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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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APR1400 SUBCOMMITTEE

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WEDNESDAY

APRIL 19, 2017

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ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

RONALD G. BALLINGER, Chairman

DENNIS C. BLEY, Member

MICHAEL L. CORRADINI, Member

WALTER L. KIRCHNER, Member

JOSE MARCH-LEUBA, Member

DANA A. POWERS, Member

JOY REMPE, Member

GORDON R. SKILLMAN, Member

JOHN W. STETKAR, Member

MATTHEW W. SUNSERI, Member

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER BROWN

CHRISTIANA LUI

ALSO PRESENT:

ROSS ANDERSON, ENERCON

AARON ARMSTRONG, NRO

ODUNAYO AYEGBUSI, NRO

JEFF CIOCCO, NRO

RAYMOND DREMEL, ENERCON

STEVE D. FLOYD, Jensen Hughes

ANNE-MARIE GRADY, NRO

GARY W. HAYNER, Jensen Hughes

SUN HEO, KHNP

SEOKHWAN HUR, KEPCO E&C

KYUHO HWANG, SGH

TAEHEE HWANG, KEPCO E&C

YOUNG H. IN, ENERCON

BYUNG JO KIM, KEPCO E&C

JAE GAB KIM, KEPCO E&C

JEFF LEARY, ENERCON

DONGWON LEE, KEPCO E&C

ROBERT LICHTENSTEIN, ENERCON

JAESOO LIM, KHNP

MARK LINTZ, NRO

MICHAEL MCCOPPIN, NRO

JILL MONAHAN, Westinghouse

LYNN MROWCA, NRO

TONY NAKANISHI, NRO

ALISSA NEUHAUSEN, NRO

DAE GEUN OH, KEPCO E&C

JIYONG OH, KHNP

CHAN Y. PAIK, FAI

CHAN EOK PARK, KEPCO E&C

CHANG SUN PARK, KEPCO E&C

HANH PHAN, NRO

STEVE PHILLIPPI, ENERCON

MARIE POHIDA, NRO

ROBERT ROCHE-RIVERA, NRO

JAMES ROSS, AECOM

GREGORY ROZGA, ENERCON

IN CHUL RYU, KEPCO E&C

SUJIT SAMADDAR, NRO

JASON SCHAPEROW, NRO

RAY SCHNEIDER, Westinghouse

ROB SISK, Westinghouse

COURTNEY ST. PETERS, NRO

JAMES STECKEL, NRO

VAUGHN THOMAS, NRO

ANDREA VEIL, Executive Director, ACRS

HANRY WAGAGE, NRO

*Present via telephone

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P R O C E E D I N G S

8:31 a.m.

CHAIRMAN 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 of the APR1400 Subcommittee.

ACRS Members in attendance are Mike Corradini, Gordon Skillman, Dana Powers, Matt Sunseri, Dennis Bley, John Stetkar, Jose March-Leuba, Walt Kirchner and Joy Rempe.

The purpose of today=s meeting is for the Subcommittee to receive briefings from Korea Electric Power Corporation and Korea Hydro and Nuclear Power Company regarding their design certification application and the NRC staff regarding their Safety Evaluation Report with open items specific to Chapter 17, Quality Assurance and Reliability Assurance in 19 Probabilistic Risk Assessment and Severe Accident Evaluation.

The ACRS was established by statute and is governed 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.

That said, we also set aside ten minutes for spur of the moment from members of the public attending or listening to our meetings.

Written comments are also welcome.

The ACRS Section of the USNRC 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 Wednesday, April 12, 2017. The meeting was announced as open/closed to the public meeting.

And, I=m reminded that during the presentation today, there=s a session that=s labeled as closed for the end of the day. But, if the questioning that goes on today suddenly gets into something which is proprietary, you=ll need to let us know so that we can take some action.

The meeting is announced in open public -- this means that the Chairman can close the meeting as needed to protect SRI and information proprietary to KHNP and its vendors.

No requests for making a statement to the 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 room and remember to push the button and make it green when addressing the Subcommittee.

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. The bridge number and password were published in the agenda posted on the NRC public website.

To minimize disturbance, this public line will be kept in the listen only mode. The public will have an opportunity to make a statement or provide comments at a designated time towards the end of the meeting.

Request meeting attendees and participants silence their cell phones and electronic devices.

Also, I=ve been reminded on a number of occasions that there are a number -- lots of slides, a lot to go through. And so, and I thought I was going to be issued a stun gun to keep people in order. But, keep that in mind, although we really need to have a full discussion.

MEMBER POWERS: Now, let me understand, we=re supposed to have a full and complete discussion, but not take too much time?

CHAIRMAN BALLINGER: You=ve got it right.

(Laughter.)

MEMBER STETKAR: You have your thoughts well formulated, just speak every third word.

(Laughter.)

CHAIRMAN BALLINGER: Chris Brown and Christiana Lui are the Federal -- Designated Federal Officials and they do have a stun gun.

So, let=s see, there was something else I was supposed to be reminded of.

Oh, with regard to the slides, the problem is that I have a reputation at MIT for using the most slides for anybody. So, it=s like the pot calling the kettle black.

So, I=ll turn the meeting over the Jeff.

MR. CIOCCO: Yes, thank you.

Good morning, my name is Jeff Ciocco. I=m the Lead Project Manager for the APR1400 Standard Design Certification Project. Thank you for having us back to present and defend our Safety Evaluations over these two days on Chapter 17, 19, 19.3, 19.4 and 19.5.

We will have staff and management in attendance to present and respond to questions.

Thank you, and we=re ready to get on with it.

CHAIRMAN BALLINGER: And the floor is yours.

MR. SISK: And, again, thank you very much, we look forward to a good discussion.

I do want to echo the comment, we do have a large amount of material to cover. Chapter 19 is a very busy chapter, so without any undue delay, I=m going to turn it over to Mr. Young In and he=ll get us started off on 19.1.

MR. IN: Thank you.

My name is Young In from Enercon and I will be providing the general overview of the Chapter 19.1, basically a PRA presentation.

We have five gentlemen who=s going to be presenting the subparts of the Chapter 19.1 and a few of the names associated with each key topics, in the first section will be -- the first session in the morning will be covered by Mr. Greg Rozga and Mr. Taehee Hwang.

And, then, hopefully, I don=t know if we can cover the third topic which is beyond seismic that would be Mr. Lee Dongwon.

And then, the rest of the presentation will be covered in the later part -- second session of the morning.

And then, the next slide shows the presentation that=s going to be given this afternoon and tomorrow. So, these are on the 19.3 and 19.3 through 19.5.

The section overview of the 19.1 is basically taken after the template provided in the Reg Guide 1.46. It=s very similar, basically, it=s the same order and we=ll cover each topic throughout the morning.

The PRA scope for the APR1400 is basically a full scope amount except the seismic portion which is a PRA based SMA.

The only exception here is that the internal flooding for the Level 2 was, you know, bounded by the Level 1 because, you know, such a, you know, low power in the CDR.

The basic methodology is -- and the tools, you know, that were utilized to perform the PRA, it=s basic methodology, you know, small event tree and large fault tree approaches, linked fall tree method.

The computer tools that we used is SAREX, FTREX, CAFTA and the HRA Calculator and MAAP, RELAP and MACCS.

And, the HRA Calculator was used, you know, parts of the HRA determination in the second -- played a place in the Phase 1 in the PRA, mainly for the shutdown PRA and now, we are updating that with the 27 update will be using the HRA Calculator.

And then, the -- these next slides -- the next four slides are really the conclusion portion of the 19.1.

MEMBER STETKAR: You just mentioned the word update. Can you tell us what you=re doing from the version of the PRA that=s in Rev 0 of the DCD compared to what you mean by update?

MR. IN: Yes.

MEMBER STETKAR: I know there=s a Rev 1 of the DCD coming out or is out.

MR. IN: Yes, basically, the Rev 1 of the DCD that=s coming out is, you know, results of the RAI responses, you know, that we=ve been going through.

And, the -- it=s, you know, basically, you know, all the markups, you know, that we had so far and that incorporates into one clean document. That=s the Rev 1 of the DCD that=s being submitted to the NRC.

The 2017 update, PRA update, there=s, you know, actually, update of the PRA model and that includes, you know, any design changes, you know, that occurred, you know, during the last -- during the Phase 2 review, you know, to the RAI responses.

And, we accumulated all that and then the also any RAI responses, you know, that we had in the

Chapter 19.1 that impacts the model, we are updating all that.

And then, the -- all the findings, you know, that we had, you know, from the peer review, that=s going into this update.

MEMBER STETKAR: Okay, so that update will be documented in some future revision of the DCD, is that --

MR. IN: Correct.

MEMBER STETKAR: Two or later?

MR. IN: Yes, hopefully two.

MEMBER STETKAR: Hopefully two, okay, thank you.

MEMBER REMPE: Before you leave this slide, you have here like you have MAAP and RELAP. Didn=t we hear a couple weeks ago from the staff that RELAP5/MOD3 is not really an accepted NRC code? And that was something that you had actually had in your write-up that it was.

And, do you remember what I=m talking about with the folks from the --

MEMBER CORRADINI: You=re talking to me?

MEMBER REMPE: Yes, the South Texas project?

So, this is more of a staff question, but --

MEMBER CORRADINI: But, I --

MEMBER REMPE: What is the staff=s --

MEMBER CORRADINI: I=m guessing the staff is going to tell you they=re going to evaluate this as the user using a tool, not as a generic blessing of the tool.

MEMBER REMPE: Okay. But, again, in the write-up, it had said that it was an NRC accepted code.

MEMBER CORRADINI: Oh, right.

MEMBER REMPE: And so, that was something that I was questionable. I can look for the reference on that, but it was something in your write-up.

The other thing was, could you talk a little bit about your philosophy for when you used RELAP versus when you used MAAP?

MR. IN: Yes, basically, RELAP for PRA was used, you know, to compliment the MAAP code, you know, to determine the success break criterial.

MEMBER REMPE: Okay.

MR. IN: Because, you know, MAAP, you know, on certain cases like, you know, large LOCA, you know, MAAP code is not sufficient in detail. So, you know, we used, you know, RELAP to do that.

MEMBER REMPE: Did you ever do any comparisons where you would do a similar run and say, yes, MAAP could do the thermal hydraulics fine until we got to the core damage? Did you ever look at the water level decrease, for example, and say, yes, they give the same values?

MR. IN: Mr. Hwang?

MR. T. HWANG: Yes --

MEMBER CORRADINI: This green light on, sir.

MR. T. HWANG: Yes, we have found the success correctly analysis using RELAP code per the logical sequences and some of the low power shutdown sequences.

But, basically, we used the MAAP code for the other LOCA sequences large LOCA and again with small LOCA and other transients or sequences.

And, sometimes we compared the reset for -- compared the reset for RELAP code using the RELAP and MAAP code. But, the success created for core damage is a little bit different because the RELAP code is a detailed code. So, we complied the core damage as a higher than 2,200 Fahrenheit degrees and MAAP code is a little simplified code. So we complied the core damage as 1,800 Fahrenheit degrees.

MEMBER REMPE: Thank you.

MEMBER MARCH-LEUBA: Just a modification, when -- in the regulatory basis, when a code is said approved, it means approved to be -- to do calculations referred to technical specifications.

So, it=s approved for -- it=s called approved for reference because then your technical specs can refer to it and you can only reference it if it=s approved.

You can use any code you want for engineering calculations that --

MEMBER CORRADINI: As long as staff reviews --

MEMBER MARCH-LEUBA: That calculation for --

MEMBER CORRADINI: -- and how it=s used and verified that the user is bona fide.

MEMBER MARCH-LEUBA: It=s perfectly acceptable to use a Microsoft Excel worksheet to do calculations. It=s not acceptable to do a Microsoft

Excel worksheet for set point calculations in tech specs.

MR. IN: Can we move on?

CHAIRMAN BALLINGER: Yes.

MR. IN: Okay, next full slide is, you know, really a conclusion part of the 19.1. I can cover this here now or I can cover it at the end of the -- after the other presentation is done.

MEMBER CORRADINI: So, since I=m sure Member Stetkar has lots of questions, I want to ask a delta question.

If I looked at CE80+ and I looked at this since this, with all due respect, is a derivative of CE80+, what=s the delta change? Can you at least identify the delta change from what was CE80+ and that estimate and these?

MR. IN: No, sorry.

MEMBER CORRADINI: Okay.

MR. IN: We haven=t done that comparison. Because, basically, APR1400 has a lot more redundancy range for --

MEMBER CORRADINI: So, you=d expect it to be different in many places?

MR. IN: Yes, it would be different in so many different places.

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

MR. IN: So, I=ll just, you know, go ahead with this in four slides and if you think it needs to be covered later, you know, we=ll come back to it.

So, basically, for the PRA applications or risk applications defined in the 19.1 sub and basically, there are at the COLA stage, there are two programs, you know, that has a major input to the SOR program. And then the severe accident management design on the SAMDA and that goes into the environmental report.

For the COR stage, the PRA will support there reactor oversight program and the MSPI, SDP and so forth and the maintenance role.

For the design improvements on the risk insights, basically, the APR1400, for the design certification as a reference plant which is section 43 and 4.

And, from there, we made some design improvements. And, basically, these are the -- there are five, but, you know, some may call it four.

Basically, the number of diesels went from two diesels to four diesels, one per each, and then to break up the common cause, you know, potential between EDG and the AAC, we made the -- we changed the AAC from a diesel generator to be a gas turbine generator.

And then, also to reduce the contribution from the SBO sequences, we extended the battery capacity from -- for the 125 volt DC and basically, it went from the -- the one that=s, you know, critical to the PRA is to be C&D and that went from 2 hours to 16 hours without the load sharing.

MEMBER CORRADINI: Again, I=m looking for delta. So, if I go to Shin Kori, where is Shin Kori in comparison? Is it six, five? I can=t remember which one?

MR. IN: Three and four.

MEMBER CORRADINI: Three and four? Are they two diesel generators, the AAC is a diesel generator and this is a change in the --

MR. IN: Yes.

MEMBER CORRADINI: Okay, all right. So, that was one question.

The same question is, is the accumulator the fluidic device and it=s lengthening of the -- of its operational time allow one to do the second bullet? Is that the reason you can do the second bullet?

MR. IN: No. The second bullet was, you know, basically, Shin Kori 3 and 4 has, you know, a common cause between the AAC and the EDG.

MEMBER CORRADINI: A common --

MR. IN: Common cause failure.

MEMBER CORRADINI: Oh, okay.

MR. IN: And, basically, that comes out to be one of the top causes for the SPOA sequences.

MEMBER CORRADINI: Okay. And, I=m sure Member Stetkar knows all this, but just in case.

MR. IN: And, basically, because, you know, these, you know, three improvements they are SBO LOOP and the SBO contribution from referenced plant which was about 60 percent, went down to, you know, about 30 percent -- 36 percent.

MEMBER CORRADINI: Okay, thank you.

MR. IN: And, the other design improvements that we made was that we made the changes to the Tech Spec 3.67 which is related to the equipment hatch closure in the modified.

And, the last one is the -- from the prior PRA we identified the cables for 75 components to be protected.

MEMBER KIRCHNER: May I ask a question? How early did you first do your PRA to decide to make these changes? Because these changes like the first two bullets I think have been in the DCD since Tier 1, right?

MR. IN: Tier 2.

MEMBER KIRCHNER: Tier 2?

MR. IN: Yes.

MEMBER KIRCHNER: Okay.

MR. IN: Yes, so --

MEMBER KIRCHNER: So, you did the PRA and then you went -- you looped back and you actually physically changed the design?

MR. IN: Actually, we -- when we started --

MEMBER KIRCHNER: Or did you do a scoping PRA very early on?

MR. IN: Yes, we did the scoping PRA very early and we looked at the, you know, risk profile from the reference plant. And, that=s when it started the process.

MEMBER KIRCHNER: Okay.

MR. IN: The overall results shown here, I=m not going to read the -- each individual number, but there are -- for the CDF, you know, which is, you know, from results for the Level 1, they=re all less -- around the low E-06 and the sum adds up to be about 7.8, 7.9 E-06.

And, the Level 2 is -- the results shown here is in a large release frequency. And they are all low E-07 range. The total comes out to be 5.5 E-07.

And, this overall CDF profile shows by the operational modes and the hazard, related hazard, and the largest, you know, contribution comes from the shutdown internal events which is, you know, 35 percent.

And, the second one is at power internal fire which is about 24 percent.

MEMBER KIRCHNER: So, I have another question. So, is this good? It is good that the wheel has essentially equal distribution from all the things versus one dominating?

I mean, I would think it=s good, but --

MR. IN: Yes.

MEMBER KIRCHNER: -- am I wrong?

MR. IN: Yes, it would be good. Now, it would be good to have them wholly balanced, you know, profile. Here, you know, I was going to say that the shutdown internal events which is, you know, mainly by the operation actions.

MEMBER KIRCHNER: Operators, yes.

MR. IN: It=s somewhat conservative right now at the design stage. Because, we don=t have, you know, all the procedures, you know, written down.

And, for the at power internal fire, there is, you know, somewhat conservative assumptions in there and, you know, Mr. Rozga will discuss that, you know, when the internal fire presentation.

But, yes, there are some new conservatisms in the internal fire as well.

MEMBER KIRCHNER: I=m a little surprised that the low power shutdown is such a big sector. So, is that being driven mainly by operator error?

MR. IN: Yes.

MEMBER KIRCHNER: Human reliability issues and so on? Because that=s rather large.

So, that becomes a COL task to mitigate that, to reduce that risk.

MR. IN: Yes, there are several items to make sure that the shutdown program is, you know, finalized and then that they will have to, you know, redo the shutdown PRA.

MEMBER KIRCHNER: So, what were the major contributors there?

MR. IN: We have a presentation on this.

MEMBER KIRCHNER: You=re going to get to that?

MR. IN: Yes.

MEMBER KIRCHNER: Thank you.

MR. IN: Yes.

So, that concludes the overview of the 19.1 which is, you know, PRA. And, the next, we=re going to the -- each subsection.

And, the first one is the at power internal events Level 1 which will be presented by Mr. Greg Rozga.

MR. ROZGA: Good morning, everyone. I=m Greg Rozga from Enercon supporting KHNP and KEPCO and the fire PRA and internal events PRA.

Am I speaking loud enough? Okay, thank you.

The first step in any PRA is the

initiating events analysis and, generally, it=s -- there=s three main steps.

First, we identify all potential initiators that could occur and we look at various industry generic sources.

We also do a failure modes and effects analysis on all the individual systems of the plant to see if failure of that system would result in a unique initiator that=s not in one of the industry generic sources.

Those initiators are then grouped together based on similarity of the initiator impacts on the core protection functions, common accident sequence progression and common success criteria.

And then, finally, the initiating event frequencies, they=re calculated based on generic industry data and we assumed a 95 percent capacity factor for the design certification.

MEMBER SKILLMAN: Greg, how do you know there are no major orphans when you do this review and account for these events? How do you know that you haven=t --

MR. ROZGA: That we haven=t missed anything?

MEMBER SKILLMAN: -- missed any that are significant.

MR. ROZGA: That=s the intent of the FMEA is to look all the individual systems and see a failure of those systems would cause something that isn=t already in the list of the various generic industry sources.

MEMBER SKILLMAN: Thank you.

CHAIRMAN BALLINGER: Would this be picked up on the peer review as well?

MR. ROZGA: Correct, correct.

And, there was a peer review done.

CHAIRMAN BALLINGER: Did they find anything?

MR. ROZGA: Of course, yes.

The major list of initiating events are -- we have various different sized LOCAs, tube rupture, the LOCAs include RCP, seal LOCAs both from random events as well as from system failures, failure of all seal cooling injection.

There=s a variety of transients, general transient, loss of secondary site cooling, secondary site steam and feedwater pipe breaks, loss of support systems, loss of DC, loss of instrument error, et cetera, loss of offsite power events.

And then, there=s also category, I term induced initiators, are not real initiating events, but they are initiators that are induced after the initial initiator, but because these specific failures significantly change the accident sequence progression, we create a separate event treating separate accident sequence for those.

And, those include things like ATWS, station blackout, a stuck open POSRV LOCA. In that case, it looks just like the regular LOCA, but it would start and maybe you have a general transient and then you have a stuck open POSRV so then we need to transfer over into a LOCA tree.

MEMBER STETKAR: Greg?

MR. ROZGA: Yes?

MEMBER STETKAR: It struck me that this list is, number one, completely in lock step with NUREG/CR-6928 which it always bothers me.

And, number two, that it=s notably lacking support system initiating events other than the ones that are in NUREG/CR-6928.

It also bothered me that the identification and grouping of the initiating events was not documented in the DCA, recognizing you can=t document everything in the DC.

So, things like, for example, failure of the main transformer, how is that modeled? Is that a general transient?

MR. ROZGA: Yes, yes.

MEMBER STETKAR: It strikes me that no power from the main transformer is different than a reactor trip on a sunny day. So, why isn=t failure of the main transformer a separate initiating event?

Why isn=t failure of the unit auxiliary transformers separate initiating event?

What -- where is the process that systematically shows me that I went through every electrical, fluid and ventilation system and allocated them to an initiating event category?

Because they all must be in that general transient case and it can=t be.

MR. ROZGA: Well, yes. Some may be a subset of a loss of offsite power. So you could -- you would say the loss of a UAT is a --

MEMBER STETKAR: Probably not because you took the frequencies from NUREG/CR-6928 and I don=t know what those frequencies came from.

MR. ROZGA: Yes.

MEMBER STETKAR: They=re generic frequencies for a generic site and a generic plant, not your plant.

MR. ROZGA: Yes. In the initiating event notebook, there is detail of the -- there are several hundred events that were looked at and it=s documented in there.

MEMBER STETKAR: It=s just striking that -- for a plant, things that I=ve found, for example, I know, and I have to be careful about comments here because, obviously, I have a lot, you can lose locally part of your component cooling water system, not the initiating event PLO CCW.

You can locally lose part of your component cooling water system that will give you a plant trip and put you in jeopardy of a seal LOCA and the frequency of that may be comparable to the total frequency that you=ve used for partial loss of component cooling water.

Yet, you=ve not identified that as an initiating event.

MR. ROZGA: Well, we have partial loss of CCW, partial --

MEMBER STETKAR: Yes, you do. My -- that partial loss of CCW, I don=t -- I know where the frequency came from. You=re using it from NUREG/CR-6928 which, again, is a generic plant.

I don=t know how many component cooling water trains or pumps or pipes or valves a generic component cooling water system has because they=re all different.

And, I know I read in the DC that you did site specific fault tree analyses, but you didn=t use those for the initiating event --

MR. ROZGA: Right. And the --

MEMBER STETKAR: -- frequencies.

MR. ROZGA: The numbers in 6928 were slightly more conservative than the numbers from the fault trees.

MEMBER STETKAR: Oh, okay.

I=m just curious that the -- given what I know about the design, the lack of design specific kind of goes to what Dick was saying. The lack of design specific support system initiating events seems striking.

And, I=ll ask the staff about that when they come up as far as how much of an audit they did of the initiating events.

And, in the interest of time, I=ll be quiet now.

MR. ROZGA: Next slide, please?

All right, once we have the initiators, the next thing we do for each one of those initiating events is we define the accident sequence analysis and these are modeled in the form of event trees.

And, the event trees model, the accident progression and the manner consistent with the plant design, operating procedures and expected plant response, thermal hydraulic analyses are used to determine the systemic success criteria for each branch in the event tree.

And, fault trees are used to model the mitigating system failure probabilities with respect to those success criteria.

And, the fault trees include both equipment failures as well as human failure events leading to the individual system failures.

And, since the fault threes are directly linked to the event tree branches, the inner system and sequence dependencies are inherently considered.

MEMBER BLEY: Greg?

MR. ROZGA: Yes, sir?

MEMBER BLEY: I=m going to follow up on what Mr. Stetkar asked you before.

This is a design cert PRA and I know here the requirements are a little different than for COL and certainly different from prior to fuel load PRA.

Should there actually be one of these built in the United States someday, at what point do you think it would be necessary to move from generic initiating event frequencies with what might be conservative or might be pessimistic, depending on how the real design turns out to be generic initiating events and groupings, when would that turn into design specific evaluation of initiating event groups and frequencies?

MR. ROZGA: In general, once the system design is more 100 percent complete. One of the service water and CCW system are some of the systems that are all tied to the main power block. They have their own buildings and those --

In fact, in the PRA update that we=re doing now, there have been some changes to those systems. And, those are some of the reasons why, when we initially did the support state initiating event fault trees, we ended up going with the generic data because we knew that those system designs weren=t completely locked down.

I will say that even existing nuclear plants use the generic data for their initiating event frequencies. However, they do update it with plant specific data.

And so, some things like, you know, LOCAs, tube ruptures, various other systemic initiators, some will -- well, even at the end of the design certification, we=ll use the generic data when those secondary support systems are -- when the design is finalized then those will likely use support state initiating event fault trees which will be a more accurate representation.

But, we felt it was, at this time, okay to use the generic data for those support state initiators because the numbers were -- they were close. They were a little bit greater than so we didn=t think that we were losing anything.

MEMBER BLEY: I=m not on this Subcommittee and I admit I haven=t fully read everything in detail. Do you make that at all clear in the Chapter 19?

MR. ROZGA: Make it clear when we -- I think we identify in Chapter 19 that we=re using the generic data.

MEMBER BLEY: Without regard to the design details of the system?

MR. ROZGA: Yes, for those support state initiating events, yes.

MEMBER BLEY: I=ll have to look because I didn=t see that in the quick look.

MR. ROZGA: Yes, I think it=s in a table, but don=t quote me. But, I believe there=s a footnote in the table that says that that=s where they come from.

MEMBER BLEY: Okay, thanks.

MR. ROZGA: You=re welcome.

I think part of this slide, Mr. Hwang already talked about with Joy=s question, but, ultimately the accident sequences and the success criteria analysis that=s done is for the Level 1 analysis is based on preventing, trying to prevent core damage.

And, the core damage criteria that we use is consistent with supporting requirement SEA 2 of the ASME standard as endorsed by the NRC and Reg Guide

1-200.

And, there=s the 2,200 degrees for RELAP and 1,800 for MAAP.

And, we already kind of discussed earlier, you know, where those codes are used.

Before we go any further, there=s some key PRA assumptions that feed into some of the system modeling, some of the event tree modeling. And, there was no good place to put it, so I just kind of put it here and we can go through them and I=m sure John will have many questions on this slide.

As much of all the models for the fire, flooding and seismic, as much as we had, we used the APR1400 design information.

The reference plant, Shin Kori 3 and 4, we used that design information when the design information was not available.

The digital I&C system, that one we specifically are currently using, the hardware model from Shin Kori 3 and 4.

Medium LOCA, our medium LOCA is the two to six inch range and the design basis small LOCA goes up to nine inches. So, therefore, we don=t require hot leg injection for medium LOCA. We only require hot leg injection for large LOCA.

At the time, the Rev 0 DCD was developed. The model supporting that, the RCP seal LOCA testing NW CAP was not finalized at that time. So, we used some engineering judgment. We do now have that W CAP and we are doing a detailed RCP seal LOCA analysis in the model.

But, what we have currently in the model is there is a generic number from 6928 for just a random RCP seal LOCA. You=re running fine and all of a sudden the seal just fails.

And, the size of those breaks fall within the small LOCA so we include that frequency within the small LOCA frequency and we evaluate it within the small LOCA event tree.

The seal LOCAs caused by post-trip loss of seal injection and thermal barrier cooling, that is based on engineering judgment, just based on experience with other seal LOCA models.

We have 1E-3 failure rate per reactor seal and that includes the operator action to trip the reactors, if they need to.

Or, I=m sorry, the reactor coolant pumps, if they need to.

And, the actual failure of the seals themselves.

For GSI-191, APR1400 is classified as a low fiber plant. And, therefore, although we do model some plugging, we don=t model the chemical effects.

CHAIRMAN BALLINGER: The Shin Kori 3 and 4 have a very different digital I&C system than APR1400, is that correct? Is it different -- it=s a COMMON Q, right? Is Shin Kori --

MR. ROZGA: Yes, the Shin Kori --

CHAIRMAN BALLINGER: -- 3 and 4, are the COMMON Q as well?

MR. IN: Yes.

CHAIRMAN BALLINGER: Oh, okay. I thought they were different.

MEMBER STETKAR: Greg, let=s -- reactor coolant pump seal LOCAs, you said that you=re -- that=s one of the changes, I think you said, that you=re making to the model based on --

MR. ROZGA: The PRA update based on the W CAP.

MEMBER STETKAR: Our PRA Subcommittee has been briefed rather recently on a Westinghouse model for their Generation 3 shutdown seals. And, I have to be careful because that is all proprietary and, although Rob has Westinghouse, I don=t know who has what information and this is a public meeting, so I can=t say very much.

We=ve not seen the W CAP that you=re referring to, nor do we know anything about your particular seal design.

The 10 to the minus 3 conditional failure probability per seal after loss of all cooling strikes me as numerically somewhat optimistic compared to many other models for other types of pumps and seals that I=ve seen.

So, I=m hoping that, at some time, we=ll be able to look at the W CAP report and the basis for whatever changes you=re making to your models.

MR. ROZGA: Yes, the --

MEMBER STETKAR: I=ll just leave it at that.

MR. ROZGA: Yes, the engineering judgment also, it was based on preliminary information. But, the W CAP wasn=t complete. So, it says in one of the older seal designs that did have higher conditional failure probabilities.

MEMBER STETKAR: Let me -- I don=t know where to -- at the end, I=m going to let you get through -- let me just be quiet. I=ll let you get through all of 19.1.4.1 because I have several comments to make on event modeling and it=s better to just let you get through the end and then come back to the discussion.

MEMBER KIRCHNER: Greg, may I back up one slide --

MR. ROZGA: Sure.

MEMBER KIRCHNER: -- or two?

Just, would you familiarize me with how you apply that success criteria? I don=t know the PRA standard firsthand. So, you have peak temperatures, how do you -- is it an on/off success criteria?

MR. ROZGA: Correct, there=s --

MEMBER KIRCHNER: Or is there a band. Say you=re running the temperature up to 2,150 degrees or 2,1999, how is that success criteria applied?

MR. ROZGA: Yes, well, there=s several success criteria runs that are made to support. And, you change your inputs, basically change the equipment that you=re relying on and you increase the number.

And, we tried to ensure that you are at a safe, stable state that we=re not at 1,201 and still increasing by the 24 hours. We don=t stop at that point and say, well, we=re good, because we know that we won=t be.

MEMBER KIRCHNER: Maybe I didn=t phrase my question very well.

How much uncertainty band is applied to the RELAP and MAAP calculations? In other words, you come up to some thermal limit, regardless of how the systems function or not and so on.

When do you say you=ve tripped? You don=t succeed? Is it just on/off at 2,200?

MR. ROZGA: Yes.

MEMBER KIRCHNER: Interesting. Okay, thank you.

MEMBER POWERS: I mean, I don=t think that=s -- I mean, I think that=s the way the criterion is supposed to be used. That there is a margin inherent in the 2,200 that is claimed by the NRC and doesn=t make available -- so they didn=t have -- I mean, it=s okay for him to take --

MEMBER KIRCHNER: I understand, I was

just thinking through the uncertainty in the actual analysis.

MEMBER POWERS: Yes, I mean, that=s a very legitimate concern you make. But, I think that the presumption has always been that the NRC=s already built a margin into that number.

Now, I presume if I came in with a really, really awful code that vastly under predicted temperatures and things like that, that that would get flagged in the process.

MEMBER KIRCHNER: All right, yes. Thank you.

MEMBER MARCH-LEUBA: So, let me ask the question a different way.

Is this a best estimate calculation or best estimate plus uncertainty calculation?

MR. ROZGA: Best estimate.

MEMBER MARCH-LEUBA: Okay. Not -- okay.

MEMBER KIRCHNER: Okay, thank you.

MR. ROZGA: All right.

The systems modeled in the PRA, there is several front line systems. You have your secondary side cooling water systems, your safety injection systems, CBCS, reactor protection system.

And then, all those systems that support the continued operation of your front line systems, all your cooling water systems, HVAC, chillers, instrument error, SFAS, et cetera.

MEMBER CORRADINI: So, can -- let me ask a question at this point.

So, after -- if you were to look at this PRA before Fukushima and after Fukushima, would you have -- do you operate or assume operation of the aux feed differently?

In other words, a takeaway from Fukushima is, gee, I=d like to find a way to always use aux feed as long as it=s available come hell or high water.

Is anything changed in the emergency operating procedures or have you assumed operation actions that essentially try to optimize the use of aux feed so that I don=t get into a, which I=m sure Dr. Rempe will ask a high dry low.

MR. ROZGA: With respect to the operating procedures, that I don=t know. I do know that there are guidelines for refilling the aux feed water storage tanks to extend the life of aux feed water. But, the details of the operator action, I don=t know.

Young, do you?

MR. IN: Yes, my name is Young In.

Basically, all the lessons learned, you know, from the Fukushima accident, it=s covered in the 19.3.

MEMBER CORRADINI: Oh.

MR. IN: But, yes, 19.1 does not go into the extended condition.

MEMBER CORRADINI: Okay. But, okay. But, that, in some sense, is changes. What I=m trying to get -- what I=m kind of searching for is, is there some assumptions about a change in how you try to use aux feed in the PRA or is -- if I was here in 2017 and we were doing this whole thing in 2010, I would still assume the same sorts of things in how aux feed behaves and how I use it as I get beyond the design basis, that=s what I=m trying to ask.

MR. IN: Correct, it=s in the PRA space because --

MEMBER CORRADINI: It doesn=t matter where we are compared to Fukushima on how you do the base PRA?

MR. IN: Yes, we haven=t integrated the, you know, Fukushima lessons learned in the, for instance, in -- because, you know, this is, you know, design certification stage. We didn=t integrate any of the operating guidelines such as, you know, FSG.

MEMBER CORRADINI: Okay.

MR. IN: So, those went in separately and analyzed in the 19.3.

MEMBER CORRADINI: Okay. But, you understand my question?

MR. IN: Yes, we understand.

MEMBER CORRADINI: Okay, thank you.

MEMBER SKILLMAN: Greg, a question, please.

For your front line systems, just two as an example, aux feed and main feed, how are the, for instance, lubricating oil systems for those pumps addressed in the PRA?

MR. ROZGA: The lubricating system for individual pumps is within the component boundary of the pump with the exception of the external cooling water systems that would cool the lube oil coolers, say for instance.

And, that=s a function of the data analysis when the data is collected on the pump if the pump fails because of the lubricating system failure or if it fails because of a bearing wipe of if it fails for whatever reason, that=s all included within the boundary of the individual component.

MEMBER SKILLMAN: Thank you.

MEMBER STETKAR: Greg, just one question.

MR. ROZGA: Yes, sir?

MEMBER STETKAR: And, front line systems, you didn=t list ECSBS, I got it right. I always -- and it=s kind of in there for Level 1, it=s kind of in there for Level 2.

MR. ROZGA: We don=t credit it in Level 1.

MEMBER STETKAR: Oh, okay.

MR. ROZGA: We have it in there as a sensitivity.

MEMBER STETKAR: Okay, okay. But, it is modeled explicitly for the --

MR. ROZGA: Yes.

MEMBER STETKAR: -- benefit of the rest of the Committee, ECSBS is, indeed, a FLEX system. It=s an external emergency containment spray backup system. And, it=s -- it looks like a FLEX system and it is explicitly modeled in the PRA.

MR. ROZGA: And, for the Level 1 PRA,

there is a flag that=s set to trigger that. That fails it for the Level 1 analysis. So, we do -- it=s explicitly in the model, but we do not credit it --

MEMBER STETKAR: It=s in the model? It=s just --

MR. ROZGA: -- on Level 1, though.

MEMBER STETKAR: -- this just says systems modeled in the PRA and it=s --

MR. ROZGA: Oh, okay.

MEMBER STETKAR: -- in the PRA.

MR. ROZGA: Yes.

MEMBER STETKAR: It just isn=t in the Level 1 --

MR. ROZGA: That is correct.

MEMBER STETKAR: -- PRA kind of quantification.

MR. ROZGA: Correct, it=s not in the quantification.

MEMBER STETKAR: I just wanted to make sure that the Committee was away, especially from what Mike asked about Fukushima and some things.

MEMBER CORRADINI: The reason I=m focused on this is for other studies. There=s this great intensity to try to figure out how does aux feed or

RCIC work for extended times.

And, it strikes me this would be a time to see how it would behave under your situation, particularly because I=m guessing we=re going to come back and ask about steam generator tube rupture and timing of it compared to other things.

And, one can delay that or preclude that if you had aux feed working water on the times.

MR. ROZGA: That=s fine.

Okay, Young, next slide, please?

MEMBER BLEY: Let me toss something in. Well, there=s a difference between what=s done in PRA and what=s done in licensing safety analysis.

And, PRAs almost forever, all of the systems that are available or could be available are modeled so aux feed water, RCIC would be modeled for different usages.

Fan coolers sometimes would be modeled for containment heat removal if they=re there.

And, sometimes the probabilistic likelihood of availability over long term is included.

So, in the PRA, those kind of things that are, I think, you=re bringing up, have always been part of the modeling.

MEMBER CORRADINI: Okay. But, so, here=s where I=m going with this. Is there something in the emergency operating procedures that tell you to shut off aux feed when you should try to keep it on?

In other words, is there a directive that you follow now that I would change because of what I=ve learned from Fukushima to keep this thing going longer?

MEMBER BLEY: That=s a possibility and it=s certainly design specific and it could even be plant specific at times.

MEMBER CORRADINI: Well, and the only reason I ask --

MEMBER BLEY: And, they don=t have their operating procedures yet, right, or do you?

MR. ROZGA: We have EOGs, the Emergency Operating Guidelines at this point.

MEMBER BLEY: Okay.

MEMBER CORRADINI: Well, the only reason I=m asking the question is, I=m looking back to what we=re going to discuss in other venues about steam generator tube rupture and timing of it relative to a station blackout at high pressure.

MEMBER BLEY: No, and I think that=s a really good point and, you know, depending on the PRA, you know, if you=re not -- if your procedures aren=t built to support it, it at least becomes a lot less likely that you do those kind of things.

MR. ROZGA: All right, the data and common cause data are used for equipment failure rates and it=s generally from NUREG-6928. There are some other data sources were used when 6928 did not provide that data.

And then, as a last resort, if we had no data source, then we used engineering judgment.

An example was the RCP seal failure that we discussed earlier, the 1E to the minus 3 per pump.

We don=t have any plant specific data and there is no generic data for air dryer, test and maintenance on availability.

So, yes, I don=t know why it=s not in 6928, John. But, it wasn=t. So, we have a number in there based on, you know, plant operating experience, what we think would be a reasonable number.

Human reliability analysis is a major

part of PRAs and the evaluation is based on standard industry methodologies.

The model includes about 60 pre-initiator, Type A HRAs and those are things that happened prior to the event that could impact the event.

There could be the operators miscalibrate something or they don=t realign a system post-maintenance test correctly.

So, we tried to identify those and insert them in the model.

Type B initiating events are only for initiating event fault trees. And, since we=re just using generic data, we have no Type B initiators at this point.

And, those would normally be if you fail a train, the operator would try to start the alternate train to prevent the trip.

Type C, those are the -- yes, sir?

MEMBER STETKAR: In -- remind me -- in your internal flooding models, do you include operator actions to either close flood barriers or open drain paths typically of flood?

MR. ROZGA: There are a few instances for specific flooding scenarios.

MR. DREMEL: Ray Dremel from Enercon.

The answer to that is no. We have operator actions --

MEMBER STETKAR: Okay, a few apparently is zero.

MR. DREMEL: Yes, we have operator actions to isolate a break before a certain volume of water would be released.

MEMBER STETKAR: Okay.

MR. DREMEL: But, we have no operator action to open a drain.

MEMBER STETKAR: But, that=s -- okay, the first one is one that I was asking about. Isn=t that an operator action that contributes to the initiating event frequency that you finally quantify in your model such that, if they failed to isolate the break within a certain time window, you would then have enough water entering a compartment to be a flood?

MR. DREMEL: Yes.

MEMBER STETKAR: Okay.

How do you account in your model then for dependencies between that operator action and subsequent operator actions after you draw the dotted line in the sand and say I now have an initiating event?

MR. DREMEL: We do the -- we do a dependency among all the operator actions and we saw the cut sets with all the human failure events set for relatively high.

MEMBER STETKAR: So, do you have a fault tree then that shows that operator action as a basic event in the fault tree model for the flooding initiating event?

MR. DREMEL: Yes.

MEMBER STETKAR: Okay, thank you.

MR. ROZGA: The last type is the post-initiator HEPs and we have about 70 of these. And, these are things like the operator fails to initiate feed and bleed when required.

One of the most important things with the operator actions are the dependencies between the actions. You can get several cut sets where you have several operator actions and those cut sets, because there are so many HEPs in them, would tend to get truncated.

So, what we do to make sure that we capture them is, prior to quantifying, we set all of the HEPs to a very high number, very close to one.

We quantify and then we identify all of the combinations. And then, the potentially dependent HEP combinations are evaluation -- or, I=m sorry, are evaluated based on dependency level decision trees and NUREG-1921.

And, they look at things like timing, crew, recovery from other individuals.

And then, the dependencies are inserted then at the end of the model with -- during the post-processing of the cut sets.

So, by doing that, we ensure that we capture the dependencies.

Did -- all right, make sure we=re on the right slide.

The quantification is performed. We already talked before about SAREX and CAFTA, the codes.

The truncation level that we quantified at is at E-13 for all models. And that=s about six to seven orders of magnitude below the CDF.

We use a variety of flagged files, house events and recovery files, et cetera, during the quantification to control the quantification process for each model. And, those things may turn off ECSBS, for instance, like we were talking about with John before.

We used the delete-term logic to remove unrealistic minimal cut sets, LCO, or Tech Spec disallowed maintenance combinations, for example.

And then, actually, prior to the quantification, we break circular logic loops that inevitably end up in your logic.

The final step in the quantification process is called recovery. And, there=s a recovery file where we read it, each individual cut set and then we apply recovery based on those cut sets.

Mainly used for applying the HRA dependencies, but you can all insert offsite power recovery rules and the such.

This is a CDF distribution by initiating event. As Young had mentioned earlier, it=s still dominated by SBO and LOOP. It=s much less than it was for the reference plant. There, it was about 60 percent, if I=m correct.

You see the other TLO, CCW and that=s a total loss of CCW and total loss of ESW system, those lead to RCP seal LOCAs.

And, the uncertainty analysis, we did a parametric uncertainty analysis post-quantification. And, there were selected sensitivity analyses that were performed.

There is currently an open item on generally overall on the uncertainty analysis and sensitivity analysis and we=re working with the staff at this time at resolving that.

And, unless John has no questions, then --

MEMBER STETKAR: In the words of Leslie Nielsen, surely you jest. Don=t call me Shirley, I=ll be Frank.

I promised Ron that I=d try to keep this constrained and short. So, what I=d like to do is highlight three or four issues that I identified during my reviews of the models. And, all of this information is in the DCD.

And, I only want to highlight these because I think that they=re -- I have many, many, many other comments that are kind of lower importance than this. But --

The first issue is reactor coolant pump seals. And, I mentioned a number there, that 10 to the minus 3 and I know that you=re changing those models.

When I went through the models, it struck me that the reactor coolant pump seal LOCAs are questioned in all of the usual suspect event trees.

They=re questioned in station blackout. They=re questioned in those partial and total losses of component cooling water and essential service water.

They=re not questioned in most other event trees, which is curious.

One thing that is notable on this plant compared to many plants is that although it contains four trains of some equipment, and I=ll use the term train here, like it has four component cooling water pumps nominally, and those are part of the certified design because they=re inside the walls.

It has nominally four essential service water pumps. Those are not part of the certified design, they=re kind of outside the walls. It has four safety injection plant pumps.

It basically, though, has a -- it=s a two division plant.

MR. ROZGA: That=s correct.

MEMBER STETKAR: So, for example, there are two divisions of component cooling water once you get all the plumbing taken care of.

The -- all of the cooling for the reactor coolant pumps, the auto coolers, the motor air coolers, the thermal barrier coolers, is supplied from Division 1, not supplied from Division 1 and Division 2. There=s manual crosstie valves that the model doesn=t account for.

So, what I=m getting to is there=s an asymmetry and a pretty strong asymmetry so that, for example, the reactor coolant pump seal failure contribution is much more dependent on failures of component cooling water, essential service water, Division 1 than it would be of Division 2.

MR. ROZGA: Correct.

MEMBER STETKAR: So, there=s an asymmetry.

Because of that asymmetry and the fact that there=s only one division is one of the reasons why perhaps some of those support system initiating events that are grouped with general transients on this particular plant might be more interesting than just a general transient.

The real concern I have with lack of completeness, perhaps, in modeling the reactor coolant pump seal LOCA is partially due to what we see in Level 1, core damage frequency.

It=s more important, I think, for Level 2. And, I wanted to -- this is why I want to kind of intercept it here as we go from Level 1 to Level 2.

We=ll hear in Level 2 that this PRA explicitly does account for consequential tube failures, high dry load scenarios, if you will.

This PRA also explicitly accounts for the fact that if you have a reactor coolant pump seal LOCA, there is a very high conditional likelihood that the loop seal in the affected loop will clear.

And, if the loop seal clears, there is an extremely high, like guaranteed, probability that you have a thermally induced tube rupture.

That makes the reactor coolant pump seal LOCAs really, really interesting in terms of tracking Level 2 large release frequency results.

And, I=m not sure that the models completely account for those contributions. Of course, you know, the numerical effects will depend on whatever your revised model of the conditional probability of the seal failure is.

But, even without that, there are I think some effects, for example, from the loss of instrument initiating event that may make you vulnerable to -- more vulnerable to seal LOCA. It may not guarantee a seal LOCA, but within a pump away or something like that for the seal LOCA.

But, the seal LOCA is not modeled in that particular event tree.

So, my concern about the completeness of the seal LOCA modeling under what initiating events are you asking the question, do the seals fail, is partly a concern about Level 1 core damage frequency. Because, even with the model that=s in there, we see some contribution from it.

But, to me, it=s even more important for Level 2 in the sense of, if you do have a seal LOCA and you go to melt at a relative -- at a high pressure, you get clearing the seals. And, if you then have nothing in the secondary side of one of your steam generators, it=s not a good day.

So, that=s one of the issues that I want to bring up. I don=t know if you want to reply to that. I didn=t have a real question in a sense other than trying to get a concern on the record here.

MR. ROZGA: The one thing that I will say with respect to the thermal barrier cooling coming directly from the Alpha train, if we were to split and have -- did one, did two RCPs and did another two RCPs, you basically will double.

MEMBER STETKAR: There=s no easy way around the problem.

MR. ROZGA: Yes, yes, yes.

MEMBER STETKAR: No, that=s what you said is there=s no easy around it.

MR. ROZGA: Now, there are crossties that could be credited. Right now, you have, you know, the --

MEMBER STETKAR: They could -- that all -- it=s a timing analysis.

MR. ROZGA: It=s a timing analysis and there is --

MEMBER STETKAR: And, that=s why --

MR. ROZGA: -- there is manual valves that would have to be opened. And, without procedures and without knowing exactly how far away the CCW heat exchanger belt.

So, at this point, we just --

MEMBER STETKAR: No, no, I know what=s in the model and I kind of know why it=s there. But, given that asymmetry, that=s, as I said, I don=t really have a question, I just, you know, if I said anything wrong, I would have expected you to come back and say, no, you=re lying, it=s wrong. But, I was pretty sure about that one.

So, that=s one issue, the seal LOCA, completeness of the seal LOCA modeling and its relationship to both Level 1 and Level 2.

The second one is also related to the consequential tube rupture, Level 2 more than Level 1. And, there=s kind of two parts to this one.

If you go back to -- go back to your pie chart. Well, you won=t see it on this, I=m sorry. You won=t see it on Level 1. We=ll see it on Level 2, but again, I want to get people sort of oriented before we get into the Level 2 discussion.

The current -- the PRA evaluates large steam line breaks upstream and downstream from the MSIVs and you see a little bit of a contribution here from large steam line breaks downstream. You don=t see any contribution here from large steam line breaks upstream.

When we see Level 2, you=ll see even a larger disparity between those two.

At the moment, large steam line breaks upstream of the MSIVs, it is assumed that they are all inside containment. Is that true?

MR. ROZGA: Correct.

MEMBER STETKAR: Okay.

That, in a, to me, in a one sense may be considered, I=m going to try to not use words that I don=t like to use, that might be worse for evaluating energy release into the containment.

It, however, is much better for consequential tube ruptures because, if I have a steam line break in a steam generator upstream of the MSIVs, I cannot isolate it. It will continue to blow down.

And, if the operators isolate feedwater as they=re mostly instructed to and parts of it is automatic, I can get a dry and low condition on that steam generator fairly easy.

Right now, that does not contribute to any Level 2 because all the releases are inside the containment.

If the break was between the containment wall and the MSIV outside of the containment, all of those releases would go outside, well, they=d go to the main steam valve room and, you know, out that way.

So, one of the sources of concern that I have about the steam line break upstream of the MSIV model is the fact that it=s all -- all of those breaks are allocated to only inside the containment. None of them are allocated between the containment wall and the MSIV, where there are a lot of welds.

There=s risers for all of the safety valves. There=s risers for the main steam atmospheric dump valve, so there=s a lot of welds in that line. I don=t know how big is a big or how big is a small.

So, that=s another concern that I have, more for the Level 2 analysis than the Level 1 analysis.

Dr. Corradini?

MEMBER CORRADINI: May I interrupt you in your disposition?

MEMBER STETKAR: You may because I=m going to switch gears to a different issue.

MEMBER CORRADINI: So, since you kind of educated me before we got together for this about this, so it=s the procedure you just said about isolating that I don=t understand.

In other words, I have a break or a leak and I start depressurizing and the instruction is to isolate aux feed. That=s what I=m confused about.

MEMBER STETKAR: At many plants, it is. I don=t know what=s assumed in this model. I think it is assumed that they would do that.

MEMBER CORRADINI: This why I asked the question earlier. But, he=s much more subtle in whether it=s upstream or downstream.

But, I=m just concerned that, if you isolate aux feed when you -- it=s workable, you would never then generate a low dry condition.

MEMBER STETKAR: It=s not at all clear if you got a big hole open to the environment whether you can, you know, keep it full. Eventually you can maybe. I don=t know.

MEMBER CORRADINI: But, that=s the reason I was asking the question.

MEMBER STETKAR: I know.

MEMBER CORRADINI: Because, the way you talk us through this is the aux feed is shut off by procedure.

MEMBER STETKAR: It=s not -- I know that it=s not shut off automatically in this plant, aux feed is not. Some plants do it automatically and this plant, I couldn=t find any signals or that isolate it automatically.

MR. DREMEL: It=s typical in --

MEMBER STETKAR: You have to come -- you still have to come to the microphone.

MR. DREMEL: Sorry, Ray Dremel with Enercon again.

It=s typical in most U.S. plants that, if you have a secondary side line break, you stop feeding the faulted steam generator for many concerns.

One is, the operators are concerned about killing their people. You know, you don=t want the steam going into the people tank, so to say.

So, the operators initially will turn off aux feed water until they can figure out what is safe to do and where can I send my people?

So, we model the operators isolate the feedwater -- aux feedwater to the faulted steam generator.

MEMBER STETKAR: The other one typically is they don=t like to be in a situation where you have an uncontrolled cool down going on. They like to be able to hold temperature and pressure and things in a way, you know, in a regime that they can actively control it and then walk the plant down. So, that=s another.

So, that=s -- I=m going to try to finish this stuff by 10:00 so that we can go on with the rest of it.

The other concern that I have regarding steam line breaks now is that, it=s my understanding that what I=ll call small steam line breaks upstream of the MSIVs. Some people might call them spurious opening of main steam safety valves. You can give it any name you want to, but it=s the kind of thing where you can either break the riser off or the main steam safety valve decides that it wants to open spuriously.

Those events, as I understand it, are now grouped with general transient. Is that true?

MEMBER CORRADINI: I believe so, yes.

MEMBER STETKAR: Okay.

The concern I have about that is that, if one of those happens, I don=t know whether an automatic reactor trip will occur on this plant because I=m not familiar with all of the trip set points.

I kind of know what the capacities of the safety valves are, it depends on where your high power or your turbine versus reactor power trip set points are. You may or may not get an automatic trip is -- and I=m not sure. Apparently, I don=t know.

However, if the plant does trip either manually or automatically, you then have a situation where you=re relieving a fairly good fraction of your rated steam flow out through this stuck open valve.

And, you probably will get an over cooling event. You certainly do have a -- you=ll get a safety injection.

You have an uncontrolled cool down. And, if I now go to core melt at high pressure, don=t I have now a high dry low with an open offsite relief path?

And, there=s no way for your current models to capture that because it=s all a general transient.

So, that=s -- those are my concerns about steam line breaks. Big steam line break location inside the containment, in summary, and no separate model for whether you want to call it a small steam line break or a stuck open -- a spuriously open main steam safety valve upstream of the MSIVs not distinguished as a separating initiating event to track.

Not so much Level 1, again, but progression through the Level 2 models.

The third thing that I wanted to -- and again, I=ll give you -- any comments?

MR. ROZGA: No.

MEMBER STETKAR: Okay. You guys know well enough to just interrupt me when I start lying and making up stuff. And, I=m old enough that I=ve gotten really good at lying and making up stuff.

MEMBER CORRADINI: Oh boy, is that true.

MEMBER STETKAR: The --

MEMBER SKILLMAN: John, may I ask a question, please?

MEMBER STETKAR: Sure.

MEMBER SKILLMAN: To your dissertation here, on the first item that you raised, asymmetry and the preponderance of the risk becomes because of the reactor coolant pump service is being cooled by a particular division of component cooling water.

Here=s my question, is that observation

an issue where our numbers in the PRA are misleading us? I would have thought intuitively that having the pumps serviced by different divisions would have reduced the frequency.

And, you=re theorem is, by the asymmetry that is there, the risk is probably as it should be independent of the fact that the cooling is from, if you will, an asymmetric or nonsymmetric cooling source.

The thought that was going through my head is, are we being misled by the PRA in this instance?

MEMBER STETKAR: I don=t - I don=t want to get too much internal discussions here with limited time, but we=re not being misled by the PRA provided that the PRA accurately accounts for all of the scenarios that can threaten the pumps seals.

I mean, the PRA is supposed to model the plant as its designed and operated and it actually is for the cases that you=re evaluating the seal LOCAs.

And the risk comes out to be whatever the risk is. You know, whether it would be higher or lower if you had two pumps off of Division 1 and two pumps off of Division 2, I=m not going to speculate on that right now, because I don=t like talking about numbers, I like talking about, you know, whether the plant -- the PRA is modeling the plant.

So, I don=t think we=re being misled by the PRA here.

MEMBER SKILLMAN: And, perhaps my question=s not the right question. I guess what I was really getting at is, presuming that the PRA results are accurate, are we being driven to leave the plant configured as the PRA suggests because we=re not willing to reassign the reactor coolant pumps to different divisions because that could be more beneficial.

MR. ROZGA: Could I interject?

Again, without talking numbers, if you had one division supplying two reactor coolant pumps and you had another division supplying another two reactor coolant pumps, now, if you have a failure in either division, that=s going to lead to a seal LOCA.

As now, if we have a failure in the Dib 2 CCW system, we won=t have a seal LOCA. So --

MEMBER STETKAR: It=s the difference between one division affecting four pumps or each of two divisions, each affecting two pumps.

MEMBER BLEY: We shouldn=t -- we could have something offline on this, but just a real quick close on that, the question you=re trying to phrase has different answers depending on exactly how you put the question.

Here, we=re saying, how likely is that we get one out of four having a problem and, if you=re then dependent on two different systems, that=s more likely.

If you=re asking the other question, how likely is it we have at least one that doesn=t have a problem, you get a different answer.

But, I think we ought to take that offline and --

MEMBER SKILLMAN: Fair enough.

MEMBER BLEY: -- do some detailed looking.

I think generally, our --

MEMBER CORRADINI: Until you think about it.

MEMBER SKILLMAN: Counterintuitive is where I am.

MEMBER BLEY: Our intuition isn=t always as finely tuned as we think.

MEMBER SKILLMAN: Thank you. John, I apologize for ruining the momentum you had.

MEMBER STETKAR: Oh, no, that=s okay. That=s sometimes people need to stop me.

The other issue that I wanted to bring up, and this, again, is kind of the nexus between the Level 1 and Level 2 modeling, is that the Level 1 steam generator tube rupture event tree does not question the status of steam generator -- of isolation of the ruptured steam generator.

Isolation of the ruptured steam generator is questioned in the Level 2 models. There=s a top event, I don=t remember, SGIS or something like that.

And, if it=s isolated, that=s a good thing. The Level 2, if it=s not isolated, that=s not so good for Level 2.

The question that I have is that the order in which that isolation is questioned between Level 1 and Level 2, if the steam generator is not isolated, if it=s open to the environment, and in particular, one isolation pathway is closing the main steam isolation valve that isolates everything downstream of the main steam isolation valve.

Another isolation pathway would be the

steam generator blow down lines.

And, a third isolation pathway is during the event response, the main steam safety valve, at least one of them, if not more than one, I guess two because there=s two steam lines on each steam generator, will indeed open under steam relief.

And, if one of those sticks open, does not reclose as you try to cool the plant down, you now have an unisolated ruptured steam generator and that changes the dynamics of the required operator actions in the Level 1 model to prevent core damage.

In particular, if the secondary side of the ruptured steam generator is open to the environment, the only way that the operators can stop the net loss of inventory from primary to secondary is to walk the primary system down to just about atmospheric conditions.

Once I get down to atmospheric conditions, I=m not going to have much of a driving head anymore.

But, you can=t just stop at a 1,000 pounds in the primary system. So, I think, from what I can see that there may be sources of optimism in the Level 1 tube rupture model because the status of isolation of the ruptured steam generator is not questioned in that part of the model.

It=s only questioned after the fact, after you have a core damage event already and then questioning, okay, if I got there through a tube rupture scenario, is it isolated?

Now, that also is reflected in the steam line break upstream and downstream of the MSIV models and the feedwater line break models because those models also have consequential pressure in these now tube ruptures included in them.

So, they have a simplified but not super simplified model for tube rupture response built into those event trees. And, those event trees also don=t seem to care whether or not the secondary side of the ruptured steam generator is isolated. They kind of progress oblivious to that.

So, I don=t know if you want to comment on that. That was sort of an observation.

He really ought to be up front.

(Laughter.)

MR. DREMEL: Ray Dremel from Enercon again.

And, as you pointed out, the modeling of isolation -- am I speaking loud enough?

MEMBER STETKAR: Yes, you=re on.

MR. DREMEL: The modeling of isolation of a steam generator tube rupture for Level 1 had a different focus than Level 2.

So, the events we consider failure to isolate for Level 1, if the operators don=t close the main steam isolation valve, and this is all initially, because it=s before core damage, then you have some steam going out which is taking mass out of the steam generator which is limiting to some extent the amount of water from the primary that=s going to over fill the steam generator.

So, if the operators fail to close the main steam isolation valve, that makes things better from a Level 1 point of view.

Because, the concern in the steam valve tube rupture is to prevent over filling the steam generator, the primary is leaking out, you need to put water on your main steam insolation valves you over fill.

MEMBER STETKAR: That=s one concern.

MR. DREMEL: So, from a Level 1 point of view, not closing the main steam isolation valve is a good thing. We don=t take credit for failures.

Similarly, if steam generator blow down is online and the operators don=t isolate steam generator blow down, then you=re taking water from the bottom of the steam generator, putting it in some place that=s relatively safe. I mean, it=s subprimary water that=s going to the steam generator blow down system and that makes things better from a Level 1 point of view.

The operators have more time to cool down to prevent over fill. So, that=s why those events are not modeled in the Level 1 event tree.

Now, as far as the main steam isolation valve sticking open --

MEMBER STETKAR: Safety valve.

MR. DREMEL: Safety, I=m sorry, the main steam safety valve sticking open, the first event in the steam generator tube rupture event tree is to -- the operator takes official action to cool down the initial 50 degree cool down to stop the flow from the primary to second to reduce secondary pressure less than primary.

So, to do that, the operators have to use the atmospheric dump valve and --

MEMBER STETKAR: On the intact steam generator?

MR. DREMEL: On the intact steam generator, yes.

But, if they cool down quickly, then you=re really not challenging the steam generator -- main steam safety valves because you=re cooling down the whole primary. Because initially --

MEMBER STETKAR: They -- don=t they open initially?

MR. DREMEL: There is -- it depends on the trip and what happens. So, until the operators close the main steam isolation valves and the EOP=s directing the operators to initially cool down to the condenser, and use the turbine bypass valves which should open automatically to prevent challenging your safety valves.

So, we don=t necessarily credit the turbine bypass valves, but by design, they would open to prevent challenging the main steam isolation valves.

MEMBER STETKAR: Safety valves.

MR. DREMEL: Safety valves, I=m sorry.

MEMBER STETKAR: Okay, thank you.

I still think that there=s a hole there.

The -- I agree with your concern that having a path to put water other than just in the steam generator makes it fill more slowly.

I=m more concerned in the long-term that keeping a path open to transfer the IRWST to some place that the IRWST ought not to be is also a concern.

Granted that that=s a long time, but if the operators, to stop that, if the only way that they can stop that is to make the primary system close to atmospheric. In other words, cool it down to like close to a 100 degrees C, it=ll take them some time to get there, given their cool down rates, allowable cool down rates.

MR. DREMEL: Ray Dremel from Enercon again.

And, in the Level 1 steam generator event tree, if the operators over fill the steam generator -- the faulted steam generator, the next node is, do you cool down to atmospheric and you can get there from a number of ways.

MEMBER STETKAR: That=s -- yes. But, my concern, I -- the model as it=s laid out is, if you pump the safety valves on the ruptured steam generator open by filling it with water, it=s assumed that they stick open and, indeed, the model walks you down -- has to walk you down to atmospheric conditions in the primary side.

What I=m concerned about is, other ways that before you over fill the steam generator during a cool down that you can get a stuck open safety valve or an open pathway through and MSIV that maybe you didn=t think about other ways that steam could get out or whatever.

It=s just -- we=ve had enough discussion about it. I=m three minutes over where I said I=d stop, so I=ll stop.

CHAIRMAN BALLINGER: I think this is a convenient place to stop for our break. So, we=ll recess for -- until, let=s try to come back at 15 minutes after to try to make up a few minutes.

I might remind people that, at the rate of slide production, we=ll be here until Saturday afternoon at about 5:00.

We=re in recess.

(Whereupon, the above-entitled matter went off the record at 10:03 a.m. and resumed at 10:16

a.m.)

CHAIRMAN BALLINGER: Come back in session please.

By my reckoning, we=ve advanced by two slides that weren=t discussed.

MR. T. HWANG: I present them. Can I start the presentation?

CHAIRMAN BALLINGER: Yes, but what I=m saying is I was on slide 22 and there=s 23 and 24. Okay, from there.

MR. IN: These slides were covered already.

CHAIRMAN BALLINGER: Oh, okay. Got it, okay, sorry.

MR. IN: We went back to the -- yes, 22.

The next presentation is 19.1.2 and it=s internal events Level 2 and it=ll be presented by Mr. Hwang.

MR. T. HWANG: Thank you.

My name is Taehee Hwang and I=m working for the KEPCO E&C Company PRA Group.

In this part, I briefly introduce the method approach here of APR1400 DC Level 2 PRA and its recert.

In the APR1400 Level 2 PRA, the methodology I will PDS, Plant Damage State and Containment Event Tree with the composition event tree analysis is used.

PDS model originally created in the SAREX and PDS event trees used to capture all inner system and intra system dependencies.

Following the quantification recert of PDS and containment event tree analysis, the source term variation was propounded for each release category.

To develop the APR1400 PRA model, the MAAP 4.0.8 code was used to analyze severe accident progression and system release variation.

And, SAREX code was used to develop the Level 2 PRA model.

Before presenting the Level 2 PRA recert inside, let me briefly explain the severe accident mitigation feature of the APR1400.

Which was significantly constructing the Level 2 PRA model.

First is APR1400 containment. It is designed as a pre-stressed containment with a steel liner plate and it is designed to be large dry type containment.

The containment in their pre-volume is approximately 3.1 billion cubic feet.

Next, to the reactor cavity for APR1400 is designed to minimize the challenges posed by that containment heating, pure coolant interaction and molten corium concrete interaction.

The reactor cavity has a large -- for express the corium spreading and its coolability.

Next is severe accident and mitigation picture, considering the PRA is a capped flooding system.

During the severe accident, it functions to minimize or eliminate the corium concrete attack due to MCCI after the reactor vessel breach.

And, it also functions to minimize the generation of combustible gas such as the hydrogen and/or carbon monoxide in MCCI.

Mind that the design information for severe accident mitigation feature will be, again, addressed in the 19.2 presentation.

And, hydrogen mitigation system is designed to limit hydrogen concentration in containment within ten volume percent.

The hydrogen can be generated from the 100 percent cooled clad water reaction and hydrogen mitigation system consisted of 30 PARS and 80 igniters.

MEMBER REMPE: Before you leave, I apologize, I missed the very first few slides, but I had some questions and this is as a good a time as any.

But, when I was looking through the material, there=s discussions of a term called the core debris chamber and a core cavity trap.

Is that just the base of the underneath the vessel? I mean, is there something special that makes it a core cavity trap or a core debris chamber? Do you have -- what=s the definition of those terms? Can you show me on the drawing what you=re talking about?

MR. T. HWANG: Yes, the reactor cavity has a cavity chamber to capture the excess core debris and to prevent the high pressure valve ejection or DCHE compartment.

And, the reactor cavity has a component to the flow path to the containment compartment.

MEMBER CORRADINI: So, do you have a

picture?

MEMBER REMPE: Yes, that would help. And, are there any things -- is there anything special in the cavity? Is there any extra liner or anything like that or is it just that region?

MR. IN: Yes, we don=t have a picture on these slides.

MEMBER CORRADINI: Can you at least tell us where to look? I think we=ve got all the documents somewhere. Or, that=s my problem, but if you just tell us where to look, because I also was a bit confused about what the description meant.

MEMBER REMPE: There=s a figure, I guess, under -- but again, it=s a proprietary document, but if you could point it out on a Figure 2.9-2 of your severe accident analysis report, that would help, I think. Just, I want to make sure that I understand.

MR. T. HWANG: We=ll look for that.

MEMBER CORRADINI: Yes, we can do that.

MEMBER REMPE: Sometime, yes, later, yes, it would just help because I was confused about the terminology.

MR. B. KIM: Byung Jo Kim, could you explain the detailed design of the reactor cavity?

CHAIRMAN BALLINGER: Are we into the proprietary discussion now?

MR. B. KIM: No. This is Byung Jo Kim from KEPCO Engineering and Construction Company.

In the DCD Figure 19.2.3-1, you can find what is the difference between the reactor cavity and the chamber room and so on.

And, the other information for this compartment in the lower region is given in Table -- DCD Table --

MEMBER REMPE: Slow down for a second, could you give us that figure number again?

MR. B. KIM: 19.2.3-1.

MEMBER REMPE: I=m looking.

MR. B. KIM: And, in the Table 19.2.3-2, includes the volume or elevation of the bottom of each subcompartment and height of each compartment including the rear cavity and chamber room, cavity chamber room and as I said -- and so on.

MEMBER REMPE: Okay, so, I=m looking at this figure and it=s just a containment drawing. And so, it is indeed just that whole region under the vessel is the trap or the chamber?

MR. B. KIM: Yes.

MEMBER REMPE: Okay.

MR. B. KIM: Actually, the DCD figure is not the whole information for the how the reactor cavity can trap the containment debris. So, this kind of cavity space is not exactly shown in this figure, but you can find the -- if you can look at the technical report related to it, you can see the variation.

MEMBER REMPE: Okay, then, tell me again the table number, too. So, I found the figure and what was the table?

MR. B. KIM: Yes, Table 19.2.3-2.

MEMBER REMPE: Okay. Thank you.

MEMBER SKILLMAN: Joy, it=s also Figure 1.2-4 in Tier 2.

MEMBER REMPE: And, they actually label it as -- because this does not have a label on it.

MEMBER SKILLMAN: Labeled as ICI Cavity and it shows the reactor cavity.

MEMBER REMPE: Okay, but this core debris trap --

MEMBER STETKAR: Just be careful, because those Chapter 1 things are labeled --

MEMBER REMPE: Yes, I did -- yes, they

have these funny phrases like chamber and trap that I=m not seeing on the figures and that was my confusion. And, thank you.

Actually, I have more questions, too.

What type of concrete is used on figure or in slide 28? Is it the salt based or is it limestone concrete? Did you --

So, in your analyses, did you assume a particular type of concrete in the MCCI evaluation?

MR. T. HWANG: Concrete type is limestone concrete, yes.

MEMBER REMPE: Okay.

MR. T. HWANG: So, it is good to cool the excessive core debris.

MEMBER REMPE: Okay, thank you.

CHAIRMAN BALLINGER: So, let me repeat it so I finally found the figure. I=m not as fast.

So, 19.2.3-1 it kind of looks like a PWR reactor cavity. Is there something unique about it that we need to understand?

I mean, you called it the ICI cavity, it just kind of looks like a reactor cavity with in core instrumentation coming out the bottom and going up to the seal table. Is it bigger, smaller, is there something unique about it that we should be aware of?

MR. IN: Can we discuss this question in the 19.2 because, you know, they have more --

CHAIRMAN BALLINGER: Sure.

MR. IN: -- figures there.

CHAIRMAN BALLINGER: That=s fine.

MEMBER REMPE: Okay.

CHAIRMAN BALLINGER: That=s fine.

MR. IN: Yes, this Level 2 --

CHAIRMAN BALLINGER: No problem.

MR. IN: -- doesn=t have any figures.

CHAIRMAN BALLINGER: No problem.

MR. IN: Okay.

CHAIRMAN BALLINGER: No problem.

MR. IN: Thank you.

MR. T. HWANG: Okay, next to severe accident mediation feature constructing Level 2 PRA is a pilot operated safety relief valves. It provides a means to rapidly depressurize the primary system to about 250 PSIA to prevent DCH and induced steam generator tube rupture following the severe accident.

And, the three-way valves located in the POSRV discharge path can be used to redirect release point of hydrogen from IRWST to the containment atmosphere.

And, the next feature is ECSBS.

MEMBER STETKAR: Can I ask a question about the relief path? The -- I looked at the model for that and the success criteria for rapid depressurization in the Level 2 model, or whatever it=s called, it=s top event SDR --

MR. T. HWANG: Yes.

MEMBER STETKAR: -- requires that the operators open at least two POSRVs and they redirect the discharge to the steam generator compartment, is that correct?

MR. T. HWANG: Yes, right.

MEMBER STETKAR: So that --

MR. T. HWANG: Yes.

MEMBER STETKAR: -- if I do not open the POSRVs I remain at high pressure. And, if I do not direct flow to the steam generator compartment, I remain at high pressure. Is that correct?

MR. T. HWANG: Yes, right. If the -- in our model, the success criteria for rapid depressurization is open -- is to operate more than two POSRVs and the operation of auxiliary valves.

MEMBER STETKAR: How much difference does it make to the hydrogen models if I release into the IRWST compared to into the bulk volume of the containment?

So, for example, if the operators open the POSRVs but only direct the flow into the IRWST, how -- what effect does that have on your hydrogen modeling?

MR. T. HWANG: In the IRWST area is relatively closed volume so the -- if the hydrogen generated from the vessel go to the IRWST area, then the -- it result in a hydrogen concentration inside the IRWST area.

Even if the swing panel in the IRWST upper part, but significantly, hydrogen releases go to the IRWST area, in there it can make a different condition during the severe accident.

MEMBER STETKAR: So, I was trying to ask a simple question. Is it worse if you release into the IRWST compared to the containment?

MR. T. HWANG: Yes, yes.

MEMBER STETKAR: Okay, thank you.

MR. T. HWANG: It=s very worse.

MEMBER STETKAR: The model right now,

then, and this is just a statement of my concern, the model right now is -- I try to stay away from a word that I don=t like to use -- the model over predicts the conditional probability of high pressure because failing to realign the three-way valves to the containment, despite the fact that the POSRVs are open will go to high pressure.

So, it over predicts high pressure, but it under predicts, perhaps, hydrogen effects in the IRWST because, every time you depressurize, by definition, it must be to the bulk volume of the containment. You never can get a successful depressurization with hydrogen release to the IRWST.

MEMBER CORRADINI: Can you say --

MEMBER STETKAR: Well, that=s convoluted, you can read the --

MEMBER CORRADINI: Can you say that again?

MEMBER STETKAR: -- you can read --

MEMBER CORRADINI: Sorry.

MEMBER STETKAR: Okay.

I, as an operator, can open the POSRVs to depressurize. Okay? I do that, now I have a choice between where do I put the flow from those POSRVs? I can either put it into the IRWST or I can put it out into the steam generator compartment.

Their success criteria for depressurizing requires me to both open the POSRVs and put the flow into the steam generator compartment.

If I fail to do either one of those, I stay at high pressure, either one. So, I could open the POSRVs --

MEMBER CORRADINI: I don=t depressurize or no?

MEMBER STETKAR: I can open the POSRVs and keep the flow path aligned to the IRWST. In the model, that is treated as a high pressure case.

It doesn=t have to make sense, it=s just a fact.

MEMBER CORRADINI: It=s just the way the --

MEMBER STETKAR: It=s just the way they=ve modeled it.

MEMBER CORRADINI: Okay, fine.

MEMBER STETKAR: So, that is not good for high pressure because I will actually depressurize if I blow into the IRWST.

However, if I go to melt then, I will

have, in that flow path, hydrogen released into the IRWST with a higher likelihood of a detonatable mixture in the IRSWT.

That detonatable mixture cannot occur in their model because their model requires success to always be into the containment.

I know it=s confusing, it=s on the record. I can talk to you later about the logic.

MEMBER CORRADINI: I just asked you to repeat it. Thank you.

MR. RYU: Excuse me.

MR. T. HWANG: Mr. Ryu?

MR. RYU: I am In Chul Ryu from KEPCO E&C for severe accident analysis team.

Actually, the way the valve is always open, not closed. Just closed direction change. So, if we operate the POSRVs always we can deliver the high pressure in the excess pressure.

So, the -- we don=t need to operate two valves should be open.

MEMBER STETKAR: The -- and just for --

MR. RYU: And also, the installation of the three-way valve is not -- we just -- we cannot say simple release into the IRWST is dangerous because we just want to remove the uncertainty of a hydrogen problem.

For example, another plant and if -- even though the hydrogen concentration is very high in the IRWST, but actually the combustion is required for the oxygen. So, the -- in that case, a simulating case so the IRWST is also the same.

But, why we introduced the three-way valve is the different reason. One is that we don=t argue the dangerousness about the valve.

And also, if we have the -- if we release into the IRWST, then the risk point is the annual severe area of the reactor containment. That area has many accumulations. So, in that area, we may have the diffusion frame.

So, you want to leave the environmental condition in that area. So, we change the direction into the steam generator compartment so we installed the three-way valve, not because of the just dangerousness of the IRWST.

MEMBER STETKAR: Thank you. I just was making the observation that the model, the way it=s implemented, is as I stated.

MR. T. HWANG: Thank you for your comment and I=ll take consideration about that.

Next, we --

MEMBER BLEY: I=m just going to -- I=ll let John tell me if I=m off track. But, I think where John=s headed on this, he=s telling point now what the model does. But, what=s conservative or okay for one thing you=re thinking about might not be for another thing you=re not thinking about at this time.

And, if you model it the way it really works, you get to cover all those cases when you model it in a way that takes care of what you=re thinking about. When you=re doing the modeling, you might be missing something important elsewhere.

MEMBER REMPE: And, to beat a dead horse a bit more, the ERI report did look at that case and that you could get higher concentrations if you considered that case.

MR. T. HWANG: And next mitigation feature is the ECSBS. ECSBS provides alternative means for containment spray after 24 hours following the severe accident initiation.

It delivers water from external water source to the ECSBS containment spray header and ECSBS will be a pumping device which is independent of normal and emergency AC power source.

MEMBER CORRADINI: So is that FLEX? Your terminology for FLEX?

MR. IN: This was not designed as a part of the FLEX. It was a design previous to the --

MEMBER CORRADINI: Oh.

MR. IN: -- FLEX.

MEMBER CORRADINI: Does it satisfy the FLEX? I mean, that=s what it sounds like.

MR. IN: Yes.

MEMBER CORRADINI: Okay, fine. Thank you.

MEMBER STETKAR: That, by the way, I was trying to look up my notes and, in the interest of time, I won=t quote numbers, but my recollection was that ECSBS is a rather important contribution to your conditional containment failure probability.

My recollection was that you did a sensitivity case that, without ECSBS, the conditional containment failure probability increases by something like a factor of five or so.

MR. T. HWANG: That=s right. Yes, it=s very --

MEMBER STETKAR: Okay. So, this is

important for you Level 2 folks.

MR. T. HWANG: For an internal event, at power --

MEMBER STETKAR: At power?

MR. T. HWANG: Yes.

MEMBER STETKAR: At power internal event?

MR. T. HWANG: Yes.

From now on, I=ll briefly explain the method and approach of APR1400 Level 2 PRA.

The first task for Level 2 PRA is the plant damage states analysis.

In the PDS analysis, the Level 1 event pre-sequences are extended to be additionally questioned in terms for the Level 2 PRA.

For example, the status of containment isolation is an important parameter question in the Level 2 PRA.

Also, if not questioned in the Level 2 PRA model, the status of containment sprays or the status of a steam generator condition, wet or dry, are also questioned in PDS event trees.

The bridge tree --

MEMBER STETKAR: Can I just stop you here and I promise I=m not going to say much more on Level

2.

Containment isolation, there was in Chapter 9 of the DCD describes a containment high volume purge system that=s operated during plant shutdown.

And, it also describes a containment low volume purge, smaller line, that=s operated, and I quote, when required during plant operation.

MR. T. HWANG: Yes.

MEMBER STETKAR: Many plants, in my experience, do operate their low volume purge to keep the containment atmosphere relatively clean in case people need to go in there.

I don=t -- they don=t necessarily operate it 100 percent of the time, that=s plant specific.

My only question is, does the containment isolation model account for isolation of that low volume purge for whatever fraction of the time that it=s open? Is it included in the model?

MR. T. HWANG: The APR1400 PRA model the low volume purge line is cleaned out because, as you said, the low volume pressure line is normally closed and can be operated intermittently.

But, the line is -- the line has to be

closed during the extent because the containment low volume line is closed by actuation of containment isolation signaler or the containment depressurize in isolation signal.

And, the lines of valves are being monitored, will be monitored in the MCI operators. So, we screened out this line has a very low probability of isolation.

MEMBER STETKAR: Okay. Many things in the PRA had low probability. The reason I bring this up is that, if that line is open and it is not isolated, it may represent a large enough hole in the containment that removes containment energy, just removes energy.

Now, that=s a good thing for not over pressurizing containment. It=s not a good thing for offsite releases.

So, here=s another case where assumptions about what may or may not be included in the model can have an effect, depends on the fraction of time that the line is open, depends on the reliability of the isolation signals and so forth.

So, I=ll just make that comment, if it=s not in the model, it can do good things for you in terms of not over pressurizing the containment. It can do bad things in terms of increasing the conditional large early release frequency.

CHAIRMAN BALLINGER: I need to remind people that we=ve got 80 some odd slides and, right now, they have to be finished, at least by the schedule, at 11:30, which means we have to do it almost like a movie.

So, we=ve got to do something about this.

MEMBER CORRADINI: I thought -- I mean, just looking at the schedule, I thought 19.1 goes through the afternoon.

CHAIRMAN BALLINGER: But, there=s an NRC presentation 19.1 also.

MEMBER CORRADINI: Okay, fine.

CHAIRMAN BALLINGER: I=m just -- just to remind -- I mean, it=s our fault, not our fault, we=ve been asking good questions.

MR. T. HWANG: Is it okay?

CHAIRMAN BALLINGER: Optimistic schedule. Keep going.

MR. T. HWANG: Yes.

Yes, so the bridge tree sequences are grouped into the PDS group based on the similarities in the extent progression based on the PDS grouping parameters such as containment bypass or the status of containment isolation, the LOCA or transient sequences, the ICS pressure and so on.

As a Level 2 PDS binning, 108 PDS groups were defined and quantified to capture all Level 1 and Level 2 dependencies.

To develop the Level 2 containment event tree model, we needed to estimate the APR1400 stress peak containment UPC.

In the APR1400 Level 2 PRA, the plant specific containment to the facility was determined by ultimate pressure capacity calculation which approximates the realistic probability for keeping pressure on.

For APR1400 containment, two containment failure modes such as loss of power mode and leak power mode determined based on the NUREG-1150 and NUREG/CR-6906.

In the Level 2 containment event tree analysis, the various containment failure modes and the major severe accident phenomena are represented a top events of the containment event trees.

Detailed variation of the phenomena for

each top event of CET is created in the composition event trees.

The Level 2 CET considered the following containment challenges, direct containment to bypass, containment isolation, system failure, induced SGTR during the severe accident, high pressure mass ejection and direct containment heating or blow down post steam explosion, hydrogen phenomena, steam over pressurization and MCCI and basemat melt through.

In the development of the APR1400 Level CET, the generic data were used.

NUREG-1570 was utilized for developing induced ISGTR combustion event tree, including the condition probabilities of induced ISGTR developed for the current generation plants.

MEMBER REMPE: So, 1570 was for a Westinghouse plant design.

MR. T. HWANG: Yes.

MEMBER REMPE: Your steam generator, I would think, would be more like a CE type of steam generator design. How do you justify using those conditional probabilities?

Because the geometries are a bit different, right?

MR. T. HWANG: Yes, the geometry -- the 1570 plant is different from the APR1400, but the -- at this time, the plant specific in this ISGTR probability cannot be developed in this design stage.

So, we just assumed that the condition probability of this in this ISGTR for NUREG-1570 was used APR1400 event tree.

MEMBER REMPE: So, the staff has been working hard looking at CE designs and they have this NUREG-2125 that they issues as a draft and it actually has quite different numbers of the CE designs.

Now, they also said things are very design specific, so you=ll need to look at what they did and see if it applies to your design. But, it does have quite higher conditional probabilities for the CE design and you might want to consider it.

MR. T. HWANG: Yes, yes, it=s a good comment and we=ll take consideration for another around -- as your comment directs 2512 and --

MEMBER REMPE: It=s 2125 and it=s a just NUREG, it=s not a NUREG/CR.

MR. T. HWANG: Yes, okay. Okay, thank you.

And, NUREG/CR-6475 and NUREG/CR-6109 for induced hot leg rupture probability was referenced for the -- referenced to the PRA.

And, NUREG-1150 and NUREG/CR-4551 were considered for various phenomena such as in vessel core recovery or rocket mode failure or steam explosion and so on.

After the quantification of PDS and CET analysis, the number received at the end points is very large, and detailed system analysis for all the CET end points in not feasible.

Hence, CET end points are grouped into the system release categories based on the similarities of release characteristics such as magnitude of the timing releases.

Source term release calculations are performed using MAAP 4.0.8 code. And, to determine the large release frequency for the APR1400 Level 2 PRA, the large release is defined as released of greater than 2.5 percent of volatile or semi-volatile fission products that is iodine, cesium, tellurium.

This is a second depiction of large releases in the NUREG/CR-6595.

In addition, the APR1400 Level 2 PRA defined all the releases as that the release people, they calculated the evacuation of the surrounding public after the generic emergency declaration.

This slide shows the Level 2 PRA result for at power internal events. The conditional probability of an in tank containment is 86 percent of internal CDF.

The conditional probability of containment failure, including large releases and small releases is 14 percent.

Where the small release categories includes the release categories of basemat melt through or steam generator tube rupture, with the SGTR for scrubbing -- and so on.

Finally, the conditional probability of large releases is 9 percent of internal Level 1 event CDR.

The most significant containment failure contributor is the containment bypass which is 6 percent of CDF.

Unisolated steam generator tube rupture sequences prior to core damage contributes 5 percent to at power internal events CDF.

Severe accident induced SGTR such as severe accident induced SGTR contributes 1 percent to core damage frequency.

Level 2 depressurization by using POSRVs is effective to prevent induced SGTR.

And, the single tube rupture with pool scrubbing, with wet pool scrubbing, do not result in a large release due to pool scrubbing inside loss of the steam generator.

Second dominant containment failure contributor is late containment failure which is 5 percent of CDF.

ECSBS is effective to prevent containment failure due to steam over pressurization and PARs and flooded cavity by cavity flooding system is operate effective to prevent the buildup of high hydrogen concentration inside the containment so it prevents the containment failure due to hydrogen.

This is the end of my presentation.

CHAIRMAN BALLINGER: So, can you --

MEMBER REMPE: I have to correct myself. Professor Ballinger got that I was saying the wrong number, it=s NUREG-2195. I apologize.

MR. T. HWANG: NUREG-2195.

MEMBER REMPE: Yes, I apologize.

MR. T. HWANG: Okay, yes, thank you.

Thanks very much.

CHAIRMAN BALLINGER: You have a much better memory than I do.

MEMBER REMPE: Yes, well, I forgot.

MR. T. HWANG: It is a final issue?

MEMBER REMPE: It was a draft issued a while ago, but the final -- we=re having an ACRS meeting to discuss it this next -- in two weeks from now. But, there is an earlier draft and the numbers were higher in that draft, too.

MR. T. HWANG: Okay.

MEMBER REMPE: So, in a week or two you should see a new one.

MR. T. HWANG: Thank you very much.

MEMBER REMPE: Yes.

MR. SCHNEIDER: Ray Schneider, Westinghouse.

There was an error in the draft where they treated the CE pumps as Westinghouse pumps. I think we sent them a note to fix that, so there may be some changes coming from.

MEMBER REMPE: Okay.

MR. IN: Shall we move on to the next presentation?

The next presentation is on the seismic assessment and it will be presented by Dongwon Lee.

MR. D. LEE: Good morning, ladies and gentlemen. My name is Dongwon Lee from KEPCO E&C PRA team.

I=m going to present to you seismic risk assessment.

For the seismic risk assessment, we considered three different methodologies.

First, a Staff Review Memorandum to SECY-93-087; and second, DC/ISG-020 provides the guidance for the implementation process for performing PRA based SMA. And, three, SSC, structure, systems and the components can be evaluated by either Conservative Deterministic Failure Margin method or Separation of Variables method.

Next?

For the PRA based SMA, we considered seismic input motion. We certified seismic design response spectra which is from Spectral Reg Guide 1.60 enhanced in high frequency.

CSDRS anchored to a peak ground acceleration of 0.3g and defined at free field ground surface.

For the PRA model assessment, seismic margin earthquake is considered at 1.67 times the CSDRS.

Look at the right side of the figures, the blue line is the certified seismic design response spectra and red one is the margin earthquake.

Next?

APR1400 FSTS, we did design specific capacity. We do the specific coolant capacity.

First we do the specific building structures capability analysis such as reactor containment building and the concrete internal structures of the building, EDG and diesel fuel tank room building.

And, we considered seven RCS at design specific capacity such as the following.

Next?

MEMBER STETKAR: On that slide, it struck me as curious that the structural analyses do not include the emergency service water component cooling water heat exchanger building or the tunnels that connect the component cooling water heat exchanger building to the auxiliary building.

And, I know that those are not part of

the certified design, but the implication is here that they must be much, much stronger than the containment because they never fail in your seismic margin analysis.

So, that struck me as curious because now, as a COL applicant, must I build those building much, much stronger than the seismic category one containment to satisfy the seismic margin analysis?

Or, will I be surprised when I do my COL seismic margin analysis and discover that if those buildings fail, they could be an important contribution to my seismic risk?

MR. D. LEE: I want to introduce Mr. Kyuho from SGH, he might give you the details for that.

MR. K. HWANG: I=m Kyuho Hwang from SGH to support KEPCO E&C on the seismic evaluation.

Is it on?

CHAIRMAN BALLINGER: Just a little closer.

MR. K. HWANG: All right.

Well, actually, the system the yard buildings, the safety related yard buildings are not in the scope of DC certification. So, we just, right now, so those buildings are never designed at this stage. But, we just show that those buildings are seismically rugged so we can screen them out.

MEMBER STETKAR: I=m sorry, the reactor containment building, to me, sounds like a seismically rugged building, and yet, you have a specific fragility evaluation. It has -- it does not have a zero failure probability.

So, by implication, these other buildings must be more -- must be much stronger, much stronger, than the containment. Because they have precisely zero failure probability.

MR. K. HWANG: Well, actually, our approach is like a deterministic approach. So, we never enveloped the probability of the failure of those buildings in question.

MEMBER CORRADINI: But, just to clarify, so you left it out of the analysis? That=s what I hear you really saying. I mean, John is trying to pulse you to say that, but you=ve left out of the analysis, is that fair to say?

MR. DREMEL: Ray Dremel from Enercon.

And, not the CCW building, the ESW buildings have zero failure probability. We don=t have a plant specific fragility for those, so we assumed that they will have at least a .5g HCLPF. They are included in the seismic margins assessment model.

MEMBER STETKAR: I=m sorry, they are not included in the seismic margins model. I could not find them anywhere.

MR. DREMEL: Are we talking Rev 0 or Rev 1?

MEMBER STETKAR: They were screened out.

MR. D. LEE: Rev 0.

MEMBER STETKAR: Rev 0, they were screened out.

MR. DREMEL: They were added in Rev 1.

MEMBER STETKAR: Okay. Thank you.

MR. D. LEE: Can we move on?

Okay, this slide is for approach for HCLPF capacity evaluation.

First, the critical failure mode should be identified. First this step, APR1400 design specific report calculations and the drawings should be reviewed.

And, potential failure modes by comparing design seismic demand to design capacity already identified areas of design margin.

And then, governing failure modes for HCLPF capacity evaluation was selected.

For the seismic demand, APR1400 design specific seismic demand were used and CSDRS applied at plant finished grade in the free field for eight generic certified and a fixed base case such as design in Chapter 3, what it is.

And, static capacity equations were used coded capacity for ACI 349 and ASME Section III Service Level D Level of EPRI NP-6041-SL Revision 1.

And, for the ductile failure mode, we considered inelastic energy absorption capacities.

And, finally, we conducted the HCLPF capacity of SSCS CDF method in EPRI NP-6041 applied to demonstrate HCLPF is equal to or greater than seismic margin earthquake that is 1.67 times CSDRS.

I=m going to introduce the major HCLPF from the structures. This table shows you the summary of the building structures.

As you can see, the results varied from 0.51g to 1.09g. The lowest capacity was the auxiliary building which is 0.51g. The governing failure model is all 15 at the basemat.

Next?

And, this table shows you the RCS component HCLPF and it is varied from 0.51g and 1.31g.

The lowest one is the pressurizer and the reactor internals. Reactor internals governed by core support barrel lower flange to primary membranes stress and the pressurizer was governed by pressurizer spray nozzle.

Next?

The other SEL components, it is related to RAI question 19-73, a. It requests such as provide the basis and justification for the assumption HCLPF. And, second, provide a detailed description of the methodology.

So, we provided detailed description of HCLPF and shortages and the basis and the justification for ISGS shortage.

The following is the answer for this RAI and HCLPF of ESWIS CCW heat exchanger building and BOP components because the detail of design information is not available in this phase.

And, also, the SSCF design spec is a COL item as well in Chapter 3. That=s why we assigned to COL items and assume to have a 0.5g HCLPF.

Next?

MEMBER STETKAR: Except, the only point I was trying to make, if you went back a couple of slides, is that, you showed that the lowest HCLPF capacity for the buildings in scope that you evaluated was the auxiliary building at .51g.

You used a nominal 1g earthquake for your seismic margin evaluation. And, that auxiliary building has a non-zero probability of failure at 1g, hence, the service water and component cooling water building would have a non-zero probability of failure at 1g.

MR. D. LEE: Mr. Ray could give you the details.

MEMBER STETKAR: And, I think we=ve heard you=re fixing that up in Rev 1.

MR. DREMEL: Right. The answer is --

MEMBER STETKAR: I just wanted to kind of close the loop here.

MR. DREMEL: Yes, and the answer is that the values that are being presented here are for Revision 1 of the seismic margins analysis.

MEMBER STETKAR: So, that the -- just to make sure, and we haven=t seen Rev 1, I would expect then for the nominal 1g earthquake that you use to propagate -- to solve your model, I don=t want to say quantify frequency, but to solve your model that the ESW CCW building and those tunnels will have some measurable probability of failure.

MR. DREMEL: Yes, yes.

MEMBER STETKAR: Okay, thank you.

MR. T. HWANG: Okay, continue.

MR. D. LEE: Okay.

To develop the seismic equipment list, the following three things are considered.

First, seismic initiating and the consequential events were defined such as the direct core damage scenarios such as a building collapse, loss of all instrumentation control, SBO, LOCAs, adverse and loss of offsite power.

And, the safety functions needed for response were determined such as the following and then steps needed to fulfill safety functions were identified based on internal event PRA and powered by onsite emergency AC sources.

Next?

Here we designed the logic model. First, the seismic event trees were considered following seismic events, direct core damage and also I&C at the large LOCA, medium LOCA, small LOCA and LOOP.

And then, seismic event trees inserted the seismic failures into failure trees and lastly, to the plant level HCLPF, the following will be performed to solve the seismic event tree models.

And, through the Min-Max method, the final plant level was developed at 0.5g and beside from that, we assumed the generic failure of SSGS compound building collapse and turbine building collapse was 0.5g HCLPF, as I mentioned earlier.

Next?

Here is my conclusion, major APR1400 SSCs were evaluated by following ISG-020.

APR1400 design specific seismic demands and design data was used.

And, CDFM method in EPRI NP-6041-SL Rev 1 was adapted for HCLPF capacity evaluation.

And, HCLPF capacity over the SSCs for major -- the SSCs greater than 1.67 times CSGRS.

Thank you.

MEMBER STETKAR: I have only one observation, I=ll just make this comment quickly.

When I went through the results that are documented in Table 19.1-44, I noted several instances of nonsymmetric failures that I could not understand. And, I=ll just list the contributors and point you to them.

For example, number 17, 19, 23, 26 and 40 involve turbine driven auxiliary feedwater Pump A. I could not find what I would expect, symmetric failures of Pump B anywhere.

Similarly, scenarios 28, 29, 31, 33, 34 and 44 include seismic -- combinations of seismic and hardware failures that disable Emergency Diesel General B but I couldn=t find the symmetric combinations with A.

So, it just struck me as curious that I didn=t see at about the same number. It=s called a frequency, but the same ranking, let=s call it, those symmetric combinations which may mean that there=s something in the model that didn=t quite get set correctly. I don=t know. I=ll just make that as an observation.

MR. D. LEE: Thank you.

MR. T. HWANG: Yes, shall we move on to the next presentation? We=ve got three more presentations.

CHAIRMAN BALLINGER: Yes, we have what amounts to a hard stop for one of our Members at 11:30. So, we=ll have to find a way to stop then at a convenient place and --

MEMBER STETKAR: The Member can become very arrogant and let the other people wait if we run past.

CHAIRMAN BALLINGER: Okay. Like I said, we have a hard stop for one of our Members at 11:30.

MR. IN: Okay, the next presentation is the internal fire PRA which will be presented by Mr. Greg Rozga.

MR. ROZGA: Again, I=m Greg Rozga from Enercon. We=ll be discussing the internal fire.

I=ve put together this presentation based on the PRA tasks numerically. The tasks aren=t done in order, I kind of have it put together in the order of how you actually do the work.

The first thing we do is we divide the plant into physical analysis units. There=s approximately 390 PAUs identified for the APR1400.

Some of the major highlights is that the auxiliary building has 279 physical analysis units. It=s very highly compartmentalized. And, that results in many, many fires only impacting small amounts of equipment.

So, with respect to fire, the compartmentalized design is actually a good design.

All of these fire areas are -- have rated barriers with the exception of separation that we take credit for for the yard transformers.

And, actually, the DCD says that they will either be 50 foot separation of there=ll be a 3-hour barrier.

And, the low power shutdown model uses the same PAUs. There are some removable barriers. However, those barriers are used for things like if you=re replacing a pump motor or something and those things are very infrequent, rare occurrences and there=s an assumption and a COL item that those barriers will be removed during defueled operations so you don=t have to worry about spreading a fire because of that.

Next slide?

Tasks 2 and 3 are the equipment and cable selection and the at power fire PRA equipment list is based on the at power internal events PRA equipment list with the addition of some nonmodeled spurious operations and they have been screened out because of low probability in the internal events that they need to be added back in.

Same thing with the low power shutdown FPRA. It=s basically the low power shutdown internal events PRA equipment list with the additional cable for the low power shutdown LOCAs. And, that=s the CDCS line, it=s called a JL LOCA, if you=ve seen it in the documentation.

The cables for all the equipment is routed most -- 99 percent of it is based on the referenced plant, Shin Kori 3 and 4.

There is some assumed cable routing, the new diesel generators, the ESW and CCW that aren=t part of the referenced plant. We had to assume cable routing.

And, I=ll just tell you that that is -- that=s not uncommon even in existing plant PRAs. There=s some cables where they just don=t know where it is and so we assume cable routing.

We also -- the type and number of penetrations between the PAUs is from the reference plant.

Cables for new equipment, so the

additional equipment that was in the database but wasn=t part of the internal events, we were actually able to route that based on the reference plant.

Tasks 9 and 10 are the detailed circuit failure analysis and failure mode likelihood analysis.

For our fire PRA, we assumed worst case failure modes. So, if a controlled cable failed, if the worst case is that there was a spurious operation, we would assume the spurious operation occurred.

If the worst case was that the component wouldn=t operate, we assumed that it wouldn=t operate.

There=s a lot of fiber optic cable in the plant between the main control room and the group controllers. And, what=s good about that is that you don=t -- they=re the -- well, you don=t have spurious operations.

Yes, sir?

MEMBER BLEY: Is the worst case the same for all possible scenarios in the PRA or did you evaluate worst case on a scenario by scenario basis?

MR. ROZGA: In most cases, it=s --

MEMBER BLEY: Yes, but --

MR. ROZGA: Yes, yes. For Level 1 PRA, it=s always that, you know, you want flow to happen. So, the worst case would be that you would have a spurious closure.

Now, there is the possibility that in Level 2, the worst case might be that you would later on want to close that valve.

And, I believe we have that covered but that=s something that we can check into and verify.

MEMBER BLEY: Well, just for example, over and under cooling can both affect Level 1. And, but, you assumed loss of flow was the worst case for everything?

MR. ROZGA: Yes, yes.

MEMBER BLEY: Okay.

MR. ROZGA: Yes.

MEMBER BLEY: Eventually, you might want to reconsider that.

MR. ROZGA: And then, for the spurious operations, we didn=t take credit for clearing of the short. We assumed the short happens at a probability of 1. We didn=t do any analysis.

With respect to qualitative screening, there was no qualitative screening done for the at power fire. It was always assumed that you at least have a transient fire in every room and that you will at least have a plant trip.

For low power shutdown, there was some qualitative screening. Because the plant=s already shut down, if a fire in the room does not disrupt your shutdown cooling, then we were able to screen out those rooms.

MEMBER SKILLMAN: Greg, please --

MR. ROZGA: Yes?

MEMBER SKILLMAN: -- go back to slide 52.

MR. ROZGA: Okay.

MEMBER SKILLMAN: Task 9, detailed circuit failure analysis, so you identify for a power cable loss of function control cable failure to operate.

For the power cable loss of function, how does that apply to a valve, say a motor operated valve that only functions 50 percent of its intended direction that is, if it=s to go open and only goes open halfway or if it=s to isolate, go closed, it only closes halfway?

MR. ROZGA: We assume full failure, we don=t assume partial failures. If the function in the PRA is for the valve to change state, open or closed, if the power cable is damaged, we assume that it does not change state.

MEMBER SKILLMAN: What I was really asking is, is there another set of scenarios where the devices don=t fail as you=ve predicted, but they actually fail approximately halfway and you=re only stuck because you can=t -- because of the failure, proceed to isolate or proceed, if you will, to vent or open, you=re stuck halfway?

MR. ROZGA: No, no.

MEMBER SKILLMAN: That=s just not a feature of the PRA?

MR. ROZGA: Yes, yes, correct. Yes, it either doesn=t move to its designed position or it does.

And, even if the fire doesn=t damage it, it may randomly fail. The fire PRA doesn=t only include the fire failures, it also looks at the random failure probabilities.

MEMBER SKILLMAN: Thank you.

MR. ROZGA: Okay, you=re welcome.

MEMBER POWERS: I have a question which the answer is probably, no, we didn=t address it.

But, I=ll ask anyway.

If I have a fire in one of your areas and smoke is distributed into other areas, accumulates at critical contacts, do you attempt to adjust any of your failure probabilities in the PRA for the fact that those contact points and things like that might corrode because of the corrosive nature of the smoke?

MR. ROZGA: No, that=s corrosion from smoke, that=s long-term action. If the fire is in the immediate area, that --

MEMBER POWERS: Not interested in the immediate area.

MR. ROZGA: Right, right.

MEMBER POWERS: It=s the disbursal beyond that.

MR. ROZGA: Yes, yes, we do -- no, no. The answer is just no.

MR. DREMEL: Is that beyond the current state of the art for fire PRA?

MEMBER POWERS: You know, an argument would certainly make -- could be based on that exact argument that it=s beyond the current state of the art. But, it=s the state of the art that I tend to mess with. So, I ask and not very critical that you didn=t take it into account. But, it=s a persistent concern.

MR. ROZGA: Okay.

And, did we do this? We actually already did this one.

Task 5, I don=t know if we noted, FIRM is fire induced risk model. That=s just the PRA model with your fire inputs.

The at power and low power shutdown FIRMs are both based directly on their respective internal events model and then we manipulate that model for a different fire scenarios.

We identify the equipment that would be damaged and we force that equipment failed.

If we impact operator actions, local operator actions that may have to take place, et cetera.

And, the at power fires in each physical analysis unit were assumed to either result in a transient, loss of CD, loss of DC A or B, loss of feedwater LOOP, PLO CCW or a small LOCA and those are all based on the equipment damage in the immediate room that the fire took place.

No other fire induced initiators were

identified. If no PRA equipment or cables in the room were damaged, again, we always assume that you at least have a plant trip, the fire might be bad enough that the operators are going to trip the plant.

And, all fire induced failures are assumed nonrecoverable including offsite power. So, we don=t take credit for any recovery.

The low power shutdown fire induced risk model screen, POS 7, 8 and 9 and they=ll probably discuss this in the low power shutdown. Eight is defueled, 7 and 9, you=re at high water elevation and the times associated with that boil down are extremely long.

We took credit for that same screening that was done for the internal events.

The unscreened low power shutdown fires are assumed to either result in a loss of CC, the JL LOCA which is the CVCS line, spurious operation LOCA, loss of 4KV to the operating train, loss of offsite power, the loss of level control event or just the unrecoverable failure of the operating shutdown cooling train.

I do want to reemphasize for those who are unfamiliar with fire PRA, when we screen something, that fire frequency doesn=t go away. We bring it back in when we look at the multi-compartment analysis.

So, there may be a fire in an area and that area doesn=t result in any of these initiating events, so we screened it from the single compartment analysis.

However, when we do the multi-compartment analysis, we bring all those back and then we say, well, what happens if the fire spreads from that area?

So, screening doesn=t mean it=s completely out. We do look at it again when we look at multi-compartment analyses.

Task 6 is calculation of the ignition frequencies. And, they=re based on, again, generic data. And, that generic data is continuously updating.

We recognize that there=s a new NUREG-2169 that is part of our PRA that we=ll evaluate.

The low power shutdown ignition frequencies are currently based on NUREG/CR-7114. And, for transient fires, there are transient influencing factors and it=s part of this NUREG-6850 methodology where you try to apportion where your transient fires are going to occur.

So, areas were high maintenance areas, areas where you store equipment that might be a transient initiator.

Highly populated areas, well, for low power shutdown, we reevaluated those transient initiating factors. And, as an example, you know, we increased all the containment building transient influencing factors because there=s additional work that=s done in containment.

And, we did a PAU by PAU assessment and we made adjustments to those factors.

Task 12 is HRA. The initial HEPs were estimated using the NUREG-1921 screening analysis and then the top HEPs ranked by F-of-Vs were reevaluated using your normal THRP, CDBMT, whatever is the appropriate methodology.

The fire PRA HRA used the same level of dependency among dependent HFEs. And, the current PRA update is reevaluating all of the HEP -- human failure events using the detailed HRA methodologies and we=re going to be reevaluating all the dependencies.

CHAIRMAN BALLINGER: Okay, I=m not sure

whether this is a convenient place to stop. Say again?

MR. SISK: There are three slides remaining.

CHAIRMAN BALLINGER: Oh, three slides? Okay, continue.

MR. ROZGA: Okay.

These slides mostly do with fire modeling. Without a plant to walk down, you can=t do any reasonably accurate fire modeling. So, we made some overarching assumptions.

We assumed for any single compartment that anything in the compartment burns out every time, no matter the size of the fire, we don=t know the location of the equipment in relation to the fire, so we just assume at time zero, everything has failed.

We do our initial CDF quantification, that=s the Task 7 quantitative screening. And, again, I want to reemphasize that things that are quantitatively screened, their CDF doesn=t go away, we just don=t do additional work on them because the CDF is low enough where you want to spend your resources on your higher CDF areas.

The initial high CDF areas include the

main control room containment, turbine building, and there were about 35 other single compartment physical analysis units.

Again, since we couldn=t do detailed fire modeling, the only thing we did is we took credit for automatic suppression.

There was a generalized assumption that the design of the -- if the suppression system was designed to cover this electrical panel, that it=s -- that the design is correct and it would put the fire out.

And, again, if there was a failure of the suppression system, then we=d go back to a full room burnout.

One other thing I=ll say about suppression, we did credit manual suppression only in cases where we knew there was going to be somebody there. So, we took credit in the main control room. We know what=s continuously manned.

And, we took credit for prompt manual suppression for hot work fires. You at least have the person that=s doing the hot work. There=s generally a fire watch. You=ll also generally have welding blankets or something put out as he=s doing his work.

We also have no knowledge of intervening combustibles. So, when it comes to the multi-compartment analysis, we assume that if the barrier between the two compartments fails that any fire in the exposing compartment is sufficient to fail all the equipment in the exposed compartment.

The other detailed analysis were the main control room where we have to deal with a control room abandonment scenarios.

Containment was high because you have all four trains of instrumentation for RPS SFAS and there=s the potential for small LOCA.

Turbine building, it just has a very high ignition frequency because of the size. There=s also offsite power cables in there.

And, the other 35 were just various reasons they had higher CDF, multi-compartment analysis is also part of the detailed quantification.

Task 13, seismic fire interaction analysis, without a plant to walk down, we just had to do a qualitative analysis based on the current design information.

There=s a total of 480 single compartment analyses, and 1,054 multi-compartment analysis scenarios.

And then, due to the highly compartmentalized nature of the APR1400, the CDF is generally distributed. Only 24 scenarios are higher than 1 percent, 50 percent of the CDF is in the top eight scenarios. Most of that is the main control room because of the conservative analysis that we did.

We don=t have an alternative shutdown procedure at the time the analysis was done.

Next?

And then, for low power and shutdown, there are 918 single compartment analyses and 6,071 multi-compartment scenarios and a low power shutdown scenario is a combination of the initiator and the POS.

The initiating event might change based on the POS. And, like at power results, due to the highly compartmentalized nature, the CDF is generally well distributed.

MEMBER POWERS: I can=t resist pointing out that you assume -- you have assumed away the Brown=s Ferry Fire because of your credit for manual suppression.

MR. ROZGA: Well, manual suppression can fail. It can fail and, if it failed, then the room continues to burnout, correct, yes.

MR. DREMEL: And, Brown=s Ferry was not hot work, Brown=s Ferry was an inspection.

MEMBER POWERS: Close enough.

(Laughter.)

MR. DREMEL: But, from a fire PRA, there=s a big distinction. Transients can happen anywhere, a transient fire source. But, hot work --

MEMBER POWERS: You=re splitting a hair that I don=t even think needs to be done.

MR. DREMEL: We would not have assumed away the Brown=s Ferry Fire because that is a transient. Manual suppression is not credited for transient fires.

MEMBER POWERS: Useful information.

MEMBER STETKAR: I have one -- a couple comments on the main control room analyses, I wanted you to get through the whole thing here, the way those were performed, and make sure that I understand it, is that you accounted for manual suppression in the main control room as you said.

That time window was assigned for a ten minute time window and, that if it was suppressed within ten minutes, abandonment was not required.

If it was not suppressed within ten minutes, you assume that --

MR. ROZGA: Yes.

MEMBER STETKAR: -- people would abandon the main control room.

And, from there, it was just .1 conditional core damage probability without any further evaluation.

MR. ROZGA: Right.

MEMBER STETKAR: I=m -- but, I don=t want to get into why .1 because that=s a made up number.

What I=m more concerned about is the large fraction of the fires that are extinguished within ten minutes, but do damage inside the main control room and could certainly affect subsequent operator performance inside the main control room.

Those fires, as best as I can tell, are simply ignored.

MR. ROZGA: There are -- there were some tests done on the main control room enclosure. I don=t remember the NUREG that it=s documented in.

And, they had their main control room fire scenarios and the result of that analysis was that the driving force in the control room abandonment was obscuration. It wasn=t necessarily heat or heat flux, it was obscuration.

And, it occurred somewhere between 6 and 16 minutes.

The -- that test enclosure was about half of the volume of the APR1400 containment. And, the -- for transients, we actually assumed eight minutes and that=s the time to the peak heat release rate for transient fire for cabinet fires.

We used ten minutes which gets you to about 70 percent of the peak heat release.

And, those numbers fell well within the -- that 6 to 15 minute range given the fact that, you know, the size is smaller.

MEMBER STETKAR: Greg, in the interest of time --

MR. ROZGA: Yes?

MEMBER STETKAR: -- if you=ll allow me to interrupt you, I=m not arguing about what criteria you used for me to leave this room, I=m raising a concern about for the fraction of time when the fire is extinguished, I don=t have to leave the room.

MR. ROZGA: Yes.

MEMBER STETKAR: I stay here, but some fraction of this control panel is now burned.

Now, that burned fraction of the control panel, first of all, I probably can=t use the controls on that fraction to do anything.

Second of all, they may have created spurious signals because of the fire in the control panel.

Third of all, maybe my performance isn=t quite the same as it would have been in a plain vanilla reactor trip.

And, those are the scenarios that I=m concerned about --

MR. ROZGA: Yes.

MEMBER STETKAR: -- is how -- because I don=t see any accounting for those effects where the people extinguish the fire, whether it=s 8 minutes from a transient or 10 minutes for a cabinet, extinguish it, stay in the control room with some degraded either human performance or degraded ability to manually operate stuff or perhaps with some spurious signals from the fire damage within whatever fraction of the console is.

Those seem to be ignored in the model.

MR. ROZGA: Yes, we --

MEMBER STETKAR: I don=t know how important they are, but they seem to be ignored.

MR. ROZGA: Yes. We do impact the operator actions. We do for all. We impact all control room fire actions or, I=m sorry, we impact all --

MEMBER STETKAR: But that=s for fires outside of the control room.

MR. ROZGA: Right, right. And, recall, we use the 1921 screening criteria for most of them and then the top ten.

Regarding the control panels, most of the fires in the containment, there are a couple of control panels and those control panels are generally away from where the operators -- I don=t know if you have a --

MEMBER STETKAR: I don=t want to split hairs on the nomenclature of stuff that contains stuff that controls other things. I know that there are some panels, if you want to call them that, that are physically separated from the, I will call operator consoles, and that I know that you have evaluated fires in those separated panels.

I=m talking about fires in the operator console. I=m sitting here at my console right here and a fire in this thing that I=m sitting in front of, whatever you want to call that, I=ll call it a console.

MR. ROZGA: Yes, and that the electronics in there consist of a PC and a monitor and a mouse and --

MEMBER STETKAR: Never seen a phone burn?

MR. ROZGA: I have not, but I=ve heard that -- I=m sure that --

MEMBER STETKAR: It probably has a DC to DC power converters in it. It=s probably got power supplies for monitors. It probably -- might even have CPUs in it, I don=t know what=s in those consoles.

MR. ROZGA: Yes.

MEMBER STETKAR: That=s only an observation.

MR. ROZGA: Right. And, also understand that if the -- if the computer or the monitor or the phone at that station has a fire that there is --

there is the safety console in the room, there=s -- from the other operator stations, they can take over some of those actions.

And, regarding the smoke, there is -- the control room does have a main control room smoke HVAC system. So, as time goes on, the, you know, the conditions in the control room would be expected to get better.

CHAIRMAN BALLINGER: Okay, well, okay. I think this is a good place to stop for a recess.

We=ll recess until 12:30.

(Whereupon, the above-entitled matter went off the record at 11:41 a.m. and resumed at 12:30 p.m.)

CHAIRMAN BALLINGER: Okay. We are back in session.

MR. IN: Our next presentation is on the Internal Flooding PRA, and it will be presented by Mr. Ray Dremel.

MR. DREMEL: Good afternoon. I=m Ray Dremel with Enercon. I=ll be presenting information about internal flooding, and then I=ll continue on with other external events. So the Internal Flooding PRA, the guidance we use, we try to meet all the supporting requirements of Regulatory Guide 1.200 Revision 2, and the ASME PRE Standard 2009 edition. To the extent possible for a plant that doesn=t exist. We also tried to meet all the requirements for Standard Plan 19.1. For the initiating event frequencies, we used the pipe failure data presented at EPRI-TR-1021086. Next slide.

So the APR 1400 design greatly limits the risk from internal flooding. The auxiliary building is designed to have four quadrants. In the basement, the quadrants are sealed to a level of nine feet for flooding, at least nine feet. So there=s almost no propagation from one quadrant to another. Within the quadrants, there was what was called emergency overflow lines, which are big holes in the floor that pass a lot of water from one elevation down to another. And that maintains any water that might be released an upper revelation within the same quadrant until it gets down to the basement.

Also in the auxiliary building, most of the flood sources are finite volume due to being closed loop. So there=s no essential service water in the auxiliary building. The large volume sources are fire water, domestic water, and raw water, which have very limited flow rates of a couple hundred to four hundred gallons a minute per pump. So if you have large volume, then the flow rate is limited. But that gives you a lot of time to isolate a break.

Also, within any elevation there=s a lot of rooms. So you have a flood in one room, you have big concrete walls just to prevent you from spraying equipment in another room and affecting multiple systems or multiple trains of equipment. The turbine building, it=s isolated from all other buildings. You can=t get from the turbine building to the auxiliary building at grade. It=s a large, open building. There=s a large overflow from the grade level to the outside, which is designed to pass hundreds of thousands of gallons per minute.

The emergency diesel generator building for the alpha and bravo steam generators - again, it=s isolated from other buildings. The flood sources in there are limited. It=s limited to the diesel fuel oil, diesel lube oil, and some fire protection. The compound building has no PRA equipment in it, and there=s very limited potential for a flood in a compound building to propagate - excuse me - to the auxiliary building and cause damage. We do have a few scenarios of that happening.

It=s because we don=t have design of the as-built plan. If we were actually able to go in and look at a plant, we=d probably say you=re not going to propagate across to the auxiliary building. All the water is going to go down. But without having a real plant, we can=t do it. And those scenarios are not significant.

The CCW heat exchanger building, there=s very few active components in there. The only large volume source there is ESW. We have a ESW model. The frequency of pipe breaks is very low compared to the random loss of ESW frequency. The ESW building, that only has ESW in the pumps. Because the pumps are located below grade.

In order for that to pump any significant amount of water that could potentially propagate to other buildings, these large pumps would have to operate submerged under many feet of water for a fairly long period of time. So we just don=t consider propagation from an ESW break in an ESW building through the tunnels to anything else to be credible scenario. And 4KB motors just don=t run submerged under many feet of water.

So defining a flood-induced initiating event. It=s an uncontrolled release of any fluid that also fails PRA equipment. Not just water. It could be fuel oil, lube oil, anything that=s liquid. Steam. We define an initiating event as anything that causes an immediate reactor trip or requires a tech spec shutdown within twenty-four hours.

So if you have a fail in equipment, and you have tech spec say be shut down within seventy-two hours, we didn=t consider that an initiating event if it says be shut down within eight hours. We didn=t include that as an initiating event. We took no credit for recovery there.

When we did the flood propagation, because we don=t have a real plant to look at, we had to make some conservative assumptions. So we took credit for flood barriers to remain intact up to their design level. That=s a pretty safe, pretty standard assumption for an existing plant. We did not take credit for any flood mitigation above the design level. So in some of the upper elevations between quadrants or between rooms, you may have a design flood barrier for six inches of water.

But it=s a concrete wall that bounds a

switch gear room. I would not expect to see any penetrations going through there down low. But because we don=t know, we say as soon as you get to six inches, water propagates across that barrier.

MEMBER BLEY: You mentioned - excuse me, Ray. You mentioned that you looked at all the other fluids besides water. That gas turbine generator, what fuel does it use?

MR. DREMEL: That I don=t know. But that is outside -

MEMBER BLEY: It=s on the outside?

MR. DREMEL: Yes.

MEMBER BLEY: Okay.

MR. DREMEL: And failure of that would not have caused a reactor trip. Or require a reactor trip.

(Laughter.)

MEMBER BLEY: But it=s outside.

MR. DREMEL: Yes.

MEMBER SUNSERI: You also said a couple of times that you don=t have a real, actual plant to go look at. What about Shin Kori or the Amaritz plants? I mean, those are pretty far along. Aren=t they the same physical layout?

MR. DREMEL: They are - I would expect them to be similar. But when we did this analysis, they weren=t in a state that we could have done that.

MEMBER SUNSERI: Okay. Because it=s just a plane trip, right?

MR. DREMEL: Yes.

MEMBER SKILLMAN: Ray, did you confirm - for instance - these assumptions that you communicated?

MR. DREMEL: I=m sorry?

MEMBER SKILLMAN: You communicated that you made assumptions -

MR. DREMEL: Yes.

MEMBER SKILLMAN: In order to complete this work. Did you confirm that those assumptions are communicated as COL items in the DCD? You=ve made assumptions to give you a success path. That success path needs to be communicated into the design control document.

MR. DREMEL: And the assumptions are that the design barriers are as designed. So I don=t believe we have a COL item to confirm that the -

MEMBER SKILLMAN: Oh, for the period?

MR. DREMELL Yes. Then their design basis for the flood barriers doesn=t change.

MEMBER SKILLMAN: Let=s back up. You said we don=t have a plan to look at.

MR. DREMEL: Right.

MEMBER SKILLMAN: But we=re making assumptions.

In my view, that means that the assumptions that you are making must be communicated into the COL items for what will become an as-built plan.

MR. DREMEL: Okay.

MEMBER SKILLMAN: Are those items there?

MR. IN: There isn=t a COL item for the - to verify that, you know. Once the design is in place, it has to be looked at. There isn=t a COL item.

MR. SKILLMAN: Is that an ITECH or just a COL item?

MR. IN: It=s a COL item. Because it=s not only - they have to re-do the assessment. The PRA assessment.

MR. SKILLMAN: Based on what will be the as-built?

MR. IN: Yes.

MR. SKILLMAN: Okay. Thank you.

MR. DREMEL: Okay. Next assumption we made was non-watertight doors are going to fail once water level on one side of them reaches one foot, if that failure makes the situation worse. If the failure of the door helps you, we assume that the doors remain intact. An example of that would be you have a pipe break in a hallway. You have a door that goes down a stairwell within the same quadrant. And you have a door that goes across the quadrants to a different quadrant.

So even though the door to a different quadrant might be up on a six inch curb, if we did not say the door going down the stairwell would fail first. We said, that door remains intact. Therefore, propagation to the other quadrant would occur. So any barrier failures or propagation that would ameliorate the event - we didn=t consider those. We did credit flow through drains, emergency overflow lines, or other pathways to the extent that they are credited in the design basis.

So there=s a certain place where the design basis credits flow through a drain up to a certain gallon per minute. Because that=s a design requirement, we credited that. The flow through those drains is occurring. The emergency overflow lines are basically a big hole in the floor that pass a lot of water just down to the next elevations. We did credit those.

Based on all the accident sequences, we have one hundred and thirty events that we explicitly analyzed. Most of these events are analyzed because they are related -- we assumed that a manual shutdown was required. Some of them we assumed would have caused a reactor trip. High energy line breaks are a unique case, Regulation Guide 1200 and the ASME Standard says you have to treat high energy line breaks in a conservative manner.

So within the auxiliary building, the auxiliary feed water and steam lines are analyzed inside a HELB barrier that=s designed for a complete severance of that line. So those HELB barriers remain intact up until the design of the HELB. There are some other HELB barriers, where you have auxiliary steam that runs or operates intermittently for rad waste processing. And for those, we said if you have a break bigger than the design basis, the barrier is going to fail. It=s going to fail everything in that room and the first barrier to the next room. It=s a conservative assumption, it=s auxiliary steam. But it=s consistent with other places we=ve looked at in the industry. It gives you the insight you need.

We also assumed that any high energy line break will actuate all the fire protection systems in the room. Steam doesn=t give you anything, but the fire protection will turn things off. The CDF we see is two times ten to the minus seven per year. That=s a pretty low frequency. And there=s no one significant event to flooding. Most of the breaks that do contribute to damage are beyond design basis breaks to the fire protection system.

So a design basis break of that pipe is the old standard divided by two times the thickness of the pipe divided by two. And we are going beyond those. We are looking at double ended breaks of the fire protection system. If we had an actual plant to go look at, I would expect the risk would go down quite a bit. Because we can now look at the doors. You know, what level will the doors fail. Where are the holes in the walls? Just because a wall is not a flood barrier, if you have a foot-thick concrete wall with no penetrations in it, no water is going to go across. Or no significant amount of water is going to go across. So if we look at that, I would expect flooding risk to go down.

The other thing that is interesting is that most breaks don=t require isolation. Or there is a very long time available to isolate the break. Each quadrant of the auxiliary building can contain hundreds of thousands of gallons of water before you could potentially propagate - or go above nine feet, where we assume propagation could occur. And that is internal flooding in a nutshell.

MEMBER STETKAR: You didn=t think you were going to get away unscathed, did you?

(Laughter.)

MEMBER STETKAR: Just a couple of observations. When I looked at the results, this is again similar to what I mentioned for the seismic analyses. I noticed that there were cut sets that involved flooding in turbine-driven auxiliary feed water pump room D - as in dog - for that. I couldn=t find any for C - Charlie - I don=t know why that is?

MR. DREMEL: In the flooding, there are physical asymmetries.

MEMBER STETKAR: Okay. That might

explain it. That was for the at-power. And in the shutdown, if I look at tables 19.1-107 and 19.1-108 and 19.1-10, it=s got all of the flooding. About - if I do a rough cut, table 19.1-107 shows that about ninety-eight percent of the core damage frequency in plant operating states five and eleven comes from prior protection flooding. Which is mostly around the plant.

In plant operating state eleven, a very small fraction is in plant operating state five. And the duration of plant operating state five is about three and a half hours longer than plant operating state eleven. And of course, the heat levels are higher in plant operating states. So I was curious, why that asymmetry? That one I couldn=t figure out. That=s just plant operating state, I=m not talking about locations. It=s just this slice. That one I don=t get. So anyway, that=s on the record.

MR. DREMEL: It could be because for low power shutdown PRA, we assume in the first half of the outage they are working on one division. And in the second half of the outage, they are working on the other division. So if I have -

MEMBER BLEY: Of everything?

MR. DREMEL: Yes, yes. So it=s just like consisting with - you have a division one outage, division one part of the outage and a division two part of the outage. So if I have a flood in my division two pump room, while I=m working on my division two pump, it contributes nothing to risk. Because that shut down cooling pump is out of service anyway. When you go to the other half of the outage, if I have a flood in my division one pump room, and my division two pump is out of service -

MEMBER STETKAR: Just be careful, because the tech specs require you to have both divisions available when level is low. I=m talking particularly about five and eleven which were mid-loop.

MR. DREMEL: Okay, but you can work on - there=s other equipment that you can work on.

MEMBER STETKAR: That=s okay. I just raised it as an observation. I don=t know the answer.

MR. DREMEL: Okay. Now I can move on to other external events. Analysis of most of the other external events is identified as a COL item or COL items. The external events considered identified in DCD chapter table 2.0.1. For transportation accidents, there=s a COL item to confirm that they are not risk significant. Obviously, we don=t have a plant location to analyze. We don=t have a site to analyze.

Turbine missiles were looked at. The APR 1400 has a favorable orientation for turbine missiles. In DCD chapter 3.5.1.3, the probability of two point one times ten to the minus nine per year of a turbine missile was determined based on a twelve-year inspection interval. The events that were analyzed were high winds, including tornadoes. A design basis tornado was two hundred and thirty miles per hour. That=s based on region one of Reg Guide 1.76, region one. That tornado has an exceedance frequency of ten to the minus seven per year. So we screened that out as a conservative screening.

Similarly, design basis hurricane is two hundred and sixty miles per hour and per Reg Guide 1.221, every place except Southern Florida - that hurricane has an exceedance frequency of less than ten to the minus seven per year. High winds are not considered a problem. Next.

COL items identified for the COL applicant to do. They have to do a site-specific risk assessment for the events on the left. And then for the events on the right, they just have to confirm they are not an outlier. You=re not going to build a nuclear plant next to an unstable mountain or next to a volcano. So there=s a COL item to confirm these events. Based on that, we consider that risk from other external events is going to be a negligible contributor. That=s all I have for other external events.

MR. IN: The next presentation is on the low power end shutdown PRA. That will be presented by Mr. Kim.

MR. J. KIM: My name is Jae Gab Kim from KEPCO E&C. I=m going to discuss low power shutdown PRA. A key document to follow for shutdown is called regulatory industry support. Associated with the NRC=s report. Every PRA report is also reported through the shutdown initiating event. Next slide.

These are touched on in low power shutdown. And singularly, the power of plant operating stage development. So detailed analysis has been performed for POS development.

MEMBER REMPE: On your selection of the success criteria. I was reading up on it, and I guess you put 1300 degrees after the peak clouding temperature instead of 1340. And the basis of 1340 was attributed to an ASME standard, which I didn=t have access to. But to the Reg CR report that you cited in the prior slide. If I pull that string, it=s a Surry analysis and it was the clouding temperature where you would have core damage within a short period of time. So basically, that success criteria is based on a Surry analysis if I=m understanding this philosophy. What gives you confidence that forty degrees difference is enough to have for that success criteria?

MR. J. KIM: As you said, and as in the standard as in the Reg CR inspection manual report. 1300 Fahrenheit is from the report. But these evaluations, that's something that's assumed. 1300 Fahrenheit.

MEMBER REMPE: I=m sorry. I=m having trouble following. Maybe a little slower and louder.

MR. J. KIM: This must be a little continuity assumption. 1300 Fahrenheit is much lower than 1340 Fahrenheit. So, this PR group time is much shorter than 1300 Fahrenheit.

MEMBER REMPE: So if you think forty

degrees is sufficient conservatism, because I would assume APR 1400 is a much higher power plant than Surry.

MR. DREMEL: There are also some NRC inspection manual chapters out for shutdown risk assessment. And they reference the 1300 degrees as - keep your temperatures less than 1300 degrees, you should be okay for shutdown. And what we found when we did the success criteria runs, 1300 is here. You=re either way down here, or you=re way up here.

MEMBER REMPE: Okay.

MR. DREMEL: So it=s really - you can debate, but -

MEMBER REMPE: Okay. I was not sure. I didn=t have time to go look through those inspection manuals. So that=s comforting that apparently they across the board said 1300 Fahrenheit is fine. And then you=ve done some analyses that make you feel comfortable.

MR. DREMEL: Yes. Well, there=s no place where we got, you know -

MEMBER REMPE: Real close?

MR. DREMEL: Yes.

MEMBER REMPE: Okay. Thank you.

MR. J. KIM: Next slide. To decide POS division, make use of prior experience with current and next generation analyses. At this table, it=s just to show the Reg CR report. The total number of POS is fifteen. Next slide.

CHAIRMAN BALLINGER: Excuse me. We are either being serenaded by the workout in the gym below, or there=s somebody on the phone. We are being serenaded? Okay, we can=t do anything about that.

MEMBER CORRADINI: It will stop at one o=clock.

(Laughter.)

CHAIRMAN BALLINGER: Keep going.

MR. J. KIM: Okay. This table is APR 1400 Plant Operating States and states definition. Total number of POSs is fifteen, which determined the base they=re on. Our primary system water level and the pressure and the temperature and the TS mode. Which is related to the substance criteria, as related to the variable time after event. And the system arrangement. Next slide.

APR 1400 Initiating event has been determined in this table. Most of the data is from

Reg and NS data, but some specific initiating event from the two, the initiating -- and the APR TR report 1003113. Next slide. So an appropriate combination of generic and design-specific event frequencies used.

MEMBER STETKAR: Can I stop you here? I have several comments on low power shutdown. The first one deals with partly this topic. You have the only event models really documented in the DCD are for plant operating states five and eleven. So I=ll only speak to them, since I know nothing about the others.

There are statements in the DCD that says that one train of shutdown cooling is operating and the other is in standby. And there are also statements in the systems= analyses saying that no changes were necessary for complement cooling water or essential service water from the full power PRA models to the low power and shutdown PRA models. If I look at sections of the DCD. For example, 5.4.7 in the DCD, it explicitly says that in the early part of the outage, two train cool down. Assuming two trains are in service in the early part of the outage, that seems to extend through plant operating state five.

There are things in the DCD that explicitly say that in the early part of the outage - this is in chapter nine, section 9.2.2.2.4.2. Four component cooling water pumps in 9.2.1.2.3.2 for essential service water pumps are operating. That is different from the full power model, and it=s different than the configuration that you used in plant operating state five.

Now, why do I bring it up now? Well, you have initiating events that says the normally running train of shutdown cooling fails. Or it=s interrupted. That=s S1. And the normally running train fails, that=s S2. If both trains are running, those frequencies are much different. The consequences are much different. The recoverability is much different.

So now I=m confused about what is the actual configuration of running in standby equipment in each of the plant operating states? All the way from plant operating - every plant operating state. And I don=t know what they are. I=m only left with things that are contradictory between the PRA and other parts of the design certification.

MR. DREMEL: One point is, and you

mentioned section 5.4.7 that talked about a two train cool down. So you need - the DCD analyzed as needing two trains to remove decay deep and cool down so that you can get into refueling. But I believe one train is adequate to remove decay heat.

MEMBER STETKAR: I will give you the quote so we have it on the record. The shutdown cooling system or SCS reduces the RCS temperature as follows. Two train cool down, normal operation. I can continue to give you all of the temperatures. I can continue to give you all the way down to 120 degrees Fahrenheit within ninety-six hours. If I look at the timeline for the plant operating states, that takes me out into plant operating state five.

Now if that=s normal operation, and I=m in the power plant, and I=ve been in power plants, we normally like to get cold kind of as quickly as possible to get the outage started. So we ran pretty much everything that we could do to get cool down. Not so much at the end of the outage. So it might be different in eleven compared to five. But at the front end of the outage, a lot of plants have pretty much everything running. I=m just making the observation. That, by the way, affects both the initiating event frequencies. You can=t use generic initiating event frequencies because there is no such thing.

It also affects the recoverability because if I now have a common cause failure that affects both of my operating trains, I can=t recover shutdown cooling. I have to go to feed and bleed cooling and so forth. It affects time windows, I mean it affects everything. So I=ll just make that observation. The other sections, chapter nine, also indicates that it=s normal operation during the initial cooling.

MR. J. KIM: Okay. Next slide. This slide is the Accident Sequence Analysis. The AS analysis models the combinations of system responses and operator actions that could occur during the event. Event Tree is used to delineate these combinations to present these events. This diagram is one example of an accident sequence. Next slide.

Success criteria is the ability to be using MAAAP 4 and RELAP5. Considering the initiating event, limiting plant conditions for each POS, and equipment availability specified for each accident sequence. And the core damage temperature for the

RDR are 1300 Fahrenheit. Based on the ASME PRA Standard and NRC Inspection Manual. Next slide.

At this Level 2 analysis, for POSs with RCS and containment intact, Level 2 conservatively estimated using the full power conditional probability of large release. For POSs with RCS intact but containment hatch open, failure to close hatch assumed to be large release. The successful closure of the hatch before boiling evaluated using full power CPLR.

And for POSs with RCS head removed, detailed Level 2 PRA developed. Also for portions of the analysis, the full power Level 2 methodology are considered conservatively. And LPSD Level 2 Fire modeling are the same as internal events.

MEMBER STETKAR: Now, there=s an assumption in the PRA that says isolation or containment demonstrations is assumed to be identical to the containment isolation modeling in the Level 2 model. I will tell you that I=m pretty darn sure that during plant shutdown modes, the large containment, high volume containment purge is operating. That certainly is not modeled in the full power PRA model.

I=ll make the same comment that I made this morning. If that is operating and is not isolated, that=s a big hole. That=s a much bigger hole in the side of the containment which has both of the effects that I mentioned earlier. That it=s good that you might not get an over pressure failure of the containment. It=s bad because it might be a contributor to large releases. So, I=m kind of questioning this notion that you didn=t have to change the containment isolation models from full power to low powering shutdown.

MR. J. KIM: Could you show me that?

MEMBER STETKAR: Sure.

MR. J. KIM: Could you give me more detailed information?

MR. LEARY: Jeff Leary with Enercon. The statement you made is true. It=s something we will take a look at. The additional contribution from it is going to be an additional line out of -

MEMBER STETKAR: It=s not necessarily an additional line. Remember, this is a big line. So it=s a big enough line - certainly this one is big enough. I don=t know its physical size, because I couldn=t find it in the DCD. But I know it=s big.

And it=s certainly big enough to exhaust containment heat. If that line is open, there=s no way you can ever get an over pressure failure of the containment. It just is not going to happen.

That=s good - you know, that=s the good news part of it. The bad news is if the line is open, everything=s going out. So it=s different than any of the other small water isolation lines that also have to be isolated to satisfy whatever the containment isolation criteria are. That one behaves differently.

MR. LEARY: But your statement is true, that the LPSC model used the same containment isolation model as the F power.

MEMBER STETKAR: Okay. Thank you.

MR. J. KIM: To continue. Results are dominated by operator recovery failures. And our results indicate, as expected, that the draindown and reduced inventory POSs are highly risk significant. During power shutdown, you cannot use secondary generator, and also the signal is bypassed.

MEMBER STETKAR: The two comments I=ll make - and I=ll try to be quick - well, I have to make three. Because results indicate, as expected -

the results will always indicate, as expected, if that=s where you concentrate all of your activity. Sometimes people are surprised when they look at other plant operating states and find out that they are more important than the mid loop operations. There have been studies that have found that.

So as expected is a warning to me that says, well, we thought these were going to be most important. That=s why those are the only ones that I can see in the DCD. And that=s why you concentrated all of your effort there. That=s sort of philosophical.

Two comments that I had on this notion of the importance of operator actions - I have a lot of comments on the Level 2 models. But when you reevaluate, and I heard earlier that you are reevaluating all of your HRA for the next update. There were several scenarios when I could not understand the relative timing and the success criteria for operator actions.

In a particular operator action MI for makeup or isolation of a drain down path, I had no idea how much time was available for the operators to do that. Some things led me to believe that it was four or five minutes. Some things led me to believe that it was a couple of hours. That needs to be clarified and make sure that the HRA is crisp there.

There is one combination of things that I was especially puzzled by. And that is in the Level 1 model. There is an operator action - it=s called feed and bleed, but it=s basically makeup to the primary system when you don=t have shutdown cooling. It=s put more cold water in and you can call it feed and boil if you want. Or you can call it feed and spill - anything. It=s that sort of thing.

And there=s a statement in there that says that the available time window for that action is 2.2 hours based on the start of core damage. Then in the Level 2 model, there is a top event called melt stop, which requires that the operators start putting water into the vessel to either prevent core damage or stop a melt in progress. It=s not quite clear to me which of those two apply. And there is separate credit for that.

Now if I as an operator have sat around for 2.2 hours and have not decided to put water in the vessel, and that=s the amount of time before the start of core damage, why am I suddenly going to get really really smart in a relatively short period of time and start putting water in from the same sources. So I was really puzzled. Because there are two distinct actions that happen to be separated by this artificial Level 1 and Level 2 split for the same people putting the water in the same place with the same pumps.

(Laughter.)

MEMBER BLEY: It=s the kind of thing that if a shift change had occurred maybe that=s -

(Simultaneous speaking.)

MEMBER STETKAR: This is 2.2 hours. You know, fortuitous maybe.

MEMBER BLEY: Twenty-five percent chance of a shift change.

MEMBER STETKAR: Okay.

MR. LEARY: Jeff Leary with Enercon again. The action that you=re referring to in the Level 2 part of it is taking credit for additional indication that would be occurring with the SAMGs when core exit thermal couples reach 1200 degrees Fahrenheit. Taking the credit for additional cues and information that would not have popped up sooner. So there=s a dependency analysis between the actions.

MEMBER STETKAR: Yeah, my dependency would have been one, but the problem is the story about FB which is the initial makeup. It also mentions core exit thermal couples and the onset of core damages, the critical condition that would be the end point of that time window. So it=s hard - in principle, I could wait for 2.19 hours and still win because the core exit thermal couples start to go up for the feed and - whatever we want to call it - feed and boil, feed and whatever.

MEMBER CORRADINI: Since John finds these interesting things, is this just a mechanical - a bureaucratic separation of Level 1 and Level 2?

MEMBER STETKAR: I believe that it is.

MEMBER CORRADINI: That=s what it sounds like.

MEMBER STETKAR: The problem is the stories - again, if I just read the stories about these actions, the story about melt stop, the later one says, well yeah. We recognize that early on in the event, people could have tried to make up and they might not have, but there would be additional cues. I can buy that, except for the early action seems to consume the entire time until core damage begins. And that time is important, because the time affects the human error probability for that initial action.

MR. LEARY: I can say that the dependency was evaluated and it=s something that is -

MEMBER STETKAR: I know you said that. I just wanted to raise - that one bothered me in particular. The other one that I mentioned about time available for feed and boil in some scenarios - I call it feed and boil to distinguish from the other thing. And for make up - it=s called make up and isolation, but there are very few that you can isolate. So basically getting water in before something undesired happens. Those time windows, to me, were not documented very well.

And I just wanted to raise those because you said you=re re-doing the HRA. And in terms of flags that they raise to me where the HRA people might need some more clarity. The second one being a Level 1 to Level 2 issue. The first one all being kind of Level 1.

MR. LEARY: Okay, thank you.

MEMBER STETKAR: Thanks. And that=s important, obviously because as you=ve mentioned, HRA is the whole story on the low power end shutdown. Thank you. Sorry.

MR. IN: Thank you, that concludes the presentation - all presentations for 2.1.

CHAIRMAN BALLINGER: Thank you. Thank you. We=ve got a change in the schedule in that some people have flight schedules this afternoon. And so what we=re going to do is to continue with KHMP to do chapter 19.2. And then we=ll so the staff presentations in order.

MR. SISK: Rob Sisk, Westinghouse. Thank you, Chairman. If we can, that would be very helpful. We=ll call our people up and be ready to go in just a minute.

CHAIRMAN BALLINGER: Okay, so you=ve just go to change out?

MR. SISK: Exactly.

(Simultaneous speaking.)

MR. B. KIM: Good afternoon, ladies and gentlemen. My name is Byung Jo Kim from KEPCO Engineering and Construction Company. Before I begin, let me introduce my co-workers. In Chul Ryu, he=s the Team Leader of the Central Analysis Team in my company. And Dr. Chan Y. Paik from the Fauske &

Associates in Chicago. He is the Technical Consultant for this project, the main calculation.

This morning and in the previous session, we discussed the probabilistic risk assessment for a very wide spectrum. This session, I would like to talk about the Chapter 19.2, Severe Accident Evaluation from the deterministic viewpoint. Here is the section overview.

Today I have four technical topics. First one is the severe accident prevention design. The second one is the severe accident mitigation features, and the deterministic evaluation method origin, why there are assumptions for the evaluation and the variation. The third topic is containment for MELCOR and analysis, it will be discussed. Next we will talk about the severe accident management framework, and a short summary will be given at the end of my presentation.

Severe accident evaluation is performed to confirm to APR 1400 design, with the relevant guidance such as the SECY 93-087 and 10 CFR 50.44 and Reg Guide 1.216. Today=s first topic is the severe accident prevention design. The APR 1400 is designed to prevent severe accidents from the anticipated transient without scram by digital safety system and diverse protection system.

Severe accident initiated from the mid-loop operation can be prevented by instrumentation or shutdown operation. Shutdown cooling system design, steam generation nozzle dam integrity, and alternate decay heat removal method. Severe accident following station blackout will be prevented by alternate current, starting alternate AC, and manually aligned to provide power to Class 1E 4.16 kV when EDGs fail.

Fire detection, automatic and manual fire suppression and fixed fire prevention are designed in APR 1400 to prevent severe accident following the fire incident. Intersystem loss of coolant can be recorded at safety injection system, shutdown cooling system, chemical and volume control system, containment system, and et cetera. Because all sections of this system and interfaces are designed to withstand full RCS operating pressure or have a leak-test capability, valve position indicators in the control room and high-pressure alarms to warn the operators.

So CVCS from ISLOCA can be prevented in APR 1400. There are other features incorporated in

APR 1400 to prevent severe accidents, including two independent of turbine-driven feedwater pumps, when AC power is not available. And shutdown cooling pumps can be used as a backup of containment spray pumps during a LOCA event. Feed and bleed operation using safety injection system and pileup operating safe repairs.

Second topic is severe accident mitigation to keep up of today=s presentation. I already discussed the overview of containment design in terms of severe accident management. And severe accident progression, both in-vessel and ex-vessel. Then I will introduce design features equipped in APR 1400 and the performance variation as a result of those features.

Containment is the role of the rest of the severe accident - so it is a more simple kind of structure in the severe accident mitigation. APR 1400 containment is concrete structure with a cylindrical and dome part. Concrete second is 4.5 feet and 6mm thick steel liner plate is installed on the inside of the dome and cylindrical wall to prevent leak-tightness as on the basemat concrete area.

Design characteristics of the containment in terms of severe accident management is that it has a large free volume and dry-type containment. In order to accommodate the condensable and non-condensable gases generated during the severe accident. Also inside the containment, natural mixing is achieved throughout the containment atmosphere.

Design pressure limit of the containment should be designed to meet the severe accident internal pressurization challenges. In other words, design pressure limit of the containment should meet the factored load category criteria, as noted in Reg Guide 1.2016. I will discuss this issue later.

MEMBER REMPE: So is this good time to bring up this question about is there anything special in the cavity? I think the answer is no. It=s just that when I look at the drawings of the containment building. This thing about a debris trapper or a core debris chamber. That=s just an area that=s - there=s nothing special in that area, right?

(Simultaneous speaking.)

MEMBER REMPE: Okay, thank you.

MR. B. KIM: Here you can see the key phenomena in the progression of severe accident for

APR 1400 design. So while there is an initiating event, in the surface begins with insufficient indications. Then the cores start to heat up, creating oxidation. Fuel marking and COL information in the lower half, and finally, direct result failure is inevitable if operators recovery has been failed.

Regarding the vessel failure mode, five mechanical events are integrated in the severe accident code MAAP, such as ejection of a penetration tube, creep rupture of the lower head, and attack of vessel wall of overlying metal layer.

MEMBER REMPE: So again, I guess I=d like to interrupt you here. In this accident analysis report, there=s some tables in Appendix D that carefully detail the corium composition and mass in the lower plenum for different types of vessel failure evaluations. And so I know how much is in the lower plenum, but I don=t know how much of the material went ex-vessel. For example, if you had this attack of the vessel wall by the overlying metal layer, would you just release the metal and keep the UO2 materials within the vessel? If you have a penetration at the bottom, everything goes out? Is that a true assumption? Because I could not find that, but maybe

I missed it in all the documentation.

MR. B. KIM: Right. Yes, you are correct.

MEMBER REMPE: Okay, good.

MR. PAIK: This is Chan Paik from FAI. Whatever the corium debris, above the failure location. Elevation is locating as an initial pressure failure. And then we could have subsequent to creep rupture of the lower half. Whatever is remaining still heats up. So that can have later failure. And then later failure, we assume that failure will cause at that bottom of the vessel and that will locate the rest of the material in the lower plenum.

MEMBER REMPE: Okay.

MR. PAIK: But we can still have some material left in the core.

MEMBER REMPE: Okay. So I=m not sure I found anywhere that told me how much is in the vessel and ex-vessel in some of these analyses.

MR. PAIK: There were most sequence, and here is no recovery sequence. So eventually all the corium relocates.

MEMBER REMPE: Good.

MEMBER CORRADINI: But this has the IVR

methodology. You just allow for failure of the IVR.

MR. PAIK: The IVR is not applied in these sequences.

MEMBER CORRADINI: But it=s there.

MR. PAIK: It=s feature is there.

MEMBER CORRADINI: But you don=t consider it?

MR. B. KIM: Yes, in this variation.

MEMBER REMPE: Well, that=s something I=d like to discuss later. They don=t take credit for it. But then it is there, so I=m wondering if you can have some issues because the insulation around the vessel could collapse and you might - I=m not quite sure how you - I mean, you say you don=t take credit for it. But it=s there. Do you ever consider adverse effects because it=s there? You know, like the AP 600 and AP 1000 reinforced the entryway for the water to come in. So did you consider unintended aspects of it? You don=t take credit, but that was something I was going to bring up later. But it is kind of a nuance that they don=t take credit for it. But it=s there.

MEMBER CORRADINI: I don=t remember. But there was some previous certification. If I knew who it was, I probably couldn=t say it anyway. But there was a previous certification where the coming applicant basically said that their mitigation measures were not credited. And they showed that with the presence of the mitigation measures, it didn=t make it any worse than essentially ignoring. That=s what I think is being said here.

MR. PAIK: Similarly in APWR, even though they -

MEMBER CORRADINI: In which one?

MR. PAIK: APWR.

MEMBER CORRADINI: Oh, okay.

MR. PAIK: Yes. They have similar features, but not critically.

MEMBER POWERS: Why does the metal float over the uranium dioxide?

MR. PAIK: The current - when you have -

MEMBER POWERS: Can you show me a single experiment that=s ever been done to show that it happens?

MR. PAIK: I think there is some experiment that shows that light matters. It=s kind of moving upward. MASK experiment.

MEMBER POWERS: MASK experiments were

explicitly done to show that it did not.

MR. PAIK: I have to get back on that with the particular experiment that has that.

MEMBER POWERS: Only when they deliberately constructed the oxide phase to have no - to be hyper-stoichiometric could they get the metal to float. How do you guarantee that your oxide phase is hyper-stoichiometric? When you=ve got zirconium metal that=s incompletely oxidized?

MR. PAIK: The current model assumes the light metal layer is floating above oxide.

MEMBER POWERS: What are the consequences of being wrong on that assumption?

MR. PAIK: If you credit the in-vessel or ex-vessel cooling, then that becomes an issue. That was one of the reasons this ERVCs -

MEMBER POWERS: If you look at the heat transfer to the ex-vessel cooling, how thin does the wall have to be in order to get a boiling flux outside?

MR. PAIK: I think it=s typical when the ERVCs are available, the vessel wall can go down to two to three centimeters.

MEMBER POWERS: So any kind of -

MR. PAIK: High pressure sequence -

MEMBER POWERS: Any kind of collapse in material, it will fail?

MR. PAIK: I mean, as long as there is two to three centimeters, it is still strong enough to retain the corium in low pressure.

MEMBER POWERS: Yes, but if the internals collapses?

MR. PAIK: The internal plenum still is a relatively cold.

MEMBER POWERS: What I=m asking is, that=s a pretty good radiation heat flux coming off the melt up into the upper internals. If they collapse and hit the bottom of the vessel when it=s only two or three centimeters thick, that still holds together?

MR. PAIK: If you do the ERVC, then corium temperature can get very hot and that radiation can raise the internal temperature. And that kind of thing potentially could happen. But at least in APR 1400, the ex-vessel cooling is not credited.

MEMBER CORRADINI: So can I ask Dana=s question a little bit differently? So the orientation of where the metal is compared to where the oxide is - since you=re not crediting the ex-vessel in-vessel retention, it=s of no consequence?

MR. PAIK: Right. It=s not that important.

MEMBER CORRADINI: That=s what I think what you were trying to get at. The uncertainty - the stuff comes out regardless. Because they are not crediting and thus a retention.

MEMBER POWERS: The failure modes they have explicitly recognize the orientation. If you don=t have that orientation, you=ve got a different problem. And for instance, if I have a metal melt streaming on the lower head, it fails instantly. If I fail because of the metal attack low instead of high, drain everything out instantly. It=s a little different problem.

MEMBER CORRADINI: Right. But I=m not answering for them. On the other hand, what I thought Chan was saying is that given that they didn=t credit the in-vessel cooling, it all comes out eventually. The rate would change as to what comes out first. Is that your point? And the chemical reactions are related with that rate.

MEMBER POWERS: Yeah, you have a different situation. And I=m uncertain of what the consequences are.

MR. B. KIM: Upon vessel failure, the MAAP progression moves to the ex-vessel with the following key parameters. RCS suppression, corium vessel failure mode and timing, corium releasing characteristics, cavity floor concrete type, availability of cavity flooding at the time of vessel failure. And during its vessel phase, the various events can cause the containment failure, such as the high pressure melt ejection and direct containment heating, ex-vessel steam explosion, molten core-concrete interaction, and hydrogen combustion.

Ex-vessel, EVSE, contains considerable core uncertainty. In order to reduce the uncertainty related to the ex-vessel event, the following approaches are applied in severe accident evaluation. Ex-vessel steam explosion, initial conditions are established for the realistic case. The sensitive cases need bounding parameters.

For direct containment heating, the sampled input is prepared by Latin Hyperbolic Sampling technique. And for molten core containment and hydrogen risk, we use conservative input to increase concrete ablation depth and hydrogen generation. And commonly, for the selection of the ex-vessel sequences, we used a combination of probabilistic and deterministic approaches in the variation.

MEMBER POWERS: When you did your DCH analysis, you say sampled input values. I=m unclear what you mean by that. What input values do you sample?

MR. B. KIM: There are known key parameters. Each has a dominant effect on the DCH measurement, such as the ICS pressure or the containment -- as shown. And we can determine the allowable band of each parameter. The pressure is from the low-band and higher-band, which is available in the APR 1400 -

MEMBER POWERS In the case of this particular -

MR. B. KIM: We collect random data from each parameter and prepare it for than more than one thousand data points. And we variate these randomly established input - like that.

MEMBER CORRADINI: Can I ask Dana=s question a little differently? There are some old experiments done at Argonne and then at Sandia about co-ejection of melt with water. Forget about the pressure. So I=m trying to decide what happens with the presence of water in the cavity. And is that considered in the DCH? Is that considered?

MR. B. KIM: No.

MEMBER CORRADINI: Okay. Because if memory serves me, it kind of matters.

MR. B. KIM: I=m sorry?

MEMBER CORRADINI: The presence of water matters. So if you have water there, it kind of turns into a pressurization event, if I remember the old Argonne and Sandia experiments. That to me would be an interesting input variable, or variation that I=d be curious about. I mean, to put it in a different way, DCH dry - interesting but of no consequence.

DCH with water -

MR. PAIK: I think that these rapid steam generation due to these ejections can raise the pressure. But still does not reach containment failure.

MEMBER CORRADINI: Okay. So that bounding calculation was done?

MR. PAIK: I don=t know if it was

officially done.

MEMBER CORRADINI: Okay.

MEMBER KIRCHNER: Well, you=re trying - you know, the natural thing you=re trying to do is get as much water into these scenarios as possible. So there=s going to be a large amount of water to generate steam. And then, with your reactor vessel melt through, it doesn=t have to be a violent result and a fairly significant pressurization of the containment.

MEMBER POWERS: I=m not even talking violent. That=s why I asked if a bounding calculation was done.

MEMBER KIRCHNER: Yeah, if they bounded that.

MR. PAIK: For the containment performance calculation actually, we used a mechanistic calculation of what those re-entering the pool and generating the steam and hydrogen. And then the effect on pressurization. That was considered.

MEMBER POWERS: It seems to me that for this particular design, for the direct containment heating, a dominant uncertainty is the amount of expelled core debris that gets ejected up around the vessel and into the dome. And it seems to me that that=s a fairly complicated process. If you estimate that transport, why wouldn=t you sample the parameters affecting that? Rather than just the input parameters?

MR. B. KIM: The approach used in the DCH evaluation is by following the Reg Guide. So this is talked about later. Again, I can give you more detailed information at a later time.

MEMBER POWERS: What I=m driving at is would you have sampled things - of these things, how much melt gets expelled? What metal fraction of it is, what the driving pressure is? And those are all admittedly uncertain, and you can formulate some sort of distribution. And the nice thing is, nobody=s going to be able to prove you wrong on those. But that=s not what dictates the pressurization for you. What dictates the pressurization for you in this is how much pre-existing hydrogen you have, and how much of the debris comes up around the vessel and goes up into the dome.

Because it=s only that debris that goes up in the dome that can fully impart its energy to the pressurization of the atmosphere. Now that process of expelling debris and having it come up into the dome region involves particles bouncing off things, going up through channels and what not. There are lots of things that I personally don=t know how to mechanistically calculate. So if I were having to do it, I would have to do some sort of an uncertainty analysis or bounded like Mike does on everything he ever encounters.

(Laughter.)

MEMBER POWERS: In some way, look at a range of things. But apparently, you only sampled over the inputs and nothing associated with that discharge up into the dome region. And I=m just trying to understand why?

MR. PAIK: I don=t remember every detail of the DCH calculation. But actually how much can go through the analysis to the dome - we probably have done some analytical calculation to provide us some of those values.

MEMBER POWERS: I would think that would be a first order in importance. Because, I mean, sampling the input values - I think that=s great. And I=m sure you put in distributions that we can argue over until the cows come home and it won=t make any difference. It=s fine, whatever you did. That=s not the crucial thing. The crucial thing is how much energy you put into the atmosphere, and that seems like it=s challenging. And I think, particularly for this design, it=s crucial.

MEMBER CORRADINI: I=m trying to find the Sandia report, but Marty Pilch did a series of Sandia reports for the NRC to try to think of these effects that Dana has mentioned. So that=s, I guess, is where I would start.

MEMBER POWERS: Yeah, I mean that=s certainly exactly right. That=s where I would start. They=ve done their time pressure, and they spent all their time looking at Westinghouse designs and things like that. Here, you=ve got a substantially different situation because what=s in there is different. I mean, there=s just a lot of things that are different.

And what goes on below the operating deck really doesn=t matter for containment pressurization. It=s what you get up into the dome. And the amount of pre-existing hydrogen that you have up there that can get ignited. In your case, you might not have very much because you=ve got your igniter, your passage systems, and things like that. So you=re in good shape to start with.

MR. PAIK: The methodology certainly follows the Sandia results.

MR. B. KIM: Okay now, let=s move to the mitigation features of APR 1400. First to be discussed here is hydrogen control systems. Hydrogen control system is designed to accommodate the hydrogen generation from one hundred percent metal water reaction and to limit the hydrogen concentration less than ten percent, as required in these two criteria.

The mitigation features of hydrogen risk is - the first one is containment. The second one is pressure recombiners, and the third one is igniters. Containment tests large free volume, as I told you previously. Rather than the three million cubic feet. And thirty PARS and eight igniters is installed throughout the containment with the seismic category 1 requirement.

MEMBER POWERS: When you think about your severe accident, and you say gee - I=ve got melt down interacting with concrete. And concrete always has a certain amount of gypsum in it. So I=m getting sulphur-bearing gasses coming off of that. Do you look at the poisoning by the PARS by those sulphur-bearing gasses?

MR. PAIK: That particular thing was not considered. But the effectiveness of the PARS would be reduced to about fifty percent. Yes. The base would be twenty-five reduction of PAR capability. So in order to consider this time of consulting -

MEMBER POWERS: What I=m - I mean that=s great, except I don=t know where the twenty five - why wouldn=t it be one hundred percent if I=m putting up volumes of hydrogen sulfide? If you=ve ever done a melt concrete experiment, you know they stink of hydrogen sulfide.

MR. B. KIM: Yeah, I don=t remember seeing the actual PAR data using those things yet. But instead of a mechanistic model, the chemicals reduce the effectiveness of PARS. It=s one way of trying to address some of our uncertainties.

MEMBER POWERS: Well, the difficulty I have is reducing - if the potential reduction is one hundred percent and I take twenty-five percent, then I haven=t done anything. That=s the dilemma I run into. I certainly don=t know. I=ve never put a PAR in the presence of a melt concrete interaction. And it may well be, though unlikely in this particular geometry, that the hydrogen all burns up the melt concrete interaction for all I know.

But I do know - what I absolutely know - is that most of the PARS that are used, that are proposed for use in nuclear power plants were originally developed for use on diesel engines. And they had to require people use low-sulphur diesel, because the sulphur irreversibly poisons the palladium-platinum alloy.

MR. PAIK: In APR 1400, when you have an MCCI, we still have about seven or eight meters of water pool on top of the corium. So whatever the off-gas from MCCI and aerosol generation has to go through these seven or eight meter of water pools. And most of them, especially aerosols, will be scrubbed. Some gas can escape.

MEMBER POWERS: I would recommend that you bubble hydrogen sulfide into water and sniff over the top.

MR. B. KIM: Three, hydrogen control system performance. We applied MAAP 4.08 code. And with highly probable sequences from PRA Level 1 study is represented deterministic sequences including five major initiatives. Such as three LOCAs and station blackout, and total loss of feedwater.

Regarding the hydrogen source, besides the hydrogen mass, you can bound it to the one hundred percent MWR inside the in-vessel, as criteria requires. An additional generation of hydrogen during each vessel phase, such as the MCCI were also considered in the variation.

MEMBER REMPE: So in your accident analysis report, you discuss the sub-nodal method that=s in MAAP. Could you talk about that a little bit? Because you mention that seems very helpful in matching the data such as the HDR tests. I also appreciated the fact that you said, although this matches here, there=s a lot of uncertainty when we extrapolate to a large scale facility or containment. But could you talk a little bit about that? Because I think MAAAP or melt core doesn=t have such a sub-nodal method, and so I was curious on how it helps.

MR. B. KIM: Right. MAAP has two or actually three models. The typical lump - the parameter core doesn=t have. One is, it has a counter-current flow. So when we have a heavier or colder gas on top of a lighter gas, then the heavy gas will come down and light gas goes up. So we have a counter current flow. In terms of sub-nodal physics, a model is essentially - we are modeling a plume rising.

Let=s say we have a hollow LOCA and hydrogen coming out. And then a plume will generate going through the low compartment, steam compartment to upper compartments. So essentially, using these plume paths, we can push some of the lighter gas to the top portion. That=s one aspect of sub-nodal physics.

Second aspect of sub-nodal physics is like a couple kind of features. One is if a lighter gas is trying to come down. And let=s say it goes below this table. Then only a portion of it will be covered by lighter gas. It=s not mixing all of the way. Then a lump of the parameter, that cannot be modeled. So we use a sub-nodal physics model to check if the lighter gas only penetrates the certain length. And it doesn=t really contribute to the overall nature of circulation.

So depending on that condition, essentially we shut off the junctions to prevent these numerical mixing. So you have the sub-nodal physics pushing to one, shut off the junction to prevent these lighter gasses mixing to the lower part. And second part is let the lighter gas plume rise to the top.

MEMBER REMPE: Okay. Thank you.

MR. B. KIM: Once the hydrogen steam and air mixture conditions are calculated for all containment in use from the MAAP study, the flame acceleration, and different operation to deflagration detonation rendition, DDT, is analyzed by applying the sigma criterion and seven lambda criteria, respectfully.

Another possible hydrogen bonding mode inside the containment is a slow deflagration. The pressure prediction for this slow deflagration is analyzed on the conservative assumptions and bounding pressure predicted by the adiabatic isochoric complete combustion approach.

So here, we again applied a hydrogen source to the one hundred percent method of the reaction. Hydrogen control analysis results indicate that awareness of containment must be achieved, and leave less than ten percent hydrogen concentration. Also, there is no potential for DDT and pressure by

AICC assumption meets the FLC requirement of the containment integrity, as discussed later.

MEMBER POWERS: I find that no possibility of flame acceleration or DDT remarkable. It=s just because you never get high enough concentration of hydrogen to satisfy the seven lambda requirement?

MR. B. KIM: Yes.

MEMBER POWERS: Assuredly in your dome, seven percent hydrogen lambda is on the order of half a meter or something like that? Am I remembering that roughly correctly?

MR. PAIK: I don=t know the exact lambda value.

(Simultaneous speaking.)

MR. PAIK: What the TSMP did is they applied these OECD - the standard methodology for flame acceleration and also lambda defined the geometry, defined in -

MEMBER POWERS: So I suspect you just always fell below the initiating criteria? I suspect that=s what happened. Assuredly, the dimensions in the lambdas are okay, but they cut that off. And there=s no good reason to cut it off. We=ve just never done experiments that are in that regime.

MR. B. KIM: The second accident issue is molten core concrete interaction. The goal of MCCI mitigation is to secure the basemat liner integrity by minimizing the corium concrete attack and removing it from the core debris in the reactor cavity. So mitigation features of APR 1400 for MCCI is the reactor cavity and cavity floor concrete area, and cavity floating system CFS.

APR 1400 cavity is designed with cavity floor concrete and almost to the empty space. And to achieve the complete corium spreading on the floor, we dug out any obstacles for spreading. And protective concrete layer is installed in the basemat liner plate. Cavity floating system is designed to flood the cavity with water from the IRWST initiated by operation of the time of the severe accident entry.

MEMBER CORRADINI: Can you explain that part? I=m kind of curious where you=re going to send the operator when I have severe accident open valves. Tell me the procedure that has to be done by the operator to go open a valve at this point in a severe accident to flood the cavity.

MR. B. KIM: Yes. When the accident goes to the CFS, the operator tries to open the MOV in the cavity flooding system. Once the MOV is open, then IRWST - water in the IRWST starts to flow into the cavity by the -

MEMBER CORRADINI: I get that part. But the way - maybe I=m misunderstanding. I didn=t read this section, I=ll admit it. It says manual. So what=s the timing and where=s the manual operation occurring?

MEMBER POWERS: How do they know?

(Laughter.)

(Simultaneous speaking.)

MR. B. KIM: Core temperature exceeds one thousand two hundred degrees Fahrenheit.

MEMBER CORRADINI: So is this the same manual operation for the IVR?

MR. B. KIM: No.

MEMBER CORRADINI: I would think it has to be. If the water is going to the same place, isn=t it?

MR. B. KIM: Yes, but the -

MEMBER CORRADINI: Do you know what I=m asking? I=m trying to understand the logic.

MR. PAIK: If they open the MOV, what happens is the volume of the tank and cavity level will be similar. And that level is below the vessel. So you cannot have these expressive coolings. So if they will not do an IVR, then you have to - the valve is not open and you have to inject using the shutdown cooling pump - inject the water into the cavity. So the cavity to whole - whatever HVT -

MEMBER CORRADINI: Ah, so this valve is connecting the HVT to the cavity?

MR. PAIK: Yes. Right.

MEMBER CORRADINI: Okay.

(Simultaneous speaking.)

MEMBER CORRADINI: Okay. All right. So that was my mistake. So now I=ve got a valve between the HVT and the cavity, and how does the operator get to it?

MR. B. KIM: MOV.

MEMBER CORRADINI: MOV? So it=s a manual actuation? So I=ve got DC power? Okay, fine. All right, I misunderstood you. Thank you. And this floods into a level below the reactor vessel bottom area?

MR. B. KIM: Right.

MEMBER CORRADINI: Got it. And - I=m sorry. And so you=re not crediting an IVR, but if it were an IVR, you would pump it in?

MR. PAIK: Then they have to close the valve.

MEMBER CORRADINI: Right. Okay.

MR. B. KIM: By using the CFS, we applied to free-floating strategic power to the vessel. So analysis of the MCCI by using MAAP studies performed with conservative approaches with a supporting MCCI and CORQUENCH. So MAAP 4.08 incorporates two MCCI motors. One is the jet breakup, and the other one is a heat removal to override water pool in order to mimic the water integration effect. So to decide the usually dependent key parameters of MAAP motor - for these two kinds of motors.

The supplemental study using CORQUENCH is done by Dr. Mitch Farmer at Argonne National Lab. So Dr. Farmer evaluates the very conservative operation. That=s under the assumption of full core relocations at once. And without jet breakup in case of large LOCA sequence. So the conservative CORQUENCH studies project ablation depths of .27 meter. So MAAP parameters related to the these two MCCI motors is then decided to get a comparable ablation depth with CORQUENCH study for the conservative Large LOCA case.

The selective MAAP model parameters used for all of the last test on sequence variation. This is the conservative variation. So the MAAP study result credits that the highest tabulation depths of .24 meters from the Large LOCA case, which is obviously less than the thickness of the LCS concrete, which is nineteen centimeters. So therefore, we have compounded the integrity of basement liner plate against an MCCI event.

MEMBER KIRCHNER: May I ask, what=s the pressure rise in the containment in this scenario? How high does the pressure go?

MR. B. KIM: Yes. Of course, the pressure goes up due to the continuing steam evaporation from the cavity. It=s of importance, so we look at the pressure behavior due to the MCCI for long-term. And we found that it=s not - it does not go over to the design barrier.

MEMBER KIRCHNER: Is it close?

MR. B. KIM: No. It=s lower than the containment performance requirement.

MEMBER KIRCHNER: I understand that it might be lower. But how close does it get to the yield plate? Or your containment pressure?

(Simultaneous speaking.)

CHAIRMAN BALLINGER: The design pressure is about sixty psi?

MEMBER KIRCHNER: That=s normal.

CHAIRMAN BALLINGER: Normal? Yeah. But how close does it get?

MR. B. KIM: The criteria, the pressure limit of the containment is not sixty psi. It=s much higher than the design pressure. It=s more than one hundred twenty psi. I don=t remember the exact barrier of the pressurization on the inside, but -

MEMBER KIRCHNER: Is it sixty or a hundred psi?

MR. PAIK: One hundred ten psi.

MEMBER KIRCHNER: Okay. So now it raises the question of how well mixed this core collapse is with the water. Because that could change the rate at which the steam is generated and the pressure pulse that you get. Do you see what I=m saying? That you can get a pressure, a fairly significant pressure pulse from a collapse of the corium into the water pool.

MR. B. KIM: Okay. I think your concern is more related to the steam explosion, not the -

MEMBER KIRCHNER: More to the pressure. The pressure that is generated inside the containment.

MEMBER CORRADINI: I think Dr. Kirchner is asking, you=re getting a quasi-steady pressurization with time. Do you ever get to the failure pressure of one hundred and ten? Or does it just happen or occur days later?

MR. PAIK: Yes, this one hundred and ten psi occurs in twenty-four hours.

(Simultaneous speaking.)

MR. PAIK: At the time of pressure failure, we will have a pressure spike. But that spike is typically low. Initially it goes up and then it comes down, and then we will have a gradual increase.

MEMBER KIRCHNER: Increase, okay. Is that pressure - is the pressure time history, is that a function of your assumptions for how well-mixed the corium is as it comes through the vessel and collapses?

MR. PAIK: Yes.

MEMBER KIRCHNER: So do you look at a span of bounding calculations without getting too wrapped up in exactly how it jets down. Do you see where I=m going with this? Do you just mix that corium with the water pool to see the amount of surface area that is available for quick pressurization?

MR. PAIK: The current MAAP 4 model, as heat transfer is a factor as it is coming down and generating, or transferring heat to the water and generating steam. Once the corium contacts the cavity, then that particulate generated is mixed with the remaining corium. So that particulate was not tracked.

MEMBER KIRCHNER: No, I was just curious. A simple experiment. If you ever take a thermos and put hot water in - just very near boiling water - shake up the thermos and try to lift the top off, and you=ll see the kind of pressure spike you get from that. And I=m just curious what kind of pressure spike you might get, depending on how the vessel ruptures and the corium is just dumped into the pool.

MR. PAIK: I think that depends on the vessel pressure, and how deep the water pool is. And then we typically use a kind of a jet entry using an aid that can get to our fuel bar or pressure spike.

MEMBER KIRCHNER: And what=s the steady

state when this happens? Is it pretty low?

MR. PAIK: The - I mean, as I mentioned earlier, the pressure is gradually increasing.

MEMBER KIRCHNER: It=s gradually going up. Okay, all right. Thank you.

MR. B. KIM: Okay. The third severe accident issue is high pressure melt ejection and direct containment heating. The goal is to prevent early containment failure and minimizing entrained debris to upper containment. So the mitigating feature for HPME, DCH, and for the APR 1400 is rapid depressurization system and reactor cavity with convoluted flow path. So RCS pressure reduction system performance is analyzed by using MAAP 4.08 code and DCH event is analyzed by following the NUREG/CR-6338 methodology.

Analysis results indicate that the present pressure reduction system can make the RCS pressure less than the DCH pressure of 250 psi at the time of vessel breaching. And conditional containment failure probability in APR 1400 containment by DCH event is calculated less than .01 percent.

The fourth issue is fuel coolant

interaction. For a steam explosion in both in-vessel and ex-vessel. The first step of analysis and methodology is to set up the initial boundary conditions. For the base case and sensitivity cases. Second step is to evaluate the energetic loads by using the TEXAS-V code. Then finite element modal code is employed to investigate the vessel load head and the cavity wall response against those steam explosion energy.

The analysis will indicate that the integrity of the lower half and the cavity were preserved. In addition, the effect of the in-vessel retention and external reactor vessel cooling, IVR-ERVC, stretches on the ex-vessel steam explosion. It will be assessed by COL time by applicant as described in the item 19.2.

The fifth issue is the equipment survivability, or ES. The proposal of ES assessment is to conform the equipment and instrumentation can operate under the severe accident environment over the required time window. So the first step of ES assessment is to identify and to restore the equipment and instrumentation used in the severe accident mitigation. And second step is evaluation of environmental conditions of those equipments by using MAAP 4.08 code for selected sequences.

Then, we can decide what is the bounding temperature, pressure, and radiation for each increment from the MAAP calculation. Lastly, the instrument survivability is assessed by comparing them with suppliers= test data or experimental test data by thermal lag analysis.

MEMBER REMPE: So I had a couple questions on this. First of all, is there a cutoff frequency? You have here that you used MAAP3 code for selected sequences. How did you select the sequences? Did you have a cutoff frequency where you said, okay - I mean, you could really have a bad sequence if you don=t have a cutoff frequency. So, how did you select those sequences. That=s one of my questions.

And then the other question is that you have hot junction thermal couples and core exit thermal couples that are type Ks. And in the accident analysis report, it claims that you have data - KHNP has data that says that type Ks can withstand up to 1533k for long durations, which amazed me. Because I just don=t know of any type Ks that can do that. What kind of sheath does it have? Because usually, there=s chromium or something in the sheath that will migrate through and attach to the thermal elements. And they=ll cause those thermal couples to degrade. So I was real curious on what thermal couples you had that could withstand that high of temperatures.

MR. B. KIM: Okay, for the first question. We selected accidents by following the required 1.216. Simply, I can say we chose accident sequences which correspond to more than ninety percent of coding frequency. And in addition, we also applied them to the accident sequences, initiated by the dominant sequences.

MEMBER REMPE: Okay. Okay.

MR. B. KIM: And for the second question, some thermal couple K-types can survive that temperature, elevated temperatures for a long time. I need to check from the physical description for that information.

MEMBER REMPE: Yes, please. It actually stated that in your report. And so I was real puzzled about that one, because -

MEMBER POWERS: Well I would think that it would depend on what you mean for a long time. The degradation of the seabed coefficient is like effusion. So it=s relatively slow. The question is - I mean, you=re not looking for a particularly accurate measurement here. So the degradation is slow, but I mean -

MEMBER REMPE: It just goes downhill when I=ve run thermal couples in furnaces.

MEMBER POWERS: I had -

MEMBE REMPE: - Eleven hundred. And we=ve had a large number of different types.

MEMBER POWERS: I=ve had them do everything known to man. Up, down, sideways, open, and what not. But if all you=re looking for is an indication, the temperatures are very hot and you don=t care about how hot, then yes. They can do it for a while. If you=re interested in three days, you=re in trouble. If you=re interested in three minutes or three hours, probably it=s okay.

MEMBER REMPE: It said long duration. So I=d be curious about the definition.

MEMBER POWERS: Yes, what=s the definition of long?

(Laughter.)

MEMBER REMPE: What data they have to support that claim?

MEMBER POWER: Yeah, I mean - I=ve run type Ks in obnoxious environments and used every bit of the 1327 you=ve got.

MR. B. KIM: The containment conditions will never reach that high.

MEMBER REMPE: Well, this is the core exit temperature in a hot junction thermal couples that are used for water level measurement in the vessel.

MR. B. KIM: Basically, the thermal couple is only - we expected can survive the continual onset of the core damage. So beyond this time span, if the thermal couple survives, it is okay. But we don=t take credit for a long time.

MEMBER REMPE: Okay. It was on page five hundred out of six fifty-nine of your accident analysis report. And so I=d be curious on the data that supports that claim, okay?

MR. B. KIM: Okay. I will check for you. So because the major assessment is related to the suppliers= or the vendors= data, so the sections for ES just gives those instrumental lists. And how we do the ES assessment based on the reference plant experience. And what expected bounding environmental conditions are. So the rest part of the ES assessment will be done later as addressed in the first core item of 19.2.

The containment performance - containment withstands to the severe accident challenges is performed to confirm to containment performance. Criteria for the containment is given in SECY 93-087 and Reg Guide 1.216. As per Reg Guide 1.216, applicant needs to show the containment integrity against the pressure load from the hydrogen combustion and more likely, severe accident challenges. Positions two and three, respectfully.

Containment integrity regarding the position two - the hydrogen burning load are varied conservatively under the key assumptions to hydrogen mass from one hundred percent MWR, and no credit of recombiners and igniters, and AICC bonding methodology. In addition, the various initial impression is considered to get the maximum AICC pressure under the given condition.

So, determined AICC pressure is a bounding value of hydrogen, deflagration is a 123.7 psi is then applied be an input value for containment response. So finite element modal, FEM, studies indicate that the maximum strength of the liner plate to not reach the allowable limit. Therefore, conservative hydrogen combustion load meets the factored load category, FLC, requirement.

Regarding the position three, the pressure from more likely severe accident challenges is evaluated. So according to the commission=s recommendation addressed in the Reg Guide, accidents are selected to be ninety percent of cumulative CDF from PRA Level 1 study. In addition, the five representative initiators such as Large and Small LOCA, station blackout, SGTR, and total loss of feedwater are taken into account in the light of deterministic approaches with conservative accident progress.

Then, a MAAP 4.08 study was done to determine a bounding pressure profile and peak pressure with realistic [ESA] appropriation, such as the success of cavity flooding system and RCS pressure reduction and emergency containment spray backup system. Determined bounding profiles then again applied to the three dimensional finite element modal study of containment structure.

MEMBER CORRADINI: Can I just make sure of something? So, you don=t have to pull it up. But on slide thirty-six for the Level 2 PRA Section 19.1, this connects up to these dots. Which is, you delay containment failure with the ECSPS and with cavity flooding, you delay any sort of over pressurization so late in time. So you have no early failures in the first day or two?

MR. B. KIM: Yes. We dealt with containment.

MEMBER CORRADINI: Okay. And so the dominant contributors to the fourteen percent containment failure probability is containment bypass and delaying all failures until late times. So, all the other stuff we were talking about and quizzing you about are much smaller fractions of containment failure. In other words, of the - I don=t know how to ask this properly, but - of the fourteen percent, I=m worrying about DCH and explosions, et cetera, et cetera, et cetera.

It=s a small fraction of everything we=re talking about. It=s really the operation of the ECSBS cavity flooding delaying pressurization that dominates that late fourteen percent containment failure. Do I have that correctly? Do you understand my question?

MR. T. HWANG: My name is Taehee Hwang from KEPCO & ENEC. He is right, the containment failure is probably based on internal events. This contributes to fourteen percent of CDF. Opposed to containment -- containment bypass, SGTR including severe accident induced SGTR. And then second containment failure modules or rate containment failure due to low pressurization and/or the hydrogen -

MEMBER CORRADINI: Yeah. So let me ask my question this way. If tomorrow magically explosions, direct containment heating, the other things we were asking about all went to zero probability - would fourteen percent go to thirteen percent?

[Simultaneous speaking.]

MEMBER CORRADINI: Do you understand my question? There=s a fourteen percent containment failure probability. If all these physical processes were wrong and zero probability, does fourteen become thirteen? I think the delta change is very small. I=m trying to get an idea of what the delta change is. Am I making sense? No? Member Stetkar says no. I=m just trying to understand the contribution of all -

MEMBER STETKAR: I know what you=re trying to ask. But the way you=re asking is not making sense. I don=t know how to ask it, is the problem.

MR. PAIK: What=s the containment failure frequency when ESPSBS is not available?

MR. LEARY: Excuse me. If I may, Jeff Leary with Enercon. I believe the question that you=re trying to express is the significance of or contribution to containment failure from DCH and steam explosions?

MEMBER CORRADINI: Yes. Let=s take those two as an example.

MR. LEARY: Yeah, I don=t have the exact percentages. But it=s not a significant contribution.

MEMBER CORRADINI: Okay. It=s a small fraction of the fourteen percent, is what I was trying to estimate.

MR. LEARY: That=s correct.

MEMBER CORRADINI: So I=m curious about the timing of the failures. Does the timing of the failures, as long as it=s beyond - I=m not sure exactly what you call long - but a day or two, does it matter after that in terms of large release frequency? Or it also is - because the other thing I was going to ask you is going back to slide thirty-six, I was trying to understand the difference between the fourteen percent for conditional containment failure probability and nine percent for large release frequency. So there=s five percent going away, and I=m trying to figure out physically why there=s a difference there. And it must be timing of the failure.

MR. LEARY: I=m sorry. The five percent or so is a small containment failure.

MEMBER CORRADINI: So it=s leaking, but it=s just not leaking fast enough?

MR. LEARY: Including things like basalt and things that are not large. So the nine percent that you saw on there is large releases, and the fourteen percent is all. So large and small combined together.

MEMBER CORRADINI: But the difference is really leakage at lower rates?

MR. LEARY: By rates, you mean?

MEMBER CORRADINI: Percent per day.

Through out of the containment or its leakage paths. I think that=s what you=re saying.

MR. LEARY: It=s a total integrated at the end is what determines large versus small, but yes.

MEMBER CORRADINI: Okay. Fine. Thank you.

MR. LEARY: Thank you.

MEMBER CORRADINI: I=m just trying to figure out how much I have to worry about all the things we ask you.

MR. B. KIM: The last part of 19.2 is severe accident management framework. Accident management encompasses those actions taken into during the course of accident by the plant operating and technical staff. The first step is to prevent core damage. The second step is to terminate the progress of core damage. The third step is to maintain containment integrity as long as possible. And the last step is to minimize the offsite release.

At each step, operator will try to act what he can do according to the accident management guidelines. For example, operator tries to recover the RCS inventory by safety injection. And if safety injection has failed, he will try to open POSRVs to depressurize RCS and to allow injection using shutdown cooling or containment spray pump. ERVC strategy can be achieved by flooding using shutdown cooling pump to remove the heat on the outer surface of the vessel.

MEMBER REMPE: I have a question about this slide. Because this is what I was thinking about with the vessel retention. The staff doesn=t review the severe accident management guidelines, right? So you=re going to include them in the guidance to the operators. And how does this work? Because it=s like something that=s out there, we don=t take credit for it. And that was my question earlier about - you know. Do you reinforce it so you know it works?

I know the AP 1000 and at least the AP 600 had to change the design to make sure that the insulation was robust. And you=re chugging all that water through and generating the vapor for the external reactor vessel cooling. So you=ve got this mitigating strategy in your design that=s being credit for, so I guess the staff doesn=t review it very much. How does something like that work? And how are you going to come up with the guidance with late term water addition, and things like that? When we=ve been talking about flecks - is this something that - you know. If this plant were built in the U.S., would it be part of the PW Owner=s group? And this would be something that=s considered by the PW Owner=s Group?

MR. PAIK: I think in the APR 1400, they are looking at the ERBC plus the water injection. Because AP 1000 and ERBC is just a passing. But the APR 1400 and ERBC is you need a pump to inject the water. And if you have a pump that=s available, the question is why not inject into the vessel? Why only inject on the outside? So what they need, what they want to do is it takes about thirty to forty minutes to fill up the cavity.

In the meantime, ICS will be depressurized. So as soon as they finish the cavity flooding, then they can switch to the injecting into the vessel. So I don=t know the details, but I think that that=s the kind of approach they will go use. So not only the express cooling, but also trying to inject after the RCS is depressurized.

MEMBER REMPE: Well it=s just something

I=m not quite sure how the staff will deal with that. Is this something - well, they don=t. What if the guidance is not optimum? And who reviews that guidance?

MEMBER CORRADINI: I don=t want to speak for the staff, but my understanding is as long as it=s there, that=s it.

MEMBER REMPE: I hope that it=s done right.

It sounds like it=s unreviewed. I mean, this is a different thing that=s not standard in the existing fleet and although it is - I guess in the AP 1000 - they are supposed to have it. And I just am kind of curious about this long-term phenomena. You want to inject in the vessel. It seems like a reasonable thing to do. If the vessel has - if it=s at a high pressure. I hope they would have the right criteria for doing it. I guess I=ll leave it at that.

And then I didn=t hear anything about, did you consider how much you need to reinforce that insulation? Like the folks did for the AP 1000? How much scrutiny did the design have? I mean, the staff=s not reviewing it. So I=m asking. Did you consider, did you have sufficient opening for water to get in and for the steam to go out of this vessel retention capability?

MR. RYU: But for reference, we make some change to enhance the flow rate in the cooling channel. Each situation the design is varied.

MEMBER REMPE: Is it in the Shin Kori design? It is? Okay.

MR. RYU: This is in Shin Kori.

MEMBER REMPE: Okay.

MR. B. KIM: Regarding the accident management framework, detailed development implementation and maintenance of accident management plan is clarified in 19.2. It is time to close my talk with some summaries from 19.2. The severe accident prevention and mitigation features of APR 1400 are designed to confirm to associated criteria and requirements. Hydrogen risk, MCCI, DCH, and ES are investigated, and we found the requirements are satisfied. Containment integrity or containment performance is confirmed according to the Reg Guide 1.216. And accident management plan will be developed and established as described in COL 19.2. This is the end of my presentation. Thank you for your kind attention.

MEMBER STETKAR: Before KHNP vacates the premises, I just wanted to make a couple of comments. I know I was pretty critical of a few items this morning. I just wanted to, for the record, say that my review of this particular design certification PRA - in my opinion - it is much, much better than the vast majority of design certification PRAs than we have seen over the nine and a half years I=ve been on the committee. And I=ve looked at five or more design certifications.

In particular, not so much the Level 1 internal events because everybody sort of does the same thing there. But I think your treatment of fires is much more coherent. Your treatment of internal flooding is much more coherent. Your low power and shutdown models, you actually have low power and shutdown models. And the fidelity of the way that you did and presented the PRA base size margins. So I hope this - from the staff=s perspective is a good example going forward to any other designs that we might see. Because it=s a heck of a lot better than what I=ve seen, anyway.

CHAIRMAN BALLINGER: That=s a big compliment.

(Laughter.)

CHAIRMAN BALLINGER: He doesn=t say that to all of us. In fact, that=s the first time I=ve ever heard him say it.

MEMBER CORRADINI: Are we sure it=s him?

(Laughter.)

MEMBER POWERS: And once again, he=s probably wrong again.

(Laughter.)

CHAIRMAN BALLINGER: Well thank you very much. What we will do now is we will recess for fifteen minutes. Well, until quarter of. And then we will pick up with the staff.

(Whereupon, the above-entitled matter went off the record at 2:28 p.m. and resumed at 2:47 p.m.)

MR. STECKEL: My name is Jim Steckel, and I=ve had the privilege of working with this PRA team for the last couple of years.

A couple of things -- Lynn Mrowca is over on this side. She is the branch chief who will, sadly, soon be leaving the agency. I wanted to make sure everyone knew that.

MEMBER POWERS: Is it because she doesn=t like us?

MEMBER BLEY: That=s pretty much it. I heard it was one person.

(Laughter.)

MEMBER POWERS: Okay. Well, I=ve done my share of harassing on her, I=ll have to admit.

MR. STECKEL: One and done.

The technical staff will be up here to make the presentations for their particular sections that they have reviewed. These are the names and their areas where they work and their titles, and of course Jeff Ciocco is lead. I=m Jim Steckel, the chapter PM. And these will be the actual areas of review that each of the members has taken on.

And just for your clarification, this is what we call Chapter 19, consists of 19.1 and 19.2. And we have considered the rest of the 19s to be separate chapters, at least during the review process and the SER preparation.

So you may hear me or someone refer to Chapter 19.3 when we get to the phase 6 where everything -- there are no more open items, et cetera. I believe everything will be consolidated under one Chapter 19 with those different sections, 19.1

through 19.5. But the rest of it will be tomorrow.

We are going to get all of these reviewers to come up here and participate, which means there will be some seat swapping going on. We will try to keep that to a minimum -- a minimum disruption for the proceedings.

And I=m going to ask the staff to please leave your name tags here. Chris will either collect them or they need to be here for us tomorrow, and I believe he wants to retain those for the future.

So with that, we are ready to proceed for our first reviewer, which is Mr. Hanh Phan. He is the senior reviewer for PRA.

MR. PHAN: Good afternoon. My name is Hanh Phan. I am the lead reviewer for APR1400 design certification.

MEMBER BLEY: Could you speak a little more directly into the mike? It=s hard to hear over here.

MR. PHAN: Okay. Better now? Thank you.

First, on behalf of the staff participating in the review of the APR1400 PRA severe accident evaluation, we=d like to thank all of you for the opportunity to share with you our phase 2 SECY reviews and any comments you have on our assessment.

We=d also like to take this opportunity to recognize the guidance from our branch chief, Mrs. Lynn Mrowca, and the lead project manager, Mr. Jeff Ciocco, since both of them are going to retire late this month or early next month.

Next slide, please.

This slide outlines the staff today presentations for Chapter 19.1. In this presentation, the staff will cover internal events, Level 1 and 2; internal fires, Level 1 and 2; internal flooding, Level 1 and 2; PRA-based seismic margin assessment, and other external events.

This PRA covers both at-power and during low power and shutdown operations. The staff will also present you the reviews of the PRA quality and the use of this PRA during the design certification stage. For each technical area, the staff will briefly present you the depth of our review and the key technical issue. It should be noted that for this presentation it only covers the information provided in the DCD Revision 0, and any support information for this revision.

The staff is totally aware and on top of the PRA update being conducted by the applicant. However, because the DCD revision was -- is not yet available to us; therefore, it is immature at this point to make any conclusion on the PRA adjustment.

Next slide, please.

The first topic for today=s PRA presentation is on the quality of APR1400 PRA. The applicant addressed the qualities of the PRA by conducting peer reviews and provide stratifications in Section -- DCD Sections 19.1.2, and specifically in Table 19.1.1, that the PRA is sufficient to support this application.

Following the guidance provided in the SRP, the staff ensured that the levels of details, the scope, and the PRA qualities, including PRA update and upgrade are reasonable and acceptable. The staff estimates the peer review=s report to an audit to ensure that any deficiencies identified from these peer reviews would not have any significant impacts on the PRA. Due to the issues identified in the staff evaluation report with open items, at this point we are unable to make any conclusion that the PRA quality of APR1400 is acceptable.

Next slide, please.

Under the quality, two topics we=d like to present to you. The first one is the PRA conversions from SAREX to CAFTA. During phase 1=s review, the applicant notified the staff that they had intended to convert their models from SAREX to CAFTA.

The duration for this conversion taking place from June 2015 through July last year. During the conversion, the applicant incorporates some of the peer review findings, some of the staff findings, and their self-identified issues.

During the public meetings last year, they sent to us their preliminary results from the CAFTA models. The staff noted that those two models not identical; however, the difference is from the CDF, the LRF, CCFP, and the PRA insights are not much different, and they are just minor difference.

In addressing the staff concerns, the applicant agreed to complete the following tasks during the staff phase 2 review, including perform self-assessment on the CAFTA model, notify the staff of the changes and the results, update PRA notebook, revisit all of the sensitivity studies using the SAREX, and revise the DCD.

Next, please.

The next topic is on the peer review. I mentioned previously the applicant justified their quality -- the quality of their PRA. They conducted a peer review prior to the submittal.

This was performed during the week of June 24, 2013, against the ASME/ANS PRA standard. The peer reviews were conducted by a team of six PRA experts with over 170 years of diverse PRA experience.

The scope of this review included at-power internal events, Level 1; at-power internal flooding, Level 1; and loss release frequency modelings.

This peer review resulted in 90 facts and observations. Within those F&Os, 59 are findings, 27 are suggestions, and four are best practices.

The peer report concluded that, and I quote it directly from that report, that the PRA1400 PRA substantially meets both the ASME PRA standard and the draft ALWR standard at capability 2 or better for 88 percent of the applicable supporting requirements, with 90 percent met at capability 1 or better.

Next slide, please.

MEMBER KIRCHNER: Could you just for the non-PRA practitioners, I find it interesting that it=s 88 percent of the applicable supporting requirements. Is that just -- you just tabulate all of the requirements in the standard and then match the PRA against it?

I mean, what does it mean to be 88 percent or 90 percent or better? Is that like an A, or is it a good vintage, or is it -- or a lot of work left to be done? It doesn=t sound like it. So I=m just curious. Is this the way these peer reviews are typically done?

I=m having a little fun, Mr. Chairman.

MR. PHAN: In the --

MEMBER KIRCHNER: It=s interesting that the peer reviewers used percentages to grade it. I=m just --

MR. PHAN: In the PRA standard, there are 327 supporting requirements. Based on their initial evaluation, they believe that 49 of them is not applicable, like sub-requirement for dual unit. For those that may not applicable to them, at this point they take that out from the 327 SR.

And for the rest of that 278 remaining SR, they say that 245 of them, or 88 percent I mentioned here are the capability 2 or higher, which means that those supporting requirements sufficient to support risk-informed application. But that is not the case here. This is the design certification.

So according to the SRP, capability 1s would be sufficient for this application. So with that, they say 90 percent of 278 supporting requirements met the capability 1.

MEMBER KIRCHNER: Now I will be a little more serious. You know, in the construction management business, one is always concerned about percent complete because you could count milestones, but not all milestones are created equal. So in lay terms, plain English terms, what=s your assessment of the -- what does that mean when it=s 88 percent or 90 percent?

This is quite good. It would --

MR. PHAN: Yes.

MEMBER KIRCHNER: -- seem to indicate that it=s a mature -- a relatively mature PRA at this point, given where they are in the design.

MR. PHAN: The 88 percent would not tell anything.

MEMBER KIRCHNER: Right.

MR. PHAN: The staff taken that to ensure that for those that not met, the applicant, they have to justify why not met, and what the impact is on the application.

MEMBER KIRCHNER: Okay.

MR. PHAN: Secondly, the staff focused on the findings not met -- are not met at this point, because the finding would tell us any -- there are, you know, issues with the PRA models, and that=s where the staff paid the attention on.

MEMBER KIRCHNER: Thank you.

MR. PHAN: And for your information, we asked the applicant to resolve all of the findings and give us their resolutions by the end of phase 4 review.

MEMBER KIRCHNER: Thank you.

MR. PHAN: Yes.

MEMBER BLEY: Just for members of the Committee who haven=t been here the last many years, in a series of letters over at least five years, maybe going on 10, the Committee has been urging the staff to actually ask that the design cert PRA be a category 2, except in cases where it is not possible because of the state of construction. They haven=t agreed with us.

MR. PHAN: Okay. Through the regulatory audit, the staff had the opportunity to estimate the peer reviewers' report, and open insights with the PRA has been assigned to the capability of the PRA standard.

For the staff, this peer review was used to identify the strength and the weakness of the PRA and get confidence in the PRA models and results. Based on our initial review and the audit, the staff finds that the applicant had not completely dispositioned all the 59 findings.

Some of those findings the applicant assessed by perform such DCD study. And, furthermore, we found inconsistencies of the information provided in the report and the information provided in the DCD.

In response to the staff concerns, the applicant agreed to disposition all 59 findings and update the DCD to incorporate the findings into their model and the DCD during phase 4 review.

And that is the end of the PRA quality. The next topic is on the internal events at-power Level 1 PRA. And I will introduce Mr. Ayegbusi and Ms. St. Peters.

MR. AYEGBUSI: All right. Next slide.

All right. Good afternoon. My name is Ayo Ayegbusi, and I was responsible for reviewing the initiating events, success criteria, and accident sequence analysis, and the quantification section of the DCD.

All right. So my review was performed in accordance with the SRP-19.0 acceptance criteria while using the PRA standard as well as a guide. As Hanh mentioned, he had already talked about the peer review. So because the applicant had a peer review performed, I was able to perform a less detailed review and focus on issues of the design and the consistency between the DCD and the PRA notebooks.

I was also able to audit the PRA notebooks for my sections in detail, partly because the DCD was light on information. In addition, my review included ensuring the applicant appropriately used the data from NRC, NUREGs, and other sources that they referenced, as they were applicable to the applicant=s design.

Subsequent to that, I was -- I used RAIs for areas that were not adequately described in the DCD as expected by the SRP acceptance criteria or areas that, based on our experience, we felt needed to be covered in the DCD.

Overall, the applicant=s responses to my RAIs were acceptable, and the RAIs are now confirmatory items.

Next slide, please.

So now I would like to discuss some items of interest that came up during my review, and they are mainly under the initiating events analysis and the success criteria analysis section of the DCD.

What I identified was that some initiating events, such as very small LOCA, were not screened during the analysis, even though that was -- it=s one of the initiating events that is identified in NUREG-6928, and in the applicant=s application they mentioned that they had used that in developing and performing the analysis.

We also did not identify any new or unique events. The applicant didn=t. We looked at it from our point of view, and we also didn=t identify any.

All right. To move on to success criteria, so for this --

MEMBER STETKAR: Before you get to success criteria, I=ll ask you the same question I asked the applicant. It=s really curious to me why I don=t see many support system initiating events, and why the support system initiating events that I can see are in lockstep with that NUREG.

Did you review the initiating event notebook and the initiating event grouping? And let me ask you a couple of questions. Are you okay with the fact that failure of the main transformer is a general transient initiating event?

MR. AYEGBUSI: Which question would you like me to answer first?

MEMBER STETKAR: Both of -- well, the first one you can answer first.

MR. AYEGBUSI: Okay. So I did review both notebooks. Well, I did review the initiating event analysis notebook.

MEMBER STETKAR: Are you okay with the fact, then, that failure of the main transformer is a general transient initiating event?

MR. AYEGBUSI: I did not identify any

issues with main transformer failure being a general transient initiating event.

MEMBER STETKAR: Thank you.

MR. AYEGBUSI: Okay.

MEMBER STETKAR: I completely disagree with your finding. Are you okay with the fact that spurious opening of a main steam safety valve is a general transient initiating event?

MR. AYEGBUSI: I have to go back and look at that.

MEMBER STETKAR: Okay.

MR. AYEGBUSI: I don=t recall off the top of my head.

MEMBER STETKAR: Did you look at the fact that there are many failures, valve failures, that can give you spurious isolation of all component cooling water to all four reactor coolant pumps?

And that if you use the valve failure rates in NUREG/CR-6928, the frequency of that event would be comparable to the total frequency of partial loss of component cooling water. That to me sounds like it=s a design-specific initiating, support system initiating event. Did you look at that?

MR. AYEGBUSI: So for the -- in looking

at the design-specific initiating events, I was trying to identify if any were there. As far as support system initiating events, what I looked at was the fault trees that developed for each support system, and what frequencies they developed there and compared that to what the NUREG initiating event frequencies were and to see what they eventually used.

MEMBER STETKAR: Do you have any idea what the configuration of the component cooling water system in the NUREG is that gave them those frequencies in that NUREG?

MR. AYEGBUSI: I don=t have that off the top of my head.

MEMBER STETKAR: No, you don=t, because it=s an amalgam of chunk.

MR. AYEGBUSI: Okay.

MEMBER STETKAR: Okay? The particular thing that I was trying to mention here is that it=s nothing to do with pumps failing. It is valves closing spuriously that isolate all component cooling water to all four reactor coolant pumps. And this is -- this does not depend on whether the pumps are split between division 1 or division 2. It=s strictly the plumbing in the plant.

The frequency that I can calculate using data from the same NUREG that you referred to gets me a larger fraction of the total frequency that they use that ostensibly accounts for pumps failing. But I don=t know whether it=s pumps failing because, you know, we don=t know what=s in that NUREG frequency; do we?

Greg, do you want to add something?

MR. ROZGA: Greg Rozga from Enercon. Is this on?

MEMBER STETKAR: Yes, it is.

MR. ROZGA: Okay.

MEMBER STETKAR: Well, speak into it.

MR. ROZGA: When you did your calculation, did you also need to fail seal injection cooling?

MEMBER STETKAR: No, no. That=s -- I=m sorry. If I fail component cooling water to all four reactor coolant pumps, I will have an initiating event. I will have no reactor coolant pumps running because the operators will shut them down.

MR. ROZGA: Correct. I=m sorry. I thought --

MEMBER STETKAR: They don=t fail by

themselves.

MR. ROZGA: -- you were referring to a seal LOCA.

MEMBER STETKAR: No. I didn=t say seal LOCA; I said initiating event.

MR. ROZGA: My error. Sorry.

MEMBER STETKAR: Okay. Thanks.

Last question I have for initiating events is -- and I=m trying to keep these as concise as I can. The LOCA initiating event frequencies in this magic NUREG that we have to use the numbers from have a small LOCA size of .5 inches to two inches, a medium LOCA size of two inches to six inches, and a large LOCA size of greater than six inches.

And that is from the January 2012 version of NUREG/CR-6928, just to give you a NUREG. It is actually listed as Update 2010, but if you look at the footer of the page it=s January 2012.

How do we know that those are the appropriate LOCA size ranges for this nuclear power plant?

MR. AYEGBUSI: So the -- if I remember correctly, that was the initial -- looking at the -- looking through the notebooks, right, there was --

and that was something I was going to get to under success criteria -- there was some inconsistency between the DCD and the notebook, right?

And so I don=t recall if it was a question that was asked via the RAI process or a question that was asked during our audit, but that was a question that was asked. And what we got back from the applicant through their analysis was that those break sizes were applicable to this design.

And so what I did then was look at the results of what was provided. I believe some of the results were provided in the -- in this -- I want to say the success criteria notebook, in one of the PRA notebooks, and that was -- I found that acceptable, so --

MEMBER STETKAR: Did you find -- I mean, did they do a systematic -- typically, the break sizes are based on physics and thermal hydraulics. You know, what is the largest break that cannot directly remove decay heat determines the upper end of the small LOCA size range. That depends, in my notion, on how big a power plant you have.

MR. AYEGBUSI: It=s not just a two-inch piece of pipe. The size break between medium and large LOCA depends on the ability of low-pressure injection alone to prevent core uncovery, not to be careful about core uncovery, to allow reflood. That also depends on the plant design.

MEMBER STETKAR: So did the success criteria notebook do that? If I have a 300 megawatt plant compared to a 1,500 megawatt plant, my break sizes are going to be much different.

MR. AYEGBUSI: I agree.

MEMBER STETKAR: Okay. So that=s, again, why -- because the magic generic NUREG/CR-6928 gives me frequencies for a half-inch to a two-inch break, why is that an appropriate small LOCA frequency for this particular plant?

MR. AYEGBUSI: So when we looked at that probably over a year ago, the analysis that I recall bounded the break sizes in NUREG-6928. So I don=t -- so, for example, such as for large LOCA, I don=t recall specific -- the specific details of what was done, but I do recall that that analysis was done to determine the threshold between small and medium and medium and large.

MEMBER STETKAR: Okay. Thank you.

MR. AYEGBUSI: Okay. So moving on to

success criteria, in one instance during our review we identified the Chapter 19 success criteria for SI pumps in response to a large LOCA was more conservative than the Chapter 15 success criteria. We also found that there were similar inconsistencies between the DCD and the PRA notebooks during our audit.

In discussions with the applicant, these items were resolved, and they have been closed to confirmatory actions -- confirmatory action items.

On the final bullet that I have, the PRA model software conversion, so the applicant is -- they are converting the model, the PRA model, and we are expecting them to incorporate some of the issues that they identified during -- that were identified during the PRA review, some of the issues that we have raised and resolved by the RAI and audit process.

And once that=s done, and I guess later revisions to the DCD is sent in, I will have to review those changes and then determine if there is any impact on the PRA results and insights.

MEMBER MARCH-LEUBA: Can you give us an example of those success criteria that were more conservative?

MR. AYEGBUSI: All right. So I=ll just --

MEMBER MARCH-LEUBA: Just one example.

MR. AYEGBUSI: Okay. So, for example, as you=re aware, Chapter 19 is the accident analysis, right? And in Chapter -- sorry, did I say Chapter 19? Chapter 15.

MEMBER MARCH-LEUBA: You said 19; you meant 15.

MR. AYEGBUSI: 15, yes. So, in Chapter 19, the success criteria for large LOCA for the safety injection pump was three out of four pumps, right? In Chapter 15, it was two out of four pumps, right? And, you know, typically, with Chapter 19 and the PRA, we are expected to be more realistic, right? So --

MEMBER MARCH-LEUBA: And do you expect that that was because they did a preliminary Chapter 19 before they actually had a design, and then didn=t come back to fix it?

MR. AYEGBUSI: I have no idea.

MEMBER MARCH-LEUBA: But they=re fixing it now?

MR. AYEGBUSI: That is correct.

MEMBER MARCH-LEUBA: Okay. Thank you.

MR. AYEGBUSI: That=s all I have, unless there are any questions.

MS. ST. PETERS: Good afternoon. My name is Courtney St. Peters. I also reviewed part of the internal events at-power. I was responsible for reviewing data analysis, system analysis, and human reliability analysis.

Next slide, please.

This slide just goes over my review approach. As you can see, I revised part of the peer review report, a sampling of the PRA and system notebooks during the audit, but during my audit I also asked RAIs if I needed additional information, and I also asked questions during the audit as well and had technical topic discussions at the public meeting.

I also had to ensure consistency with other DCD chapters, because some of my sections covered things that were in other chapters, such as human factors and digital I&C. I reviewed the key assumptions which involved following up on some of those. And in the course of my review, I found my DCD sections were mostly acceptable, but I do have some confirmatory items, along with an open item.

Next slide.

So one of the technical topics that I=ll be talking about was digital I&C. This is one of the -- this is the open items that I do have. During my review, there was a lack of information regarding digital I&C, in particular the common cause failure analysis relating to their digital I&C system.

At the time of writing my SC, I still did not have enough information. After the SC was issued, we did hold a public meeting with KHNP and KEPCO staff, and they committed to providing additional information, and we do have an approach to closing out this open item.

So, if you go to the next slide, the next slide highlights the staff commitments from KHNP and KEPCO. Some of those where they plan to discuss the COMMON Q software similarities with Westinghouse, they are going to evaluate the level of detail -- of the model detail that they currently provide. There are also plans to evaluate the architecture of the digital I&C compared to the reference plant.

They will be adding the software common cause failure events to their PRA, and this information will be incorporated into their next PRA update, and they are going to plan to update the DCD. During these updates, I am in communication as well with the digital I&C staff to ensure that this is consistent with what their information is as well.

Next slide?

The other technical topic I had to highlight was related to RCP seal LOCAs, which I know was discussed earlier.

No, no. I was hoping you guys already had all the questions out of the way.

So during the course of my review, and along with other reviewers, the question related to RCP seal LOCAs came up. It was evaluated by KHNP and KEPCO as a model uncertainty, a sensitivity analysis, and it was a key assumption as well.

During our review, we noticed they have the failure probability of $1E^{-3}$ per pump, which was based on engineering judgment. Before they performed their seal LOCA model, we requested additional justification. In the course of that, they did provide their seal model testing results, which was proprietary information, but the results did support their assumption.

They plan to review and revisit the model uncertainty and their sensitivity analysis during their PRA update, and this item is considered a confirmatory item.

MEMBER STETKAR: I read the RAI response, and I won=t quote any numbers from it because it=s a proprietary document.

MS. ST. PETERS: Yes.

MEMBER STETKAR: It raised several questions in my mind, and I would -- does the staff have the WCAP report?

MS. ST. PETERS: Yes.

MEMBER STETKAR: You do?

MS. ST. PETERS: During the audit, yes, we=ve had access to it.

MEMBER STETKAR: No, I=m sorry. Do you have it in hand?

MR. PHAN: We don=t have that report.

MEMBER STETKAR: We would like to have, somehow, access to that WCAP report. I would anyway. I don=t know how we make that happen because it=s not submitted on the docket. But I will tell you that numbers in that RAI response, especially given what we=ve learned about other pumps= designs and other claims for reliability of advanced seals, are suspect.

So I would like to have the opportunity to somehow be able to read the source document and see what types of engineering, what types of design, and in particular what amount of testing has been done on those seals under what conditions.

MR. SCHNEIDER: Excuse me, but I=d like to basically add -- Ray Schneider, Westinghouse. I=d like to add some information.

MEMBER STETKAR: This is, by the way, an open meeting, so be careful about what you say.

MR. SCHNEIDER: Right.

MEMBER STETKAR: This is a public meeting.

MR. SCHNEIDER: I understand. But I=m going to basically do is just try to put everything in perspective, because I think there is a misinterpretation as to which seal design we are looking at.

The RCP seal design used in the KNGR plant is a derivative of the combustion engineering design, which is totally different than the -- which is a hydrodynamic seal design, which is totally different than the design of those typical Westinghouse seals.

The hydrodynamic designs typically -- and this design has been evaluated by the NRC, and an approved consensus guidance document for a very similar kind of construction in 2004 applied to the CE fleet.

And the version of this seal that is being used for the KNGR design is a three-stage. Each stage is fully -- full pressure stages. They=re dynamic seal stages, so they -- all stages have to fail sequentially in order for a substantial leak above this by -- of gpm above about 10 gpm to occur per pump.

In the testing on the old versions of a parallel seal design, not the KSB design, years ago, which just tested these to 72 hours under station blackout conditions. We recently completed testing of the KSB seal design in Germany for the advanced seal version of the type F seal, which is going to be going into these plants.

The type F seal was a modified redesign of the KSB pump seal, specifically for the intent of dealing with station blackout scenarios. They modified some very small interference, so the seal would be able to grow and contract without any interference or distortion.

They reevaluated and redesigned or rerequired the composition of the elastomers such that the elastomers would be able to take -- been tested to at least 72 hours at -- in the 560 degree range without any impact. And there was a full-scale test also completed a few months back, which basically shows that operation for 120 hours, of which 72 remain station blackout conditions, had minimum leakage.

And we also looked at the possibility of going to low subcoolings to find out if there is impact to pop opening, and we couldn=t see any observable pop open, which we didn=t expect to see anyways, but we had them do that special test.

So we provide -- based on this new information, a new topical was written which slightly modified the existing information that was part of the old combustion engineering plant topical, which I believe is in the hands of the NRC now.

And these values will basically show that the failure rates dropped a little bit based on these new enhancements, but they are in the 10^{-3} range. And this typical of what you see in the combustion engineering fleet, and this was also tied to the question you had with the -- with how they may have been treating these seal leakages in the consequential steam engineering and tube rupture report.

So it=s a different design. There=s a 10 -- leaking seal, and I just wanted to basically make that clear, so we=re not mixing apples and oranges.

MEMBER BLEY: Do you know if you are submitting that topical?

MR. SCHNEIDER: I=m putting --

MEMBER BLEY: You don=t know. That=s okay.

MEMBER CORRADINI: But I thought John=s initial question was, can we get the topical?

MEMBER STETKAR: It=s not a topical report. It=s simply a -- it=s a --

MEMBER CORRADINI: A technical report.

MEMBER STETKAR: -- it=s a WCAP technical report.

MEMBER CORRADINI: Technical report.

MEMBER STETKAR: It has not been submitted on the docket.

MR. J. OH: This is Andy Oh, KHNP,

Washington office. That WCAP document is not submitted to the NRC, but it is admitted to the electronic reading room, and the staff can audit and see that document.

MEMBER MARCH-LEUBA: So do ACRS members have access to the electronic reading room? Or can we get access?

MEMBER STETKAR: We don=t need to work out the logistics at this subcommittee meeting. This is just a request.

MR. CIOCCO: This is a document that we requested. This is a document that we audited. So we don=t -- that=s as far as we=re going with it. And, I mean, if you have a request for the document, I guess you could provide it to KHNP. But for the staff, it=s an audit document.

MEMBER STETKAR: We=ll work through the -- I think we did, but it=s on the record. They can say no.

MEMBER MARCH-LEUBA: Courtney, back to digital I&C, once you put the common cause failures on software, my gut feeling is that the reliability is going to be achieved through the watchdog and not through the software itself. And there is going to be a big reliance on the watchdog, which we will review under Chapter 17, how it=s implemented.

I=m just giving you a heads-up that whenever this PRA gets done of the whole digital I&C system, please make sure to review the watchdog implementation and how it=s built into the PRA, because that=s what is going to give the numbers.

MS. ST. PETERS: Okay. And I=ve been talking quite frequently with the Chapter 7 reviewers, and we=ve been communicating on the information we had and what we=ve seen. And so I=ll definitely keep that in mind.

MEMBER MARCH-LEUBA: Just make sure the PRA evidence is the watchdog.

MS. ST. PETERS: Okay.

MEMBER MARCH-LEUBA: Because that=s where the numbers are going to come from.

MEMBER REMPE: So I think I should have asked this for Odunayo instead of you, but did you -- were you present earlier today when we talked about the use of the NUREG-1570 Westinghouse conditional consequential steam generator tube rupture probabilities?

MR. AYEGBUSI: Yes, I was.

MEMBER REMPE: So you heard my question about using Westinghouse data for CE, and you=re aware of the staff=s work in research on NUREG-2195?

MR. AYEGBUSI: I=m not aware of the -- I wouldn=t say that I heard your question and was able to follow your question. Neither am I aware of the NUREG you mentioned.

MEMBER REMPE: So I did -- I think -- I believe I heard KHNP say, AYes, we will be looking at this, and that we might want to change those values.@ And I think it would behoove the staff to also look at that report and think about whether the conditional probabilities for consequential steam generator tube rupture should be changed.

MR. AYEGBUSI: Understood. We=ll take a look at that.

MEMBER REMPE: Thank you.

MR. AYEGBUSI: Thank you.

MS. ST. PETERS: So if there=s no other questions, that=s the end of my presentation, and I believe the end of Level 1 at-power.

MR. NAKANISHI: Good afternoon. My name is Tony Nakanishi, and I=ll be discussing the internal fire and flood review. This morning I think the applicant provided a good summary of the individual tasks that are performed for both Level 1 floods and fires. So I=d like to focus more on kind of the high-level staff thinking relative to how the fire and floor PRA is supporting the design certification application.

Can you go to the next slide, please?

So, overall, we=re finding that the fire PRA -- we=re fairly comfortable with the applicant=s approach and assumptions that are being used. You know, we have a few questions related to -- mainly with respect to documentation and the DCD.

But, overall, you know, the staff -- we=re finding that the applicant used the -- you know, the industry standard approach, NUREG-6850, and applied it to the extent practical and appropriate for design certification stage by, you know, assuming appropriate assumptions like, you know, full room burnout for most of the fire compartments, assuming at least, you know, transient, general transient, given a fire cable, you know, in lieu of doing a detailed cable or circuit analysis, you know, assuming sort of a, you know, bounding approach.

So at this point, you know, I think we=re fairly comfortable with the methodology and assumptions. I will say that Hanh had mentioned they=ve gone through a conversion activity, and we need to sort of circle back and make sure the results -- kind of confirm the results.

But overall we are finding that the applicant is also, you know, using risk insights to, you know, propose things that need to carry through the design and operation, like, you know, in particular the routing of the transformer cable was judged to be placed outside the turbine building, for example.

They have identified some cables that need to be protected to ensure the risk profile, which leads to my next slide, if you could -- so one item that we do want to highlight that -- you know, the applicant had mentioned this also I think in the morning presentation, but they have identified in certain risk-significant fire compartments some cables that need to be either, you know, protected or, you know, can be shown later that it won=t affect the component that it controls.

So to ensure that this sort of carries through, we asked the applicant to identify a COL item to make sure that, you know, this key PRA assumption remains in place.

So, again, overall, you know, we are fairly comfortable with the fire PRA approach. And if you have no questions, I can move on to the flooding analysis.

MEMBER STETKAR: I have questions. You asked them a question about why they didn=t model fire-induced -- if we want to call it spurious safety injection or a fire-induced safety injection, and their response was, well, the normal reactor coolant system pressure is higher than the shutoff pressure of the safety injection pump, so it=s not a problem.

In my experience, a safety injection is a lot more than just starting the safety injection pumps. It=s isolation of the containment. It=s other things that get isolated, like main feedwater gets isolated.

So for spurious safety injection, we only care about whether the pumps can pump water into a higher pressure.

MR. NAKANISHI: So that=s a good point.

MEMBER STETKAR: Okay.

MR. NAKANISHI: We=ll look at that and

make sure --

MEMBER STETKAR: What I=m trying to probe here, by the way, is the level of detail that the staff thought about these types of initiating events, not just looking at some NUREG someplace and accepting what somebody says with a focus on pump deadheading. So, okay, thank you.

The second question I had on the fires -- you heard -- were you here this morning?

MR. NAKANISHI: I was.

MEMBER STETKAR: Okay. Or this afternoon, whenever the heck it was. Why are you okay with the way they treated main control room fires, given the fact that I put out the fire in 9.9 minutes and have some amount of damage on the thing that I=ve taken to be calling a control of console to avoid confusion with something that somebody else might call a cabinet?

MR. NAKANISHI: So you gave me something to think about also there. We=ll look at that and see if there needs to be any adjustments.

MEMBER STETKAR: And, finally, something that I didn=t know reading anything in the DCD or looking at the models or looking at success criteria or reading about human reliability analysis until I got to the staff=s review of the fire stuff is that apparently one of the two motor-operated valves in the line for each pressurizer POSRV is normally deenergized and somebody has got to run out and connect power to -- you know, close a breaker or something like that, in order for the operators in the control room to initiate feed and bleed cooling.

MR. NAKANISHI: That=s correct. We didn=t know that initially either.

MEMBER STETKAR: It=s interesting it=s not documented anywhere. Did the staff look at all at the human reliability analysis? And now, because I didn=t learn about this until fire, but it applies during power operation, and it applies even during low power and shutdown.

MR. NAKANISHI: Absolutely. Actually, we --

MEMBER STETKAR: Do you look at how that affects the human reliability analysis for --

MR. NAKANISHI: Right. So we --

MEMBER STETKAR: -- those actions?

MR. NAKANISHI: That=s exactly the question that=s outstanding still.

MEMBER STETKAR: It is?

MR. NAKANISHI: Yes. We actually noticed it as part of the internal flooding analysis where -- there was some inconsistency and assumptions where during internal events -- well, let=s see, for flooding they have an assumption where for external -- you know, operator actions external to the control room will basically fail.

Now, that may be a potential thing that needs to be looked at if the feed and bleed will require ex-control room action. So we have an RAI to kind of figure out -- you know, figure out that assumption.

MEMBER STETKAR: But -- and I read that for both the internal fires where I first learned about this and the internal flooding that you just mentioned. Did you also try to follow up on how they treated it during a plain vanilla loss of feedwater initiating event that propagates -- this is full power operation, loss of main feedwater, internal initiating event, that eventually gets to feed and bleed cooling?

MR. NAKANISHI: Right. So --

MEMBER STETKAR: It strikes me that the

human error probability for failure to initiate feed and bleed on this particular design would be higher than the human error probability for a design where the operator simply has to walk up to the control board and open the valves.

MR. NAKANISHI: So we did look at the HRA notebook, and, you know, their methodology identifies that particular step as --

MEMBER STETKAR: They do.

MR. NAKANISHI: -- external to control room action. So they=ve considered that.

MEMBER STETKAR: Okay. Thank you. Do you know how many operators they have to dispatch to -- given the distribution of things throughout the auxiliary building, I=m assuming that these -- they have to connect -- they have to close breakers in four separate rooms.

MR. NAKANISHI: I don=t know.

MEMBER STETKAR: Okay. By the way, if KHNP wants to answer this, you can come up and get it on the record.

MR. J. OH: Yes. This is Andy Oh, KHNP, Washington office. In order to implement a POSRV feed and bleed operation for -- circuit breaker for the belt should be closed in a local area. That is -- based on my memory, that is located in second floor of the aux building, and that takes time for -- within 30 minutes operator can close the circuit breaker, and then the NCR, the POSRV belt can be open, and then implemented to the feed and bleed.

The reason that we -- the design of that feature is inadvertent open for the POSRV makes some LOCA. So in order to prevent that inadvertent just per function of opening valve in -- on the NCR, that leads to the direct LOCA. So that=s the reason we just made some of the redundant features to the -- to make some circuit is in different location.

MEMBER STETKAR: Thank you. You didn=t answer how many people do this. Is it single operator or --

MR. J. OH: Yes. Per -- I think that one single people can dispatch to that place that can close the circuit breaker.

MEMBER STETKAR: Okay. When you say Athat place,@ it=s -- I know the rooms they have to go to. They have to go to four separate rooms. No? Greg, come on up to the mike. I=m actually trying to understand this because there=s no description of it whatsoever in the DCD, and it=s a different design feature than --

MR. ROZGA: Greg Rozga from Enercon. Yes, there is two rooms. There is an I&C equipment room, Div 1 and Div 2, that they have to go to. They=re in the Alpha and Bravo quadrants. Whether it=s on the second or third level, that I don=t recall.

MEMBER STETKAR: But this says the power is disconnected to the motor-operated valve, there=s four POSRVs, so I=m assuming there=s four motor-operated valves. And maybe I=m wrong; I was just assuming that one would be in each of the four AC or DC divisions. But I don=t even know if they=re AC- or DC-powered motor-operated valves.

MR. ROZGA: They are -- correct me if I=m wrong, they are AC motor-operated valves, but they have a DC power supply through an inverter.

MEMBER STETKAR: Okay.

MR. ROZGA: Is that correct?

MEMBER STETKAR: Well, I don=t want to -- time on the details, but --

MR. ROZGA: Thank you.

MR. NAKANISHI: So we=ll move on to

internal flooding. If you could just -- the other way.

So, again, internal flooding -- you know, overall the staff finds that the applicant basically used the industry standard approach, you know, consistent with the ANS standard and the staff=s SRP.

As Hahn mentioned, this model was peer reviewed also. This is one of the other models that was peer reviewed, and basically the peer review team found that, you know, the PRA essentially meets capability category 1.

There was one finding relative to uncertainty or certain assumptions being not documented and things like that. But overall, you know, I think the applicant provided a lot of detail in terms of characterizing the flood scenarios. You know, the partitioning represents the design, and so, again, we=re fairly comfortable with the overall methodology and assumptions.

One thing, if you=d go to the next slide, one item of potential interest is regarding the maintenance-induced floods. The applicant initially screened out this as a potential initiating event where an operator would inadvertently operate a component leading to a flooding event. And there was a quantitative approach that we felt that wasn=t quite defensible.

So the applicant identified a COL item for the -- the COL applicant or holder to do a plant-specific analysis when more details in terms of procedures and things like that are available. So we=re thinking that=s an acceptable approach.

So that=s it in terms of internal flooding. Are there any questions? Thank you.

MR. WAGAGE: My name is Hanry Wagage. I=ll be discussing the results for internal events, internal fire, and internal flood at-power Level 2 PRA.

MEMBER MARCH-LEUBA: Do you have your light green?

MR. WAGAGE: My name is Hanry Wagage. I=m discussing my review of internal events, internal fire, and internal flood at-power Level 2 PRA.

I reviewed DCD Section 19.1 related to Level 2 using SRP 19.0 guidance. Reviewed Level 2 methodology, demonstrates containment event trees, decomposition event trees, and release categories. Some of these informations were missing or discussed, for example, how the decomposition event trees were analyzed, were not discussed. So then I discussed -- I asked the applicant during audit or issued RAIs and got changes to DCD adding this information.

Audited APR1400 PRA notebooks related to this area, and I looked at the review topics in this large release frequency in the PRA review report. Discussed -- as I said, I discussed technical issues with the applicant during audit and during public meetings.

To highlight one of the areas I review -- I have described, when we noticed that there are two source term categories in internal events PRA, they differ only by 10 percent release area. One is containment leakage, .1 cubic foot opening, and other one contains a breach, one cubic foot area.

So when they looked at the source term, cesium iodine, which represents iodine release, we found that it does not justify the significant change from the area because the release was so high. And we asked the applicant to explain.

Finally, the applicant explained that the reason was that for one case, STC-17, containment pressure stays high because the leak containment pressures stays high. Because of that natural circulation, cooling is high. So because of that, there is less revaporization of isotopes in the pressurizer. That=s the reason that the STC-17 data load found this justification acceptable and reasonable.

Next slide?

Internal fire and internal flood Level 2 PRA, there wasn=t much information in DCD to bring it to a Level 2. But applicant described Level 1/ Level 2 PRA for internal events, but not much for internal fire and internal flood.

So I asked the applicant to provide this information, and applicant updated -- proposed updates to DCD providing this information. Applicant stated that applicant used the same methodology used for internal events for internal fire and internal flood Level 2 PRA. We find that acceptable because initiating events would not affect how the containment would behave in Level 2.

Next slide? Next?

MS. NEUHAUSEN: Good afternoon. My name is Alissa Neuhausen. I=m a technical reviewer in the Structural Engineering Branch. I was responsible for the review of the PRA-based seismic margin assessment. In addition to myself, Robert Roche and Hanh Phan also contributed to the review.

The intent of the review was to ensure that the applicant=s size of margin assessment is reasonable and acceptable. To reach a conclusion, the staff reviewed the scope, level of details, and technical adequacy of the applicant=s approach. The staff followed guidance in ISG-20, implementation of a PRA-based seismic margin analysis for new reactors, and SECY 93-087.

The staff focused on information provided in DCD Section 19.1.5.1, seismic risk evaluation, and Table 19.1-43, seismic fragility analysis results summary. The number on the slide is incorrect. It should say 43.

In the next two slides, I will discuss the applicant=s seismic fragility evaluation and the status of the plant level, high confidence of low probability of failure, or HCLPF capacity.

The applicant=s initial submittal, Rev 0, included HCLPF capacities for structures and components based on the reference plant design response spectra. Based on this submittal, the staff issued eight RAIs to clarify the SMA fragility evaluation approach.

The fragility evaluation HCLPF capacities are referenced in the APR1400 CSDRS, thereby ensuring an adequate margin of design based on DCD level information.

The applicant applied the conservative deterministic failure margin approach to determine the HCLPF capacities. The staff found that the fragility evaluation is in accordance with guidance in ISG-20. The fragility evaluation demonstrated that site-independent structure HCLPF capacities are greater than or equal to .5 g, component HCLPF capacities are greater than or equal to .5 g, and site-dependent structure HCLPF capacities are greater than or equal to 1.67 times the GMRS PGA.

For the at-power seismic margin assessment, the staff found that the method used was acceptable. For low power and shutdown modes, the PRA-based SMA was not addressed in the DCD. The staff, additionally, requested seismic-induced dominant mixed cutsets containing seismic failures, random failures, and operator actions in sequence level HCLPF capacities during at-power and low power shutdown modes, before a conclusion can be made on the acceptability of the PRA-based SMA.

That=s the end of this portion of the presentation, if there are no questions.

MR. PHAN: Thank you, Alissa.

The next topic is on other external events. For other external events --

MEMBER KIRCHNER: Hanh, before you go on, Mr. Chairman, can I ask a couple of questions, at risk of backtracking?

CHAIRMAN BALLINGER: You can ask more than a couple.

MEMBER KIRCHNER: While the presenters are here. The first one is on sensitivity analysis. I just -- I looked at that, and you had some cases where, without going into numbers because I think this must be proprietary, the GSI-191 sensitivity case showed a significant increase on CDF, and then the statement, AWell, this is well within the Commission=s goal.@

So what do you do when you look at these sensitivity analyses? What do you -- and I think that statement is made several times. So I was left somewhat perplexed as to what the takeaway is on sensitivity analysis.

MR. PHAN: I would ask Courtney or Ayo to --

MEMBER KIRCHNER: Yes, we lost Courtney already. Sorry.

MEMBER MARCH-LEUBA: They=re hiding behind the --

MEMBER KIRCHNER: They=re hiding from me.

MR. AYEGBUSI: Can you please repeat the question? Because some --

MEMBER KIRCHNER: Yes. My question was on sensitivity analyses. You had five or six cases -- I will not go into the numbers because I believe this is proprietary, but at least one case GSI-191 showed a significant increase in CDF.

But then the tag line was although that appears significant, it=s well within the Commission=s goal, and that was stated several times. So what are you looking at when you look at these sensitivity analyses?

(Pause.)

CHAIRMAN BALLINGER: We=ve got a little too much dead air here. How about we refer this?

MEMBER KIRCHNER: Okay. And then I=m

processing a lot of information, so I=m one or two presenters behind. I was -- I wanted to go back to the case of the release of fission products. The explanation given was that, in part, the -- you had a significant, larger opening in the containment, and I expected that to be the culprit.

But then you go on to say that the reduction in containment pressure made natural convection cooling of the pressurizer less effective, so you had revaporization of iodine. How much versus the -- how much was that a factor versus the opening size? I would have thought that would be in the noise. I=m just intuiting that the pressurizer temperature isn=t going to be a lot different.

MR. PHAN: Actually, the applicant used a natural circulating heat transfer correlation. I plugged the numbers to point out that when the pressure is high, then it increases heat transfer by a factor of three or -- I don=t remember the exact number. I found that that was reasonable explanation for significant change in release, although the area is not --

MEMBER MARCH-LEUBA: You=re talking about the pressure in the containment air?

MR. PHAN: Yes. We have one case --

MEMBER MARCH-LEUBA: You said one --

MR. PHAN: -- of this containment air pressure. In one case, it=s leakage .1 cubic foot. It does not drop the pressure in the containment. In the other case, in one case it=s 1.1 square foot area. The other case it=s one square foot area. You drop the containment pressure significantly. When the pressure is low, density of air is low.

MEMBER KIRCHNER: I understand all that, but the thermal inertia of the pressurizer and the -- and what=s going through it is going to dominate the temperature, not the external cooling of it. I suspect it was more the area than this phenomenon, but I --

MR. PHAN: Actually, area only could not explain this. Area change is 10 times. That release is 357 times. So that=s why we got the applicant to explain it. Applicant provided --

MEMBER KIRCHNER: They=re blowing down the containment with a bigger area. As you said, it depressurizes the containment. That=s going to take a lot more out with it.

MR. PHAN: That=s right. That case

viewed the higher release. Because the depressurization dropped the pressure, then it does not provide sufficient cooling for the pressurizer, then revaporizes happen in the pressurizer, in that case only at high release dose.

MEMBER KIRCHNER: Thank you, Mr. Chairman.

CHAIRMAN BALLINGER: Where are we?

MR. PHAN: We are on the other external events, Slide 32. For the other external events, the staff reviews ensured that the applicant=s assessment is comprehensive in scope, the approach used for evaluating and screening out the external events conforms to the guidance, the screening criteria and the justifications used to support the screening out of these external events are rational, and the external events treatments are reasonable.

The applicant assessed the external events following the guidance in Part 6 of ASME/ANS PRA standard, specifically in Appendix 6-A, which identifies the external events that require considerations and supporting requirement EXT-B1, which is the initial preliminary screening for screening out an external event. And to the staff this is acceptable.

Next, please.

The staff review finds that the applicant did not follow the SRMs on the SECY-93-087 to perform quantitative or bounding analysis for the external events, such as high winds, hurricanes, tsunami, and so on.

In addition, the DCD Revision 0 does not discuss how the main control room would cope with the external fires. Furthermore, the COL information item 19.1(8) is not complete, missing events, tsunami, and others.

Therefore, at this point, the staff concludes that the external event assessment, that appears in the DCD Revision 0 is incomplete.

In addressing the staff findings, the applicant agrees to revise the DCD to include the quantitative or bounding analysis and address the main control room=s issues. In addition, applicant will revise the revise the COLs information items 19.1(8) to include those missing events.

Up to this point, we have presented you the PRA during at-power. The next topic is on the PRA during low power at shutdown.

MS. POHIDA: Good afternoon. My name is Marie Pohida, and first I=ll be discussing the approach that I took for evaluating the low power and shutdown PRA for internal events.

Consistent with the SRP and the draft low power and shutdown standard, I reviewed the plant operating state definitions for completeness. All POSs were defined, including reduced inventory operations, water solid conditions, and cavity flooded conditions.

For each POS, the time to boiling and the time to core uncovery was determined, along with the status of all open RCS penetrations, RCS level, and decay heat. I reviewed the event trees for each POS. They were not included in Revision 0 of the DCD, but they will be added to the DCD Revision 1. Okay?

I also reviewed the implementation of Generic 88-17 regarding RCS level and temperature instrumentation, the availability of pumped injection, the installation of steam generator nozzle dams, the potential for vortexing of the shutdown cooling pumps, and containment closure during reduced inventory conditions.

I also reviewed the risk insights to

identify what SSEs should be considered for potential tech specs, LCOs, under Criterion 4 of 50.36. An example of that would be containment closure during reduced inventory conditions.

We did perform a confirmatory calculation of the applicant=s low power and shutdown MAAP analyses for source terms, and Jason will be addressing that topic later this afternoon.

And I also made sure that significant operational assumptions were included as risk insights or tech specs, as applicable. And an example of that would be the order of nozzle dam installation, you know, such as the hot leg nozzle dams, they=re always installed last, and the steam generator or hot leg nozzle dams are always removed first.

MEMBER STETKAR: Before you flip to the next slide, I=ll ask this one here now. I noticed when I read the SER that there was some discussion about interfacing system LOCAs during low power and shutdown, and that apparently the applicant will provide a DCD update to state that once a primary event is established there is a negligible IS LOCA vulnerability.

It sounds to me like that whole discussion focused only on the classic narrow focus of overpressurizing a low pressure piping system. When shutdown cooling is operating, they have a low pressure letdown flow path open.

That indeed does not go through the flow path that is discussed, at least what I could read in the SER, and yet if that flow path is open, and charging is not available to put water back in, that to me is an interfacing system LOCA. More water is going out of the reactor coolant system than is going in, and where it=s going to is outside of the containment.

So I was curious, since you=re happy with the fact that they don=t have any IS LOCAs, how do you disposition that? In fact, it=s called initiating event JL, but that=s strictly pipe breaks. I=m talking about other things that can happen that keeps water draining out and not going in.

MS. POHIDA: Okay. I=m going to -- this is going to be a reach of my memory here. I may have to go and take that question back.

MEMBER STETKAR: Okay. That=s --

MS. POHIDA: JL breaks I believe were

evaluated in the Level 2 portion of the SER, but to the details I=m going to have to go back and check on that.

MEMBER STETKAR: JL -- but my point is, JL is a break, and I don=t care about pipe breaks.

MS. POHIDA: I understand.

MEMBER STETKAR: I care about flow paths that deliver water outside of the containment, and no water going back in. That to me is an interfacing system LOCA. I don=t have to have a pipe rupture.

MS. POHIDA: Operator-induced flow diversions --

MEMBER STETKAR: No, no, no. I didn=t say operator-induced; did I?

MS. POHIDA: Okay. Not --

MEMBER STETKAR: This is a normal flow path that is open, such that if the water is going out and no water is going back in, the water goes out and goes away and doesn=t go back in.

MS. POHIDA: Yes.

MEMBER STETKAR: Now, that can happen for a variety of different causes that are not related to overpressurizing a system or operators or anything. Operators could be part of it. It=s hard to overpressurize it when there=s no pressure, but --

MS. POHIDA: Okay.

MEMBER STETKAR: And all of -- everything that I read in terms of rationalizing why that type of phenomenon was not important dealt with things like, well, we have orifices, and those are high pressure orifices as well. This line connects downstream of those high pressure orifices. In fact, those high pressure orifices are isolated when you=re shut down because you can=t get any flow through them at low pressure.

So just take -- I just wanted to make that comment to see if you had thought about it. That=s all I have on this one. You can go to the next slide.

MS. POHIDA: Thank you. I=d like to take that question back and evaluate that.

May I go on to the second slide, please?

MR. ANDERSON: Excuse me, Marie?

MS. POHIDA: Yes, sure.

MR. ANDERSON: If I could -- hi, Ross Anderson with Enercon. We deliberately included diversion events in our review and development of initiating events. We took a look at industry history of diversion events of any sort, not necessarily to another system, but water being taken out of the primary, put elsewhere, and, therefore, you have what could be called a LOCA, called a diversion event, because we don=t see much in the way of pipe breaks with the system depressurized and cooled down.

But there were a number of events, typically not always high volume, where a lot of water was removed from the primary system, and that was the basis for our -- what was our small LOCA term ESL, Sierra Lima initiator.

So we did include that, and we judged at the time, because we reviewed the issue of intersystem LOCA, we thought we were covered by the diversion since --

MEMBER STETKAR: But on the other hand, Sierra Lima events are inside the containment.

MR. ANDERSON: We didn=t limit the definition that way. We called them a diversion of any sort to anywhere. So could be in, could be out.

In terms of the Level 2 implication, I can=t address that for you. But in terms of the Level 1 impact, it=s covered.

MEMBER STETKAR: You think you=re covered on Level 1 in terms of the scope of --

MR. ANDERSON: Yes.

MEMBER STETKAR: -- that particular initiating event.

MR. ANDERSON: Yes.

MEMBER STETKAR: Okay. I=ll have to think about that. Level 2 is -- you=re right, Level 2 can be a different issue because that=s --

MR. ANDERSON: It=s a different table.

MEMBER STETKAR: -- that is a release pathway, but -- okay. Let me think about that one. Thank you.

MS. POHIDA: Shall I continue?

MEMBER STETKAR: Yes.

MS. POHIDA: Thank you. Okay. Based on staff questions, the applicant added or augmented the following tech specs and DCD descriptions. One is regarding containment closure when the RCS is open via the pressurizer manway until the refueling cavity is flooded, 23 feet above the reactor vessel flange.

Two trains of safety injection are operable in hot shutdown, cold shutdown, and refueling, when the refueling cavity is less than 23 feet above the reactor vessel flange. Midloop operation was defined as taking place greater than 96 hours post-shutdown.

The availability of the PARs and igniters during shutdown was documented as a risk insight in Chapter 19. And procedures to ensure that a steam generator or hot leg manway is open to prevent a rapid loss of inventory when any cold leg penetrations exist was included as a risk insight.

So I found the applicant=s approach to be consistent with our guidance, subject to closure of the open and confirmatory items.

MEMBER STETKAR: One that I=m honestly really -- I=m not trying to -- I=m really puzzled about this one. In the low power shutdown model -- now talking Level 2, low power shutdown, get you oriented. The models right now say, AWell, in plant operating state 3B and 4A, the equipment hatch may be open.@

MS. POHIDA: That is correct.

MEMBER STETKAR: And that to isolate the containment, personnel need to reclose the equipment hatch and at least secure it with a minimum of four bolts.

MS. POHIDA: Yes.

MEMBER STETKAR: Okay. I got that. One question is, in the DCD, it says something to the effect, and I can=t find the quote right at the moment, but if a station blackout occurs, no AC power whatsoever, that activity is failed. In other words, it=s assumed they can=t reseal -- can=t move the equipment hatch in place.

It strikes me that I don=t know the plant design, but most plants I have looked at have one power supply for a crane that can move the equipment hatch not for independent safety-related power supplies. So it=s not clear to me why that=s only impossible if I have a station blackout, why it=s not impossible when I have failure of power at some bus.

So I don=t know if you=ve looked at that, which might be a lot more likely than station blackout.

MS. POHIDA: Okay. Could you please restate the question?

MEMBER STETKAR: Okay. The statement in the DCD -- and I can=t find the quote right now; I can find it later -- is that that activity is failed --

MS. POHIDA: Yes.

MEMBER STETKAR: -- if you have a station blackout --

MS. POHIDA: Yes.

MEMBER STETKAR: -- meaning no AC power anywhere. In my experience, it might be different for this plant design. In my experience, you do not have four independent safety-related power supplies to a crane that can move the equipment hatch. You typically have one power supply, and it often is not safety-related.

So my question is: does the model actually account for the real power supply to that crane, and where is it powered from? Because I would bet -- I could be wrong -- that it is not -- does not have redundant power supplies from all four safety buses.

MS. POHIDA: Okay. In POS 3B, that is -- let=s see, that is hot shutdown.

MEMBER STETKAR: I=m not -- I don=t care about POS here.

MS. POHIDA: Okay.

MEMBER STETKAR: I care about the power -- what -- let me phrase this very explicitly. What is the power supply to the crane that can move the equipment hatch? Is that clear enough?

MS. POHIDA: Mm-hmm. I --

MEMBER STETKAR: If I know the answer to that question, I will then know how vulnerable you are to not being able to move the equipment hatch.

MS. POHIDA: Okay.

MEMBER STETKAR: So what is the power supply to that crane?

MS. POHIDA: I will have to go back and check on that.

MEMBER STETKAR: More importantly, though, is there is a discussion in the DCD -- and I didn=t stumble across this until I found it in the SER, which is good -- in the SER it says according to tech spec -- I=ll get you the right number -- 3.6.7, POS 4B, 6, 10, and 12A may have the equipment hatch open. So that got my attention because it=s assumed that it=s closed in those plant operating states in the model.

So I went and looked up the tech specs, and indeed the tech specs say that containment integrity is required during modes 1, 2, 3, and 4. Section 3.6.7 indicates that the hatch must be closed and secured with at least four bolts. One door in each personnel airlock must be closed, and the containment isolation valves must be closed or operable, during reduced inventory configurations in mode 5 or 6, which is consistent with what you have.

Section 3.9.3 further requires those same containment integrity conditions whenever fuel is being moved in mode 6. However, other than that, the tech specs are silent regarding the need to have the containment hatch closed at any other condition during mode 5 or mode 6.

The SER says, well, the question is considered closed, but issues remain unresolved and related to RAI 8546, question 16-149. So I dutifully went to look at that, and it seems that question pertains only to tech spec requirements during those reduced inventory configurations.

So my real question is: according to the law, can the equipment hatch be open in POS 4B, 6, 7, 8, 9, 10, and 12A?

MS. POHIDA: Thank you for bringing up this question because we spent a lot of time on this during the review. When I initially reviewed Revision 0 of the DCD, there was an inconsistency between the Level 2 portion of the PRA that said we defined -- we built one containment event tree to describe shutdown operations when the pressurizer manway is open and when the refueling -- when the reactor vessel head is off.

And so in the Level 2 section, Level 2 portion of the PRA -- I=m going to speak very slowly so I make sure I get this right -- from POSs 4A and to, you know, POS 12, one containment event tree was used, and it was predicated back that the containment hatch and all penetrations were closed.

I went back and looked at Chapter 16 of the tech specs -- tech spec, you know, 3.6.7, and it says reduced inventory operations. Reduced inventory operations is defined in Generic Letter 88-17 as three feet below the reactor vessel flange. Okay? That leaves a gap. Okay? Because 4B covers those POSs for that one containment event tree that was built, cover operations during vessel head removal and reinstallation.

So if you read the letter of the law, there is -- you know, there was a technical inconsistency between tech specs and the Level 2 portion of the DCD for shutdown. So we asked a bunch of RAIs, and tech specs were modified, and they were modified -- they were modified, and this is still confirmatory. I believe it=s still an open item, to change the applicability for tech spec 3.6.7 so it includes hatch closure during reactor vessel head removal operations and installation operations.

So that=s -- and that is identified and being resolved through RAI 16-149, the change in applicability for tech spec 3.6.7. Does that help?

MEMBER STETKAR: Yes, I am confused. I --

MS. POHIDA: Yes. How can I help?

MEMBER STETKAR: I have a simple question.

MEMBER CORRADINI: Some of the members enjoy confusion on his part. Let=s just let him sit there for a minute.

(Laughter.)

MEMBER STETKAR: Can the equipment hatch be open in what the PRA calls POS 4B, POS 6, POS 7, POS 8, POS 9, POS 12A, POS 12B, and maybe a little bit of 13? That is a simple question. Do the tech specs allow the equipment hatch to be open under any of those conditions?

MS. POHIDA: Okay. Based on DCD Revision 0 or the changes that we=re expecting in Revision 1 of the DCD?

MEMBER STETKAR: I don=t know, because the follow up question is, if I=ve got to have the containment hatch closed during the whole outage, I don=t want to work in that plant. I=m trying to find out the -- how the tech specs and what somebody may or may not be committing to for closing and sealing the containment hatch line up to the condition of the containment hatch that is assumed in the PRA model. And right now, from what I can read, those do not align.

And what Marie has said is the staff has identified that inconsistency, but I don=t know how the tech specs are being revised. And if they=re only being revised to when they=re removing or installing the head, that still does not satisfy all of the other plant operating states that I identified that the model now assumes that the hatch is sealed.

MEMBER BLEY: Okay. So the question arose because in the PRA model you see assumptions about requiring the hatch be sealed.

MEMBER STETKAR: Yes. The PRA model

strictly says only in something called 3B and 4A the hatch can be open and --

MEMBER BLEY: Yes.

MEMBER STETKAR: -- that=s the only --

MEMBER BLEY: That=s the assumption in the PRA, but my question is sort of the other way around. The PRA doesn=t govern operation of the plant, although if we=re using the PRA it ought to match operation of the plant. But I don=t recall that that -- anything like that that was in any tech specs I=ve seen.

MEMBER STETKAR: Well, they have actually instituted a tech spec on this plant, which I think it=s kind of a risk-informed tech spec that says at midloop operation they want the containment equipment hatch closed with four bolts in place. It=s not fully bolted --

MEMBER BLEY: Okay.

MEMBER STETKAR: -- because they feel they are more risk-sensitive during those --

MEMBER BLEY: And I=ve seen other people implement something similar --

MEMBER STETKAR: Right.

MEMBER BLEY: -- although I didn=t think

it was in the tech specs, but this plant --

MEMBER STETKAR: This is actually in the tech specs.

MEMBER BLEY: Okay.

MEMBER STETKAR: It=s tech spec 3.6.7. They have the --

MEMBER BLEY: It=s not a bad idea.

MEMBER STETKAR: No. And they have the standard one that says whenever you=re moving fuel, which is actually what they call POS 7 and POS 9 here, it also has to be in place with four --

MEMBER BLEY: Need to close quickly.

MEMBER STETKAR: No. It=s got to be in place with four bolts.

MEMBER BLEY: Okay.

MEMBER STETKAR: And that=s also pretty standard. What I=m interested in is the other ones. The reason I=m interested in the other ones is that the human error -- the inability to reclose that equipment hatch is a big contributor to large releases during those two relatively short plant operating states when it=s open.

So if it actually is -- can be open during --

MEMBER BLEY: More.

MEMBER STETKAR: -- more, then the PRA model is wrong. On the other hand, if they=re going to write tech specs saying that it=s got to be closed the whole time, I probably don=t really want to have an outage in that plant.

MEMBER BLEY: And I guess the related piece -- and I haven=t seen a tech spec on this either -- is in times when it=s allowed to be open in other plants -- I don=t know what they=ve got here -- it is often open in such a way that you aren=t going to close it for hours because there=s cables and tubing and all kind of stuff running through.

MEMBER STETKAR: Here they -- well, you can get into the timing and stuff like that. That was the first question I had about they say, well, you can=t close it. If you have a complete loss of all AC power, you can=t close it. My allegation is that if you don=t have AC power at some bus, you can=t close it.

They claim that it=s supposed to be clear enough that you can get it closed within whatever time window they have to close it, and they have some criteria in the PRA that says, you know, what triggers that motion.

So I don=t know if we=ve discussed it enough. I just want to understand what -- if the tech specs are being changed, that=s fine. I mean, we=ll pick it up in Rev, you know, whatever the heck it is, one of the PRA and whatever.

MS. POHIDA: But this point is very important because --

MEMBER STETKAR: Yes.

MS. POHIDA: -- you know, times to boiling are exceptionally quick. You know, the time to boiling, whether you=re at -- we do at midloop or with reduced inventory operation, which is defined as three feet below at flange --

MEMBER STETKAR: Early on.

MS. POHIDA: -- or at the flange. It=s only minutes. So the goal of this RAI was to ensure that what was modeled in the Level 2 portion of the PRA was consistent with tech specs. And so 16-149 is supposed to tweak tech specs, so that the tech specs are consistent with Level 2 of the shutdown PRA.

MEMBER BLEY: But John is also bringing up, should the applicant overtweak the tech specs, they can=t operate this plant.

MEMBER STETKAR: It=s going to make life difficult to actually move stuff inside and out of the containment when you --

MEMBER BLEY: Like you need to do during the outage.

MEMBER STETKAR: Somebody is shaking their head, so let=s get feedback from them.

MR. ANDERSON: Hi. Ross Anderson with Enercon again. Just wanted to weigh in. Per tech spec, hatch is closed modes 1 through 4.

MEMBER STETKAR: Yes.

MR. ANDERSON: Below 4, reduce inventory or -- hatch is closed. Otherwise, it can be open, and I believe that=s a standard tech spec convention, so there shouldn=t be any surprises there.

MEMBER STETKAR: That=s right.

MR. ANDERSON: Again, I haven=t gone through to verify that that has been integrated into the Level 2 analyses, but nothing unusual about tech specs and the analyses should be consistent.

MEMBER STETKAR: Good. From what I just heard, then, unless something is going to change, the hatch can be legally open in POS 4B, 6, 8, 10, 12A, 12B, and maybe part of 13, because 13 is kind of --

you=re coming out of mode 5 and 13. It=s a transition-type POS. So I can give up on mode 13 -- or POS 13.

But if it can legally be fully open in 4B, 6, 8, 10, 12A, and 12B, that condition is not modeled in the current PRA. It=s assumed that it=s closed in the current PRA.

MR. ANDERSON: I want to be careful with what I say because I haven=t inspected these parts of the model. But the way they were defined is that it was consistent with tech specs or it should be.

MEMBER STETKAR: The POSs that I read, 4B, 6, 8, 10, 12A, 12B, use the standard containment event tree from the full power PRA, which assumes that the containment hatch is closed because it=s closed during full power.

MR. ANDERSON: And you folks may want to revisit that.

MEMBER STETKAR: You may want to revisit that. A big deal is made out of that narrow window of 3B and 4A. Special analyses are done. A special analysis says I=m exposed to having the equipment hatch open, and I need to close it if only in those two plant operating states.

The analysis for 5 and 11 says that it=s closed with four bolts, and because of that there is a higher conditional probability for overpressure failure of the containment because it=s not sealed as tightly.

It, similarly, is closed in 7 and 9, which are the two when you=re actually moving fuel --

MS. POHIDA: For alteration tech specs.

MEMBER STETKAR: Right. And those plant operating states are basically ignored in the low power and shutdown PRA because the refueling pool is full of water. So they=re not even addressed in the low power and shutdown PRA, and that=s why I=m concerned about 4B, 6, 8, 10, 12A, and 12B. And I think I=m not going to say it again.

MEMBER SUNSERI: Yes. I=ll just add one thing from my plant operating experience. And maybe the staff found this or you=ll learn this from KHNP, but the last few outages I=ve been in in the last couple of plants I was at, the utility had to demonstrate that they could close that hatch within 30 minutes or meet the time to blow requirements, including having a temporary generator stage, if necessary, for the electric hoist. And we ran drills every outage multiple times to demonstrate compliance with that requirement.

MEMBER STETKAR: And, indeed, in POS -- just to follow up on this, in POS 3B and 4A, the models explicitly evaluate whether people can get it closed within whatever time window. And, indeed, you know, like everything in PRA, it=s not guaranteed failed, and it=s not guaranteed success.

The failure probability that=s in there is a relatively large contribution to large releases in those particular plant operating states. In fact, it=s the largest contribution to large releases in those, too.

If the hatch was closed with the same conditional -- I=m sorry, if the hatch was open with the same conditional probability for getting it closed in all of the other plant operating states that I mentioned, the overall large release frequency during low power and shutdown would be much higher than it currently is.

MEMBER SUNSERI: Yes. John, I wasn=t challenging. I was just giving some direction or some advice to the staff of where they could look.

MEMBER STETKAR: okay.

MS. POHIDA: So the intent is that it=s -- this discrepancy between the tech specs and the PRA will be resolved through changing the applicability of tech spec 3.6.7, you know, for containment closure. So that single containment event tree that encompasses plant operation state 4B all the way to 12 is reasonable.

MEMBER STETKAR: Well, that=s one way to do it. That=s making my life as a plant operator more miserable because you want the plant to emulate a PRA. The other way is to make my life what I=d like it to be and make the PRA emulate what the plant is. You have either of those options. Usually, we want people to operate the plants the way that they can manage an outage and get things done, and make the PRA consistent with that, meaning the models, the Level 2 models for those plant operating states would need to change, rather than changing the tech specs.

Anyway, I think the issue is clear. It=s just a matter of what KHNP -- my concern is that this has been punted off into a Chapter 16 issue, and I want to make sure that however it gets resolved that it circles back to the PRA because it has now been punted out of the PRA into Chapter 16, which is tech specs, with apparently the presumption that the tech specs will be changed such that that Level 2 model that=s in there now is valid.

MS. POHIDA: Yes.

MEMBER STETKAR: Yes. So we=ll let KHNP struggle with that one.

MS. POHIDA: Okay.

CHAIRMAN BALLINGER: Can we continue?

MS. POHIDA: I=m completed with my presentation. Are there any more questions or -- well, thank you very much for your time.

MEMBER SKILLMAN: Yes, I do have a question, please.

MS. POHIDA: Oh, I=m sorry.

MEMBER SKILLMAN: I=m on page 19-78 of the safety evaluation. And the text here is as follows, AThe staff also recognized that installed reactor internals did shorten the time to core boiling given possible limited communication between the RCS inventory around the core and inventory in the refueling cavity.@

And the RAI is requesting action, and one of the items is an evaluation documenting the time to core damage given an extended loss of the decay heat removal function with and without installed reactor internals.

MS. POHIDA: Yes.

MEMBER SKILLMAN: And my question is: the reactor internals that that action item is referring to is the plenum or that piece that fits above the core in this design? Is that what that is?

MS. POHIDA: Okay. Now --

MEMBER SKILLMAN: This is part of the low power shutdown.

MS. POHIDA: Oh, I understand.

MEMBER SKILLMAN: It=s in POS 7 and POS 9.

MS. POHIDA: Yes. And this -- the reason why this question was asked is that all plant operational states were evaluated in the PRA before POSs 7 and 9 were quantitatively screened. We asked the applicant for some, you know, thermal hydraulic analysis to identify the time to, you know, core damage when reactor vessel level is 23 feet above the reactor vessel flange.

As you said, two sensitivity studies were performed, one with installed reactor internals and one without, to look at the differences of the time to core damage. And I=m trying to think if -- my mind is drawing a blank here to everything that constitutes the reactor internal package.

But I do know that they are installed; before they are removed that they can limit the communication between water in the reactor vessel cavity and what=s in the core.

If you want a list of the specific components, I would have to go and take that back.

MEMBER SKILLMAN: No. I think what this is referring to is what is known as the plenum or the -- I don=t know what the device is called in the APR1400. But it is the device that really rests above the core. You remove the head and you remove that piece, and if it=s that piece then I understand it=s the chimney of the heat coming up from the fuel up to the refueling canal. I understand that.

I was just -- my first reaction was I don=t think you can have a core without internals because the internals hold the core. But I think this is just a nomenclature issue about this device that rests on top of the fuel.

MS. POHIDA: May I take that back?

MEMBER SKILLMAN: Please.

MS. POHIDA: Thank you.

MEMBER SKILLMAN: Yes. I was just trying to get clarification. Again, it=s on page 19-78 of the safety evaluation.

MS. POHIDA: Okay. Thank you.

MEMBER SKILLMAN: Thank you. Okay.

MS. POHIDA: Are there any additional questions? Thank you for your time.

MR. NAKANISHI: Good afternoon. This is Tony Nakanishi again, and I just want to quickly cover the staff review of internal fire and floor during low power and shutdown.

So I did want to mention that there currently is no staff-endorsed guidance for performing low power shutdown internal fire or flood. But basically the at-power approach can be applied for shutdown conditions, and that=s exactly what the applicant did. And NUREG/CR-7114 provides a little more guidance in terms of how one might take the 68.50 approach and apply it to low power shutdown.

We went through the underlying documentation within -- you know, during our audit, and basically the staff confirmed the approach that was taken, and we find it -- that it=s a reasonable approach.

One thing that=s important for low power shutdown fire and flood is the integrity of the barrier, and that=s one thing that -- we wanted to make sure there are adequate provisions to make sure there are controls.

For example, you know, having -- so the COL items actually cover ensuring appropriate, you know, fire barrier management procedures, configuration control procedures that will ensure that, you know, risk-significant doors and such are monitored with a fire watch or a watch.

So, overall, we think that the applicant approach -- the shutdown fire and flood in a reasonable manner.

So that=s all I had, if there=s any questions.

MR. PHAN: The last topic in this PRA presentation is on the use and application of the PRA. As listed on this slide, the APR1400 PRA was used as an input for many DCD chapters, including Chapter 19.6, physical security; Chapter 14.3, ITAAC; Chapter 16, technical specifications; Chapter 17.4, reliability accuracy programs; Chapter 18, human factor engineering; Chapter 19.2, severe accident evaluations; and also in the environmental report, SAMDA.

Note that there are no risk-informed initiatives included in this application. In conformance with the policy statements on the use of the PRA, the applicant did use the PRAs to improve the design, such as the numbers -- the numbers of the diesel generators and the battery=s depletion time to optimize the plant safety.

The staff reviews ensure that the APR1400 PRA is commensurate with the issues and the applications, the inputs used for the programs is sufficient, and the information in Chapter 19 and other chapters are consistent.

The staff expected that during phase 4 the applicant will revisit these chapters and update the PRA input with the PRA final models and final resource.

Next slide?

In conclusion, due to the phase 2 findings, the staff is currently unable to accept and make final conclusions on the APR1400 PRA in -- of appropriate scope, level of detail, and technical adequacies. And also the APR1400 PRA reasonably reflects the as-designed, as-to-be-built, and as-to-be-operated plant.

This is the end of our presentation on APR1400 PRA. At this point, I would ask, do you want us to continue with 19.2, severe accident evaluation, or we should stop here for additional questions on the PRA?

CHAIRMAN BALLINGER: I have another question. Do we think we need a short break? Okay. I think we=ll take a 10-minute break, come back at five of. We=ll be in recess.

(Whereupon, the above-entitled matter went off the record at 4:46 p.m. and resumed at 4:55 p.m.)

CHAIRMAN BALLINGER: Okay. We=re back in session. I don=t know which one is going to be which, but whichever one it is, please start.

MR. WAGAGE: My name is Hanry Wagage. I=ll be presenting -- leading off Section 19.9 on severe accident evaluation. We have several reviewers presenting this, but first I will go ahead. I will be presenting severe accident prevention.

And using the recommendation of SECY-90-016 and SECY-93-087, applicant addressed severe accident prevention issues, anticipated transient without scram, midloop operation, station blackout, fire protection, and interfacing steam loss of coolant accident.

I have some highlighting issues specifically on station blackout. This morning, also, applicant discussed how the applicant addressed station blackout by having diverse power systems, or, in addition to that, increasing two emergency diesel generators to four. They added more diverse systems to address the station blackout.

MEMBER STETKAR: Hanry, why is the staff -- and don=t give me SECY numbers because I know the SECY numbers, but why technically is the staff concerned with these, and only these, severe accident prevention issues for any new plant that we may review?

MR. WAGAGE: Prevention --

MEMBER STETKAR: Let me ask you, why are we not questioning steam generator tube rupture?

MR. WAGAGE: That comes on this interfacing system loss of coolant.

MEMBER STETKAR: Okay. We=ll talk about that later then. My question is that these were derived from a limited number of PRAs that were performed in the late 1980s and early 1990s for currently operating plants at that time, and said, AGee, these things look like they would be important to risk. You=d better pay attention to them.@

What relevance does that necessarily have for any new plant that might come into us? In other words --

MR. WAGAGE: To address these, some of the new plans address some other issues. For example, some of the plants address ex-vessel steam explosion issues. This one, the plant design is that the issues are addressed differently. The steam explosion applicant is doing analyses, ensuring that they can prevent by -- containment threat by design in the containment. But these aren=t ones the applicant addressed.

MEMBER STETKAR: We can go on. I=m just -- it was a rhetorical question, that the staff was spending time looking at these and only these because of some SECY that was written 25 years ago. And you may be missing other more important things to severe accidents because you=re focusing only on these.

MR. WAGAGE: Other issues that applicant addressed by designing -- for example, the applicant -- on the mitigation, we will be talking about other issues.

Interfacing system loss of coolant accident, there were two methods recommended -- having low pressure systems designed to full RCS pressure and providing means of testing pressure isolation valves and indications.

Applicant identified the systems interfacing with the RCS, and what I found is that the shutdown cooling system -- it was not clear how the applicant had it, because in one place applicant said that the shutdown cooling system is designed to have full pressure or leak test capability, and also it mentioned eliminating interfacing lines.

And it was not clear which way it is doing with eliminating interfacing or having design for full pressure. Then applicant clarified that eliminating unnecessary interfacing lines.

Next, I will be discussing severe accident mitigation progression and features. Severe accident mitigation -- there is a severe accident analysis report that provides details of how applicant addressed severe accident. Four of the areas I reviewed was MELCOR concrete interactions and core debris coolability, and decontam in heating and high pressure melt ejection.

There was a question this morning how the applicant -- how much melt was going to occur to the upper containment. Applicant used the area ratio, area of the annulus and area of the cavity flow area, to determine how much melt would go to the upper containment.

In severe accident analysis report, applicant identified input parameters but did not give the input values. And I asked -- we are going to provide the input values, and found them reasonable, and will update the severe accident analysis report.

In-vessel and ex-vessel steam explosions and containment bypass -- there was a question this morning about concrete type used in this containment. There is a sump in these -- the containment floor. The sump is closer -- the bottom of the sump is closer to the liner, and also recall it is a constricted area. It can accumulate melt to a higher depth. Because of that, general melt and core concrete interactions would not work because it=s a significant higher melt depth.

And applicant mentioned that report Dr. Pilch, Marty Pilch, analyzed that, and I reviewed -- I audited that report on ERR. And I found that the sump being -- having a higher melt depth, it has to -- the cooling or quenching of melt is not guaranteed for basaltic concrete because other type -- two types of concrete are limestone and limestone common sand.

Those two types of concrete, gas generation from the ablation was significant to break the melt crust on the top, but it could not have sufficient justification for breaking the crust for basaltic concrete. Because of that, that those analyzed type of concrete, I asked the applicant to identify that the DCD -- the DCD is going to be updated to limit the type of concrete to basalt and to limestone and limestone common sand.

MEMBER REMPE: I somehow missed that in your SC that you asked them that as an RAI. It=s in there?

MR. WAGAGE: This is during audit. Audit we are -- we asked that, then applicant provided that report.

MEMBER REMPE: Okay. So it=s not included in this Chapter 19 anywhere? Because I didn=t see that. I wouldn=t have asked the question if I had seen it earlier. Is it in what we reviewed in your Chapter 19?

MR. WAGAGE: I stated that I reviewed sump evaluation, but I did not discuss it further.

MEMBER REMPE: Okay. Thank you.

MEMBER POWERS: One point the applicant made in his discussion was the amount of ablation of the concrete when melt streams down onto it versus the depth of the liner below the concrete. And he has assumed limestone concrete, limestone aggregate in his concrete.

Since the ablation depth is dependent on the heat of the fusion of the concrete, the limestone concrete has a much, much higher heat effusion than does a basaltic concrete. And the differences are on the order of a factor of two, which means that a small ablation in the case of limestone concrete would be much bigger in the case of basaltic concrete. Is that a point of issue here?

MR. WAGAGE: I didn=t catch the last part of your --

MEMBER POWERS: Is that a point of issue here? That -- whereas his analysis on streaming suggested that he would ablate a small fraction of the concrete. I mean, it will be roughly -- roughly speaking, twice that much. And you would get to the point where the concrete may not be able to sustain the load on it, and he would in fact fracture out whatever remaining concrete is and expose the embedded liner directly. Is that an issue to consider?

MR. WAGAGE: I think the issue that -- is the cavity filled with water when it transfers heat toward it, not calculate much less ablation, and MELCOR calculates higher-than-MAAP ablation rate for water-filled cavity. This is a water-filled cavity because of that, and there have been melt spread in large area, and the thickness of melt layer is small, and that it does not have --

MEMBER POWERS: No matter how much you spread it, or how much you try to quench it, if you get any concrete ablation you will get more ablation with a siliceous concrete than you will with a calcareous concrete, simply because the heat effusion is so much less in the case of the siliceous concrete. And whereas they ablated like a foot with the limestone concrete, you will ablate like two feet with a siliceous concrete.

MR. WAGAGE: I mean, siliceous concrete is not going to be used in this plant.

MR. PAIK: This is Chan Paik from Fauske and Associates. The difference between the siliceous concrete and limestone or a limestone common sand has -- limestone has a lot higher decomposition kind of energy to require.

But the main issue here with the water is the gas generation, and this gas generation essentially, like an eruption, so this gas is going through the accordion floor and entering the molten core into the outside become a particle, and that particle can be cooled by water.

So the main difference between the siliceous concrete and limestone, limestone common sand, would be the water. It=s a decodable gas-induced eruption mechanism for coolability.

MEMBER POWERS: If you get any ablation at all thermally, you will get more with siliceous concrete than you will with calcareous concrete, simply because of the heat it takes to -- I mean, you=ve got to decompose the calcium carbonate, and that is an extremely energy-intensive process.

I=m just wondering if that=s an issue. If the application specifies, AThou shalt use calcareous concrete,@ okay, fair enough. If it=s left as is conventional to whatever is locally available, about a third of your sites have a siliceous aggregate commonly used in construction.

MR. WAGAGE: Actually, it is limited by -- because of the sump mainly because sump is -- has a larger thickness of melt to break the melt from melt -- solidifying melt, you need to produce some gas. Siliceous concrete was not producing that gas because --

MEMBER POWERS: I defy you tell -- in looking at a melt concrete interaction, you cannot tell the difference between calcareous and siliceous concrete based on gas generation because in the siliceous case you are decomposing a larger volume of concrete, and so you get steam; whereas, in the calcareous you also get steam and carbon dioxide. They look about the same.

If you sit down and calculate, the molar generation of gas is about twice as much in the case of calcareous concrete per unit of concrete evolved. But you decompose twice as much with a siliceous concrete. So you get about the same amount of gas generation. I mean, it=s not exact, but it=s roughly the same.

MR. WAGAGE: In addition to gas generation, it was that -- the calculation found that if each of the line -- line of failure, it would kind of guarantee the failure of preventing line of failure. Line is three feet below. Siliceous concrete showed line of failure if sued within certain time.

So that=s why that applicant decided to limit types of concrete to limestone and limestone common sand.

Next I will be talking about in-vessel steam explosions. A while ago, the NRC had studies and those studies found that threat to the containment by in-vessel steam explosion is minor. However, the applicant performed analysis with TEXAS-V computer code and used ABAQUS code to do structure analysis and found that it is still sticking.

Ex-vessel steam explosion is an open item because of several reasons. The applicant used a one-dimensional TEXAS computer code, one-dimensional meaning it can take radius in vertical direction but not in horizontal direction. The radial direction conditions are assumed to be the same. It=s mixed -- well mixed in the radial direction.

In that case, if one uses much larger radial distance of radial pool of water, then it can quench the melt, and there will be less energy of explosion. If someone uses much smaller area of water pool, cross-sectional area, then there will be so much steam generation, then melt will not be in the vicinity of water to produce any.

That means between there is optimal value. For in-vessel steam explosion, the applicant showed that applicant will use the optimal value during higher energy release. But for ex-vessel, you are revealing that you -- that applicant -- how the applicant is justifying that.

Other area of review is that opening of how the pressure is attenuated, this computer code calculates the pressure. And after it calculates the pressure in the given calculation area or volume, it has to attenuate the pressure and get the -- calculate the pressure on the structures to calculate the structures.

I found there was an error in the equation used. By the way, for in-vessel steam explosion, that equation was not used; only for ex-vessel. That means in-vessel steam explosion that attenuate pressure, attenuation was not used. Whatever the maximum pressure calculator was applied to the bottom of the vessel, but ex-vessel, you know, is attenuation of pressure used, and there was an error in the equation that I am discussing with the applicant.

Staff is reviewing the structural evaluation of cavity structures. We this morning also discussed that in-vessel retention system, ex-vessel external reactor vessel cooling system operation.

There wasn=t much information on the system. The question came about installation. We found that mentioned in DCD Tier 2, Section 5.3.5. That is on shipment and installation. There is one sentence I can read. AThe installation for the reactor vessel is designed to have an annular flow path suitable for the external reactor vessel cooling, the RVC, operation during a severe accident.@

I was not concerned about the external reactor vessel cooling because applicant did not take credit in the PRA, and I assume that system is not working.

When the applicant does not take credit, there is no reason for me to ask how good is the system, how -- whether it=s going to work. For example, I was not concerned about how the installation is going to work because applicant did not assume it=s going to work. That system is going to work.

However, for the ex-vessel steam explosion, there was a concern it=s the opposite. Our thinking was that, yes, the applicant did not take credit, but during a severe accident when operator sees this, operator is attempting to use this to flood the cavity to cover part of the reactor vessel, and with the intention of quenching melt inside the vessel so it will not come out.

However, there is a possibility that melt will be unfrozen, the layer on the top, that that melt layer can keep attacking the vessel and vessel may fail, giving a much larger MELCOR. But that --

MEMBER CORRADINI: So can I ask a question? This goes back to the AP1000.

MR. WAGAGE: Yes.

MEMBER CORRADINI: Right? The AP600 in fact. Exactly the same analysis was done for AP600 and 1000. But my memory is that the person who did it claimed -- and I think it was certainly accurate -- is the MELCOR rate would actually be smaller, not larger.

MR. WAGAGE: No, no.

MEMBER CORRADINI: So why would it be larger?

MR. WAGAGE: No. It=s much larger. It=s -- I think I -- I think it=s much larger because assume that it=s going to open like a clamshell and pour a much larger -- actually, AP600 did analysis for that situation and found the cavity is going to fail but containment would stay intact.

MEMBER CORRADINI: So you=re looking at the whole thing on Zippy.

MR. WAGAGE: Yes. Not whole thing like -- opening like a clamshell that --

MEMBER CORRADINI: Like a --

MR. WAGAGE: -- for significant amount of melt.

MEMBER CORRADINI: Like an upside-down tin can, like an upside-down soup can.

MR. WAGAGE: Like Pacman opening, yes. Then AP600 did structural analysis for that and found that cavity is going to fail, but they do not detail the containment. But this APR1400 has not gone that far.

MEMBER CORRADINI: But they didn=t do the analysis.

MR. WAGAGE: They did not do the analysis. They did not take credit. But we are debating whether to -- oh, then many odd questions --

MEMBER CORRADINI: So now I was going to ask you the binary question. If it went away, does it matter? And if it was there, does it matter, in terms of the overall containment failure probability?

MR. WAGAGE: Okay. Applicant did sensitivity and --

MEMBER CORRADINI: Before you make them go through this excruciating analysis, does it matter?

MR. WAGAGE: Actually, the one for which

applicant did sensitivity analysis for the -- operating the system and found that it did not matter much to the system. So they did sensitivity analysis. Even if the system works, that does not buy much.

MEMBER REMPE: If you claim it=s not there, if you really want to prove that it doesn=t hurt things, do you assume an adiabatic out-of-surface on the vessel? Because your -- what was the heat transfer coefficient they assumed on the exterior surface of the vessel to the -- you know, through the insulation into the cavity area, because with it there you are hoping that you=re going to have nice heat transfer from the ERVC, and they aren=t taking credit for it.

So what did they assume for the heat transfer coefficient off a vessel? Because if it adversely affected things, if you couldn=t get water in between the vessel and this insulation for ERVC, then you would have a very reduced heat transfer and you might actually cause the vessel to fail earlier.

So did they assume an adiabatic outer surface if they didn=t take credit for it?

MR. WAGAGE: Actually, they did not

assume the cavity being filled to cover the vessel. The ex-vessel -- external reactor vessel cooling was not considered because they did not take credit. They did not show any calculation for that one, though it did not --

MEMBER REMPE: So, basically, they assume an adiabatic, or do they have natural convection to the wall? I mean, they would have to -- the thing is, one of the things we worried about with AP600 was, you know, if it wasn=t sufficiently robust, and you mentioned in Chapter 5 apparently they=re saying they have a good flow area, so they have considered this, but, you know, if you had some sort of collapse of that flow path that it would limit the flow of water down there, and could you have a heat transfer condition on the outer surface that is --

MR. WAGAGE: Our situation was that because the applicant did not take credit, there was no reason for us to ask how good the system was. Now, with the opposite --

MEMBER REMPE: Right.

MR. WAGAGE: -- applicant does take credit. However, the operator may use the system. Then what happens? That=s what I was --

MEMBER CORRADINI: I=m trying to remember back for AP1000. But if my memory serves me, AP1000 there was an estimate by ERI. I=m looking for the gentleman; he has left.

MR. AYEGBUSI: Hossein is sitting in the back of the room. He did that work.

MEMBER CORRADINI: Hossein did it, but that isn=t the person in charge. But my memory was it=s a fractional amount that was estimated that the IVR wouldn=t work. It was something like about 17 or 24 percent. And in that time period you assume you go to film boiling, and you essentially create a hole. I don=t remember the whole hole unzipping. I remember that was the bounding calculation to get them so that it=s not a problem.

So are you asking the applicant to do some sort of bounding calculation, or just consider this and it=s up to them to figure out what to do?

MR. WAGAGE: Actually, when asked the question, applicant proposed a COL --

MEMBER CORRADINI: That=s a copout.

MR. WAGAGE: -- information item telling that it has --

MEMBER CORRADINI: Sorry.

MR. WAGAGE: -- that COL applicant will be addressing ex-vessel steam explosions if SAMDA uses this system.

MEMBER CORRADINI: Okay.

MEMBER REMPE: And then it will already be a certified design, and we don=t look at the AMGs, and so it=s -- it=s not evaluated.

MEMBER CORRADINI: Well, I=m back to -- I=m back to -- I don=t mean to be so brutal, but I=m back to, why do I care? I either care because of accident management guidelines, or I care because it affects the containment failure probability. So if it doesn=t affect the containment failure probability, then I don=t care. Does it affect the accident management guidelines?

Those are the only two reasons I would do this analysis. Otherwise, I just wouldn=t do it. It=s interesting. I find it fascinating. But I wouldn=t do it unless it affects one of those two items.

MR. WAGAGE: I think if you put probability here, then I think it would be a much more probability because it has to --

MEMBER CORRADINI: It would be what? I=m

sorry.

MR. WAGAGE: It must be small probability for the --

MEMBER CORRADINI: But I think Dr. Rempe=s question is a fair one. If you=re asking the applicant to do this, and they can already show by essentially binary, either it=s there and I get a failure of cavity but it doesn=t do anything to containment, or it=s not there and that change is -- delta is so small as to the overall containment failure probability it only comes down to the accident management guidelines is where it matters. I think that=s where Joy was going with it.

MEMBER REMPE: Well, if you can -- I think what I remember is more AP600-based, but they basically finally said there is too much uncertainty, and so they did do the bounding and say, ADoesn=t matter. If you can make them do that, that=s cool.@ But right now, it=s kind of like out there and we=re not really analyzing it.

So then I start wondering about adverse effects. You aren=t taking credit, but then you need to think about that there=s something in this cavity and containment area that -- and show that it doesn=t adversely affect things. So I would prefer that a bounding calculation show that it can=t -- you know, it doesn=t affect things, and there=s no adverse effects.

But I=m not sure how one -- I mean, we can write something about it in the letter, but I don=t know what the staff could do. Mike, is there something?

MEMBER CORRADINI: I just want to understand what Hanry is forcing them into.

MR. WAGAGE: No, I=m not -- right now that is --

MEMBER CORRADINI: Oh, you=re giving them a choice.

MR. WAGAGE: That=s the status. Actually, I proposed that COL information item and --

MEMBER CORRADINI: That=s acceptable. Okay.

MR. WAGAGE: -- we are reviewing it with the -- that=s right. We are reviewing it right now.

MEMBER CORRADINI: But I guess --

MR. WAGAGE: We don=t have a position which way to go.

MEMBER CORRADINI: But I think, to clean it up, since the AP1000 part of this open, they can just look and see what was done there, right? Okay.

MR. WAGAGE: Yes.

MEMBER REMPE: It seems like that would be -- making it a COLA item is kind of -- I don=t know, that doesn=t seem as satisfactory.

MR. WAGAGE: Thank you. I am done with --

CHAIRMAN BALLINGER: We should move on.

MR. WAGAGE: -- this part of the presentation.

CHAIRMAN BALLINGER: Maybe we shouldn=t move on.

MEMBER STETKAR: In your presentation this afternoon, you didn=t address one of the bullet items on your Slide 50, and in particular that would be containment bypass. And we=re not going to talk about it anywhere else, so I=m going to talk about it now.

In the SER, there is a section that does address containment bypass. And in the interest of brevity, let=s just say it says two types of accident scenarios that are of interest are steam generator tube rupture and interfacing system LOCA.

There is then a discussion of an exchange between the staff and the applicant regarding tube ruptures, both as an initiating event and as a consequential tube rupture. And it basically cites the fact that for consequential tube ruptures they can open up the POSRVs and depressurize the primary side. And for initiating events, they can do the standard steam generator tube rupture response.

And the final conclusion of the staff is -- and here=s where I will quote -- AGiven the design features described above,@ which is all of the tube rupture stuff I just summarized, Aand evaluated in Section 19.2.2.5 of this report, which is a discussion of interfacing system LOCA initiating events, which are consistent with SECY-90-016 recommendations, the staff concludes that the containment bypass is not a significant contributor to severe accidents for the APR1400 design.@

To me, that=s really interesting because in the DCD I learn that 49 percent of the large release frequency, just slightly less than half, is from containment bypass from tube rupture, both initiating event and induced, where most of it is actually from the consequential.

For those of you interested in whatever is going down around in that cavity, that is invisible. Twenty-seven percent comes from a late rupture with no containment sprays, 12 percent is from a containment rupture prior to core damage. That=s energy release overpressure early --

I know. I know you=re fascinated by it. I heard that you are.

And then there is 10 percent from containment leakage and, I don=t know, it gets really small after that. So my curiosity is the staff is saying, well, containment bypass is no big deal on this plant, and yet it=s half --

PARTICIPANT: More than half.

MEMBER STETKAR: No, it=s 49 percent.

PARTICIPANT: Oh, 49.

MEMBER STETKAR: Forty-nine percent. It=s actually a little less than half.

Of their large release frequency. I=m curious. This comes back to my initial question about, why do you focus on only those things that were identified back in some SECY paper 25 years ago, and not focus on what is important for this plant?

And is there anything that can be done to make the stuff that=s important less important?

They have already accounted for everything that you cite in your discussion, and yet even accounting for all of that it=s still half of the large release frequency. I=m not going to say any more. It=s on the record.

MR. WAGAGE: I=ll take that question back.

CHAIRMAN BALLINGER: Let=s continue.

MS. GRADY: Good afternoon. I=m Anne-Marie Grady, and I=m here to address the severe accident mitigation feature of equipment survivability.

CHAIRMAN BALLINGER: Is your light on?

MS. GRADY: It=s green. Does that sound better?

CHAIRMAN BALLINGER: Yes.

MS. GRADY: Okay. The objective of equipment survivability comes from SECY-93-087, and it requires mitigation features be designed to operate in the severe accident environment for which they are intended over the timespan for which they are needed.

10 CFR 50.44(c)(3), which covers combustible gas control, requires containments to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen equivalent to that of a fuel clad and coolant interaction involving 100 percent of the fuel cladding. That is an additional condition over severe accident mitigation.

Severe accident mitigation is after any severe accident. This is an additional condition put on it by the burning of hydrogen.

Staff finds -- hold on. I guess I=ve got to read that. Okay. The applicant selected accident scenarios from the most probable core damage sequences in the Level 1 PRA and from several LOCAs. The applicant then identified mitigation functions of reactor coolant system inventory, reactivity control, and containment integrity as the mitigation functions that needed to be satisfied.

The applicant in the DCD specified specific equipment which would be needed to achieve those functions, and it=s found in DCD

Table 19.2.3-4. entitled Systems and Equipment and Instrumentation Required for Equipment Survivability.

Staff agreed with the equipment that was identified in that table, and in addition requested two other items be added. One of them is the emergency containment spray backup system check valve that is sitting inside the containment, and the other one is the integrity of the containment isolation penetrations.

The containment isolates much earlier in an accident than this, but this is just to maintain integrity of the penetrations where the containment isolation valves are.

The applicant has agreed to add those two items to the existing list, and we agree with the equipment that has been identified.

The accident conditions characterized by the applicant and the environmental conditions for equipment survivability establish sufficient guidance to demonstrate compliance with 50.44(c)(3) and 10 CFR 50.34(f). The temperature profiles were confirmed in the staff confirmatory calculations, and the applicant=s AICC pressure of 110 psia bounded the staff=s confirmatory pressures.

Additionally, the applicant calculated the severe accident radiation dose of 4.4E+05 Gy or 4.4E+07 rad using a MAAP dose code. Staff did not do a confirmatory calculation on the dose, but rather compared the dose calculated by the applicant with other advanced light water reactors of similar size and design and fuel type, and found them comparable.

MEMBER POWERS: This is dose over some period of time?

MS. GRADY: Twenty-four hours.

MEMBER POWERS: Twenty-four hours.

MS. GRADY: Yes.

MEMBER POWERS: Okay.

MS. GRADY: Staff also found the containment atmospheric assessments of temperature, pressure, and radiation described in DCD Section 19.2.3.3.7 acceptable for evaluating equipment survivability.

There was, and still is, a COL information item whereby the COL applicant will then take the equipment identified, the conditions of the severe accident, and then ascertain, once they=ve specified this equipment, and once they=ve purchased it, they will do the evaluation to show that that equipment can in fact last for the time period it needs to and function in that time period. But that is a COL item.

So this is basically a partially complete evaluation. This is what staff has -- this is what the applicant has proposed. This is what we have agreed to.

MEMBER POWERS: This is -- this dose is a severe accident dose, and it is for a 24-hour period, but it=s not going to be radically different than the design basis dose because it=s dominated by the noble gases.

MS. GRADY: I=m sorry. Would you repeat that last part of the question?

MEMBER POWERS: The dose you get for severe accidents is not radically different than the design basis dose, because it=s dominated by the noble gases.

MS. GRADY: Okay.

MEMBER POWERS: And so that the equipment that must survive to deal with the design basis dose must deal with a much larger total dose, because it has to last for 30 days; is that correct?

MS. GRADY: You said 30 days. I don=t

know where that requirement would have come from. But it could be longer than 24 hours, yes.

MEMBER POWERS: Yes. Okay. So --

MS. GRADY: But it depends on what you need the equipment for and for how long you need it. All it has to do is survive and function. Yes, I agree with you.

That=s all I have to say unless somebody has a question.

MR. SCHAPEROW: I=m Jason Schaperow with the Office of New Reactors, and I would like to present to you our MELCOR independent confirmatory analysis that we did for Chapter 19.

MEMBER REMPE: Jason, could you kind of explain to me when you say Athe independent calculations you did,@ I know there was a calc notebook, which I actually looked through that ERI created, but that=s just the model description. Where does one find the calculation results documented?

MR. SCHAPEROW: Okay. I think it should be a reference in the draft SC that you received.

MEMBER REMPE: The only place I=ve seen plots with MELCOR --

MR. SCHAPEROW: And it=s in ADAMS. There=s a reference to the SC that you received; it=s in ADAMS.

MEMBER REMPE: I=d like to see a copy. The only place I=ve seen calculations are those combustible gas ones that was -- or an ERI document, but maybe I missed --

MR. SCHAPEROW: This is actually an SPRA Branch document that we produced.

MEMBER REMPE: Okay. And it=s dated 2015. And, you=re right, I missed that. I apologize. Okay.

MR. SCHAPEROW: So the objective of our confirmatory analysis was to confirm the applicant=s use of MAAP for the PRA and for the severe accident analysis, Chapter 19. Our approach was to perform independent analysis for select scenarios, so we did -- we took a sample of scenarios.

We ran the calculation for the MELCOR, and then we compared the MELCOR results with the MAAP results for these scenarios.

MEMBER REMPE: Jason, I take it back. I=m sorry to keep interrupting you, but that was a meeting that Walt showed me. It says it=s to be developed in the draft SC, the calculations. If I look at that last reference, it says this report is under development. SPRA, is that --

MR. SCHAPEROW: We finished it in November 2016. So it=s possible that you have -- the SC version you have is --

MEMBER REMPE: That must be -- what we have is outdated, because --

MR. SCHAPEROW: I=ll get it to you. I=ll get it to you.

MEMBER REMPE: Yes. I would like a copy is the bottom line, please.

MR. SCHAPEROW: Yes. Especially -- I and Sean Campbell from Research, the two of us put it together in November.

MEMBER REMPE: Okay. Yes, I --

MR. SCHAPEROW: He was on rotation. That was the first thing he did when he got there was to put the report together because we wanted to have a document of the calculations.

Okay. I just want to note up front that there is two remaining issues after this work that we did. One is the applicant has committed to assess the impact of their sensitivity calculations they did in response to our RAI on their SAMDA analysis. And the second one is that the applicant is revising their shutdown analysis with MAAP, and I=ll get into that a little more on my last slide.

Okay. Next slide, please?

Okay. We reviewed the DCD and the supporting documents that we had access to in the electronic reading room in order to select scenarios, and this graph shows the -- this chart shows the five scenarios that we analyzed with MELCOR for our confirmatory analysis.

The first four rows are at-power scenarios. These scenarios are very similar, with the exception that they have different new -- these new severe accident features are different. So for the Q03 scenario, there is no -- it=s a really plain vanilla station blackout, nothing works. And then as you go up higher on the chart, then we start adding systems that they have for severe accident mitigation.

Finally, the last row of the chart, we actually did a calculation for a shutdown accident, and this is for scenario POS 5.

Next slide, please?

So we did the calculations, and when we compared the results we found that we actually had a couple of assumptions a little bit different than what was in the MAAP calculations. The assumptions that we used for MELCOR were based on what we read in the DCD, but we learned a little more when we did the comparison and we said, AWell, why is this a little different?@ and they said, AWell, it=s because we did this.@

Some of the differences -- MELCOR had -- we had modeled the safety injection tanks in the MELCOR calculation where in the MAAP calculation they didn=t have that included.

For the second row here, we had a hot leg creep rupture included in our model, but for the one calculation that we looked at from KHNP, which was the high pressure calculation, they assumed that there was no creep rupture of the hot leg.

Similarly we assumed that there was seal leakage and failure. This is a holdover kind of from the sort of work that we did. For MAAP, they assumed no seal leakage or failure.

And the last two rows deal with timing of operator actions. From what I saw in the DCD, it looked to me like they were going to perform these operator actions opening the POSRV, and opening the three-way valves and the cavity flooding valves. I thought this would be done when the core-exit thermocouple hit 922K.

Again, when we got to comparing the results, we found out this wasn=t what was done with MAAP. For opening the POSRVs, the operator opened them. They assumed the operator opened them after the first life of the POSRVs, which was consistent with their feed and bleed procedures.

And, similarly, for opening three-way valves and the cavity flooding valves, we found that they actually assumed a delay, that they thought it would take a little while until those valves got open.

Next slide, please?

As a result of our comparisons, KHNP went back and they ran some sensitivity cases to look at these differences and assumptions. And for the cases that we looked at, for the source term categories, for example, they decided -- they concluded that the new MAAP calculations didn=t really make any difference in the PRA.

We took a look at that also, and we

extended that to look at other parts of the other source term categories beyond source term categories 11 -- 10, 11, and 16. We looked at all of the other source term categories to see if we thought the differences in the calculations would be significant.

Regarding large release frequencies, we scaled the releases to account for the sensitivity calculations, and we decided that it wouldn=t --it wouldn=t change any small release to a large release, so we figured that wouldn=t be a difference.

We also looked at the SAMDA analysis, and, again, scaling the cesium releases by the differences that we saw, we didn=t think it was going to affect the SAMDA analysis because SAMDA analyses do have typically quite large margins in them, and this one did.

Regarding quantification of the CET, the Case Q03 was used in the containment event tree to look at containment pressures. And while they did get a little different containment pressure when they did sensitivity calculations with Case Q03 with MAAP, it didn=t make that big a difference, especially in comparison with the ultimate containment failure pressure, which is 162.7 pounds gauge.

Next slide, please?

MEMBER POWERS: Jason, that containment failure fragility number there, that=s a membrane failure, right?

MR. SCHAPEROW: It=s listed in the document as the median ultimate containment failure pressure. I don=t know what it=s based on, though.

MEMBER POWERS: But it=s the --

MR. SCHAPEROW: The structural folks would know.

MEMBER POWERS: -- it=s the rupture of the steel liner as a membrane.

MEMBER STETKAR: It=s actually around the equipment hatch is where they fail it. So it=s not --

MR. SCHAPEROW: It=s used within the PRA as part of their determination of the likelihood of overpressure failures, and they use these MAAP calculations to decide what the likely -- they enter this table of failure pressure to figure out what -- the likelihood of each of these calculations resulting in containment failure.

MR. ROCHE-RIVERA: This is Robert Roche, Structural Engineering Branch in NRO, and I agree with Jason=s last point. I think it did contain not only liner failure limits, but it looked at other aspects such as rebar, tendon, failure limits for the rebar, tendon, and even like leakage failure criteria to come up with the combined median ultimate pressure capacity.

MR. SCHAPEROW: We also used MELCOR to simulate a midloop accident. The deck that we started with was an at-power deck, but we made the changes needed to make it mimic a shutdown accident. In particular, we wanted to do a calculation for a plant operating state 5, which was a big part of the core damage frequency for this design. So we reduced the decay heat.

We assumed the accident happened during POS 5, which was after -- a bit after shutdown. We added nozzle dams by blocking flow paths. We added an open manway on the top of the steam generator. We took the safety injection tanks away by isolating them, and then we changed the RCS pressure temperature and water level to mimic the start of an accident for midloop.

We did comparisons. We looked at the MAAP results. We compared them with the MELCOR results. We asked a lot -- we asked a number of questions to the applicant, and the applicant has decided to revise their MAAP model for this -- for the POS 5 calculations. And last I saw, they had actually redone several of these calculations and they were documenting them and folding them back into the PRA.

They also did find a code bug as part of the -- as a result of the questioning, and they told us about that as well.

CHAIRMAN BALLINGER: Continue. Next?

MR. SCHAPEROW: Finished.

CHAIRMAN BALLINGER: Thank you.

MS. NEUHAUSEN: Good afternoon again. My name is Alissa Neuhausen, and I=m a technical reviewer in the Structural Engineering Branch. I was responsible for the review of the containment performance capability, along with Robert Roche.

Next?

The staff review ensures that the applicant meets the Commission=s deterministic containment performance goal as described in SECY-90-016 and 93-087. The staff focused on information provided in DCD Section 19.2.4, containment performance capability.

The staff also reviewed the applicant=s finite element analysis of containment subjected to severe accident pressure and temperature loadings. The staff followed guidance provided in Regulatory Guide 1.216, Regulatory Position 3, for severe accidents.

The staff confirmed that the ASME factory load category for concrete containments are met for severe accident loading. The staff has completed the review of the deterministic containment performance goal.

Next, please? Thanks.

The deterministic containment performance goal establishes that the containment used to maintain its role is a reliable leak-tight barrier for approximately 24 hours following the onset of core damage and continue to provide a barrier against the uncontrolled release of fission products after 24 hours.

The applicant=s approach was to select a conservative severe accident load. The applicant demonstrated that the most significant pressure loading history is generated from a large loss of coolant accident, station blackout, and total loss of feedwater events are bounded by the selected severe accident load.

The applicant demonstrated that for the selected severe accident load the strains in the liner plate do not reach the allowable limit strain values as defined by ASME Code Section 3, Division 2, subarticle CC-3720, factor load category.

If there are no questions, that=s all I have.

MEMBER CORRADINI: So there=s a stylized pressure temperature history that bounds those three?

MS. NEUHAUSEN: The pressure bounds those three.

MEMBER CORRADINI: And along with some temperature.

MS. NEUHAUSEN: Along with some temperature.

MEMBER CORRADINI: Okay.

CHAIRMAN BALLINGER: Thank you.

MR. WAGAGE: My name is Hanry Wagage. I=ll conclude with phase 2 staff findings that due to the remaining issues, the staff is unable to make final conclusions on the severe accident evaluation of the APR1400 design.

MEMBER CORRADINI: Pending open items.

CHAIRMAN BALLINGER: Thank you.

MR. PHAN: Ladies and gentlemen, that=s the end of our presentation on PRA and severe accident evaluation, understanding that there are outstanding questions that the staff will provide you with additional evaluations and information. But for now, again, we thank you for all of your comments and your advice, and the staff will incorporate those in the next phase of our review.

And with that, if you have any additional questions, please raise them at this point.

CHAIRMAN BALLINGER: Any questions? Now, should we maintain the line open? I=m not sure anybody is on it.

So any comments from anybody in the room? Anybody want to -- hearing none, the line is open. Is there anybody out there? If there is, can you identify yourself or say that you=re there?

MEMBER BLEY: Just ask for comments.

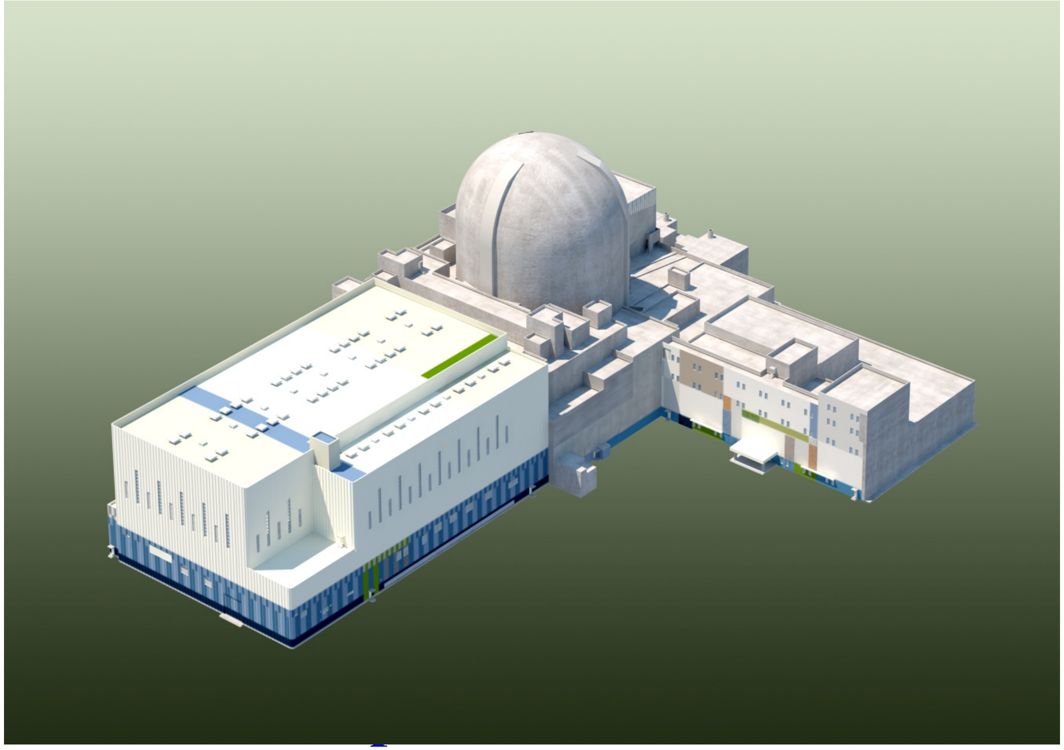
CHAIRMAN BALLINGER: Any comments from people on the line? Hearing none, thank you very much.

I think that -- I haven=t heard anything about needing a closed session. We had on the schedule something here and a note -- I don=t think we have it, so we don=t need it.

That being the case, we have gone from likely to be very late to being five minutes early. So, in that case, we are adjourned, released, recessed.

(Whereupon, the above-entitled matter went off the record at 5:55 p.m.)

APR1400 DCA
Chapter 19.1: Probabilistic Risk Assessment



KEPCO/KHNP

April 19, 2017

Overview of Chapter 19 (1/2)

- **Section Overview**

Section	Title	Presenter
19.0	**Probabilistic Risk Assessment and Severe Accident Evaluation**	Young In
19.1	**Probabilistic Risk Assessment**	
19.1.1	Uses and Applications of the PRA	
19.1.2	Quality of PRA	
19.1.3	Special Design/Operational Features	
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	Tae-Hee Hwang
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
19.1.5.4	Other External Events Risk Evaluation	
19.1.6	Safety Insights from the PRA for Other Modes of Operation	Jaegab Kim
19.1.6.1	Level 1 Internal Events PRA for Low Power and Shutdown Operations	
19.1.6.2	Level 2 Internal Events PRA for Low Power and Shutdown Operations	
19.1.6.3	Internal Fire PRA for Low Power and Shutdown Operations	
19.1.6.4	Internal Flooding PRA for Low Power and Shutdown Operations	
19.1.7	PRA-Related Input to Other Programs and Processes	Young In
19.1.8	Conclusions and Findings	

Overview of Chapter 19 (2/2)

- **Section Overview**

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19.2	**Severe Accident Evaluation**	Byungjo Kim
19.2.1	Introduction	
19.2.2	Severe Accident Prevention	
19.2.3	Severe Accident Mitigation	
19.2.4	Containment Performance Capability	
19.2.5	Accident Management	
19.2.6	Consideration of Potential Design Improvement under 10 CFR 50.34(f)	
19.3	**Beyond Design Basis External Event**	Chan-Eok Park
19.3.1	Introduction	
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19.4	**Loss of Large Area**	Gary Hayner
19.4.1	Introduction and Background	
19.4.2	Scope of the Evaluation	
19.4.3	Conclusions	
19.5	**Aircraft Impact Assessment**	Randy James
19.5.1	Introduction and Background	
19.5.2	Scope of the Assessment	
19.5.3	Assessment Methodology	
19.5.4	Conclusions	

Overview of Chapter 19.1

● Section Overview

19.1 APR1400 PRA Scope

Operation Mode		Level 1	Level 2
At-power	Internal Events	O	O
	Internal Fire	O	
	Internal Flooding	O	
	Seismic*	O	-
Low Power and Shutdown	Internal Events	O	O
	Internal Fire	O	O
	Internal Flooding	O	△

* PRA-based SMA, △ Bounding approach

19.1 PRA Methodology & Tools

- Small Event Tree & Large Fault Tree Approach
  - Linked Fault Tree
- Computer Tools
  - KEPCO E&C SAREX™
  - KAERI FTREX™
  - EPRI CAFTA*
  - EPRI HRA Calculator 5.1 (2017 PRA Update)
  - MAAP 4.0.8
  - RELAP5/Mod3
  - MACCS2

* For LPSD Internal Fire and Internal Flooding Level 1, and LPSD Level 2. The PRA updates planned in 2017 will be using CAFTA.

19.1.7 Risk Applications

- Current Applications (DC Phase)
  - Reliability Assurance Program (RAP)
  - Severe Accident Management Design Alternative (SAMDA)
  - Environmental Report
- Future Applications (COL Phase)
  - Reactor Oversight Program
  - MSPI, SDP, etc.
  - Maintenance Rule

19.1.7 Design Improvements from Risk Insights

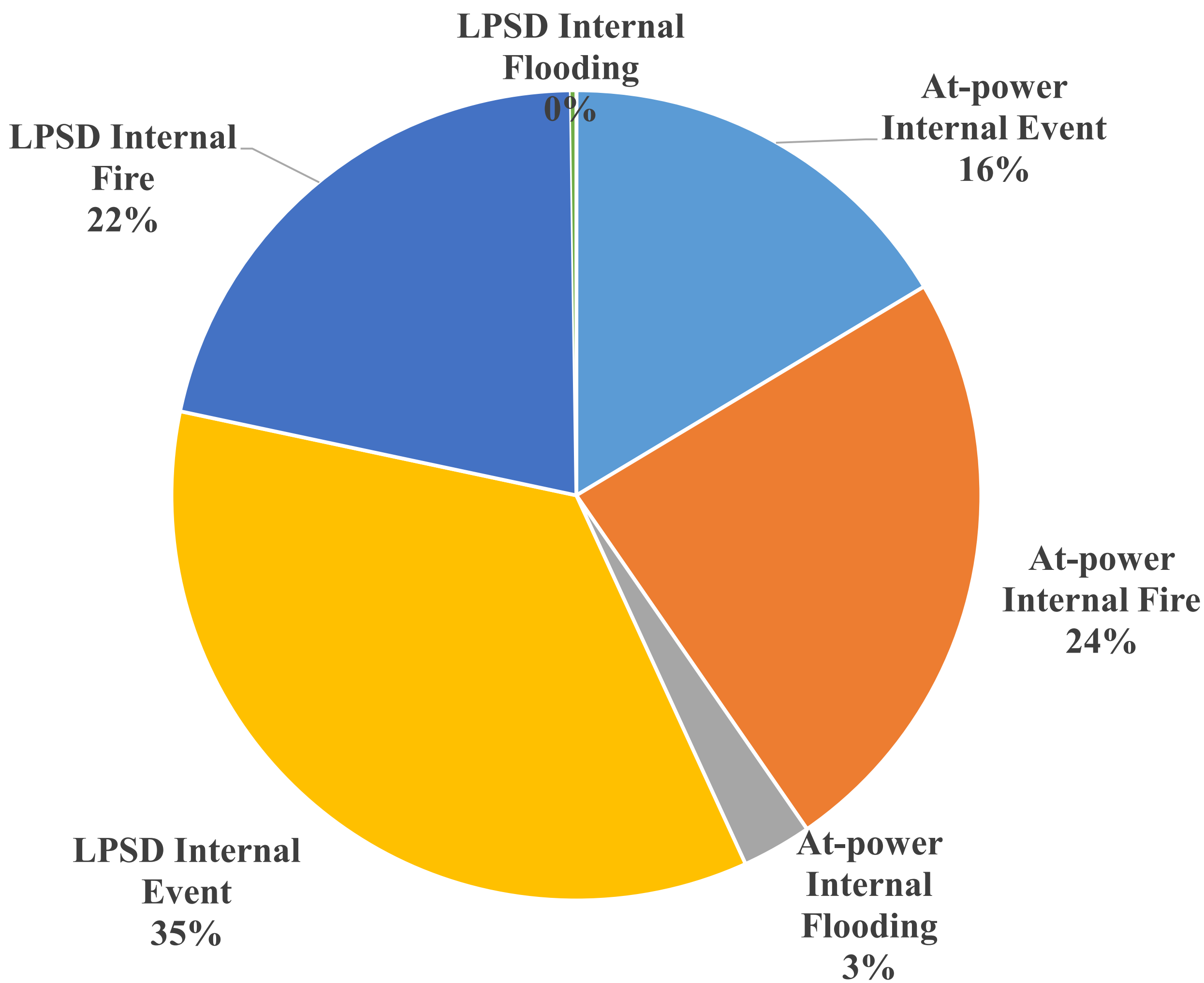
- EDG: two (2) EDGs to four (4) EDGs
- AAC: from Diesel Generator (DG) to gas turbine generator (GTG)
- 125V DC Batteries: Increased capacities
- Technical Specifications 3.6.7: equipment hatch closure in Mode 5
- Cables to be protected: cables for 75 components in 59 fire compartments

19.1.8 Overall Results

Operation Modes	Hazards	Level 1 (per yr)	Level 2 (per yr)
At-Power	Internal Events	1.3E-06	1.1E-07
	Internal Fire	1.9E-06	1.7E-07
	Internal Flooding	2.2E-07	1.7E-08
LPSD	Internal Events	2.6E-06	1.2E-07
	Internal Fire	1.7E-06	1.3E-07
	Internal Flooding	1.8E-08	---
Total		7.9E-06	5.5E-07

Note: The CDF and LRF values are point-estimates.

19.1.8 Overall CDF Profile



19.1.4.1 Internal Events Level 1

19.1.4.1 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

19.1.4.1 Initiating Event Analysis

- LOCAs
  - Large, Medium, Small, ISLOCA and RVR
  - SGTR
- Transients
  - General Transient
  - Loss of Main Feedwater
  - Loss of Condenser Vacuum
  - Large Secondary Side Breaks (MS and MFW)
  - Loss of Support Systems (IA, DC, CC, SX)
- Loss of Offsite Power Events (Plant, SWYD, Grid and Weather Related)
- Induced Initiators
  - Not true initiators, but rather unique events induced post-initiator by certain plant response failures
    - ATWS
    - Grid Disturbance LOOP/SBO
    - Station Blackout (SBO)
    - Stuck Open POSRV LOCA

19.1.4.1 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

19.1.4.1 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 FSAR Chapter 15 for consistency**

19.1.4.1 Internal Events at Full Power

- **Key PRA Assumptions**
  - The Internal Fire, Internal Flooding and Seismic Assessment
    - Based on the APR1400 design information
    - The SKN 3&4 design information is used, if the design information is not available
  - Digital I&C system
    - Uses the hardware model from the SKN 3&4 design
  - MLOCA
    - Assumed not to require Hot Leg Injection to prevent boron precipitation
  - RCP Seal LOCA Probability
    - Modeled based on the engineering judgment
  - GSI-191
    - Sump plugging modeled, but chemicals effect is not modeled since there no fibrous materials in the containment

19.1.4.1 Systems modeled in PRA

Front Line Systems

- Auxiliary Feedwater (AF)
- Containment Spray (CS)
- Shutdown Cooling (SC)
- Safety Injection (SI)
- Chemical & Volume Control (CV)
- Main Feedwater (FW)
- Main Steam (MS)
- Safety Depress. and Vent (SDVS)
  - Reactor Coolant (RC) & Gas Vent (RG)
- Reactor Protection (RP)

Supporting Systems

- Component Cooling Water (CC)
- Essential Service Water (SX)
- Essential Chilled Water (WO)
- Electrical Systems (AC, DC, EDG, AAC)
- Heating, Ventilation and Air Conditioning (HVAC)
- Instrumentation Air (IA)
- T/G Bldg Open Cooling Water (WH)
- T/G Bldg Closed Cooling Water (WT)
- Engineered Safety Feature Actuation (EF)

19.1.4.1 Data and CCF Analysis

- **Data Source used in internal level 1 full power PRA**
  - **Component Unreliability Data**
    - **2010 update for NUREG/CR-6928**
    - **Vendor specific data (e.g., Sempell Co.) for POSRV**
    - **Alternate industry sources (e.g., IEEE STD-500)**
  - **Component Unavailability Data**
    - **2010 update for NUREG/CR-6928**
    - **NUREG/CR-5500, Vol. 2, 3, 10, and 11**
    - **Engineering judgment**
  - **Common Cause Failure Data**
    - **2010 Update for NUREG/CR-5497**
  - **Special Events Data**
    - **NUREG/CR-6890 for LOOP non-recovery probabilities**

19.1.4.1 Human Reliability Analysis

- 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

19.1.4.1 Human Reliability Analysis

- **Methodology**
  - Re-quantify model with HRA valves set to a value near 1.0
    - This is done to ensure that all risk significant HRA combination are addressed.
    - The combination of HRA probabilities when not set to a high value can be truncated and therefore not addressed by the dependency analysis
  - Analyze the HRA combinations for dependency
    - The HRA events are analyzed the same procedure usage
    - The HRA events are analyzed to ascertain if the failure of one events will fail the other HRA events in the combination.
    - The HRA events are analyzed to ascertain the affect of one event on the operator and whether the other events will be attempted.
  - Dependency Level Evaluation
    - Dependency level is determined by dependency level decision tree in NUREG-1921, "EPRI/NRC-RES, Fire HRA Guidelines"
    - Decision Tree Branches : Intervening Success / Crew / Cognitive / Cue Demand / Manpower / Location / Sequential Timing / Stress
  - The EPRI HRA Calculator provides the tools to perform the dependency analysis (being used for 2017 PRA update)

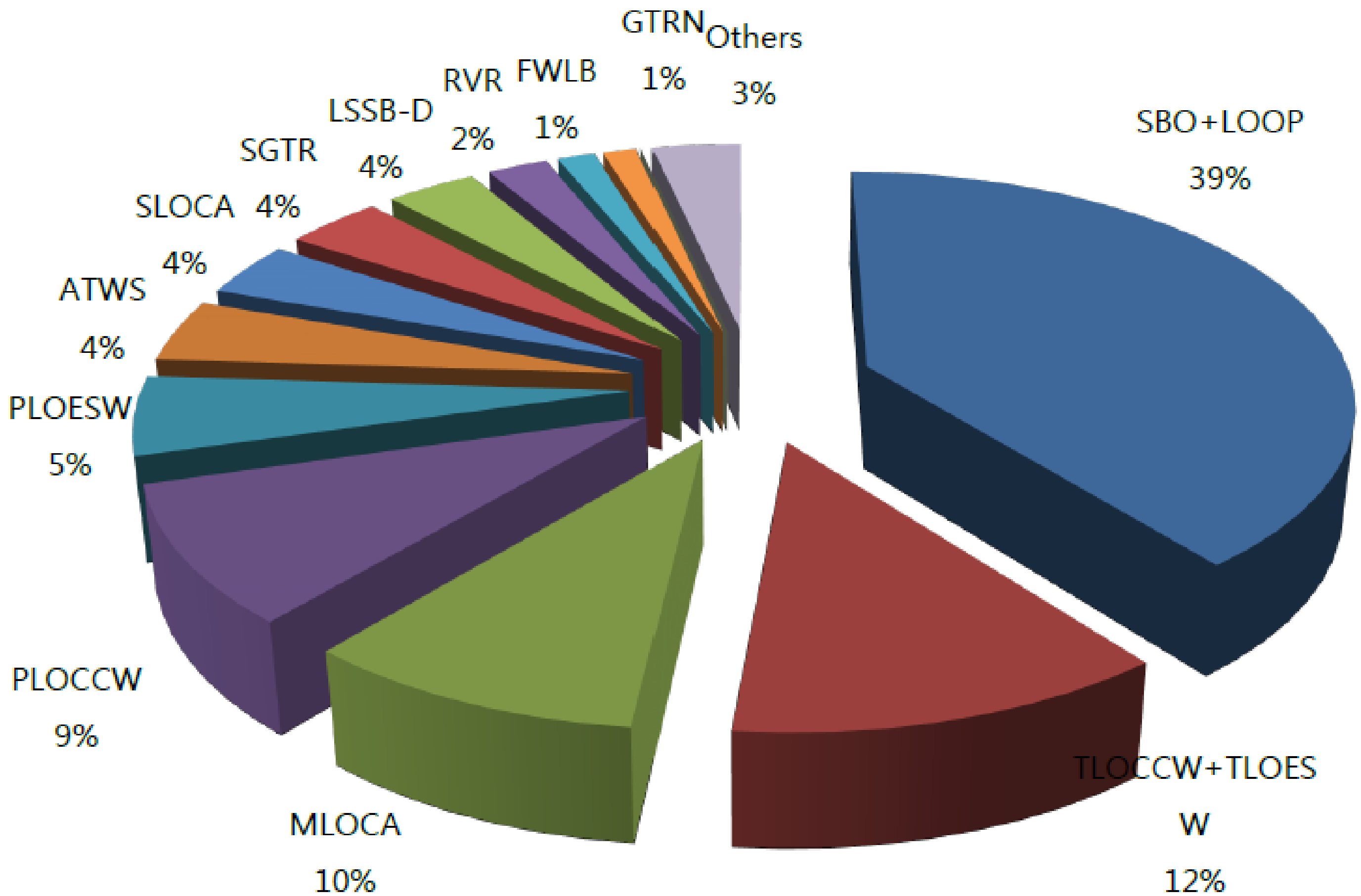
19.1.4.1 Quantification

- **Quantification Process**
  - Quantification performed using SAREX and CAFTA
    - 10^{-13} is applied as cutoff value for CDF quantification
  - Use of flag events
    - To consider various IE conditions, both house event and double initiators are used together
  - Use of delete-term logic in the model
    - To remove unrealistic MCS combinations, delete-term logic is used in quantification process
  - Logical loop treatment
    - Circular logic in the supporting system were broken at DC batteries.

19.1.4.1 Quantification

- **Quantification Process**
  - **Cutset recovery process using recovery file**
    - **HRA dependency rules, double counted events, transferred initiators with tag events are replaced with meaningful event names**

19.1.4.1 Internal Events CDF by Initiating Events



19.1.4.1 Uncertainty Analysis

- Parametric uncertainty
  - State of Knowledge Correlation (SOKC) will be addressed.
  - Selected sensitivity cases to be re-performed (e.g., HEPs and CCF factors)
- Modeling uncertainty
  - Design-specific sources of uncertainty
    - Based on key assumptions
  - Generic sources of uncertainty
    - NUREG-1855, EPRI 1009652
  - Uncertainty characterization
- Selected sensitivity analyses performed

19.1.4.1 Summary and Insights

- The CDF from internal events for at-power operation is low 10^{-6} per reactor year.
- The CDF contribution from LOOP and SBO is dominant.
- For final PRA model, 2017 update is in progress.

19.1.4.2 Internal Events Level 2

19.1.4.2 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 for analyzing severe accident progression and source term release
  - SAREX 1.3 for developing Level 2 PRA model

19.1.4.2 Severe Accident Mitigation Features (1/3)

- **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

19.1.4.2 Severe Accident Mitigation Features (2/3)

- **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**

19.1.4.2 Severe Accident Mitigation Features (3/3)

- **Pilot-Operated Safety Relief Valves (POSRVs)**
  - **Provides a means to rapidly depressurize the primary system to about 250 psia to prevent DCH and induced SGTR following severe accidents**
  - **Three-way valves located in the POSRV discharge path can be used to redirect the release point of hydrogen from IRWST to the containment atmosphere via SG compartment**
- **ECSBS (Emergency Containment Spray Backup System)**
  - **An alternate means of providing containment spray water after 24 hours following severe accidents**
  - **Deliver water from external water source to the ECSBS containment spray header**
  - **Use mobile pumping device independent of normal and emergency AC power sources**

19.1.4.2 Plant Damage States (PDSs)

- Level 1 event tree sequences are extended to be additionally questioned important for Level 2 PRA. For example,
  - Containment Isolation?
  - 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 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.

19.1.4.2 Containment Structural Analysis

- **An Ultimate Pressure Capacity (UPC) calculation approximates the realistic failure pressure of the containment.**

- **Two containment failure modes (i.e., Rupture and Leak) are determined based on NUREG-1150 and NUREG/CR-6906.**

19.1.4.2 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)
  - Steam Over-pressurization
  - Molten core-concrete interaction, including Basemat melt-through

19.1.4.2 Generic Data Used in Level 2

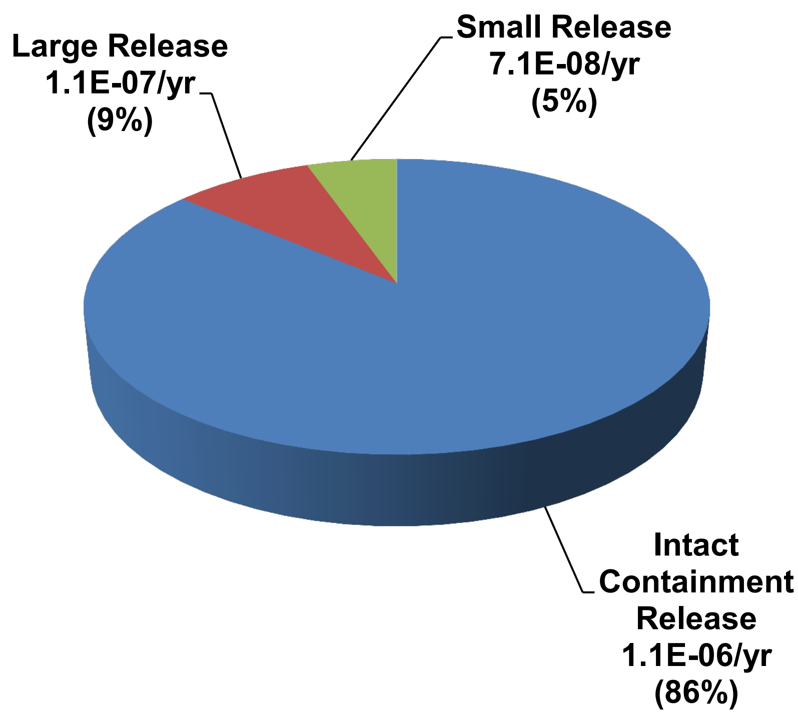
- NUREG-1570 for ISGTR, including the conditional probabilities of ISGTR developed for current generation plants.
- NUREG/CR-6475 and NUREG/CR-6109 for induced hot leg rupture.
- NUREG-1150 and NUREG/CR-4551 were consulted for various phenomena (in-vessel recovery, rocket mode failure, steam explosion, etc.)

19.1.4.2 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 before effective evacuation of the surrounding public after the general emergency declaration.**

19.1.4.2 Level 2 Results for internal events

- Intact containment = 86 % of CDF
- Containment failure (including large and small release) = 14 % of CDF
- Large Releases = 9 % of CDF



19.1.4.2 Level 2 PRA-based Insights

Level 2 Insights (for at-power internal events)
Containment Performance for APR1400 - Containment failure frequency : 1.8E-07/yr (conditional probability : 14 %) - Large Release Frequency : 1.1E-07/yr (conditional probability : 9 %)
Dominant Contributors for CFF/LRF ○ 1st dominant contributor : Containment Bypass (6 % of CDF) - SGTR prior to core damage : 5 % of CDF - Severe accident-induced SGTR : 1 % of CDF - Rapid depressurization is effective to prevent the severe accident-induced SGTR. - SGTR sequences with wet SGs do not result in a large release due to pool scrubbing ○ 2nd dominant contributor : Late Containment Failure (5 % of CDF) - ECSBS is effective to prevent containment failure due to steam overpressure - PARs and flooded cavity (by CFS) are effective to prevent the build-up of hydrogen to the high concentration inside the containment

19.1.5.1 Seismic Assessment

19.1.5.1 The Results of Fragility Analysis for PRA-Based SMA

1. Methodology

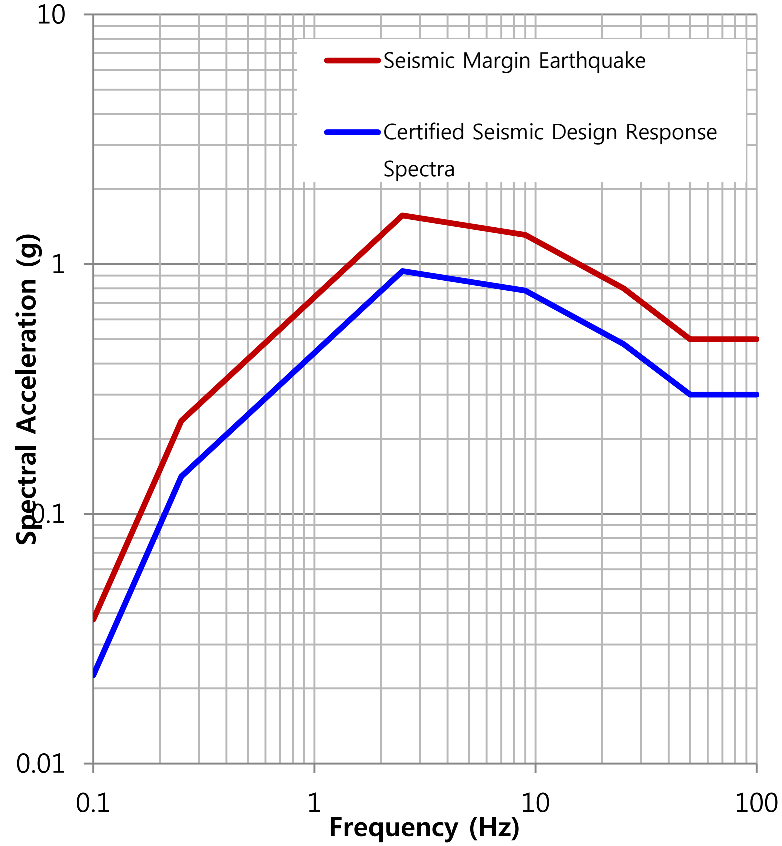
- Staff Review Memorandum to SECY-93-087
- DC/ISG-020 provides guidance for implementation process for performing PRA-based SMA
- Structures, systems, and components (SSCs) can be evaluated by either Conservative Deterministic Failure Margin (CDFM) method or Separation of Variables (SOV) method

19.1.5.1 The Results of Fragility Analysis for PRA-Based SMA

2. 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



19.1.5.1 The Results of Fragility Analysis for PRA-Based SMA

3. APR1400 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

19.1.5.1 The Results of Fragility Analysis for PRA-Based SMA

4. 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

19.1.5.1 The Results of Fragility Analysis for PRA-Based SMA

4. 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 is equal to or greater than Seismic Margin Earthquake, i.e., 1.67 times CSDRS

19.1.5.1 The Results of Fragility Analysis for PRA-Based SMA

5. 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

19.1.5.1 The Results of Fragility Analysis for PRA-Based SMA

6. 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

19.1.5.1 The Results of Fragility Analysis for PRA-Based SMA

7. HCLPF capacities of other SEL components

- Related to RAI Q. 433-8363, 19-73, a)

- ESWIS, CCW Hx Building, and BOP components
  - Detailed design information is not available in DC phase
  - Assigned to COL items (COL 19.1(7)) and assumed to have 0.5g HCLPF

19.1.5.1 The Results of Fragility Analysis for PRA-Based SMA

8. Seismic Equipment List

- Define Seismic Initiating and Consequential Events
  - Direct to core damage scenarios such as building collapse
  - Loss of all Instrumentation and Control
  - Station Blackout (SBO)
  - LOCAs
  - Anticipated transient without SCRAM (ATWS)
  - Loss of offsite power (LOOP)
- 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

19.1.5.1 The Results of Fragility Analysis for PRA-Based SMA

9. Seismic Logic Model

- **Seismic Event Trees**
  - Direct Core Damage
  - Loss of I&C
  - ATWS
  - LLOCA
  - MLOCA
  - SLOCA
  - LOOP
- **Inserted Seismic Failures Into Fault Trees**
- **Plant-Level HCLPF**
  - Solved Seismic Event Tree Models
  - Min-Max Method
  - HCLPF = 0.5g
    - Generic Failure of SSCs
    - Compound Building Collapse
    - Turbine Building Collapse

19.1.5.1 The Results of Fragility Analysis for PRA-Based SMA

10. Conclusion

- ❖ Major APR1400 SSCs were evaluated by following ISG-020
  - ➢ Used APR1400 design-specific seismic demands and design data
  - ➢ CDFM method in EPRI NP-6041-SL Rev. 1 was adapted for HCLPF capacity evaluation
- ❖ HCLPF capacities of the SSCs are greater than 1.67 times CSDRS

19.1.5.2 Internal Fire PRA

19.1.5.2 Internal Fire PRA – Task 1

- **390 Physical Analysis Units (PAUs) identified**
  - PAUs are based on Fire Areas with rated barriers
    - Majority are 3 hr rated barriers, but some are 1 or 2 hr barriers
    - Separation may only be credited for yard transformers
      - DCD 8.2.1.4 50 ft separation OR 3 hour barrier
  - ~279 PAUs in Auxiliary Building – highly compartmentalized
  - PAUs are mainly single rooms, but may contain 2 or more rooms
    - No credit for PAU internal barriers because walkdowns cannot be performed to determine whether the internal barriers would "substantially contain the adverse effects of fires"
  - 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

19.1.5.2 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
  - Cable database is based on reference plant (SKN 3&4)
    - Added assumed routing for 2 new EDGs, AAC, ESW and CCW HX Buildings, and Offsite Source Permissive (OSP) cables
    - Number and type of penetrations (for MCA) are directly from reference plant
  - Cables for new equipment were routed based on reference plant data, and incorporated into the At-Power and LPSD fire risk models (FIRMs)

19.1.5.2 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

19.1.5.2 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

19.1.5.2 Internal Fire PRA – Task 5

- At-power and LPSD FIRMs both based directly on their respective internal events models
  - In some cases surrogate events were used for spurious operations if the impact is the same (e.g., failures to close (FTC) basic event may be used as surrogate for spurious operation (SO)).
  - Basic events added for non-modeled spurious operations
- 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

19.1.5.2 Internal Fire PRA – Task 5

- LPSD FIRM screens POS 7, 8, 9 (same basis as LPSD internal events model)
- Unscreened LPSD PAU fires assumed to result in either CC, JL, KV, LP, S2 or SL based on scenario equipment damage AND the POS
  - Loss of support system on operating train dependent upon POS
    - Assumed outage schedule is basis for operating trains and T/M schedule
      - Due to assumed outage schedule, PAU initiator may be different for each POS, e.g., "A" SC train assumed operating during POS 3A - 6, "B" train assumed operating during 10 - 13, so a PAU only damaging the "B" SC train results in S2 event only during POS 10 - 13, no initiator during POS 3A - 6
  - All fire induced failures are assumed non-recoverable including OSP

19.1.5.2 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
  - Not currently updated to NUREG-2169
- LPSD Ignition Frequencies from NUREG/CR-7114
- Transient Influencing Factors for LPSD based on At-Power TIFs with adjustments based on Engineering Judgement
  - e.g., All Containment Building TIFs increased from low to high (1 to 10) for all LPSD POS due to increased work in containment
  - No effort was made to do a POS by POS adjustment

19.1.5.2 Internal Fire PRA – Task 12

- **Initial HFE HEPs estimated using NUREG-1921 Screening Analysis**
- **Top 10 HEPs (ranked by F-V) were re-evaluated using detailed HRA methodologies**
- **Fire PRA HRA used same level of dependency among dependent HFEs as at-power models**
- **Current PRA update is re-evaluating all HFEs using detailed HRA methodologies, and re-evaluating the dependency among fire HFEs**

19.1.5.2 Internal Fire PRA – Tasks 7, 8 & 11

- 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

19.1.5.2 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

19.1.5.2 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)

19.1.6.3 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

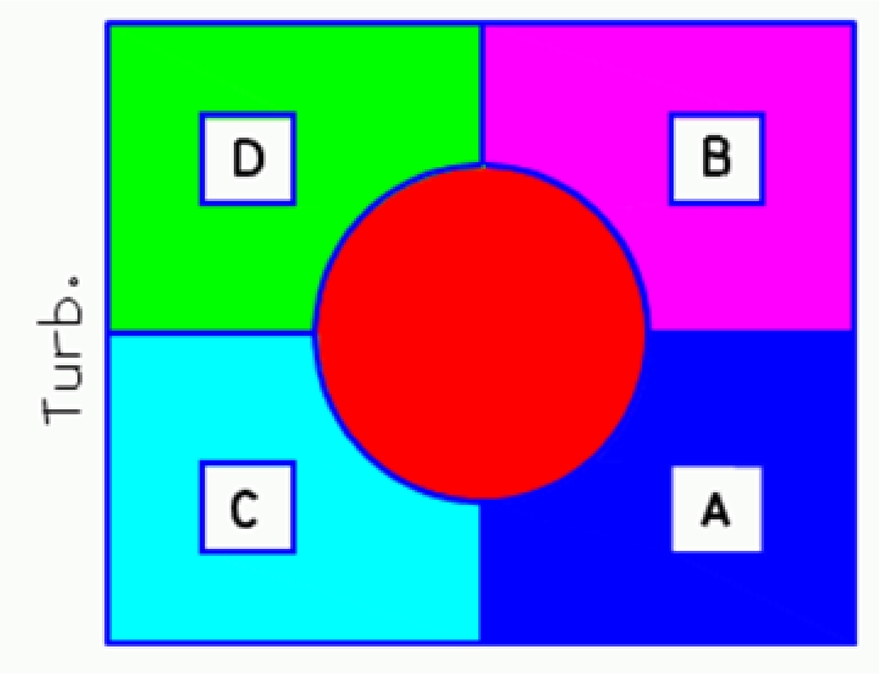
19.1.5.3 Internal Flooding PRA

19.1.5.3 Internal Flooding PRA

- **Guidance**
  - **RG 1.200 Revision 2**
  - **ASME/ANS RA-Sa-2009, “PRA Standards”**
  - **SRP 19.1**
  - **EPRI 1021086, “Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments” Revision 2**

19.1.5.3 APR1400 Characteristics

- **Auxiliary Building**
  - Quadrant Design
  - Emergency overflow lines limit accumulation on upper elevations.
  - Limited flood sources



19.1.5.3 APR1400 Characteristics

- Turbine Building
- EDG Building (A/B)
- Compound Building
- CCW Heat Exchanger Building
- ESW Building

19.1.5.3 APR1400 Internal Flooding Analysis

- **Flood-induced initiating event**
  - Uncontrolled release of fluid that also fails PRA-related equipment.
  - Considers failures that result in reactor trip or require 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

19.1.5.3 APR1400 Internal Flooding Analysis

- 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 fail first barrier and actuates fire protection

19.1.5.3 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

19.1.5.4 Other External Events

19.1.5.4 Events Considered

- **Most External Events Identified as COL Items**
  - Reference DCD Table 2.0-1
- **Transportation Accidents**
  - COL Confirm
  - RG 1.76, 1.91, SRP 3.5.1.6
- **Turbine Missiles**
  - Favorable orientation
  - DCD 3.5.1.3 – 2.1E-09 per year with 12-year inspection interval
- **Events Analyzed**
  - High winds (including tornadoes)
  - External floods

19.1.5.4 COL Item

- **Site-Specific Risk Assessment**
  - Aircraft crash
  - External flooding
  - Extreme winds and tornadoes
  - Industrial or military facility
  - Lightning
  - Pipeline accident
  - Release of onsite chemicals
  - River diversion/flooding
  - Toxic gas
  - Transportation accidents
  - Storm Surge
- **Confirm No Outliers**
  - Avalanche
  - Biological events
  - Coastal erosion
  - Dam failure
  - Drought
  - Forest fire
  - High summer temperature
  - Hurricane
  - Landslide
  - Low lake/river water level
  - Low winter temperature
  - Sandstorm
  - Tsunami
  - Volcanic activity

19.1.6 Low Power and Shutdown

19.1.6 Key Documents

- Regulatory
  - Trial ANS standard for shutdown PRA quality (March 2015)
  - RG 1.200 Revision 2
  - SRP 19.1
  - NUREG/CR-6144
- Industry
  - Approved submittal for AP1000
  - Submittals for EPR and US-APWR
- Support
  - EPRI shutdown initiating event data (TR-1003113)

19.1.6 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**

19.1.6 Analysis philosophy of POSs

Make use of prior experience with current and next-generation analyses such as US-APWR and EPR : POSs Definitions from Various Sources

NUREG/CR-6144 Section 3.5 (June 1994)
(1) Low Power Operation & Reactor Shutdown
(2) Cooldown with Steam Generators to 345°F
(3) Cooldown with Residual Heat Removal to 200°F
(4) Cooldown to Ambient Temperatures (using RHR)
(5) Draining the RCS to Mid-loop
(6) Mid-loop Operation
(7) Fill for Refueling
(8) Refueling
(9) Draining the RCS to Mid-loop After Refueling
(10) Mid-loop Operations After Refueling
(11) Refill RCS Completely (After Mid-loop Operation)
(12) RCS Heatup Solid and Draw Bubble
(13) RCS Heatup to 350°F
(14) Startup with Steam Generators
(15) Reactor Startup and Low Power Operation

19.1.6 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 dep ressurized; <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 vacuu m 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 manwa y 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 S ystem 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.

19.1.6 Initiating Event Selection

APR1400 uses a consistent set when compared to other current and next-generation PWRs

Event	Initiating Event Description	# of Events	# of Demands	Time (yrs)	Point Estimate	degrees of freedom	Upper Confidence Limit	Lower Confidence Limit	Error Factor
		n	D	T	PE	f	UCL	LCL	EF
S1	Recoverable Loss of Shutdown Cooling System (*)	9		66.9	1.4E-01	19	2.25E-01	7.56E-02	2
S2	Unrecoverable Loss of Shutdown Cooling System	1		66.9	2.2E-02	3	5.84E-02	2.63E-03	5
S0	Over-drainage During Reduced Inventory Operation	2	874		2.9E-03	5	6.33E-03	6.55E-04	3
SL	Operation	1		5.1	2.9E-01	3	7.66E-01	3.45E-02	5
LL	Large LOCA	Generic NRC data used							
ML	Medium LOCA	Generic NRC data used							
	Small LOCA above reduced inventory	2		72	3.5E-02	5	7.69E-02	7.95E-03	3
SL	Small LOCA at reduced inventory	11		72	1.6E-01	23	2.44E-01	9.09E-02	2
	Small LOCA in transition modes	Generic NRC data used							
JL	Unrecoverable LOCA (CVCS Letdown Line)	NUREG/CR-6144 estimate used							
PL	POSRV Fails to Reclose	Generic NRC data used							
RL	LTOP Relief Valve Fails to Reclose	Generic NRC data used							
LP	Loss of Offsite Power	Generic NRC data used							
LX	Station Blackout	Transfer from LOSP fault tree							
CC	Partial Loss of Component Cooling	At-power fault tree used							
TC	Total Loss of Component Cooling	At-power fault tree used							
ES	Partial Loss of Essential Service Water	At-power fault tree used							
TS	Total Loss of Essential Service Water	At-power fault tree used							
KV	Loss of 4 kV Emergency Bus (SCS Power Supply)	2		72	3.5E-02	5	7.69E-02	7.95E-03	3
DC	Loss of 125 VDC Bus	At-power fault tree used							
SG	Steam Generator Tube Rupture	Generic NRC data used							

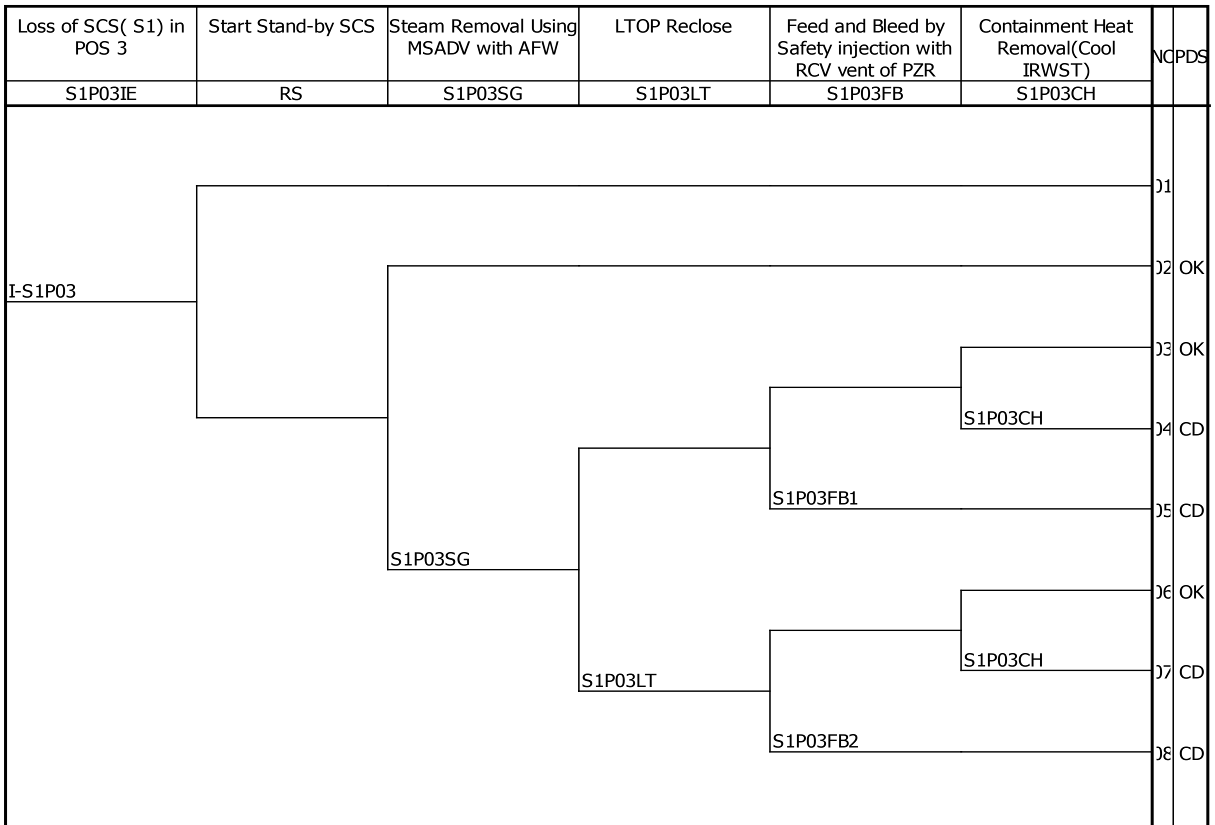
IE Point Estimates from TR-1003113 Data Since 1994

19.1.6 Initiating Event Point Estimates

- An appropriate combination of generic and design-specific initiating event frequencies used
- Design-specific initiating events used fault tree development for the point-estimates

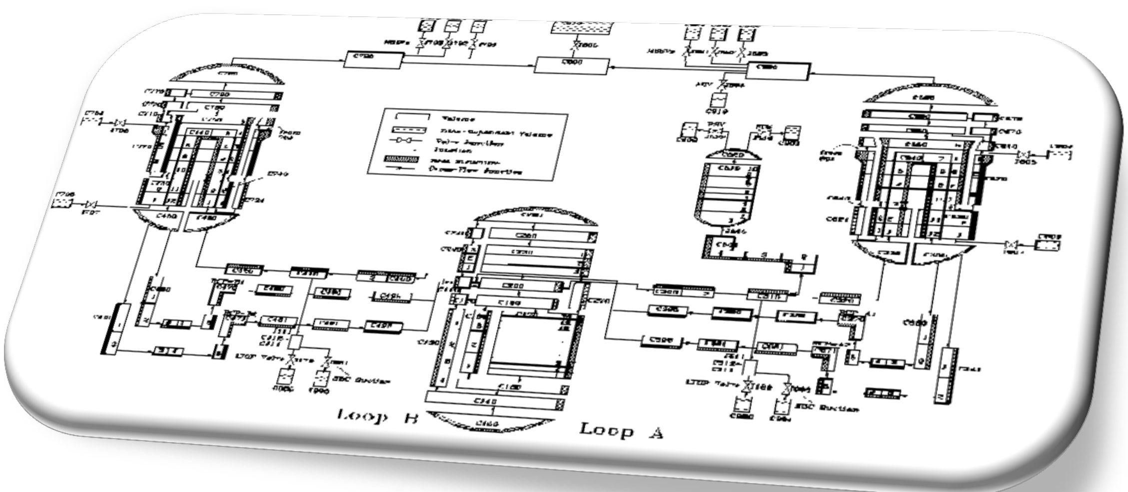
19.1.6 Accident Sequence Analysis

- The AS analysis models the combinations of system responses and operator actions that could occur during the event
- Event Tree analysis is used to delineate these combinations to present these events



19.1.6 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 (2005)



19.1.6.2 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

19.1.6 LPSD Insights

- **Results are dominated by operator recovery failures**
- **Results indicate, as expected, that the draindown and reduced inventory POSs are highly risk significant**

Attachment: Acronyms & Abbreviations

AAC	alternate alternating current
AC	alternating current
AF	auxiliary feedwater
ANS	American Nuclear Society
APR1400	Advanced Power Reactor 1400
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BOP	balance of plant
CC	component cooling water
CCDP	conditional core damage probability
CCF	common cause failure
CCW	component cooling water
CDF	core damage frequency
CET	1) containment event tree, 2) core-exit thermocouple
CFF	containment failure frequency
CFS	cavity flooding system
COL	combined license
CS	containment spray
DC	direct current
DCD	Design Control Document
DCH	direct containment heating
DET	decomposition event tree
DG	diesel generator
ECSBS	emergency containment spray backup system
EDG	emergency diesel generator

Attachment: Acronyms & Abbreviations

EOL	emergency overflow line
EPRI	Electric Power Research Institute
FCI	fuel-coolant interaction
FIRM	fire risk model
FMEA	failure modes and effects analysis
GTRN	general transient
HCLPF	high confidence of low probability of failure
HELB	high-energy line break
HEP	human error probability
HFE	human failure event
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control
IRWST	in-containment refueling water storage tank
ISLOCA	interfacing systems loss of coolant accident
KEPCO	Korea Electric Power Corporation
KEPCO E&C	KEPCO Engineering & Construction Company
KHNP	Korea Hydro & Nuclear Power Company
LOCA	loss-of-coolant accident
LOCV	loss of condenser vacuum
LODCA	loss of dc power (Train A)
LODCB	loss of dc power (Train B)
LOFW	loss of main feedwater
LOOP	loss of offsite power
LPSD	low power and shutdown
LRF	large release frequency
MAAP	modular accident analysis program

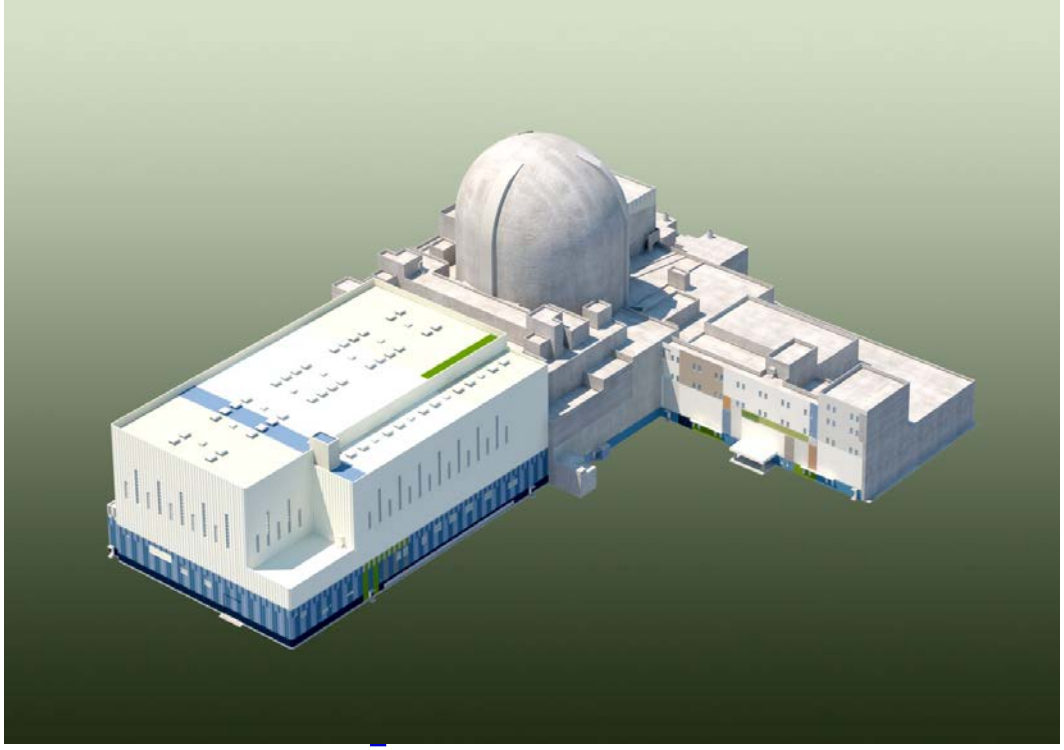
Attachment: Acronyms & Abbreviations

MCA	multiple compartment analysis
MCCI	molten core-concrete interaction
MCR	main control room
MFW	main feedwater
MSPI	mitigating systems performance index
NRC	U.S. Nuclear Regulatory Commission
NUREG	NRC technical report designation
NUREG/CR	NRC technical report designation – performed by contractor
OSP	offsite source permissive
PAR	passive autocatalytic recombiners
PAU	physical analysis unit
PDS	plant damage state
POS	plant operational state(s)
POSRV	pilot-operated safety relief valve
PRA	probabilistic risk assessment
RCP	reactor coolant pump
RCS	reactor coolant system
RG	Regulatory Guide
RPV	reactor pressure vessel
RVR	reactor vessel rupture
SBO	station blackout
SC	1) success criteria analysis, 2) shutdown cooling
SDP	significance determination process
SEL	seismic equipment list
SG	steam generator
SGTR	steam generator tube rupture

Attachment: Acronyms & Abbreviations

SI	safety injection
SLOCA	small break loss of coolant accident
SMA	seismic margin analysis
SOV	solenoid operated valve
SRP	Standard Review Plan
SSCs	structures, systems, and components
SX	essential service water system
UPC	ultimate pressure capacity

APR1400 DCA
Probabilistic Risk Assessment and Severe Accident Evaluation : Chapter 19.2



KEPCO/KHNP

April 19, 2017

Overview of Chapter 19 (1/2)

- **Section Overview**

Section	Title	Presenter
19.0	**Probabilistic Risk Assessment and Severe Accident Evaluation**	Young In
19.1	**Probabilistic Risk Assessment**	
19.1.1	Uses and Applications of the PRA	
19.1.2	Quality of PRA	
19.1.3	Special Design/Operational Features	
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/ Ray Dremel/ Robert Lichtenstein
19.1.4.2	Level 2 Internal Events PRA for Operations at Power	Tae-Hee Hwang/ 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/ Kyu-Ho Hwang/ Ray Dremel
19.1.5.2	Internal Fire Risk Evaluation	Greg Rozga
19.1.5.3	Internal Flooding Risk Evaluation	Ray Dremel
19.1.5.4	Other External Events Risk Evaluation	
19.1.6	Safety Insights from the PRA for Other Modes of Operation	Jaegab Kim/ Ross Anderson/ Jeff Leary
19.1.6.1	Level 1 Internal Events PRA for Low Power and Shutdown Operations	
19.1.6.2	Level 2 Internal Events PRA for Low Power and Shutdown Operations	
19.1.6.3	Internal Fire PRA for Low Power and Shutdown Operations	
19.1.6.4	Internal Flooding PRA for Low Power and Shutdown Operations	
19.1.7	PRA-Related Input to Other Programs and Processes	Young In
19.1.8	Conclusions and Findings	

Overview of Chapter 19 (2/2)

- **Section Overview**

Section	Title	Presenter
19.2	**Severe Accident Evaluation**	Byung Jo Kim
19.2.1	Introduction	
19.2.2	Severe Accident Prevention	
19.2.3	Severe Accident Mitigation	
19.2.4	Containment Performance Capability	
19.2.5	Accident Management	
19.2.6	Consideration of Potential Design Improvement under 10 CFR 50.34(f)	
19.3	**Beyond Design Basis External Event**	Chan-Eok Park
19.3.1	Introduction	
19.3.2	NTTF Tier 1 Recommendation	
19.3.3	NTTF Tier 2 and 3 Recommendation	
19.4	**Loss of Large Area**	Gary Hayner
19.4.1	Introduction and Background	
19.4.2	Scope of the Evaluation	
19.4.3	Conclusions	
19.5	**Aircraft Impact Assessment**	Randy James
19.5.1	Introduction and Background	
19.5.2	Scope of the Assessment	
19.5.3	Assessment Methodology	
19.5.4	Conclusions	

Overview of Chapter 19.2

● Section Overview

19.2.1 Introduction

- **Regulation Considerations**
  - **Severe Accident evaluation for APR1400 design is consistent with the applicable guidance:**
    - **SECY-93-087**
    - **10 CFR Part 100**
    - **10 CFR Part 50 Appendix A,**
    - **10 CFR 50.34 (f)**
    - **10 CFR 50.44**
    - **RG 1.216**
    - **SECY-90-016**

19.2.2 Severe Accident Prevention (1/2)

- **Anticipated Transient Without Scram**
  - **Digital safety system and diverse protection system**
- **Mid-Loop Operation**
  - **Instrumentation for shutdown operation, SCS design, SG nozzle dam integrity, Alternate decay heat removal methods**
- **Station Blackout**
  - **AAC automatically start and is manually aligned to provide power to a Class 1E 4.16 kV when EDGs fail**
- **Fire Protection**
  - **Fire detection, automatic and manual fire suppression, and fixed fire barriers**

19.2.2 Severe Accident Prevention (2/2)

- **Intersystem LOCA**
  - ISLOCA can be occurred at SIS, SCS, CVCS, CSS, etc.
  - All sections of the system and interfaces are designed to withstand full RCS operating pressure,
    or have a leak-test capability,
    valve position indicators in the control room,
    high pressure alarms to warn operators

- **Other SA preventive Features**
  - Two independent turbine driven AFPs when on/off-site AC power are not available
  - SC pumps can be used as a backup of CS pumps during LOCA event
  - Feed-and-Bleed operation using the SIS and POSRVs

19.2.3 Severe Accident Mitigation

- **19.2.3.1 Overview of Containment Design**
- **19.2.3.2 Severe Accident Progression**
- **19.2.3.3 Severe Accident Mitigation Features**

19.2.3.1 Overview of Containment Design

- **Generals**
  - Prestressed concrete structure with cylindrical and dome part
  - Cylinder wall thickness: 1.37 m (4 ft 6 in)
  - Steel liner plate with 6.0 mm thickness on the inside of the dome and cylindrical wall to provide leak-tightness

- **Design characteristics in terms of SA management**
  - Large free volume and dry-type containment
  - Accommodation of condensable and non-condensable gas
  - Natural mixing throughout the containment atmosphere

- **Pressure limits**
  - Design to meet SA internal pressurization challenges
  - Meet the FLC requirements for a period of 24 hours from the onset of core damage, and following this initial 24 hour period

19.2.3.2 Severe Accident Progression (1/3)

- **In-Vessel melt progression**
  - Core heatup resulting from loss of adequate cooling
  - Metal-water reaction and cladding oxidation, core damage
  - Melting and relocation of cladding, structural materials, and fuel
  - Formation of melt pool and crust, and failure of crust in core region
  - Drainage of molten material to the lower head
  - Formation of melt pool and crust in the lower plenum
  - Reactor vessel breach

- **Five vessel failure mechanism in MAAP code**
  - Local ablation of vessel wall by molten jet impingement
  - Melt ingress into a penetration tube
  - Ejection of a penetration tube
  - Creep rupture of the lower head
  - Attack of the vessel wall by overlying metal layer

19.2.3.2 Severe Accident Progression (2/3)

- **Ex-Vessel melt progression**
  - **Key parameters:**
    - **RCS pressure, vessel failure mode and timing, corium releasing characteristics, cavity floor concrete type, availability of cavity flooding, etc.**

- **Events can cause the containment failure:**
  - **HPME and DCH**
    - **High RCS pressure at the time of vessel breach**
  - **EVSE**
    - **Dynamic load generated from rapid mixing with water in the cavity**
  - **MCCI**
    - **Containment basemat melt-through**
    - **Pressurization from evolved steam and non-condensable gases**
    - **Production of combustible gases**
  - **Hydrogen combustion**
    - **Dynamic explosion or deflagration**

19.2.3.2 Severe Accident Progression (3/3)

- **Effort to reduce uncertainty considered in ex-vessel accident progress**
  - **EVSE:**
    - **Base case with realistic input parameters**
    - **Sensitivity cases with bounding input parameters**
  - **DCH:**
    - **Sampled input values by Latin Hyperbolic Sampling technique**
  - **MCCI and H_2 risk:**
    - **Conservative input values to increase concrete ablation depth and hydrogen generation**
  - **Selection of accident sequences:**
    - **Combination of probabilistic and deterministic approach**

19.2.3.3 Severe Accident Mitigation Features (1/8)

- **Hydrogen Generation and Control**
  - Criteria: 10 CFR 50.34(f), 10 CFR 50.44(c)
    - Generation: 100% metal-water reaction (MWR)
    - Control: less than 10% of H_2 concentration inside the containment
  - Mitigation features: Containment, PARs, Igniters
    - Large free volume containment
    - 30 PARs and 8 igniters throughout the containment
    - Meet the Seismic Category I
  - Analysis methodology: Generation and Distribution of H_2
    - MAAP4.0.8
    - Highly probable sequences from PRA Level 1 study && Representative deterministic sequences (LLOCA, MLOCA, SLOCA, SBO, and TLFOW)
    - 100% MWR in "in-vessel phase" + additional source in "ex-vessel phase"

19.2.3.3 Severe Accident Mitigation Features (2/8)

- **Hydrogen Risk (cont'd)**
  - **Analysis methodology: Flame Acceleration and DDT possibility**
    - **H_2-steam-air mixture for all containment nodes from MAAP study**
    - **Applying σ-criterion for FA, 7λ-criterion for DDT**
  - **Analysis methodology: Slow deflagration**
    - **Hydrogen source equivalent to 100% MWR**
    - **Conservative and bounding pressure by AICC assumption**
  - **Analysis results**
    - **Achieving a well-mixed containment atmosphere**
    - **Achieving a H_2 concentration less than 10 %**
    - **No possibility of FA and DDT occurrence**
    - **AICC pressure meets the FLC requirement (see 19.2.4.2)**

19.2.3.3 Severe Accident Mitigation Features (3/8)

- **MCCI and Core Debris Coolability**
  - Goal: Secure the basemat liner integrity
    - Minimize corium-concrete attack
    - Remove heat from the core debris
    - Minimize generation of gases
    - Scrub fission products
  - Mitigation features: Reactor cavity, cavity floor concrete, CFS
    - Cavity with large floor area and no obstacles for corium spreading
    - Concrete layer on the basemat liner with 3 feet thickness
    - Flooding the water from IRWST initiated by manual opening of MOVs at the time of severe accident entry
    - Supplying the sufficient water to cavity by gravity driven flow

19.2.3.3 Severe Accident Mitigation Features (4/8)

- MCCI (cont'd)
  - Analysis methodology: Conservative preparation of model parameters
    - MAAP4.0.8's user-dependent key parameters:
      - Jet breakup
      - Heat removal to overlying water pool
    - CORQUENCH study for conservative Large LOCA sequence
      - High decay heat, full core relocation, no jet breakup
      - Very conservative ablation depth is predicted (0.27 m or 0.86 ft)
    - Decision of MAAP model parameters
      - To get a comparable ablation depth from CORQUENCH study for conservative Large LOCA sequence
  - Analysis results:
    - Large LOCA: 0.24 m (0.79 ft) << 0.91 m (3 ft)
  - Therefore the integrity of basemat liner can be preserved

19.2.3.3 Severe Accident Mitigation Features (5/8)

- **HPME and DCH**
  - Goal: Prevent early containment failure by DCH
    - Provide reasonable depressurization system to prevent HPME
    - Minimize entrained debris to upper containment compartments
  - Mitigation features: Rapid depressurization system, reactor cavity
  - Analysis methodology:
    - Rapid depressurization analysis using MAAP 4.0.8 code
    - DCH analysis using NUREG/CR-6338 methodology
  - Analysis results
    - RCS pressure at reactor vessel failure: less than DCH cutoff pressure (17.6 kg/cm^2 [250 psi])
    - CCFP in APR1400 containment: less than 0.01 percent (0.0001)

19.2.3.3 Severe Accident Mitigation Features (6/8)

- **FCI (In-vessel / Ex-vessel steam explosion)**
  - **Analysis methodology**
    - **Setup the initial conditions**
      - **IVSE: single and multi-jets configuration**
      - **EVSE: base case & sensitivity cases based on MAAP prediction**
    - **Evaluation of energetic loads**
      - **TEXAS-V code**
    - **Assessment of structural integrity**
      - **Vessel load head and cavity wall response using FEM code**
  - **Analysis results**
    - **Integrity of the lower head and the cavity wall are preserved**
  - **EVSE under IVR-ERVC: COL 19.2(3)**

19.2.3.3 Severe Accident Mitigation Features (7/8)

- ES (Equipment Survivability)
  - Purpose
    - Provide reasonable assurance that the equipment and instrumentation can operate under severe accident environment over the required time span
  - Assessment methodology
    - Identification of required equipment and instrumentation
    - Evaluation of severe accident environmental conditions
      - Using MAAP4.08 code for selected sequences
    - Determination of bounding conditions for equipment
      - Containment gas temperature, pressure, radiation
    - Assessment the survivability
      - Comparison with equipment suppliers' test data
      - Comparison with equipment survivability test data
      - Analytical methodology: Thermal lag analysis
      - Alternative means

19.2.3.3 Severe Accident Mitigation Features (8/8)

- **ES (Equipment survivability) (cont'd)**
  - **Assessment results**
    - **Site-specific equipment survivability assessment: COL 19.2(1)**
      - **COL 19.2(1): The COL applicant and/or holder is to perform and submit site-specific equipment survivability assessment 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.**

19.2.4 Containment Performance Capability

- **19.2.4.1 Containment Performance Goal**
- **19.2.4.2 Containment Performance Analysis**

19.2.4.1 Containment Performance Goal

- **Criteria for Containment**
  - **SECY 93-087**
    - **For the first 24 hours, containment maintain its role as a reliable, leak-tight barrier (meets the FLC requirement) under the more likely severe accident challenges**
    - **Following initial 24 hours, containment should continue to provide a barrier against the uncontrolled release of fission product**
  - **RG 1.216**
    - **Position 2, Combustible Gas Control Inside Containment**
      - **Pressure load from H_2 mass and energy releases generated from a 100% MWR accompanied by the burning of H_2**
    - **Position 3, Commission's Severe Accident Performance Goal**
      - **Pressure load from more likely severe accident challenges**

19.2.4.2 Containment Performance Analysis (1/2)

- **Combustible Gas Control (RG 1.216 Position 2)**
  - **Analysis methodology**
    - **H_2 mass from 100% MWR and ignoring PARs, igniters**
    - **Adiabatic Isochoric Complete Combustion (AICC) approach**
    - **Various steam fraction is considered to get a maximum pressure**
  - **Analysis results**
    - **Predicted AICC pressure (123.7 psia) is applied to FEM study**
    - **FEM study indicates maximum strain of liner plate do not reach the allowable limit**
  - **Therefore, conservative H_2 combustion load meets FLC requirement**

19.2.4.2 Containment Performance Analysis (2/2)

- **More likely SA challenges (RG 1.216 Position 3)**
  - Accident selection (Position 3.1 a)
    - Sequences are selected to cover around 90% of cumulative CDF from PRA Level 1 study (draft)
    - 5 representative initiators (Large and Small LOCA, SBO, SGTR, LOFW) are taken into account in the light of the deterministic approach with having conservative accident progress
  - MAAP 4.0.8 calculation for the selected sequences (Position 3.1 b)
    - CFS and POSRVs assumed to be operable
    - ECSBS operates at 24 hours from the onset of SA
    - Determine a bounding pressure profile and peak pressure
  - 3-dim. finite element model study (Position 3.1 c)
    - Maximum strain in the liner plate remains in elastic region
  - Therefore, SA load does not threat the containment integrity

19.2.5.1 SA Management Framework (1/2)

- **Actions taken during the course of an accident to**
  - prevent core damage,
    - Ex) non-LOCA:
      - Secondary side cooling by two MD and two TD AFW pumps.
      - If secondary cooling failed, once-through cooling of the core using SI and SC or CS pumps after depressurization by POSRVs
  - terminate the progress of core damage if it begins and retain the core within the reactor vessel, including ERVC strategy,
    - Ex) Try to inject water by SI. If SI failed, open POSRVs to depressurize RCS and allow injection using SC or CS pumps
    - ERVC may be achieved by using SC pumps to submerge the reactor vessel lower head in water
  - maintain containment integrity as long as possible, and
    - Ex) isolation the containment, operation of CFS to terminate MCCI,
  - minimize offsite releases.
    - Ex) operation of CS to remove fission products by CS/SC pumps or ECSBS

19.2.5.1 SA Management Framework (2/2)

- **COL applicant is to develop and submit an AM plan (COL item 19.2(3)) that addresses**
  - **a systematic evaluation of plant functions during potential SA,**
  - **implementation of the necessary enhancements,**
  - **severe accident management guidelines and training,**
  - **In Vessel Retention-External Reactor Vessel Cooling strategy**

Summary of Chapter 19.2

- **APR1400 Severe Accident Evaluation**
  - SA prevention and mitigation features are designed to conform to associated Criteria and Requirements
  - H_2 risk, MCCI, FCI, DCH, ES are investigated and meet the relevant requirements
  - Containment integrity is consistent with RG 1.216
  - AM plan will be established (COL19.2(3))

Attachment: Acronyms & Abbreviations

AAC	Alternative Alternating Current
AICC	Adiabatic Isochoric Complete Combustion
AFP	Auxiliary Feedwater Pump
ATWS	Anticipated Transient Without Scram
CCFP	Conditional Containment Failure Probability
CDF	Core Damage Frequency
CFS	Cavity Flooding System
CVCS	Chemical Volume Control System
CS	Containment Spray
DCH	Direct Containment Heating
DDT	Deflagration-to-Detonation Transition
ECSBS	Emergency Containment Spray Backup System
EDG	Emergency Diesel Generator
ES	Equipment Survivability
EVSE	Ex-Vessel Steam Explosion
FA	Flame Acceleration
FCI	Fuel Coolant Interaction
FEM	Finite Element Model
FLC	Factored Load Category
HPME	High Pressure Melt Ejection
IRWST	In-containment Refueling Water Storage Tank
IVR-ERVC	In-Vessel Retention and External Reactor Vessel Cooling

Attachment: Acronyms & Abbreviations

IVSE	In-Vessel Steam Explosion
LHS	Latin Hypercube Sampling
LOCA	Loss Of Coolant Accident
LOFW	Loss Of Feed Water
LOOP	Loss Of Offsite Power
MCCI	Molten Core Concrete Interaction
MD	Motor Driven
MOV	Motor Operated Valve
MWR	Metal Water Reaction
PAR	Passive Autocatalytic Recombiner
POSRV	Pilot Operated Safety Relief Valve
PRA	Probabilistic Risk Assessment
RCS	Reactor Coolant System
SA	Severe Accident
SAMDA	Severe Accident Mitigation Design Alternatives
SBO	Station BlackOut
SCS	Shutdown Cooling System
SGTR	Steam Generator Tube Rupture
SIS	Safety Injection System
TCE	Two Cell Equilibrium
TD	Turbine Driven
TLOFW	Total LOFW

Non-Proprietary

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Presentation to the ACRS Subcommittee

Korea Hydro & Nuclear Power Co., Ltd (KHNP) and Korea Electric Power Corporation (KEPCO)

APR1400 Design Certification Application Review

Safety Evaluation Report with Open Items

Chapter 19.1 - PROBABILISTIC RISK ASSESSMENT
Chapter 19.2 - SEVERE ACCIDENT EVALUATION

April 19-20, 2017

Staff Review Team

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- **Technical Staff**
  - **Hanh Phan** (Lead), Senior Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Odunayo Ayegbusi**, Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Anne-Marie Grady**, Reactor Systems Engineer, Containment & Ventilation 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
  - **Jason Schaperow**, Senior Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Courtney St. Peters**, Reliability and Risk Analyst, PRA & Severe Accidents Branch
  - **Hanry Wagage**, Senior Reactor Engineer, Containment & Ventilation Branch
- **Project Managers**
  - **Jeffrey Ciocco,** Lead Project Manager
  - **James Steckel**, Chapter Project Manager

Presentation Outline

Chapter 19.1 - Probabilistic Risk Assessment (PRA)

1) Quality of APR1400 PRA (Phan)
2) Internal Events At-Power Level 1 PRA (Ayegbusi, St. Peters)
3) Internal Fire and Flood At-Power Level 1 PRA (Nakanishi)
4) Internal Events, Fire, and Flood At-Power Level 2 PRA (Wagage)
5) PRA-Based Seismic Margin Assessment (Neuhausen, Roche-Rivera, Phan)
6) Other External Events Evaluation (Phan)
7) Internal Events During Low-power and Shutdown Levels 1 and 2 PRA (Pohida)
8) Internal Fire and Flood During Low Power and Shutdown Levels 1 and 2 PRA (Nakanishi)
9) Uses and Applications of PRA (Phan)
10) Results and Conclusion (Phan)

Quality of APR1400 PRA

Hanh Phan

Quality of APR1400 PRA

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- Ensured the applicant's justification is reasonable and acceptable
  - ♦ Scope
  - ♦ Level of details
  - ♦ Technical adequacy
  - ♦ PRA maintenance and upgrade
- Focused on the information provided in DCD Section 19.1.2 and Table 19.1-1
- Examined the results from the peer review
- Confirmed that the deficiencies would not significantly impact the PRA results and risk insights
- The staff is currently unable to finalize its conclusion on the acceptability of PRA scope, level of details, technical adequacy

PRA Conversion from SAREX to CAFTA

- Initiated during Phase 1 review, June 2015
- Completed in July 2016
- Incorporated:
  - Some of 59 findings from the peer review
  - Some of the staff findings
  - KHNP/KEPCO self-identified issues
- The staff finds:
  - SAREX model and CAFTA model results are not identical
  - Differences between CDFs, LRFs, CCFPs, and risk insights are minor
- Applicant agreed to:
  - Perform self-assessment on the CAFTA model
  - Notify the staff on the changes and results
  - Update PRA notebooks
  - Revisit all sensitivity studies & RAI responses using CAFTA model
  - Revise the DCD

Peer Review

- Peer review was performed during the week of June 24 - 28, 2013, against the PRA Standard, ASME/ANS RA-Sa-2009
- Conducted by a team of 6 PRA experts with over 170 years of diverse PRA experience
- The scope included at-power IEs Level 1, at-power IF Level 1, and LRF
- Peer review resulted in 90 Fact & Observations (59 “Findings,” 27 “Suggestions,” and 4 “Best Practices”)
- Peer review report concluded that *“The APR1400 PRA substantially meets both the ASME PRA Standard and the draft ALWR Standard at Capability Category II or better for 88% of the applicable Supporting Requirements, with 90% met at Capability Category I or better”*

Peer Review (continued)

- The staff obtained insights of the degree to which the APR1400 PRA has been assigned to the capability categories of ASME/ANS Standard
- Together with the staff safety review, the peer review was used to identify the strengths and weaknesses of the PRA and to gain confidence in the PRA model and results
- The staff finds:
  - ♦ Peer review “Findings” have not been completely dispositioned
  - ♦ Some “Findings” were addressed by performing sensitivity studies
  - ♦ Several inconsistencies between the DCD and the peer review report
- The applicant agreed to:
  - ♦ Disposition all “Findings” and update the DCD during Phase 4 review

Internal Events At-Power Level 1 PRA

Odunayo Ayegbusi
Courtney St. Peters

Internal Events (IEs) At-Power Level 1 PRA

Initiating Events, Success Criteria, Event Sequences, and Quantification

Odunayo Ayegbusi

IEs PRA - Initiators, Success Criteria, Event Sequences, and Quantification

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- Reviewed APR1400 DCD Section 19.1.4.1.1 in accordance with SRP 19.0 acceptance criteria (SRP 19.0.II.7)
- Reviewed peer review report and planned resolution which led to less detailed review of the PRA
- Audited APR1400 PRA notebooks at varying levels of detail (audited 70% of each notebook)
- Reviewed DCD references for applicability and use
- Held public meetings with KHNP/KEPCO staff about technical issues and RAIs leading to proposed DCD markups
- The staff found these DCD sections mostly acceptable

Initiating Events and Success Criteria

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- New or unique initiating events to the APR1400 design
  - KHNP/KEPCO did not identify new or unique events
  - A few NUREG/CR-6928 initiating events initially not evaluated
  - RAI issued and closed to confirmatory action item
- APR1400 success criteria
  - Chapter 19 success criteria more conservative than Chapter 15
  - KHNP/KEPCO evaluated inconsistencies and revised DCD
  - RAI issued and closed to confirmatory action item
- PRA model software conversion
  - New platform will incorporate peer review findings and observations
  - Potential impact on PRA modeling and results
  - RAI issued to review changes and impact on current DCD revision

Internal Events At-Power Level 1 PRA

Data Analysis, System Analysis, and Human Reliability Analysis

Courtney St. Peters

IEs PRA - Data Analysis, System Analysis, and Human Reliability Analysis

- Reviewed applicable APR1400 DCD sections in 19.1
- Reviewed peer review report
- Reviewed a sampling of PRA and system notebooks during audit
- Discussed questions during audit and RAIs if additional information was needed for staff findings
- Discussed technical topics at public meetings
- Ensured consistency with other DCD chapters (e.g. I&C, human factors)
- Reviewed key assumptions and followed up on additional justifications (i.e., room heatup/HVAC)
- The staff found these DCD sections mostly acceptable

Digital Instrumentation and Controls

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- The staff review found a lack of detail regarding digital I&C system modeling in the PRA including, but not limited to:
  - System description
  - Key assumptions
  - CCF analysis of both hardware and software
  - Failure effects
- Staff has no findings currently due to continued lack of information
- Public meeting with KHNP/KEPCO staff was held on March 17, 2017, where KHNP/KEPCO committed to provide additional information and an approach to close out this open item

Digital Instrumentation and Controls (continued)

- KHNP/KEPCO staff committed to:
  - Discuss COMMON Q software similarities with Westinghouse
  - Evaluate level of model detail currently provided in the PRA
  - Evaluate architecture of digital I&C compared to reference plant
  - Add software CCF events to the PRA
  - Incorporate information provided into next PRA update and update the DCD

RCP Seal LOCA

- RCP seal LOCA is evaluated as a model uncertainty and sensitivity analysis
- KHNP/KEPCO assumed a failure probability 1×10^{-3} per pump based on engineering judgement before performing a seal LOCA model
- Staff requested additional justification
- APR1400 RCP Seal Model testing results (proprietary information) supported this assumption
- KHNP/KEPCO will revisit model uncertainty and sensitivity analysis as part of the PRA update

Internal Fire and Flood At-Power Level 1 and Level 2 PRA

Tony Nakanishi

Internal Fire Level 1 PRA

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- Reviewed the extent to which applicant's FPRA information is consistent with the applicable methods in NUREG/CR-6850
  - ♦ Certain tasks were not performed or used simpler analyses since design details are unknown at design certification stage (e.g., specifics of cable routing, ignition sources, and target locations)
- Review focused on methodology and assumptions since significant model changes in conjunction with PRA model conversion were expected

Cable Protection

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- In certain risk-significant compartments, cables were assumed to be either protected (e.g., circuits rerouted or redesigned to prevent failure), or can be shown through detailed circuit analysis to not result in the modeled failure mode
- Applicant identified a COL item to ensure that fire protection features required for preventing fire-induced damage of the PRA-credited components will be properly incorporated in the cable design for the as-built condition

Internal Flood Level 1 PRA

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- Reviewed the extent to which applicant's internal flood PRA information is consistent with ASME/ANS standard requirements
- Staff considered results peer review, which found that the internal flood PRA generally met the ASME/ANS requirements for at least Capability Category I
- Staff confirmed detailed identification and characterization of flood areas and flood sources, systematic development of flood scenarios

Maintenance-Induced Floods

- Applicant screened out flooding initiating events caused by inadvertent operation or erroneous operation of a plant component during maintenance
- In response to staff RAI, applicant identified COL Item for COL applicant/holder to demonstrate that maintenance-induced floods are negligible contributors to flood risk when plant-specific information is available

Internal Events, Internal Fire, and Internal Flood At-Power Level 2 PRA

Hanry Wagage

Internal Events Level 2 PRA

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- Reviewed APR1400 DCD Section 19.1 related to Level 2 internal events PRA in accordance with SRP 19.0
- Reviewed Level 2 methodology (PDS, CET/DET, and Release Categories)
- Audited APR1400 PRA Notebooks
- Reviewed peer review report
- Discussed technical issues with the applicant during public meetings

Release Category Analysis

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- CsI release fraction for STC-21 is 357 times higher than that for STC-17 (5.0 versus 0.014 percent of total core inventory) while the release opening area was only 10 times larger (1.0 versus 0.1 ft^2)
- The applicant found a lack of re-vaporization of fission products for STC-17 caused by high pressure in the containment providing better cooling of external surface of the pressurizer
- The staff found that the applicant's justification for the difference reasonable and acceptable

Internal Fire and Internal Flood Level 2 PRA

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- In response to staff RAIs, the applicant stated that the quantification process for Level 2 internal fire and internal flooding is the same as that for Level 2 internal events and proposed DCD changes
- The staff finds that the applicant’s evaluation of Level 2 internal fire and internal flooding acceptable

PRA-Based Seismic Margin Assessment (SMA)

Alissa Neuhausen
Robert Roche-Rivera
Hanh Phan

PRA-Based SMA

- Ensured the applicant's seismic margin assessment is reasonable and acceptable
  - Scope
  - Level of details
  - Technical adequacy
- Focused on the information provided in DCD Section 19.1.5.1 and Table 19.1-47
- Examined the results of the seismic margin assessment (HCLPF calculations report is available for audit)
- Status of plant level HCLPF capacity

Seismic Fragility Evaluation

- The staff examined DCD Section 19.1.5.1
  - ♦ Initial submittal included HCLPF capacities for structures and components based on reference plant design response spectra
- The applicant clarified approach to SMA fragility evaluation
  - ♦ Applicant provided fragility calculations
  - ♦ Conservative Deterministic Failure Margin Approach
  - ♦ HCLPF capacity referenced to APR1400 CSDRS
- The staff finds:
  - ♦ Seismic fragility evaluation is in accordance with guidance in DC/COL-ISG-20
  - ♦ Site-independent structure HCLPF capacities >= 0.5g
  - ♦ Component HCLPF capacities >= 0.5g
  - ♦ Site-dependent structure HCLPF capacities >= 1.67xGMRS PGA

PRA-based SMA Scope and Method

- The method used to assess the at-power seismic margins is acceptable
- The PRA-based SMA during LPSD was not addressed in the DCD
- Staff requested seismic-induced dominant mixed cutsets containing seismic failures, random failures, and operator actions, and, the sequence-level HCLPF capacities during at-power and LPSD modes

Other External Events Evaluation

Hanh Phan

Other External Events

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- Staff ensured that:
  - ♦ The applicant's assessment is comprehensive in scope
  - ♦ The approach used for the screening conforms to the guidance
  - ♦ The screening criteria and/or justification used to support the screening out of an external event are rational and acceptable
  - ♦ APR1400 external hazard treatment is reasonable
- ASME/ANS PRA Standard, Part 6, "Requirements for Screening and Conservative Analysis of Other External Hazards At-Power"
  - ♦ Appendix 6-A, "List of External Hazards Requiring Consideration"
  - ♦ Supporting Requirement EXT-B1, "Initial Preliminary Screening for screening out an external hazard"

Incomplete assessment

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- The staff finds:
  - ♦ No quantitative or bounding analyses for the external hazards specified in the SRM on SECY-93-087
  - ♦ No discussion as to how the main control room would cope with an external fire, i.e., smoke
  - ♦ COL Information Item 19.1(8) is not complete (e.g., missing aircraft crash event, tsunami)
- The applicant agreed to revise:
  - ♦ DCD to include quantitative/bounding analysis and address the main control room issue
  - ♦ COL Information Item 19.1(8)

Internal Events (IEs) During Low-power and Shutdown Levels 1 and 2 PRA

Marie Pohida

IEs LPSD Levels 1 and 2 PRA

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- Reviewed POS definitions for completeness:
  - Time to boiling and core uncovery
  - Status of RCS penetrations, RCS level, decay heat
- Reviewed Event Trees for each POS
- Reviewed GL 88-17 implementation regarding RCS level and temperature instrumentation, availability of pumped injection, installation of nozzle dams, vortexing, and containment closure during reduced inventory conditions
- Reviewed risk significant equipment considered for TS LCO under 10 CFR 50.36(c)(2)(ii)(D)
- Performed confirmatory calculation of applicant's LPSD MAAP analyses for source terms
- Reviewed significant operational assumptions included as risk insights or TS as applicable

LPSD TS and DCD Additions

- Applicant added the following TS and DCD descriptions based on staff questions:
  - ♦ Containment Closure when RCS open via pressurizer manway until refueling cavity water level 23 feet above the reactor vessel flange (TS LCO 3.6.7)
  - ♦ Two trains of SI are operable in hot shutdown, cold shutdown and refueling with refueling cavity water level less than 23 feet above reactor vessel flange (TS LCO 3.5.3)
  - ♦ Midloop operation defined in TS requires >96 hrs post shutdown (DCD Chapter 16)
  - ♦ Availability of PARs and igniters during LPSD documented as risk insight (DCD Chapter 19)
  - ♦ Procedures to open hot leg manway to prevent rapid loss of inventory when cold leg penetrations exist included as risk insight (DCD Chapter 19)
- Staff finds applicant’s approach consistent with guidance, subject to successful closure of open and confirmatory items

Internal Fire and Flood During Low-power and Shutdown Levels 1 and 2 PRA

Tony Nakanishi

Internal Fire and Flood LPSD Levels 1 and 2 PRA

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- No staff-endorsed guidance on performing LPSD internal fire or internal flood PRA
- NUREG/CR-7114 provides framework for performing LPSD fire PRA
- Staff review focused on changes made relative to the full-power fire and flooding and the internal events LPSD PRA models (e.g., induced initiating events, barrier integrity), and on adequacy of COL item addressing key assumptions

Uses and Applications of PRA

Hanh Phan

Uses and Applications of PRA

- PRA Results and Insights were used as an input to
  - ♦ Chapter 13.6, *“Physical Security”*
  - ♦ Chapter 14.3, *“ITAAC”*
  - ♦ Chapter 16, *“TS”*
  - ♦ Chapter 17.4, *“RAP”*
  - ♦ Chapter 18, *“HFE”*
  - ♦ Chapter 19.2, *“SA”*
  - ♦ Environmental report, *“SAMDA”*
- Influenced the selection of design features, i.e., four EDGs and battery depletion time
- Ensured that:
  - ♦ PRA is commensurate with the uses and applications
  - ♦ Input used for above programs is sufficient
  - ♦ Consistency between Chapter 19 and other Chapters

Results and Conclusion

Hanh Phan

Phase 2 Staff Findings

- The outstanding issues are:
  - ♦ Appropriate scope, level of detail, and technical adequacy of APR1400 PRA for its identified uses and applications
  - ♦ The reasonableness of APR1400 PRA to reflect the as-designed, as-to-be-built, and as-to-be-operated plant

Section 19.2

Severe Accident Evaluation

Presentation Outline

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Chapter 19.2 - Severe Accident Evaluation (SAE)

1) Severe Accident Prevention (Wagage)
2) Severe Accident Mitigation (Wagage, Pohida, Grady)
3) MELCOR Confirmatory Analysis (Schaperow, Campbell)
4) Containment Performance Capability (Roche-Rivera, Neuhausen)
5) Conclusion (Wagage)

Severe Accident Prevention

Hanry Wagage

Severe Accident Prevention

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- Reviewed prevention issues:
  - ♦ Anticipated transient without scram
  - ♦ Mid-loop operation
  - ♦ Station blackout
  - ♦ Fire protection
  - ♦ Interfacing system loss-of-coolant accident

Station Blackout

- Provides one alternate ac (AAC) source, which is independent and diverse from the Class 1E EDGs
- Design change from two EDGs to four EDGs
- Extension of 125 Vdc battery life to 16 hours from 8 hours
- Successful startup of the AAC together with turbine-driven auxiliary feedwater pumps would prevent core damage during SBOs

Interfacing System Loss of Coolant Accident

- SECY-90-016 recommends low-pressure systems to be designed to withstand full RCS pressure or to provide means of testing pressure isolation valves and indications
- SIS, SCS, and CVCS are directly connected to the RCS:
  - SIS and SCS interfaces are designed to withstand full RCS operating pressure or have a leak-test capability
  - CVCS letdown and charging lines each has a high-pressure alarm to warn the operator when the pressure is approaching the low-pressure system design pressure
- Designing SCS lines to reduce ISLOCA was not clear as DCD stated designing for RCS full pressure or leak test capability and eliminating interfacing lines. In response to audit question, the applicant proposed updating DCD to clarify that deletion of unnecessary interfaces (e.g., the purification return line)

Severe Accident Mitigation “Progression and Features”

Hanry Wagage

Severe Accident Mitigation

- Reviewed DCD Section 19.2 and APR1400-E-P-NR-14003-P, "Severe Accident Analysis Report"
  - ♦ MCCI and Core Debris Coolability
  - ♦ DCH and HPME
  - ♦ FCI: In- and Ex-Vessel Steam Explosions
  - ♦ Containment Bypass
- Audited the sump evaluation, "Ex-Vessel Severe Accident Analysis for the APR1400 with the MELTSPREAD and CORQUENCH Codes"
- Reviewed severe accident mitigation features identified by the applicant using SRP 19.0 and SECY-90-016

In-vessel Steam Explosions

- The FCI expert review group concluded in NUREG-1116 and NUREG-1524 that the probability of containment failure was vanishingly small or physically unreasonable
- The applicant used one-dimensional TEXAS-V computer code for calculating steam explosion loading
- The staff asked for justification for the chosen cross-sectional area of the pool for TEXAS calculation
- The applicant used ABAQUS 6.10 code for structural analysis and Shockey criteria of 11% plastic strain for allowable limit

Ex-vessel Steam Explosions

The remaining issues are as follows:

- Cross-sectional area of the pool used for one-dimensional TEXAS computer code calculations
- Calculation of impulse on cavity structures using TEXAS results
- Structural evaluation of cavity structures
- Impact of IVR/ERVC operation of steam explosion loading

Ex-vessel Steam Explosion: Impact of IVR/ERVC operation

- The SCS provides water from the IRWST for active cavity flooding to perform ERVC strategy that leads to flood water to appropriate elevation
- The applicant did not assume the operation of IVR/ERVC system for DCD Chapter 19 analysis
- Although the active flooding may cool the core melt in-vessel, a possibility exists for the vessel bottom to fail on side causing a larger melt jet that would generate higher steam explosion energy than analyzed
- The applicant proposed a COL information item to analyze the above condition under accident management plan

Severe Accident Mitigation Equipment Survivability

Anne-Marie Grady

Equipment Survivability

- Objective:
  - ♦ SECY 93-087 requires mitigation features be designed to operate in the severe-accident environment for which they are intended and over the time span for which they are needed
  - ♦ 10 CFR 50.44(c)(3) requires containments to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen equivalent to that generated from a fuel clad-coolant reaction involving 100 % of the fuel cladding

Equipment Identified

- Applicant selected accident scenarios from the most probable core damage sequences in the Level 1 PRA, and, from several LOCAs
- Applicant identified required mitigation functions of RCS inventory control; RCS heat removal; reactivity control; and containment integrity
- Applicant identified specific equipment required to achieve each function in DCD Tier 2, Table 19.2.3-4, “Systems and Equipment/Instrumentation Required for Equipment Survivability Assessments”
- Staff agreed with equipment identified, but requested addition of ECSBS-V1014, and, CIV and penetration integrity (applicant agreed)

Accident Conditions Characterized

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- Staff finds the methodology, identification of equipment required for accident mitigation, and environmental conditions for equipment survivability establishes sufficient guidance and input for the COL applicant to demonstrate compliance with 10 CFR 50.44(c)(3) and 10 CFR 50.34(f)
- The temperature profiles were confirmed in the staff confirmatory calculations
- The applicant's AICC pressure of 110 psia bounded the staff confirmatory pressures

Accident Conditions Characterized (continued)

- The applicant calculated the severe accident radiation dose of 4.4E+05 Gy (4.4E+07 rad) using the MAAP4-DOSE code
- Staff compared these results to those from other ALWR's of similar size, design and fuel type, and severe accident scenarios, and found the containment doses comparable
- Staff also found the containment atmospheric assessments of temperature, pressure and radiation described in DCD section 19.2.3.3.7 acceptable for evaluating equipment survivability
- COL Information Item 19.2(1) will be revised to state that the COL applicant and/or holder is to perform and submit site specific equipment survivability assessment based on the above equipment and containment atmospheric assessments

MELCOR Independent Confirmatory Analysis

Jason Schaperow
Shawn Campbell

MELCOR Confirmatory Analysis

- To confirm applicant's use of MAAP for PRA and severe accident analysis
- To perform independent analysis for select scenarios using MELCOR and compare with MAAP results
- Remaining issues
  - ♦ The applicant is assessing impact of MAAP at-power sensitivity calculations on the SAMDA analysis
  - ♦ The applicant is revising MAAP shutdown analyses

MELCOR Confirmatory Analysis

MAAP Case	Objective of MAAP Case	Accident Initiator	New Reactor Severe Accident Feature		
			Rapid Depressur-ization	Cavity Flooding	ECSBS at 24 hours
STC10	Source term for containment leakage	LOCCW	✓	✓	✓
STC11	Source term for basemat melt-through	LOCCW	✓		✓
STC16	Source term for containment leak at 24 hours	LOCCW	✓		
Q03	Containment pressure for dry cavity	SBO			
POS5*	Time to core damage, lower head failure	Loss of SDC and injection			

* Mid-loop accident

At-power Accidents

Comparison of MELCOR and MAAP results led to identifying differences in assumptions

Assumption	MELCOR	MAAP
SITs	✓	
Hot leg creep rupture	✓	
RCP seal leakage/failure	✓	
Operators open POSRVs	CET = 922K (SAMG value)	After first automatic lift (feed & bleed procedure)
Operators open 3-way valves and cavity flooding valves	CET = 922K (SAMG value)	30 minutes after CET = 922K (assumed delay)

CET - core-exit thermocouple

At-power Accidents

- The applicant ran MAAP sensitivity cases to address differences in assumptions
- Staff scaled the cesium releases in the DCD by the differences between DCD Cases STC10, STC11, and STC16 and the Cases STC10, STC11, and STC16 sensitivities
  - ♦ LRF – scaling the cesium releases does not change the release from small to large for any source term category
  - ♦ SAMDA – scaling the cesium releases is unlikely to affect the SAMDA analysis given its margins
- CET quantification
  - ♦ The containment pressure differences between Case Q03 sensitivities in the RAI response and Case Q03 in the DCD are unlikely to affect CET quantification, because the median ultimate containment failure pressure is 162.7 psig

Mid-loop accidents

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- Staff modified the MELCOR at-power deck to simulate a mid-loop accident
  - ♦ Reduced decay heat to account for time until reactor is in mid-loop configuration
  - ♦ Simulated nozzle dams by blocking flow paths connecting the steam generators to the hot legs and cold legs
  - ♦ Added open manway at top of pressurizer
  - ♦ Isolated the SITs from the RCS
  - ♦ Set initial RCS pressure, temperature, and water level to 14.7 psia, 330K, and hot leg mid-plane, respectively
- The applicant is revising its MAAP analysis and documentation
  - ♦ More realistic modeling of RCS and containment configuration
  - ♦ Code bug

Containment Performance Capability

Robert Roche-Rivera
Alissa Neuhausen

Containment Performance Capability

- Ensured the applicant meets the Commission's deterministic containment performance goal as described in SECY 90-016 and 93-087
- Focused on the information provided in DCD Section 19.2.4
- Compared the applicant's evaluation to staff guidance
- Examined the results of the applicant's finite element analysis
- Confirmed that the ASME Factored Load Category criteria are met for severe accident loading
- The staff has completed the review of the deterministic containment performance goal

Factored Load Category

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- Deterministic containment performance goal
  - Containment maintains its role as reliable leak-tight barrier
    - Containment stresses do not exceed ASME service level C limits/Factored Load Category for 24 hrs following onset of core damage
  - Containment continues to provide a barrier against the uncontrolled release of fission products after 24 hrs
- The applicant selected a conservative severe accident load
- The applicant demonstrated that the most significant pressure-loading histories: LLOCA, SBO, TLOFW are bounded by the severe accident load selected for the FLC
- The applicant demonstrated that the strains in the liner plate do not reach the allowable limit strain values
- Staff concludes strain limits meet ASME Code

Conclusion

Hanry Wagage

Phase 2 Staff Findings

Due to the remaining issues, the staff is unable make final conclusions on the severe accident evaluation of the APR1400 design

ACRONYMS

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- **ACRS -** Advisory Committee on Reactor Safeguards
- **AICC -** adiabatic isochoric complete combustion
- **ALWR -** advanced light-water reactor
- **ANS -** American Nuclear Society
- **ASME -** American Society of Mechanical Engineers
- **CCF -** common-cause failure
- **CCFP -** conditional containment failure probability
- **CDF -** core damage frequency
- **CET -** containment event tree or core-exit thermocouple
- **CFR -** Code of Federal Regulations
- **CFS -** cavity flooding system
- **CIV -** containment isolation valve
- **COL -** combined license
- **CSDRS -** certified seismic design response spectra
- **CVCS -** chemical and volume control system
- **DC -** design certification
- **DCD -** design control document
- **DCH -** direct containment heating
- **DET -** decomposition event tree
- **ECSBS -** emergency containment spray backup system
- **EDG -** emergency diesel generators
- **EE -** external events
- **ERVC -** external reactor vessel cooling
- **FCI -** fuel-coolant interaction
- **FLC -** factored load category
- **GMRS -** ground motion response spectra
- **HCLPF -** high-confidence-and-low-probability-of-failure
- **HFE -** human factors engineering
- **HPME -** high-pressure melt ejection
- **HVAC -** heating, ventilation, air conditioning
- **HVT -** holdup volume tank
- **I&C -** instrumentation and control
- **IE -** initiating event

ACRONYMS (continued)

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- **IF -** internal flood
- **IRWST -** in-containment refueling water storage tank
- **ISLOCA -** interfacing system loss-of-coolant accident
- **ITAAC -** inspections, tests, analyses, and acceptance criteria
- **IVR -** in-vessel retention
- **KEPCO -** Korea Electric Power Corporation
- **KHNP -** Korea Hydro and Nuclear Power Co.
- **LCO -** limiting condition for operation
- **LLOCA -** large-break loss-of-coolant accident
- **LOCA -** loss-of-coolant accident
- **LOCCW -** loss of component cooling water
- **LPSD -** low power and shutdown
- **LRF -** large release frequency
- **MAAP -** modular accident analysis program
- **MCCI -** molten core-concrete interaction
- **PARs -** passive autocatalytic recombiners
- **PDS -** plant damage state
- **PGA -** peak ground acceleration
- **POS -** plant operational state
- **PRA -** probabilistic risk assessment
- **RAI -** request for additional information
- **RAP -** reliability assurance program
- **RCP -** reactor coolant pump
- **RCS -** reactor coolant system
- **RPV -** reactor pressure vessel
- **SA -** severe accident
- **SAE -** severe accident evaluation
- **SAMDA -** severe accident mitigation design alternatives
- **SAMG** - severe accident mitigation guidelines
- **SBO -** station blackout
- **SCS -** shutdown cooling system
- **SDC -** shutdown cooling

ACRONYMS (continued)

- **SI -** safety injection
- **SIS -** safety injection system
- **SIT -** safety injection tank
- **SMA -** seismic margin assessment
- **SRM -** staff requirements memorandum
- **SRP -** Standard Review Plan
- **STC -** source term category
- **TLOFW -** total loss of feedwater
- **TS -** technical specifications