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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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U.S. APWR SUBCOMMITTEE

OPEN SESSION

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MONDAY

JULY 9, 2012

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ROCKVILLE, MARYLAND

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

SUBCOMMITTEE MEMBERS:

JOHN W. STETKAR, Chairman

SANJOY BANERJEE, Member

CHARLES H. BROWN, JR. Member

DANA A. POWERS, Member

JOY REMPE, Member

WILLIAM J. SHACK, Member

NRC STAFF PRESENT:

GIRIJA SHUKLA, Designated Federal Official

ANDREW BIELEN, RES

JOSEPH DONOGHUE, NRO

FRED FORSATY, NRO

MICHELLE HART, NRO

MICHELLE HAYES, NRO

STEVE MONARQUE, NRR

JEFFREY SCHMIDT, NRO

AMY SNYDER, NRO

MICHAEL TAKACS, NRO

ALSO PRESENT:

DAVID CARAHER, ISL

HIROSHI FUJISHIRO, MHI

YUKO FUJITA, MNES

HIROSHI HAMAMOTO, MHI

TERRY HEAMES, Alion

HIDEAKI IKEDA, MHI

MASAAKI KATAYAMA, MHI

KEVIN LYNN, MNES

YUTA MARUYAMA, MNES

REIKO NARUTOMI, MHI
JUNTO OGAWA, MHI
MARCOS ORTIZ, ISL
DAN PRELEWICZ, ISL
RYAN SPRENGEL, ISL
REBECCA STEINMAN, MNES
RURIKO TAKAMASHI, MNES
TETSUYA TERAMAE, MHI
DOUG WOOD, MHI
DON WOODLAN, Luminant Power
TSUYOSHI YOSHIDA, MHI

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C-O-N-T-E-N-T-S

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

(8:36 a.m.)

CHAIR STETKAR: The meeting will now come to order. This is a meeting of the United States Advanced Pressurized Water Reactor Subcommittee. I am John Stetkar, Chairman of this Subcommittee meeting.

ACRS Members in attendance are Sanjoy Banerjee, Bill Shack, Charlie Brown and Joy Rempe. I believe that we will be joined by Dana Powers and we may be joined by Steve Schultz, not particularly clear there.

Mr. Girija Shukla of the ACRS staff is the Designated Federal Official for this meeting. The Subcommittee will discuss Chapter 15, Transient and Accident Analysis of the Safety Evaluation with open items associated with the US APWR Design Certification, and the Comanche Peak combined license application, and related topical reports, MUAP-07011 large-break LOCA code applicability report for US APWR, and MUAP-07013 small-break LOCA methodology for US APWR, and their Safety Evaluation

Reports.

We will hear presentations from the NRC staff and Mitsubishi Heavy Industries. We received no written comments or requests for time to make oral statements from members of the public regarding today's meeting.

There is a bridge line. It will be kept on mute during the meeting so as to not disturb the proceedings. We will open it if there is reason we need support from people on the other end, or if there are members of the public, we will open it for comments at the end of the meeting.

The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions as appropriate for deliberation by the full Committee.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the Federal Register.

Parts of this meeting may need to be closed to the public to protect information proprietary to MHI or other parties. I am asking the NRC staff and the applicant to identify the need

for closing the meeting before we enter into such discussions, and to verify that only people with the required clearance and need to know are present, and I am going to be relying very heavily on MHI and MNES to monitor that. If we delve into --

MS. STEINMAN: The material today should only be publicly available.

CHAIR STETKAR: Okay, but if questions come up during the discussion and we delve into proprietary responses, we can close the meeting. It's just a matter of --

MS. STEINMAN: Okay.

CHAIR STETKAR: controlling the logistics of doing that.

MS. STEINMAN: Okay.

CHAIR STETKAR: So I'll ask you to keep us on the straight and narrow there. Parts of this -- sorry, I read that already. A transcript of this meeting is being kept and will be made available as stated in the Federal Register notice. Therefore, we request that participants in this meeting use the microphones located throughout the meeting room when addressing the Subcommittee.

The participants should first identify

themselves and speak with sufficient clarity and volume so they may be readily heard. I'd ask you all to please silence your cell phones and let me give you a little bit of information about the schedule.

We are scheduled for a day and a half. We need to finish tomorrow at noon time, at 12 sharp, because we do have other conflicting meetings tomorrow. So we can't run long tomorrow.

We, depending on people's stamina, have a little bit of flexibility to run a little late this afternoon, so we'll just, we'll see how the presentations go.

So, if it looks like we are running behind today, too badly, we are going to try and extent it a little bit the end of this afternoon because we do need to finish promptly at noon tomorrow.

And with that, I will turn the meeting over to NRC management.

MS. SNYDER: Good morning Chairman Stetkar and ACRS Subcommittee. It is a pleasure to be here today. My name is Amy Snyder. I am acting branch chief for Licensing Branch 2, which is

responsible for the project management of the US APWR.

Today, as you said, we are going to be focusing on Chapter 15, transient and accident analysis. This is a major milestone for the staff. We have been working on this since mid-2007.

The project manager for the design certification is Jeff Ciocco. He is not here today. The project manager for Chapter 15 is Michael Takacs we have technical staff from the Reactor Systems Branch, Joe Donoghue's branch, and also we will have presentations from the -- our contract support Information Systems Laboratories.

CHAIR STETKAR: Nothing more? Thank you, and with that I will turn the meeting over to MHI.

MR. IKEDA: Yes, thank you Chair. Well, good morning everybody. My name is Hideaki Ikeda. I am responsible for the LOCA code development and safety analysis, safety design, MHI, Mitsubishi Heavy Industries.

This presentation provides a brief overview of MHI's small-break LOCA methodology for US APWR. Next slide please.

Masaaki Katayama, besides me, is co-presenter and he is deputy manager of LOCA section and also our lead engineer of US APWR small-break LOCA in our team. Next slide please.

The objective of the topical report is to present the MHI's comprehensive methodology for US APWR small-break LOCA analyses.

The key topics of the topical report are: first is identification of the US APWR design feature; next, development of the phenomena identification ranking table and assessment matrix; third, development of M-RELAP5, a modified version of what we called RELAP5-3D; and finally the assessment for the evaluation model adequacy of M-RELAP5 and applicability to the US APWR's small-break LOCA will be discussed in the topical report.

Now, M-RELAP5 is being applied to US APWR small-break LOCA safety analysis and the present topical report is referred to in Chapter 15.6.5 of the US APWR Design Control Document, DCD.

Next slide. First of all, the future of US APWR. This slide summarizes the primary specification of US APWR related to LOCA. The US APWR employs a four-loop configuration with

U-tube-type steam generators.

The fuel assembly is a 17 by 17 rod lattice layout, and 257 fuel assemblies are loaded in the core, providing 4,451 megawatts thermal output. Also, the thermal output is larger than the current load of PWR. The average linear heat rate is reduced as well to 4.6 kW per foot by extending the average linear heat rate to 14 feet.

The engineered safety features are the high-head injection system, HHIS and the advanced accumulator passive safety device. Next slide please.

MEMBER BROWN: Excuse me. I've missed a couple of meetings but how do the steam generators for this APWR compare with other steam generators you have made or plants in the U.S. or in Japan? Are they similar to other designs?

MR. IKEDA: Yes, the design is similar, but the size is larger than the conventional four-loop PWRs.

MEMBER REMPE: So they are larger than the ones, for example, that you made for San Onofre?

MR. IKEDA: For example, heat transfer -- sorry, excuse me.

MR. SPRENGEL: This is Ryan Sprengel with MNES. The quick answer is no, they are smaller. They are significantly smaller than SONGS.

MEMBER REMPE: Okay, thank you.

MR. SPRENGEL: His reference point was the conventional Japanese plant.

MEMBER REMPE: Okay.

MR. IKEDA: Yes.

MEMBER REMPE: Thank you.

MR. IKEDA: Thank you very much.

MEMBER BROWN: Is the design the same, the issue that came up in terms of the manufacturing and the other issues that have popped up or is that going to be discussed or what?

MR. SPRENGEL: I don't think we are going to get into that, because it's a little cart before the horse. We are waiting on the evaluations to be completed at SONGS before we proceed with anything --

MEMBER BROWN: Okay, so this will be a more generic evaluation --

MR. SPRENGEL: That's correct. We have already actually received them --

MEMBER BROWN: relative to the actual

components themselves?

MR. SPRENGEL: Right, we have received an RAI requesting us to evaluate and incorporate any lessons learned, so that is work that will be done and will be presented.

MEMBER BANERJEE: So far as the tube side flow area is concerned, the megawatt of power, is that smaller or larger than the conventional Japanese systems?

MR. IKEDA: And, sir, you are talking about heat transfer area?

MEMBER BANERJEE: No, flow area.

MR. IKEDA: Flow area is larger than the conventional --

MEMBER BANERJEE: Per megawatt.

MR. IKEDA: Maybe as a flow area multiplication factor --

MEMBER BANERJEE: Can you give us that number for both the hot leg and the steam generator?

MR. IKEDA: Yes. Okay, I have a backup presentation slide for the tomorrow discussion. I will explain tomorrow. So --

MEMBER BANERJEE: But one of the issues we will explore with the small break will be

refluxing, so we need to know the flow areas.

MR. IKEDA: Yes, so I remember the ratio of thermal output compared with the conventional four-loop. Typically the four-loop PWR allowed 1.3. On the other hand, the flow area of the steam generator is allowed 1.4.

MEMBER BANERJEE: So, just give me the numbers.

MR. IKEDA: Yes.

MEMBER BANERJEE: Okay. Both for the steam generator and the hot leg.

MR. IKEDA: Steam generator.

MEMBER BANERJEE: And the hot leg. That is required.

MS. STEINMAN: Ask him tomorrow to bring both those --

MR. IKEDA: Yes, I remember the exact number for the hot leg so I will give you information tomorrow.

MEMBER BANERJEE: Right.

MR. IKEDA: But anyway, the flow area of the hot leg and also for the steam generator flow area is rather comparably --

MEMBER BANERJEE: In absolute terms, I'm

sure it is.

MR. IKEDA: Yes. Yes.

MEMBER BANERJEE: It is from the power generated during the, you know, operation and the small-break LOCA, that's very interesting.

MR. IKEDA: Can I move on?

MEMBER BANERJEE: Yes.

MR. IKEDA: Okay, thank you. Next slide please. Okay. This slide --

CHAIR STETKAR: Folks up front, just be a little bit careful with your paper on the microphones.

MR. IKEDA: Oh, okay.

CHAIR STETKAR: They are really sensitive and they give our reporter a lot of problems, so be careful not to hit them. Thank you.

MR. IKEDA: Okay. So, this slide explains the US APWR, the small-break LOCA scenario. The transient evolution is similar to that of the Westinghouse type four-loop PWR.

The accident scenario is divided into five phases -- blowdown, natural circulation, loop seal formation and this clearing, boiloff and core recovery phases.

The figures show the RCS, reactor coolant system, pressure response and the core liquid level response activity.

MEMBER BANERJEE: Will you go into more detail in the future on how you do your loop seal clearing calculations?

MR. IKEDA: Yes.

MEMBER BANERJEE: So you will, okay. So let's defer that for the moment.

MR. IKEDA: Okay.

MEMBER BANERJEE: And this is based on how many? Is this just one loop seal clearing?

MR. IKEDA: Exactly all the loop allowed to -- allowed clearing if possible. No penalty is given for M-RELAP5 calculation in terms of loop seal clearing. And then --

MEMBER BANERJEE: But the most conservative would be just to clear one loop seal, right?

MR. IKEDA: Yes. That -- for the US APWR small-break LOCA calculation, we identified all the loop clear in our representative small-break LOCA case.

MEMBER BANERJEE: Why did you do that?

Will you come back and talk about the -- this is of course, as you know, very important, for small-break LOCA. So we want to go into some detail as to how you do the loop seal clearing, when you do it, why you do it, all the details. So we can do it later --

MR. IKEDA: Okay.

MEMBER BANERJEE: when you show it.

MR. IKEDA: Unfortunately the presentation material today does not include such information, so I will provide a discussion, Sanjoy, I think, tomorrow.

MEMBER BANERJEE: Okay.

MR. IKEDA: Next slide please. So the -- our small-break LOCA methodology is based on the M-RELAP5 code as I mentioned, and M-RELAP5 is a conservative code, conforming to regulatory requirements specified by Appendix K to 10 CFR Part 50.

Because US APWR employs ZIRLO cladding materials, figures of merit are identical to those specified in 10 CFR Part 50.46, so, PCT, ECL, hydrogen generation, coolable geometry, and long-term cooling criteria.

Next slide please. The basis of our SBLOCA methodology, M-RELAP5 code has been developed based on RELAP5-3D code as I mentioned. The complete two-fluid thermal-hydraulics model is applicable to important processes and phenomena occurring under US APWR small-break LOCAs.

MHI's, our modification to M-RELAP5 is limited to implementation of conservative models and our plant-specific models such as the advanced accumulator models.

M-RELAP5 code has been developed and assessed in accordance to evaluation model development and assessment procedure in regulatory guide 1.203.

MEMBER BANERJEE: So with regard to the advanced accumulator model, which we will hear about later, this fall, could you tell us in more detail at some point what that model that you put in, is? Is it just a flow versus pressure or what that model is, would be useful to know.

MR. IKEDA: So you need --

MEMBER BANERJEE: You're telling us just assume the model for now and we'll discuss this later?

MR. IKEDA: And do you need a transient response under the small-break LOCA? So you said --

MEMBER BANERJEE: Well, at the moment I am just interested in knowing what is the format of the model you have put in, you know. We can go into the model itself when the time comes, but right now, just informational.

MR. IKEDA: Okay.

MEMBER BANERJEE: Okay.

MR. IKEDA: I see. Next slide.

MEMBER BANERJEE: Also, the other specific area I am interested in is whether you have put a model for the elbow region of the hot leg for flooding.

MR. IKEDA: CCFL.

MEMBER BANERJEE: CCFL. And what that model is.

MR. IKEDA: Now, for the hot leg CCFL we employed the type of conductive rod theory.

MEMBER BANERJEE: Whatever it is, if you just -- right now, this is, again, just informational.

MR. IKEDA: Okay. Okay.

MEMBER BANERJEE: What it was that you

used. Thank you. Okay.

MR. IKEDA: Can I move on?

MEMBER BANERJEE: Yes.

MR. IKEDA: Thanks. So, this slide summarizes the M-RELAP5 Appendix K models. In conformance to Appendix K, to 10 CFR 50, the following models are implemented on M-RELAP5 -- ANS-1971 decay heat models for the conservative use, they multiplied the heat -- I'm sorry -- the decay heat model is multiplied by 1.2; the fuel gap conductance model equivalent to that used in the fuel design code. Our code is FINE for US APWR.

And next, the Baker-Just metal-water reaction models and the cladding swelling and rupture models applicable to ZIRLO cladding material, and the Moody, two-phase critical flow models and finally, no return logic and no return to transition or nuclear boiling during the blowdown phase.

MEMBER REMPE: Excuse me again. Could you elaborate a little bit about what you mean by the fuel design code, and what data supports that fuel design code and where it is with respect to NRC approval?

MR. IKEDA: Yes. That code, the fuel design code is under review now. Is that correct? The FINE code -- I'm sorry. The FINE code is under the review now?

MR. SPRENGEL: The FINE code topical report has an SER base to be completed by the staff but it has not come on to the ACRS yet.

CHAIR STETKAR: We have a September subcommittee meeting that will be addressing the fuel models.

MEMBER REMPE: Okay, and there are data that you obtained from Haldon to support that code?

MR. IKEDA: Yes.

MEMBER BROWN: Okay. We will look forward to it.

MEMBER REMPE: Okay.

CHAIR STETKAR: Put it on your calendar. September 20th.

MEMBER REMPE: I will.

MEMBER BANERJEE: I'm sure you'll have more questions when the LBLOCA comes up.

MR. IKEDA: Okay, so the last topic is in order to simulate the advanced accumulator flow rate, we have developed the advanced accumulator

model based on the RELAP5-3D accumulator models. Next slide please.

So hereafter, I'd like to give a summary of US APWR small-break LOCA analysis for the Safety Evaluation.

M-RELAP5 can treat the LOCA with the break sizes up to one square foot.

MEMBER BROWN: Can -- not being a thermal hydraulics guy, they said they've scaled -- they've got their scaling bias set up from the halfscale data using the CFD calculations. Is that a common --

MEMBER BANERJEE: They will go through that, I guess, later.

CHAIR STETKAR: Yes, we've got an October subcommittee.

MEMBER BROWN: Yes, okay.

CHAIR STETKAR: We are going to hear about the detailed models for the accumulator.

MEMBER BANERJEE: No, it's not obvious.

MEMBER BROWN: Okay. I just --

CHAIR STETKAR: At least that's what's scheduled right now.

MEMBER BROWN: I just wanted to know how

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it was validated. That's --

CHAIR STETKAR: That's, that's --

MR. SHUKLA: It's a good question.

CHAIR STETKAR: It's an excellent question.

MEMBER BANERJEE: How do you know everything?

MEMBER BROWN: Thank you. Okay, my questions end. You can go on. Thank you.

MR. IKEDA: Thank you. So the technical report, MUAP-07025, small break LOCA sensitivity analysis for US APWRs, which is referred to in the US APWR DCD, provides the following sensitivity calibration, like break location, break size, break orientation, plant noding, time step size, offsite power availability, single failure assumption and accumulator flow rate, to check the sensitivity of the flow rate bias for the advanced accumulator.

By performing this sensitivity calculation, it is identified that the loop seal PCT occurs during the 7.5-inch cold leg bottom break and that the boiloff PCT occurs in the one square foot, cold leg bottom break.

In addition, the condition of the loss

of offsite power and two ECCS trains available due to -- the two ECCS is not available, provides the limiting PCT in the safety analysis.

CHAIR STETKAR: The sensitivity calculations that you did for the small-break LOCA, I know that you did, I think, one calculation from the pressurizer steam space, and it, I believe it said that the size of that break was about 6.7 inches, is what it said in the report, about.

How large is the depressurization valve line? Not the safety depressurization valves, but the -- the severe accident depressurization valve line. How large is that pipe?

MR. IKEDA: How large --

MR. HAMAMOTO: This is Hiroshi Hamamoto, MHI. The size is six-inch.

CHAIR STETKAR: It is six-inch?

MR. HAMAMOTO: Yes.

CHAIR STETKAR: Thank you. So that bounds -- the 6.7 would bound a break in that line.

MR. HAMAMOTO: Thank you.

MR. IKEDA: This slides provides a representative small-break LOCA result for US APWR one square foot cold leg break, the limiting PCT

case. Following the break occurrence, the reactor coolant system pressure rapidly decreases and the natural circulation and loop shield phases are not found clearly.

Due to the rapid loss of RCS coolant, the cladding heatup starts around 100 seconds. That's almost the same time the advanced accumulator and high-head injection system start injecting the safety coolant and the PCT is suppressed below 1328 degree air.

The RCS inventory recovers by safety coolant injection and the core is quenched at approximately 400 seconds. It is noted that the figure for the ECCS flow indicates the total flow rate for the accumulator and HHIS high-head injection system respectively with the unit of pound per second.

Okay. I would like to summarize my presentation today. The US APWR small-break LOCA important phenomena have been identified and M-RELAP5 modeling capability has been assessed in conformance to EMDAP.

M-RELAP5 is capable of predicting all the high-ranked phenomena of small-break LOCAs, and

detailed information will be provided in tomorrow's presentation.

And now, M-RELAP5 is currently being applied to small-break LOCA safety analysis for US APWR reported in the design control document, Chapter 15. Thanks for your attention.

MEMBER BANERJEE: Have we got that previous slide you showed? This doesn't seem to be -- yes, it doesn't seem to be in my package. Is that just --

MEMBER REMPE: They were sent to us electronically.

MEMBER SHACK: It's in the electronic version.

MEMBER BANERJEE: Ah, it's in the electronic version. So, I don't have it here. Is that -- is that because it's proprietary or what?

MS. STEINMAN: No, that would have been a mistake if it's not in there. The intent was that it was to be included.

MEMBER BANERJEE: Oh, okay.

MEMBER BROWN: There's three or four slides that aren't included in the package here that they went through. So --

MEMBER BANERJEE: Make sure that we have a full package.

MR. SHUKLA: I'll do that.

MEMBER BANERJEE: Because otherwise we get mixed up. So the PCT here is referring -- can you walk us through this just once more?

MS. STEINMAN: Each slide, just describe.

MR. IKEDA: Okay. These figures show the reactor coolant system pressure and here, the occurrence of the break, and the break size is large, comparably large, so as RCS system pressure rapidly decrease.

And in the small-break case, I mean the comparably small-break cases, here the plant transits to the natural circulation. But in this case, in this limiting, one square foot break case, no clear natural circulation exists.

CHAIR STETKAR: This is a big small LOCA.

MR. IKEDA: Yes.

MEMBER BANERJEE: A foot square.

CHAIR STETKAR: Yes.

MEMBER BANERJEE: I mean it was a big --

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MR. IKEDA: Yes, and no clear loop seal formation in the clearing. And here is pressure level probably equivalent to the secondary pressure level, and -- but soon the loop seal clear and the RCS pressure decrease again soon.

MEMBER BANERJEE: So, how do you clear the loop seal in your -- is it by differential pressure or -- is there a specific module in your code which clears the loop seal or --

MR. IKEDA: No specific module.

MEMBER BANERJEE: It just clears it by pressure difference or --

MR. IKEDA: Yes, and the capability of -- predictability of the code was assessed and validated using the UPTF number 5 test data, and also, using IET, integral effect test obtained in the ROSA test facility.

Using the IET test data, we verified our code capability to predict the loop shield clearing. Unfortunately, for this case, no clear loop shield -- I'm sorry, no clear loop shield clearing here. But tomorrow -- tomorrow I will give the example, the transient including the loop shield formation and the clearing, exactly a 7.5-inch break. 7-5-

inch break cases provides a limiting loop shield PCT -- I'm sorry, loop shield clearing PCT.

MEMBER BANERJEE: So, you allowed the loop seals to clear by just the pressure difference across them?

MR. IKEDA: Yes.

MEMBER BANERJEE: So, whichever ones will clear, they will clear. It -- you don't, you do not do a conservative calculation which says that once a loop seal has cleared, only one will clear?

MR. IKEDA: Exactly --

MEMBER BANERJEE: And all of them clear.

MR. IKEDA: Exactly no conservative exemption is made for M-RELAP5.

MEMBER BANERJEE: And where do you go into refluxing? Do you have a refluxing situation?

MR. IKEDA: For the DCD calculation, the reflux condensation mode is not included, because in the US APWR design, the reflux condensation occur when the operator manually decreases the secondary pressure level.

MEMBER BANERJEE: So if you don't reduce the secondary pressure, you don't get into reflux condensation?

MR. IKEDA: Significant reflux condensation does not occur -- not occur if the manual secondary pressure reduction is activated.

MEMBER BANERJEE: If it is not.

MR. IKEDA: Not, yes, not.

MEMBER BANERJEE: So is that because the secondary system stays at high pressure and may even be a heat source, or --

MR. IKEDA: Yes, but --

MEMBER BANERJEE: A heat sink ever?

MR. IKEDA: Yes, that's right, about there -- better for the --

MR. KATAYAMA: I want to say the capacity of HHIS for US APWR is very large, and HHIS can suppress the PCT -- can suppress the PCT enough, and long-term cooling can be accomplished by HHIS.

And if operator decided to start shutdown mode, then the secondary pressure level is to be decreased by manually. During the manual operation, the PCT does not reheat up, and even if automatically the secondary pressure level is decreased, I think heat-up does not occur in such a case --

MEMBER BANERJEE: So, it would be

helpful for us to see, for some of these limiting breaks, what the pressure that you calculate on the secondary side and the primary side is, to understand, you know, the phases of the accident, because if the secondary side pressure becomes lower than the primary side, then it will be a heat sink and obviously the steam generator will act as a condenser, you know, and I'm not sure that I can --

CHAIR STETKAR: But I think what he is saying, for this size break, that's not going to happen.

MEMBER BANERJEE: Yes, but I'm saying that as you look at the spectrum of breaks, I don't know what the phenomenon looks like. I don't have a good feel for this plant so -- because you have no low pressure injection --

CHAIR STETKAR: I think you are asking probably down in the three or four inch size, right?

MEMBER BANERJEE: What other size do you have? By the way, the staff will come up, but I wanted to ask you, did the staff do any confirmatory calculations on this?

MR. TAKACS: Yes, we did do confirmatory calculations.

MEMBER BANERJEE: Over a range or break sizes?

MR. TAKACS: That is correct.

MEMBER BANERJEE: And will you be presenting some of those?

MR. TAKACS: Yes, we will be.

MEMBER BANERJEE: Okay. So we will catch up with you on that. Okay.

MR. TAKACS: Okay.

MR. IKEDA: Do you need additional explanation for the other figure?

MEMBER BANERJEE: Why don't you go through it, yes. So --

MR. IKEDA: Okay. Okay, this is RCS pressure and ECCS flow rate figure gives advanced accumulator flow rate. In total three advanced accumulator here, so not flow rate, not the flow rate are one ECC advanced accumulator, three accumulator flow rate, and high-head injection flow rate -- and okay.

When the RCS pressure decrease below approximately 600 pound, the advanced accumulator is activated and start injection. This is large flow rate model.

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And the flow rate is comparably lower compared with the case for large-break LOCA cases. But anyway, here the advanced accumulator flow rate model is changed -- to the small flow rate injection model.

And here the temporal degrees of RCS pressure appear and check valve is activated and advanced accumulator is tentatively stopped and restart again.

And at almost the same time, high-head injection system starts injecting the safety coolant like this.

MEMBER BANERJEE: So advanced -- the accumulator stops due to the check valve. That's by a pressure difference.

MR. IKEDA: Yes.

MEMBER BANERJEE: And then, due to the pressure difference changing at about, what is it, 250 or something, it starts to open the check valve automatically?

MR. IKEDA: Yes.

MEMBER BANERJEE: And it goes on injecting?

MR. IKEDA: Yes. In case of the

advanced case of the cold leg break. The accumulator flow rate motion is dependent on the break size and the break location.

And here is the --

MEMBER BANERJEE: So, what is the purpose of that check valve?

MR. IKEDA: To protect the accumulator.

MEMBER BANERJEE: From inflow?

MR. IKEDA: Yes, reverse flow. Reverse flow means from cold leg to the advanced accumulator tank.

MEMBER BANERJEE: Okay, let's go to the next slide -- the next panel.

MR. IKEDA: Okay, this is the liquid level for the core region and the upper plenum region. As you know, the large scale for the break is comparably larger so the RCS inventory rapidly decrease and also cold leg level is depressed rapidly.

And here, the time, ECC injection and advanced accumulator injection starts, and the liquid level certainly recovers like this.

And this figure indicates the RCS mass inventory, and similar to the core and upper plenum

liquid level, the -- by activating the advanced accumulator and high-head injection, the RCS inventory will recover like this.

This is the heat -- figure for the PCT and the cladding heatup. Here at approximately 100 seconds following the break, the cladding heatup starts like this, and also, the advanced accumulator injection starts. That's -- heatup keeps, because a large amount of core coolant is lost due to the large break size.

And that, the PCT occur along 160 seconds and it starts decreasing like this.

MEMBER BANERJEE: So if you go to the third plot, 160 seconds coincides to when the -- you start to recover the core -- the liquid level, is that it?

MR. IKEDA: Yes.

MEMBER BANERJEE: It's about there?

MR. IKEDA: Yes. It mostly coincides.

MEMBER BANERJEE: What is happening physically to change the slope that much there? I'm just trying to understand. It was -- is there some water coming from somewhere, the accumulator filled something up or what happened?

MR. IKEDA: I see. I estimate, I'm sorry, maybe the downcomer filled with the advanced accumulator and HHRS safety coolant. So --

MEMBER BANERJEE: It would be useful, at least, you know, for me, to try to understand --

MR. IKEDA: Okay.

MEMBER BANERJEE: how well the core does. I'm always skeptical about cores.

MR. IKEDA: Okay.

MEMBER BANERJEE: Because it has to be associated with some physical phenomena. So if you are filling the downcomer it makes sense, okay? So could you check whether that is due to the downcomer filling or not?

MR. IKEDA: Okay. The --

MEMBER BANERJEE: So actually the maximum in PCT depends on the -- how well you predict this filling, right? Okay.

MR. IKEDA: Anyway, the information about the other parameters, I think, are -- all the parameters you need are included in the DCD. But I will try to provide tomorrow.

MEMBER BANERJEE: Yes, it's hard for us to decipher, you know, all this stuff, so if you can

just give us -- you know, it's much easier if you can lead us through what's happening in the critical case, let's say, the worst case, and associate it with some physical phenomena, like this is filling, that's why the temperature turns around, or whatever is happening.

MR. IKEDA: Yes, I see.

MEMBER BANERJEE: And then you can see, you know, how well do you predict that filling, and what can be there. Okay. You have quite a bit of margin here, so --

MR. IKEDA: Yes.

MEMBER BANERJEE: not very sensitive.

MR. IKEDA: That's all. Any other comments?

MEMBER BANERJEE: And what was the maximum oxidation that you saw?

MR. IKEDA: Very low. Less than 0.2 percent, less than --

MEMBER BANERJEE: Even for the most critical --

MR. IKEDA: Yes.

MEMBER BANERJEE: Okay. So tomorrow we are getting loaded up with a lot of stuff.

CHAIR STETKAR: We may have to start at 3 o'clock in the morning. We'll see how we do. We'll see how -- we have to finish at 12 tomorrow. That's a firm time. We'll see how we get through this. This is our first introduction to this topic and it may not be our last.

Any other members' questions for MHI? If not, thank you, appreciate it, and we'll ask the staff to come up.

MEMBER BANERJEE: John, just a question for you. Are we expected to write an interim letter based on this, or what?

CHAIR STETKAR: We'll talk about that. We are writing interim letters. The next one will probably be in October. What we have been doing is bundling together chapters. Unless we determine that there is a fairly substantial issue and we can kind of move up the schedule and priority for those interim letters.

The simple answer to your question is yes we will.

MEMBER BANERJEE: Okay, yes, but in October?

CHAIR STETKAR: In October. And we have

some flexibility about those, especially because we won't be hearing about the transient topical report and the advanced accumulator until the October subcommittee meeting. We may not write about the LOCA indices in that October interim letter.

MEMBER BANERJEE: But you'll integrate everything --

CHAIR STETKAR: Yes. I mean we'll talk a little bit about the schedule later. Mike, it's all yours.

MR. TAKACS: Good morning Chairman Stetkar and Subcommittee members. My name is Mike Takacs. I am the chapter PM for transient and accident analysis. I have been involved in this chapter since May of 2008. The last time we had a similar meeting where we gave an informational briefing was back in February, 2009, in Pittsburgh, and that included all the topical report status.

Today, the staff is going to provide an overview of the safety evaluation for small-break LOCA methodology.

MEMBER BANERJEE: I remember we visited that facility, right?

MR. TAKACS: Yes we did. Several of us

were there with you.

MEMBER BANERJEE: Okay.

MR. TAKACS: With me today, some of the NRO staff, the technical staff here, presenters up front. Jeff Schmidt, Andrew Bielen will introduce themselves when they get there.

Jeff Ciocco, as you know, is the lead PM. Supporting this whole review throughout the time, the four-year period, Information System Laboratories is here as well, in the audience. Their names are up on the screen.

And with that, I am going to turn over the presentation of the technical substance to Jeff Schmidt.

MR. SCHMIDT: Hi, I'm Jeff Schmidt, and I was the lead reviewer for the small-break LOCA and I'm also the reactor systems lead for the APWR in general. Today we are going to talk about the small-break LOCA and another person who was key in the review of this is Andy Bielen.

A little bit about myself real quick. I was -- I'm a nuclear engineer from Iowa State University, undergrad, and graduate at University of Illinois, spent about three and a half years at

Westinghouse, the core design group, moved on to Salem nuclear power plant when they were taking on their own reload methodology, spent about seven years there, then moved on to Palo Verde where I spent the last 11 years, both as a senior engineer there in the reactor -- nuclear analysis and safety analysis group, and then the last six years there as the supervisor of that group. So that's a real quick history on myself. Andy.

MR. BIELEN: Hi, I'm Andy Bielen, like Jeff said. I have Bachelors degrees in mathematics and nuclear engineering and a Masters degree in nuclear engineering from Penn State.

I came to the NRC in 2008 and spent three years in reactor systems and currently I'm at the University of Michigan under Tom Downar and Annelisa Manera doing my PhD under Research sponsorship.

MR. SCHMIDT: Okay, so let's get into the staff review, the overview slide here. As MHI had mentioned, the M-RELAP model is based on the Idaho National Labs RELAP-3D code.

The methodology only applies as licensed for the US APWR because of the unique small-break

LOCA PCT response and we'll get into that. Sanjoy has kind of touched on that already. We'll have some slides to hopefully make Sanjoy happy.

MEMBER BANERJEE: Am I ever happy?

MR. SCHMIDT: Well, that's up to you, Sanjoy. We did a detailed confirmatory analysis. We looked at a lot of the issues Sanjoy kind of brought up already, both with RELAP-5 and the Mitsubishi code and RELAP-5. So we ran both codes. The RELAP-5 is the MOD3.3 version that NRC controls.

We found a number of significant issues during the review. We will go through some of those significant issues here. One of them, of course, was the critical flow model we'll get into. But there were some other --

MEMBER REMPE: Just out of curiosity, why did you decide to use RELAP-5 3.3 for your confirmatory analysis instead of something else, like TRACE?

MR. SCHMIDT: We had both RELAP-5 and TRACE running in parallel, one through Research and one through commercial contract, and when we got the TRACE results there were some modeling issues that really weren't addressed properly.

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So to fix those up would have caused a schedule impact. We had RELAP-5 MOD3.3 up and running and it was performing well, so to minimize the impact on the overall review schedule, we went with MOD3.3 but we did have both in parallel for a while.

MEMBER SHACK: Did you develop your own accumulator model?

MR. SCHMIDT: No. It's taken from the Mitsubishi model.

MEMBER BANERJEE: Does ISL have TRACE running right now?

MR. SCHMIDT: I guess I'm going to have to ask ISL if they do run TRACE, but I don't know --

MR. PRELEWICZ: This is Dan Prelewicz from ISL. We have run TRACE for a lot of other cases but we didn't run it for MHI for the US APWR.

MR. SCHMIDT: Yes, I know that but I wasn't sure if Sanjoy was asking just for US APWR or just in general. Okay. Next slide.

One outstanding issue we had, which we have talked about with Chairman Stetkar, is that the final accumulator bias is still to be determined. We did this review in parallel with the accumulator

model. At some point we thought the accumulator model was somewhat resolved and that we could take a scaling bias associated with it, put it in the small-break LOCA model and proceed from there.

But as we -- as we proceeded in the accumulator review we found that the scaling bias that was used in this topical report may not be the final value.

So there still is some uncertainty associated with the final bias that is going to be accounted due to the fact that there was half-scale testing and that we are trying to come up with a bias that would account for the full scale.

MEMBER BANERJEE: So when you say bias, it's some multiplier that reduces the flow?

MR. SCHMIDT: Yes, that's correct.

MEMBER BANERJEE: And the timing, it stays about the same for everything?

MR. SCHMIDT: The bias accounts is -- for their time zero.

MEMBER BANERJEE: Okay.

MR. SCHMIDT: And is always there, it's a constant.

MEMBER BANERJEE: Okay. Maybe at some

point, if you aren't going to explain it now, when we talk about the accumulator, we'll go through this bias and how it's being set to take account of scale effects.

MR. SCHMIDT: Okay. And the path forward is, once the accumulator SE is finalized and we come up with what the staff considers an adequate scaling bias, then we are going to revise the small-break LOCA topical report and revise the SE reflecting the final scaling bias and of course the Chapter 15.6.5 will be revised, showing the final PCT values.

But really, the small-break LOCA methodology, we feel, is fine. You know, it's been benchmarked, validated. But the final PCT values are still to be determined based on that accumulator scaling bias.

MEMBER BANERJEE: How significant is the reflector on the heat sink -- a heat source, sorry, in this case?

MR. SCHMIDT: The reflector on the heat source? It's pretty small compared to the decay heat. We did look at that. It's a very small contributor.

MEMBER BANERJEE: But this is a massive reflector.

MR. SCHMIDT: Yes, it is. But still, it's a massive core, it's 257 assemblies, 14 feet high. We did --

MEMBER BANERJEE: The other thing, two feet higher as well.

MR. SCHMIDT: Right, it's got a lot of decay heat.

MEMBER BANERJEE: Right. Yes. Okay.

MR. SCHMIDT: The decay heat is still down there. Next slide. Mitsubishi covered some of this already so I will go quickly through this, some of the design features of the APWR.

MEMBER BROWN: Can I ask a real quick question?

MR. SCHMIDT: Sure.

MEMBER BROWN: Okay, again, on the accumulator. You say you didn't -- you used their model for the accumulator.

MR. SCHMIDT: Right.

MEMBER BROWN: Why did you come to the conclusion that you were comfortable with their model as opposed to independently verifying that

they had a reasonable model that would predict its performance?

MR. SCHMIDT: Well I mean, we are really handling in that in the advanced accumulator topical report, right? The purpose of that report is to come up with something that is satisfactory to the staff in that advanced accumulator topical report.

If we find in that report for some reason that the correlations that were developed, were not applicable, we will have to come back and revisit this. But we weren't really -- we really considered that a separate item under the advanced accumulator, and it took this really as input for the small-break LOCA.

MEMBER BROWN: Okay, so you -- go ahead. I'm sorry.

CHAIR STETKAR: I was going to say, there are obviously some ends that need to be tied together. That's one of the benefits for the way that we have been conducting these reviews, is that this meeting gives us an opportunity to kind of weigh in on any issues, if you will, with the overall methodology.

The results, we will not see those

results until the final review of the advanced accumulator, the changes to the topical report and the changes to Chapter 15, when we do our final review of, you know, the SER with no open items, which will be some time not this year.

MEMBER BANERJEE: Is there a schedule in rough terms even?

CHAIR STETKAR: Yes, in rough terms. I don't know what that is, but I'm sure there is one. And I'm sure it will change. So it's sort of a moot point.

MEMBER BANERJEE: The question of when all this will come to a head, right, when you have no more open items?

MR. SCHMIDT: Right now I think the tentative schedule is March, March next year --

MEMBER BANERJEE: So that's fairly soon.

MR. SCHMIDT: for the advanced accumulator.

MEMBER BANERJEE: Okay.

CHAIR STETKAR: That's for the advanced accumulator.

MR. SCHMIDT: That's advanced accumulator, right.

CHAIR STETKAR: The SER on the advanced accumulator. The SER on the overall design certification with no open items is -- I'm sure the staff, I don't know if they want to comment, I'm sure they have a schedule. It changes periodically. It's probably not next year. Some of the chapters, I suspect, will be next year. But --

MS. SNYDER: But they just reissued a scheduling letter and I believe it's some time in 2015.

CHAIR STETKAR: '15, the final.

MEMBER BANERJEE: I was just wondering whether I'll still be here on this committee.

CHAIR STETKAR: If we can nail your feet to the floor, you will.

(Laughter)

MR. SCHMIDT: Okay, all right. Again, you know, it's a very similar design to the Westinghouse four-loop. The main bullet here is the increased high pressure injection flow, safety injection flow, on a megawatt basis and we gave a number here of 2.8 times at the runout conditions, and I'll show a plot later on of just how significant that increased high pressure safety

injection flow is.

Again, increased heat generators, MHI had already talked about that. They have a larger pressurizer, 14-foot active port, neutron reflector, as we know, and the core power is about 25 percent higher but it still has a lower average kilowatt per feet because of the large core, large number of assemblies and high height.

Next slide.

The methodology we talked about before, it's basically RELAP-3D, it's run in a 1D mode and Mitsubishi added the Appendix K features and the advanced accumulator model.

Again, as they mentioned, they followed Reg Guide 1.203, Transient and Accident Analysis. Again, the acceptance criteria, which we are all familiar with, that's 10 CFR, Appendix K, ECCS Evaluation Models.

Staff closely reviewed the 15.6.5, which is the sensitivity analysis, and that's documented in MUAP-07025 and we will talk a little bit about that, actually during this presentation on the break size sensitivity, there is an overlap.

MEMBER BANERJEE: So, the additional

features mainly were -- the Appendix K features were mainly heat transfer correlations that were made conforming to --

MR. SCHMIDT: Whatever made -- yes. For instance the decay heat right, heat transfer correlations, all the thing that make it acceptable for Appendix K. But they had it.

MEMBER REMPE: So 3.3 is older. There were a lot of updates to 3D. And so after MHI has modified 3D, and you were trying to do confirmatory calculations with an older code, don't you find corrections that were implemented causing differences, or have they improved 3.3 so that you don't notice any discrepancies?

MR. SCHMIDT: I mean, there are differences in the models. One of the questions we did ask was are all the error notices for RELAP-3D accounted for --

MEMBER REMPE: In this version of 3.3

MR. SCHMIDT: In this version of their 3D, which is the M-RELAP version if you will, the Mitsubishi version. And they did address all the known problems between the version that they took from Idaho National Labs and their final M-RELAP --

MEMBER REMPE: I'm not making my question very clear.

MR. SCHMIDT: Okay.

MEMBER REMPE: 3D has corrected things that were in 3.3, just over the years.

MR. SCHMIDT: Oh, I see.

MEMBER REMPE: What I'm asking is, were the differences in the ISL NRC 3.3 version, were those corrections implemented when you compared it to the Mitsubishi RELAP-3D version?

MR. SCHMIDT: Right, right.

MEMBER REMPE: And were they?

MR. SCHMIDT: I'm going to have to ask Dan, who keeps track of all the changes to MOD3.3.

MR. PRELEWICZ: It's Dan Prelewicz from ISL. We have been exchanging error reports for the last several years with INL. So we, if there is something that has been corrected in 3D, we correct it in 3.3 and vice versa.

MEMBER REMPE: So it's not just the 3.3 that is referenced in the NUREG that I found in your topical report? It's an improved version of 3.3?

MR. PRELEWICZ: Well, it's the version of these, the NRC version of 3.3. We -- again, we

have been correcting the errors, so there are --

MEMBER REMPE: Okay.

MR. PRELEWICZ: The differences are -- no significant differences in terms of errors.

MEMBER REMPE: Okay.

MR. PRELEWICZ: The models might be different but any errors that were found in either code, are corrected in both codes.

MEMBER REMPE: Okay.

MR. SCHMIDT: Thanks, Dan. Okay, next slide. The acceptance criteria, again, just a repeat of what MHI had said, PCT below 2200, local maximum cladding thickness less than 0.17, core wide oxidation less than one percent and then the core remains in a coolable geometry and assurance of long-term decay heat removal.

Again, we really focused on M-RELAP and small-break LOCA on the first three bullets.

The PIRT process, as MHI described, broken down into the five phases: blowdown phase; natural circulation; loop seal clearance; boiloff; and core recovery. We agreed with those basic five steps or stages of the small-break LOCA.

The model assessment cases as a part of

the validation were selected to evaluate the high-ranking phenomena for each transient phase. And the staff agreed that the thigh-ranked phenomena had been correctly identified in the PIRT.

MEMBER BANERJEE: So, refluxing was not considered high-ranking phenomena?

MR. SCHMIDT: I don't remember of the reflux was in the PIRT on a high-ranking phenomenon. I'd have to get back to you. I don't know if that was a high-ranking parameter.

We did add assessment cases to address reflux which you actually see on the next slide.

MEMBER BANERJEE: Okay.

MR. SCHMIDT: But I can't for sure say off the top of my head if it was a --

MEMBER BANERJEE: It's just a matter of the design, as to how significant the heat sink stimulator can be.

MR. SCHMIDT: Yes, we did looking into the CCFL correlations fairly heavily as part of this review. And you know, we found that the UPTF data for the 15-bar and the 3-bar were conservative for CCFL correlations.

MEMBER BANERJEE: Okay, we'll get into

that.

MR. SCHMIDT: Yes. Yes. The staff did ask questions on the validation test matrix. The staff requested that larger break sizes be evaluated. Originally the Rev 0 of the topical repoirt had a five percent break size, which you know, covered both loop seal and core boiloff, but because of the nature of this plant and its systems, we thought that larger break sizes should also be evaluated, so we requested that 10 percent and 17 percent cases for the ROSA be added.

Also, the TMI action items requests require that LOFT 3.1 and Semiscale be added as part of the validation matrix so MHI did that.

And then we looked at some of the reflood heat transfer as a separate effects test -- because of the large-break nature we wanted to make sure what the heat transfer coefficients looked like under the large-break regimes.

And through all the validation cases, these are -- what's listed here are just the ones the staff, MHI agreed to add. The M-RELAP demonstrated the ability to represent the test data fairly well.

Go ahead. Next slide. One of the unique features of this plant, as we talked about earlier, is the higher -- high pressure injection flow rates. You know, one of the first things that caught our attention was there was no typical heatup in the two to four inch break of a typical PWR, a Westinghouse PWR.

Like I said, we asked that validation case be added to address the larger break sizes. We also asked additional cases for the loop seal clearing phenomenon, and then an overall explanation of why we don't see the heatups in the two to four inch range, and why we do see the large heatups in what we consider a fairly large, small break.

So we kind of grouped -- had a lot of questions in that regard, both small breaks, loop seals and the large break phenomenon. Next slide.

Here is a -- this is a confirmatory analysis where the staff artificially stopped loop seals from clearing. You can see the circles are all loop seals clear, and then as you move up to the squares and the diamonds, we are letting less loop seals clear.

And the purpose of this --

MEMBER BANERJEE: This is what I was looking for.

MR. SCHMIDT: Yes, I thought it might be. What we were trying to gauge here is, because we didn't see the heatup in the lower break ranges that we are used to seeing, we wanted to understand why.

So we stopped, we artificially stopped the loop seals from clearing, and even with, you know, the two loop seals, you know, can clear, case, we still didn't see a heatup in the low break sizes.

And the real reason for that is the high pressure safety injection flow is just so great it can make up for breaks up to six inches, and that's really why you don't see the two to four inch break in it, but I'll have some followup slides which will further describe this too.

But, so this was kind of what --

MEMBER BANERJEE: Just the -- that you have more flow is that it?

MR. SCHMIDT: Yes, that's it.

MEMBER BANERJEE: Because this is very unusual results.

MR. SCHMIDT: Yes.

MEMBER BANERJEE: Yes.

MR. SCHMIDT: Yes, no, we agreed -- it was the first thing we noticed. It was very unusual, and actually if you go to the next slide, there's your flow curves.

What we did is we compared the US APWR with two pumps running with the, you know, the minimum safeguards flow rates, to a Seabrook case that was the flow is scaled up to like a per megawatt basis.

And you can still see, even after Seabrook is scaled up, there's significantly higher flow in the MHI design than say a scaled up Seabrook case. You know, if RCS pressure typically hangs out for a lot of these small breaks around 1,000 psia, you know, you are over twice the flow of these pumps than you would, in, say a conventional four-loop plant. So this went a long way to understanding the break size behavior. Next slide.

Now we did the opposite. What we did is say let's take the MHI plant and scale down their flow to what the Seabrook scaled value would be on a per megawatt basis, to see if we see the typical response to a small-break LOCA.

And you can see the blue line there, the diamond line, you do see the peak that you typically see with the Westinghouse plant in the two to four inch range when the HPI flow is significantly reduced.

MEMBER BANERJEE: How many loop seals cleared here?

MR. SCHMIDT: Let's see.

MEMBER BANERJEE: All of them?

MR. SCHMIDT: I don't -- all allowed, I believe. We never -- we didn't artificially control the loop seal clearing in this case.

MEMBER BANERJEE: On this one.

MR. SCHMIDT: Yes. Right. Only the one where we were trying to understand the break size behavior initially. So this would just be whatever the model represents, right?

The other thing we had looked at was the green curve here, which isn't that significant. We were also looking at, as part of he understanding of the break size phenomena, we looked at steam generator size too, because it's bigger than the model on a four-loop Westinghouse plant.

So we were looking at both parameters,

but, you know, as the graph shows, by and large it's really the -- it's the increased safety injection flow that really makes the transient behave different --

MEMBER BANERJEE: But you used steam generator size in the four-inch case, gives you quite a bit lower peak clad temperature, right?

MR. SCHMIDT: Quite a bit -- well, I mean that's with -- that's only changing the steam generator. It doesn't have the reduced flow.

MEMBER BANERJEE: Oh, okay.

MR. SCHMIDT: That's the difference. So they are two separate cases. So the green would be the MHI flow rates with just the reduced steam generator.

MEMBER BANERJEE: Yes, so you are getting the effect of the reduced -- sorry, I got mixed up.

MR. SCHMIDT: Oh, that's fine. That's just -- I probably didn't do a very good job of --

MEMBER BANERJEE: No, but it's there in the caption. I just didn't read it correctly. Go ahead.

MR. SCHMIDT: Okay. Next slide. Some

of the other issues that the staff found that are in the review was the critical flow model that was being used.

MHI was using the Moody correlation and the pull through model and there were issues associated with the modeling there, associated with the stub pipe, which is a connection between the cold leg and the containment basically.

I don't want to get too much into the modeling of that, because I think it's considered proprietary, so we can talk more about that tomorrow. But the general gist was the staff was finding, as we varied the stub pipe size and dimensions, that we were seeing vastly different PCT responses, and I have a curve that will show that.

And that kind of led the staff to start to question things like the pull through model and the break flow correlation an its uses.

So, let's go to the next slide and we'll take a look at -- confirmatory calculations show that the Moody was not consistently applied when two phases were present as required by Appendix K, and there were some issues with the pull through model switching logic that caused numerical oscillations.

MHI corrected these issues and effectively eliminated the strong dependence on the PCT response to the actual size of this stub pipe or this phantom pipe.

And as MHI described, the bottom leg break became the limiting break, which is consistent with other PWRs. And I think the next slide probably shows the real interesting portion of it.

The black line there is this PCT versus time for the limiting case, the one square foot break case, and it's with the two versions of the code -- 1.4, which had some of the issue with the Moody break correlation and the pull through model, and then the final version, which is version 1.6.

So, the version 1.6 is the DCD value and you can see that with the stub pipe -- the stub pipe sensitivities, we plotted here the max and the min PCTs.

So we looked at a range of different stub pipe sizes, and since it's an imaginary pipe, it can take on a variety of imaginary dimensions. So what we did here is, this is a case where 1.4, which is the green and the red line, you see a huge different in PCT based on the stub pipe dimensions.

Now, granted, this is the worst case. But you can see why the staff was concerned and had a lot of followup questions in this.

MEMBER SHACK: Is this model something that MHI did to RELAP, or is this something that occurred because you are now using RELAP and forcing it to do Appendix K-type calculations?

MR. BIELEN: We should probably -- they're very protective of their modeling choices as far as this goes, so I think we should probably defer that to either closed or tomorrow.

MR. SCHMIDT: Yes, I think that would be a good idea. I'd probably go to --

MEMBER BANERJEE: Are we going to have a closed session, John?

CHAIR STETKAR: Yes.

MEMBER BANERJEE: Okay. So let's guide this as a --

CHAIR STETKAR: We'll see how we go today. I mean, if things -- if we get through material in a timely fashion today, we can close part of the session later this afternoon.

MEMBER BANERJEE: It would be interesting to know more about this.

CHAIR STETKAR: We don't want to let too many things pile up until tomorrow because of the schedule problems we have.

MEMBER BANERJEE: Okay.

MR. SCHMIDT: Next. This is a plot the staff put together really showing -- and the red there is the Henry Fauske correlation and the black is the Moody correlation and you can see that the code was switching back and forth between correlations and that caused some of the PCT response you saw versus time. This is just to illustrate that, that it was a switching problem back and forth between the two break flows.

The other major issue, or another major issue we had was this --

MEMBER BANERJEE: How was this resolved finally?

MR. SCHMIDT: It went to Moody only.

MEMBER BANERJEE: And for the -- was there a period of subcooled discharge? Did they use Henry Fauske at that point?

MR. SCHMIDT: They used whatever was the worst case in the part of the -- during part of the transient. The logic would pick whatever the worst

case scenario would be.

MEMBER BANERJEE: I'm just trying to imagine. You've got a hole, let's say, in the cold leg. So for a period this is going to be sub-cooled and they, presumably if we are using Henry Fauske, I don't know, maybe we should ask --

MR. SCHMIDT: If you go back to the previous slide there, you'll see, let's see --

MEMBER BANERJEE: It's not clear what they are doing here.

MR. SCHMIDT: There is --

MEMBER BANERJEE: The M flow J --

MR. SCHMIDT: Oh, actually this is the -- this is the uncorrected version, so you are really asking about the corrected version.

MEMBER BANERJEE: So, M flow J is some model of their own?

MR. SCHMIDT: No, it's just a variable that we are -- it's kind of a time average variable of what both correlations are doing.

MEMBER BANERJEE: Oh, okay.

MR. SCHMIDT: Perhaps maybe Dan or ISL can better answer Sanjoy's question on the --

MEMBER BANERJEE: So, during sub-cool

discharge, what's being used?

MR. PRELEWICZ: Not sure I exactly understand the question. There was a switching logic problem where instead of -- whenever the -- Appendix K requires whenever it's two-phase, you are supposed to use the Moody model.

MEMBER BANERJEE: Right, I'm talking about sub-cooled. What did you use during sub-cooled?

MR. PRELEWICZ: Well, the sub-cooled, they can use the Henry Fauske, yes, when it's sub=cooled.

MEMBER BANERJEE: Yes, so that's what I was saying. Did they use Henry Fauske during sub-cooled?

MR. PRELEWICZ: Yes.

MEMBER BANERJEE: And then switched to Moody when it was two-phase.

MR. PRELEWICZ: Well, that's what it does now. Before, there were some cases when it was two-phase and it was still using Henry Fauske.

MEMBER BANERJEE: Ah, I get the problem now. Okay.

MR. PRELEWICZ: And it was switching

back and forth --

MEMBER BANERJEE: Between the two --

MR. PRELEWICZ: during the event.

MEMBER BANERJEE: But why was it doing that? Was that just a cooling error or --

MR. PRELEWICZ: Well, the switching -- the switching logic wasn't based on two phase. It was phased on -- if you have non-equilibrium, there was two phases there. But if you looked, you mixed them all together, you would be sub-cooled. So they had like sub-cooled voids and that sort of thing in there that was causing it to switch back to the Henry Fauske --

MEMBER BANERJEE: So it was taking a quality based on averaging between the two phases, and they were in non-equilibrium, so the water was sub-cooled and the steam could be superheated or whatever.

MR. PRELEWICZ: Yes, but there was steam there.

MEMBER BANERJEE: Yes, okay.

MR. PRELEWICZ: And it was supposed to be using Moody but it wasn't.

MEMBER BANERJEE: Right.

MR. PRELEWICZ: As you can see, it was switching back and forth. Wherever the red, it was using -- on this curve, it was still way down in, it was using the Henry Fauske where it should have been using Moody, should have been all the black color. The red showed up. That means it was using the wrong critical flow correlation.

MEMBER BANERJEE: So, rather than using an average quality it should use a void fraction to switch or something?

MR. PRELEWICZ: We did a lot of testing. We looked at things like the void fraction, whenever there was any voids we should use Moody. I don't remember exactly how -- what criteria it wound up in. It's now, both codes have a criteria that uses only the Moody. You have the same problem with 3.3 --

MEMBER BANERJEE: Henry Fauske basically goes to homogeneous, doesn't it, at two-phase flows? I mean, there is no sub-cooling what happens to Henry Fauske? Remind me.

MR. PRELEWICZ: When there is no sub-cooling?

MEMBER BANERJEE: Yes.

MR. PRELEWICZ: Well --

MEMBER BANERJEE: You know the Henry Fauske correlation, where does it go? It actually applies, even in two-phase flow, doesn't it?

MR. PRELEWICZ: Well, the code was calculating Henry Fauske in two phase, that's correct, and I don't know, it may have been used again improperly. I'm not sure. But it was calculating using Henry Fauske in two phase.

MEMBER BANERJEE: It should go -- it should go into two phase as well, but I'll look it up myself. Don't worry. Okay. Go ahead.

MR. SCHMIDT: The other issue, as I started to mention, was there was no detailed scaling evaluation in the original Rev 0 of the topical report. There was some comparisons and some verbiage regarding four-loop -- a comparison to a four-loop Westinghouse plant.

The staff had questions about the scaling to a four-loop plant because of the size differences, and requested that a quantitative scaling analysis be performed.

Initial scaling analysis was limited to a single facility and a single test and did not

cover the range of phenomena. MHI provided a final scale and report with multiple ROSA cases.

The top-down scaling demonstrated that the relative rankings of the non-dimensional coefficients between the plant and the test were preserved.

No new or different system interactions from those exhibited in the test facility, and the same dominant processes and phenomena occur in the US APWR and the integral test experiments that were used for validation.

Experimental data are acceptable for validation of the M-RELAP5 code, and there were some minor scaling distortions identified in the top-down analysis were evaluated in the bottom-up --

MEMBER BANERJEE: So what were these scaling distortions? Do you remember?

MR. SCHMIDT: No, I'm going to turn to Marcos for hopefully remembering what the distortions were.

MEMBER BANERJEE: You have to come to the mic, Marcos.

MR. ORTIZ: This is Marcos Ortiz from ISL. I don't exactly remember which ones. What I

remember that we did was, we looked at the coefficiency questions and made sure that they were in the same magnitude, and the past the couple of more important ones, in the end there were variants, and then we checked those.

But they were in the non-important --

MEMBER BANERJEE: So, let me try to understand this. We are trying to look at the applicability of the code to this system, which has quite a lot of unusual attributes, for example the accumulators and the lack of a low-pressure injection system

I would have expected significant distortions in the top-down scaling because of these -- the flows are so different and everything else. And you know there would be== it's just an unexpected result to think that the distortions are small, in a case where you've got three times as much HP flow.

MR. ORTIZ: Well, the accumulator was not in the part that I reviewed. It was -- that was a separate --

MEMBER BANERJEE: Yes, the real question to me is do the ROSA integral effect tests provide

an adequate database with the accumulators being a purely separate effects test. There is no integrated test done with the accumulators, right?

MR. ORTIZ: Yes. That's correct.

MEMBER BANERJEE: So I would expect large distortions.

MR. ORTIZ: They were not in the analysis that I reviewed, you know, the --

MEMBER BANERJEE: But you did it without the accumulators, right?

MR. ORTIZ: Well, yes, at such, yes.

MEMBER BANERJEE: So to me that has always been an open question and I should say this, that it could be that -- thanks Marcos, but maybe you can answer this. But it can be that we can have a bunch of integral tests, which are the ROSA, or whatever else the integral tests there are, then you have te separate effects tests with the accumulators. We had no integral effects tests with the accumulators. So there seems, at some point, a fairly -- you are in some area of scaling where there isn't a large database available to hang your hat on.

It doesn't mean that the results don't

hold, but I would expect there would be distortions. That's why I was startled by this.

MR. SCHMIDT: In some sense, you know, during the large-flow portion of the accumulator, it acts just like a regular accumulator.

MEMBER BANERJEE: Right.

MR. SCHMIDT: You know, and in that phase, it's -- and most of these --

MEMBER BANERJEE: So you think you can appeal to the database there?

MR. SCHMIDT: Right. Right.

MEMBER BANERJEE: Okay.

MR. SCHMIDT: And most, for instance, like the limiting case, the one square foot break case, you know most of that flow, as MHI pointed out, is during the large-flow phase of the accumulator.

So, again, it's acting like a regular accumulator at that point.

MEMBER BANERJEE: Yes, the thing that was sort of different, is where you expected to see the peaks in the Westinghouse-type plants, which is the four-inch breaks. You don't' see the breaks.

MR. SCHMIDT: Right, right.

MEMBER BANERJEE: You see quite different sort of phenomena occurring with regard to PCT. You are getting no peaks and then you are getting something at a larger break size.

So -- and you know, the Westinghouse plants have been well scaled against experimental data, the codes have been validated etcetera. It's just that here it seems you are seeing different things, different phenomena.

MR. SCHMIDT: I would say it's probably not different phenomena. I would say it's due to a plant design difference that causes the difference in behavior, not the phenomenon, but the plant.

MEMBER BANERJEE: Yes, I agree. But the integral effects tests that were used to validate the codes and so on, and show the applicability of the codes to the SBLOCAs for these Westinghouse plants, you know, the concern is, are these set of integral tests sufficient? Should there have been a set down with, you know --

MR. SCHMIDT: Right, an integral system basically --

MEMBER BANERJEE: Yes, an integral system. It seems to be missing.

MR. SCHMIDT: You know, obviously there weren't.

MEMBER BANERJEE: Yes.

MR. SCHMIDT: And again, I think the only thing I can really say is that in most cases, this accumulator acts like a normal accumulator for these --

MEMBER BANERJEE: Well, I can read that, yes.

MR. SCHMIDT: break sizes. So I mean, that's really, I think, one of the reasons why we are saying that the ROSA tests are still valid for the APWR.

MR. BIELEN: And they also did a ROSA test recently, in 2009, for the 17 percent equivalent break to try to look at APWR-specific --

MEMBER BANERJEE: Ah, okay. That's more interesting.

MR. SCHMIDT: But it was still with a traditional accumulator --

MR. BIELEN: It was still with the --

(Simultaneous speaking.)

MR. BIELEN: -- but at least you are still looking for the same kind of thermo-hydraulic

phenomena that you would expect in the US APWR specifically.

MR. BIELEN: But they added the 17 percent --

MR. SCHMIDT: Right. That matched up with the one square foot limiting break here.

MEMBER BANERJEE: Because when you look at that nice curve you showed with the loop seals not being allowed to clear, you know, it's sort of an interesting curve, because --

MR. BIELEN: Agreed.

MEMBER BANERJEE: Right. Well this is all very interesting, because you are seeing sort of different phenomena because in the small-break business, you know, we look and you are getting this sort of -- but you see a big separation by the number of loops he's clearing as you get to the larger breaks.

MR. ORTIZ: I -- the different interactions. The interactions are delayed. There's a little different -- the phenomena are still the same though.

MEMBER BANERJEE: I believe the phenomena are the same, but the pie groups are

supposed to be measures of well your experiment applies to your full-scale system, right?

MR. ORTIZ: Yes. Yes. And they were quite similar, but -- and more importantly, they kept the same relative ranking.

MEMBER BANERJEE: Well, if you didn't specifically look at accumulator effects, then --

MR. ORTIZ: Well, it's a separate thing, but if the flow was there, the numbers were there, so the parameters were there, and the pies accounted for whatever they --

MEMBER BANERJEE: So you took the flows from the accumulators in calculating your pie groups, or how did you do it?

MR. ORTIZ: I reviewed what they had in terms of those pie numbers. It was not like our AP600 deal where I did the whole thing. I reviewed what they had.

MR. SCHMIDT: But it has the accumulator flow --

MR. ORTIZ: It has the numbers in there.

MEMBER BANERJEE: It has the numbers. Okay. All right. Well, there is a report on this, right?

MR. SCHMIDT: Yes, there is a separate scaling --

MEMBER BANERJEE: Scaling report.

MR. SCHMIDT: Yes.

MEMBER BANERJEE: And we have access to it, clearly.

MR. SCHMIDT: Yes.

MEMBER BANERJEE: Girija, let me take a quick look at that at some point. Okay. Thanks. It's an MHI report.

MR. SCHMIDT: And I looked up while you guys were speaking, one of the distortions was in a volumetric flow out the break, was one of the distortions that was looked at from a bottom-up perspective too.

MEMBER BANERJEE: Yes, the distortion that would be here is the break flow to the accumulator flow, or the break flow to the HPI flow. There would have to be some significant distortions there. But I'll take a quick look at it. Okay, go ahead.

MR. SCHMIDT: Okay. Next slide please. Conclusions. MHI small-break LOCA methodology is acceptable for calculating PCT, hydrogen generation

and fuel cladding surface oxide for the US APWR with the appropriate Appendix K evaluation models.

What we listed there are the preliminary or I would say today's PCT values. Again, as we discussed, the accumulator scaling bias is not final, it's not been determined. But there is significant margin to the 2200 here.

MEMBER BANERJEE: Unless only one loop seal clears.

MR. SCHMIDT: Unless only one loop seal clears. That's right, which for these large breaks, we anticipate --

MEMBER BANERJEE: Yes, these are fairly big breaks.

MR. SCHMIDT: These are big small breaks, right, because of the high pressure injection. So as MHI discussed, you know, there were no artificial means of controlling the loop seals if you well, so -- and the reason why staff found that acceptable is because of the HPI flow, that limiting breaks are big breaks, and you would expect the loop seals --

MEMBER BANERJEE: Yes, I think that's the argument, because the staff, for example, but

EPUs really for the small breaks, as required that you keep the loops seals, only let one clear.

MR. SCHMIDT: Only want one clear. Right, right. We do not have the requirement here.

MEMBER BANERJEE: Yes, but that was -- that's in the smaller sizes.

MR. SCHMIDT: Right, right.

MEMBER BANERJEE: So, we need to think through that logic, what, you know --

MR. SCHMIDT: We did consider that quite a bit --

MEMBER BANERJEE: Right.

MR. SCHMIDT: in the review, and that was one of the reasons why, our first curve, we were examining, you know, as we've artificially closed off loops, we've stopped them from clearing, we wanted to see the response to see if it did push it down to the four-inch and then we were going to reconsider whether we had to add some artefacts to the loop seal clearing.

MEMBER BANERJEE: Okay. All right.

MR. SCHMIDT: And this is, the final slide here, is just to give an idea again on break spectrum versus the staff's RELAP model. Again you

see a very good match up of behavior between the two models, again, this is with their final corrected version 1.6.

As you can see, you know, again, you don't see the heatups on the small breaks. You see a small increase in the MHI model on 7.5 inch loop seals clearing break, and then again as you move up in break size the PCT just goes up.

MEMBER BANERJEE: And this is with two trains of high pressure?

MR. SCHMIDT: Yes. Okay. All right. Well, thank you very much.

MEMBER BANERJEE: Interesting.

CHAIR STETKAR: Any other questions or comments for the staff? If not, thank you very much. Very good presentations.

MR. SCHMIDT: Thank you.

CHAIR STETKAR: You guys did really well.

MEMBER SHACK: These background slides you have, you didn't go through.

MR. TAKACS: Some of them may be proprietary.

CHAIR STETKAR: Unless Dr. Shack really

wants to hold us here, we will recess until 10:25.

MEMBER BANERJEE: Well, if we are having a proprietary session, we probably should.

CHAIR STETKAR: Well, what I want to do is, let's see how we get through this stuff. I don't want to push too much stuff until tomorrow because otherwise --

MEMBER BANERJEE: Well, if we don't have to write a letter, we can always --

CHAIR STETKAR: We don't have to write a letter, but I also don't want to -- if any issues come up regarding the proprietary information, I want to make sure we have enough time to get them out on the table, either today or tomorrow, by the time, you know, by the time we finish tomorrow noon time. So --

MEMBER BANERJEE: Well, what is happening afternoon, the P&P?

CHAIR STETKAR: No, well, at noon we have got PMP and tomorrow afternoon there are two other subcommittee meetings, Seabrook and --

MEMBER BANERJEE: Oh, right.

CHAIR STETKAR: And flooding. So we can't run into tomorrow afternoon and we can't push

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the P&P because we are backing up against subcommittee meetings, so we are --

MEMBER BANERJEE: Yes sir.

CHAIR STETKAR: We are stuck. We originally had this scheduled for two full days and we had to -- I got negotiated back to a day and a half.

MEMBER BANERJEE: I'm sure you are a good negotiator John?

CHAIR STETKAR: Not too well on this one, but at least I held it to days where you didn't have to come back from Switzerland.

(Whereupon, the meeting went off the record at 10:10 a.m. and resumed at 10:26 a.m.)

CHAIR STETKAR: Okay, let's reconvene and we'll hear from Mitsubishi about the large LOCA methodology. Dr. Banerjee will join us.

MR. TERAMAE: Thank you, Chairman. Good morning everyone my name is Tetsuya Teramae. I am working for the Mitsubishi Heavy Industries.

This morning I am talking about the large-break LOCA code applicability report for US APWR. MHI has committed to the NRC to supplement its US APWR design certification application.

And today's -- this shows that today's presenters, I will be supported today by my co-presenter, Ms. Narutomi and as well as MHI staff.

Here are the contents of our presentation today. We will be presenting the LBLOCA code and methodology. I will provide an overview of the topical report and then I'm going to present some of sample plant analysis with analysis and result, with WCOBRA/TRAC M1.0 code. Finally, I will summarize our presentation.

So let me start with a brief overview that highlights the main aspects of the topical report. The objective of the topical report is to present a comprehensive assessment of the applicability of WCOBRA/TRAC M1.0 code with ASTRUM methodology to US-APWR LBLOCA analysis.

The key topics in the topical reports are, first, the US APWR design features to be evaluated for the code applicability have been identified and discussed.

Second, the applicability of the code and the methodology of -- to US APWR has been examined and confirmed, based on the code scaling applicability and uncertainty evaluation

methodology.

In relationship with the design certification document, design control document, DCD, WCOBRA/TRAC M1.0 with ASTRUM methodology is being applied to the LBLOCA analyses for US APWR design certification. The topical report is referenced in section 15.6.5 of the US APWR DCD.

I'd like to begin with slide 6 plus, and then I'm going to go back slide 5 and then slide 4. Sorry for this inconvenience but I think the explanation will be more clear in this order.

Let's show page 6 of the slides, it starts applicability of the code and methodology. The applicability of the code and the evaluation methodology to the US APWR has been examined and confirmed based on the code scaling, applicability, and the uncertainty evaluation methodology.

That means it has been considered under the following five main viewpoints: for the US APWR PIRT, the modeling requirements for performing LBLOCA calculations for the US APWR were identified and discussed.

And at the same time, the features of the US APWR that affect the applicability of the

existing PIRT were specified and estimated.

In the place of the code applicability determination, the applicability and assessment for features of the US APWR was examined based on the PIRT result.

The nodalization for the US APWR has been developed based on the same scheme approved for current three- and four-loop PWRs in the U.S.

Then, the demonstration for typical LBLOCA characterization with WCOBRA/TRAC M1.0 has been provided.

For the statistical analysis, the uncertainty treatment for new or improved features and other plant-specific parameters of the US APWR are determined.

MEMBER REMPE: I have a question on the previous slide. When you talk about he same scheme for nodalization approved for current three- and four-loop PWRs, do you really mean what you did was follow recommendations to come up with your nodalization? Or what -- could you elaborate what you mean by that?

(Off the record comments)

MR. TERAMAE: Nodalization is originally

based on the approved methodology, ASTRUM, developed by the Westinghouse methodologies, and --

MEMBER REMPE: So it was an ASTRUM-approved methodology.

MR. TERAMAE: Yes. That methodology has nodalization scheme, basically, and we, based on the same scheme nodalization.

MEMBER REMPE: Okay. And they gave you guidelines, perhaps, on how big of a volume you might use, or something like that, or how did you adapt it to your plan? What kind of guidance?

(Off the record comments)

MR. TERAMAE: The original scheme basically depends on the loops, number of loops. For example the four-loop plant, the downcomer is divided from the four divided channel.

MEMBER REMPE: So you had, like, maybe in some cases a larger volume but you used the same number of --

MR. TERAMAE: Yes.

MEMBER REMPE: nodes. For the core height, for example, you might have this 10 or something still, even though the plant has a larger core?

MR. TERAMAE: Core height, core height divides based on the grid, portion of the grid, grid spacer. And we refer to grid and a half, grid to grid on the series methodology of that main.

MEMBER REMPE: Okay, thanks.

MR. TERAMAE: Therefore the longer core are only the ones where stability is increased.

MEMBER REMPE: Okay, thank you.

MR. TERAMAE: Can I move on?

MEMBER REMPE: Yes.

MR. TERAMAE: Back to the slide number 5. The applicability of WCOBRA/TRAC M1.0 to the US APWR has been evaluated and confirmed especially respective to simulating high-ranked LBLOCA phenomena and new or improved design features, such as: longer core; advanced accumulator; and direct vessel injection; and neutron reflector.

This slide lists computer code and uncertainty evaluation methodology, using the best estimate LBLOCA analysis for the US APWR.

The WCOBRA/TRAC M1.0 code is MHI's version of WCOBRA/TRAC which incorporates the US APWR design features of the advanced accumulator and neutron reflector, and it calculates the thermo-

hydraulic behavior during the LBLOCA.

The HOTSPOT code calculates the fuel rod behavior, such as the cladding burst, metal-water reaction, and fuel relocation following burst phenomena, using the same model, same fuel rod model as that using the WCOBRA/TRAC. The code can calculate the effect of the uncertainties.

ASTRUM, standing for automated statistical treatment of uncertainty method, is a non-parametric order statistical methodology that does not assume the peak cladding temperature distribution.

This statistical method determines that 124 cases must be run in order to assure the 95/95 level, peak cladding temperature, local maximum oxidation and core-wide oxidation.

Next slide will provide some result of the DCD section 15.6.5 reference case as a sample. Major LOCA parameters for WCOBRA/TRAC M1.0 are listed in the table 3.6-5 in the topical report.

The value of statistical parameters are basically nominal for the reference cases. As I said before, the nodalization was developed based on the same scheme as approved for current four-loop

PWRs.

This slide presents some of the results of sample plant analysis for the double-ended guillotine break case, with discharge question to 1.0.

The left side of the figures on the slide show the transient of the core flow rate and the pressure, core pressure, during the blowdown phase.

The core flow rate that changed from the upward flow to downward flow, changed in about 10 seconds after the break.

MEMBER BANERJEE: What pump model was used here?

MR. TERAMAE: What pump model?

MEMBER BANERJEE: Pump. Pump. For the homologous curves for the pump --

MR. TERAMAE: Yes.

MEMBER BANERJEE: What did you use?

MR. TERAMAE: Yes.

MEMBER BANERJEE: Did you do experiments or --

MR. TERAMAE: Yes. Single phase homologous obtains based on the flow data.

MEMBER BANERJEE: What scale? What scale was the data?

MR. TERAMAE: One third.

MEMBER BANERJEE: And did you do any two phase tests?

MR. TERAMAE: Two phase data is not obtained MHI, but US APWR plant and current operating Westinghouse plant are all near very similar to the pumps, and very near value of the specific feed.

Therefore we -- MHI decided 93A, two-phase data adopted to the US APWR two-phase data.

MEMBER BANERJEE: So you used the similar curves to Westinghouse or --

MR. TERAMAE: Yes.

MEMBER BANERJEE: what did you -- yes. So are your pumps similar? It seems bigger.

MR. TERAMAE: Bigger. It was just bigger.

MEMBER BANERJEE: Just bigger. And did you, did you assess the size effect on your two-phase flow behavior, the curve pumps?

MR. TERAMAE: It's --

MEMBER BANERJEE: How much bigger?

(Pause)

(Off the record comments)

MR. TERAMAE: I -- we evaluated the -- used the based on the 93, based to two-phase data, to the US APWR two-phase -- US APWR pumps.

MEMBER BANERJEE: So you did some single-phase tests --

MR. TERAMAE: Yes.

MEMBER BANERJEE: with your pump, one PIRT scale, right?

MR. TERAMAE: Yes.

MEMBER BANERJEE: From what you said, you did no two -phase tests.

MR. TERAMAE: Yes.

MEMBER BANERJEE: You assumed same -- I'm trying to understand and repeat, so tell me if I'm right or wrong. So you did only single phase flow tests with one-third scale pump, no two-phase flow test --

MR. TERAMAE: Yes.

MEMBER BANERJEE: But you assumed two-phase characteristics are similar to Westinghouse pumps or some other existing pumps? Is that correct?

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MR. TERAMAE: In that part.

MEMBER BANERJEE: Even though these pumps are bigger.

MR. TERAMAE: Bigger.

MEMBER BANERJEE: How much bigger?

MR. TERAMAE: I don't have exactly value and -- but we adopt the one-third 93 two-phase data before the two-phase data, based on the -- degraded from the single-phase.

How to validate is only the -- from the Westinghouse data, based homologous used single-phase. Single-phase based on how degraded in the phases data or adopted to the Westinghouse two-phase data.

MEMBER BANERJEE: Yes, so let me ask a different question. How sensitive is your results here to the pump model?

MR. TERAMAE: US APWR are limiting cases always

an L-O-O-P case, offsite power off. Therefore the two-phase -- the two-phase condition is very short and therefore the effect of the two-phase condition RCP is small impact to the PCT.

MEMBER BANERJEE: Small impact?

MR. TERAMAE: Yes.

MEMBER BANERJEE: So, your peak clad temperature is always coming the second peak, not the first peak? There is two peaks, right? In the blowdown peak and the reflood peak?

MR. TERAMAE: Yes.

MEMBER BANERJEE: So you are always in the reflood peak?

MR. TERAMAE: That depends on the break type or sites.

MEMBER BANERJEE: I'm saying large break, usually what do you find?

MR. TERAMAE: Usually the reflood PCT is --

MEMBER BANERJEE: Controlling? The second peak is bigger than the blowdown peak.

MR. TERAMAE: Yes.

MEMBER BANERJEE: Okay. But you have a lot of emergency cooling, right? So -- I'm trying to understand the phenomena because of your -- if, if you have a lot of emergency cooling, shouldn't that reduce your reflood peak? Or does it come too late maybe?

I don't know, when does your high

pressure injection start? Around 100 seconds or what?

MR. TERAMAE: Yes. The single failure condition on the offsite power off condition, the high head injection, about 124 seconds.

MEMBER BANERJEE: Ah, okay. That explains -- 124 seconds.

MR. TERAMAE: Yes.

MEMBER BANERJEE: It's delayed a little bit.

MR. TERAMAE: Delay time is 100.

MEMBER BANERJEE: Okay. That's because of your gas turbine right?

MR. TERAMAE: Yes.

MEMBER BANERJEE: Okay. So, I'm just looking at it. You don't have the flow anywhere here. It would be useful to know what the different flows are, you know, from which sources they are coming, and how much flow, because your high head injection is 124 seconds.

MR. TERAMAE: Yes.

MEMBER BANERJEE: So that is on when you get the core quench. But where does the water go? Does it fill the downcomer and get into the core?

It would be nice to know where the water sources are, and where they are going, the water, okay? Because this, the plant is a little bit different. Your high head is coming a bit later, right?

So -- and when do the accumulators come?

MR. TERAMAE: Accumulators come from the cold leg.

MEMBER BANERJEE: No, I'm saying what time is the --

MR. TERAMAE: Accumulator injection started --

MEMBER BANERJEE: Okay, you are showing -- okay.

MR. TERAMAE: is 30 minutes -- 30 seconds.

MEMBER BANERJEE: Okay. Okay. So 10 seconds, right? Thirteen seconds. Thirteen, okay.

MR. TERAMAE: Thirteen.

MEMBER BANERJEE: One-three.

MR. TERAMAE: One-three. Thirteen.

MS. STEINMAN: It is 13 seconds.

MEMBER BANERJEE: Okay, well that also explains, yes.

MR. TERAMAE: Modeled results presented

tomorrow's session will.

MEMBER BANERJEE: Okay. Now, so you are saying that the two-phase pump model is not important. Did you do some calculations of this, to show it's not very important for the loop case?

MR. TERAMAE: Loop case, loop case also the short of the two-phase flow, because US APWR is automatic pump trip logic.

MEMBER BANERJEE: But you have a pump rundown, right?

MR. TERAMAE: Yes.

MEMBER BANERJEE: Over how long does the pump run down?

MR. TERAMAE: About 20 seconds from the ESF signal, it shoots.

MEMBER BANERJEE: But the pump, does it affect the blowdown peak?

MR. TERAMAE: Yes.

MEMBER BANERJEE: Yes, it does, right?

MR. TERAMAE: Yes.

MEMBER BANERJEE: So -- but the blowdown peak is not affected by your two-phase flow model in the pump?

MR. TERAMAE: Two-phase model -- yes.

MEMBER BANERJEE: Not affected?

MR. TERAMAE: Yes.

MS. ARNDT: Not affected.

MEMBER BANERJEE: Not affected. Can you show us a little bit tomorrow?

MS. ARNDT: We did an analysis on the loop condition and it was delayed, so we can show you the result.

MEMBER BANERJEE: Yes, it would be useful. It was -- okay. So we can hold it until tomorrow. Thank you.

MR. TERAMAE: Could I move on?

MEMBER SHACK: Yes, you may.

MR. TERAMAE: So, the right side figures show, of this right, the figures, the transient of the peak cladding temperature and the downcomer liquid levels. The upper figure indicated the free PCT.

The transient of the peak cladding temperature shows at 20 seconds, 23rd second on the blowdown PCT, 1563 degrees Fahrenheit at elevation of 10.7 feet from the bottom of the heat rings.

There is no safety injection yet. The 77 second, the reflood PCT of 1669 degrees

Fahrenheit occurs at the elevation 10.4 feet. This time, after this time the advanced accumulator injected is already activated.

The whole core is quenched at 220 seconds. We can see that these figures that transient during the LBLOCA for US APWR is very similar to the typical kind for PWRs.

MEMBER BANERJEE: But your clad PCT temperature is lower. It seems lower to me.

MR. TERAMAE: Yes.

MEMBER BANERJEE: Why?

MR. TERAMAE: This case is a reference case. I used the -- many permit I use the aluminum except that active types on the power relative to parameters.

The core power is 100 percent and if there is edge on each assembly, there possibly is a tech spec value. But other parameters use nominal parameter. That is a reason for the low PCT.

MEMBER BANERJEE: But in your ASTRUM calculations later, what is the range of the peak clad temperature above this? Is there a number of runs?

MR. TERAMAE: There are limiting cases

above this case but the range is very --

MEMBER BANERJEE: Wide.

MR. TERAMAE: wide.

MEMBER BANERJEE: Will you speak of this later? Tomorrow. Because we would like to know what effects the reference case, the most, what parameters.

MR. TERAMAE: Can I move on? Let me summarize our presentation. The WCOBRA/TRAC code with ASTRUM methodology is used for -- being used for the US APWR LBLOCA analysis.

Second, the new or improved US APWR features, such as a longer core, advanced accumulator, direct vessel injection and neutron reflector have been evaluated for the code applicability.

Third, the applicability of WCOBRA/TRAC M1.0 code and ASTRUM methodology to US APWR has been examined and confirmed based on the CSAU approach.

Fourth, WCOBRA/TRAC M1.0 has been applied to a sample US APWR plant analysis and its capability to simulate the US APWR LBLOCA transient was demonstrated.

Last, the US APWR LBLOCA analysis is

reported in DCD Section 15.6.5. Thank you very much for your attention.

CHAIR STETKAR: Any other questions for MHI at this time? If not, thank you very, very much. Appreciate it. And we'll hear from the staff.

MEMBER BANERJEE: So, John, we are putting a lot of things back to tomorrow.

CHAIR STETKAR: I'm aware of that. Tomorrow --

MEMBER BANERJEE: I mean, maybe it's worth having a closed session today.

CHAIR STETKAR: That's, well, we'll see how -- there's a lot of material to cover today also. It's -- becoming a bit concerned about that, so. Let's go through the staff's presentation on this, and before we break for lunch, we'll talk a little bit --

MEMBER BANERJEE: Right.

CHAIR STETKAR: I think about strategy for the next 24 hours.

MEMBER BANERJEE: It's pretty hard to take a two-day meeting and make it into one and a half days.

CHAIR STETKAR: Yes. And your point?

(Laughter)

MEMBER BANERJEE: That the P&P could get pushback on that.

MR. SHUKLA: well, Dr. Banerjee, one of the reasons --

CHAIR STETKAR: It's fine. Some days you win. Some days you lose.

MR. SHUKLA: Yes, one of the reasons it was done that way, because the SER for these topical reports might change.

MEMBER BANERJEE: Yes, I realize the reason, yes. But it's not a --

CHAIR STETKAR: We'll get through it. We have -- quite honestly we have another window of opportunity in October, to -- to bring other things up if we don't get them resolved. That may be a strategy. I don't like to do that but that may be our only -- unless you want to come back from Switzerland in the middle of August.

MALE PARTICIPANT: You'll love it.

CHAIR STETKAR: The heat particularly.

MR. TAKACS: All right. Good morning everybody. I'm Mike Takacs again here. We are going

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to provide an overview of the large-break LOCA safety evaluation applicability to the US APWR.

This presentation is slightly shorter because the code that was utilized here, WCOBRA/TRAC, the derivative used here for WCOBRA/TRAC M1.0, did not -- was previously approved code, if you will.

We have a different cast of reviewers here. With me is Fred Forsaty, the lead reviewer, and as well as the ISL support staff, the names are up there, and of course Jeff Ciocco being the lead PM for this review.

And that's it for my opening, and what I'm going to do, Fred, you can take over for the technical side of the review now, the presentation.

MR. FORSATY: Thank you. It is still morning, right? Good morning. My name is Fred Forsaty. I work for the agency, NRO. We have come to you previously for similar topic, large-break LOCA topical report for the previous design, so we have some kind of experience doing this review.

We, in the initial design review, we had some concern about thermal conductivity and the range of the variables that have been used as a part

of the sampling methodology.

MEMBER BANERJEE: What did you do about this TCD degradation thing, thermal conductivity degradation?

MR. FORSATY: I don't know --

MEMBER BANERJEE: I don't want to go out of sequence but it will be a question right?

MR. FORSATY: That issue was resolved -- for that design, actually, there was a few hundred degree impact on the PCT.

MEMBER BANERJEE: Right.

MR. FORSATY: And the good thing is we didn't see that issue on this design. The FINE code has the latest Haldon data and that basically resolved that biggest concern that we had in other --

MEMBER BANERJEE: So they took it into account because their code already had the Haldon data in there?

MR. FORSATY: In here or in the other design?

MEMBER BANERJEE: No, in this, this.

MR. FORSATY: Yes, we looked at it. We did some confirmatory runs. We compared it with the

other codes that we have in house, and we are convinced at this point that we don't have any thermal conductivity issue.

The FINE code is still under review, but as fa as I am concerned, we should not have any thermal conductivity issue that could impact the PCT.

As a partner of the review, we looked at the CSAU-related issues that we have had previously, scaling issues. We looked for some detail under WCOBRA/TRAC Mitsubishi version, especially with the features that are unique to the design, especially the accumulator, neutron reflector, and we also looked at the thermal conductivity issue these are the three areas we focused on.

MEMBER BANERJEE: So you looked at the reflector of the heat sink as well?

MR. FORSATY: To some extent.

MEMBER BANERJEE: Heat source, sorry.

MR. FORSATY: Yes. Again, I am going to say the same thing. I am going to repeat whatever Jeff said. You know, the reflector does not impact the analysis or doesn't have a major impact on the heat source.

MEMBER BANERJEE: Okay.

MR. FORSATY: Our confirmatory run was performed by ISL. They supported us. They used RELAP5 and we had the opportunity to run the WCOBRA/TRAC Mitsubishi version, and the idea behind that was to basically have some understanding of the new design, especially the accumulator design, because using that -- the modeling that is in the WCOBRA/TRAC Mitsubishi for accumulator, we took that modeling, we incorporated that to RELAP code, the MOD3.3.

And prior to that we ran the WCOBRA/TRAC Mitsubishi version to be comfortable with the code and the new design that is -- it is in the code itself.

Again, --

MEMBER BANERJEE: So you used both codes?

MR. FORSATY: We ran both codes.

MEMBER BANERJEE: And HOTSPOT, and what did you do -- what did you use for the fuel code yourself?

MR. FORSATY: Okay. That's a good question. We are going to come to that. HOTSPOT is

basically --

MEMBER BANERJEE: It's their code, right?

MR. FORSATY: Yes, it uses the boundary conditions that are given to it by the WCOBRA/TRAC. It takes all the boundary condition and provides -- it creates or calculates the fuel response to that, basically. Did I answer your question?

MEMBER BANERJEE: Yes, so what we are all concerned about all the time is the stored energy.

MR. FORSATY: Of course.

MEMBER BANERJEE: In this case, of course, since you are in a reflood peak situation, it's less important than if you were in the blowdown peak.

MR. FORSATY: That's right.

MEMBER BANERJEE: But if I understand it, HOTSPOT is used to calculate the stored energy, using some gap conductance source --

MR. FORSATY: Yes, use a thermal conductivity, gap conductance, the temperature profile and everything else, uses all the heat transfer coefficients that are given by the

WCOBRA/TRAC, and it does its neutronic thermal conductivities calculation.

MEMBER BANERJEE: Okay, and --

MR. FORSATY: It calculates the temperature profile and picks the hot rod temperature and that would be the PCT that would be used for that --

MEMBER BANERJEE: Right, now, HOTSPOT incorporates the thermal conductivity out of Haldon?

MR. FORSATY: That's right. It has the latest Haldon --

MEMBER BANERJEE: Haldon data. Is that your understanding too?

MEMBER REMPE: I've not seen anything on HOTSPOT, so I don't --

MEMBER BANERJEE: No, but you've read about it, but you've --

MEMBER REMPE: I haven't read anything about HOTSPOT. They said they were going to have a meeting later. So I don't know what HOTSPOT does or doesn't do.

MEMBER BANERJEE: So you assume right now that it takes into account TCD? We'll look into it later.

MR. FORSATY: The big issue previously was thermal conductivity as a function of burnup was modeled only for the first 10, 15 percent of the core life. After a certain time, as a function of burnup, thermal conductivity would not change. They would just keep it constant.

So as a result of that, there was no thermal conductivity degradation model previously. But in the FINE code, they have the latest data, so all the latest data is incorporated in calculating the thermal conductivity. You see the thermal conductivity degradation, and they ran some sample cases in-house to justify that, you know, there is no thermal conductivity issue.

MEMBER REMPE: So you ran FRAPCON in-house?

MR. FORSATY: Yes, we did.

MEMBER REMPE: And then did you put some sort of correction into RELAP 3.3 to account for what you learned with the updated FRAPCON?

MR. FORSATY: No, we didn't make any correction to that. We just ran the WCOBRA/TRAC and compared its result with what we had in-house available to us, for the latest data.

MEMBER BANERJEE: Let me understand, and maybe -- Joy, sorry for interrupting -- but when you put the calculations and did the RELAP5 calculations, you used the stored energy that came out of HOTSPOT, or -- because I guess she is trying to understand, which I am trying to, for your confirmatory calculations, the WCT HOTSPOT we understand. That's a -- what did you do for the fuel with RELAP5?

MR. FORSATY: I am going to give you my opinion and I am going to ask Dave, the person who ran the code. I believe the information that was in the WCOBRA/TRAC was extracted.

We compared it with FRAPCON. And all that information was used to create the database for the RELAP5 to run the analysis. Dave, can you help me out?

MR. CARAHER: My name is David Caraher. I work for ISL. And on these calculations that were done fo RELAP5 calculations, then what we did was we would match the center line temperature and we would put in a gap -- vary the gap conductance so we had the same stored energy.

We did not do anything where we ran a

fuels code. That's -- when we ran WCOBRA/TRAC, we generally looked at the reference case, which doesn't have a HOTSPOT calculation, but we did repeat their calculations for their limiting cases, which means we ran HOTSPOT.

But we never did a corresponding transfer of data from RELAP5 to FRAPCON.

MEMBER BANERJEE: So, let me give it back to you to see if I understood. For some of the limiting cases, like loop or whatever there were, you used a HOTSPOT, full HOTSPOT calculation or the results of it, with WCOBRA/TRAC to do some confirmatory tests.

With the RELAP5, you used the same center line temperatures and gap conductances to get roughly the same -- not exactly, but roughly the same stored energy, and ran those. Is that how I understood it, or did I get it wrong?

MR. CARAHER: No, I think you got it correct.

MEMBER BANERJEE: Okay. Do you understand what they did, Joy?

MEMBER REMPE: I think I understand what they did, but then is it really an audit, an

independent audit calculation?

MEMBER BANERJEE: Well, it depends what the audit is about, I guess. If it is about the thermal hydraulics, they have got roughly the same stored energy, right? So --

MR. CARAHER: Yes, and remember, the way this methodology works is they run a reference case. That reference case does not involve HOTSPOTs. That's where our confirmatory calculations ended as far as comparing RELAP5 and WCOBRA/TRAC results.

MEMBER REMPE: Okay.

MEMBER BANERJEE: Let's go over that again. The reference case does not involve HOTSPOT.

MR. CARAHER: That's correct.

MEMBER BANERJEE: But it has to have an estimate of the stored energy, right? How did they arrive at that then?

MR. CARAHER: How do they arrive at that? Well, they, I mean, steady state code, with the powers and thermal -- just the thermal properties would give you that when you initialized the code.

MEMBER BANERJEE: Yes, but is there degradation of thermal conductivity or not, in that

calculation?

MR. CARAHER: You'd have to ask Mitsubishi to be sure in that case, but as far as I know, no.

MEMBER BANERJEE: Right, this is --

MR. CARAHER: It's a, I mean, it's a beginning of life, the calculation, the reference case.

MEMBER BANERJEE: Yes, okay. So the reference case is -- that I didn't know, was a beginning of life calculation. Okay. If that's the case, then it may not matter. Right.

Does that make sense? I mean, you've got the hottest fuel, is the beginning of life, right? So I suppose that --

MEMBER SHACK: But that's true for other people that see an effect of TCD, so presumably it may shift the worst case a little bit.

MEMBER BANERJEE: We have to think this through. All right, we will revert to you and we need to -- because it, in some of the calculations we have seen, particularly where the blowdown peak is important, TCD has a big effect on the results, right? Without naming who they are.

MEMBER SHACK: Which, maybe it's that case.

MEMBER BANERJEE: Yes. Which was under your chairmanship actually. Okay. All right. Let's --

MR. FORSATY: As a part of our confirmatory runs we'd identified a couple of coding errors that we will discuss later, and those coding errors were corrected by the applicant, Mitsubishi.

MEMBER BANERJEE: What were the coding errors?

MR. FORSATY: I'm going to discuss it later. It's going to come up. Next. You are going too fast. Yes. Four. The outstanding open item basically has to do with how the accumulator biases are being calculated. That's --

MEMBER BANERJEE: That's for the small break.

MR. FORSATY: That's right, and our feel on that is that you know, we are aware of the changes, and we do not -- once they finalized the accumulator review, the applicant is going to run a new LOCA run, we do not expect the PCT change would be more than 20, 30 degrees, if at all it would be

that much.

But again, it's an open item and it would update the SE based on the information we are going to get later on from the applicant.

MEMBER BANERJEE: Do you agree with the applicant that the pump model doesn't have much effect on the PCT?

MR. FORSATY: It's a good question. We asked a question related to that. They asked them, it's not exactly the same question but probably the outcome would have the same result, they asked them to run the -- first the question, the LOOP, if they should run, be allowed to have LOOP all the time or not.

And my recollection is we asked them, some time ago, to run the LOCA runs with LOOP and without LOOP, we didn't see much of an impact in the final result. We looked at that.

Again, we just ran through the SE to see if we asked the applicant to run -- have LOOP all the time as a part of the analysis. We couldn't find that. But I remember that we asked that question, and LOOP and without LOOP, didn't have any impact on this. Does that help?

MEMBER BANERJEE: Yes.

CHAIR STETKAR: Fred, when they ran the no LOOP, they ran it really no LOOP, they didn't --

MR. FORSATY: I don't --

CHAIR STETKAR: They didn't just look at it through the, whatever other --

MR. FORSATY: No, I believe --

CHAIR STETKAR: delay times that they put --

MR. FORSATY: Yes, yes, they had the similar question in other designs, so that was one of the questions we asked, and they ran it for the whole --

CHAIR STETKAR: And they ran -- okay.

MR. FORSATY: Okay, next slide. Again, the US APWR large-break LOCA is a best methodology, it uses the ASTRUM methodology. The applicant is using the WCOBRA/TRAC Mitsubishi to do the nuclear sink supply system calculation.

As I mentioned before, HOTSPOT is used through the neutronic side of the code. And we asked the applicant, the applicant basically ran 124 cases and it's not mentioned here, we had two questions on the cases, the range of the IRWST and

the accumulative temperature was not realistic. If I remember right, the range of the temperature for these two systems were somewhere between 50 degree Fahrenheit to 130, and that's not a realistic temperature for the containment at 100 percent power.

That -- Mitsubishi corrected that and I believe the new number was 90 degrees to 130. The impact was 30, 40 degree Fahrenheit on PCT only.

MEMBER BANERJEE: So, let me ask on this one. We saw that for some cases the check valve comes in, right?

MR. FORSATY: For the accumulator?

MEMBER BANERJEE: For the accumulator. How sensitive -- the first question is, does it activate before or after the peak in the temperature? The peak , if I remember the reference case, occurred around 100 seconds or so, correct?

MR. FORSATY: That's right.

MEMBER BANERJEE: And if I remember, the accumulator was injecting very strongly, still, and it was only after that, that when it started to fall, that the check valve came in.

But that was for this large-break LOCA

or was that for the small-break LOCA?

MR. FORSATY: I think the picture you saw was for a small-break LOCA.

MEMBER BANERJEE: Right, the picture was for -- but does it give a similar behavior for the large break?

MR. FORSATY: Well, it's -- that's a good question. I think that is specific to a small-break LOCA because you don't have that the high pressure drop for a small break compared to a large break, so you would see that closure of the valve.

But also for a small break, it doesn't happen for a large break. So it basically acts like a regular accumulator -- it has a good benefit also that some of the water is still trapped in the accumulator, in the baby accumulator after vortexing, so to me, that helps prevent the nitrogen injection --

MEMBER BANERJEE: Going in, yes.

MR. FORSATY: So I think that's a benefit. And when it comes to small-break LOCA, you know, closure of the valve would take probably 10 seconds to open, 10 seconds to close, so that's about 20 seconds, so you are not going to be cycling

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on and off for 100 seconds.

You don't -- it is not going to happen so many times, because for 100 seconds, it probably opens once and then after that, it closes maybe once and that's it and then it should not be a kind of cycling of that event for the duration of interest.

Is this, I think that's something that probably --

MEMBER BANERJEE: It would be good if somebody, if not the staff, the applicant, would walk us through what happens, you know the phenomena. Because this is a little bit unusual plant compared to what we are used to seeing, and I am not completely familiar with all the phenomena here that would occur, like when the accumulator flow drops, and whether the nitrogen goes in or not, the points you were mentioning right now.

In fact, I don't have a good feel for it like I would have for a Westinghouse plant, you know, so maybe the applicant, this is for the applicant to note, maybe should walk us through all the things that are happening, so we see what the physical phenomena are, when is the downcomer filling, when is various things happening.

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MR. FORSATY: I think that some time ago, I had the same question and I looked at some data. I don't have it with me but I think --

MEMBER BANERJEE: Well you could do it if you have got it, at some point.

MR. FORSATY: I will search. I think the nitrogen, 90 percent of it is still going to be in the accumulator after 100 seconds or so, so 10 percent would be released. That's, I think the number I have. I don't know how accurate that is. But that's based on the data I collected a long time ago. So I thought that's a good thing.

MEMBER BANERJEE: No, that is definitely a good thing. So okay, but just note, somebody should note that because this is so new to us, the phenomena, we should walk through the phenomena carefully and see that we understand what is happening. I don't have a really good picture of the physics of what's going on.

MR. FORSATY: All the data that you need to create that is probably something that the applicant has more access to the data than --

MEMBER BANERJEE: Right. Yes. Fine. Girijiya please note this. Thanks.

MR. FORSATY: So again, compared to the previous design, you know, they are running 124 cases, not 59 cases. The thermal conductivity, should that not appear to be an issue here, is not an issue here.

We --

MEMBER BANERJEE: So, 124 is the normal ASTRUM --

MR. FORSATY: That's right. That's what it should be. And then the applicant agreed and corrected the temperature range for the accumulator and the IRWST temperature to make sure that the range is reasonable and realistic.

Again, these are some of the features of the APWR design. You've got the new accumulator design. And compared to a Westinghouse four-loop plant, we have basically higher or increased HPI flow. We have significantly bigger or higher heat transfer area for the steam generators. It's almost twice as much.

We have larger pressurizer for 10 foot fuel. They have a linear portion in there. We have heavy neutron reflectors with the smaller holes that would cool a reflector, and of course, larger core,

larger volume, more fuel, more power.

Next please.

MEMBER BANERJEE: The volumetric power per unit volume is the same more or less, right, or lower?

MR. FORSATY: No, it is -- it's not the same. We're going to get to that. We basically -- we have that information later. We'll give it to you and if you need more data --

MEMBER BANERJEE: Right.

MR. FORSATY: This slide, you have seen it many times. The PCT is -- plenty of margin for the PCT, compared to 2200. The -- no issues with the cladding oxidation, and under this LOCA condition, which we do not have any issues with the coolable geometry, there is no deformation of fuel, and assurance of long-term decay heat removal, I think we are going to discuss that as a part of 15.65 in the afternoon.

We have two open issues, three RAIs, the open issues related to a small-break LOCA criticality, boron dilution, one RAI on that, and then we have boron precipitation, we are questioning that and that's an open issue.

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MEMBER BANERJEE: Is that -- we always have these issues with how much mixing we allow and things like this. I mean, we can visit it when we talk in the afternoon.

MR. FORSATY: Yes, we will -- but actually --

MEMBER BANERJEE: Ideally we would want to know the assumptions that are going into this.

MR. FORSATY: Yes. This afternoon is, not closed session, but one thing I can tell you, they did some testing on that and mixing, recently, and the result is under review. But they did a complete testing.

MEMBER BANERJEE: Did you guys ever get in touch with Len Ward on this?

MR. FORSATY: Len Ward? I don't remember. On which issue, on boron precipitation or mixing?

MEMBER BANERJEE: Both.

MR. FORSATY: We are closely working together again as a part of consistency, we are, you know, if not daily then weekly communications with the NRR, and we are -- on this specific issue, we are working very closely --

MEMBER BANERJEE: With NRR.

MR. FORSATY: with NRR. They are going to help us to run some confirmatory runs and you probably already know, our branch chiefs, they meet every week, once a week and they discuss all those issues.

MEMBER BANERJEE: Okay.

MR. FORSATY: Yes, Len is very much involved.

MEMBER BANERJEE: Yes, because we have some long-term running concerns about these mixing and --

MR. FORSATY: Boron precipitation. If you look at those two RAIs, we put it on the table two years ago, so that's in line with your concern. So --

MEMBER BANERJEE: Okay.

MR. FORSATY: That will be addressed. So if you can go to the next one, we looked at, again, we put more time on this than we did with the other design. We had more time. We looked at the PIRTs for the blowdown, refill and reflood. We looked at the -- especially the highly ranked variables and again, we didn't see any issue with

their choice.

MEMBER BANERJEE: So this is absolutely standard, right?

MR. FORSATY: That's right, I mean we followed it to the extent possible. Again, as a part of that we basically identified that the generic issue with the temperature range for a few of the systems needed to be corrected and it was correct.

MEMBER BANERJEE: They have of course DVI injection in this system, right?

MR. FORSATY: Yes.

MEMBER BANERJEE: A little bit different from cold leg injection. It didn't raise any issues in the downcomer and -- if I remember they have a design where they try to lead the flow downwards into the downcomer, right?

MR. FORSATY: The DVI injection is right -- in case inject the core in the downcomer and, Dave, if you can help answer this question.

MR. CARAHER: They have some pads located on the --

MEMBER BANERJEE: You have to come to the mic unfortunately, or fortunately, identify

yourself.

MR. FORSATY: He is a better person to answer that, but I think I know the answer.

MR. CARAHER: On the --

MEMBER BANERJEE: Name please.

MR. CARAHER: David Caraher, ISL, and they are -- just opposite the DVI nozzles there are some pads located that directs the nozzle flow downwards.

MEMBER BANERJEE: Right, but the reason I was asking this question is, did this introduce any new factors into the PIRT, the DVI injection and the fact that you had these pads? Were there some new phenomena that had to be taken into account? It was really related to the PIRT, I was asking the question.

MR. CARAHER: There are no new issues. No.

MEMBER BANERJEE: There are no new issues, okay. Well, let us consider that, because it is a little bit -- it's a different injection mechanisms from what the normal PIRT is, right?

And so the -- I imagine there could be no new issues, but I just want to flag that as a

difference, and try to understand what consideration was given to it, and then maybe it didn't introduce any new issues.

MR. FORSATY: I can get back to you on that tomorrow, but --

MEMBER BANERJEE: Okay.

MR. FORSATY: I forgot at the beginning to give you a little bit of history of my -- but I will do it at the end, how about that?

MEMBER BANERJEE: You don't have to. It's okay.

MR. FORSATY: Okay. We looked at actually the early -- I'm sorry?

MEMBER BANERJEE: The last point is what I was --

MR. FORSATY: Yes. Yes. At the early part of this review, we -- the staff discussed the plan on how we are going to do our review, and at that time the TRACE codes didn't have the accumulator model. So what they had to do is, we got the WCOBRA/TRAC Mitsubishi model, we ran some cases to make sure we get familiar with that, and then we borrowed the accumulator model once we were kind of comfortable with that. We used that for the

RELAP5 M3.3 and that's the version we used to do our confirmatory runs.

And again, the basic differences here in this -- for the WCOBRA/TRAC, and the WCOBRA/TRAC Mitsubishi model is basically the core length which is for 10-foot fuel, should not have that much of an impact but then accumulator design and neutron reflector and DVI injection, they are looked at as a part of our confirmatory runs.

As a part of this review, initially, before we do our actual analysis, we identified an issue with the code and that was the gravitiational head was calculated --

MEMBER BANERJEE: What was wrong with that?

MR. FORSATY: I knew you were going to ask me. The gravitational head is basically a simple calculation, right, rho g, delta g, I think at some point some of the history of the calculation was not reset, and it was not properly calculated.

But, again I'm going to call Dave to make sure it was -- confirm what I just told you. We used that -- we communicated that issue with the applicant. The issue was resolved.

This is -- my understanding this is something that existed prior to the code WCOBRA/TRAC to be modified to Mitsubishi. I have no guarantee on that. But again, that issue was communicated to applicant. It was resolved and the impact was minor.

MR. CARAHER: David Caraher with ISL again. The first thing we did when we had -- took WCOBRA/TRAC and started doing confirmatory analysis, is just do a null calculation where you turn off the core power and the RCPs and see if the -- all the LOOP flows should go to zero. They didn't. So then we just kicked it back to MHI and asked why.

And after their analysis, they found that there was, when they did all the momentum source terms for the RCPs, there was term missing or an extra term added that kept the gravity head.

MEMBER BANERJEE: So how could you have gravity head if there was no temperature difference?

MR. CARAHER: That's the point.

MEMBER BANERJEE: Right, but did they --

MR. CARAHER: They had a momentum source term that, I mean, they had the DZ, you would think, in this case, right? rho g delta z. The rhos are

the same.

MEMBER BANERJEE: Could we get a little bit more -- what was the term that was causing this issue?

MR. CARAHER: I can't recall, but we have the coding, I mean, we looked and reviewed their coding when they changed it, and how it was changed.

MEMBER BANERJEE: No, I know. But was there a term added or what was it -- what was the nature of this term, could you tell? We can ask the applicant.

MR. CARAHER: Yes I think it would be better to ask the applicant.

MEMBER BANERJEE: Yes, could somebody from the applicant tell us a little bit more about this term which caused flow in the absence of gravity head?

MR. SPRENGEL: Yes, this is Ryan Sprengel. Give us just one second to confirm if we can provide that information now or if we will provide it later today.

MEMBER BANERJEE: Sure. If it was related to the gravity head, that's one thing. It

can be that sometimes, at bends and things, if you don't do the momentum --

MEMBER SHACK: Dr. Wallis would be familiar with that.

MEMBER BANERJEE: It acts as a pump, so I just want to make sure that you have got the problem sorted out.

MR. SPRENGEL: Okay, we are going to take some time to review this and provide that information with our later presentations.

MEMBER BANERJEE: Okay. Girija, in case I forget. My memory is fading in old age. Keep going.

MR. FORSATY: Next page please. Again, HOTSPOT is, as I've discussed, is the coding part of the WCOBRA/TRAC. That's the neutronic part that takes the boundary condition from WCOBRA/TAC, the thermo-hydraulic boundary conditions, and calculates fuel response.

During the confirmatory runs, we noticed that the HOTSPOT -- there was an error in HOTSPOT. The way HOTSPOT is set up, the heat transfer coefficient has a top boundary condition which should not be exceeded, and that top boundary

condition, if I remember right, it was like 30 or 35 BTU per hour square foot, and during the calculation we noticed that number exceeds.

So, that error was identified, communicated to the applicant and it was resolved. And by the way, all these errors also have been communicated through NRR. They are very much aware of this and they have taken all the necessary action to make sure that if that exists in the previous approvals, it would be addressed.

MEMBER BANERJEE: So, the -- let me try to understand. The reflood heat transfer coefficient could go beyond -- what was the upper limit? Was there -- I'm just trying to recall, is there a set upper limit on this, or what?

MR. FORSATY: Yes, there is a set upper limit.

MEMBER BANERJEE: Is that an Appendix K limit, or what?

MR. FORSATY: No --

MEMBER BANERJEE: Or is it =- can you just enlighten me as to what that is --

MR. FORSATY: The upper limit is basically -- is an upper limit that after that is

just basically you are going to a different phase, it's not applicable to the phase that you are analyzing, and that basically would not be realistic.

So that upper limit was set up some time ago, a long time ago, to be -- not to be violated. And then ISL ran the code that discovered that upper limit.

MEMBER BANERJEE: Is this in some reg guide or something, that this is going to be an upper limit?

MR. FORSATY: It is in some --

MR. CARAHER: This is David Caraher, ISL again. The way it was developed is that when Westinghouse was -- started using this methodology, it was found that HOTSPOT could predict numbers that -- the staff asked at that time, well, you can predict numbers that are higher than what we ever see, or may see in Flecht-Seaset tests.

So they went, the staff requested Westinghouse to put an upper limit and that upper limit was based upon the best -- the highest value they might see during the reflooding phase, not during, you know, some really insurge phase, some

steady state value.

MEMBER BANERJEE: So this was in the pre-cooling phase, in the precooling region, above the quench front?

MR. CARAHER: Yes.

MEMBER BANERJEE: So it was in dispersed flow --

MR. CARAHER: Yes, that's right. And so the staff asked, can you give me -- identify a maximum value and Westinghouse went back to the Flecht-Seaset test, and said, okay, we'll fix a maximum value, here it is, here's the number.

And now, HOTSPOT will always -- it will always have a number below that until it quenches, okay?

MEMBER BANERJEE: Okay. So there's some limit based on experimental data.

MR. CARAHER: Yes.

MEMBER BANERJEE: Fine, thanks. I wasn't aware of that actually. Okay. Thank you.

CHAIR STETKAR: See what happens when you come to these meetings?

MEMBER BANERJEE: Yes, you learn something. Always new.

MR. FORSATY: Okay then. Next. I think you had some question on the scaling, the US APWR power is 1.3 or 30 percent more than the conventional plant. This RCS volume is about 37 percent more than conventional plants.

MEMBER BANERJEE: But is the core volume the same?

MR. FORSATY: Core volume is 25 percent bigger.

MEMBER BANERJEE: The volume.

MR. FORSATY: Volume.

MEMBER BANERJEE: So it's about 1.3 over 1.25.

MR. FORSATY: Flow area for the hot leg is 12 percent higher than the flow area for a conventional hot leg.

MEMBER BANERJEE: So it doesn't go in proportion?

MR. FORSATY: No, but the good thing is we looked at this as a possible issue and we did some confirmatory runs. We looked at the steaming rate and the velocity in the hot leg, all the droplets would quickly be transferred to -- would not be holding that, and would be transferred

quickly to the steam generator, so --

MEMBER BANERJEE: So from the point of view of steam binding, there are no issues that can occur due to this?

MR. FORSATY: I do not see any issue. We spent quite a lot of time on that, and --

MEMBER BANERJEE: Somebody looked at it carefully?

MR. FORSATY: Yes.

MEMBER BANERJEE: Right.

MR. FORSATY: And you have some backup slides and maybe tomorrow we can show you that to show you that --

MEMBER BANERJEE: So let me try to understand this again. The hot leg is 12 percent larger in area, but the power is 30 percent larger.

MR. FORSATY: That's correct.

MEMBER BANERJEE: What about the steam generator flow area? Is that --

MR. FORSATY: The steam generator area -- the area was, I think it's -- Dave, correct me if I'm wrong -- I think it's like almost twice as much area.

MEMBER BANERJEE: No, the area may be,

but the flow area. Is it twice as much?

MR. FORSATY: No.

MEMBER BANERJEE: No. What is the ratio of the flow area?

MR. FORSATY: It should be 30 percent.

MEMBER BANERJEE: Is it 30 percent? Does it go in proportion or -- we are simply trying to find if it's in proportion.

MR. DONOGHUE: This is Joe Donoghue. Does Mitsubishi have that information?

MR. HAMAMOTO: This is Hiroshi Hamamoto. Sorry, we don't have it now. We need to confirm.

MEMBER BANERJEE: Okay, so Mitsubishi agree with the 12 percent, because that was a question that arose -- to them as well, we asked them how much was the hot leg increased compared to the power increase.

So, it's come as 12 percent now, so we need to confirm --

MR. FORSATY: That's the number I have, 12 percent, and the steaming rate for the hot leg velocity is basically 15 percent more than it. The steaming rate is basically a little bit more than the flow area for the hot legs, about 15 percent

higher compared to what --

MEMBER BANERJEE: Why is the steaming ratio -- why isn't it 30 percent higher based on the power? Is there a reason?

MR. FORSATY: Why --

MEMBER BANERJEE: The power is 30 percent higher.

MR. FORSATY: That's right.

MEMBER BANERJEE: So why would the steaming rate be only 15 percent higher?

MR. FORSATY: I think that's a good question. It should be more on the, you know, the pressure drop differential and --

MEMBER BANERJEE: Yes, it probably is related to a lot of things, but you know --

MR. FORSATY: And that tells you that the steam generator flow area is probably not 30 percent.

MEMBER BANERJEE: Yes.

MR. FORSATY: So it should be within the range of 15 to 20 percent higher and not 30 percent.

CHAIR STETKAR: Why don't we wait? I think MHI has it, you know, as an action item to come back with --

MEMBER BANERJEE: Okay.

CHAIR STETKAR: with those areas and --

MR. FORSATY: But again, when it comes to large-break LOCA, we didn't see --

MEMBER BANERJEE: It doesn't matter too much.

MR. FORSATY: It doesn't. For us it doesn't matter.

MEMBER BANERJEE: It's just the steam binding --

MR. FORSATY: We don't have that issue. Okay. And so you went through this slide. Next slide please. Overall, our review indicates that in our WCOBRA/TRAC, the reference base, you know, I believe with our confirmatory runs and the differential was maybe 10, 20 degree Fahrenheit.

And we did some sensitivity runs, we didn't find any issues that could impact the acceptance of the code. Again, I just want to emphasize the fact that WCOBRA/TRAC is approved. We focused on the differential, on the changes, and we didn't identify any issues that would impact the results. And there's plenty of margin on the PCT. Next slide please.

And the conclusion is that the Mitsubishi large-break LOCA methodology is acceptable for calculating peak clad temperature, hydrogen generation, fuel cladding surface oxidation for the US APWR, and the PCT that we have today is close to 1800 degree Fahrenheit, which, again, has a considerable margin, 400 degree.

A little background on me. My name is Fred Forsaty. I have been the agency five, six years, worked in the private industry at numerous facilities, at Beaver Valley, Niagara, Mohawk, Yankee Atomic, VC Cook, built the simulator for Arizona Power, I didn't see Jeff there when I was working. I built a simulator for Perry plant and I have been here six years working on the new reactor site.

MEMBER BANERJEE: So, thank you. Just wanted to ask you a question about the PCT. I'm always interested in the physical reasons for this, and I assume that what is happening here is they have plenty of HHS, high head safety injection, which is why they shift out of the blowdown peak into the reflood peak, which is why you're getting relatively low PCT.

It would be good if we could get a physical explanation for this, either from the applicant or the staff, because there has to be a reason for it, and it's probably related to the -- I mean, I'm just thinking back.

It's not -- sorry, it's not the HHSI, because it comes only, it must be the accumulators that come on very fast.

MEMBER SHACK: It's over by the time the --

MEMBER BANERJEE: Yes. By the time that comes in, the peak is there already, so it must be the accumulators. But we need to understand the physics. I haven't understood the physics of this plant well, yet.

MR. FORSATY: You know, we did some work on the other designs and similar phenomena was occurring, you know, and so this is nothing new to us at least --

MEMBER BANERJEE: Because we have encountered calculations in plants where they have come very close to the 2200 limits with -- even with the ASTRUM methodology, and thermal conductivity degradation. So --

MR. FORSATY: Also, the good thing is there is no nitrogen injection because in some other designs we made them to take the nitrogen injection out to see what the impact would be on both peaks, to make sure they are as well as possible.

And then, again, for large-break LOCA you are basically, accumulator, it's a normal accumulator when it comes to its function.

MEMBER BANERJEE: Yes, but really, why does the blowdown peak come so low compared to the reflood peak here, at least the reference case we saw, that they showed us, the blowdown peak was quite a bit lower than the reflood peak.

So there has to be a good physical explanation for that.

MR. FORSATY: Well, the stored energy has to, you know, get a little bit more time to --

MEMBER BANERJEE: Get out for some reason.

MR. FORSATY: so that it's free to get out, and so that, and the fact that, you know, they have got decay heat, they have got to store energy and al of these things that would tend to, you know, provide a little bit more time for the second peak

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to occur.

MEMBER BANERJEE: Yes. Well let's walk through the processes, you know, the physics of this later on.

MR. FORSATY: To make sure I understand what my actions are, did somebody with you summarize this, or when they come back tomorrow, we know who is going to come back with some answers?

CHAIR STETKAR: Let's -- let me first go -- make sure the -- any other members have any questions for the staff on what we just heard? Joy?

MEMBER REMPE: Actually, more of a question for you on process. At the beginning of this you discussed the fact, and throughout this morning we have heard that there's several points that are outstanding, about the accumulator model, the steam generators --

CHAIR STETKAR: That's, that's -- I was going to get to that. Does anybody have any questions for the staff on what we just heard?

Okay. Now, in about 10 minutes -- now I would like to address the process, because I'm not real clear in my own mind where we should plan to go over the next 24 hours.

What we do need to accomplish is I want to be sure that we get through MHI's presentation on all of Chapter 15, and we get through the staff's SER on all of Chapter 15. We need to do that.

We also need to address the COL, which is short, but we need to do that on Chapter 15. We have time allocated tomorrow morning for closed sessions on both the large and small LOCA methodology.

My sense is that we probably do not have enough time to do the types of in-depth analyses we have been discussing this morning. Given the fact that there are still outstanding questions about the performance of the accumulator, the modeling of the accumulator, we will need to revisit the LOCA analysis at some time in the future.

We have a subcommittee meeting scheduled in October. Right now it is slotted for two days, so we have the two days allocated to it. The topics of that meeting right now were for the non-LOCA analysis methodology.

There's a topical report on that that we are not discussing during this meeting because the SER is not ready yet for that, and information on

the CFD models, at least, for the accumulators.

I don't have an answer about what -- either from Fred's question or Joy's question, what we should plan to discuss tomorrow. I don't know what the most efficient way to spend the remaining time is, regarding the LOCA analysis, and if anybody has any suggestions, I'd like to entertain them.

My sense is that we are going to have to revisit the LOCA analysis in more detail in October. By that time there will be better information about the accumulator and people will have more time available to prepare the information for the actual modeling. But --

MR. FORSATY: When you say LOCA, you mean small or large or both?

CHAIR STETKAR: I didn't distinguish, did I?

MEMBER REMPE: It's not just the accumulator. There's the thermal conductivity degradation that's supposed to be reviewed, the steam generator, so it could all affect the plant's ability to transfer heat etcetera.

And so I'm just kind of -- I assume someone is tracking this, but at some point you

might decide all of these analyses need to be redone or some fraction of them needs to be redone, because of a model not being --

CHAIR STETKAR: Well, I think at a minimum, they will need to be redone to develop what the temperatures are --

MEMBER REMPE: Right.

CHAIR STETKAR: for example. That's certainly true. Whether -- I think the purpose of this exchange, during this day and a half, is to identify whether we have any fundamental concerns about what we hear about the overall methodology that may then factor into those re-analyses.

I mean that's my spin on what we are up to, you know, in this day and a half session.

MR. FORSATY: When it came to large-break LOCA I think the stored energy was properly modeled, you know. Conservatively, then, based on our previous experiences, I do not see that that would be an issue, and if you want us to come back -- if that would help I can come back to you with some of the questions that is on the table for large-break LOCA.

CHAIR STETKAR: Okay. Let's -- and I'm

clueless about the topic. And I'm old enough to admit that. Sanjoy and Joy, because you are most actively involved, perhaps you might want to think over lunch and when we come back from lunch, if there are specific kind of focus issues, questions, that we can address in tomorrow morning's session, if we can kind of elaborate those for either the staff, if it's for the staff, or for MHI, so that everybody is clear, you know, what we expect tomorrow morning, and I suspect they will not have enough time to do the detailed types of information that you were asking for, Sanjoy. I think that's something that we will need to revisit, probably in October.

But if there are specific issues that you want to -- topics, questions, to make sure that either the staff, if it's for the staff, or for MHI, so they can be prepared to come and give us those answers in a closed session tomorrow.

MEMBER BANERJEE: I think that I have a pretty good idea of the -- from these presentations of the differences in phenomena that you encounter compared to the conventional PWRs they are used to.

Nonetheless, there are effects which are

somewhat unusual that need to be thought about more carefully and the physics behind them clarified. For example, the shifting of the critical small break from the usual four-inch small breaks to the much larger sizes, you know, it's something we need to get our own sort of thoughts around.

It may not be that they have to supply more staff. It's simply that we need to do our homework better, you know, on that, and try to understand, and then you have much more pointed questions.

So, because of that, for example, a phenomenon like loop seal clearing may have a different importance than it does conventionally in a, let's say, a Westinghouse plant, or similarly, reflux condensation may not be quite as important, so you need to recalibrate your mind in some way and that hasn't happened with my mind yet. So --

CHAIR STETKAR: The only reason I bring it up is obviously the MHI in particular, you know, has presentations prepared for tomorrow morning, and unless there are very focused specific issues that we need to absolutely have addressed, you know, I'd ask them to sort of proceed with their normal

presentation.

On the other hand, if -- if, during this morning's or for that matter, this afternoon's discussion, if there are very, very focused questions that we really need an answer to --

MEMBER BANERJEE: Well, I'll give you one --

CHAIR STETKAR: Other than the broader sort of understanding that --

MEMBER BANERJEE: I'll give you one which is sort of important in my mind. You know, we typically do calculations for other plants where we force loop seals not to clear.

Now, for small -- relatively small breaks, I think that is a reasonable thing to do and the staff does that.

CHAIR STETKAR: And they did that.

MEMBER BANERJEE: Now, in this case, we did not and maybe there are good reasons, because as you go through larger breaks becoming more important, the loop seals will clear more easily.

But we do need to have a very clear understanding of that rationale, that indeed, loop seals will clear more easily, and what is the basis

for that, other than a warm fuzzy feeling that they will clear more easily.

CHAIR STETKAR: Okay.

MEMBER BANERJEE: You know, they'd have to be made more --

CHAIR STETKAR: So if MHI could at least try to address that in the discussion of the small LOCA --

MEMBER BANERJEE: Yes, because clearly if they don't clear, and you have only one loop seal cleared, you are going to get a pretty high temperature for your larger small breaks.

CHAIR STETKAR: So that's one thing for tomorrow, if you can keep, MHI, if you can keep that --

MEMBER BANERJEE: So, clearly we are going to go over the rationale for saying why loop seals clear, why they don't clear, what sizes do they clear. You know, this thing has got to be exposed in its gory detail.

CHAIR STETKAR: Yes.

MEMBER BANERJEE: Okay. And similarly, you know, I can see that refluxing is probably not important here because you are going into larger

breaks so you are getting more heat out of the break --

CHAIR STETKAR: Well, but everything you are saying is that rationale of what -- why that critical break size is --

MEMBER BANERJEE: All of this has to be thought through very carefully.

CHAIR STETKAR: Right.

MR. SHUKLA: Dr. Banerjee, as a background, when we were planning for this meeting, there was some question about whether these topical reports and their SERs should be presented to us or not, so, because they are subject to be revised based upon the advanced accumulator analysis.

MEMBER BANERJEE: Until the accumulators are done --

CHAIR STETKAR: Well, but I mean, in some sense that's details of the calculation. It's not the, you know, a fundamental part of the methodology of what they are doing.

MEMBER BANERJEE: Unless there's some huge surprise due to this, I don't think we want to rehash what we have already discussed for --

CHAIR STETKAR: Yes, but that's the

whole point of having, you know, these subcommittee meetings at the time during the review process, is that if there -- if we, as a subcommittee, had some fundamental issue with the methodology, not necessarily the specific results, because we know they are going to change.

MEMBER BANERJEE: So we are very, very concerned, and have been always, about the ability of these codes to predict loop seal theory, you know, so that's why the staff has always taken a position to hold them from clearing, and see what the effect is, and in this case, maybe there is more reason to allow it, because the breaks are larger, but we really need to go over this aspect of it in detail.

CHAIR STETKAR: So, if nothing else, then, for tomorrow, in terms of -- in terms of organizing MHI's time, if you can at least focus a bit on that issue for the small LOCA discussion, and maybe reorganize your time a little bit to focus a little more on the small LOCA, because I think that is a bigger concern.

MEMBER BANERJEE: And I'm also sort of concerned, not concerned, but would like to know

why, what the physical reason is, that the blowdown peak is not dominant here --

CHAIR STETKAR: That's a large LOCA.

MEMBER BANERJEE: which it is in many plants, compared to the reflood peak.

CHAIR STETKAR: And MHI, there's something for your large LOCA discussion.

MEMBER REMPE: Is it worthwhile to spend a little bit of time looking at -- well I looked at the slides at the closed session tomorrow, and they show nodalization diagrams that they have selected and to discuss briefly those nodalization diagrams and the basis for it?

CHAIR STETKAR: I mean, that's something they -- you know --

MEMBER REMPE: that the method, it's not a result.

CHAIR STETKAR: they are planning to do. What I'm trying to do is identify --

MEMBER REMPE: But if we don't get through the whole presentation, that's the part that was of most -- is of interest to me, knowing that the results can change.

CHAIR STETKAR: Okay. Okay.

MR. FORSATY: So, on the large-break LOCA, to make it clear, any questions on methodology that I can --

MEMBER BANERJEE: I don't think on the methodology, really, trying to understand the phenomena.

MR. FORSATY: Okay, I take that action, and --

MEMBER BANERJEE: I mean that's the phenomenon.

MR. FORSATY: I take that action. Any other concerns that --

MEMBER BANERJEE: I think the methodology -- we have agreed to this more or less.

MR. FORSATY: Okay.

MEMBER BANERJEE: You know. This is a little bit larger size and longer core, more reflector, but --

MR. FORSATY: I'll do my homework tonight, try to have something for you tomorrow.

MEMBER BANERJEE: Yes, and let's make sure that the DVI injection is something that the methodology can handle.

MR. FORSATY: And the PIRT can handle.

Those are the two actions. Okay. Thank you.

CHAIR STETKAR: Anything else, any of the members? If not, thank you and we will recess until one o'clock.

(Whereupon, the meeting went off the record at 12:55 p.m. and resumed at 1:02 p.m.)

A-F-T-E-R-N-O-O-N S-E-S-S-I-O-N

(1:02 p.m.)

CHAIR STETKAR: Okay, we're back in session. This afternoon we are going to hear about all of the Chapter 15 transient and accident analysis from both MHI and the staff, and we will start off with MHI's presentation.

MR. MARUYAMA: Thank you. Good afternoon. My name is Yuta Maruyama. We are representing Mitsubishi Heavy Industries. Today we will be discussing the US APWR DCD Chapter 15 transient and accident analysis.

We also have several technical experts from MHI here today. The technical experts represent non-LOCA, small-break LOCA, large-break LOCA and dose sections.

This slide and the next slide provide a list of acronyms used during our presentation that

may be a helpful reference for you today.

Chapter 15 consists of the analysis of various anticipated operational occurrence, and postulated accident event. The US APWR design is very similar to the design of current operating PWRs in the US and Japan. Therefore, there are no new AOO or PA events that need to be defined specifically for the US APWR.

Here is the summary of the main plant parameters for the US APWR as compared to a typical four-loop plant in the US. As you can see, the US APWR has a larger core thermal power. The US APWR also has a lower average heat rate, which provides larger thermal margins. The US APWR also uses 14-foot fuel.

Most of the design features of the US APWR are similar to operating PWRs in the US. However, there are a few differences. This table provides a list of the key design features for the US APWR and how these features impact the safety analysis. In most cases, these design features are explicitly modeled in the safety analysis.

Chapter 15 is divided into nine sections and one appendix. Each of these sections will be

discussed in detail during this presentation. Section 15.0.0 of the DCD contains introductory material. This section describes generic information that is meant to be applicable to all Chapter 15 events.

This information includes classification of events as either AOOs or PAs, analysis assumptions for initial conditions, reactor trip and ESF signals, single failures, operator actions, loss of offsite power, and so on.

These descriptions are very general. For each event, any information that is event-specific is discussed in the applicable section for that event.

There is one open item in the Safety Evaluation Report for section 15.0.0 related to the revision of the reload evaluation methodology technical report.

Section 15.0.2 of the DCD concerns the methodology of the safety analysis. MHI's methodology follows the regulatory guidance in SRP Chapter 15. This section also discusses the computer codes used in the safety analysis methodology.

For the non-LOCA event, MHI uses following codes -- MARVEl-M, TWINKLE-M, VIPRE-01M, and ANC. The details of these computer codes and how they are used in non-LOCA analysis are described in the non-LOCA methodology topical report, MUAP-07010.

For the LOCA event, MHI uses three codes. WCOBRA/TRAC and HOTSPOT are used for large-break LOCA analysis and M-RELAP5 is used for small-break LOCA analysis.

The details of these computer codes and how they are used in LOCA analysis are described in the two LOCA topical reports, MUAP-07011 and 07013. The dose analysis for Chapter 15 is performed using the NRC's RADTRAD code.

There are our open items for section 15.0.2 related to the NRC review of the relevant methodology topical reports. As you know, the topical reports for small-break and large-break LOCA are being discussed during this ACRS meeting.

Section 15.0.3 provides an overview of the dose analysis performed for Chapter 15. The dose evaluation methodology is based on Regulatory Guide 1.183.

Doses are evaluated for the exclusion area boundary, the outer boundary of the low population zone, the main control room and the technical support center.

This section describes the source term assumptions, atmospheric dispersion factors, dose conversion factors and airborne radioactivity removal coefficient.

The details of a representative dose analysis will be provided later in this presentation, during the section 15.6 discussion.

CHAIR STETKAR: I had a question on the dose analysis general information in 15.0.3, and tell me if it's better to discuss it in 15.6. But let me ask you.

There are two questions. One, they are related to controlling the pH in the RWSP water after an event. One question is something that we had from an earlier meeting and that is why is it necessary for the US APWR to operate with a 4,000 ppm boron concentration in the RWSP?

We asked that question during an earlier meeting and we were told, well MHI would answer that as part of the accident analysis. So I'm asking

that question again.

It's related -- I'd like to know the answer to that question of why you need that boron, but that is then related to the control of the pH in the water during an event response, and the control of the pH depends on the rate at which the sodium tetraborate, whatever it is, NaTB, is dissolved.

In the accident analysis, it seems that you assume that the buffer is dissolved over a 12-hour period -- I believe I read that somewhere -- which requires 12 hours' continuous containment spray operation.

And I had a question about whether or not that's a realistic assumption considering how your operators may be instructed to really operate the containment sprays.

In many plants the operators are instructed to operate the containment sprays only until containment pressure is reduced below the actuation set point and then to turn off sprays, and that's not continuous operation over a 12-hour period.

So I'm not sure whether you had the answers to those questions now or whether you may

address them in 15.6, but I had those two questions, and they are -- one is related to accident response, but it has a feedback into controlling the pH in the sump, because of the high boron concentration.

MR. MARUYAMA: So, but your question is firstly why do we need 4,000 ppm --

CHAIR STETKAR: Yes, compared to most other plants that are in the 2,000 to 2,500 ppm range, what particular feature of the US APWR design requires that much, much higher boron concentration.

MR. FUJISHIRO: Used to be the Westinghouse plant and also operating plant in Japan had boron concentration is 2,000 ppm.

CHAIR STETKAR: Right.

MR. FUJISHIRO: However, after the United States and also in Japan eliminated boron injection tank, at that time MHI increased the RWST boron concentration 3,500 ppm or 4,000 ppm. It's --

CHAIR STETKAR: Well, but there are many operating plants in the United States that eliminated the boron injection tank and kept their boron concentration at 2,000 to 2,200, 2,300 ppm, without doubling it.

MR. FUJISHIRO: Yes, I know. And it depends on the core characteristics in some plant in Japanese domestic plant, MOX core or high burnup. So we increased RWSP boron concentration.

CHAIR STETKAR: Well, what I'm trying to understand is that, you know, it isn't always necessarily true that more of boron is better for everything. And I'm trying to understand whether -- whether there's a --

MR. FUJISHIRO: There is no such critical issue in the APWR control than -- as concern over the accidental, transient and accidental release is the higher boron concentration can put in the core more negative radioactivity.

CHAIR STETKAR: That's true, but it increases the likelihood of boron precipitation with a need for, for a hot leg recirc, and also it increases the acidity in your RWSP, which means you need to add much more buffer at some rate to maintain a mutual or slightly basic, and you are very, very slightly basic.

MR. FUJISHIRO: Yes, in Japanese regulations, they are a little bit too strict to getting back some critical after steam line break,

so --

CHAIR STETKAR: Well, but what I'm interested in -- we are reviewing this design for certification in the United States, and I'm simply asking a technical argument of what's the basis for that 4,000 ppm? In other words, do you need 4,000 ppm? If you do not need 4,000 ppm, what is -- what is the minimum concentration that is necessary to mitigate the spectrum of events for this particular design, considering your core design, considering your burnup, you know, all features of this design.

And I'm trying to understand that. I'd like to understand that.

MR. FUJISHIRO: Steam line break, hot zero power case, it -- I remember 1.5 something, there is still a little bit too margin to safety analysis limited. However, I don't know how much we can make a deduction.

Maybe we can, but I don't know.

CHAIR STETKAR: Perhaps we could take that as an open item, if you had some notion of what margin you have in there. I mean, even if it was 3,000 ppm versus 4,000 ppm --

MR. FUJISHIRO: Optimization of RWST?

CHAIR STETKAR: Essentially that's what I'm asking about, is if you have looked at that or if you could give us some insight about -- recognizing it probably is a steam line break that's your limiting event for boron concentration, but if you could give us some sense of what -- how much margin you do have with the 4,000 ppm, it would help.

MR. FUJISHIRO: That is, we need to basically -- we must calculate the DNBR of the steam line break. It's based on the 3D calculation of the core and also using some by power, the MARVEL, ANC and the bypass, then. It takes time, sorry.

CHAIR STETKAR: Well, let's see, if you can, perhaps we can get some information.

MR. MARUYAMA: Yes, we understand your comment, so then as Fujishiro-san said, we confirm that 4,000 ppm has enough margin in calculation of steam line break analysis, so we don't know how --

CHAIR STETKAR: That's right. It's clear that you do have enough margin. The question is, the question is, can you reduce that and by reducing that, provide additional margin for some other functions, and particularly boron

precipitation and control of pH in the sump water.

MR. SCHMIDT: This is Jeff Schmidt from the staff. One of the reasons maybe for their plant, and I know it's where the plant I came from was, is that the 4,000 ppm is set sometimes for refueling. If you remove your control rods while you still have fuel in the pot, to reach k = 0.95 you may have to have a higher boron concentration.

You know, a lot of the Westinghouse plants, for example, keep their control rods disassembled and put in the fuel. So the 4,000 may not be a safety analysis value, but may be a refueling convenience if they remove their control rods and store them next to the plant.

CHAIR STETKAR: That would be an answer, if that's the answer. I'm just looking for, I haven't seen it anywhere, and given the margins that I have seen, and all the analyses, it's just not clear to me why they need that much boron.

MR. SCHMIDT: Yes, I know for instance, I came from Palo Verde and that was the reason why

they have 4,000 ppm still today, is because they --

CHAIR STETKAR: Palo Verde runs with 4,000? I didn't know that.

MR. SCHMIDT: Yes.

CHAIR STETKAR: Okay.

MR. SCHMIDT: And their k was 0.95. That's what they --

CHAIR STETKAR: I didn't know that. Thank you. The other part of my question was let's suppose that 4,000 ppm is, you know, it's certainly the value that's included in the design right now. I believe, if I read the accident analysis, it says that the buffer is dissolved over a 12-hour period, and because the buffer relies on the containment spray flow to flow through the baskets, that that presumes the containment spray flow will be maintained constant for 12 hours, which at least in current operating plants that I'm familiar with, is not consistent with the guidance that the operators are typically given. Guidance that the operators are typically given in emergency operating

procedures is to monitor containment pressure, and when containment pressure goes below, either below the initiation set point or with some margin, to either reduce or completely stop containment spray, reset it, and you know, if containment pressure rises again, then to initiate spray.

I have no idea how -- I know you do not have emergency operating procedures for the US APWR yet. I read somewhere that emergency response guidelines, at least ERGs were being developed. I don't know whether they are available yet or whether they have that level of detail.

But if the analyses are including debt for operating containment spray for 12 hours, and the operators really would not operate the containment spray for 12 hours, I think we'd like to understand how the analyses include credit for full dissolution of the buffer in the sump water, and whether or not those -- the analyses are consistent with what the operator's guidance would actually be.

MR. HAMAMOTO: This is Hiroshi Hamamoto.

We assumed that 12 hours time. As you see, the figure 6.3-10, and the line under 6.3-10, there is cooperation with containment and the basket configuration.

CHAIR STETKAR: Yes.

MR. HAMAMOTO: And also, 6.3-10 is a spray that is part of the spray water is filled up containment vessel.

CHAIR STETKAR: Yes.

MR. HAMAMOTO: We assumed there are minimum spray through, assuming, even if the minimum spray flow fills up containment, 12 hours is sufficient to dissolve all of the energy.

We have 23 energy containment and I forgot the detailed numbers, what we assumed that spray minimum flow under time under dissolved time. I confirm.

MR. SPRENGEL: Okay. I think we understand your question.

CHAIR STETKAR: I was going to say, I understand all of that. I know you only have two

spray trains running and you took the minimum spray flow, but my point is that --

MR. SPRENGEL: What if you stop.

CHAIR STETKAR: Suppose you stop, and in most plants, at least that I'm familiar with, the EOPs tell people to stop.

MR. SPRENGEL: And that's zero, right. We'll take that as a question.

CHAIR STETKAR: Or half.

MR. SPRENGEL: We'll have to follow up on that.

CHAIR STETKAR: I mean, it's an integration between the emergency operating procedures and what's in the accident analysis, is really what I'm asking about. Okay. You want to say something?

MR. YOSHIDA: Our calculation --

CHAIR STETKAR: Make sure -- you have to say your name also, so we have you on the record.

MR. YOSHIDA: Oh, sorry.

CHAIR STETKAR: That's fine. These

microphones are very sensitive, so you can speak here.

MR. YOSHIDA: I am Tsuyoshi Yoshida from MHI. In our pH analysis of the site, we don't consider operator manual action. We assume safety injection signal is actuated, spray is actuated and automatically pH is going up.

CHAIR STETKAR: I understand that, but in my sense, the accident analyses should also consider -- in this case, not considering the operator action to stop spray as a function of containment pressure may result in optimistic results from your accident analysis.

In other words, assuming that spray comes on, minimum spray flow is available, only two trains and so forth, but that it stays on for at least 12 hours, that assumption allows you to dissolve all of the buffer over that 12-hour period, but if that assumption is optimistic with respect to normal operator performance, then it leads to a question about how reasonable is your accident

analysis.

This is not necessarily an operator error. It is an operator doing what the operator is normally instructed to do, at least from other plants.

Now you know, you can write procedures any way you want to write procedures, but --

MR. SPRENGEL: Right, but you just want to confirm that the analysis, though, is --

CHAIR STETKAR: Is not optimistic.

MR. SPRENGEL: Right, is appropriately conservative. So --

CHAIR STETKAR: Is appropriately conservative considering expected operator response, because other analyses that you perform, for tube rupture and a few other transients, account for expected operator performance.

MR. SPRENGEL: Okay. Okay. We'll follow up on that.

CHAIR STETKAR: Okay. Thanks.

MR. MARUYAMA: Thank you very much,

okay, can I move on?

CHAIR STETKAR: Yes.

MR. MARUYAMA: Okay, now we are in slide number 14. This table shows all of the Chapter 15 events that have been analyzed for radiological consequences. Each event is classified as a postulated accident. The RADTRAD computer code is used in each case. There are no open items for section 15.0.3 now.

This table shows the increase in heat removal event in section 15.1. To save time, MHI will not discuss each event in detail. Instead, a representative event in each of the major sections will be presented, as indicated by the red text in the table.

The representative events were generally selected as a limiting event in a section. For the increase in heat removal events, MHI has selected the steam system piping failure in section 15.1.5.

A steam system piping failure will cause a large increase in steam flow. This will result in

a very large decrease in core inlet temperature due to the increased heat transfer in the steam generator.

MHI analyzes a lot of -- a hot standby and a hot full power case. For the hot standby case, the largest break size without loss of offsite power is the most limiting scenario.

The radioactivity increases until criticality occurs, and the reactor returns to power. Safety injection actuation occurs on a low steam line pressure signal.

The borated water from SI reduces reactor power and restores subcriticality. The DNB analysis shows that DNBR remains above the safety analysis limit, and therefore no fuel failure is predicted.

Both reactor coolant system and SG pressures are not limiting, because they decrease during this event.

For the hot full power case, the most limiting break size is an intermediate break of

approximately 0.4 square feet.

Since the reactor is already at power, the colder coolant result in an increase in reactor power above the full power. Eventually, reactor trip occurs on overpower delta T for the break size. So reactor trip protects against DNB such that the DNBR remains above the safety analysis limit, throughout the event.

Again, RCS and SG pressure are not limiting as they decrease during the event.

In addition to these analyses, MHI also uses a steam mass and energy release result in order to perform a containment integrity analysis in Chapter 6.

There are no open items for section 15.1.

This table shows the decrease in heat removal event in section 15.2. As before, the red text indicates the representative event that MHI will discuss in detail.

For the decrease in heat removal events,

MHI has selected the feedwater system piping break in section 15.2.8.

CHAIR STETKAR: Before you do that, because I have some questions about that, the loss of external load analysis that you performed, accounted for loss of offsite power at the time -- I think it's three seconds after the reactor trip. Is that correct?

MR. FUJISHIRO: Well, there is -- you mentioned LOOP, loss of offsite power.

CHAIR STETKAR: Right.

MR. FUJISHIRO: However, this is initiating event is, precisely speaking, loss of external electrical load demand.

CHAIR STETKAR: Okay, I understand that. Let me then ask the question that I really want to ask. Have you evaluated an event that is initiated at T time T=0 by a loss of all non-essential AC power?

MR. FUJISHIRO: That is loss of offsite power, different initiator, and this one is a --

CHAIR STETKAR: No, I understand that. That's --

MR. FUJISHIRO: Oh, okay.

CHAIR STETKAR: What I'm asking is, I think I do understand what you analyzed with loss of external load, and that is not loss of offsite power. It's loss of -- loss of something out there that will accept the electrical load.

What I'm asking is, in any of your analyses, have you evaluated an event that starts at zero with no non-1E AC power, Such that the turbine must then trip, the reactor must then trip, the RCPs start to coast down at T=0, that you don't have any force flow for some period of time. And I am not sure whether you have evaluated that condition. That condition, obviously, can occur. It happens at plants.

MR. MARUYAMA: Yes, the RCS, we have a variation that assumed that loss of offsite power at time=0.

CHAIR STETKAR: You do.

MR. MARUYAMA: So that variation is in section 15.2 that --

CHAIR STETKAR: That's part of the LNEP? That's part of LNEP?

MR. MARUYAMA: Yes.

CHAIR STETKAR: Okay. Okay. I thought it might have been, and I was hoping you were going to say that. I just couldn't' necessarily draw all of the lines together from what was in the DCD. So that is part of LNEP.

MR. MARUYAMA: Yes.

CHAIR STETKAR: Okay, thank you. And now I'll let you talk about feedwater pipe breaks.

MR. MARUYAMA: Okay. So, then, we will talk about 15.2.8, the feedwater piping break. The analysis for the feedwater break assumes that feedwater is lost at time 0 that feedwater pipe break is assumed to occur at the same time as the SG water level reaches -- the low SG water level reaches a reactor trip analytical limit.

During the resultant heatup, the

pressurizer safety valves and the main steam safety valves open to reveal primary and secondary pressure. As a result, the RCS and SG pressures remain 110 percent of the design pressure.

The analytical -- the analysis shows that DNBR decrease but the DNBR remains above the safety analysis limit throughout the event. This event also results in an increase in pressurizer water level. The analysis results shows that the pressurizer does not overfill and therefore no water relief occurs through the pressurizer safety valves.

Radiological consequences of this event are not analyzed because they are bounded by the steam line break analysis due to the larger steam release for that event.

MEMBER POWERS: I'm curious about your second bullet. Suppose I have a double-ended feedwater line break with steam water -- steam generator water level above the low steam generator water level reactor trip, what happens in that case?

(Off the record comments)

MR. FUJISHIRO: We have -- conservatively at time 0, all steam generator no main feedwater. That the story is if we -- imagine we assume the some intermediate size of feed line break, then all main feedwater will spill out that hole.

Then no main feedwater or steam generator. Only main feedwater goes out from break area.

MEMBER POWERS: I guess what I'm asking really is suppose that I don't trip on break.

MR. FUJISHIRO: It is a very conservative scenario to cover the small or intermediate size of feed line break.

MEMBER POWERS: I guess the question I'm asking, is that the conservative --

MR. LYNN: I think one of the things is that this is a heatup event, so what we are trying to do is maximize the possibility for heatup. So if you have a feedwater line break that happens at time 0, it would obviously affect that faulted SG, and it

would drain the water level in that SG very quickly, and you would get a reactor trip, and the intact SGs would be relatively unaffected and their inventory would then be available to help cool down the plant.

Now, in the scenario we analyzed, we drain all of our SGs down to that low, low level, and then have a break such that when the reactor trip does occur, all of the SGs, including the unaffected ones, have the least amount of inventory possible that you could postulate, and so therefore you get more of a heatup than the other case --

MEMBER POWERS: What I'm asking you, is follow your scenario exactly, but now give me two feet of water in the steam generators so that I don't -- so that I get some time at power before I trip. Where is the worst case in that scenario? Is it in fact the case you have selected, or is it a somewhat higher water level?

MR. FUJISHIRO: Guillotine break at time 0 guillotine break, the steam generator water level is going down very fast, then maybe after 10 seconds

the reactor trip signal will come in.

At that time, intact steam generator have lot of water inventory, then long time it is not remitting. Then reducing the break area from routine break, the -- after the steam generator water level is close to intact steam generator, water level goes down. Then more small case, maybe all main feedwater will spill over, then all steam generator going down, same slope and hit the lower steam generator or at the same time, it is the most severe case and after that, the reactor trip, we assume guillotine break.

This one is -- covers intermediate break.

MR. MARUYAMA: So, we believe this case is most conservative case, compared to the case we assumed that the -- in fact, rupture this overfill -- not overfill -- spillout occurs at the time 0.

MR. FUJISHIRO: No, we not to assume the blowdown from rupture of the steam generator before reactor -- because if we assume some flowdown

rupture of the steam generator, that steam generator will -- water level going down fast, and the heat to the reactor trip set point earlier, at that time, intact steam generator have more water inventory, and long-time, intact steam generator water inventory will be consumed by decay heat.

Rupture of the steam generator sooner or later will dry out, then that doesn't matter. It -- our concern is, at that time, at that point, is that intact steam generator with an inventory minimum case is worst case. Then --

CHAIR STETKAR: You want to think for a minute? Let me ask him another question if --

MEMBER POWERS: Please.

CHAIR STETKAR: Give you some thinking time. I had a question also about this event. And that question is whether it is optimistic from an energy release into the containment, the last bullet on your slide says, "The radiological consequences from this event are bounded by a steam line break due to the larger steam release," larger energy

release.

The -- if you start this with a loss of all main feedwater flow, allow levels to go down to the trip set point and then break the feedwater line, by definition because you started with no main feedwater flow, there is no main feedwater flow.

If I go break that line, at time T=0, and I'll give you nominal steam generator levels, you are right, the broken steam generator level will fall more quickly than the other three.

You will get a reactor trip signal perhaps a little bit earlier. But I still have main feedwater pumping a lot of energy into the containment through that broken feedwater line, a lot of energy, and that doesn't stop until main feedwater is isolated.

Main feedwater isolates automatically on, in this case an ECCS actuation signal, which has some delay time associated with it. I don't know when it comes in.

The question I have is, is it really

true that a main steam line break deposits more energy into the containment than a main feedwater line break at T 0 with delayed isolation of that main feedwater line?

I have seen analyses of other plant designs where indeed the limiting event is the main feedwater line break because of the delayed feedwater isolation, so I was curious, in your particular analysis, you don't need to consider that because by definition, you started with no main feedwtaer, so there is no need to isolate main feedwater. There is zero flow.

So I was curious whether you actually looked at that other case to assure yourself that indeed, you are still limited by the steam line break. That's a different -- that's a little bit different topic than what Dana was asking about.

MR. FUJISHIRO: Feedline break, main feedwater spill over the containment, however relatively low energy compared to steam release.

CHAIR STETKAR: For steam.

MR. FUJISHIRO: Yes, the -- MHI currently believes a steam line break is limiting, however we have not evidence right now. Do you have something?

MR. MARUYAMA: Yes, actually, the mass and energy release analysis is a form -- described in Chapter 6 and in Chapter 6.

CHAIR STETKAR: Yes, I mean we haven't reviewed Chapter 6. That's a -- I try to bounce back and forth a little bit, but you run out of time after a while.

(Simultaneous speaking.)

CHAIR STETKAR: In Chapter 6 you did? Okay.

MR. LYNN: We have both analyses and the steam line break is limiting. The only question, then, that we are trying to determine, is whether the assumptions for the Chapter 6 analysis are the same as this.

CHAIR STETKAR: That's -- I couldn't, I couldn't -- I couldn't find that quickly.

MR. LYNN: And so I think we may have to look that up and get back to you.

CHAIR STETKAR: Okay. Okay. I tried to check. I did go look at Chapter 6, and at least quickly, what I could find there, I couldn't determine exactly what those assumptions on the feedwater line break were. So thanks, if you can confirm that, that would help.

MR. MARUYAMA: So we will confirm the assumption in chapters -- difference in the assumptions in Chapter 6 and --

CHAIR STETKAR: Right, as long as the Chapter 6 analysis, where you really do address this issue --

MR. MARUYAMA: Yes. We think there are some differences but I will check about this.

CHAIR STETKAR: Dana, did you have anything to follow up on --

MEMBER POWERS: No, I just don't believe they have gotten the most conservative case.

CHAIR STETKAR: Yes. So the question is

still there, whether some demonstration that indeed that assumption about low initial levels is, is you know, with a trip happening at effectively T 0 when the break occurs, is that the most limiting condition.

MEMBER POWERS: The problem is they would get into troubles if they delayed the trip.

MR. WOOD: I'm Doug Wood. I have worked with these people on some analyses. I just wanted to point out, for those of you who don't have that event, the feedline break, in front of you right now, the peak -- all the peak parameters, you know, are way out in time.

This is really a global energy, you know, balance on the RCS. The event, you know, the worst case parameters for these events peak many, many, many seconds after the reactor trip, so this is really an energy balance event and the -- I think that if you think of it in that way, it's logical to conclude that the worst case situation occurs where you assume the least amount of heat removal

capability in your secondary. I just wanted to point that out so as to --

CHAIR STETKAR: For the short term --

MR. WOOD: Short term meaning hundreds of seconds, not like immediately.

MEMBER POWERS: It should be -- you should be about 900 seconds or something like that.

MR. WOOD: Right. That's correct. Yes.

MEMBER POWERS: The whole point being the few, 30 seconds' worth of additional power from the core during that time change things a lot. Ten seconds without power will change things. You described a higher level.

MEMBER BROWN: So, then we answered your question, because this is 900 seconds later like you --

MEMBER POWERS: No.

MEMBER BROWN: Okay, it didn't change.

MEMBER POWERS: No, the question stays the same.

MEMBER BROWN: To the uninitiated, I was

trying to connect the dots on the two. I'm done.

CHAIR STETKAR: It's strictly just a question of, even in the short term, how much energy is added.

MEMBER BROWN: I understand that part of it. I just --

MR. MARUYAMA: Okay, I'll go on to the next page.

MR. SPRENGEL: Real quick, Yuta. Does MHI understand the outstanding question?

MR. MARUYAMA: Regarding the timing of the --

MR. SPRENGEL: I don't know what the question is. That's why I asked.

CHAIR STETKAR: Dana, if you could --

MEMBER POWERS: Well, I guess just how you perturb this, this thing. What they assume here is an all steam generator comedown to this level set point. They get a break, a double-ended guillotine and instantaneously trip the reactor.

CHAIR STETKAR: With some minor time

delay.

(Simultaneous speaking.)

MEMBER POWERS: No, all I'm asking is that we in fact consider suppose you came down to a foot above the low level at a double-ended guillotine type break, and so now that some time delay becomes a little longer time delay, and if it goes long enough, then clearly this is more limiting. But is there an intermediate case that is in fact more limiting than this instantaneous time trip? It's a very simple question. The alternative that's been posed is in fact the other steam generators are not low, okay? And I will admit, this is more conservative than those. But suppose -- I'm just following from their hypothesized situation except saying they didn't quite get that low, so you have a little longer period of power, so you put a little more oomph into the system before you trip.

At what point are you most conservative?

(Off the record comments)

MR. LYNN: So just to clarify, you agree that starting at -- starting the break at time 0 with all of the SGs at their nominal levels would be less limiting than what we presented, but you are saying somewhere between here and there might be --

MEMBER POWERS: Well, in between the two --

MR. LYNN: Okay.

MEMBER POWERS: have you looked to see if in fact you found the most limiting case or not? Because you are right, I mean, as long as those other generators are full you have got all inventory to help your cooling down. When does that water stop being your friend and start being part of the pressurization? Maybe you are absolutely correct for all I know, but it would surprise me if you were absolutely correct.

I am often surprised. Yes. Well you didn't need to agree so enthusiastically.

CHAIR STETKAR: Well if you admit it.

MEMBER POWERS: I admit it, but it

should be viewed as something of humility on my part and not an obvious truth.

CHAIR STETKAR: Humility and power are never used in the same sentence. Do you actually -- do you understand the concern? It's not necessarily saying you know, we are not saying start it, you know, from nominal level, because it's clear that --

MR. MARUYAMA: So you're asking if there is intermediate conservative case or not. So --

CHAIR STETKAR: Could there be some intermediate level that indeed the combination of somewhat reduced heat removal from the other steam generators, with the additional delay time for the reactor trip, are more bounding than what you used.

MR. MARUYAMA: You are asking intermediate size break or --

CHAIR STETKAR: No, take the big break but start it, instead of -- you started it at this level. If this is the normal level, is there some intermediate level in here that may be more limiting --

MR. MARUYAMA: Oh, however, steam generator water level is controlled by feedwater control system and as initial condition, it should be a condition one situation then steam generator water level should be programmed level. We must take account some failure in feedwater control system. It's quite a real case. And it should not be considered as an accidental condition.

If we have a feedwater controlled system have some there, then we cannot keep the steam generator water level trip either.

MR. SPRENGEL: Okay, we'll take some time to discuss. I think we understand the point, some of us understand the point.

CHAIR STETKAR: Okay.

MR. SPRENGEL: And we'll discuss internally. We'll move on for now.

CHAIR STETKAR: That might be an argument, in terms of this is an accident analysis.

MEMBER BROWN: I'm trying to relate from my past to this one and each generator has its own

control system, so that three -- when you have a break in one, doesn't mean that the other ones don't try to drive the other one up so that you don't break any of those down to generate a trip.

So I don't know how many trips it takes, just one? So it's when the one that has the guillotine break comes down, it's not a coincidence type thing for that, for these -- for this plant, for these designs.

CHAIR STETKAR: Low level in any one --

MEMBER BROWN: Any one.

CHAIR STETKAR: Any one of the four will --

MEMBER BROWN: Okay, but I mean, so if they start right at the trip point you don't have time then for the other one to be driven up and then drive yourself into the scenario that Dana is talking about.

CHAIR STETKAR: That's why they assumed the loss of feedwater event at time minus or something, if you want to think of it, as some

rational way to get all of the levels, all of the levels down.

MEMBER BROWN: I just wanted to relate to his comment, relative to the level control systems interaction, and now I understand what the difference is. I thought it was a coincidence, the two of them, and so not understanding that changes the --

CHAIR STETKAR: Low level in any one of the four steam generators will do it for you.

MEMBER BROWN: Okay, I'm done. Thank you.

MR. MARUYAMA: So, actually there are no open items for section 15.2 now. The next table shows the decrease in reactor coolant system flow rate event in section 15.3.

For the decrease in RCS flow events, MHI has selected the RCP rotor seizure in section 15.3.3. the seizure of one RCP rotor causes a rapid decrease in RCS flow, which results in an increase in RCS temperature and decrease in DNDR.

A reactor trip will occur on the low reactor coolant flow signal. Even with the reactor trip, the DNBR decreases below the safety analysis limit. Therefore, MHI calculates number of rods in DNP and this is significantly less than 10 percent.

MHI also calculates the peak cladding temperature. It remains below the 2,200 degree limit. RCS and SG pressure both remain below 110 percent of the system design pressure.

A dose analysis is then performed by assuming 10 percent of fuel rods have failed. This fuel failure bounds the actual analysis results described above.

MEMBER POWERS: When you do a dose analysis where 10 percent of rods failed, what do you do? What do you release from the failed rods?

MR. FUJISHIRO: Would you ask me again? Ten percent of fuel rods --

MEMBER POWERS: You have failed -- 10 percent failed rods --

MR. FUJISHIRO: Maybe some credit

calculations, some things that we assume that were going to fail, and account in radioactivity calculation.

MEMBER POWERS: What I'm asking is what is the radionuclide release you associate with 10 percent railed rods?

MR. YOSHIDA: This is Tsuyoshi Yoshida with MHI. We assume radioactivity from broken fuel rod released into RCS and primary to secondary leakage is assumed, so contaminated tube water leaks to secondary site, and from main steam relief valve contaminated steam is released. That is our assumption.

MEMBER POWERS: What radionuclide release do you assume from the rod itself?

MR. YOSHIDA: We assume noble gas and iodine and cesium.

MEMBER POWERS: Just a gap inventory?

MR. YOSHIDA: Pardon?

MEMBER POWERS: Just a gap inventory?

MR. YOSHIDA: Yes.

MEMBER POWERS: Then you take that gap inventory to be what?

MR. YOSHIDA: Gap -- pardon?

MEMBER POWERS: What fraction of the inventory of radionuclides are in the gap?

MR. YOSHIDA: Ah, okay. We assume 10 -- based on radioactivity at 1.13, we assumed --

MEMBER POWERS: Five percent.

MR. YOSHIDA: Yes.

MEMBER POWERS: And so you don't leach anything. You don't leach anything out of a bit of fuel. No.

MR. MARUYAMA: There are no open items for section 15.3 now. Okay the next table shows the reactivity and power distribution anomaly events in section 15.4.

In this case MHI has selected two events. MHI has selected the control rod assembly withdrawal at power event as an AOO to present. MHI has also selected the rod ejection as a PA to present.

Rod withdrawal at power insert radioactivity to the core, and therefore increases with the power from its initial value. MHI analyzes different combinations of radioactivity insertion rates, BOC and EOC and initial powers to determine the limiting case or cases.

The event is terminated by a reactor trip. The reactor trip that occurs depends on the conditions of the particular case, but it is one of the four trip signals listed here.

The DVD provides transient results for two cases -- a high reactivity insertion rate case an a low reactivity insertion rate case. The reactivity rate case is more limiting from a DNBP perspective.

For both cases, DNB decreases until reactor trip occurs, but remains above the safety analysis limit such that fuel failure is not predicted.

This event is not limiting for either RCS or SG pressure.

MEMBER POWERS: What kind of energy deposition do you get in the fuel?

MR. MARUYAMA: Excuse me, could you say that again?

MEMBER POWERS: What kind of energy deposition to get in the fuel?

MR. MARUYAMA: Energy deposition?

MEMBER POWERS: Right. You've got a reactivity insertion, you are putting energy suddenly into the fuel, how much do you put in?

MR. MARUYAMA: Well, we evaluate the fuel centerline temperature and it is a little bit to increase, but energy deposition is basically the criteria for the reactivity initiating event. That is from zero power.

There is no such criteria if you interpret something. There is no. Only we are looking for the fuel melting, not melting. That is -- may I continue now?

MEMBER POWERS: You don't get centerline melting?

MR. FUJISHIRO: No. The power is very increasing slowly and at maximum power is a little bit overshoot, maybe 120 percent, something the -- not so much big energy deposition and fuel centerline temperature well below the melting point.

MEMBER BANERJEE: The DNBR calculation is based on full-scale experiments?

MR. FUJISHIRO: You mean the DNBR test itself?

MEMBER BANERJEE: Yes, well the 14-foot test, did you do?

MR. FUJISHIRO: It is already done I think.

MEMBER BANERJEE: Finished? It was full-scale? Who can tell us?

MR. MARUYAMA: Yes, we did full-scale test.

MEMBER BANERJEE: Full scale with -- where was it done?

MR. MARUYAMA: Where?

MEMBER BANERJEE: Full power, full

scale? It's a big, big LOOP.

MR. SPRENGEL: The DNBR testing that was done, was done in Germany.

MEMBER BANERJEE: Okay, at AREVA?

MR. SPRENGEL: At an AREVA facility, yes.

MEMBER BANERJEE: Because this is a large facility. This was done at full scale, so did you look at transient effects as well?

MR. MARUYAMA: Transient effects?

MEMBER BANERJEE: Because I guess --

MR. SCHMIDT: This is Jeff Schmidt with the NRC. They looked at a spectrum of cases to bound the Chapter 15 events and their cases were confirmatory in nature, using a certain existing --

MEMBER BANERJEE: Correlation.

MR. SCHMIDT: correlation, yes.

MEMBER BANERJEE: Which we can't name.

MR. SCHMIDT: Which I cannot name, which shall remain nameless, at least until tomorrow. So --

MEMBER BANERJEE: It starts with a B.

MR. SCHMIDT: So the transient cases, or the cases cover the transient cases and it was five by five array, 14 feet high.

MEMBER BANERJEE: Not a 17 by 17.

MR. SCHMIDT: No it was not a 17 by 17.

MEMBER BANERJEE: And you are satisfied with this five by five area?

MR. SCHMIDT: Yes, it's typical DNBR testing, yes.

MEMBER BANERJEE: All right. Will you be speaking to these tests later? I mean, staff will give us some of your views on these AOOs?

MR. SCHMIDT: I mean it's covered as part of the VIPRE topical report. That's where the DNBR testing is referenced and VIPRE is referenced in Chapter 15.

MEMBER BANERJEE: Okay.

MR. SCHMIDT: But you know, mainly the discussion of the actual testing will be done under the VIPRE topical report.

MEMBER BANERJEE: Okay, thank you.

MR. MARUYAMA: Okay, next page. This page. Page 29. MHI has also chosen rod ejection postulated accident as a representative event for section 15.4. A rod ejection causes a large reactivity insertion, and also large power peaking.

The initial power increase is mitigated by the Doppler feedback effect. Reactor trip will also occur, but reactor trip that occurs depends on the condition of the particular case, and can be any one of the four trip signals listed here.

MEMBER BANERJEE: For the calculations do you use a spatial kinetics code which is coupled or you decouple the spatial modes from --

MR. MARUYAMA: No not coupled.

MEMBER BANERJEE: Not coupled. So you do, you have a time and space separate calculations?

MR. MARUYAMA: I mean, the time and space -- we use 3D calculation for HOTSPOT case, this is time and space combined. But we mean the -- some are calculation and kinetic calculation is not

coupled.

MEMBER BANERJEE: But your rod ejection is a fully coupled code, right? Or not.

MR. FUJISHIRO: He explained our calculation covers 3D, however initial temperature is constant and the pressure is also constant, and internally, there is of course thermal hydraulic and also 3D nuclear calculation, to include 3D calculation.

MEMBER BANERJEE: The 3D nuclear calculation is a transient 3D?

MR. FUJISHIRO: Yes.

MEMBER BANERJEE: The nuclear calculation is fully transient 3D? But it's not coupled to the thermal hydraulics.

MR. OGAWA: Not coupled with the fuel calculation, sorry.

MEMBER BANERJEE: Not coupled. So let me understand what it is coupled to.

(Off the record comments)

MR. MARUYAMA: Sorry, I'd like to modify

what was said. The fuel calculation on the 3D kinetic calculation is performed at the same side as also, as we explained, the 3D transient calculation, using TWINKLE-M code.

MEMBER BANERJEE: Okay. So the fuel calculation, the temperatures, everything, get fed into this 3D neutronic calculation? It's fully coupled. Everything is coupled, temperature as well as the --

MR. FUJISHIRO: Fuel temperature and coolant temperature.

MR. BIELEN: This is Andy Bielen from the staff. They start with the neutronics calculation, with TWINKLE-M, so they do a full 3D kinetics calculation. Then they make some adjustments to it and put it into VIPRE to do temperature evaluation.

MEMBER BANERJEE: And that's fed back or --

MR. BIELEN: No. Nonono. It's one way -- one way.

MEMBER BANERJEE: Thank you.

MR. BIELEN: Sure.

MEMBER BANERJEE: It's clear what you did. Thank you.

MR. MARUYAMA: Okay, so where am I? Now, so, RCS pressure --

MEMBER BANERJEE: Does this give you a conservative --

MR. MARUYAMA: Yes.

MEMBER BANERJEE: Conservative bound on the effect, right, of the rod reaction?

MR. MARUYAMA: You mean the one-way calculation?

MEMBER BANERJEE: Yes.

MR. MARUYAMA: Yes. If we considered feedback from the thermal hydraulic calculation, that will, you know, reduce the --

MEMBER BANERJEE: All right.

MR. MARUYAMA: power so that is conservative.

MEMBER BANERJEE: Okay.

MR. MARUYAMA: The rod is -- so may I continue?

MEMBER BANERJEE: Yes.

MR. MARUYAMA: Okay. The rod ejection will also cause a breach of the reactor coolant pressure boundary that will result in a decrease in RCS pressure. MHI analyzed several cases in order to consider different combinations of EOC and BOC along with initial power.

For the full power cases, BOC and EOC, MHI uses the measured neutron flux detector power considering a single failure to determine when the reactor will trip.

For these full power cases, the peak fuel centerline temperature and other fuel pellet enthalpy at the hot spot remain below their respective limits.

For the zero power cases, the react will trip on the high neutron flux low setpoint. For these zero power cases, the average fuel pellet enthalpy at the hot spot under hot spot peak fuel

enthalpy remains below their respective limits.

The zero power cases also look at PCMI. The analysis results show no PCMI failed, no PCMI failed the fuel rods.

MHI also performs analysis to determine the number of rods in DNB for the hot full power cases. In the DNB analysis, the limiting case occurs when the reactivity insertion is such that reactor trip on neutron flux does not occur.

Instead, reactor trip occurs later, during the RCS depressurization caused by the breach of the reactor coolant pressure boundary. No DNB occurs during the initial prompt power increase. However, DNB eventually decreases below the safety analysis limit during the RCS depressurization.

MHI calculates the number of rods in DNB and it is less than 10 percent. MHI performs additional analysis for RCS pressure, thus shows that RCS and SG pressures remain below 110 percent of the design pressures.

MEMBER BANERJEE: How do you establish

this 10 percent? Do you have a probability distribution curve of some sort? This is right in the tail side, right, of the curve? Like it's just 10 percent means if you have a curve like this, it's just at the tail right? So how do you establish this? I know there is a well-established methodology, but are you, what do you --

MR. MARUYAMA: Yes, actually for DNB failure calculation, first we calculate the consensus, consensus if there is an HN. Then, on the other hand, we calculate the limiting that reaches the DNBR in transient calculation.

So comparing those two results, we will find a fuel that have more -- that have more than limiting, then we assume that those fuels are failed.

So such kind of methodology is described in detail in non-LOCA topical report methodology.

MEMBER BANERJEE: Because you are in a very small region.

CHAIR STETKAR: We'll see that report in

October.

MEMBER BANERJEE: Okay.

MR. LYNN: The relevance of the 10 percent is that 10 percent is the number assumed in the dose calculation, so --

MEMBER BANERJEE: I know that you have this, but it's a question of how you establish it. That's -- but as John says, we will see it.

CHAIR STETKAR: Yes, it's just a problem in terms of timing of our meetings and staff reviews and things that are beyond our meager control at the moment.

MEMBER BANERJEE: Okay.

CHAIR STETKAR: And that's something that we will not discuss tomorrow, that's not on the table. So --

MEMBER BANERJEE: Fine, as long as it's discussed at some point.

CHAIR STETKAR: Yes,, it's on our agenda for October.

MR. MARUYAMA: Okay, then we will

discuss our methodology in detail. Okay. Then in page 32, there are three open items for section 15.4.

For each of our items, MHI has provided a response to the associated RAI. These responses are currently under industry review.

And this table shows the increase in reactor coolant inventory events in section 15.5. In this case only the CVCS malfunction event in section 15.5.2 is applicable to the US APWR.

Therefore MHI has selected this event to present in detail. The CVCS malfunction is assumed to be a full open failure of the charging flow control valve.

The boron concentration of the injected water from the CVCS is the same as RCS, therefore the main concern is an increase in pressure, pressurized water level that could fill the pressurizer.

The US APWR has an automatic CVCS isolation feature that occurs on high pressurized

water level. This automatic isolation will terminate the increase in pressure the water level before pressurizer overfill occurs.

There are no open items for section 15.5. And this table shows the decrease in reactor coolant inventory events in section 15.6. In this case MHI has selected two events. MHI has selected the steam generator tube rupture and LOCA events to present.

Both of these events are PAs.

The steam generator tube rupture event considers the double-ended rupture of one SG tube. MHI analyzes two cases, one to evaluate doses and the other to determine the margin to SG overfill.

In both cases, manual operator actions are necessary to terminate the event and are therefore assumed in the analysis.

These manual actions -- these manual actions include reactor trip, isolation of the ruptured SG, cooldown and depressurization of the RCS and termination of SI.

This slide provides additional detail for the dose evaluation case. A manual reactor trip is assumed 15 minutes after the tube rupture. Isolation of the ruptured SG is assumed at 20 minutes.

Then, the operators begin an RCS cooldown at 25 minutes. The remaining operator actions are performed as conditions allow, and are therefore determined by the analysis.

The event is not terminated until the primary to secondary leakage finally stops at approximately 70 minutes. The RCS cooldown is performed by releasing steam from only the ruptured SGs.

However as an additional failure the main steam release valve of the ruptured SG is assumed to open at the same time, remain stuck open until it is later isolated by operator.

CHAIR STETKAR: Is that -- I'm wanting to understand that isolation of the stuck-open relief valve, because in one place I thought it said

it was operated -- isolated by the operator. In another place it seemed to say it was isolated automatically.

Does the analysis account for operator action or is it an automatic isolation?

MR. FUJISHIRO: Main steam relief valve will have motor-operated isolation valves and it close well below the setpoint motor isolation valve is automatically closed.

If operator opens as steam relief valve, or automatically open and close steam generator MHO will maybe working something like that, and sometimes stuck open then motor operated isolation valve will automatically close it.

What does that -- you're mentioning main steam isolation valve?

CHAIR STETKAR: No, no, I'm talking about the -- they are either called -- it's the MSRV isolation valves.

MR. FUJISHIRO: Then if MSRV is stuck open --

CHAIR STETKAR: Right, and =

MR. FUJISHIRO: then --

CHAIR STETKAR: and in the analysis, that closed -- the isolation valve for the stuck-open MSRV closes at 1,826 seconds after the initiating event, approximately.

Now, in some places, in the DCD, it seems that the operator closes it, and in other places it seems that it closes automatically. What I'm asking is, is that an automatic isolation signal or is that an operator action? In the analysis.

MR. LYNN: I believe it's an operator action, because the method of closing it is to close a block valve before the valve.

CHAIR STETKAR: I understand, but that block valve also gets an automatic signal from some things, and that's what I'm trying to understand about the analysis versus the real world.

So, you're saying in the analysis, it's closed by an operator action?

MR. MARUYAMA: Yes.

CHAIR STETKAR: Okay, thank you.

MEMBER POWERS: What do you assume on the iodine spike?

MR. FUJISHIRO: Could you say again?

MEMBER POWERS: What do you assume on the iodine spike?

MR. FUJISHIRO: Iodine spike didn't halt the steam generator.

MR. YOSHIDA: Of course we assume iodine spike two case. One is pre-accident iodine spike case and the second is coincident iodine spike case, based on Regulatory Guide 1.183.

MEMBER POWERS: 500-fold increase over the water coolant.

CHAIR STETKAR: Make sure, if you are answering something, make sure you are at microphone.

MR. YOSHIDA: For iodine spike, it's based on technical specifications. We assume 60 microcuries per gram concentration, and a coincident iodine spike case based on Regulatory Guide 1.183,

335 times of iodine concentration.

CHAIR STETKAR: In your dose evaluation analysis for the tube rupture case, do you include credit for filtering through the main condenser?

MR. YOSHIDA: In both analyses, we don't consider main condenser.

CHAIR STETKAR: So any releases to the main condenser are treated as a direct release to the atmosphere?

MR. YOSHIDA: Yes.

CHAIR STETKAR: Everything? Noble gases, everything?

MR. YOSHIDA: We assume all the released radioactivity goes through main steam release valve and main steam safety valve. So most go through condenser.

CHAIR STETKAR: Well, I guess my confusion is in the DCD there's a table, table 15.6.3-3, that says the releases -- this is steam mass -- released to the atmosphere from the ruptured steam generator is 109,000 pounds released to the

atmosphere. The release to the main condenser is 1,280,000 pounds, and the difference is because of the timing that you have assumed on various automatic and manual signals.

So, my question -- and it's clear, there's a plot that shows the release to the atmosphere starting at 900 seconds and ending at 1,800 seconds, let's say, and they are released to the atmosphere from the steam -- the ruptured steam generator before 900 seconds.

Now my question is, before 900 seconds, you are releasing to the main condenser, from the ruptured steam generator. Does the dose analysis treat those releases during the first 900 seconds as a release to the atmosphere?

MR. YOSHIDA: In dose analysis, we assume release that from time -- when SG release occur. Yes. So -- but in safety analysis, I'm not sure, but I guess the result is not -- in safety analysis result, steam release is not occur the time is zero.

CHAIR STETKAR: I guess I didn't quite understand.

MR. FUJISHIRO: We got to you --

CHAIR STETKAR: Let me -- the reason I ask this is that if I follow the time line of the analysis, the time line says that the tube rupture occurs at zero, which is good. I understand that.

At two minutes, 120 seconds, it's assumed that the operators discover the tube rupture because of the nitrogen 16 alarms. It's just noted in the analysis.

At 10 minutes, they identify the ruptured steam generator and at 15 minutes, they then manually trip the reactor, close the main steam isolation valve and at the same time, a loss of offsite power happens.

Then the analysis proceeds, oh, and also at 20 minutes, they isolate emergency feedwater to the ruptured steam generator, so that 15 to 20 minute time window is an important time window.

Then, from 15 minutes, 900 seconds,

until 1,826 seconds, when the MSRV block valve is closed, you have a release to the atmosphere from the ruptured steam generator. That's clear.

My question is, if the dose analyses only account for the release between 900 and 1,800 seconds, that's one issue. If I'm an operator and at two minutes I see that I have a ruptured steam generator, I will probably trip the reactor very quickly, and if my atmospheric release then starts at two minutes or two minutes and 30 seconds and is not terminated until 30 minutes, that's a much different release than something that starts at 15 minutes. In other words I have, you know, something on the order of 13 or -- 12 or 13 more minutes of release.

So my question is, does your -- does your dose analysis only account for the 900 to 1,800 second release, or does it account for the release from zero?

MR. FUJISHIRO: I think that before 900 the -- during zero and 900, this period, steam flow

is going to the condenser. However, DF is two types, steam generator and condenser, then it should be reflected in dose evaluation.

However, we said to dose people total amount of the leaks, and it should be noble gases all the leak is discharged to atmosphere in dose calculation and also we send to those guy the safety valve and relief valve steam amount.

However, we don't send to you the zero to 900 total steam flow. However, its effect is small.

MR. YOSHIDA: Almost all he said is correct and --

MEMBER POWERS: Almost all?

(Laughter.)

MR. YOSHIDA: In dose analysis, release of steam has much value. So I think this 900 seconds steam must -- is included over impact condition.

CHAIR STETKAR: Can you repeat that, because I didn't quite understand?

MR. YOSHIDA: In dose analysis, we based our result of safety analysis, we -- our input data for dose analysis includes some margin, about 20 percent, so this 900 seconds period of release must -- steam loss -- is -- effect of this 900 second steam loss is included in our input condition.

MR. FUJISHIRO: He wanted to say the total equivalent is from time zero. The noble gases entire one is calculated in dose calculation.

CHAIR STETKAR: From zero. I understand that because I read the noble gases go out through the condenser event. I'm more interested in how you are treating iodine and other things.

MR. FUJISHIRO: The iodine --

CHAIR STETKAR: You assume one percent failed fuel in all of your leakers and things like that.

MR. FUJISHIRO: Iodine is weak, assume the -- none of the -- the contamination factor, 199, then one percent only to atmosphere --

CHAIR STETKAR: Because of the main

condenser?

MR. FUJISHIRO: Yes. Then not only the condenser, but the steam generator, right?

CHAIR STETKAR: No, I understand -- the steam generator is the steam generator. It all -- I'm not arguing about the steam generator because that's always there. What I am concerned about is are you taking an additional DF for releases, in particular for iodine, through he main condenser during that first 900 seconds?

MR. YOSHIDA: No, only steam generator water retention.

CHAIR STETKAR: Just steam generator?

MR. YOSHIDA: Yes.

CHAIR STETKAR: So you are treating from zero -- let me make sure that I understand it -- are -- from zero you are treating all of the releases, not just noble gases but iodine, cesium, anything else, as if it was a direct atmosphere release?

As if you were steaming from that -- let's say through the main steam safety valve. Is

that true?

MR. YOSHIDA: Yes, and we assume also flushed break fall, so iodine, including this flushed break fall, we assume they have to be released to the environment. Can you understand my English?

CHAIR STETKAR: I think so. And let me just say it back, that you have assumed the iodine is released offsite. Okay. From zero?

MR. YOSHIDA: Yes.

CHAIR STETKAR: Okay.

MR. YOSHIDA: Okay.

CHAIR STETKAR: So the timing of the reactor trip and the isolation -- effectively the timing of the reactor trip, what I'm trying to understand is, is the dose analysis affected by the timing of the reactor trip, and the closing of the main steam isolation valve, which changes the release from the condenser to the main steam relief valve that is assumed to stick open at that time, and from what I understand, you are saying no, that

time does not affect the dose analysis. Okay, thank you. That helps. Thank you.

MR. MARUYAMA: Okay.

MR. LYNN: Before we move on, I want to clarify. You asked earlier about the main steam relief valve on the ruptured SG sticking open and how that's closed I said that it was -- you said at 1826 seconds, and I said that it was, I believe that it was at an operator action, but I did confirm that the closure of that valve at 1826 is an automatic signal that happens due to low main steam line pressure.

CHAIR STETKAR: Thank you.

MR. MARUYAMA: The main purpose of this analysis is to determine the primary-to-secondary leakage and the release rate from the ruptured interfaces to use as input in the dose analysis as we discussed.

However, the result also shows that DNBR remains above the safety analysis limit, and RCS and SG pressures are within limits.

Both large-break LOCA and small-break LOCA event are presented as represented event for section 15.6. The large-break LOCA analysis uses the ASTRUM methodology with WCOBRA/TRAC and HOTSPOT.

The event begins with a cold leg pipe break and a coincident loss of offsite power.

CHAIR STETKAR: I'm sorry. I was writing, and I -- before we start discussing the LOCA, regarding the steam generator tube rupture, in the Safety Evaluation Report, there is some discussion about operator actions and the amount of time available and the amount of time required to perform the operator actions, and in the SER -- you'll see why I asked you in a moment her -- it says that in the MHI response to RAI 808-5921, question 15.06.03-3 -- do this so it's on the record -- and now I've lost my train of thought. Thank you so much.

The -- don't you have other things to do?

(Laughter.)

But you supplied a table that showed the times available for each of the operator actions and the time that was required, and you noted that though the available times and the instructions for the operator actions were consistent with your emergency response guidelines. I don't have the RAI response. We can get that. That's not a problem.

My question was, it was also noted in that, in the SER, that you had indicated that the ERGs would be available by the end of December 2011. My question now is are they available? Have you finished the ERGs?

MR. LYNN: I guess, to clarify your question, available to whom or --

(Laughter.)

CHAIR STETKAR: No one except himself.

MR. LYNN: I wouldn't give them to Dr. Powers.

CHAIR STETKAR: No, I -- okay. No. It is a relevant question.

MR. SPRENGEL: I can answer that.

CHAIR STETKAR: Yes, thanks.

MR. SPRENGEL: They are completed.

CHAIR STETKAR: They are completed.

MR. SPRENGEL: But they are not going to be submitted to the staff. So --

CHAIR STETKAR: And typically they are not submitted to the staff because there is support for the EOPs, which is typically a COL item which is typically after the COL is issued, and I understand that.

MR. SPRENGEL: But they are completed, and --

CHAIR STETKAR: But they are completed and they are available. Okay. But not for formal review because they are not submitted as part of this licensing.

MR. SPRENGEL: That's correct.

CHAIR STETKAR: Understand. Thank you. Okay. Thank you. I'm sorry. Now you can talk about LOCAs.

MR. MARUYAMA: Okay. Thank you. So,

about LOCA, in this stage I was explaining about the large break LOCA. The event begins with cold leg pipe break, and the coincident loss of offsite power.

ECCS actuate on low pressurizer pressure but only two of the four SI pumps, minimum safeguards are assumed. The values of peak cladding temperature, LOCA maximum oxidation and core-wide oxidation in the ASTRUM analysis satisfy the 10 CFR 50.46 acceptance criteria.

The small-break LOCA analysis uses MRELAP-5 based on the Appendix K methodology. The event begins with a cold leg pipe break. Reactor trip occurs on low pressurizer pressure and a loss of offsite power is also assumed at the time ECCS actuates on low pressurizer pressure.

But only two of the four SI pumps and minimum safeguards are assumed. The values of PCT, LMO and CWO in M-RELAP5 analysis satisfy the 10 CFR 50.46 acceptance criteria.

The LOCA event is also analyzed for

radiological consequences. A guillotine break is assumed. All fuel rods are assumed to be failed and the primary coolant activity is assumed to be at the t spec limit initially.

The radioactivity in the fuel is gradually released into containment and then escapes from containment through specific release pathways. Radioactivity is assumed to be removed from the air by containment spray, natural deposition and the filtering systems.

The results meet all of the SRP 15.0.3 acceptance criteria.

MEMBER POWERS: When you do your spray calculation, what do you assume about your spray?

MR. YOSHIDA: Pardon?

MEMBER POWERS: I was wondering about droplet size distribution, spray volume, unsprayed volumes.

MR. YOSHIDA: Yes, we assumed that 60 percent of containment volume is sprayed region and the remaining 40 percent of containment volume is in

dose analysis.

MEMBER POWERS: And this droplet -- have you picked your spray hitters yet? Your spray nozzles?

MR. YOSHIDA: No. I have not picked yet to --

MR. HEAMES: This is Terry Heames. Dana, are you talking about do they have the spray nozzles set up for this particular design as opposed to what we used for the dose analysis?

MEMBER POWERS: Yes.

MR. HEAMES: Okay. We can't answer, I can't answer that one. Do you have a particular spray nozzle that you have chosen to use the US APWR? That's his question.

MEMBER POWERS: Well really all I want to know is what the droplet size distribution is that you set, so like, if you knew the nozzle, then I'd probably know the droplet size distribution.

MS. STEINMAN: So, at this time, no we have not selected --

MEMBER POWERS: So, how do you do a spray decontamination if you don't know the droplet size distribution?

MR. YOSHIDA: I use for spray model a -- we used SRP model for particulate, and for natural division I used Powers model for --

MEMBER POWERS: What the hell does he know?

(Laughter.)

MR. YOSHIDA: -- for particulate and for natural division, for airborne iodine I also used SRP model.

MR. MARUYAMA: There are seven open items for section 15.6. All of the open items are associated with the LOCA event. The first two open items here are associated with the initial review of related technical and topical reports.

The third open item, shown here, is related to the RCP seal integrity issue. That is not thoroughly a Chapter 15 item.

CHAIR STETKAR: Yes, and we had also in

our subcommittee meetings, extensive discussion over that. So we are following that.

MR. MARUYAMA: These two items are associated with technical reports for PARAGON/ANC and for GSI-185. For these two open items, MHI has provided a response to the associated RAI. These responses are currently under NRC review.

This table shows all of the section 15.7 that are analyzed for radioactive release. Some of these events are analyzed separately as a part of Chapter 11. Only the fuel handling accident is specifically analyzed in Chapter 15 using RADTRAD.

There are no items for section 15.7.

CHAIR STETKAR: Your analysis about the cask drop says it's not physically possible in this design to move the cask over the spent fuel pool. Is that right? I think I recall that.

MR. YOSHIDA: As you guessed, our design, we applied for single failure approved claim, so spent fuel cask does not drop out to lower floor and --

CHAIR STETKAR: That's a different -- I'm not asking you about the probability of dropping it. I'm asking about the possibility of dropping it. They are two different things.

MR. YOSHIDA: And then transfer pathway of spent fuel cask does not over the spent fuel.

CHAIR STETKAR: Okay. That's what I seem to remember. Thanks. That's a possibility rather than a probability.

MR. YOSHIDA: Yes, I think I was asking a possibility question.

CHAIR STETKAR: No, if it's not possible, I can handle zero.

MR. MARUYAMA: There are no open items for section 50.7. Then this is the section 15.8 covers anticipated transients without scram, ATWS. The US APWR has an analog diverse actuation system that is diverse from the digital normal reactor trip system and engineering safety feature.

If reactor trip does not occur by the normal reactor trip system, the DAS has an automatic

reactor trip that occurs on any of the three signals showing here. The DAS also has automatic SI and emergency feedwater actuation features.

MHI has performed defense D3 coping analysis in the technical report, MUAP-07014 to evaluate chapter 15 AOOs with a concurrent common cause failure.

The D3 coping analysis of some event creates automatic DS features. In all cases, the acceptance criteria are met.

CHAIR STETKAR: Thanks, and just for the record, we have not yet reviewed Chapter 7 of the DCD. We will be doing that review, our first review of that, in November of this year. So I am sure we will be examining the DAS and automatic features at that time.

MEMBER BROWN: Yes, I wanted to make one, yes, one piece of input, in that there's only three signals that result with this system. There ought to be, when you present this, or when we discuss this later --

CHAIR STETKAR: We'll get to it.

MEMBER BROWN: There ought to be a full basis for why those three cover all other possible combinations where other reactor trips are not done and why they cover the waterfront, umbrella-wise. So there ought to be a good basis for that, as to why there's a small microcosm of the reactor trips actually will actuate from the diverse actuation system.

CHAIR STETKAR: Take that as a preview for your I&C folks who are not here who are --

MEMBER BROWN: It just gets better --

CHAIR STETKAR: not here today.

MEMBER BROWN: It just gets better, right?

MR. MARUYAMA: Thank you very much.

MEMBER BROWN: Thank you.

MR. MARUYAMA: Okay. There is one other item --

MEMBER BROWN: Oh, one other point, okay? On the diverse actuation for engineered

safeguards features. And this will help me understand this when you all get to this point, those are typically fail as -- ESF systems are typically fail as is, as opposed to fail to actuate. I presume they are the same in your plant as they are in other ones, I'm making that assumption. And John is shaking his head at that, so he obviously knows.

How -- one of the other questions I will be asking is how, if they are fail as-is, what is the backup if in fact you really did need an ESF actuation, but because of a failure in your automatic normal systems as well as a failure in here, don't allow that system to actuate. How do you handle those?

So that's just a heads-up on part of this interface between the two, particularly in the automatic, normally automatic actuation systems, because they are fail as-is. Now all of a sudden you need it, uh-oh, it didn't actuate, now what are the operator response times and other monitoring

things that may allow operator action to take, and is that good enough from your safety -- from your ECCS and other type systems? So, just a question.

CHAIR STETKAR: The way I look at this is, we are obviously going to have discussions about both the normal protection system and the diverse systems.

If those discussions identify any areas of concern, they might feed back into the accident analyses if they are not resolved. I mean, but for the moment, until we have had the opportunity to kind of finish those reviews, it's -- we don't know.

I mean, you have done, for the accident -- right now, you have done for the accident analysis what you need to do.

MR. MARUYAMA: Thank you for the comment. I really appreciate your comments. And so the next slide is the last slide of this presentation, explaining Appendix 15A in Chapter 15, which provides additional details on the models and parameters used for radiological consequences

analysis.

This detail includes description of the RADTRAD code, description of the model for the analysis, and associated input parameters, assumptions for offsite doses, and assumptions and models -- model descriptions for the main contributing dose.

There are no open items for Appendix 15A.

MEMBER POWERS: When you do your analysis on the main control room dose, what's your unfiltered in-leakage?

MR. YOSHIDA: For dose analysis initially, we assume filtered intake and unfiltered in-leakage.

MEMBER POWERS: And how much is the unfiltered in-leakage?

MR. YOSHIDA: Filtered intake, intake flow is 1200 cfm and unfiltered in-leakage flow, it's 120 cfm.

MEMBER POWERS: 120 cfm. Substantial.

Biggest problem is how you ensure that that stays for all time, but that's a licensee problem.

CHAIR STETKAR: That's a -- that's keeping the people from coming in and out of the doors.

MEMBER POWERS: Or keeping the maintenance guy from not putting a hole in the building.

CHAIR STETKAR: Well, that too. There are some ways to be able to do that.

MEMBER POWERS: Well, some plants haven't found it.

CHAIR STETKAR: I know. I know.

MEMBER POWERS: Thank you.

MR. MARUYAMA: Okay, this is the end of this presentation. So thank you for your attention. We will welcome your comments or questions.

CHAIR STETKAR: Any members have anything else for MHI? If not, thank you very much. You covered a lot of material very efficiently, and I really appreciate that. I thought it was a very

good presentation, and we will recess until 3:20.

(Whereupon, the above-entitled matter went off the record at 3:02 p.m. and resumed at 3:20 p.m.)

CHAIR STETKAR: We are back in session. Mike, it's your show again.

MR. TAKACS: Okay. Good afternoon Chairman Stetkar and ACRS members. My name is Mike Takacs. I am the chapter PM for this Chapter 15. Now we are moving into the DCD overview of the safety evaluation. Unlike the topicals, this will have open items and that's where we will focus the presentation on this afternoon.

The first slide here of course mentions the reviewers that are involved. You'll see Jeff Schmidt and Fred Forsaty who will be presenting along here. And of course the same staff members from the Information System Laboratories are here as well in the audience.

And with that we are going to go ahead and move into the technical side of the

presentation. Jeff?

MR. SCHMIDT: Thanks, Mike and I'm Jeff Schmidt from the NRC staff. Just going to go over kind of the major review areas and then the as Mike said, cover the open items that we have.

You know, we reviewed, obviously, the plant characteristics and initial conditions assuring that they are appropriate, conservative, and that the acceptance criteria are met for each one of the Chapter 15 accidents; reviewed the worst single failures to make sure that they are appropriately chosen and analyzed with respect to, again, the acceptance criteria; loss of offsite power was also reviewed as part of each Chapter 15 accident and the review of long-term cooling. Next slide.

Review of sequence of events for each one of the scenarios to evaluate the mitigating systems, instrument responses, and assumed operator actions if there were any; assure that the transients are supported by the design of the plant

systems and future EOPs. Like you said, we haven't seen the EOPs and it sounds like we won't see the EOPs so --

Review of the analyses include transient predictions to assure that the consequences of each analyzed limiting case meet the specific criteria again and all the regulatory requirements.

So we are going to list the open items here.

CHAIR STETKAR: Jeff, I wanted to ask you, this is kind of a procedural issue. I have a few questions that I wanted to ask the staff, but essentially none of them are related to the open items. They are related to other issues.

Is it -- would you prefer to go through the open items first and kind of get through that part of the presentation and then we'll handle the questions? Okay, let's do it that way then.

MR. SCHMIDT: The first open item we have is the general reload methodology topical report, with the staff's interactions with MHI there

were, I'd say, a fair number of changes to the Chapter 15 accident analysis, and the staff is just keeping this one open to do another review of the reload methodology to make sure it's consistent with the Chapter 15 changes, really, and that's why we have it open.

We are going to make another pass through it to make sure it's consistent with the changes we have made in Chapter 15, which like I said were fairly significant.

So the second open item there is the non-LOCA methodology. That we are going to come I guess in October, right Mike, with the non-LOCA topical report?

MR. TAKACS: That's correct, yes.

MR. SCHMIDT: I believe that's with the OGC?

MR. TAKACS: It will be today.

MR. SCHMIDT: It will be today. Okay. So that's with the OGC. It's, right now it's out of the staff's hands with the OGC.

Thermal design is also with management for signatures, which is the VIPRE topical report, there's some thermal analysis, so a lot of these are just keeping track of open items that support Chapter 15, like the non-LOCA topical and the VIPRE topical report.

Next slide. We talked about large break still being open because of the accumulator issues. That's of course is an open item in Chapter 15.6-5. The same with small break LOCA and then on 15.8, which is the ATWS, the staff is still going through the coping analysis for the Chapter 15 events that assume DAS operation and common cause failures. Our input will be provided to the Chapter 7 folks.

Again, 15.4-4, as MHI mentioned, there is still an open item on control rod misalignment or misoperation. I'm still looking to see whether I'm comfortable that all the reasonable misalignments have been covered. So there's still an open item with that.

Startup of an inactive loop, still has

questions about lower mode operation, where in modes 1 and 2, RCPs are all on, you know, there's no additional ones to start.

Lower modes, you can still have one RCP and you can have another one start, staff is still questioning whether there's adequate evaluation for the lower mode operation.

And then boron dilutions in modes 4 and 5, with RCPs running, that's still an open issue. We are still working on some wording in tech specs that the initial wording on the latest review implied that there was -- there could be some dilution when no RCP is running. The staff is a little uncomfortable with that because of mixture issues so we are still reviewing the wording on the tech specs change for RCPs in the lower mode operation for mixing of boron.

Again, this is just a tracker one, also, 15.6-5 is a sump strainer that affects, could potentially affect the LOCA analysis as far as the SI pumps, so we are leaving that as an open item.

Accumulator topical report we talked about a couple of times at this point, we are still waiting to finalize the accumulator topical report. The main issue there is the scaling bias.

Open item 15.6-3, which is a TMI action item. Again, this is kind of a tracking item as far as 15.6-5 is concerned. This has to do with the number 2 RCP seals and how well it would last under station blackout. We are working with DE and the Chapter 9 and Chapter 8 folks on that.

Again, this is not primarily handled here but it is brought up in 15.6-5.

Our first long term cooling issue, which is just another open item, a tracking item for the ANC code, which is the physics code, advanced nodal code. It's Chapter 4.3. We will check if 4.3 is completed and it's going through, I believe, OGC. Again there's no real issues -- technical issues there, it's just going through the internal process.

And I'm going to turn over to Fred now for the long term -- the rest of the long-term

cooling issues.

MR. FORSATY: Thank you. Again, this is Fred Forsaty. I am the engineer responsible for the long-term cooling. Here we have two topics that are open, three RAIs.

The first topic has to do with the GSI-185, small-break LOCA, recriticality mixing, boron dilution. They just completed -- Mitsubishi just completed a test. They have the result of the test back. We are going through that. It's under review.

That's for boron dilution, then we have two RAIs on boron precipitation, 93 and 100. 93 is basically -- the question is simple, explain to us at what time boron would precipitate and what's the switchover time.

Their analysis and their response to the RAI basically are based on the BACCUS test result and the -- we are doing some confirmatory run on that to understand, you know, their four and a half switchover time.

And then RAI-100 is basically RAI-03 with debris. You are asking to let us know what the boron precipitation is in case you have a debris in the lower plenum.

And that again, once we complete our review of 93 and understand the proper -- if you accept the boron precipitation, and its switchover time, then we would look at that and its impact on the debris on the same two issues. So --

MEMBER BANERJEE: Are you doing any analysis yourself or are you just waiting for their response right now?

MR. FORSATY: A combination of both. On boron precipitation, we are doing some confirmatory runs. Mitsubishi is helping us to give us the boundary condition, the information that we need. The confirmatory runs should be probably complete within a month or two at the latest.

And once we finish that, I think we can have a better feel of what we need to do with the presence of debris in RAI-100.

MEMBER BANERJEE: So remind me of what the current switchover time is?

MR. FORSATY: Four and a half hours.

MEMBER BANERJEE: And is that -- are you reviewing that as well to see if it --

MR. FORSATY: No, we are just -- asking questions to ensure that that four and half hours time is accurate.

MR. SPRENGEL: Small correction, the switchover time right now is four hours.

MR. FORSATY: Four hours, I'm sorry, four hours. I think the --

MEMBER BANERJEE: So we just have to wait to see what happens here.

MR. FORSATY: I'm sorry?

MEMBER BANERJEE: We just have to wait to see what happens here.

MR. FORSATY: Yes, it should be we need a month or so, and we should complete our confirmatory run hopefully by then.

MEMBER BANERJEE: Have they made any

assumption about the amount of lower plenum mixing for this already?

MR. FORSATY: Yes, they have some assumptions --

MEMBER BANERJEE: After lower plenum?

MR. FORSATY: Some, and some of the BACCUS test result is also issued as a part of the response. The -- I think some of the information that I have to discuss probably is better to discuss it tomorrow.

MEMBER BANERJEE: For Friday? Okay. Okay. Somebody will be here from the staff tomorrow to --

MR. FORSATY: I'm sorry?

MEMBER BANERJEE: Somebody will be here from the staff to tell us --

MR. FORSATY: I'll be here too.

MEMBER BANERJEE: give us a little bit of --

MR. FORSATY: Yes, I will discuss with you all the information you need. I think I'm done

here.

MR. TAKACS: And that's the end of the open items.

CHAIR STETKAR: That was good.

MEMBER BANERJEE: Everything is proprietary.

CHAIR STETKAR: Now we're getting not so good. I have a few questions. First one, and I'll -- these are not in any particular order, so I'll just walk through them -- as I understand it, MHI revised their pressurizer high-level tech spec so that now the maximum level during plant power operation is 60 percent I think.

And as I understand it -- correct me if I'm wrong -- if they have a feedwater line break with level at that maximum level, the pressurizer will go water solid and they will have water relief through the pressurizer safety valves. Is that correct?

MR. TAKACS: That's correct.

CHAIR STETKAR: Okay, and safety valves

are not qualified for water relief, is that correct?

MR. TAKACS: That's correct.

CHAIR STETKAR: Okay. Now, as I read the SER, you are okay with that. I mean, the analysis that MHI, the analysis of record in the DCD is initiated at normal pressurizer level plus uncertainty.

MR. TAKACS: Plus uncertainty, right.

CHAIR STETKAR: But it's a substantially lower level than the 60 percent, and they show no -- they show small but margin --

MR. TAKACS: Yes, right.

CHAIR STETKAR: to filling the pressurizer at that level.

MR. TAKACS: That's correct.

CHAIR STETKAR: In the SER, section 15.0.0-4, you say that the staff agrees that water relief is allowed for postulated accidents, and you find that the analysis is acceptable.

Now, the question I have, is do you allow water relief of a postulated accident because

LOCAs are postulated accidents so it's okay for a feedwater line break to also transition to a LOCA?

MR. TAKACS: Well, in the case where it starts at nominal plus uncertainty, it doesn't, right?

CHAIR STETKAR: I understand that but everything else, everything else is done --

MR. TAKACS: From the tech spec value --

CHAIR STETKAR: From the tech spec value.

MR. TAKACS: Right. Yes. Effectively, what we were trying to do there, is -- I don't know if you are familiar with the older Westinghouse tech spec, where it was around 92 percent level. What we were trying to do is say that's an unreasonable level to be at, because there is no margin even for AOOs to occur.

CHAIR STETKAR: Sure right.

MR. TAKACS: So we brought it down for AOOs that you have got to survive to your tech spec level.

CHAIR STETKAR: Okay.

MR. TAKACS: For postulated accidents, you are right, we started at the nominal plus uncertainty and said okay, well, if you -- if -- you know the idea is not to pass water. If you do at the tech spec value and you do pass water, then yes, you have a LOCA that's effectively been analyzed.

CHAIR STETKAR: And it's okay to have a feedwater line break transition to a LOCA?

MR. TAKACS: In this case, we said yes. I mean it's not unusual, for instance another example would be Palo Verde has that nominal plus uncertainty for their --

CHAIR STETKAR: Okay that's what I wanted to ask, is, because I'm personally less familiar with design basis accident analysis rules.

MR. TAKACS: Right.

CHAIR STETKAR: If -- there are other currently-licensed plants that could have that same condition.

MR. TAKACS: That's correct.

CHAIR STETKAR: Okay, right. That's --

MR. TAKACS: Well, actually if you want to really look at it, a lot of licensees allow it to be at 92 percent.

CHAIR STETKAR: Which, yes --.

MR. TAKACS: Clearly can't survive anything.

CHAIR STETKAR: Which you clearly can't survive anything. So --

MR. TAKACS: Right. So our, our -- our goal is to --

CHAIR STETKAR: Yes, that -- that's enough. That helps. That helps me. Let me write a note here, only because I -- my memory span is about 15 seconds these days.

All right. One question and I had, and this is a nit but I hate things that are not accurate, and perhaps it is, but I need to be educated if it is. In many places in the DCD, and consequently in the SER, there are statements regarding a single failure of the reactor trip

system, and you know, one of the quotes that I excerpted was, "A single failure was assumed in one train of the reactor trip system, but this has no impact on the safety analysis because any one of the remaining trains is adequate to provide the trip function."

As best as I can understand the reactor trip system, that statement is not logically correct, because the tech specs, for example for instrumentational, require that you have three channels of instrumentation available, three, if one of them is removed from service then you must trip that one, which leaves you two.

MR. TAKACS: Right.

CHAIR STETKAR: If you have a single failure of one of those -- what I'm getting to is you can have one out for maintenance that is not tripped, and a single failure, and one more will not get you a trip function. You need, you need two more trains of censors and if you look at the logic that -- the train logic to the reactor, I'll call

them reactor trip breakers, the same is true.

So it's not clear to me, and I know -- I know this was excerpted directly from the DCDs so it may be more of a question to MHI. It doesn't make any difference in the analyses, because you can still survive a single failure, but characterizing that single failure with the notion that any one of the remaining things is good enough, is, I don't think, quite accurate. So you may want to re-look at --

MR. TAKACS: Right, yes, I think you're right.

CHAIR STETKAR: at that in particular.

MR. TAKACS: We need to revisit that.

CHAIR STETKAR: I don't think it changes, and I tried to think about it every time I bumped into it, and I don't think it changes any of the analyses, I don't think it changes your conclusions about whether reactor trip is the -- you know, whether you assigned the single failure to reactor trip or to a different function. But you

may want to look at that. As I said, that's a -- it's a nit but --

MR. TAKACS: I think you're right.

CHAIR STETKAR: But giving the wrong impression about the available redundancy in things is something that we should be careful about.

MR. TAKACS: Right, I agree.

CHAIR STETKAR: Okay. Oh, this -- I tried to follow the story about when loss of offsite power is a assigned throughout the spectrum of transient events, and -- transients and LOCAs -- and it kind of is triggered at different times for different transients.

MR. TAKACS: Right.

CHAIR STETKAR: For most transients, I think it's three seconds after the reactor trip occurs --

MR. TAKACS: Right.

CHAIR STETKAR: And there might be different rationales for why those -- what the three seconds, and I don't want to get into that. It's

kind if three -- you've got four circulations for like three seconds, after the trip.

You did ask several questions, as I understand it, for the turbine trip transient analysis with specific regard to the timing of the loss of offsite power and I think I followed that whole story. I -- you seemed to probe very well in that area.

The conclusion in the SER is that, it says, "The analysis results show that while turbine trip is the most limiting Chapter 15 event with respect to primary system pressure, no acceptance limits are exceeded."

Now, all of this is kind of derived from RAIs and things that I don't have immediate access to. I always tell people, and I don't want immediate access to it, because I can't read the things that I have immediate access to yet.

So, don't view that as a request. Then when I look at -- I'm trying to read my notes here -- then when I look at the results for the main

feedwater line break, there's a statement in the SER that says this is the most limiting Chapter 15 event with respect to primary pressure.

So I have two, I have a turbine trip with the updated analysis of a consequential loss of offsite power and there was some timing analysis that apparently went through to identify the most conservative time for loss of offsite power. I don't know what that -- when that is, whether it's -- I just don't know what that time is.

But apparently some optimization was done and the conclusion is that that turbine trip with that, if I can call it an optimized loss of offsite power is the most limiting for primary pressure in one part of the SER, and in another part, there is the feedwater line break with some consequential timing of loss of offsite power.

Do you happen to know which -- I mean, are they really lose? I --

MR. SCHMIDT: I don't remember off the top of my head.

CHAIR STETKAR: I kind of lose the sense of how the plant is running for each of these things.

MR. SCHMIDT: Michelle is here to help me.

CHAIR STETKAR: Okay.

MR. SCHMIDT: Because I think we ended up with turbine trip was the answer, right? So Michelle is helping me here, and she is saying it's turbine trip is the limiting case. She was just mentioning that she thought that one was maybe like a PA and one was a --

CHAIR STETKAR: Okay, that may very well be, but it just says most limiting Chapter 15 --

MR. SCHMIDT: Yes, okay.

CHAIR STETKAR: -- which kind of lumps everything together but --

MR. SCHMIDT: Yes, that doesn't really break it out, right.

CHAIR STETKAR: If you do -- okay. If you do separate them --

MR. SCHMIDT: So we've got to make that clear.

CHAIR STETKAR: Yes.

MR. SCHMIDT: Okay.

CHAIR STETKAR: All right. I was just trying to get my -- I do have sort of an intuitive feel, I don't have an intuitive feel for LOCA response on this plant. But transient response I kind of have a feel for, and --

MR. SCHMIDT: Yes, I know we had a lot of questions on turbine trip, and turbine trip became a more limiting event --

CHAIR STETKAR: Okay, and I could -- when I finally worked through all of that, I could kind of understand why.

MR. SCHMIDT: Right.

CHAIR STETKAR: And then got all of that sent and then I went to the feedwater and I said well wait a minute --

MR. SCHMIDT: You went to the feedwater and it said the same.

CHAIR STETKAR: I thought this other -- okay.

MEMBER BANERJEE: How close does it get to the pressure --

MR. SCHMIDT: Michelle, what was the values? They were pretty close. What were the feed values ===

MEMBER BANERJEE: 2200 or something?

MS. HAYES: 2635 and 2616.

CHAIR STETKAR: They're fairly close.

MR. SCHMIDT: Yes, they're fairly close, yes.

CHAIR STETKAR: I mean, you know, percentage, they are fairly close, absolutely.

MEMBER BANERJEE: But turbine trip is limiting, right?

CHAIR STETKAR: Turbine trip, as I understand it, an optimized time for when you lost offsite power. The plant -- and I still don't quite understand why it's so sensitive to keeping forced flow for a short period of time. There seems to be

a lot of stuff -- a lot of attention to that.

MR. SCHMIDT: think about that. There was a lot of attention paid to that. Andy, do you remember why? No, okay.

CHAIR STETKAR: Okay.

MR. SCHMIDT: I'll have to get back to you. I don't remember why. There was a reason.

CHAIR STETKAR: I mean it's kind of an interesting curiosity, but it's turbine trip with loss of the forced flow at just the right point to optimize the pressure transient, and I actually don't know what that time delay is.

MR. SCHMIDT: We'll get back to you on that. I just have to review it again.

CHAIR STETKAR: I mean I haven't read -- it's probably buried in all of those responses. I just haven't read them nor do I want to.

But if you could find it --

MR. SCHMIDT: We'll give you a summary tomorrow.

CHAIR STETKAR: That would be good.

Thanks. Now, and here's another case where I don't know if it makes any difference, so please tell me whether it does. There was quite a bit of discussion about whether or not the uncontrolled withdrawal of a single rod cluster control assembly, RCCA, is an AOO or a PA.

And apparently you asked questions about it, MHI classified it as a postulated accident. They apparently did a failure modes and effects analysis, did some fault tree work, did some quantitative analysis to show that the frequency -- it would require more than a single failure and that the frequency was less than frequencies of other postulated accidents, and you essentially said, well, you have the latitude of throwing it into one box or another box, and you agreed that it could be thrown into the postulated accident box.

Functionally, does it make any difference whether it's a --

MR. SCHMIDT: What do you mean by functionally?

CHAIR STETKAR: In terms of the -- in terms of the consequence analysis for design basis accident analysis.

MR. SCHMIDT: You know, you know, if you make their conservative assumptions for single rod withdrawal, you end up needing to go to a PA because you failed fuel.

CHAIR STETKAR: Okay.

MR. SCHMIDT: If you use, say, a best estimate neutronics code that's, you know, space time dependent, I don't think you get to that failure point. But they -- they took a simplified method and said look, I am going to use a simplified method, I'm going to end up with a PA and I'm going to do a dose analysis on it.

And we accepted that because it took more than one single failure to get there.

CHAIR STETKAR: Okay. Okay. I think I understand that rationale.

MEMBER BANERJEE: So what is the simplified method that they used?

MR. SCHMIDT: They used just a large reactivity withdrawal rate, overly large reactivity withdrawal rate. I guess I don't want to go too much into the details until tomorrow, but --

CHAIR STETKAR: Okay, let's talk about it tomorrow then a little bit.

MR. SCHMIDT: Okay.

CHAIR STETKAR: The question I had, is -- and I've looked at a few plants and just arguing that it takes more than one failure doesn't satisfy me because a lot of plants take more than one failure

and arguing about whether or not a single human error could do it, well, I've done human reliability analyses and you could argue whether or not that's the case for any other plant, as you can argue about, you know, numerical estimates of how frequently this kind of thing occurs.

So I'm more concerned about whether or not there is an actual difference in the design of

this plant and its core compared to other plants that makes it more vulnerable to withdrawal -- uncontrolled withdrawal of a single rod assembly compared to other plants.

MR. SCHMIDT: You know, I think MHI could speak better to that, but you know, in my opinion no, there is probably nothing different in their design versus anybody else's.

You know, it takes an operator action to go into an individual, select mode for an RCCA, and you usually only do that if you have a dropped rod and I don't even know many operators who will even go recover a dropped rod.

So you know, you have to have that single failure to occur, somebody purposely going in there and going in to individual select to control an individual RCCA cluster.

CHAIR STETKAR: Yes, but given that, it is one of the design basis accident --

MR. SCHMIDT: And historically you know, it has been evaluated as a PA. The SRP allows you

to go, either with a PA or an AOO. Again, if you use a more detailed neutronics model and more best estimate type calculations, even if you say best estimate with uncertainties, reasonable uncertainties for your physics model, you know, you are going to end up with an AOO.

CHAIR STETKAR: Okay.

MR. SCHMIDT: It's just that historically, another vendor has also analyzed this as a postulated accident.

CHAIR STETKAR: Okay. That at least, that perspective, as I said, I'm personally kind of less familiar with design basis accident analysis, so that helps.

MR. SCHMIDT: Right, you know, the SRP does give you latitude to go both ways on this.

CHAIR STETKAR: Yes. Okay. Boron dilution events, and I know you have an open item regarding dilution with the reactor coolant pumps running, I'm going to drop back to the events that were analyzed.

In the events that were analyzed in modes 3, 4 and 5, the analyses show time windows for the operator to terminate those events in modes 3 and 5 of 16 minutes and mode 4 at shutdown of 19 minutes.

And apparently the conclusion was made by MHI, and that's the reason I didn't ask them about it was, the conclusion was made that because 16 minutes is longer than the 15 minute criterion in SRP section 15.4.6, and obviously 19 minutes is even much longer than that, therefore everything was acceptable.

What's the technical basis for a universal 15 minute acceptance criterion for operator response that applies to every single plant in the world with every single scenario and the equipment configuration.

MR. SCHMIDT: I have to answer that it's just the SRP criteria that it's probably more of a human factors response time. They really have, you know, significantly more time. I mean, the 16

minutes is effectively back-calculated to give really effectively the core designers the maximum margin they can for boron concentrations.

So in essence, they really have a lot more time than the 16 minutes would imply. There are other alarms that are -- that could be credited earlier that would give them an indication of a boron dilution event occurring.

So, in essence it says 16 minutes, but it's really a back calculation of 16 minutes.

CHAIR STETKAR: Yes, I got that sense. I mean I didn't look at the --

MR. SCHMIDT: Okay. It wasn't clear initially to us.

CHAIR STETKAR: No, it -- well I mean reading the DCD and then reading the SER, I kind of got the sense that it was calculated such that if these conditions were available, you know, did apply, then indeed we could show that we miraculously had 15 minutes and 17 seconds or something like that.

MR. SCHMIDT: Right. Right.

CHAIR STETKAR: But where -- I guess we just then rely on the human factors engineering to determine whether or not it's, for this particular design, feasible to accomplish the actions within the time window that the -- now this is the formal accident analysis, regardless of what you say about how much additional time might be available in a best estimate analysis.

This is indeed the basis for certifying this design, so that in principle, I guess the COL applicant would be required to demonstrate that indeed they can meet that. Is that correct?

MR. SCHMIDT: Right. That -- right.

CHAIR STETKAR: Are there ITAAC, I mean other than the generic -- other than the generic you shall perform verification validation studies on all of your -- all is not the correct word -- on a sample of your human factors engineering tasks, which is kind of a generic requirement, are there specific ITAAC that are developed for showing those

-- the feasibility, demonstrating the feasibility of the operator actions, with a margin for any operator actions that are explicitly included in the design basis accident analyses? You follow my line of questioning?

MR. SCHMIDT: Right, no, there's no specific, for example boron dilution ITAAC.

MR. FORSATY: But normally, in a situation like this, I think what we -- the action we take is to create a COLA item, and then that would be kind of tracked.

CHAIR STETKAR: Well, that's what I was going to, I mean essentially it's -- I understand it's not done during the design certification. It's done -- essentially the COL applicant, when they perform their human factors engineering task analyses, you know, the selection of the scenarios that they use, essentially for their V&V of the human factors engineering, in one sense, there could be a COLA item that says, you know, this has to be one of the ones that you select because it has

implications on the accident analysis in the certified design.

MR. FORSATY: Yes, but that would be creating a lot of these, like, for the containment cleanliness programs, and you know that the applicant or the licensee agrees to keep the debris to such a level, and as a result of that, they have got to conform to some of the --

CHAIR STETKAR: But that one's a little bit different. That's sort of a status of the plant. This in particular would affect the population of scenarios that the COL applicant, or at that time you'd be the COL holder because they always do this stuff after the COL is issued.

MR. DONOGHUE: Mr. Stetkar, this is Joe Donoghue. I think there's a general ITAAC in part of the review, for Chapter 18 I think it is, on human factors, but I want to get back to an answer on anything specific to this event or other assumptions for different events, all right?

CHAIR STETKAR: That's -- okay -- okay.

Because we -- okay thanks Joe.

MR. DONOGHUE: I mean I guess unless we could do it today.

CHAIR STETKAR: Right, it isn't -- well, it's just a matter of whether or not the staff in their review, I you identify, for example, in your Chapter 15 analyses, critical operator actions.

For example, this boron dilution is one, steam generator tube rupture is another, whether or not there is a vehicle for the staff to essentially put a hook into the COLA, into the COL applicant that says, well, because in Chapter 18 it's just sort of a generic you'll go out and select a bunch of scenarios and try to make sure that they are important scenarios but there is no specification about which ones you select.

MR. DONOGHUE: Yes, I don't recall us specifically making that connection here.

CHAIR STETKAR: Good.

MS. HAYES: This is Michelle Hayes and 15.6.3, with those operator actions we tied it back

to the ITAAC.

CHAIR STETKAR: I saw that. That's why I asked about this one.

MS. HAYES: that applies to all operator actions for Chapter 15.

CHAIR STETKAR: Oh, all?

MS. HAYES: Yes.

CHAIR STETKAR: Oh okay. I must have misinterpreted -- I thought it was --

MS. HAYES: assumed in the safety analysis.

CHAIR STETKAR: I thought that only applied to the tube rupture.

MS. HAYES: No.

CHAIR STETKAR: Since it was in -- okay. I must have misinterpreted that then. Thanks. That helps. That -- and that's in 15.6.3, you said?

MR. BIELEN: Yes.

CHAIR STETKAR: Okay. With that, that is all that I had. Any other members have anything for the staff? If not --

MEMBER BROWN: Just one question, relative to your comment on the operator action of 15 minutes, how does that relate to the other reg guides that talk about 30 minutes for operator reaction to accident response and have some justification for that, whereas there's none for -- I mean, we had this discussion on time available on --

CHAIR STETKAR: You're looking at me as if I would have an answer.

(Laughter.)

CHAIR STETKAR: It is a valid question. There are many -- that's why I tend to fall back on if an applicant takes credit for 15 minutes or 16 minutes or 19 minutes or 27 minutes or whatever minutes in their accident analysis, there should be justification somewhere that indeed the actions that they are taking credit for can be accomplished -- can be accomplished, that they are feasible within that time window, with some margin or some confidence of reliability, because I've given up on

whether or not it should be 30 minutes or 15 minutes or some arbitrarily assigned number that differ depending on where you might find those numbers, Charlie.

That's kind of my take and that's why I was asking about whether, since the boron dilution, you know, analysis of record basically takes credit for operator response within 16 minutes' or 19 minutes' work, you know, in principle for some of the other scenarios that they looked at was 61 minutes I think and 73 minutes.

I mean, the same questions in principle would apply for those, that there be some, at least, requirement for the people who developed the training and the procedures and do the task analyses for the human factors engineering, to demonstrate that indeed people can do that, and if they can't demonstrate that, that then says well, there's an issue, because the certified design says that they will do that.

MR. FORSATY: I think that as a part of

licensing the operation, some of these items are built into the training, and I don't know with this specific one, but I have seen some of these --

CHAIR STETKAR: But in -- the nice thing about the process that we have now is there is a -- as opposed to the previous licensing process there is a greater focus on human factors engineering, you know, and before you kind of relied on training and generic procedures and things like that.

Here, here it is an element of getting the plant licensed and a lot of the demonstration is left to post-COL issuance once the procedures are in place and the training programs are developed.

But without kind of reminders to close those loops, to put the hooks in there to say that there are indeed some scenarios that do have some implications on the licensing basis for the plant. They can easily fall in the cracks, especially things like boron dilution.

Tube ruptures are things -- people tend to look at tube ruptures but boron dilutions, not so

much, and especially boron dilutions from low power, you know, shutdown conditions that people tend not to put a lot of care into, or at least traditionally have not.

MR. SHUKLA: Yes, I think there is an explanation for 30 minutes in the digital I&C ISG 33.

CHAIR STETKAR: We've run into that in the past and the digital I&C world has been evolving more toward the kind of, what I'd call a performance-based assessment.

MEMBER BROWN: Well, yes I have no -- even there, there was a 30 minutes. If you can't make a decision, you default to a certain thing, but there's a time available, time required, analysis required and that comes under human factors. But I just polled these numbers in the analysis basis, they are all -- 15 minutes is thrown around all over the place, because I just took the 15 minutes and just ploughed through the entire now and there's lots of 15 minutes in there.

CHAIR STETKAR: There are, aren't there?

MEMBER BROWN: With no basis at all for any of them, other than we make this assumption.

CHAIR STETKAR: Well, they're in the Standard Review Plan.

MEMBER BROWN: I didn't say I agreed with that. I just said they're all over the place, okay and they're in direct contradiction to standard other assumptions that us I&C guys make.

CHAIR STETKAR: Thanks.

MEMBER BROWN: You don't want me to talk anymore, is that what you are trying to say?

CHAIR STETKAR: Pretty much.

MEMBER BROWN: Okay.

(Laughter.)

CHAIR STETKAR: I think we got it. I'll go back and look at that 15.6.3, because as I said, I saw that and I probably misinterpreted it as only related to the tube rupture.

MR. DONOGHUE: Michelle just pointed out that in 15.0.0-4 there is a discussion about

operator actions and it links to a COL and then the ITAAC that she just -- 15.6.3, so it is --

CHAIR STETKAR: It does -- okay, I, I apparently missed that link. So thanks. Anything else for the staff? If not, thank you very, very much. Now, we are suddenly -- suddenly have a wealth of time on our hands and because so many things were postponed to the ubiquitous tomorrow, I've been told that the staff is prepared to discuss the Comanche Peak information regarding Chapter 15. Is that correct, Steven?

MR. MONARQUE: Yes, this is Steven Monarque with New Reactors. Yes, we are prepared to go ahead with that and discuss our safety evaluation. And I would ask Luminant if they are ready to discuss their slides.

CHAIR STETKAR: Is Luminant -- I hate to sandbag you, but you have been sitting there so quietly, and --

MR. WOODLAN: Yes Luminant is ready.

CHAIR STETKAR: Okay great. Let's see

if we can get that out of the way. I am assuming it will be fairly painless to everyone concerned and it will give us a little bit more time tomorrow.

There was some discussion about whether even to have the poor folks from Comanche Peak come all the way up here, but I just love seeing people fly on airplanes and -- yes, and in some sense actually to check off our boxes it's important for you to at least present on the record what you have done, what you do have, and the same is for the staff. So Don, I guess this is finally your show.

MR. MONARQUE: If I could go ahead?

CHAIR STETKAR: Steven, I'm sorry. I didn't want to --

MR. MONARQUE: Good afternoon. My name is Steven Monarque. I am the lead PM for the Comanche Peak COL application. I want to thank the committee members for giving us this opportunity to present the safety evaluation as well as giving Luminant an opportunity to present their analysis to the subcommittee today.

As you all know, today we present five chapters to the subcommittee, and with that I will go ahead and introduce Luminant.

MR. WOODLAN: My name is Don Woodlan. I am the licensing manager with the new built project for Luminant Power and I am here to talk, as Steve said, about Chapter 15 as it relates to the Comanche Peak FSAR and our SER.

Here's the -- the administrative slides are longer than the meeting slides. The agenda, we will do an introduction which I have kind of already done, a subsection discussion and then a summary.

Chapter 15, I think all of you are aware, is pretty much incorporated by reference. This is standard plan information. We took no departures from the work that was done by MHI for Chapter 15.

The primary information that is in Chapter 15 related to the Comanche Peak design is really Chapter 2 information, which is the chi over q values, and that is discussed in Chapter 2. I'm

sure it will be addressed in detail in the Chapter 2 SER. It is in our FSAR.

And the key is that the standard plant values bound the Comanche Peak values so that all the analyses that are done, especially with the doses, remain applicable for Comanche Peak.

And we have no contentions pending with respect to Chapter 15. In fact, our ASLB has been closed.

CHAIR STETKAR: Don, where is your emergency operations facility going to be for Comanche Peak units?

MR. WOODLAN: It's on site. It's maybe a quarter of a mile from the actual plant. It'll be the same facility we use for units 1 and 2.

CHAIR STETKAR: Do you evaluate in your COLA dose assessments for the EOF or is it just limited to the main control room and the TSC?

MR. WOODLAN: No, I'm pretty sure -- let me look real quick -- but I'm pretty sure it includes the -- I don't have it here. It's

primarily the TSC and the control room, the EAD and then the distant food generation areas.

CHAIR STETKAR: But since your EOF is onsite, it's somewhere between the TSC and the distant food generation?

MR. WOODLAN: Yes, it's outside the EAD, so EAD values would bound it.

CHAIR STETKAR: Okay. Okay. Okay. And it is outside the EAD?

MR. WOODLAN: Yes.

CHAIR STETKAR: Okay. Okay. Thanks.

MR. WOODLAN: Next slide. Don't go into any details on the sections, because as I said up front, without departures we are incorporating by reference all the Chapter 15 sections. Last slide.

As I said, the key values that are in the FSAR are the chi over q values. We are bounded by the standard plant and we have no outstanding issues in the SER.

And that concludes my presentation.

CHAIR STETKAR: Well, you did get a

question, so --

(Laughter.)

CHAIR STETKAR: Any of the members have any questions for MHI?

MEMBER POWERS: For Luminant.

CHAIR STETKAR: I'm sorry, Luminant.

MEMBER POWERS: He wanted the questions for MHI. It was okay with him.

CHAIR STETKAR: Yes, because they incorporated everything -- I'm sorry. My mind is turning to oatmeal. If not, thank you very much. It was a brief and succinct --

MEMBER BROWN: And illuminating.

CHAIR STETKAR: I didn't want to say that. I was, I was --

MEMBER POWERS: You have no self-control whatsoever, do you?

MEMBER BROWN: No.

CHAIR STETKAR: Mike? It's yours.

MR. TAKACS: It's my turn? Okay, good afternoon again, it's Mike Takacs, the chapter PM

for the reference COLA, Chapter 15, the safety evaluation with open items.

As you heard earlier, all the sections with the exception of one, all the sections are IBR section -- before I bounce ahead let me point out Michelle Hart is the main reviewer of this one particular section that is not IBR. It is section 15.03 of the DBA, radiological consequences section and you have probably seen that information that's about two pages in the safety evaluation of the total of seven pages.

So if there are any questions, this is the end of my presentation.

(Laughter.)

MR. SHUKLA: And you were saving this for tomorrow?

MR. TAKACS: I was going to offer to do it today but I -- it's done.

CHAIR STETKAR: Anything from any of the members? Thank you, also for keeping us on schedule. It's, as I said, it's -- at some level it

sounds silly, but as part of our function to make sure that indeed there is the opportunity for people to ask questions if they have them, and make sure that we have the presentation on the record so that there is some record that we have at least discussed the matter, and we have done that.

With that, I have -- Ryan is looking at me as if there might be something that MHI wants to contribute or --

MR. SPRENGEL: We can try to fill the time.

CHAIR STETKAR: Let me, let me first do something that I need to do. Well, were you going to try to answer some of the questions from this morning, or what did you want, what were you proposing?

MR. SPRENGEL: Or consider -- we would consider starting the presentations today, because I think actually some of the presentations --

CHAIR STETKAR: Closing the meeting and start --

MR. SPRENGEL: Yes.

CHAIR STETKAR: Okay. If -- if we are going to consider that, let me do something first, while the meeting is still open, and that is, are there any members of the public here, Girija, can you open up the bridge line if there is anybody out there? Anybody from the public here who has any comments?

Seeing none, I am going to wait a couple of seconds and open up the bridge line to see if there is anyone out there who wants to make any comments or ask any questions. And then we can consider closing the meeting so that I don't have to go back and reopen it.

MR. SPRENGEL: I don't know if any of the public are on the phone. So --

CHAIR STETKAR: I know we had a bridge line open and there is always the opportunity for someone to be out there, so I'll just wait to see if there is.

MR. SHUKLA: The line is open.

CHAIR STETKAR: Okay. I've been told the bridge line is open. If there is someone out there listening in, would you do me a favor and just make some sound so that I know that the bridge line is open and that there is someone out there?

PARTICIPANT: Hello

CHAIR STETKAR: Thank you very much. That at least confirms that we are open. Now, is there -- are there any members of the public out there who wish to make any comments or add anything to the record?

Hearing none I will assume that we don't have any other comments and we will close the bridge line again, please. And now, members and MHI, it's 4:20. We do have a little time left on our agenda for this afternoon. MHI has offered to start presenting some of the proprietary information.

I'm certainly willing to entertain that. I have absolutely no personal life whatsoever.

What are your constraints? And I know you are not supposed to have any constraints, but I

try to be humane about these things.

COURT REPORTER:. I have a personal life, but I'll give it up for the ACRS.

(Laughter.)

MEMBER BROWN: She's much more accommodating than some of us are.

CHAIR STETKAR: So let's assume that we can run for about an hour.

MEMBER REMPE: Is Sanjoy still here?

CHAIR STETKAR: And Sanjoy isn't here. I'd like to find --

PARTICIPANT: Where is he?

CHAIR STETKAR: I don't know. It's not my day to watch him. We should try to find Sanjoy because he obviously would have an interest in this. I would certainly, I think, I think any amount of material that we can get in, we should probably try to do that. I'd like to find Sanjoy and then we will need to close the meeting.

Presuming that we can find Sanjoy, we need to make sure that there is no one in the room

from MHI or the staff who are not privy to the information. I have a little delay here while we try to find Dr. Banerjee because without him I fear that this exercise might not be worthwhile

MR. SPRENGEL: And would any portion of the meeting be open tomorrow?

CHAIR STETKAR: At the -- I mean, we can open and close meetings at will. At the moment, we were not -- our agenda for tomorrow showed closed presentations by MHI, and then opening the meeting for the Comanche Peak discussion, but since we have achieved that, right now I don't see any reason for the meeting --

MR. SPRENGEL: My only concern is discussion on planning, specifically the full Committee.

CHAIR STETKAR: We can open it for that.

MR. SPRENGEL: Okay.

CHAIR STETKAR: If need be.

MR. SHUKLA: So, small-break first?

CHAIR STETKAR: Yes, I'd really like to

find Dr. Banerjee.

(Whereupon, the above-entitled matter went off the record at 4:21 p.m. and resumed at 4:26 p.m.) CHAIR STETKAR: We're back in session. This is an open session, but we are now going to close the meeting to discuss MHI proprietary information, and with that we will also close the bridge line so that we do not share any of this information publicly.

So I'll wait back to hear to make sure the bridge line is closed. Again I'll ask the staff and MHI to confirm that everyone who is in the room is okay to be here, and I just want to make sure that we hear back that the bridge line is closed.

(Whereupon, the open session was concluded at 4:28 p.m.)

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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

U.S. APWR SUBCOMMITTEE

OPEN SESSION

+ + + + +

TUESDAY

JULY 10, 2012

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ROCKVILLE, MARYLAND

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

SUBCOMMITTEE MEMBERS:

JOHN W. STETKAR, Chairman

SANJOY BANERJEE, Member

CHARLES H. BROWN, JR. Member

JOY REMPE, Member

WILLIAM J. SHACK, Member

NRC STAFF PRESENT:

GIRIJA SHUKLA, Designated Federal Official

ANDREW BIELEN, RES

JOHN BUDZYNSKI, NRO

JEFF CIOCCO, NRO

JOSEPH DONOGHUE, NRO

FRED FORSATY, NRO

MICHELLE HART, NRO

MICHELLE HAYES, NRO

RALPH LANDRY, NRO

STEPHEN MONARQUE, NRR

NGOLA OTTO, NRO

EDUARDO SASTRE, NRO

JEFF SCHMIDT, NRO

AMY SNYDER, NRO

ALSO PRESENT:

ROBERT BEATON, ISL

DAVID CARAHER, ISL

JOHN CONLY, Luminant Power

CLIFF DAVIS, INL

HIROSHI FUJISHIRO, MHI

YUKO FUJITA, MNES

ED HAOREGO, INL
HIDEAKI IKEDA, MHI
MASAAKI KATAYAMA, MHI
SCOTT KIFFER, MNES
ATSUSHI KUMAKI, MNES
HIROSHI HAMAMOTO, MHI
TERRY HEAMES, Alion
KEVIN LYNN, MNES
YUTA MARUYAMA, MNES
REIKO NARUTOMI, MHI
JUNTO OGAWA, MHI
MARCOS ORTIZ, ISL
KEITH PAULSON, MNES
DAN PRELEWICZ, ISL
RYAN SPRENGEL, ISL
REBECCA L. STEINMAN, MNES
RURIKO TAKAMASHI, MNES
TETSUYA TERAMAE, MHI
EDWARD D. THROM, ISL
DOUG WOOD, MHI
DON WOODLAN, Luminant Power
TSUYOSHI YOSHIDA, MHI

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P R O C E E D I N G S

8:35 a.m.

CHAIR STETKAR: Okay. The meeting will come to order. This is the second day of the subcommittee meeting for MHI Chapter 15 U.S. APWR. For the record, members in attendance are Sanjoy Banerjee, Joy Rempe, Charlie Brown, and Bill Shack. And the Designated Federal Official is Girija Shukla.

As I mentioned yesterday, what we'll do this morning is, we're in open session right now, but I'm going to close this session, unless, MHI, do you have any responses to questions to put on the open record? What I was planning to do was to close the session for the proprietary information and then reopen it at the end so we can discuss upcoming meeting issues and things like that. If you do have anything, answers to questions from yesterday that are part of the public record, we can do that in the open session. Otherwise, we'll discuss them in the closed session.

Staff have anything to say? Nothing? Okay. With that, I would like to close the meeting so we can discuss, continue the discussion of the

proprietary information. I'll ask the staff and MHI to confirm that everybody in here is appropriate for the information. And with that, we will close the record.

(Whereupon, the foregoing matter went off the record at 8:37 a.m. and went back on the record at 11:43 a.m.)

CHAIR STETKAR: We're now in open session, and, as we normally do, I'd like to go around the table now that we're kind of at the end of our day and a half session here to see if any of the members have any additional comments or questions, and I'll start with you, Joy.

MEMBER REMPE: Oh, I guess I'd like to reiterate something we brought up yesterday that, again, there's a lot of outstanding issues. And as you are well aware, the fact that, in addition to the outstanding issues with the steam generators, thermal conductivity degradation, the accumulator model, and some of those things will impact the thermohydraulic analyses, and so we'll have to see it repeated. In addition, there were issues brought up by Dr. Banerjee, who's not here today or at this point in the closed session, and so I would like to

see those issues addressed and staff comparisons of some of the parameters besides peak cladding temperature to see how well they've been addressed. So are you going to have a meeting later to re-look at the thermohydraulic issues?

CHAIR STETKAR: It's clear that we're not done with this topic. How we bring it to closure is something we're going to have to discuss, you know, offline. We can't decide that today. Sanjoy had a couple of ideas, and I'm hoping he'll walk back into the room here in the next five minutes or so. But we need to get to closure on some of these, at least to a better understanding of, as Sanjoy put it, the physics. We don't need precision in the numbers as long as we understand what's actually going on in the models.

MEMBER REMPE: It's reasonable and predicted consistently.

CHAIR STETKAR: Yes, that's right. Charlie?

MEMBER BROWN: Only to reiterate my past stuff from the reactor safety part of it, not this particular part --

CHAIR STETKAR: Sure.

MEMBER BROWN: -- that's dealt with here relative to what they have to do, but we've still got its basis and how the narrowing down of what's automatic and what's manual and the basis for that and all that kind of stuff. So that deals with the coping --

CHAIR STETKAR: We'll beat that to death in Chapter 7, I'm sure.

MEMBER BROWN: Yes.

CHAIR STETKAR: Dr. Shack?

MEMBER SHACK: Well, my biggest confusion relates to this break flow modeling. I'd like to know when it's conservative enough, and I didn't see any criteria for determining that and it sort of left me, you know, it sort of looked like I could get a lot of answers in various ways. That just sort of needs to be resolved.

CHAIR STETKAR: Again, I look at that -- yes, and I agree. I look at that as -- I don't like the word conservative. Understand where we are on the spectrum of reality would be a decent goal.

MEMBER SHACK: What constitutes an acceptable break flow model --

CHAIR STETKAR: Yes.

MEMBER SHACK: -- and why.

CHAIR STETKAR: And why. Dr. Banerjee is not here. He's obviously had a lot of questions and comments, so I think they'll stand by themselves. And I would just echo from what you've heard here. I think that we've learned a lot in the last day and a half in terms of, especially for the LOCA responses and how they're modeled. We haven't talked too much about the transients. I feel reasonably comfortable about the transients. We had a few questions yesterday but not an awful lot.

I think that we will need to have additional discussions about, in particular, the LOCA modeling so that we do understand a little bit better the physics, how your models are behaving, what particular assumptions are most sensitive in your modeling of the performance, whether or not your models can be driven, in a sense, to give very different results so that we better understand kind of the sensitivity of those models. And from the staff's perspective, if you can shed any light, you know, on your confirmatory analyses so that we can better understand where the differences are, if there are differences, and the reasons for those

differences, I think that would also help to kind of give us confidence about the overall conclusion.

And with that, I would like to thank you very, very much. I think we've had a very, very good discussion. I always appreciate these meetings because you do come well prepared and you have a lot of back-up material, and I really appreciate all of the hours that were probably put in last night looking at information to answer a few of our questions. So thank you very much for the presentations.

Now, with that, and I have not forgotten, Ryan, I think we need to discuss offline among MHI, the staff, and our staff how we come to closure on the particular issues of the Chapter 15 analyses. And I don't think it's worth trying to make any suppositions about how we're going to do that right now. We do have another two-day meeting scheduled in the middle of October. I think it's the 18th and 19th; don't hold me to the dates. In particular, to look at the non-LOCA methodology topical report and the advanced accumulator CFD modeling. That two-day meeting will give us another opportunity to at least revisit some of these

issues. We can't afford more than two days, unfortunately, in our schedule, but we may want to think about how best to use that meeting time. And if there's anything else that we can do, perhaps, in the interim, we'll talk about that more offline.

Now, we do, I believe, right at the moment have a full committee presentation. I think it's for October's schedule. Is that right, Ryan? You tend to know these things better than I do. And we wanted to discuss a little bit about what the topics would be for that full committee meeting, so that everyone has a chance to prepare.

And as you're aware, what we typically do as a full committee is write an interim letter just to inform the staff and the Commission of where we are in our review of the entire issue, both the design certification and the COLA process and, if necessary during those interim letters, to identify any particular issues that we feel merit special attention so that they can be addressed before we have the final review.

And since our last interim letter, which was issued in September, a year ago, September 2011, we've had subcommittee meetings that have covered

design certification Chapter 9 on auxiliary systems. We've had this meeting now today, and I'll talk about that in a second. And for, in particular, the COL process, we've now completed Chapters 5, 8, 10, 11, 12, and, again, this meeting on Chapter 15. We have a meeting scheduled in September on Chapter 4 for both the DCD and the COL. That's nominally before our October meeting. It's probably not a good topic for the full committee to address, you know, two weeks later, so I'm not going to include that.

My suggestion is that, in the full committee meeting, we cover primarily the COL. The previous letter was entirely design certification. What I'd like to do is cover kind of a status update on the COL, fairly simple chapters but it's important to let people know if we have any concerns. So it would be 5, 8, 10, 11, and 12 on the COL. And only because it's something that I don't want to dangle, we'll include also Chapter 9 on the design certification. It's going to be a bit of a split because we won't get to Chapter 9 on the COL for some time yet, but it's a big chapter and there's a lot of systems in there and it's

worthwhile. So if that seems reasonable to both the staff --

MR. MONARQUE: Chairman, what about Chapter 15?

CHAIR STETKAR: Yes. Come to the mike because we're still on the record here.

MR. MONARQUE: Sorry, Chairman. What about Chapter 15? Are you going to defer that for another --

CHAIR STETKAR: I think, I think it's prudent to defer it. I think that there are enough, that the problem is that we'll have the next subcommittee meeting where we have a chance to address Chapter 15 type issues would be after that full committee meeting. And my sense is that there are still enough questions in the minds of the subcommittee that it might not be fair to bring Chapter 15 to the full committee in October.

If we had a time slot before October to have another subcommittee meeting, I'd feel differently. But I just, I don't think it would be fair to either MHI or the staff to bring Chapter 15 to the full committee in October. I just don't. So we'll have to push it -- we can always schedule

another interim letter closer to the end of the year, December time frame for example. If we want to pick up, if it's possible, Chapter 15 and Chapter 4 and issues that might come up on Chapter 7 in November for example, we have that opportunity in December for another interim letter. We don't necessarily have to schedule them, you know, six months to a year apart.

MR. MONARQUE: I think, wasn't the general rule, I think, you wanted to include at least six chapters?

CHAIR STETKAR: Well, the general rules, I was trying to bundle chapters together. We don't want to issue a letter, you know, with only two or three chapters, or if there was some compelling reason, if we identified some major concern, to have a full committee discussion of that concern and issue an interim letter, you know, kind of as a focus topic. In my opinion, we haven't identified anything like that yet to date, so that's why I've just been kind of bundling things together in larger groups just so we can manage our time and not bother everybody with, you know, several letters.

So let's try to do that, if it sounds

okay to you and MHI, and we'll plan for that. And I think probably, we can discuss it again offline, probably tentatively plan for another interim letter. The problem is we don't have a meeting in January, a full committee meeting in January. So it would either be December or February. We don't normally have a -- thank you, Dr. Shack -- normally a full committee meeting in January, but kind of December or February to pick up probably Chapter 15 and most likely Chapter 4 and perhaps 7. Those are three heavy-hitter chapters, and I know you don't want to delay too long on getting feedback.

MR. MONARQUE: Do you want to talk offline on how we're going to move forward with 15 --

CHAIR STETKAR: Yes, absolutely offline.

MR. MONARQUE: Okay.

CHAIR STETKAR: Yes, yes, yes. Any other questions, MHI, staff, subcommittee members?

(No response.)

CHAIR STETKAR: And, again, I thank you all, MHI and the staff. I think it's been a really productive day and a half. I'm really sorry that we had to cut it short from our originally-scheduled

two days, but we have four subcommittee meetings going on today and the time pressures are just horrendous. So thank you all again. And with 15 seconds to spare, we are adjourned.

(Whereupon, the foregoing matter was concluded at 11:58 a.m.)

Presentation to ACRS Subcommittee

Overview of

MUAP-07013 R2

"Small Break LOCA Methodology for US-APWR"

July 9, 2012

Mitsubishi Heavy Industries, Ltd.

Meeting Attendees

- Hideaki Ikeda (Presenter)

  Engineering Manager

  Safeguard System Engineering Section

  Reactor Safety Engineering Department

  Mitsubishi Heavy Industries, Ltd.

- Masaaki Katayama (Co-presenter)

  Deputy Manager

  Safeguard System Engineering Section

  Reactor Safety Engineering Department

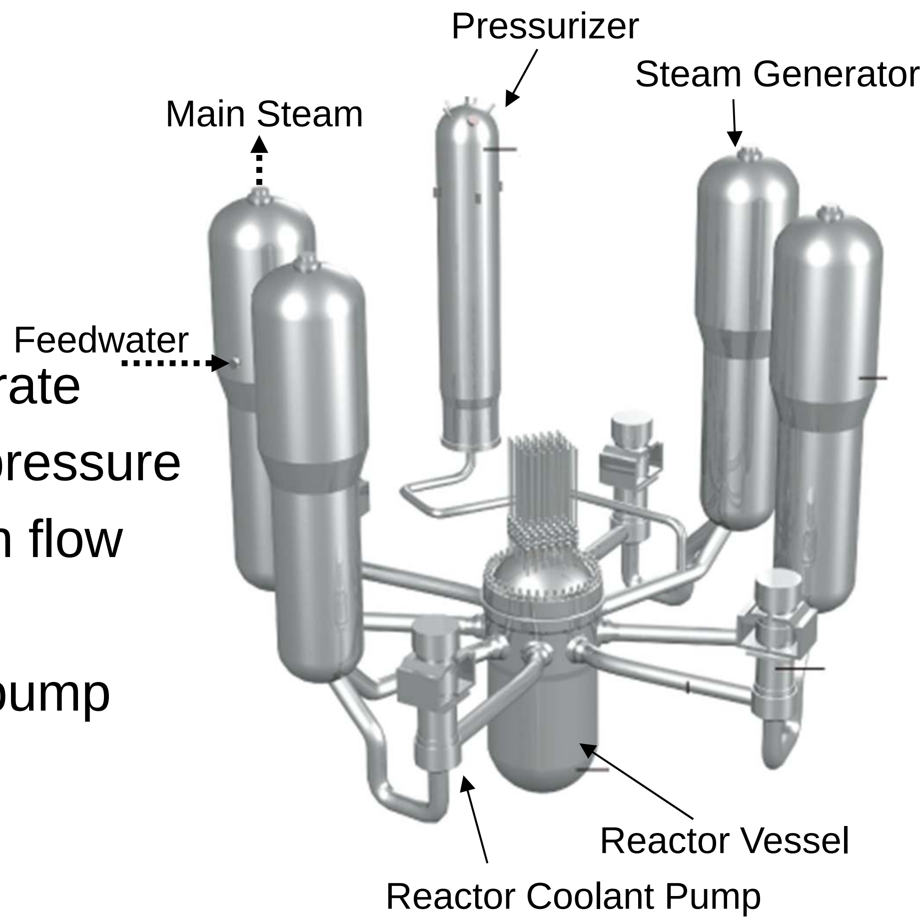
  Mitsubishi Heavy Industries, Ltd.

Highlights of the Topical Report

- Objective
  - ✓ To present MHI's comprehensive methodology for US-APWR Small Break LOCA (SBLOCA) analyses
- Key Topics
  - ✓ Identification of US-APWR design features relevant to SBLOCA
  - ✓ Development of Phenomena Identification Ranking Table (PIRT) and assessment matrix for SBLOCA analysis code
  - ✓ Development of M-RELAP5, a modified version of RELAP5-3D to incorporate 10CFR50 'Appendix K' Evaluation Model (EM)
  - ✓ Assessment for EM adequacy of M-RELAP5
- Relationship with DCD
  - ✓ M-RELAP5 is being applied to the SBLOCA analyses for the US-APWR design certification
  - ✓ The present topical report is referenced in Chapter 15.6.5 of the US-APWR DCD

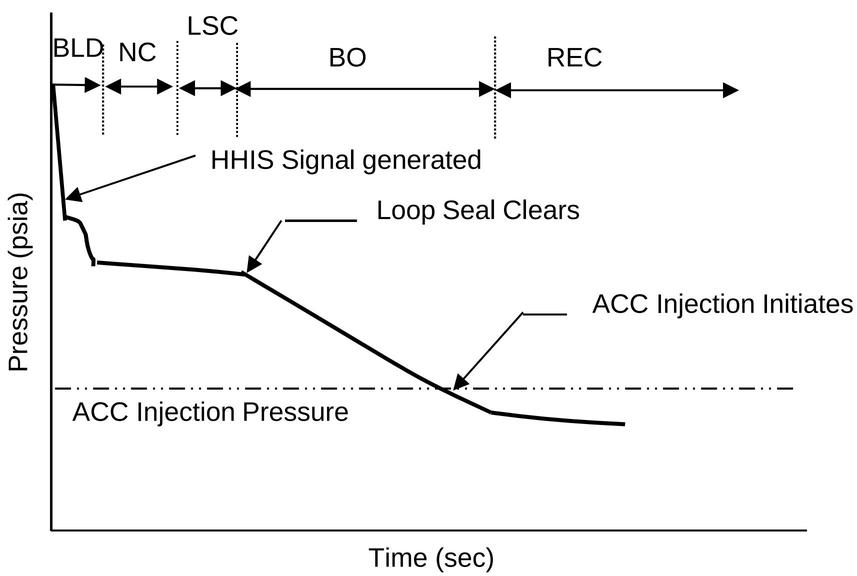
US-APWR Summary

- Plant Specifications
  - ✓ 4451 MW thermal output
  - ✓ 257 fuel assemblies
  - ✓ 17X17 fuel rod lattice
  - ✓ 14-ft active fuel length
  - ✓ 4.6 kW average linear heat rate
  - ✓ 2,250 psia primary system pressure
  - ✓ 448,000 gpm thermal design flow
  - ✓ U-tube steam generator
  - ✓ Centrifugal reactor coolant pump
  - ✓ Neutron reflector
  - ✓ Enhanced capacity of HHIS
  - ✓ Advanced accumulator

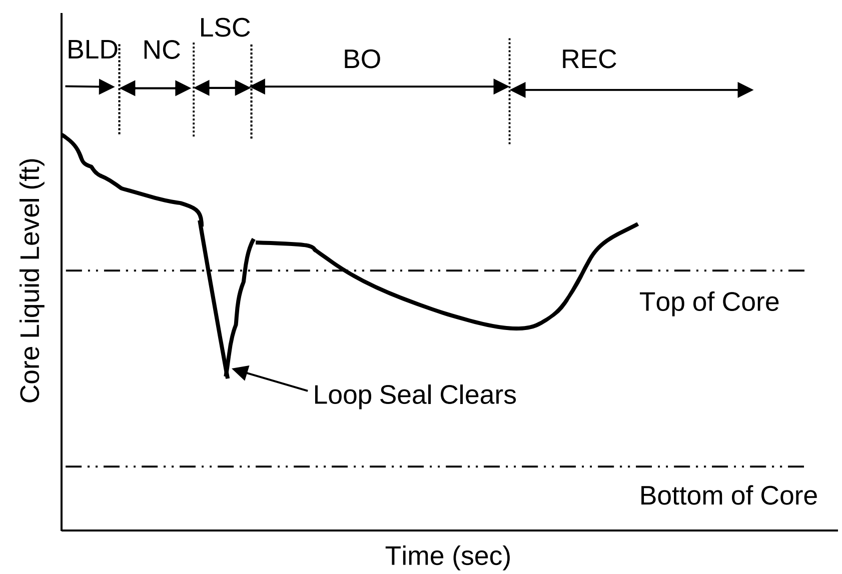


US-APWR SBLOCA

- US-APWR SBLOCA Scenario
  - ✓ Transient evolution under US-APWR SBLOCAs is similar to that for Westinghouse 4-loop PWRs, and is divided into five phases:
    - Blowdown period (BLD)
    - Natural circulation period (NC)
    - Loop seal clearance period (LSC)
    - Boil-off period (BO)
    - Core recovery period (REC)



Primary Pressure Response



Core Collapsed Liquid Level Response

Compliance with Regulatory Requirements

- SBLOCA Methodology
  - ✓ MHI SBLOCA methodology using the **M-RELAP5** code conforms to regulatory requirements specified in Appendix K to 10CFR50

- Figure of Merit
  - ✓ US-APWR SBLOCAs (ZIRLO cladding) satisfy the requirements in 10CFR50.46
    - Peak cladding temperature (PCT) shall not exceed 2200°F
    - Maximum cladding oxidation shall nowhere exceed 0.17 times the total cladding thickness
    - Maximum hydrogen generation shall not exceed 0.01 times the hypothetical amount generated in all the cladding cylinders
    - Core remains in a coolable geometry
    - Assurance of long-term decay heat removal

Basis of SBLOCA Methodology

- M-RELAP5 Code
  - ✓ Based on **RELAP5-3D** code with complete two-fluid thermal-hydraulics model applicable to important phenomena and processes occurring under US-APWR SBLOCAs
  - ✓ MHI's modification is limited to implementation of conservative models and plant-specific models so as to comply with regulatory requirements
  - ✓ M-RELAP5 code has been developed according to **Evaluation Model Development and Assessment Procedure (EMDAP)** in Regulatory Guide 1.203 "*Transient and Accident Analysis Methods*"
    - Development of SBLOCA PIRT
    - M-RELAP5 code assessment based on PIRT
    - Examination for code applicability to plant safety analyses

Features of M-RELAP5

- Appendix K Conservative Model
  - ✓ In conformance with Appendix K requirements related to the high-ranked PIRT, the following conservative models are implemented
    - ANS-1971 decay heat model
    - Fuel gap conductance model equivalent to fuel design code
    - Baker-Just metal-water reaction model
    - Cladding swelling and rupture model applicable to ZIRLO
    - Moody critical flow model
    - Rewet logic in the heat transfer model
      - No return to transition boiling during blowdown period
      - No return to nucleate boiling during blowdown period

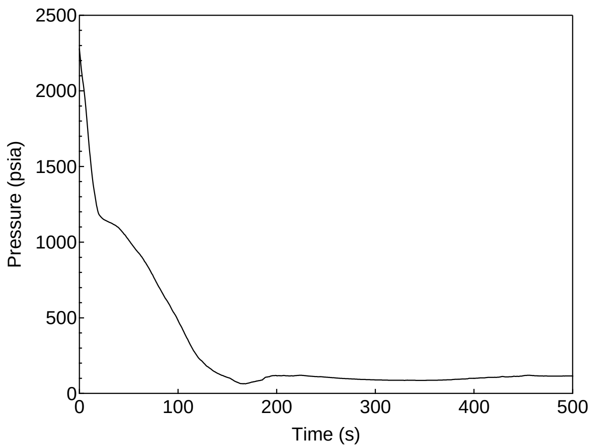
- Advanced Accumulator Model
  - ✓ Accumulator flow damper model based on ½-scale test data is newly implemented with the scaling bias determined from CFD calculations

US-APWR SBLOCA Summary

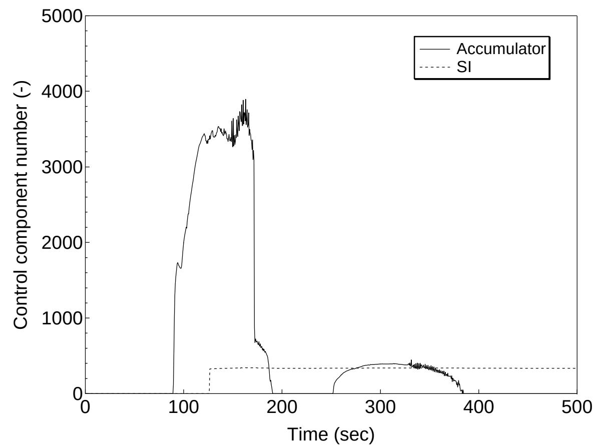
- Spectrum of Postulated Piping Breaks
  - ✓ Design-basis SBLOCAs with break sizes up to 1-ft^2 (cold leg)
  - ✓ MUAP-07025 "*Small Break LOCA Sensitivity Analyses for US-APWR*" provides the following sensitivity calculations
    - Break location
    - Break size
    - Break orientation
    - Plant noding
    - Time step size
    - Offsite power availability
    - Single failure assumption
    - Accumulator flow rate
- Sensitivity calculations to determine the limiting SBLOCA
  - ✓ Loop seal PCT : 7.5-in cold leg bottom break
  - ✓ Boil-off PCT : 1-ft^2 cold leg bottom break
  - ✓ Loss-of-offsite power and two ECCS trains not available due to single failure assumption and maintenance

US-APWR SBLOCA Results (Limiting Case)

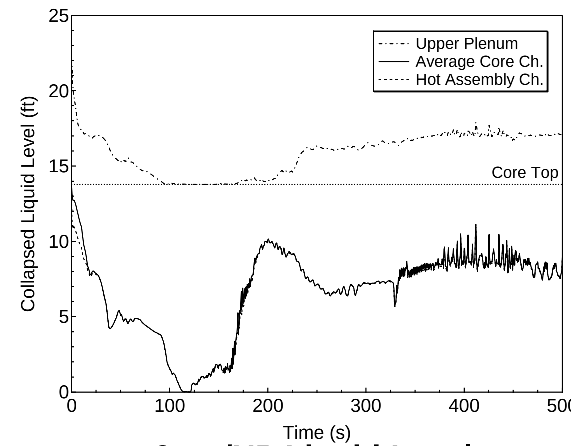
- 1-ft^2 Cold Leg Break
  - ✓ Rapid depressurization and PCT occurring during BO (PCT=1328˚F)
  - ✓ US-APWR ECCS flow sufficient to suppress PCT



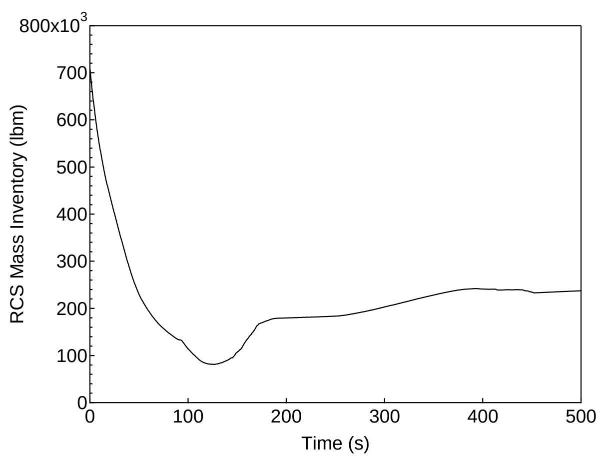
RCS Pressure



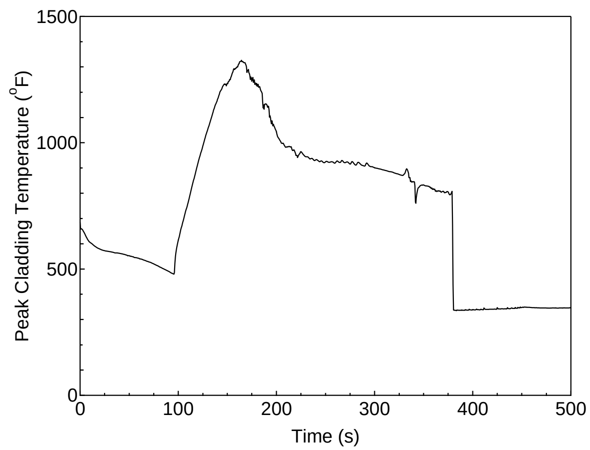
ECC Flow Rate



Core/UP Liquid Level



RCS Mass Inventory



PCT

Events	Time (sec)
Break occurs; blowdown initiation	0.0
Reactor trip (LOOP is assumed)	6.9
ECCS actuation signal	8.3
Control rod insertion starts	8.7
Main steam isolation	8.7
RCP trip	9.9
Main feedwater isolation	14.9
Main steam safety valve open	not actuated
Accumulator injection begins	89
Core upper region uncovery	96
Emergency Power Source initiates	111
High Head Injection System begins	126
Emergency feedwater flow begins	141
Peak Cladding Temperature occurs	164
Core upper region recovery	381

From MUAP-DC015 (R3) "US-APWR Design Control Document"

Summary

- US-APWR SBLOCA important phenomena have been identified and M-RELAP5 modeling capability has been assessed in conformance to EMDAP
- M-RELAP5 is capable of predicting all the high-ranked phenomena of SBLOCA
- M-RELAP5 is applied to the SBLOCA analysis for the US-APWR reported in DCD Chapter 15

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Protecting People and the Environment

Presentation to the ACRS Subcommittee

Small Break Loss of Coolant Accident Methodology for US-APWR

Safety Evaluation

July 9, 2012

NRO Staff Review Team

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Technical Staff

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  Reactor Systems, Nuclear Performance, and Code Review Branch
- **Andrew Bielen**
  Reactor Systems, Nuclear Performance, and Code Review Branch

Project Managers

- **Jeff Ciocco**
- **Michael Takacs**

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U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

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    Information System Laboratories
  - **Ed Throm**
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  - **Marcos Ortiz**
    Information System Laboratories

- **Contractor Confirmatory Analysis Support**
  - **Don Fletcher**
    Information System Laboratories
  - **Robert Beaton**
    Information System Laboratories
  - **David Caraher**
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    Information System Laboratories
  - **David Caraher**
    Information System Laboratories

Overview of Staff Review Process

- MHI's US-APWR SBLOCA methodology utilizes the M-RELAP5 code which is based on the Idaho National Laboratory RELAP5-3D® code
- Methodology applies only to US-APWR, which has unique SBLOCA PCT response
- Detailed confirmatory analysis performed using RELAP5 and M-RELAP5
- Significant issues found and resolved with critical flow model

Outstanding Issue

- Final accumulator flow bias still to be determined. This will affect final PCT, but does not affect the methodology

- Accumulator Topical Report SE will finalize scaling bias

- SBLOCA Topical Report and SE will be revised to reflect final scaling bias

US-APWR Significant Design Features

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- Advanced dual flow accumulator
- Compared to a Westinghouse 4-loop plant:
  - Increased HPI flow per MW is 2.8x greater at runout
  - Significantly increased steam generator heat transfer area (91,500 ft^2 vs 55,000 ft^2)
  - Larger pressurizer (2,900 ft^3 vs 1,800 ft^3)
- 14 ft active core region
- Neutron reflector
- Larger core (25% more power, lower average linear heat rate)

US-APWR SBLOCA Methodology

- MHI modified RELAP5-3D® by adding Appendix K features and an advanced accumulator model.

- Followed Regulatory Guide 1.203, "Transient and Accident Analysis Methods"

- Acceptance criteria for methodology is 10 CFR 50, Appendix K, "ECCS Evaluation Models"

- Staff review closely paralleled review of DCD Chapter 15.6.5, including SBLOCA break size sensitivity studies in the Technical Report, MUAP-07025-P

Regulatory Basis and Acceptance Criteria

Acceptance criteria for the SBLOCA methodology is 10 CFR 50.46 which requires that none of the criteria of paragraph 50.46(b) will be exceeded.

- PCT < 2200 ^{o}F
- Max Local Cladding Oxidation (LCO) < 0.17 initial cladding thickness
- Core Wide Oxidation (CWO) < 1% of maximum possible
- Core Remains in a Coolable Geometry
- Assurance of Long Term Decay Heat Removal

M-RELAP5 is used to verify the first three items.

PIRT Process

- Considered all PIRT items ranked high in each of the five transient phases: (1) Blowdown, (2) Natural Circulation, (3) Loop Seal Clearance, (4) Boiloff (BO) and (5) Core Recovery

- Model assessment cases were selected to evaluate the high ranking phenomena in each transient phase

- Staff agreed that high ranked phenomena had been correctly identified

Validation Test Matrix

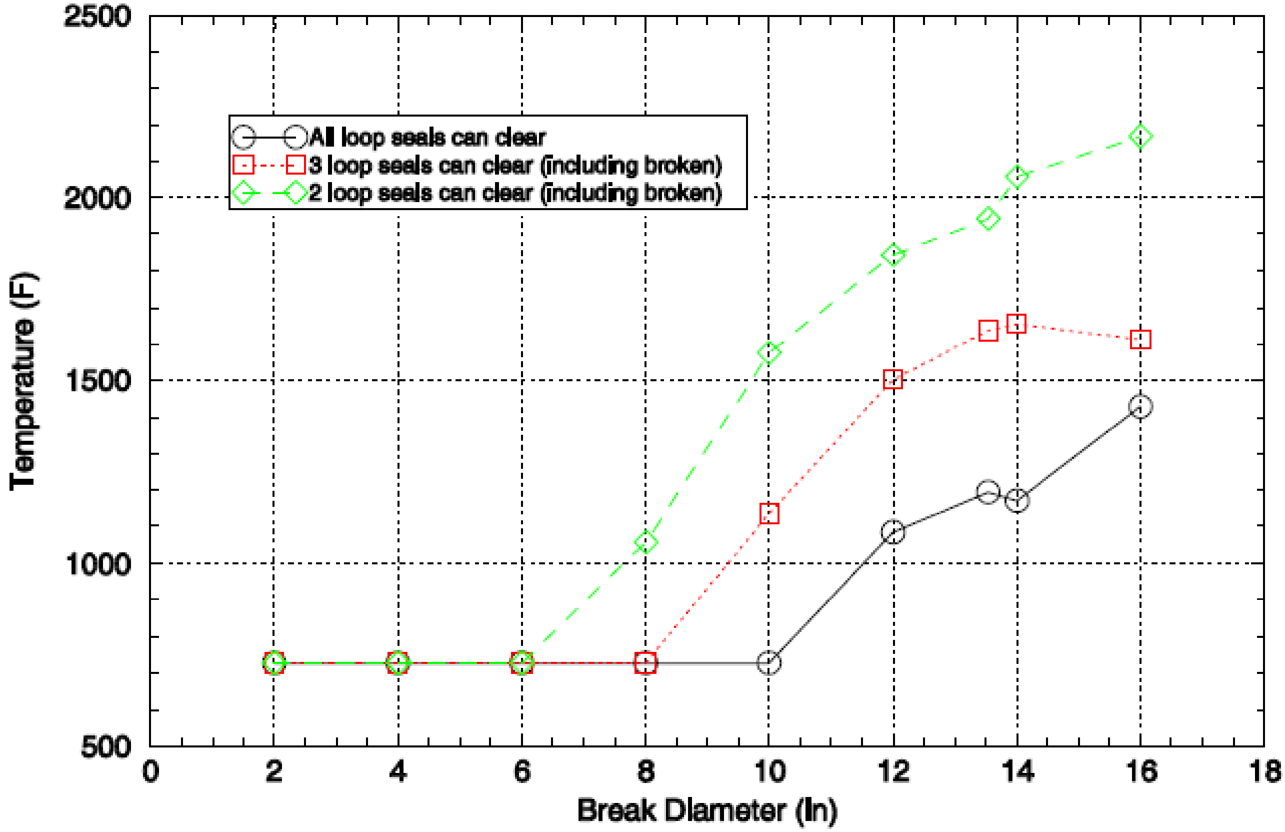
- Staff questioned the adequacy of the validation test matrix with regard to:
  - Larger break sizes: **ROSA-IV/LSTF SBLOCA (10 percent) Test (SB-CL-09) and ROSA/LSTF SBLOCA (17 percent) test (IB-CL-02) added,**
  - TMI action items: **(LOFT/L3-1 and Semiscale/S-LH-1 added),**
  - Reflood heat transfer **(Flecht-Seaset Forced-Reflood Test added)**
- MRELAP-5 demonstrated the ability to represent additional test data

SBLOCA Response

- The staff questioned the larger limiting breaks and no clad heatup in the typical 2”-4” break range
- The staff asked about other validation cases for larger break sizes
- The staff asked whether additional, larger test scale data was available to support the loop seal clearing phenomena
- The staff requested an explanation why the limiting breaks are larger than typically observed in a Westinghouse-type PWR

SBLOCA Response
(Slide 1/3)

PCT as a Function of Break Diameter and Number of Loop Seals Permitted to Clear



SBLOCA RESPONSE
(Slide 2/3)

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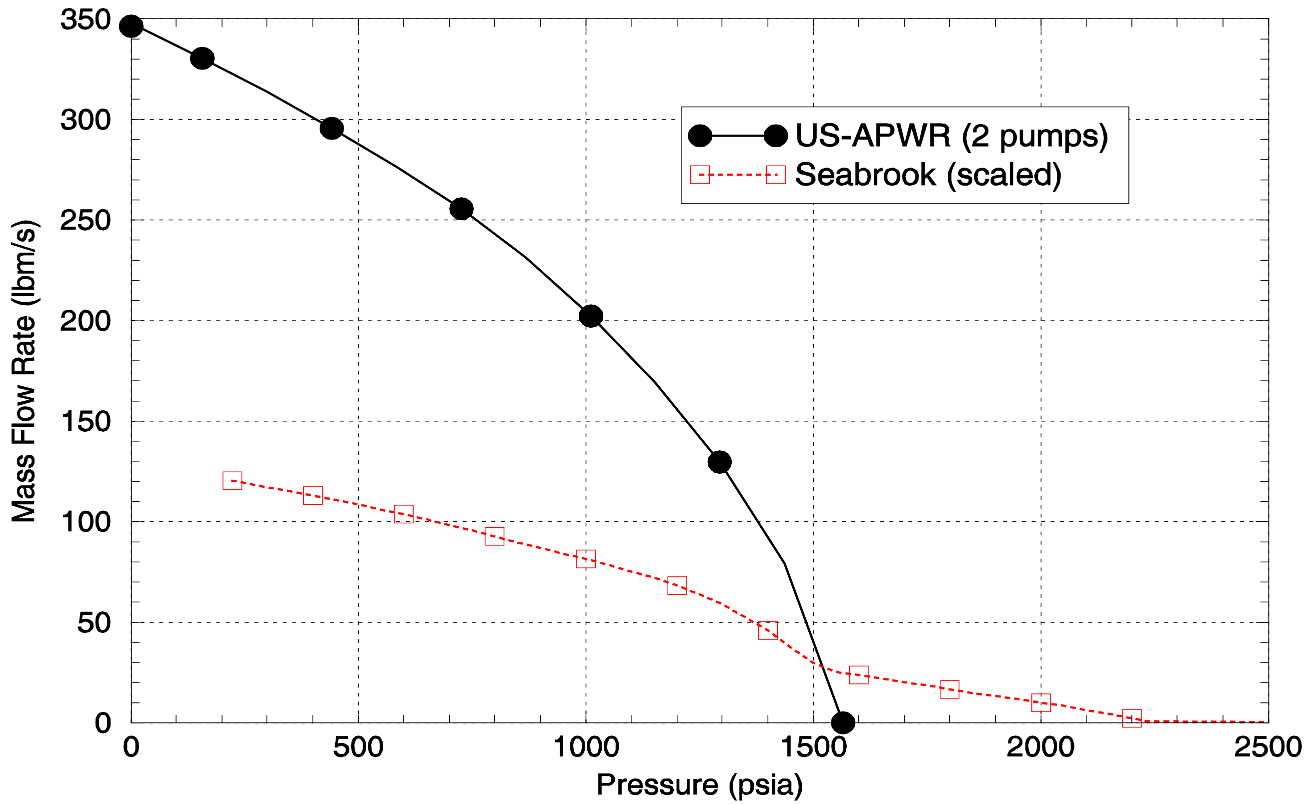
Total HPI Flow US-APWR and Core-Power Scaled Conventional PWR



SBLOCA RESPONSE
(Slide 3/3)

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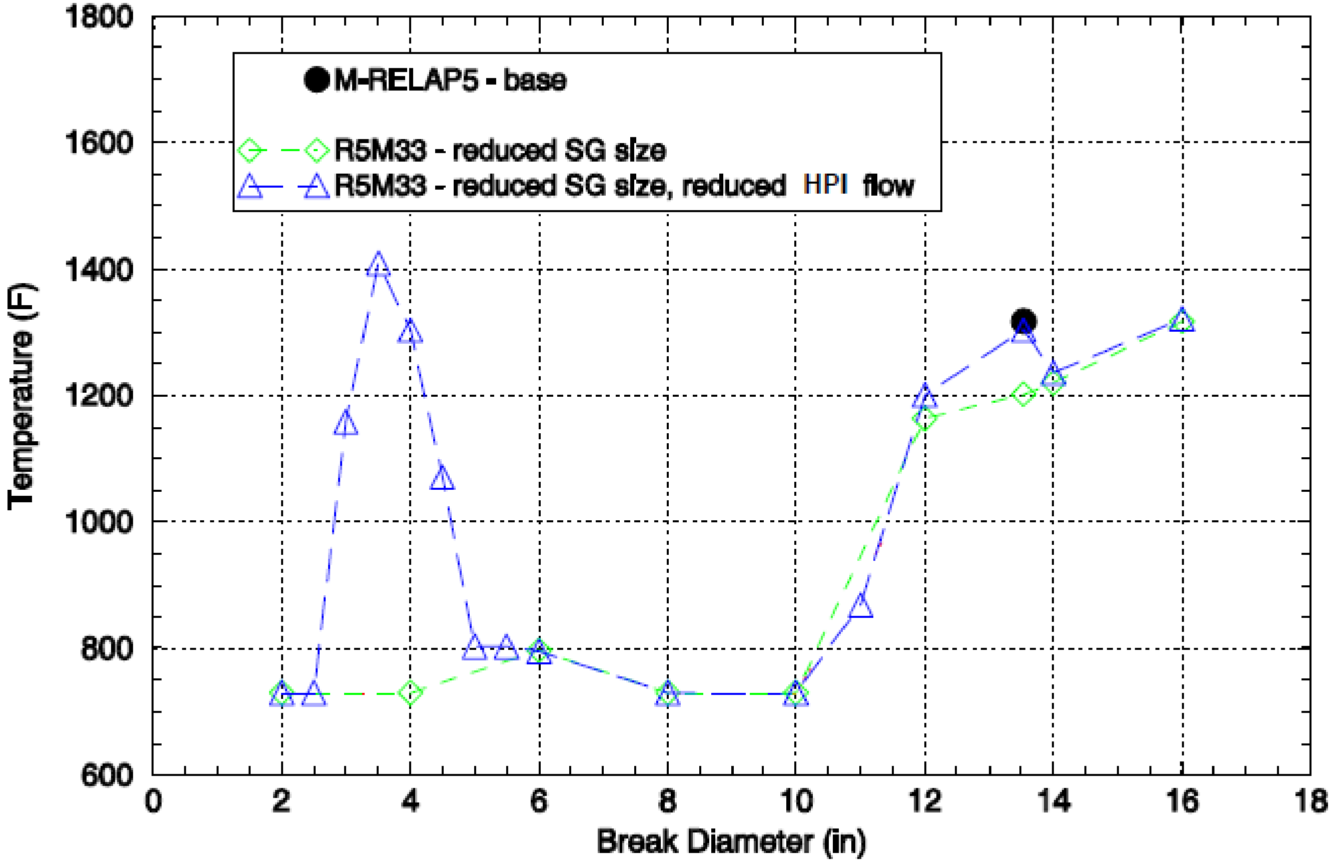


Critical Flow Model

- Moody model and pull through model are implemented using a stub pipe connected to the main coolant pipe.
- Stub pipe flow area and length are assumed values.
- Initial confirmatory calculations with RELAP5/MOD3.3 and M-RELAP5 showed strong PCT dependence on assumed stub pipe geometry.

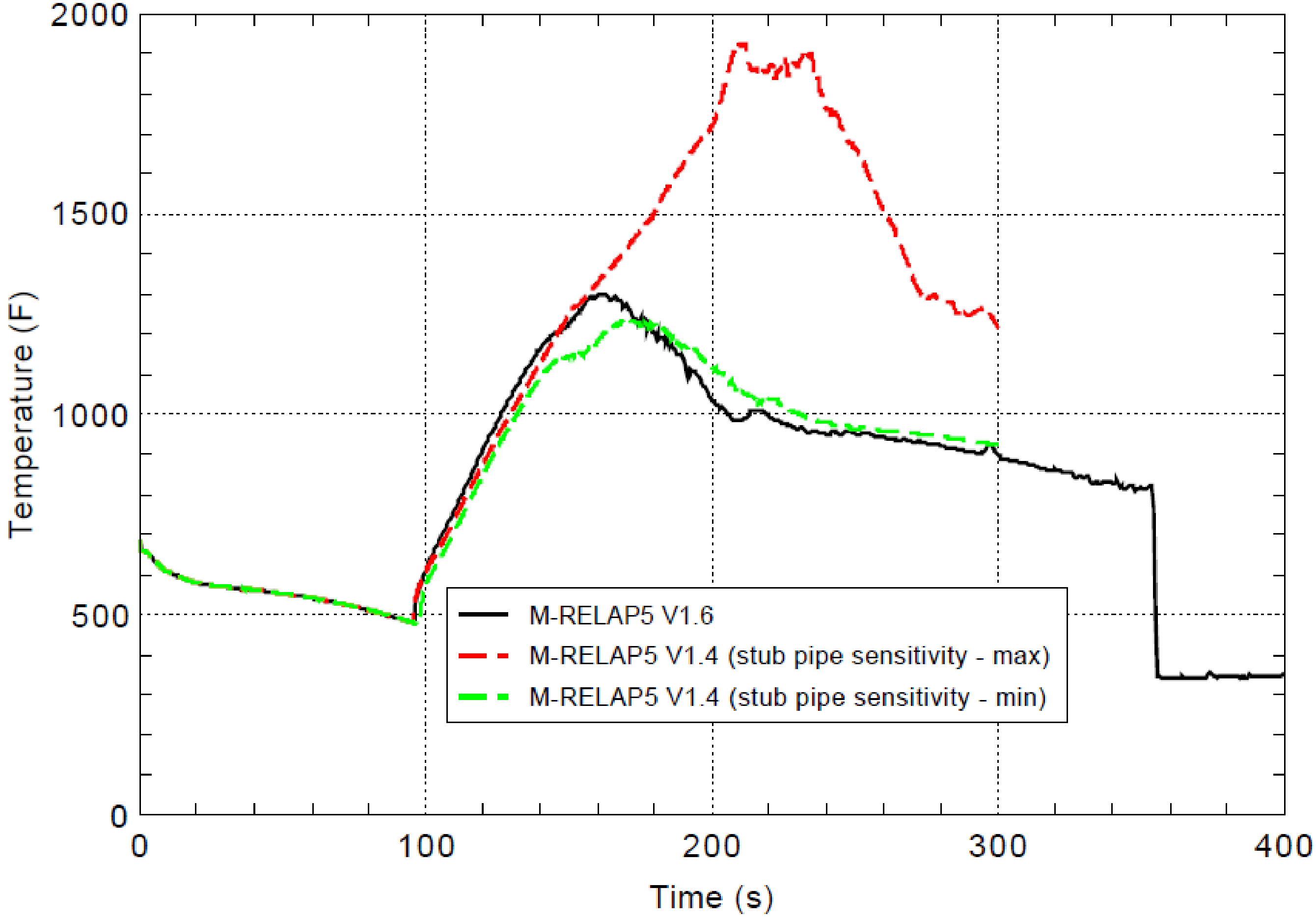
Critical Flow Model

- Confirmatory calculations with M-RELAP5 showed that:
  - The Moody model was not consistently used for two-phase blowdown conditions as required by Appendix K
  - Pull-through model switching logic problems

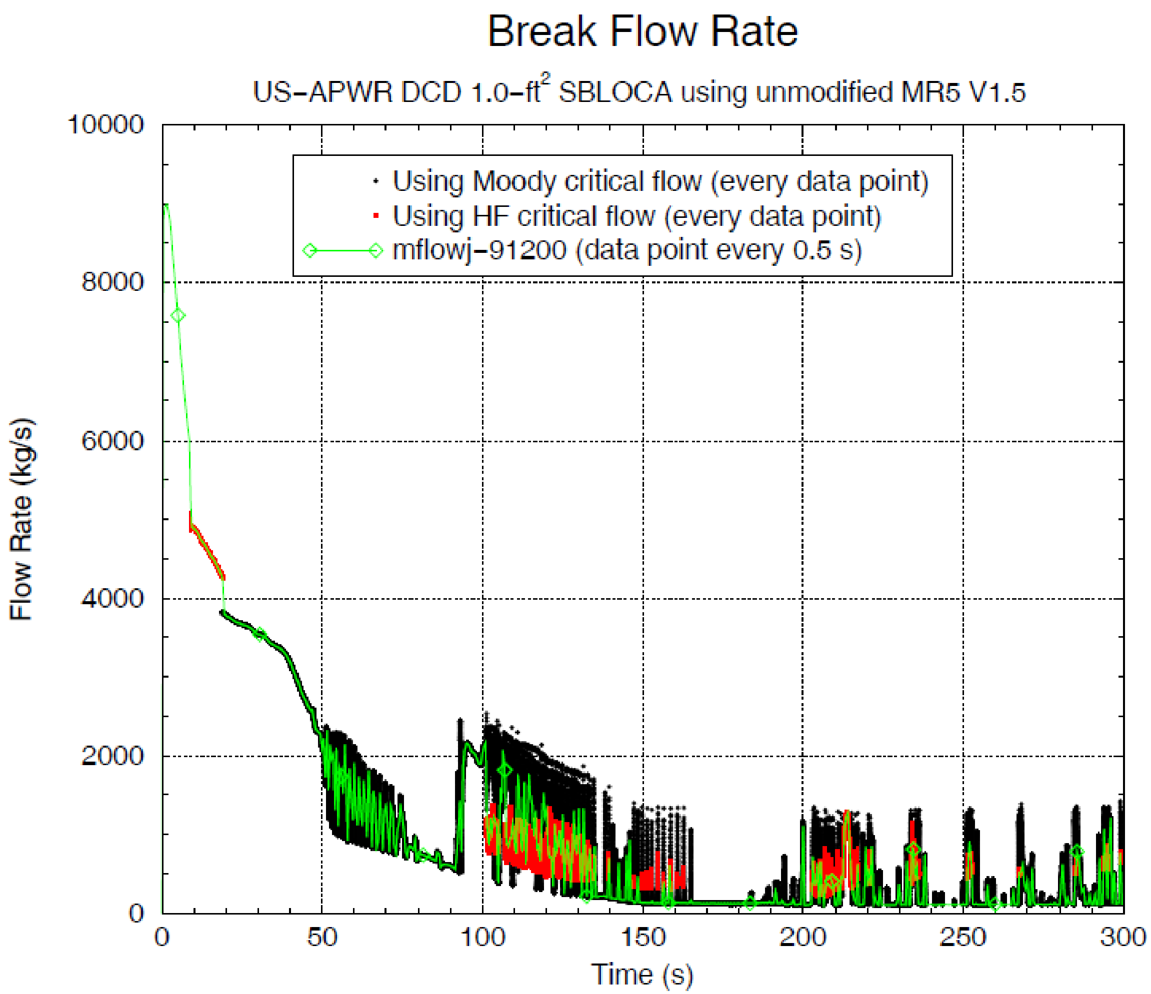
- MHI corrected issues with the horizontally stratified flow regime and critical flow switching logic;
  - Eliminated strong dependence of PCT on stub pipe geometry [reduced from >600 °F (333 °C) to 38 °F (21 °C)]
  - Bottom of the cold leg break became limiting (in line with other PWRs) instead of the top

Critical Flow Model
(Slide 1 of 2)

PCT Response with M-RELAP5 Versions 1.4 and 1.6



Critical Flow Model
(Slide 2 of 2)



Scaling

- No detailed scaling analysis for Topical Report Rev. 0
  - Provided a comparison to 4-loop W plant
- Staff questioned relevance of scaling comparison to 4-loop W plant
- A quantitative scaling analysis was requested
- The initial scaling analysis was limited to a single facility and a single test. Did not cover ranges of phenomena.

Scaling (cont)

- MHI provided final scaling report with multiple ROSA-LSTF cases.
- Top-down scaling demonstrated that the relative rankings of non-dimensional coefficients between the plant and test facility were preserved. Therefore:
  - no new or different system interactions from those exhibited in the test facilities.
  - the same dominant processes and phenomena occur in the US-APWR and the IETs.
- Experimental data are acceptable for validation of the M-RELAP5 code for SBLOCA analysis.
- Minor scaling distortions identified in top-down analysis were evaluated in bottom-up scaling confirming applicability of the test data.

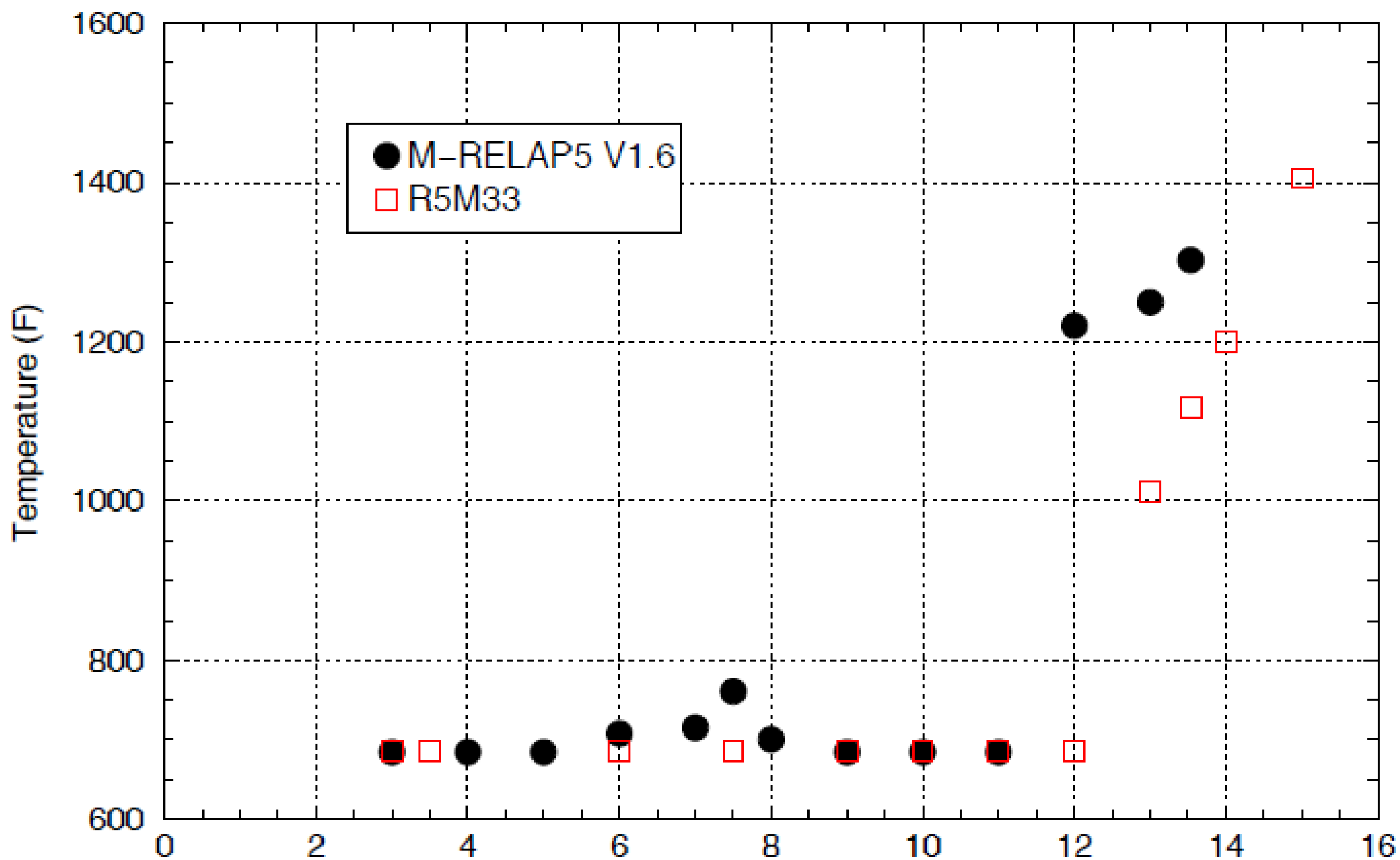
Part IV. Conclusions

MHI's SBLOCA methodology is acceptable for calculating PCT, hydrogen generation and fuel cladding surface oxidation for the US-APWR in accordance with Appendix K Evaluation Model requirements

Current preliminary PCT of 720 ^{o}C (1328 ^{o}F) shows significant margin to Acceptance Criteria of 1204 ^{o}C (2200 ^{o}F)

Confirmatory Calculation

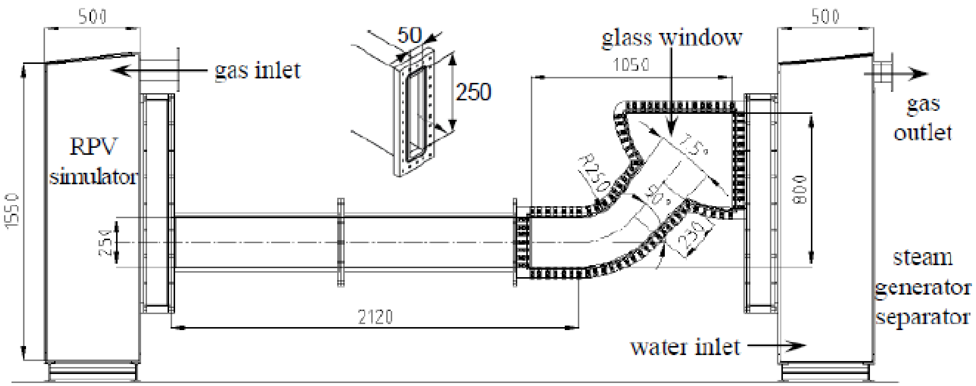
PCT Break Spectrum for M-RELAP5 and RELAP5/MOD 3.3



Break Diameter (inches)

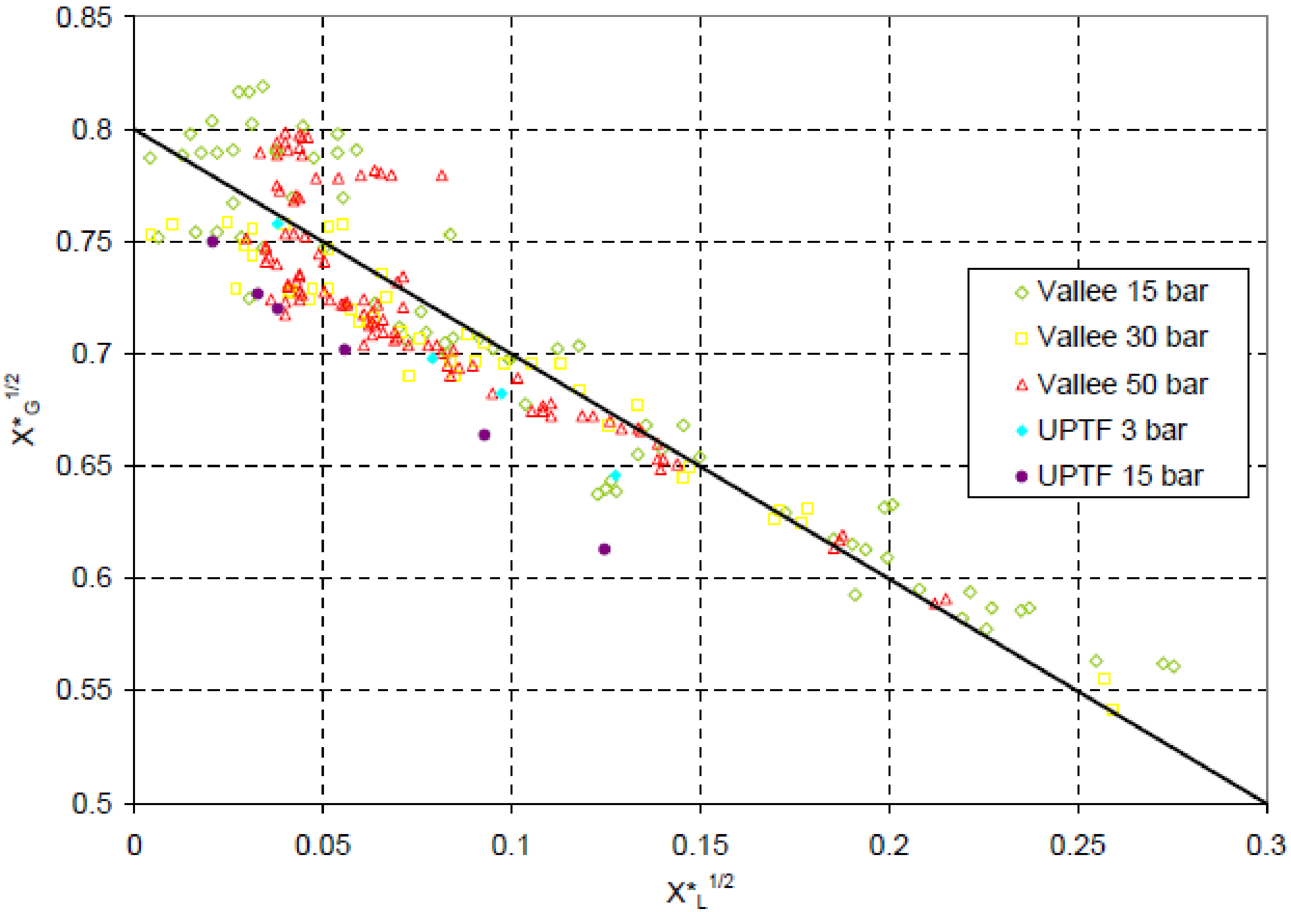
CCFL (Slide 2 of 3)

- Dresden (Vallee) conducted experiments to supply suitable data for CFD code validation.



- Hot leg model devoted to optical measurement. A rectangular cross section was used. Well suited for visual observation of the flow regimes in the hot leg.
- The geometry is not prototypical of a hot leg.
- Experiment is 1/3 scaled
- UPTF data used by MHI is full scale and better suited for use in modeling CCFL in the hot leg of a PWR.

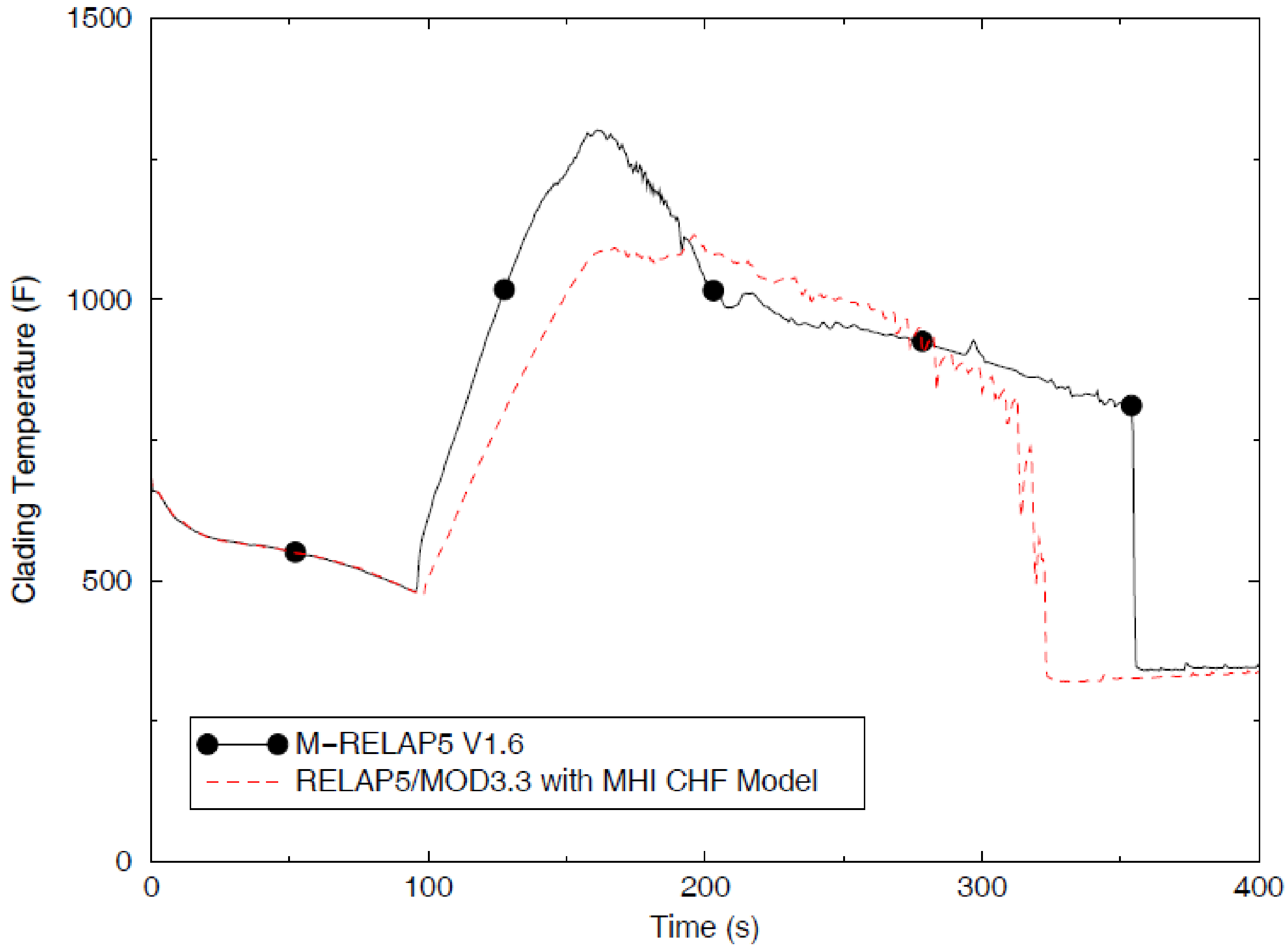
CCFL (Slide 3 of 3)



UPTF Data with Dresden (Valle) Correlation

Confirmatory Calculation

Comparison of M-RELAP5 and RELAP5/MOD 3.3 for Limiting Break



Presentation to ACRS Subcommittee

Overview of

MUAP-07011 R3: Large Break LOCA Code Applicability Report for US-APWR

July 9, 2012

Mitsubishi Heavy Industries, Ltd.

MHI Presenters

- Tetsuya Teramae (Presenter)

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  Reactor Safety Engineering Department

  Mitsubishi Heavy Industries, Ltd.

- Reiko Narutomi (Co-presenter)

  Deputy Manager

  Safeguard System Engineering Section

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  Mitsubishi Heavy Industries, Ltd.

Contents of Presentation

- Highlights of the Topical Report
- LBLOCA Code and Methodology
- Sample Plant Analysis
- Summary

Highlights of the Topical Report

- **Objective**
  - ✓ To present comprehensive assessment of the applicability of WCOBRA/TRAC(M1.0) code with ASTRUM methodology to US-APWR LBLOCA analysis
- **Key Topics**
  - ✓ Identification of US-APWR design features relevant to LBLOCA
  - ✓ Confirmation of code and methodology applicability based on Code Scaling, Applicability, and Uncertainty (CSAU) approach
- **Relationship with DCD**
  - ✓ WCOBRA/TRAC(M1.0) with the ASTRUM methodology is being applied to the LBLOCA analyses for US-APWR design certification
  - ✓ The present topical report is referenced in Section 15.6.5 of the US-APWR DCD

LBLOCA Code and Methodology (1/3)

- WCOBRA/TRAC(M1.0)
  - ✓ WCOBRA/TRAC code (Mod.7A Rev.6) with the following features:
    - ACC
    - NR
  - ✓ Calculation of thermal-hydraulic behavior during LBLOCA
- HOTSPOT
  - ✓ Same fuel rod analysis model as that used in WCOBRA/TRAC
  - ✓ Calculation of the effect of uncertainties at axial locations of the fuel rods
  - ✓ Simulation of cladding burst, metal-water reaction, and fuel relocation following burst phenomena
- ASTRUM (Automated Statistical TReatment of Uncertainty Method)
  - ✓ Non-parametric order statistical methodology that does not assume Peak Cladding Temperature (PCT) distribution
  - ✓ Statistical method determines that 124 cases must be run to assure 95/95 PCT, Local Maximum Oxidation (LMO), and Core-Wide Oxidation (CWO)

LBLOCA Code and Methodology (2/3)

- Applicability of WCOBRA/TRAC to US-APWR has been evaluated and confirmed especially in simulating high-ranked LBLOCA phenomena and new or improved design features
  - ✓ Longer Core (14 ft length)
  - ✓ Advanced Accumulator (ACC)
  - ✓ Direct Vessel Injection (DVI) for Safety Injection Pump (SIP)
  - ✓ Neutron Reflector (NR)

LBLOCA Code and Methodology (3/3)

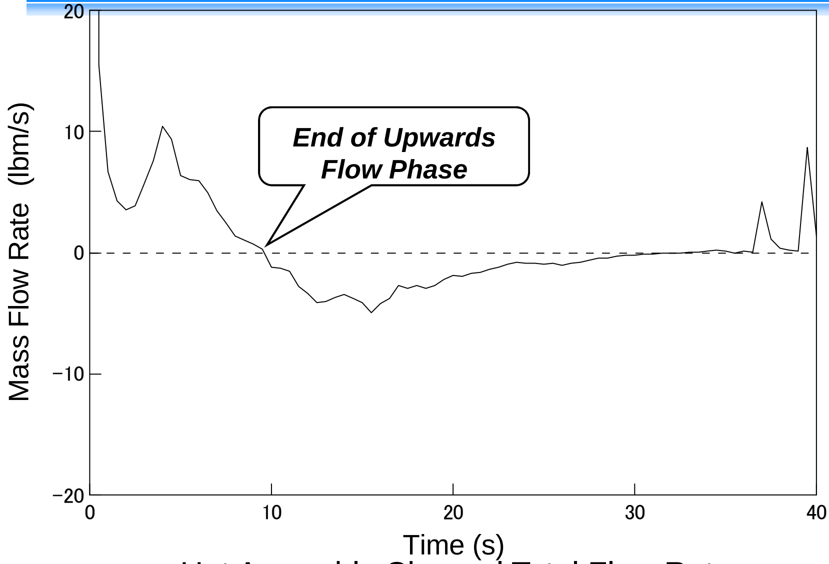
- Applicability of the code and methodology for US-APWR has been examined based on Code Scaling, Applicability, and Uncertainty (CSAU) approach
  - ✓ Identify and Rank Phenomena (PIRT)
    - Discuses and shows features of the US-APWR that affect the applicability of the existing PIRT
  - ✓ Code Applicability and Assessment
    - Applicability and assessment for features of the US-APWR has been examined based on PIRT result
  - ✓ Define Nodalization for Nuclear Power Plant (NPP) Calculations
    - Nodalization for US-APWR has been developed based on the same scheme approved for current 3- and 4-loop PWRs
  - ✓ Perform Sample Plant Analysis
  - ✓ Uncertainty Treatment

Sample Plant Analysis (1/2) –Reference Case-

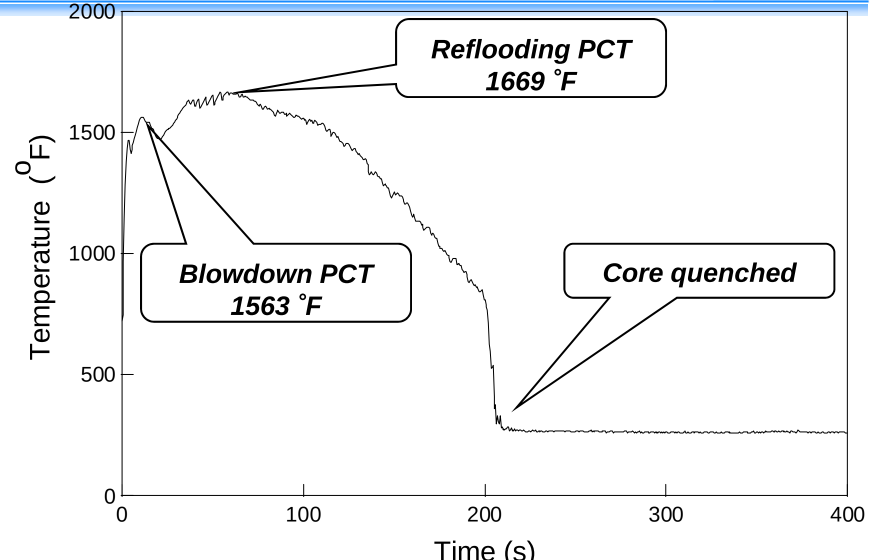
US-APWR

- Sample plant analysis adopts DCD Section 15.6.5 Rev. 3 "Reference Case"
- Major LOCA parameters for WCOBRA/TRAC(M1.0) are listed in Table 3.6-5 in the topical report.
- Value of statistical parameters are basically nominal for the reference case
- Nodalization for Plant Analysis has been developed based on the same scheme as approved for current 3- and 4- loop PWRs

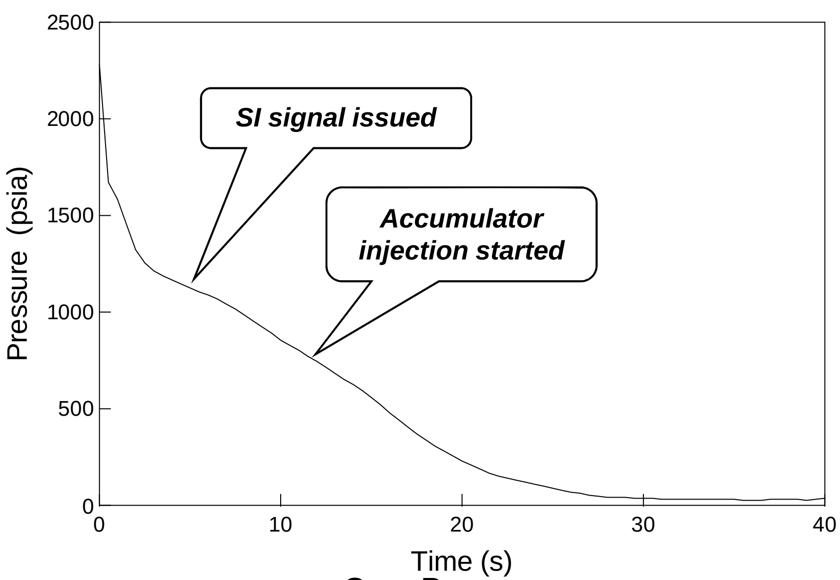
Sample Plant Analysis (2/2) –Reference Case-



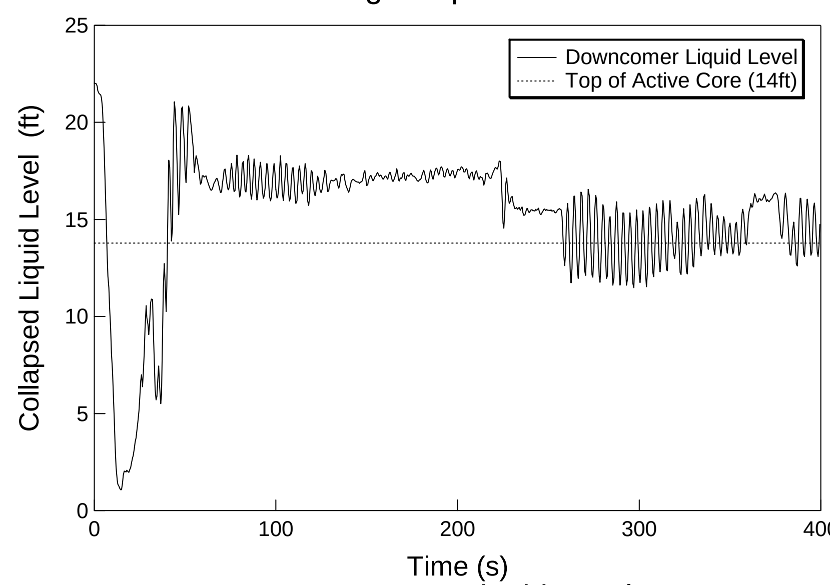
Hot Assembly Channel Total Flow Rate



Peak Cladding Temperature of Hot Rod



Core Pressure



Downcomer Liquid Level

Summary

- WCOBRA/TRAC(M1.0) code with ASTRUM methodology is used for US-APWR LBLOCA analysis
- New or improved US-APWR design features have been evaluated for the code applicability
- Applicability of WCOBRA/TRAC(M1.0) code and ASTRUM methodology to US-APWR has been examined and confirmed based on the CSAU approach
- WCOBRA/TRAC(M1.0) was applied to a sample US-APWR plant analysis and its capability to simulate the US-APWR LBLOCA transient was demonstrated
- The US-APWR LBLOCA analysis is reported in DCD Section 15.6.5

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

Large Break Loss of Coolant Accident Code Applicability Report for US-APWR

Safety Evaluation

July 9, 2012

Staff Review Team

United States Nuclear Regulatory Commission
Protecting People and the Environment

Technical Staff

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Project Managers

- **Jeff Ciocco**
- **Michael Takacs**

Overview of Staff Review Process

- Review is of the Applicability of the Approved Westinghouse ASTRUM Methodology to the US-APWR
  - CSAU Issues
  - WCT-M1 Applicability
  - Unique Features (Neutron Reflector, Advanced Accumulator)
  - Scaling Issues

- Confirmatory analysis performed using RELAP5 and WCT-M1

- Staff review identified coding errors in WCT-M1 and HOTSPOT
  - Corrected by applicant

Outstanding Issue

- Final accumulator flow bias still to be determined. This will affect final PCT, but does not affect the methodology

- Accumulator Topical Report SE will finalize scaling bias

- LBLOCA Topical Report and SE will be revised to reflect final scaling bias

U.S. APWR LBLOCA Methodology

- US-APWR LBLOCA methodology utilizes the ASTRUM Methodology (WCAP-16009-P-A)
  - WCT-M1 is used to calculate the global NSSS response
  - HOTSPOT is used to compute local fuel rod response
  - An automated process drives the statistical sampling, assignment of uncertainty parameter values in WCT-M1 and HOTSPOT, running of the 124 WCT-M1 and HOTSPOT cases, and the tabulation of results
  - Conforms to Regulatory Guide 1.157, "Best Estimate Calculation of Emergency Core Cooling System Performance"

US-APWR Significant Design Features

- Advanced dual flow accumulator
- Compared to a Westinghouse 4-loop plant:
  - Increased HPI flow. HPI flow per MW is 2.8x greater at runout
  - Significantly increased steam generator heat transfer area ($91,500\ ft^2$ vs. $55,000\ ft^2$)
  - Larger pressurizer ($2,900\ ft^3$ vs $1,800\ ft^3$)
- 14 ft active core region
- Neutron reflector
- Larger core (25% more power, lower average linear heat rate)

Regulatory Basis and Acceptance Criteria

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Acceptance criteria for BE LBLOCA methodology is 10 CFR 50.46(a)(1)(i) which requires that it be shown with a high probability that none of the criteria of paragraph 50.46(b) will be exceeded.

- PCT < 1204 C (2200 F)
- Maximum Local Cladding Oxidation (LCO) < 0.17 initial cladding thickness
- Core Wide Oxidation (CWO) < 1% of maximum possible
- Core Remains in a Coolable Geometry
- Assurance of Long Term Decay Heat Removal

PIRT Process

- Staff reviewed all PIRT items ranked high in each of the three transient phases: Blowdown, Refill, and Reflood
- Staff noted initial power level was sampled about the nominal power.
  - Staff requested no sampling at less than nominal power
- Staff agreed that high ranked phenomena had been correctly identified, and their uncertainties were either properly accounted for or they were treated in a conservative fashion

Code Applicability

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- WCT-M1 is a derivative of WCOBRA/TRAC
  - Applicability justification relies heavily on WCAP-12945-P-A, “Code Qualification for Best Estimate LOCA Analysis”
- Staff review focused on applicability of WCT-M1 to model the unique features of the US-APWR
  - Core Length – Staff requested further justification of heat transfer correlations
  - Advanced Accumulator – Additional justification of uncertainties required by the staff
  - Neutron Reflector – Staff reviewed the full scale NR test program results
  - Direct Vessel Injection of HHSI – Staff confirmed WCT-M1 treated it conservatively by ignoring downward directed momentum.
- Staff review uncovered an error in WCT-M1
  - Gravitational head improperly accounted for in the RCP
  - Applicant fixed WCT-M1 and demonstrated the error had no significant impact upon computed results.

Code Applicability (cont.)

- HOTSPOT
  - Nominal local conditions provided by WCT-M1
  - 15 uncertainty parameters treated in HOTSPOT
    - Fuel parameters, rod gap and rod surface HTCs, clad rupture parameters
- Staff review uncovered an error in HOTSPOT
  - Reflood HTC could exceed upper limit in certain cases
  - Applicant corrected coding and demonstrated no impact on calculated safety parameters

Code Applicability (cont.)

- Scaling
  - Power to Volume Ratio of the RCS is 95% of conventional 4 loop PWR
    - US-APWR Power is 1.3 of conventional plant
    - US-APWR RCS volume is 1.37 of conventional plant
  - Diameter of RPV is 20% greater than conventional plant and there are more upper plenum internals
    - Upper plenum de-entrainment expected to be greater than, and hot leg entrainment less than, conventional plant
- Staff review concluded WCT-M1 can adequately account for the larger size of the US-APWR

Confirmatory Analyses

- RELAP5/Mod3.3
  - Agreed well overall with WCT-M1 Reference Case
    - Reflood PCT's within 10 C of one another
    - Hot spot quench times within 30 seconds of one another
- WCT-M1 and HOTSPOT
  - Confirmatory calculations:
    - Identify PCT sensitivities to selected uncertainty parameters
    - Examine calculational details not reported in the DCD
    - Independently verify results reported by the applicant

Conclusion:

MHI’s LBLOCA methodology is acceptable for calculating PCT, hydrogen generation, and fuel cladding surface oxidation for the US-APWR

Current preliminary PCT of 963 C (1766 F) shows significant margin to Acceptance Criteria 1204 C (2200 F)

Hot Leg Entrainment (backup slide)

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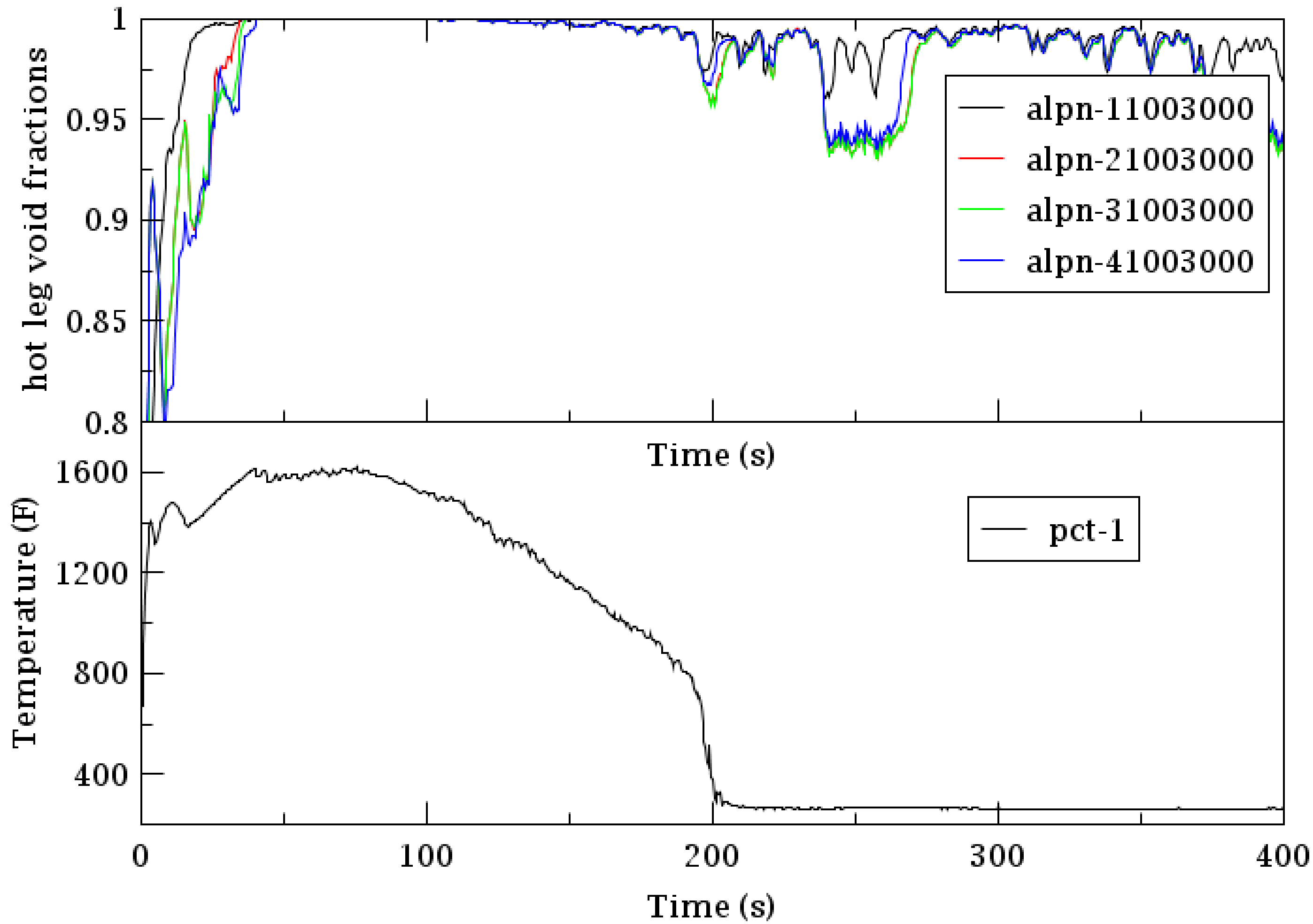
- There is no holdup of liquid in hot legs until after PCT



Presentation to ACRS Subcommittee

Chapter 15: Transient and Accident Analyses

July 9, 2012

Mitsubishi Heavy Industries, Ltd.

MHI Presenters

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Acronyms

ACRS	:Advisory Committee on Reactor Safeguards
AOO	:Anticipated Operation Occurrence
ASTRUM	:Automated Statistical Treatment of Uncertainty Method
ATWS	:Anticipated Transient Without Scram
BOC	:Beginning of Cycle
BWR	:Boiling Water Reactor
CCF	:Common Cause Failure
CFR	:Code of Federal Regulations
CVCS	:Chemical and Volume Control System
CWO	:Core Wide Oxidation
D3	:Defense-in-Depth and Diversity
DAS	:Diverse Actuation System
DCD	:Design Control Document
DNB	:Departure from Nucleate Boiling
DNBR	:Departure from Nucleate Boiling Ratio
DVI	:Direct Vessel Injection
EAB	:Exclusion Area Boundary
ECCS	:Emergency Core Cooling System
EFWS	:Emergency Feedwater System
EOC	:End of Cycle
ERG	:Emergency Response Guideline
ESF	:Engineered Safety Feature
GSI	:Generic Safety Issue
HFP	:Hot Full Power
HZP	:Hot Zero Power
LBLOCA	:Large Break Loss of Coolant Accident
LMO	:Local Maximum Oxidation
LOCA	:Loss of Coolant Accident

Acronyms (continued)

LOOP	:Loss of Offsite Power
LPZ	:Low Population Zone
MCR	:Main Control Room
MHI	:Mitsubishi Heavy Industries, Ltd.
MSRV	:Main Steam Relief Valve
MSSV	:Main Steam Safety Valve
N/A	:Not Applicable
NRC	:U.S. Nuclear Regulatory Commission
OLM	:On Line Maintenance
PA	:Postulated Accident
PCMI	:Pellet/Cladding Mechanical Interaction
PCT	:Peak Cladding Temperature
PWR	:Pressurized Water Reactor
RAI	:Request for Additional Information
RCCA	:Rod Cluster Control Assembly
RCS	:Reactor Coolant System
RCP	:Reactor Coolant Pump
RG	:Regulatory Guide
RTS	:Reactor Trip System
RWSP	:Refueling Water Storage Pit
SBLOCA	:Small Break Loss of Coolant Accident
SG	:Steam Generator
SGTR	:Steam Generator Tube Rupture
SI	:Safety Injection
SRP	:Standard Review Plan
TS	:Technical Specification
TSC	:Technical Support Center

Introduction

- US-APWR design is essentially the same as current PWRs in the US and Japan

  - ✓ Primary and secondary system configuration
  - ✓ Thermal hydraulics characteristics of the coolant
  - ✓ Fuel properties
  - ✓ Reactor control and protection system functional design
  - ✓ Active safety systems

- As a result, there are no new AOO or PA events for the US-APWR design compared to US operating PWRs

Introduction (continued)

- *US-APWR Plant Parameter Summary*
  - ✓ Large core thermal power
  - ✓ Large thermal margins due to lower average linear heat rate

Features	US-APWR	Typical US Current 4 Loop Plant
Core thermal power (MWt)	4,451	3,565
Number of loops, SGs, and RCPs	4	4
Number of fuel assemblies	257	193
Fuel rod lattice	17 x 17	17 x 17
Active fuel length (ft)	14	12
Average linear heat rate (kW/ft)	4.6	5.7
Reactor coolant pump type	Centrifugal	Centrifugal
Steam generator type	U-Tube	U-Tube

Introduction (continued)

- US-APWR Design Features
  - ✓ Very similar to US operating PWRs
  - ✓ Design features and their effects on safety analyses

Features	Effects on Safety Analyses
Neutron Reflector	Neutron Reflector is explicitly modeled in LOCA analyses
Simplified core lower plenum	Core inlet mixing among loops approximately the same
Pressurizer	Larger steam space moderates pressure transients
Steam generator	Smaller U-tube diameter moderates SGTR
EFWS	4 independent trains with one pump to each SG
Diverse actuation system	Satisfies design requirements to cope with ATWS
Advanced Accumulator	Characteristics of advanced accumulator is modeled in LOCA analyses Not expected to actuate during Non-LOCA events
RWSP	In-containment refueling water storage pit
DVI	High head SI flow goes into vessel directly

Contents of DCD Chapter 15

Section No.	Description
15.0	Transient and Accident Analyses
15.1	Increase in Heat Removal by the Secondary System
15.2	Decrease in Heat Removal by the Secondary System
15.3	Decrease in Reactor Coolant System Flow Rate
15.4	Reactivity and Power Distribution Anomalies
15.5	Increase in Reactor Coolant Inventory
15.6	Decrease in Reactor Coolant Inventory
15.7	Radioactive Release from a Subsystem or Compartment
15.8	Anticipated Transients without Scram
15A	Evaluation Models and Parameters for Analysis of Radiological Consequences of Accidents

- **This section contains generic information that applies to the Chapter 15 events:**
  - ✓ Classification of events (i.e. AOO, PA)
  - ✓ Initial conditions assumed in the analyses
  - ✓ Reactor trip system and ESF assumptions
  - ✓ Single failure assumptions
  - ✓ Non safety-related system assumptions
  - ✓ Operator action assumptions
  - ✓ LOOP assumptions
  - ✓ Long term cooling analyses
  - ✓ RCP seal integrity

- **Event specific information is included in the section for the applicable event**

US-APWR

➢ Open Items

Open Item No.	RAI No.	Question No.	RAI Topic / NRC Concern	RAI Response / DCD Impact
15.00-3	882-6237	15-37	**Reload evaluation:** Update reload evaluation methodology for consistency with Chapter 15.	➢ MHI revised MUAP-07026 to ensure consistency with the methodology in Chapter 15.

15.0.2 Review of Transient and Accident Analysis Methods

- **LOCA, Non-LOCA, and Dose analyses follow regulatory guidance in SRP Chapter 15**

- **Non-LOCA codes used for DCD Chapter 15**
  - ✓ *MARVEL-M*: plant system and transient analysis code
  - ✓ *TWINKLE-M*: multi-dimensional neutron kinetics code
  - ✓ *VIPRE-01M*: subchannel thermal hydraulics analysis and fuel transient code
  - ✓ *ANC*: three-dimensional two-group diffusion core calculation code

- **Above Non-LOCA computer codes and methodology used by MHI are described in Non-LOCA Methodology Topical Report (MUAP-07010)**

15.0.2 Review of Transient and Accident Analysis Methods

- **LOCA codes used for DCD Chapter 15**
  - ✓ *WCOBRA/TRAC(M1.0)*: thermal-hydraulic behavior calculation code during LBLOCA
  - ✓ *HOTSPOT*: effect of uncertainties at axial location of fuel rod calculation code during LBLOCA
  - ✓ *M-RELAP5*: thermal-hydraulic behavior and safety performance calculation code during SBLOCA

- **Above LOCA computer codes and methodology used by MHI are described in reports:**
  - ✓ MUAP-07011 Large Break LOCA Code Applicability Report for US-APWR
  - ✓ MUAP-07013 Small Break LOCA Methodology for US-APWR

- **Dose code used for DCD Chapter 15**
  - ✓ *RADTRAD*: dose estimation in design basis accident code

Open Items

Open Item No.	RAI Topic / NRC Concern
15.00-4 (15.4-1*)	NRC staff is currently reviewing the Non-LOCA Methodology Topical Report, MUAP-07010.
15.00-5 (15.4-2*)	NRC staff is currently reviewing the Thermal Design Methodology Topical Report, MUAP-07009.
15.00-6	NRC staff is currently reviewing the LBLOCA Applicability Topical Report, MUAP-07011.
15.00-7	NRC staff is currently reviewing the SBLOCA Methodology Topical Report, MUAP-07013.

*The identical Open Item is repeated in Section 15.4 of the Safety Evaluation using a different Open Item number.

15.0.3 Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors

- **Methodology for estimation of radiological consequences from postulated accidents is based on RG 1.183**
- **Doses are evaluated for the EAB, the outer boundary of the LPZ, the MCR, and the TSC**
- **Source term assumptions, atmospheric dispersion factors, dose conversion factors, and airborne radioactivity removal coefficients are described**
- **Representative analysis shown as part of Section 15.6 presentation**

15.0.3 Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors

US-APWR

Section No.	Event	Category	Computer Code(s)
15.1.5.5	Radiological Consequences of Main Steam Line Break Outside Containment	PA	RADTRAD
15.3.3.5	Radiological Consequences of Reactor Coolant Pump Rotor Seizure	PA	RADTRAD
15.4.8.5	Radiological Consequences of Rod Ejection Accident	PA	RADTRAD
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	PA	RADTRAD
15.6.3.5	Radiological Consequences of Steam Generator Tube Rupture	PA	RADTRAD
15.6.5.5	Radiological Consequences of Loss-of Coolant Accident	PA	RADTRAD MicroShield*1
15.7.4	Radiological Consequences of Fuel Handling Accident	PA	RADTRAD

***1:Direct radiation doses in MCR and TSC from the containment, radioactive plume and the MCR or TSC emergency filtration unit are calculated with the MicroShield code. The direct radiation shine dose in accident condition is represented by direct radiation shine dose for LOCA.**

15.0.3 Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors

- **There are no Open Items for DCD Section 15.0.3**

15.1 Increase in Heat Removal by the Secondary System

Section No.	Event	Category	Computer Code(s)
15.1.1	Decrease in feedwater temperature	AOO	MARVEL-M
15.1.2	Increase in feedwater flow	AOO	MARVEL-M
15.1.3	Increase in steam flow	AOO	MARVEL-M
15.1.4	Inadvertent opening of a steam generator relief or safety valve	AOO	MARVEL-M, ANC, VIPRE-01M
15.1.5	Steam system piping failures – Minor/Major	AOO/PA	MARVEL-M, ANC, VIPRE-01M

Representative transient analysis for Section 15.1:
- 15.1.5 Steam System Piping Failures

- **Large increase in steam flow will result in a very large decrease in core inlet temperature due to increased heat transfer in SG**

- **Hot standby cases:**
  - ✓ Largest break size is most limiting
  - ✓ Case where RCPs continue running (i.e. no LOOP) is most limiting
  - ✓ Criticality occurs and the reactor returns to power
  - ✓ SI actuates on low steam line pressure
  - ✓ Borated water injection restores subcriticality and reduces reactor power to hot standby
  - ✓ DNBR remains above safety analysis limit (fuel failure does not occur)
  - ✓ RCS and SG pressures decrease during this event

Representative transient analysis for Section 15.1:
- 15.1.5 Steam System Piping Failures

- **Hot full power case:**
  - ✓ Intermediate break size is most limiting (~0.4 ft^2)
  - ✓ Case where RCPs continue running (i.e. no LOOP) is most limiting
  - ✓ Reactor power increases from full power
  - ✓ Reactor trip occurs on overpower ΔT for intermediate break size (limiting case)
  - ✓ DNBR remains above safety analysis limit (fuel failure does not occur)
  - ✓ RCS and SG pressures decrease during this event

- **Steam mass and energy release is calculated for containment integrity analysis (DCD Chapter 6)**

- **There are no Open Items for DCD Section 15.1**

15.2 Decrease in Heat Removal by the Secondary System

Section No.	Event	Category	Computer Code(s)
15.2.1	Loss of external load	AOO	MARVEL-M
15.2.2	Turbine trip	AOO	MARVEL-M, VIPRE-01M
15.2.3	Loss of condenser vacuum	AOO	--
15.2.4	Closure of main steam isolation valves	AOO	--
15.2.5	Steam pressure regulator failure	N/A to US-APWR	
15.2.6	Loss of non-emergency AC power to station auxiliaries	AOO	MARVEL-M
15.2.7	Loss of normal feedwater flow	AOO	MARVEL-M
15.2.8	Feedwater system pipe break – Minor/Major	AOO/PA	MARVEL-M

Representative transient analysis for Section 15.2:
- 15.2.8 Feedwater System Pipe Break

- Loss of feedwater is assumed initially
- Double-ended feedwater line break occurs simultaneously with SG water level reaching the analytical limit for low SG water level reactor trip
- Pressurizer safety valves and MSSVs open to relieve pressure
- DNBR decreases but remains above safety analysis limit (fuel failure does not occur)
- RCS and SG pressures increase but remain below 110% of system design pressures
- Pressurizer water level remains below level of pressurizer safety valves (no water relief occurs)
- Subcooling margin is maintained (no boiling occurs in hot leg)
- Radiological consequences are bounded by steam line break (15.1.5) due to larger steam release

15.2 Decrease in Heat Removal by the Secondary System

- There are no Open Items for DCD Section 15.2

15.3 Decrease in Reactor Coolant System Flow Rate

Section No.	Event	Category	Computer Code(s)
15.3.1.1	Partial loss of forced reactor coolant flow	AOO	MARVEL-M, VIPRE-01M
15.3.1.2	Complete loss of forced reactor coolant flow	AOO	MARVEL-M, VIPRE-01M
15.3.2	Flow controller malfunction	N/A to US-APWR	
15.3.3	Reactor coolant pump rotor seizure	PA	MARVEL-M, VIPRE-01M
15.3.4	Reactor coolant pump shaft break	PA	--

US-APWR

Representative transient analysis for Section 15.3:
- 15.3.3 Reactor Coolant Pump Rotor Seizure

- **Seizure of one RCP rotor causes rapid decrease in RCS flow**
- **Rapid reduction in flow will result in an increase in RCS temperature and decrease in DNBR**
- **Reactor trip occurs on low reactor coolant flow**
- **DNBR decreases below safety analysis limit and all rods below limit are assumed to fail**
- **Number of rods in DNB is calculated to be <10%**
- **Peak cladding temperature case is analyzed and PCT is <2200F limit**
- **RCS pressure case is analyzed and RCS pressure is less than 110% of system design pressure**
- **SG pressure remains below 110% of system design pressure**
- **Dose analysis is performed using an assumption of 10% of rods have failed (bounds calculated value)**

- There are no Open Items for DCD Section 15.3

15.4 Reactivity and Power Distribution Anomalies

Section No.	Event	Category	Computer Code(s)
15.4.1	Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition	AOO	MARVEL-M, VIPRE-01M, TWINKLE-M
15.4.2	Uncontrolled control rod assembly withdrawal at power	AOO	MARVEL-M
15.4.3	Control rod misoperation	AOO/PA	MARVEL-M, VIPRE-01M
15.4.4	Startup of an inactive loop or recirculation loop at an incorrect temperature	AOO	N/A
15.4.5	Flow controller malfunction causing an increase in BWR recirculation loop	N/A to US-APWR	
15.4.6	Inadvertent decrease in boron concentration in the RCS	AOO	N/A
15.4.7	Inadvertent loading and operation of a fuel assembly in an improper position	PA	ANC
15.4.8	Spectrum of rod ejection accidents	PA	MARVEL-M, VIPRE-01M, TWINKLE-M, ANC

Representative transient analysis for Section 15.4:
- 15.4.2 Uncontrolled RCCA Withdrawal at Power

- **RCCA bank withdrawal causes reactivity insertion and increase in reactor power**

- **Different combinations of reactivity insertion rates (from 0 to 75 pcm/s), core conditions (BOC/EOC), and initial powers (100%, 75%, 10%) are analyzed**

- **Reactor trip that occurs depends on reactivity insertion rate and initial power but is one of the following:**
  - ✓ High power range neutron flux (high setpoint)
  - ✓ Over power ΔT
  - ✓ Over temperature ΔT
  - ✓ High pressurizer pressure

Representative transient analysis for Section 15.4:
- 15.4.2 Uncontrolled RCCA Withdrawal at Power

- **Transient results are provided for two cases:**
  - ✓ HFP, BOC, 75 pcm/s (reactor trips on high power range neutron flux – high setpoint)
  - ✓ HFP, BOC, 5 pcm/s (reactor trips on over temperature ΔT)

- **5 pcm/s reactivity insertion rate produces the more limiting DNB of the two cases**

- **For both cases:**
  - ✓ DNBR decreases until reactor trip occurs but remains above safety analysis limit (fuel failure does not occur)
  - ✓ RCS and SG pressures remain below 110% of system design pressures

Representative transient analysis for Section 15.4:
- 15.4.8 Spectrum of Rod Ejection Accidents

- **Ejection of control rod results in positive reactivity insertion and increased power peaking**
- **Power increase is mitigated by Doppler feedback**
- **Reactor trip that occurs depends on reactivity insertion rate and initial power but is one of the following:**
  - ✓ High power range neutron flux (high setpoint)
  - ✓ High power range neutron flux (low setpoint)
  - ✓ Over temperature ΔT
  - ✓ Low pressurizer pressure
- **Can also cause RCS pressure decrease due to breach in reactor coolant pressure boundary**
- **Different combinations of core conditions (BOC/EOC) and initial powers (HFP/HZP) are analyzed**

Representative transient analysis for Section 15.4:
- 15.4.8 Spectrum of Rod Ejection Accidents

- **HFP cases (BOC and EOC):**
  - ✓ Reactor trip occurs on high power range neutron flux (high setpoint) based on measured power considering single failure of neutron flux detector
  - ✓ Peak fuel centerline temperature remains below fuel melting temperature limit
  - ✓ Average fuel pellet enthalpy at hot spot remains below limits

- **HZP cases (BOC and EOC):**
  - ✓ Reactor trip occurs on high power range neutron flux (low setpoint)
  - ✓ Average fuel pellet enthalpy at hot spot remains below limits
  - ✓ Hot spot peak fuel enthalpy is within limits
  - ✓ No PCMI failed fuel rods

US-APWR

Representative transient analysis for Section 15.4:
- 15.4.8 Spectrum of Rod Ejection Accidents

- **Rods in DNB evaluation (HFP cases):**
  - ✓ Limiting case is when reactor trip does not occur on high power range neutron flux (high setpoint)
  - ✓ No DNB occurs during prompt power increase
  - ✓ Reactor trip occurs later on low pressurizer pressure or over temperature ΔT during RCS depressurization
  - ✓ DNBR decreases below safety analysis limit due to decrease in RCS pressure
  - ✓ Number of rods in DNB is calculated to be <10%

- **RCS pressure analysis:**
  - ✓ RCS pressure remains below 110% of system design pressure during early part of transient
  - ✓ RCS pressure later decreases due to breach in reactor coolant pressure boundary
  - ✓ SG pressure remains below 110% of system design pressure

Open Items

Open Item No.	RAI No.	Question No.	RAI Topic / NRC Concern	RAI Response / DCD Impact
15.4-4	888-6274	15.04.03-12	**One or more misaligned RCCAs event:** Provide additional details on the assumed RCCA misalignment configurations.	➢ MHI provided additional details in the response that was submitted to the NRC on January 31, 2012.
15.4-7	903-6325	15.04.04-15.04.05-1	**Startup of inactive loop event:** Provide additional details on why Modes 3 through 5 are not addressed.	➢ MHI provided additional details on why Modes 3 through 5 are not limiting in the response that was submitted to the NRC on March 16, 2012.
15.4-8	902-6318	15.04.06-10	**Boron dilution event:** Clarify whether the TS allow boron dilution during Modes 4 and 5 with no RCPs running and Mode 6.	➢ MHI provided additional details on the TS in the response that was submitted to the NRC on March 7, 2012.

15.5 Increase in Reactor Coolant Inventory

Section No.	Event	Category	Computer Code(s)
15.5.1	Inadvertent operation of ECCS that increases reactor coolant inventory	AOO	N/A
15.5.2	CVCS malfunction that increases reactor coolant inventory	AOO	MARVEL-M

15.5 Increase in Reactor Coolant Inventory

Representative transient analysis for Section 15.5:
- 15.5.2 CVCS Malfunction

- **Full open failure of charging flow control valve is assumed**
- **CVCS injects water at same boron concentration as RCS**
- **Pressurizer water level will increase**
- **Event is terminated by automatic CVCS isolation on high pressurizer water level prior to pressurizer overfill**

15.5 Increase in Reactor Coolant Inventory

➢ **There are no Open Items for DCD Section 15.5**

15.6 Decrease in Reactor Coolant Inventory

Section No.	Event	Category	Computer Code(s)
15.6.1	Inadvertent opening of a PWR pressurizer pressure relief valve	AOO	MARVEL-M
15.6.2	Radiological consequences of the failure of small lines carrying primary coolant outside containment	PA	RADTRAD
15.6.3	Radiological consequences of an SGTR	PA	MARVEL-M
15.6.4	Radiological consequences of main steam line failure outside containment (BWR)	N/A to US-APWR	
15.6.5	Loss-of-coolant accidents	PA	WCOBRA/TRAC, HOTSPOT M-RELAP5

Representative transient analysis for Section 15.6:
- 15.6.3 Radiological Consequences of an SGTR

- **Double-ended rupture of single SG tube**
- **Transient results are provided for two cases:**
  - ✓ Dose evaluation case
  - ✓ SG overfill case

- **Manual operator actions are assumed for reactor trip (dose evaluation case), ruptured SG isolation, RCS cooldown, RCS depressurization, SI termination**

Representative transient analysis for Section 15.6:
- 15.6.3 Radiological Consequences of an SGTR

- **Dose evaluation case**
  - ✓ Manual reactor trip at 15 minutes
  - ✓ Ruptured SG isolation at 20 minutes
  - ✓ RCS cooldown at 25 minutes
  - ✓ Remaining operator actions occur according to analysis conditions:
    - RCS depressurization ~ 45 minutes
    - SI termination ~ 48 minutes
  - ✓ Event is terminated when primary-to-secondary leakage is stopped at ~ 70 minutes
  - ✓ MSRV of ruptured SG is stuck open (at time of RCS cooldown) as additional failure
  - ✓ Primary-to-secondary leakage and SG release rates are used as input to dose calculation and onsite and offsite doses are less than SRP limits
  - ✓ DNBR decreases until reactor trip but remains above safety analysis limit (no fuel failure occurs)
  - ✓ RCS pressure decreases during the transient, so maximum pressure is initial pressure
  - ✓ SG pressures increase following reactor trip, but remain less than 110% of system design pressure

US-APWR

Representative transient analysis for Section 15.6:
- 15.6.5 Loss-of-Coolant Accidents (Large Break LOCA)

- **ASTRUM methodology with WCOBRA/TRAC(M1.0) and HOTSPOT is used for LBLOCA analysis**
- **Major assumptions:**
  - ✓ Pipe break is assumed to occur in cold leg
  - ✓ LOOP occurs coincident with break
  - ✓ ECCS actuation signal occurs on low pressurizer pressure
  - ✓ 2 of 4 SI pumps are available (single failure + OLM)
  - ✓ Minimum ECCS safeguards are assumed
  - ✓ Minimum containment pressure described in DCD Section 6.2.1.5 is applied

- **Values of PCT, LMO, and CWO (95th percentile with 95% confidence) in ASTRUM analysis (124 runs) satisfy the acceptance criteria of 10 CFR 50.46**

Representative transient analysis for Section 15.6:
- 15.6.5 Loss-of-Coolant Accidents (Small Break LOCA)

- **M-RELAP5 is used for SBLOCA analysis, based on Appendix-K to 10 CFR 50**
- **Major assumptions:**
  - ✓ Pipe break is assumed to occur in cold leg
  - ✓ Reactor trip is initiated on the low pressurizer pressure
  - ✓ LOOP occurs concurrent with the reactor trip
  - ✓ ECCS actuation signal occurs on low pressurizer pressure
  - ✓ 2 of 4 SI pumps are available (single failure + OLM)
  - ✓ Minimum ECCS safeguards are assumed
- **Values of PCT, LMO, and CWO in M-RELAP5 analysis (with sensitivity analysis) satisfy the acceptance criteria of 10 CFR 50.46**

US-APWR

Representative transient analysis for Section 15.6:
- 15.0.3 Radiological Consequences of Loss-of Coolant Accident

- Guillotine break in the primary system
- All fuel failure and TS limit for primary coolant activity
- Radioactivity in faulted fuel is assumed to be gradually released into containment (gap release phase and early in-vessel phase)
- Containment leakage, leakage collected in annulus, ESF leakage are assumed as release pathways
- Containment spray, natural deposition, and filtering system are credited as radioactivity removal methods

- Acceptance criteria for this event in SRP 15.0.3 is met

US-APWR

➢ Open Items

Open Item No.	RAI No.	Question No.	RAI Topic / NRC Concern	RAI Response / DCD Impact
15.6.5-1	352-2369	15.06.05-3	**LOCA event:** Conservatism of SI flow curves as related to sump strainer performance report (MUAP-08001).	➢ Sump Strainer Performance Report (MUAP-08001) is still under NRC staff review.
15.6.5-2 15.6.5-4	352-2369	15.06.05-15	**LOCA event:** Conservatism of accumulator flow rate bias as related to advanced accumulator report (MUAP-07001).	➢ Advanced Accumulator Topical Report (MUAP-07001) is still under NRC staff review.
15.6.5-3	352-2369	15.06.05-1	**LOCA event:** RCP seal integrity.	➢ RCP seal integrity is discussed in Chapter 8 and Chapter 9 of the DCD. (Closure of follow-up RAI related to RCP seal integrity in Chapter 9 is being handled separately.)

➢ Open Items

Open Item No.	RAI No.	Question No.	RAI Topic / NRC Concern	RAI Response / DCD Impact
15.6.5-8	719-5352	15.06.05-90	**LOCA event:** Post-LOCA long term cooling evaluation model. Applicability of core design code to recriticality analysis following the SBLOCA reflux condensation.	➢ Technical Report MUAP-07019 "Qualification of Nuclear Design Methodology using PARAGON/ANC" is being reviewed by the NRC.
15.6.5-9	861-6062	15.06.05-98	**LOCA event:** Post-LOCA long term cooling evaluation model. Validity of core recriticality analysis models and conditions.	➢ MHI agreed with NRC that MHI would issue a technical report on US-APWR GSI-185 evaluation, which includes: 1) US-APWR GSI-185 scenario 2) Vessel mixing test summary 3) Core recriticality analysis

Open Items

Open Item No.	RAI No.	Question No.	RAI Topic / NRC Concern	RAI Response / DCD Impact
15.6.5-11	861-6062	15.06.05-93	**LOCA event:** Post-LOCA long term cooling evaluation model. The timing of switchover to simultaneous vessel and hot-leg injection.	➢ MHI provided an explanation that the US-APWR ERGs ensure the manual switchover to hot leg injection occurs in a timely manner.
15.6.5-14	861-6062	15.06.05-100	**LOCA event:** Post-LOCA long term cooling evaluation model. Further explanation about the impact of sump debris on the evaluation, based on the revised technical report, MUAP-08013-P.	➢ MHI provided an explanation concerning the impact of sump debris on the long term cooling according to the technical report, MUAP-08013-P, and demonstrated its impact is very limited.

15.7 Radioactive Release from a Subsystem or Component

Section No.	Event	Category	Computer Code(s)
15.7.1	Gas Waste Management System Leak or Failure	Analyzed in DCD Chapter 11	
15.7.2	Liquid Waste Management System Leak or Failure (Atmospheric Release)	N/A to US-APWR	
15.7.3	Release of Radioactivity to the Environment Due to a Liquid Tank Failure	Analyzed in DCD Chapter 11	
15.7.4	Fuel Handling Accident	PA	RADTRAD
15.7.5	Spent Fuel Cask Drop Accident	N/A to US-APWR	

- **There are no Open Items for DCD Section 15.7**

15.8 Anticipated Transients without Scram

- **US-APWR design includes a Diverse Actuation System (DAS)**
- **DAS is analog system that is diverse from normal RTS and ESF**
- **DAS has automatic reactor trip (and turbine trip and main feedwater isolation) on:**
  - ✓ High pressurizer pressure
  - ✓ Low pressurizer pressure
  - ✓ Low steam generator water level
- **DAS also has automatic SI and automatic EFWS actuation features**
- **MHI has performed D3 coping analysis in technical report MUAP-07014 to evaluate AOOs with a concurrent CCF:**
  - ✓ DAS reactor trip is credited for some events
  - ✓ All acceptance criteria are met

➢ Open Items

Open Item No.	RAI Topic / NRC Concern
15.08-1	NRC staff is currently reviewing the D3 coping analyses in technical report MUAP-07014.

Appendix 15A Evaluation Models and Parameters for Analysis of Radiological Consequences of Accidents

US-APWR

- Description of RADTRAD code
- Description of model for radiological consequences analyses and input parameters
- Offsite dose assumptions
- MCR dose calculation model and assumptions

- There are no Open Items for DCD Appendix 15A

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Presentation to the ACRS Subcommittee

US-APWR Design Certification Application Review

Safety Evaluation with Open Items for Chapter 15, Transient and Accident Analyses

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Major Review Areas

- Review of plant characteristics and initial conditions to assure they are appropriately conservative, consistent with TS LCOs, and bounding for each of the acceptance criteria
- Review of worst single failure assumed in each event scenario to assure the worst cases are analyzed with respect to each of the acceptance criteria
- Review of Loss-of-Offsite-Power (LOOP) analysis
- Review of long term cooling (LTC)

Major Review Areas (continued)

- Review of sequence of events for each event scenario to evaluate the mitigation system, instrumentation responses and assumed operator actions to assure the transients are supported by the design of plant safety systems and future EOPs

- Review of the analyses including transient predictions to assure that the consequences of each analyzed limiting case meet the specific acceptance criteria specified for each event and all the regulatory requirements

Description of Open Items

- **Open Item 15.00-3 , Reload Methodology**

  Technical report MUAP-07026 describes the reload evaluation process, including identification of key safety parameters for each transient. The staff asked how recently proposed DCD changes would be captured in this document. The applicant's response, which includes a revised report is being evaluated by the staff.

- **Open Item 15.00-4 , Non-LOCA Methodology**

  This item will close when the final safety evaluation for Topical Report MUAP-07010-P is issued

- **Open Item 15.00-5 , Thermal Design Methodology Topical Report**

  This item will close when the final safety evaluation for Topical Report MUAP-07009-P is issued

Description of Open Items

- **Open Item 15.00-6 , Large Break LOCA Code Applicability Report for US- APWR**
  This item will close when the final safety evaluation for Topical Report MUAP-07011-P is issued.

- **Open Item 15.00-7 , Small Break LOCA Methodology for US-APWR**
  This item will close when the final safety evaluation for Topical Report MUAP-07013-P is issued.

- **Open Item 15.08-1, ATWS analysis**
  Technical Report MUAP-07014-P, Defense-In-Depth and Diversity Coping Analysis demonstrates that ATWS events maintain acceptable plant conditions. This item will be closed when the Chapter 7 SE is issued.

Description of Open Items

- **Open Item 15.4-4, Control Rod Misoperation**
  Staff is reviewing if most limiting misalignments are analyzed.

- **Open Item 15.4-7 , Startup of an Inactive Loop**
  The staff asked about startup of an inactive loop in lower Mode operation.

- **Open Item 15.4.8 , Boron Dilution in Modes 4 and 5**
  Wording in TS allows for planned dilution with no RCPs running.

Description of Open Items

- **Open Item 15.6.5-1, US-APWR Sump Strainer Performance Technical Report**
  Still under staff review.

- **Open Item 15.6.5-2 & 4, Advanced Accumulator Topical Report**
  Still under staff review. Scaling bias for accumulator flow rate.

- **Open Item 15.6.5-3 , TMI Action Item II.K.3.25, Effects of Loss of AC Power on Pump Seals**
  Tracking Open Item to ensure No. 2 RCP seal can perform its safety function.

- **Open Item 15.5.6-8, LTC, ANC, Chap 4.3**
  Tracking Open Item. Will be closed when the Chapter 4 SE is completed.

Description of Open Items

- **Open Item 15.6.5-9, LTC, SBLOCA recriticality, GSI-185**
  The staff questioned the amount of condensate generated in the SG U-tubes for a limiting small break and its impact on the core recriticality. The response to this RAI is under staff review.

- **Open Item 15.6.5-11, LTC, Boron Precipitation**
  The staff questioned the boric acid precipitation calculation and the timing of the switchover to hot leg injection. The response to this RAI is under staff review.

- **Open Item 15.6.5-14 , LTC, Boron Precipitation, with Debris**
  The staff questioned the impact of fuel blockage by debris in the reactor coolant on the US-APWR boric acid precipitation in the core. The response to this RAI is under staff review.

Backup Slides

Closed Items

- **Open Item 15.00-1** requested actuation analytical limits and time delays for all signals credited for accident mitigation in Chapter 15. **This information was provided in a proposed DCD revision; this item is now considered confirmatory.**

- **Open Item 15.00-2** requested a single-failure analysis of CVCS isolation. **This information was provided and the staff was satisfied with the response; this item is now considered closed.**

- **Open Item 15.03-1, Reactor Coolant Pump Rotor Seizure Analysis**
The staff asked how the reactor coolant pump rotor seizure analysis addressed local flow conditions in VIPRE-01M and also how the FΔHN census curve was generated. The response is being evaluated by the staff.

LUMINANT GENERATION COMPANY

Comanche Peak Nuclear Power Plant, Units 3 and 4

ACRS, US-APWR Subcommittee

FSAR Chapter 15 – Transient and Accident Analysis

July 10, 2012

Agenda

- Introduction
- Subsection Discussion
- Summary

Introduction

- R-COLA uses “Incorporated by Reference” methodology
- No departures from the US-APWR DCD for FSAR Chapter 15
- CPNPP COLA Revision 2 submitted in June 2011 and Revision 3 in June 2012
- No contentions pending before ASLB

Subsection Discussion

Sections 15.1, 15.2, 15.3, 15.4, 15.5, 15.6, 15.7, 15.8, and App 15A are incorporated by reference with no departures or supplements

15.0.3.3 Atmospheric Dispersion Factors

- **CPNPP COLA FSAR Summary**
  - **Site-specific X/Q values in FSAR 2.3.4 are bounded by values in DCD Tables 15.0-13 and 15A-18 through 15A-24.**

- **NRC SER Summary**
  - **No outstanding issues**

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Comanche Peak Nuclear Power Plant, Units 3 and 4
COL Application Review

Safety Evaluation

CHAPTER 15: Transient and Accident Analyses

July 9-10, 2012

Staff's Presentation Order

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- **Michael Takacs -** Project Manager

Technical Review Team

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