13, fire must impact shutdown cooling, or result in loss of level control, if not, then no LPSD initiator**
    - **Different from at-power which assumes every fire has a CDF impact**
    - **137 unscreened PAUs analyzed for POS 3A - 13**
  - **Differs from NUREC/CR-7114 qualitative screening criteria which says to include all PAUs which contain credited equipment even if there is no initiator**
    - **Really only applicable in POS 1, 2, 14 and 15 which are screened**

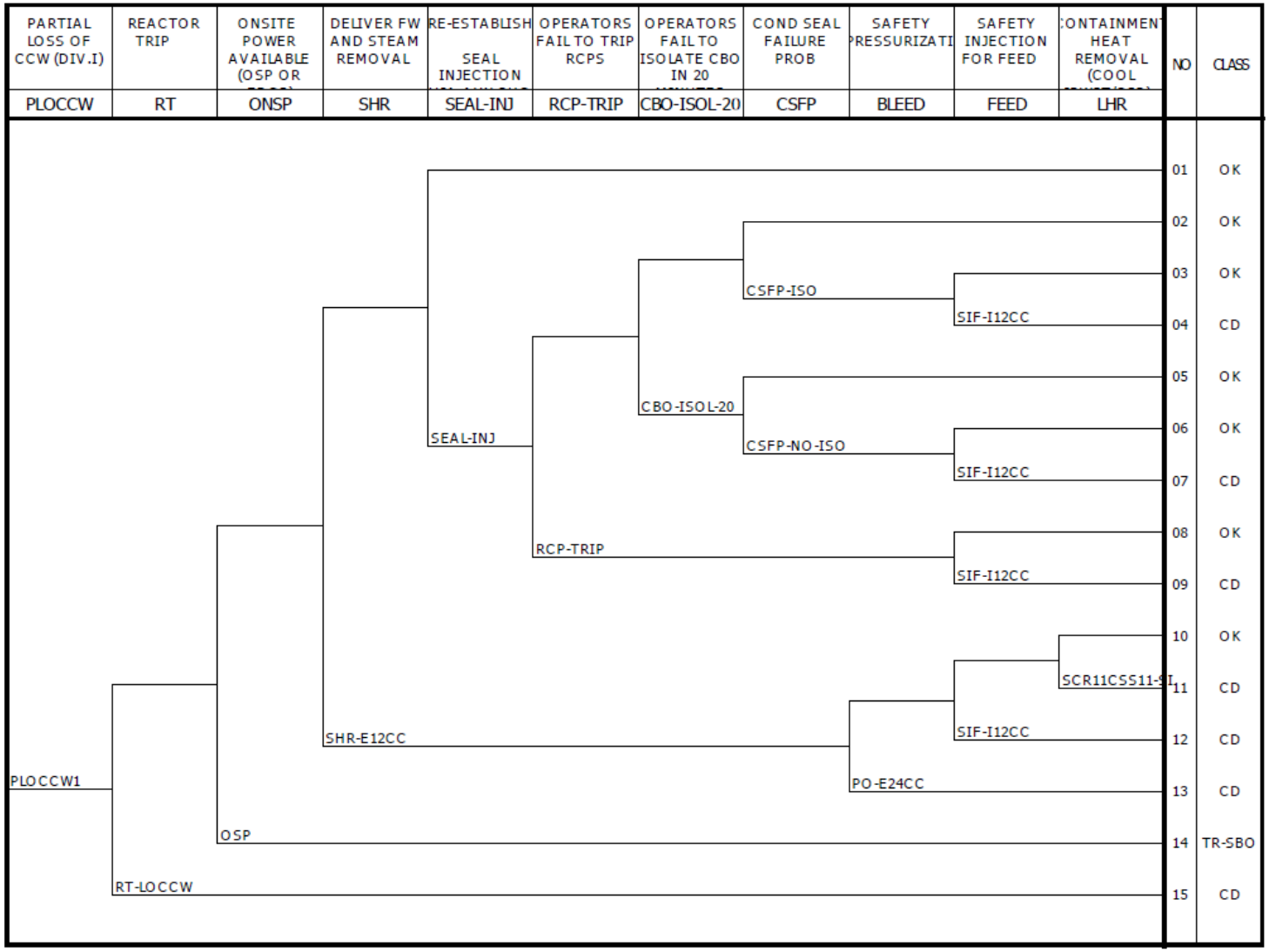
Initiating Event Frequency (IEF) Update

- Majority of IEF data from NRC Website 2010 data (Effective 2012)
  - 2015 Data was not published until after project start date
- SLOCA = SLOCA + VSLOCA + Random RCP Seal + IORV
  - SLOCA & VSLOCA from NRC Website
  - Random RCP Seal failure from NUREG/CR-5750
  - IORV based on POSRV vendor data
- Reactor Vessel Rupture (RVR) from NUREG-1829
- ISLOCA recalculated based on data update and inclusion of additional potential pathways

Data Update

- Majority of reliability and unavailability data from NRC Website 2010 data (Effective 2012)
  - 2015 Data was not published until after project start date
- SKN3&4 data used for some I&C components which are not included in NRC data
- CCF Data from NRC Website 2012 update
- ATWS Unfavorable Exposure Time (UET) values from KEPCO E&C analysis
- RCP Seal Conditional Seal Failure Probabilities from KSB RCP Seal (Type F) Failure Model

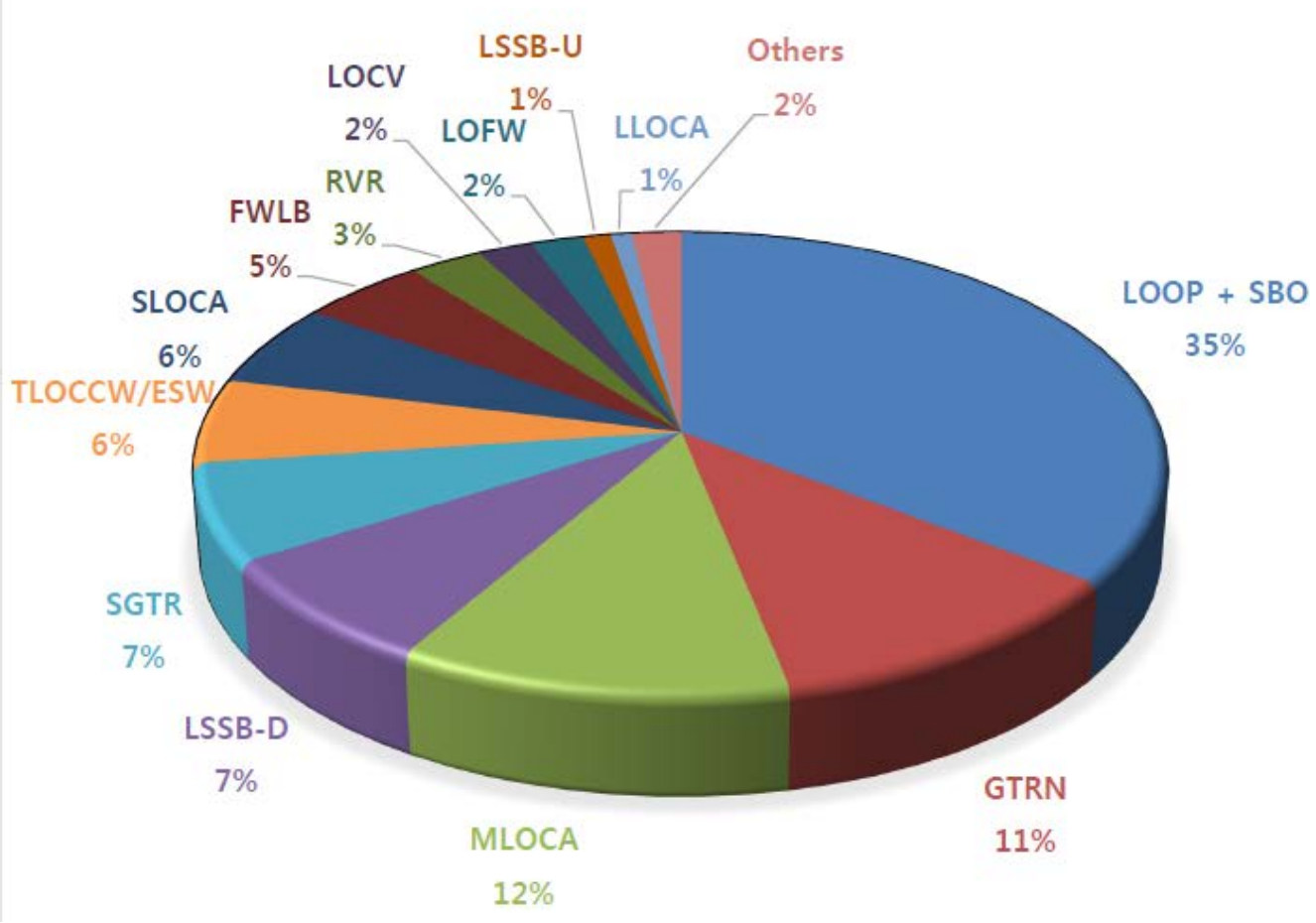
RCP Seal LOCA Model



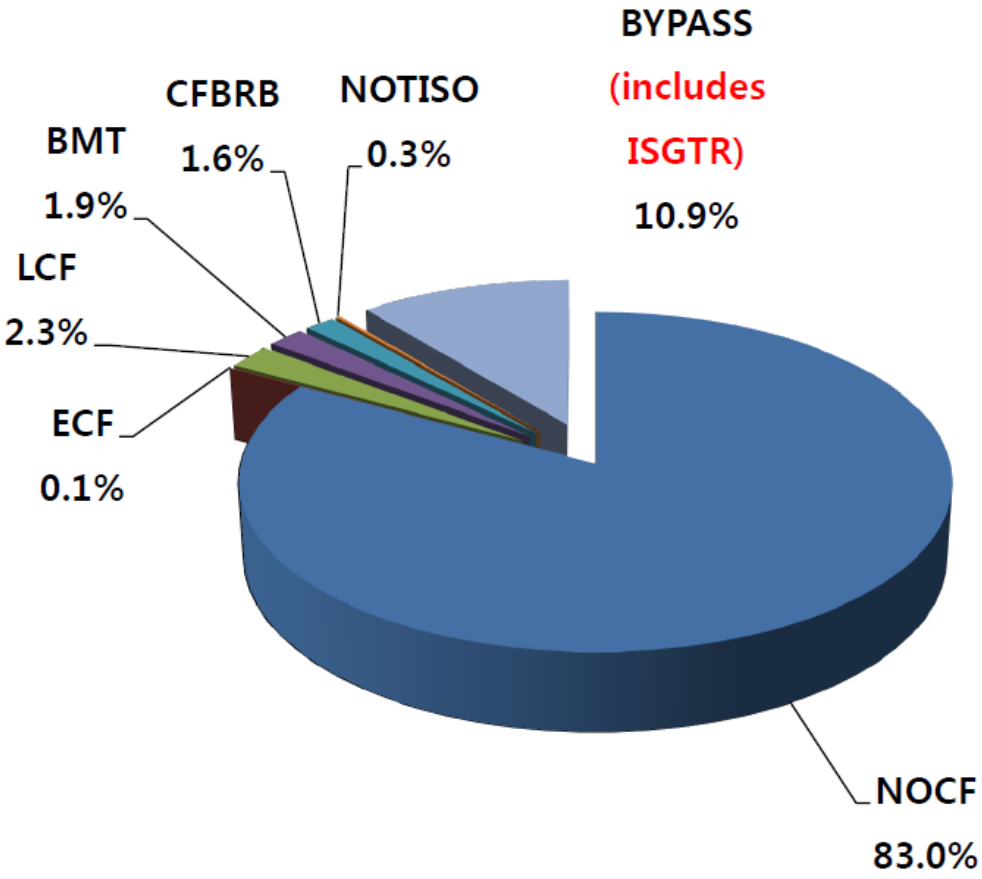
Software Failures in Digital I&C Model

- When reliability analysis (as described in Section 7.3.2.1 of IEC 61226 for Category A functions) are performed, the resultant reliability of each reactor trip signal (e.g., P-1, P-2, etc.), SIAS and AFAS can be shown to be ~1E-04 to ~3E-03
  - The above reliability analyses includes all PPS software and hardware failure modes including random, common cause, test and maintenance unavailability and miscalibration errors. Since the quantification is being performed to determine the reliability of the hardware and software, the model does not credit operator action to recover the failed signal.
  - This analysis is for PPS only, and does not consider DPS
- Note 2 of Section 7.3.2.1 of IEC 61226 states claims of reliability greater than 1E-04 are not precluded, but will need special justification; however, the reliability is less than 1E-04 in all cases.

At-Power Internal Events Risk Profile (CDF)



CET Quantification Results for Internal Events



Consideration of NUREG-2195 C-SGTR (1/2)

- **NUREG-2195 evaluated conditional probability of C-SGTR during**
  - **(1)Severe-accident progression, and**
  - **(2)As a consequence of ATWS/SSLB**

- **(1) NUREG-2195 calculated a higher probability of C-SGTR due to severe accident progression (relative to probs. used in APR1400 L2), but the APR1400 LRF does not have a high sensitivity**
  - **Sensitivity increased all APR1400 AP-IE cond. probs. of severe accident C-SGTR by 20%, and the AP-IE LRF increased by 2%.**

Consideration of NUREG-2195 C-SGTR (2/2)

- (2) ATWS and SSLB-induced SGTR is significant to the APR1400 LRF (first 10 AP-IE LRF cutsets related to ATWS or SSLB C-SGTR)
  - In APR1400 L2, cond. prob. of ATWS/SSLB C-SGTR is 2.7E-2
  - In NUREG-2195, cond. prob. of ATWS C-SGTR is 8.0E-3
  - In NUREG-2195, cond. prob. Of SSLB C-SGTR is 4.0E-3
  - Sensitivity changed all ATWS/SSLB C-SGTR from 2.7E-2 to 8.0E-3, and AP-IE LRF decreased by 31%.

Therefore, if the APR1400 Level 2 utilized NUREG-2195 for all conditional probabilities of C-SGTR, it is expected that the total LRF would decrease.

Overview of Chapter 19.1

❖ Summary of RAIs

No. of Questions	No. of Responses	No. of OI
103	103	14

Overview of Chapter 19.1

❖ List of Open Items from Phase 3 (1/2)

No.	Related RAI	Topic	ADAMS Accession #
1	RAI 232-7864 (Q 19.1-10)	The PRA-based seismic margin analysis (SMA) for low-power and shutdown (LPSD) conditions	ML18101A794
2	RAI 270-7894 (Q 19.1-14)	Summary of Other External Events Risk Evaluation	ML16077A317
3	RAI 271-8290 (Q 19.1-15)	Common-cause failure (CCF) modeling for digital instrumentation and control (DI&C) systems	ML17244A062
4	RAI 409-8325 (Q 19.1-28)	Operator action dependency analysis during LPSD midloop operation	ML17271A479
5	RAI 418-8348 (Q 19.1-43)	The truncation analyses for the APR1400 DC PRA model quantifications	ML17271A482
6	RAI 418-8348 (Q 19.1-45)	AFWS reliability analysis in accordance with TMI Action Item II.E.1.1 of NUREG-0737	ML17244A639
7	RAI 418-8348 (Q 19.1-47)	Summary of Internal Flooding Effects on POSRV Pilot Valve Operation	ML17237C042
8	RAI 432-8377 (Q 19.1-59)	The cesium iodine release fraction	ML16342C937
9	RAI 434-8352 (Q 19.1-85)	The PRA-based seismic margin analysis (SMA) for at-power condition	ML18101A155

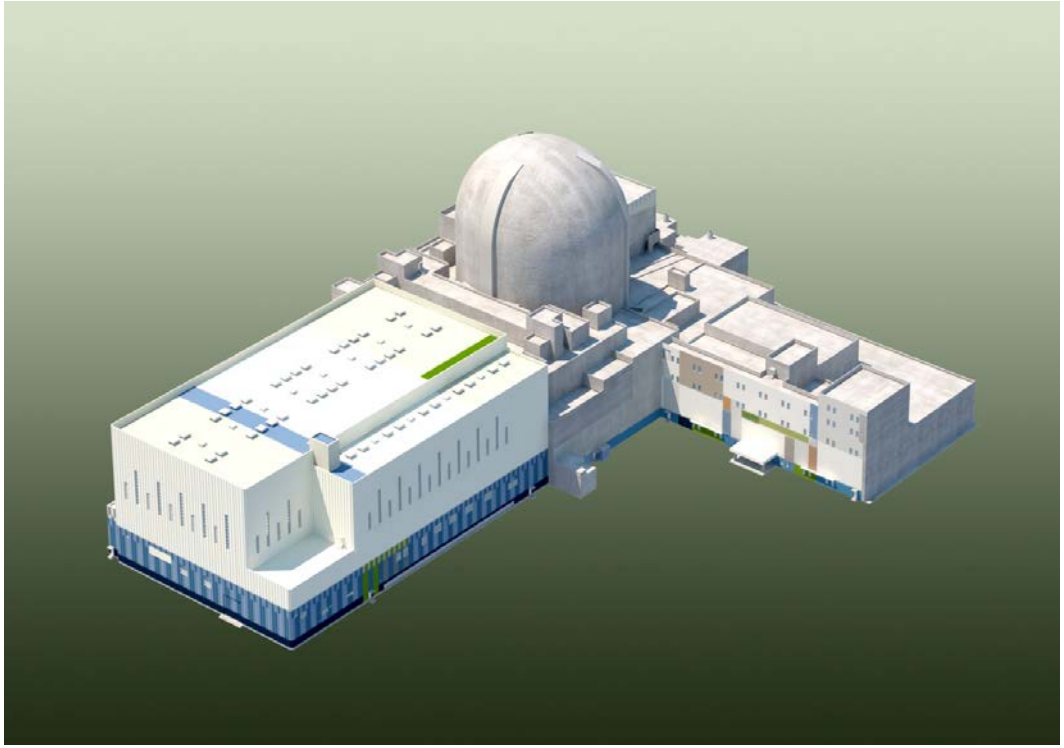
Overview of Chapter 19.1

❖ List of Open Items from Phase 3 (2/2)

No.	Related RAI	Topic	ADAMS Accession #
10	RAI 434-8352 (Q 19.1-87)	ITAAC listing associated with APR1400 PRA results and risk insights	ML17311B050
11	RAI 434-8352 (Q 19.1-88)	Technical adequacy of APR1400 DC PRA, and the Peer Review Resolutions	ML16305A430
12	RAI 434-8352 (Q 19.1-91)	APR1400 DC PRA Model Conversion to CAFTA	ML17244A645
13	RAI 434-8352 (Q 19.1-92)	APR1400 DC PRA 2017 Update Results	ML16222A925
14	RAI 426-8492 (Q 19.2-53)	Effect of issues identified on all of the MAAP calculations	ML18067A847

NON-PROPRIETARY

APR1400 DCA
Chapter 19.2: Severe Accident Evaluation



KEPCO/KHNP

June 5, 2018

ACRS SC Meeting (June 5, 2018)

Contents

- **Overview of Chapter 19.2**
  - Section Overview
  - List of Submitted Documents and Summary of RAIs
  - Major Changes of DCD
  - List of Open Items
- **Summary of Open Items**
- **Current Status**
- **Attachments:**
  - Acronyms
  - List of COL Items related to Open Items

Overview of Chapter 19.2

Section	Title	Major Contents
19.2.1	Introduction	• Regulation Considerations
19.2.2	Severe Accident Prevention	• ATWS • Mid-loop operation • SBO • Fire protection • ISLOCA, etc.
19.2.3	Severe Accident Mitigation	• Overview of Containment Design • Severe Accident Progression • Severe Accident Mitigation Features
19.2.4	Containment Performance Capability	• Containment Performance Goal • Containment Performance Analysis
19.2.5	Accident Management	• Severe Accident Management Framework • Prevent Core Damage • Retain the Core within the Reactor Vessel • Maintain Containment Integrity • Minimize Offsite Release
19.2.6	Consideration of Potential Design Improvement	• Severe Accident Mitigation Design Alternatives (SAMDA)
19.2.7	Combined License Information	• COL items

Overview of Chapter 19.2

❖ List of Submitted Documents for Chapter 19.2

Document No.	Title	Revision	Type	ADAMS Accession No.
APR1400-K-X-FS-14002 -P & NP	APR1400 Design Control Document Tier 2: Chapter 19.2 Severe Accident Evaluation	2	DCD	-
APR1400-K-X-IT-14001 -P & NP	APR1400 Design Control Document Tier 1	2	DCD	-
APR1400-E-P-14003 -P & NP	Severe Accident Analysis Report	2	TER	ML18081A098

❖ Summary of RAIs for Chapter 19.2

No. of Questions	No. of Responses	No. of OI
22	22	5

Overview of Chapter 19.2

❖ Major Changes of DCD (Rev.0 to Rev.2) - 1

Chapter	Topic	Rev.0	Rev.2
19.2.3.2.1	In-Vessel Melt Progression	No description for vessel failure mechanism	Detailed vessel failure mechanism considered in MAAP4.08 was incorporated.
19.2.3.2.2	Ex-Vessel Melt Progression	No description for uncertainty relating to ex-vessel melt progression	Considerable uncertainty was listed. And conservative approach to address the uncertainty was incorporated.
19.2.3.3.3.2 and 19.2.3.3.3.3	MCCI Calculation	Brief description for linking CORQUENCH and MAAP4.08	Detailed information such as - initial conditions and conservative approach used in CORQUENCH calculation, - modeling the heat transfer across the upper crust were incorporated. Missed figure for MAAP result was also corrected.
19.2.3.3.5.2.1	Fuel-Coolant Interactions	No description for in-vessel steam explosion results	Plant specific dynamic load was provided. Lower head integrity study against the load was also incorporated.

Overview of Chapter 19.2

❖ Major Changes of DCD (Rev.0 to Rev.2) - 2

Chapter	Topic	Rev.0	Rev.2
19.2.4.1	Containment Performance Goal	No details for accident selection	Detailed information by following RG 1.216 Position 3 was incorporated.
19.2.4.2.1	Combustible Gas Control inside Containment	No details for containment integrity against the hydrogen combustion load	In terms of RG 1.216 Position 2, detailed description including the calculated strain was incorporated.
19.2.4.2.2	Containment Pressurization Result	Brief description for containment performance analysis result	Followings were incorporated: - Accident selection based on PRA and deterministic viewpoint, - Bounding pressure and temperature from the selected sequences, - Containment modeling in FE study, - Calculated strain, and - Conclusion for the containment integrity on the basis of RG 1.216 Position 3.

Overview of Chapter 19.2

❖ List of Open Items

No.	Related RAI	Topic	ADAMS Accession #
1	RAI 432-8377 (Q 19-63)	In-vessel steam explosion	ML17058A238
2	RAI 432-8377 (Q 19-64)	Ex-vessel steam explosion	ML17240A414
3	RAI 433-8363 (Q 19-70)	Finite Element models of the containment	ML16243A525
4	RAI 433-8363 (Q 19-71)	Severe accident sequences contributing to core damage	ML16196A260
*	RAI 426-8492 (Q 19-53)	MAAP confirmatory calculations*	

* Item discussed in DCD Chapter 19.1

Summary of Open Items - 1

❖ Open Item: In-Vessel Steam Explosion (IVSE)

- Related RAIs : 432-8377 (Q 19-63)
- Description of issue
  - Staff requested:
    - ✓ Justify using one-dimensional numerical code
    - ✓ Justify the time at which triggering was assumed
- Resolution:
  - ✓ KHNP provided the description regarding the conservatisms of one-dimensional code based on optimized cross-sectional area of coolant.
  - ✓ KHNP provided the justification for the triggering by the corium jet contact at the bottom of reactor vessel, which can create the conservative loading.
  - ✓ KHNP provided the response to RAI with the revised TeR (APR1400-E-P-14003-P & NP, Severe Accident Analysis Report, Rev. 1) in February, 2017.

Summary of Open Items - 2

❖ Open Item: Ex-Vessel Steam Explosion (EVSE)

- Related RAIs : 432-8377 (Q 19-64)
- Description of issue
  - Staff requested to provide:
    - ✓ Justify using one-dimensional numerical code
    - ✓ Justify the time at which triggering was assumed
    - ✓ Justify loading on the cavity walls and possible concrete damage
- Resolution
  - See Q 19-63
  - See Q 19-63
  - KHNP provided how the loads on the walls were decided based on the conservative approach. KHNP also provided concrete and line plate damage details in August, 2017. Accordingly those are incorporated into DCD Rev.2 and TeR (APR1400-E-P-14003-P & NP, Severe Accident Analysis Report, Rev. 2) in January, 2018.

Summary of Open Items - 3

❖ Open Item: Finite Element models of Containment

- Related RAIs : 433-8363 (Q 19-70)
- Description of issue
  - Staff requested:
    - ✓ Provide the global and local nonlinear finite element models
    - ✓ Clarify the accident temperature
    - ✓ Update DCD to incorporate FE analysis results
- Resolution
  - KHNP provided the details of FE models such as the used code and model elements in the response, and incorporated those into DCD Rev.2.
  - KHNP described the selection of bounding temperature based on the severe accident analysis. The degradation of material properties by the elevated temperature was also responded.
  - KHNP incorporated the details of above into DCD Rev.2.

Summary of Open Items - 4

❖ Open Item: Severe accident sequences contributing to core damage

- Related RAIs : 433-8363 (Q 19-71)
- Description of issue
  - Staff requested to provide:
    - ✓ Identify the more likely severe accident sequences
    - ✓ Justify loading on the containment used in the analysis
- Resolution
  - KHNP responded the selection of accident complies with RG 1.216 Position 3.
  - KHNP justified the application of the highest severe accident load from the containment response study for the containment integrity assessment.
  - KHNP provided the revised DCD accordingly.

Current Status

- **Chapter 19.2 is completed.**
  - KHNP continues to monitor Chapter 19.2 to assure any conforming changes are addressed.
  - 4 open items, that were identified in Phase 2 and 3, have been resolved with adequate and sufficient discussion with the staff.

- **Changes in Chapter 19.2 as reviewed and marked-up in response to the RAIs were incorporated into the DCD Rev.2.**

Attachment: Acronyms

ATWS	Anticipated Transient Without Scram
COL	Combined Operation License
DCD	Design Control Document
FE	Finite Element
ISLOCA	Inter-System Loss-Of-Coolant Accident
KEPCO	Korea Electric Power Corporation
KHNP	Korea Hydro and Nuclear Power Co. LTD
MCCI	Molten Core Concrete Interaction
OI	Open Item
RAI	Request for Additional Information
RG	Regulatory Guide
SAMDA	Severe Accident Mitigation Design Alternatives
SBO	Station BlackOut
TeR	Technical Report

Attachment : List of COL Item related to OIs

COL Identifier	Description
COL 19.2(1)	The COL applicant and/or holder is to perform and submit site-specific equipment survivability assessment including flooding effect in accordance with 10 CFR 50.34(f) and 10 CFR 50.44 which reflects the equipment identified and the containment atmospheric assessments of temperature, pressure and radiation described in Subsection 19.2.3.3.7.
COL 19.2(2)	The COL applicant and/or holder will demonstrate that the covers for the large penetrations such as equipment hatch and personnel airlocks meet the Service Level C requirements in Subsection NE-3220 of the ASME code and explain how the consideration of containment leakage is accounted for when modeling local regions of containment.
COL 19.2(3)	The COL applicant and/or holder is to develop and submit an accident management plan including the evaluation of the effect of higher water level in the cavity on steam explosion loading when using In-Vessel Retention and External Reactor Vessel Cooling for accident management.

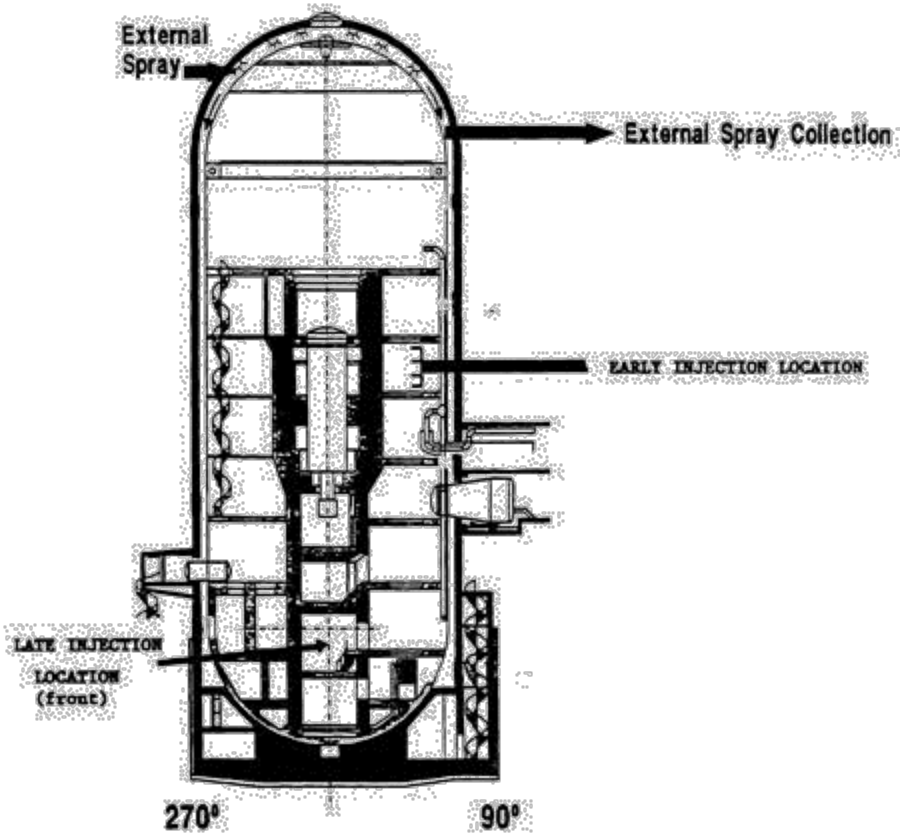
Backup Slides for 19.2

In-Vessel Retention - External Rx Vessel Cooling

- Update PRA does not rely on the presence of this system
- APR 1400 Ch. 19.2 analysis did not take credit for IVR-ERVC
- KHNP considers IVR-ERVC an accident management option
- COL applicant, at its discretion, can develop and submit as part of its accident management plans, including addressing steam explosion loading and structural integrity assessment

HDR 11-2

- HDR containment is highly compartmentalized (~ 70 rooms) unlike a large dry PWR containment
- In HDR, a dominant flow path can be identified by a pair of staircases and equipment shafts running from low elevations up to the dome at diametrically opposite positions (no flow paths between them except at the dome (30 m level) and at 10 m level)
- HDR 11-2 shows that a light gas injected mixes well with the gas above the injection elevation and remains stratified (that was observed for both early injection at mid-elevation and late injection at low elevation)
- In HDR, peak hydrogen concentration in the dome region was caused by external cooling of containment dome steel wall



Bases for APR1400 Containment Nodalization

- Fine nodalization near source locations to capture local high gas concentrations: reactor cavity and steam generator compartment
- Coarse nodalization away and above the source location such as the reactor dome, where the gas is expected to mix well
- Sufficient nodalization to represent a density driven global natural circulation flow paths

Natural Tendency Towards Mixing in Large Dry Containments such as APR1400

NON-PROPRIETARY

- Formation of a high concentration stratified layer away from the source is unlikely in a large dry containment; the gas is diluted rapidly due to entrainment while it is rising as a plume
- The concentration of the stratified layer increases only after the layer boundary comes down to the source elevation
- Mixing within the stratified layer is enhanced due to density driven natural circulation and heat transfer within the stratified layer, especially if steam is released with the gas
- In APR1400 the source location is relatively low and the gas concentration effect due to stratification (instead of mixing with the entire containment atmosphere) is moderate

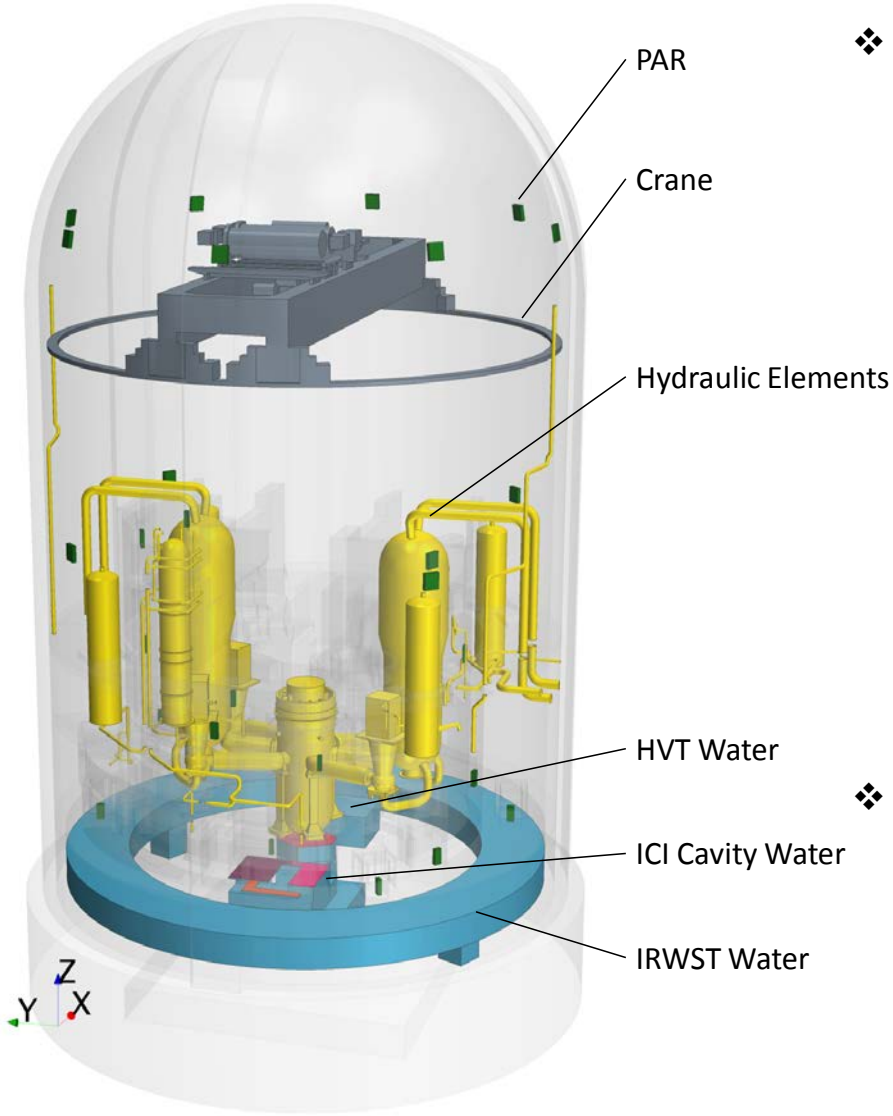
Modeling Combustible Gas Accumulation

❖ Procedure for Selection of PARs Location

- Scenario selection
- Analysis of hydrogen mixing and distribution with no PARs
  - Pre-analysis using lumped code
  - Supplementary study using three-dimensional code
  - Characteristics of hydrogen generation and release
  - Selection of draft location and capacity of PARs
- Analysis of PARs impact
  - Comprehensive analysis using lumped code with PARs
  - Analysis of Flame Acceleration and Deflagration-to-Detonation Transition
- Final decision on PARs location
  - Local corrective action according to engineering judgement

Modeling Combustible Gas Accumulation

❖ Distribution of PARs inside the Containment



❖ **Fundamental Principles in PARs Distribution**

- Place where large convective flow exists,
- Recombination capacity proportional to the free volume of compartment,
- Install above hydrogen release points,
- Easy maintenance and inspection, and
- Install in locations protected from external damage.

❖ **Because of PAR distribution, it is very unlikely to develop a stratified layer rich in flammable gas.**

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

APR1400 Design Certification Application Review

Safety Evaluation with No Open Items

Chapter 17: Quality Assurance and Reliability Assurance

June 5 , 2018

Chapter 17: QUALITY ASSURANCE AND RELIABILITY ASSURANCE

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Technical Staff Presenters and Staff Review Team

- **Aaron Armstrong (DCIP/QVIB)**
  - **Sections 17.1, 17.2, 17.3, and 17.5**
- **Odunayo Ayegbusi (DSRA/SPRA)**
  - **Section 17.4**
- **Project Managers**
  - **Bill Ward – Lead Project Manager (DLSE/LB2)**
  - **Mark Lintz – Chapter Project Manager (DLSE/LB3)**

Quality Assurance

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- The quality assurance (QA) program is described in Sections 17.1, 17.2, 17.3, and 17.5
- All regulatory requirements of Appendix B to 10 CFR Part 50 have been satisfied
- The QA Program Description (QAPD) Topical Report is approved and will provide the QA requirements for Sections 17.1, 17.2, 17.3, and 17.5
- Staff complete the QA implementation inspections in July 2017. The identified findings in Inspection Report 05200046/2016-201 (ADAMS No. ML17171A237) are closed.

Reliability Assurance Program D-RAP List Update

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Staff evaluated KHNP's revised reliability assurance program (RAP), including the design RAP (D-RAP) list
- Staff identified that the risk-significant in-containment refueling water storage tank (IRWST) sump was not provided to the expert panel for D-RAP list consideration
- An expert panel was convened, which added the IRWST sump to the D-RAP list
- Staff found that KHNP had adequately implemented the expert panel in developing the D-RAP list
- Staff found that the D-RAP list was developed in accordance with its RAP methodology and that the D-RAP list was comprehensive

Non-Proprietary

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

Korea Hydro & Nuclear Power Co., Ltd (KHNP), and Korea Electric Power Company (KEPCO) APR1400 Design Certification Application Review

Safety Evaluation with No Open Items

Chapter 19.1 - PROBABILISTIC RISK ASSESSMENT

June 5, 2018

Staff Review Team

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- **Technical Staff**
  - **Hanh Phan** (Lead), Senior Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Odunayo Ayegbusi**, Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Tony Nakanishi**, Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Alissa Neuhausen**, General Engineer, Structural Engineering Branch
  - **Marie Pohida**, Senior Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Robert Roche-Rivera**, Structural Engineer, Structural Engineering Branch
  - **Courtney St. Peters**, Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Hanry Wagage**, Senior Reactor Engineer, Containment & Ventilation Branch
- **Project Managers**
  - **William Ward,** Lead Project Manager
  - **James Steckel**, Chapter Project Manager

Presentation Outline
Technical Areas and Presenters

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

1) Staff's Review of APR1400 PRA (Phan)
2) At-Power Internal Events Level 1 PRA (St. Peters, Ayegbusi)
3) At-Power Internal Events Level 2 PRA (Wagage)
4) LPSD Internal Events Level 1 and 2 PRA (Pohida)
5) At-Power PRA-based Seismic Margin Assessment (Phan)
6) LPSD PRA-based Seismic Margin Assessment (Pohida)
7) Internal Fire and Flood PRAs (Nakanishi)
8) Staff Findings and Conclusion (Phan)

Staff's Review of APR1400 PRA

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- By the end of Phase 2, thirteen OIs on PRA were identified
- Activities during Phase 4 included:
  - Reviewing updated PRA
  - Resolving and closing OIs
  - Validating RAI responses against the final PRA model
  - Assessing impacts on PRA from design changes
  - Verifying revised DCD information against the supporting documents
  - Re-evaluating PRA input to other DCD chapters, i.e., physical security, ITAAC, RAP, human factors engineering, and environmental report
- Focused on the quality, completeness, and consistency of the information in DCD Chapter 19 to ensure that the results, conclusions, and insights obtained from the PRA are valid

Staff's Review of APR1400 PRA (continued)

United States Nuclear Regulatory Commission
Protecting People and the Environment

- PRA regulatory audit provided access to the PRA notebooks and rationales used to support PRA development
- Staff sampled supporting documents relevant to the Level 1 and Level 2 internal events PRA, internal fire PRA, internal flooding PRA, LPSD PRA, PRA-based SMA, and external events assessment
- Staff issued 21 RAIs with 103 questions and over 200 audit questions. All RAIs and audit questions have been resolved

At-Power IEs Level 1 PRA
Digital I&C

United States Nuclear Regulatory Commission
Protecting People and the Environment

- In the response to an open item, the applicant addressed system description, key assumptions, CCFs of hardware and software, and failure effects
- Multiple meetings and discussions held between NRC probabilistic and deterministic staff and the applicant
- Operating system software CCF probability of 1.2E-06
- Application software CCF probability of 1.2E-05
- The applicant evaluated 12 different sensitivity cases to better understand the impact of software reliability values

At-Power IEs Level 1 PRA
RCP Seal LOCA

United States Nuclear Regulatory Commission
Protecting People and the Environment

- RCP seal LOCA testing results submitted on the docket
  - Associated post-accident operator actions were identified but not further evaluated because they are plant-specific
- Testing results support conditional seal failure probability in DCD Revision 2, Table 19.1-16
- Multiple sensitivity cases were performed to determine impact on CDF related to conditional seal failure probabilities

At-Power IEs Level 1 PRA
Model Conversion

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Staff evaluated the APR1400 PRA model conversion documented in the conversion report
- Staff found the PRA software platform conversion acceptable because the resulting CDF from the CAFTA model was found to be very close to the previously estimated CDF using SAREX software
- Staff compared the cutsets generated by the CAFTA model with those quantified by SAREX software
- The results were determined to be close to the previous results and the applicant was able to explain changes in the ranking of important events, sequences, and SSCs
- Staff found the model conversion was properly performed and the subsequent results and insights were adequately documented in the DCD

At-Power IEs Level 1 PRA
PRA Update & Results

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Staff audited results of the PRA update to ensure the PRA model adequately reflected the APR1400 design
- Staff identified 2 issues from reviewing cutsets contributing to 50 percent of the internal events CDF:
  - The top 100 cutsets included two cutsets (#55, #73) with a single basic event (component) failure in response to an IE that results in core damage
    - Additional analysis performed showed that the hot leg injection is not needed during a hot leg break LOCA
  - Three cutsets (#28, #51, #100) with mutually exclusive basic events (both trains in T&M)
    - SRs QU-B7 and QU-B8 of ASME/ANS PRA Standard as endorsed by RG 1.200, require the deletion of mutually exclusive events from the PRA results

At-Power IEs Level 2 PRA
Severe Accident-Induced SGTR

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- APR1400 PRA assumes NUREG-1570 probabilities of severe accident-induced SGTR after increasing them for CE-type plants
- NUREG-2195 estimates probability of 0.22 to 0.31 for CE SGs
- Due to more realistic accounting for physical phenomena, the SOARCA study showed this scenario does not result in a large release
  - Aerosol deposition in the steam generator (Aerosol Trapping in a Steam Generator (ARTIST) test program)
  - Subsequent severe accident-induced hot leg creep rupture resulting in redirecting the release into the containment where it can deposit

LPSD IEs Level 1 and 2 PRA
Background

- POS definitions evaluated for completeness from hot shutdown to RCS heatup
  - Time to boiling and core uncovery calculated
  - Status of RCS penetrations, RCS level, decay heat
- Event trees developed for each POS except refueling cavity flooded operation
- Refueling cavity flooded operations evaluated qualitatively

LPSD IEs Level 1 and 2 PRA
Open Items Evaluated in Other Chapters

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- DCD discrepancy between Level 2 PRA and Shutdown Containment Closure TS LCO 3.6.7 resolved in Chapter 16
  - LCO 3.6.7 updated to require containment closure in Mode 5 with RCS loops not filled and Mode 6 with water < 23 feet above the top of the reactor vessel flange
- The potential for air entraining/vortexing SDC pump during midloop conditions impacting the RCS overdrain frequency resolved in Chapter 5.4.7
  - ITAAC added to DCD Revision 1, Tier 1, Table 2.4.1-4
  - The SCS will not be impaired by gas entrainment during mid-loop operation while the system is operating at its maximum allowable flow rate and lowest allowable hot leg level for decay heat removal

LPSD IEs Level 1 and 2 PRA
Operator Dependence

- Dependency analysis revised during PRA update
- Applicant performed sensitivity using a joint minimum HEP of 1E-5
- This sensitivity resulted in an increase in LRF from all plant operational states from 1.2E-7/yr to 1.4E-7/yr

At-Power PRA-based SMA
Applicable Guidance

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- SECY-93-087 and the related SRM, 1993
- SRP Section 19.0, 2007
- DC/COL-ISG-020, 2010
- "Seismic Probability Risk Assessment Implementation Guide," EPRI 1002989, 2003
- "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," NUREG/CR-4334, 1985
- "Recommendations to the NRC on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants," NUREG/CR-4482, 1986
- "Methodology for Developing Seismic Fragilities," EPRI TR-103959, 1994
- "Requirements for Seismic Events At-Power PRA," Part 5 of ASME/ANS RA-Sa-2009

At-power PRA-based SMA
Background

- The objective of PRA-based SMA is to evaluate the plant HCLPF and identify significant seismic vulnerabilities and insights to demonstrate the robustness of the design
- Staff focused on:
  - The framework for assessing potential significant failures induced by seismic events
  - The basis for establishing the SEL, which identifies essential equipment and structures to respond and mitigate the seismic accidents
  - Seismic fragility evaluation for determining sequence-level and plant-level HCLPFs
  - Seismic-related risk insights
- The revisions of at-power and LPSD PRA-based SMAs are confirmatory items under RAI 8352, Question 19-85 and RAI 8352, Question 19-10, respectively

At-power PRA-based SMA
Methodology

United States Nuclear Regulatory Commission
Protecting People and the Environment

Task 1 - Identify seismic hazard input

Task 2 - Develop seismic equipment list

Task 3 - Evaluate seismic fragility

Task 4 - Assess design-specific system and accident sequences

Task 5 - Calculate plant level HCLPF

Task 6 - Establish risk insights

At-power PRA-based SMA Methodology (continued)

United States Nuclear Regulatory Commission
Protecting People and the Environment

Task 1 - Seismic Hazard Input - No changes since Phase 2 review

Task 2 - Seismic Equipment List - Comprehensive, which includes:

- All components in the internal events PRA (powered by onsite AC)
- SSCs that are not explicitly modeled in the internal events PRA, i.e., piping, cable trays, tanks, ventilation ducts, cranes, etc.
- Structures that house the PRA and passive components and could cause widespread equipment failures, i.e., reactor containment building, auxiliary building, turbine building, compound building, CCW heat exchanger building, ESW building, emergency diesel generator building, and diesel fuel oil tank building

Task 3 - Seismic Fragility Evaluation - No changes since Phase 2

At-power PRA-based SMA Methodology (continued)

United States Nuclear Regulatory Commission
Protecting People and the Environment

Task 4 - Systems and Accident Sequences Analysis

a. Seismic-induced Initiating Events Analysis

- Assessed multiple seismic-induced initiating events
- Modeled plant transient, LOOP, loss of 4kv AC bus, SBO, loss of all instrumentation and control, main steam line break, loss of component cooling water, anticipated transient without scram
- Considered all LOCAs of various sizes
- Included supporting system failures, i.e., failures of engineered safety feature (component control system, group controller, plant protection system, etc.)

At-power PRA-based SMA
Methodology - Task 4 (continued)

United States Nuclear Regulatory Commission
Protecting People and the Environment

b. Evaluation of Safety Functions

- Evaluated all key safety functions to prevent core damage and large release, including: reactivity control, core heat removal, reactor coolant inventory control, reactor coolant heat removal, and containment integrity
- Identified front-line and support systems that are required to meet the safety functions
- No credit was taken for non-safety and non-seismic Category I systems

c. Plant Response and Mitigation Systems

- Developed plant response logics based on the at-power IEs PRA model
- Included random failures and human errors modeled in IEs PRA
- Developed ETs and FTs to generate core damage and large release sequences initiated by a seismic event, coupled with the random failures and human errors

At-power PRA-based SMA
Methodology - Task 4 (continued)

United States Nuclear Regulatory Commission
Protecting People and the Environment

d. Structure Failures

- Included seismic Category I structures that contain SSCs credited in the IEs PRA and Category II structures that could impact other structures that contain equipment credited in the PRA
- Failure of these structures was assumed to result in failure of all components in that building, and lead directly to core damage
- Identified eight structures:
  - Containment building
  - Auxiliary building
  - Turbine building
  - Compound building
  - CCW building
  - ESW building
  - Emergency diesel generator building
  - Diesel fuel oil tank building

At-power PRA-based SMA
Methodology - Task 4 (continued)

United States Nuclear Regulatory Commission
Protecting People and the Environment

e. Correlation between Seismic Failures

- Correlation factor between SSCs was assigned by either zero (fully independent) or 1.0 (fully dependent)
- Similar components located in the same building and elevation are treated as fully correlated
- DCD Table 19.1-42, “Seismic Equipment List,” shows the correlation factors for all SSCs in the SEL

At-power PRA-based SMA Methodology (continued)

United States Nuclear Regulatory Commission
Protecting People and the Environment

Task 5 - Sequence-level and Plant-level HCLPF Assessment

- Used “min-max” method for computing sequence-level and plant-level HCLPF values
- Random/human failure probabilities were assessed in combination with HCLPFs for each accident sequence
- Accident sequences are consistent with the APR1400 fragility evaluation, plant responses, and key assumptions
- No accident sequence has a HCLPF lower than 0.5g

At-power PRA-based SMA
Staff Findings

United States Nuclear Regulatory Commission
Protecting People and the Environment

- If all key assumptions remain valid and all seismic-related COL information items are fulfilled, the APR1400 design can withstand a review level earthquake of 1.67 times the CSDRS and meets the 0.5g HCLPF capacity
- The APR1400 PRA-based SMA does not support the determination of important contributors to seismic risk in a probabilistic sense; however, it allows identification of the plant features that are important to the design seismic capacity
- The APR1400 design satisfies the expectation of SECY-93-087 and the related SRM regarding seismic evaluation and capacity

At-power PRA-based SMA
Staff Findings (continued)

United States Nuclear Regulatory Commission
Protecting People and the Environment

- PRA-based SMA will continue to be updated
  - COL applicants will update the PRA-based SMA to include site features, design departures, site-specific SSCs, and soil effects
  - COL applicants will ensure that the results of APR1400 PRA-based SMA remain valid for the plant
  - COL holders will develop the seismic PRA in accordance with the requirements of 10 CFR 50.71(h)

LPSD PRA-based SMA
Background and Results

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Same methodology used for at-power PRA-based SMA
- LPSD SEL developed from the full power SEL
- Staff searched for equipment unique for LPSD
  - Open equipment hatch
  - Transport of the reactor vessel head (removal and reinstallation)
- Seismic induced LPSD initiators evaluated for bounding POSs to represent RCS intact and RCS not intact
  - POS 5 midloop with RCS vented via pressurizer manway
  - POS 3B cold shutdown with RCS intact and SGs available
- Seismic induced initiators selected from LPSD PRA
- All LPSD sequence HPCLFs are 0.5g or higher
- LPSD plant-level HPCLF is 0.5g
- LPSD seismic cutsets reflect significance of operator actions

Internal Fire and Flood PRAs
Staff Findings

United States Nuclear Regulatory Commission
Protecting People and the Environment

- Internal fire and flood PRAs are sufficiently detailed for design certification purposes and key assumptions are identified to account for incomplete design and operational aspects
- Sufficiently consistent with accepted guidance (e.g., NUREG/CR-6850, ASME/ANS PRA Standard)
- Estimated risks compare favorably with the Commission's goals and sufficient risk insights are identified to support design certification application

Internal Fire and Flood PRAs
Closure of Related Open Items

United States Nuclear Regulatory Commission
Protecting People and the Environment

- Documentation of accident sequences
- Assessment of model uncertainties
- Assessment of ex-control room action for feed and bleed operation
- CAFTA conversion

DCD Chapter 19.1 - PRA
Staff Findings

United States Nuclear Regulatory Commission
Protecting People and the Environment

The staff's review finds that APR1400 PRA:

- Is of the appropriate scope, level of detail, and technical adequacy for its identified uses at the design certification stage
- Reasonably reflects the as-designed, as-to-be-built, and as-to-be-operated plant to the extent possible
- Meets the Commission's CDF and LRF goals of less than 1E-04/yr and 1E-06/yr, respectively
- Meets the Commission's containment performance goals
- Provides sufficient risk insights to support DC application and COL stage

DCD Chapter 19.1 - PRA
Staff Findings (continued)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- APR1400 plant-level seismic capacity is sufficient for the seismic hazard represented by the CSDRS
- Seismic risk insights are logically revealed to illustrate the robustness and potential weakness of APR1400 design
- External events are fully addressed by analyzing or performing screening analysis
- Uncertainty, importance, and sensitivity studies are rationally performed to identify important equipment, operator actions, and risk insights
- PRA is reasonably used as an input to augment final design, TS, ITAAC, SA evaluation, RAP, human factors engineering, physical security, and environmental review

DCD Chapter 19.1 - PRA
Staff Conclusion

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- The staff concludes that the applicant has adequately addressed the Commission's objectives regarding the preparation and use of its PRA, in conformance with the SRM on SECY-89-102, "Implementation of the Safety Goals," and the Commission policy statement "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," 60 FR 42622, August 16, 1995

- The staff expects that the APR1400 PRA will continue to be updated, improved, and useful throughout its life, from design certification, through COL application, and until the end of the APR1400 operation, as required by 10 CFR 52.79(d) and 10 CFR 50.71(h)

ACRONYMS

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- **AC -** alternating current
- **ACRS -** Advisory Committee on Reactor Safeguards
- **ANS -** American Nuclear Society
- **ASME -** American Society of Mechanical Engineers
- **CCF -** common-cause failure
- **CCW -** component cooling water
- **CDF -** core damage frequency
- **CE -** Combustion Engineering
- **CFR -** Code of Federal Regulations
- **COL -** combined license
- **CSDRS -** certified seismic design response spectra
- **DC -** design certification
- **DCD -** design control document
- **ESW -** essential service water
- **ETs -** event trees
- **FTs -** fault trees
- **HCLPF -** high-confidence-of-low-probability-of-failure
- **HEPs -** Human error probabilities
- **I&C -** instrumentation and control
- **IEs -** internal events
- **ITAAC -** inspections, tests, analyses, and acceptance criteria
- **KEPCO -** Korea Electric Power Corporation
- **KHNP -** Korea Hydro and Nuclear Power Co.
- **LCO -** limiting condition for operation
- **LOCA -** loss-of-coolant accident
- **LOOP -** loss of offsite power
- **LPSD -** low-power and shutdown
- **LRF -** large release frequency
- **NRC -** Nuclear Regulatory Commission
- **OIs -** open items
- **POS -** plant operational state
- **PRA -** probabilistic risk assessment
- **RAI -** request for additional information
- **RAP -** reliability assurance program
- **RCP -** reactor coolant pump

ACRONYMS (continued)

United States Nuclear Regulatory Commission
Protecting People and the Environment

- **RCS -** reactor coolant system
- **SA -** severe accident
- **SBO -** station blackout
- **SDC -** shutdown cooling
- **SG -** steam generator
- **SECY -** Office of the Secretary of the Commission
- **SEL -** seismic equipment list
- **SGTR -** steam generator tube rupture
- **SMA -** seismic margin assessment
- **SOARCA -** state-of-the-Art reactor consequence analyses
- **SRM -** staff requirements memorandum
- **SRP -** Standard Review Plan
- **SRs -** supporting requirements
- **SSCs -** structures, systems, and components
- **T&M -** test and maintenance
- **TS -** technical specifications

Staff Review Team

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- **Technical Staff**
  - **Hanh Phan** (Lead), Senior Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Anne-Marie Grady**, Reactor Systems Engineer, Containment & Ventilation Branch
  - **Alissa Neuhausen**, General Engineer, Structural Engineering Branch
  - **Marie Pohida**, Senior Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Robert Roche-Rivera**, Structural Engineer, Structural Engineering Branch
  - **Jason Schaperow**, Senior Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Hanry Wagage**, Senior Reactor Engineer, Containment & Ventilation Branch
- **Project Managers**
  - **William Ward,** Lead Project Manager
  - **James Steckel**, Chapter Project Manager

Presentation Outline
Technical Areas and Presenters

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

1) MELCOR Confirmatory Analysis (Schaperow)
2) Steam Explosions (Wagage)
3) Conclusion (Wagage)

MELCOR Confirmatory Analysis

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Objective – confirm applicant's use of MAAP for PRA and severe accident analysis
- Approach – perform independent analysis for select scenarios using MELCOR and compare with MAAP results
- APR1400 Phase 2 conclusion
  - At-power accidents – differences seen between MELCOR and MAAP results not likely to impact LRF, SAMDA analysis, CET quantification
  - Shutdown accidents – confirmatory analysis led to identification of issues with the MAAP analysis

MELCOR Confirmatory Analysis (continued)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- APR1400 Phase 4
  - The applicant revised its MAAP analysis for shutdown accidents to address issues identified in Phase 2 and updated the FSAR
    - More realistic modeling of RCS and containment configuration
    - MAAP code bug
- Staff concluded that the applicant's revised MAAP analysis addressed the issues raised

Steam Explosions

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Reviewed and accepted the applicant's justification for the chosen cross-sectional area of the pool used in TEXAS-V calculations for in-vessel steam explosions
- Reviewed and accepted the applicant's response on ex-vessel steam explosions:
  - Cross-sectional area of the pool used for TEXAS-V calculations
  - Calculation of impulse on cavity structures using TEXAS-V results
  - Structural evaluation of cavity structures
  - Impact of IVR/ERVC operation of steam explosion loading

DCD Chapter 19.2 – Severe Accident Evaluation Conclusion

United States Nuclear Regulatory Commission
Protecting People and the Environment

Based on its review, the staff finds that the APR1400 design provides features for the prevention and mitigation of severe accidents and is in compliance with 10 CFR 52.47(a)(8) and (23) and in conformance with SECY-90-016, SECY-93-087, and SRP Section 19.0.

ACRONYMS

United States Nuclear Regulatory Commission
Protecting People and the Environment

- **ACRS -** Advisory Committee on Reactor Safeguards
- **CET -** containment event tree
- **CFR -** Code of Federal Regulations
- **DCA -** design certification application
- **ERVC -** external reactor vessel cooling
- **IVR -** in-vessel retention
- **KEPCO -** Korea Electric Power Corporation
- **KHNP -** Korea Hydro and Nuclear Power Co.
- **LRF -** large release frequency
- **MAAP -** modular accident analysis program
- **PRA -** probabilistic risk assessment
- **SA -** severe accident
- **SAMDA -** severe accident mitigation design alternatives
- **SECY -** Office of the Secretary of the Commission
- **SRP -** Standard Review Plan