Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: ACRS USAPWR Subcommittee

Docket Number: n/a

Location: Rockville, Maryland

Date: July 9, 2012

Work Order No.: NRC-1738 Pages 1-262

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1 UNITED STATES OF AMERICA

2 NUCLEAR REGULATORY COMMISSION

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4 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

5 (ACRS)

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7 U.S. APWR SUBCOMMITTEE

8 OPEN SESSION

9 + + + + +

10 MONDAY

11 JULY 9, 2012

12 + + + + +

13 ROCKVILLE, MARYLAND

14 + + + + +

15 The Subcommittee met at the Nuclear

16 Regulatory Commission, Two White Flint North, Room

17 T2B1, 11545 Rockville Pike, at 8:30 a.m., John W.

18 Stetkar, Chairman, presiding.

19 SUBCOMMITTEE MEMBERS:

20 JOHN W. STETKAR, Chairman

21 SANJOY BANERJEE, Member

22 CHARLES H. BROWN, JR. Member

23 DANA A. POWERS, Member

24 JOY REMPE, Member

1 WILLIAM J. SHACK, Member

2

3 NRC STAFF PRESENT:

4 GIRIJA SHUKLA, Designated Federal Official

5 ANDREW BIELEN, RES

6 JOSEPH DONOGHUE, NRO

7 FRED FORSATY, NRO

8 MICHELLE HART, NRO

9 MICHELLE HAYES, NRO

10 STEVE MONARQUE, NRR

11 JEFFREY SCHMIDT, NRO

12 AMY SNYDER, NRO

13 MICHAEL TAKACS, NRO

14

15 ALSO PRESENT:

16 DAVID CARAHER, ISL

17 HIROSHI FUJISHIRO, MHI

18 YUKO FUJITA, MNES

19 HIROSHI HAMAMOTO, MHI

20 TERRY HEAMES, Alion

21 HIDEAKI IKEDA, MHI

22 MASAAKI KATAYAMA, MHI

23 KEVIN LYNN, MNES

24 YUTA MARUYAMA, MNES

1 REIKO NARUTOMI, MHI
2 JUNTO OGAWA, MHI
3 MARCOS ORTIZ, ISL
4 DAN PRELEWICZ, ISL
5 RYAN SPRENGEL, ISL
6 REBECCA STEINMAN, MNES
7 RURIKO TAKAMASHI, MNES
8 TETSUYA TERAMAE, MHI
9 DOUG WOOD, MHI
10 DON WOODLAN, Luminant Power
11 TSUYOSHI YOSHIDA, MHI

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

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

2 (8:36 a.m.)

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

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

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

1 Reports.

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

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

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

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

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

1 for closing the meeting before we enter into such
2 discussions, and to verify that only people with the
3 required clearance and need to know are present, and
4 I am going to be relying very heavily on MHI and
5 MNES to monitor that. If we delve into --
6 MS. STEINMAN: The material today should
7 only be publicly available.
8 CHAIR STETKAR: Okay, but if questions
9 come up during the discussion and we delve into
10 proprietary responses, we can close the meeting.
11 It's just a matter of --
12 MS. STEINMAN: Okay.
13 CHAIR STETKAR: controlling the
14 logistics of doing that.
15 MS. STEINMAN: Okay.
16 CHAIR STETKAR: So I'll ask you to keep
17 us on the straight and narrow there. Parts of this
18 -- sorry, I read that already. A transcript of this
19 meeting is being kept and will be made available as
20 stated in the Federal Register notice. Therefore,
21 we request that participants in this meeting use the
22 microphones located throughout the meeting room when
23 addressing the Subcommittee.
24 The participants should first identify

1 themselves and speak with sufficient clarity and
2 volume so they may be readily heard. I'd ask you
3 all to please silence your cell phones and let me
4 give you a little bit of information about the
5 schedule.
6 We are scheduled for a day and a half.
7 We need to finish tomorrow at noon time, at 12
8 sharp, because we do have other conflicting meetings
9 tomorrow. So we can't run long tomorrow.
10 We, depending on people's stamina, have
11 a little bit of flexibility to run a little late
12 this afternoon, so we'll just, we'll see how the
13 presentations go.
14 So, if it looks like we are running
15 behind today, too badly, we are going to try and
16 extent it a little bit the end of this afternoon
17 because we do need to finish promptly at noon
18 tomorrow.
19 And with that, I will turn the meeting
20 over to NRC management.
21 MS. SNYDER: Good morning Chairman
22 Stetkar and ACRS Subcommittee. It is a pleasure to
23 be here today. My name is Amy Snyder. I am acting
24 branch chief for Licensing Branch 2, which is

1 responsible for the project management of the US
2 APWR.
3 Today, as you said, we are going to be
4 focusing on Chapter 15, transient and accident
5 analysis. This is a major milestone for the staff.
6 We have been working on this since mid-2007.
7 The project manager for the design
8 certification is Jeff Ciocco. He is not here today.
9 The project manager for Chapter 15 is Michael Takacs
10 we have technical staff from the Reactor Systems
11 Branch, Joe Donoghue's branch, and also we will have
12 presentations from the -- our contract support
13 Information Systems Laboratories.
14 CHAIR STETKAR: Nothing more? Thank
15 you, and with that I will turn the meeting over to
16 MHI.
17 MR. IKEDA: Yes, thank you Chair. Well,
18 good morning everybody. My name is Hideaki Ikeda.
19 I am responsible for the LOCA code development and
20 safety analysis, safety design, MHI, Mitsubishi
21 Heavy Industries.
22 This presentation provides a brief
23 overview of MHI's small-break LOCA methodology for
24 US APWR. Next slide please.

1 Masaaki Katayama, besides me, is
2 co-presenter and he is deputy manager of LOCA
3 section and also our lead engineer of US APWR small-
4 break LOCA in our team. Next slide please.
5 The objective of the topical report is
6 to present the MHI's comprehensive methodology for
7 US APWR small-break LOCA analyses.
8 The key topics of the topical report
9 are: first is identification of the US APWR design
10 feature; next, development of the phenomena
11 identification ranking table and assessment matrix;
12 third, development of M-RELAP5, a modified version
13 of what we called RELAP5-3D; and finally the
14 assessment for the evaluation model adequacy of
15 M-RELAP5 and applicability to the US APWR's small-
16 break LOCA will be discussed in the topical report.
17 Now, M-RELAP5 is being applied to US
18 APWR small-break LOCA safety analysis and the
19 present topical report is referred to in Chapter
20 15.6.5 of the US APWR Design Control Document, DCD.
21 Next slide. First of all, the future of
22 US APWR. This slide summarizes the primary
23 specification of US APWR related to LOCA. The US
24 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.

1 MR. SPRENGEL: This is Ryan Sprengel
2 with MNES. The quick answer is no, they are
3 smaller. They are significantly smaller than SONGS.
4 MEMBER REMPE: Okay, thank you.
5 MR. SPRENGEL: His reference point was
6 the conventional Japanese plant.
7 MEMBER REMPE: Okay.
8 MR. IKEDA: Yes.
9 MEMBER REMPE: Thank you.
10 MR. IKEDA: Thank you very much.
11 MEMBER BROWN: Is the design the same,
12 the issue that came up in terms of the manufacturing
13 and the other issues that have popped up or is that
14 going to be discussed or what?
15 MR. SPRENGEL: I don't think we are
16 going to get into that, because it's a little cart
17 before the horse. We are waiting on the evaluations
18 to be completed at SONGS before we proceed with
19 anything --
20 MEMBER BROWN: Okay, so this will be a
21 more generic evaluation --
22 MR. SPRENGEL: That's correct. We have
23 already actually received them --
24 MEMBER BROWN: relative to the actual

1 components themselves?

2 MR. SPRENGEL: Right, we have received

3 an RAI requesting us to evaluate and incorporate any

4 lessons learned, so that is work that will be done

5 and will be presented.

6 MEMBER BANERJEE: So far as the tube

7 side flow area is concerned, the megawatt of power,

8 is that smaller or larger than the conventional

9 Japanese systems?

10 MR. IKEDA: And, sir, you are talking

11 about heat transfer area?

12 MEMBER BANERJEE: No, flow area.

13 MR. IKEDA: Flow area is larger than the

14 conventional --

15 MEMBER BANERJEE: Per megawatt.

16 MR. IKEDA: Maybe as a flow area

17 multiplication factor --

18 MEMBER BANERJEE: Can you give us that

19 number for both the hot leg and the steam generator?

20 MR. IKEDA: Yes. Okay, I have a backup

21 presentation slide for the tomorrow discussion. I

22 will explain tomorrow. So --

23 MEMBER BANERJEE: But one of the issues

24 we will explore with the small break will be

1 refluxing, so we need to know the flow areas.

2 MR. IKEDA: Yes, so I remember the ratio

3 of thermal output compared with the conventional

4 four-loop. Typically the four-loop PWR allowed 1.3.

5 On the other hand, the flow area of the steam

6 generator is allowed 1.4.

7 MEMBER BANERJEE: So, just give me the

8 numbers.

9 MR. IKEDA: Yes.

10 MEMBER BANERJEE: Okay. Both for the

11 steam generator and the hot leg.

12 MR. IKEDA: Steam generator.

13 MEMBER BANERJEE: And the hot leg. That

14 is required.

15 MS. STEINMAN: Ask him tomorrow to bring

16 both those --

17 MR. IKEDA: Yes, I remember the exact

18 number for the hot leg so I will give you

19 information tomorrow.

20 MEMBER BANERJEE: Right.

21 MR. IKEDA: But anyway, the flow area of

22 the hot leg and also for the steam generator flow

23 area is rather comparably --

24 MEMBER BANERJEE: In absolute terms, I'm

1 sure it is.

2 MR. IKEDA: Yes. Yes.

3 MEMBER BANERJEE: It is from the power

4 generated during the, you know, operation and the

5 small-break LOCA, that's very interesting.

6 MR. IKEDA: Can I move on?

7 MEMBER BANERJEE: Yes.

8 MR. IKEDA: Okay, thank you. Next slide

9 please. Okay. This slide --

10 CHAIR STETKAR: Folks up front, just be

11 a little bit careful with your paper on the

12 microphones.

13 MR. IKEDA: Oh, okay.

14 CHAIR STETKAR: They are really

15 sensitive and they give our reporter a lot of

16 problems, so be careful not to hit them. Thank you.

17 MR. IKEDA: Okay. So, this slide

18 explains the US APWR, the small-break LOCA scenario.

19 The transient evolution is similar to that of the

20 Westinghouse type four-loop PWR.

21 The accident scenario is divided into

22 five phases -- blowdown, natural circulation, loop

23 seal formation and this clearing, boiloff and core

24 recovery phases.

1 The figures show the RCS, reactor
2 coolant system, pressure response and the core
3 liquid level response activity.
4 MEMBER BANERJEE: Will you go into more
5 detail in the future on how you do your loop seal
6 clearing calculations?
7 MR. IKEDA: Yes.
8 MEMBER BANERJEE: So you will, okay. So
9 let's defer that for the moment.
10 MR. IKEDA: Okay.
11 MEMBER BANERJEE: And this is based on
12 how many? Is this just one loop seal clearing?
13 MR. IKEDA: Exactly all the loop allowed
14 to -- allowed clearing if possible. No penalty is
15 given for M-RELAP5 calculation in terms of loop seal
16 clearing. And then --
17 MEMBER BANERJEE: But the most
18 conservative would be just to clear one loop seal,
19 right?
20 MR. IKEDA: Yes. That -- for the US
21 APWR small-break LOCA calculation, we identified all
22 the loop clear in our representative small-break
23 LOCA case.
24 MEMBER BANERJEE: Why did you do that?

1 Will you come back and talk about the -- this is of
2 course, as you know, very important, for small-break
3 LOCA. So we want to go into some detail as to how
4 you do the loop seal clearing, when you do it, why
5 you do it, all the details. So we can do it later -
6 -
7 MR. IKEDA: Okay.
8 MEMBER BANERJEE: when you show it.
9 MR. IKEDA: Unfortunately the
10 presentation material today does not include such
11 information, so I will provide a discussion, Sanjoy,
12 I think, tomorrow.
13 MEMBER BANERJEE: Okay.
14 MR. IKEDA: Next slide please. So the -
15 - our small-break LOCA methodology is based on the
16 M-RELAP5 code as I mentioned, and M-RELAP5 is a
17 conservative code, conforming to regulatory
18 requirements specified by Appendix K to 10 CFR Part
19 50.
20 Because US APWR employs ZIRLO cladding
21 materials, figures of merit are identical to those
22 specified in 10 CFR Part 50.46, so, PCT, ECL,
23 hydrogen generation, coolable geometry, and
24 long-term cooling criteria.

1 Next slide please. The basis of our
2 SBLOCA methodology, M-RELAP5 code has been developed
3 based on RELAP5-3D code as I mentioned. The
4 complete two-fluid thermal-hydraulics model is
5 applicable to important processes and phenomena
6 occurring under US APWR small-break LOCAs.
7 MHI's, our modification to M-RELAP5 is
8 limited to implementation of conservative models and
9 our plant-specific models such as the advanced
10 accumulator models.
11 M-RELAP5 code has been developed and
12 assessed in accordance to evaluation model
13 development and assessment procedure in regulatory
14 guide 1.203.
15 MEMBER BANERJEE: So with regard to the
16 advanced accumulator model, which we will hear about
17 later, this fall, could you tell us in more detail
18 at some point what that model that you put in, is?
19 Is it just a flow versus pressure or what that model
20 is, would be useful to know.
21 MR. IKEDA: So you need --
22 MEMBER BANERJEE: You're telling us just
23 assume the model for now and we'll discuss this
24 later?

1 MR. IKEDA: And do you need a transient
2 response under the small-break LOCA? So you said --
3 MEMBER BANERJEE: Well, at the moment I
4 am just interested in knowing what is the format of
5 the model you have put in, you know. We can go into
6 the model itself when the time comes, but right now,
7 just informational.
8 MR. IKEDA: Okay.
9 MEMBER BANERJEE: Okay.
10 MR. IKEDA: I see. Next slide.
11 MEMBER BANERJEE: Also, the other
12 specific area I am interested in is whether you have
13 put a model for the elbow region of the hot leg for
14 flooding.
15 MR. IKEDA: CCFL.
16 MEMBER BANERJEE: CCFL. And what that
17 model is.
18 MR. IKEDA: Now, for the hot leg CCFL we
19 employed the type of conductive rod theory.
20 MEMBER BANERJEE: Whatever it is, if you
21 just -- right now, this is, again, just
22 informational.
23 MR. IKEDA: Okay. Okay.
24 MEMBER BANERJEE: What it was that you

1 used. Thank you. Okay.

2 MR. IKEDA: Can I move on?

3 MEMBER BANERJEE: Yes.

4 MR. IKEDA: Thanks. So, this slide

5 summarizes the M-RELAP5 Appendix K models. In

6 conformance to Appendix K, to 10 CFR 50, the

7 following models are implemented on M-RELAP5 --

8 ANS-1971 decay heat models for the conservative use,

9 they multiplied the heat -- I'm sorry -- the decay

10 heat model is multiplied by 1.2; the fuel gap

11 conductance model equivalent to that used in the

12 fuel design code. Our code is FINE for US APWR.

13 And next, the Baker-Just metal-water

14 reaction models and the cladding swelling and

15 rupture models applicable to ZIRLO cladding

16 material, and the Moody, two-phase critical flow

17 models and finally, no return logic and no return to

18 transition or nuclear boiling during the blowdown

19 phase.

20 MEMBER REMPE: Excuse me again. Could

21 you elaborate a little bit about what you mean by

22 the fuel design code, and what data supports that

23 fuel design code and where it is with respect to NRC

24 approval?

1 MR. IKEDA: Yes. That code, the fuel
2 design code is under review now. Is that correct?
3 The FINE code -- I'm sorry. The FINE code is under
4 the review now?
5 MR. SPRENGEL: The FINE code topical
6 report has an SER base to be completed by the staff
7 but it has not come on to the ACRS yet.
8 CHAIR STETKAR: We have a September
9 subcommittee meeting that will be addressing the
10 fuel models.
11 MEMBER REMPE: Okay, and there are data
12 that you obtained from Haldon to support that code?
13 MR. IKEDA: Yes.
14 MEMBER BROWN: Okay. We will look
15 forward to it.
16 MEMBER REMPE: Okay.
17 CHAIR STETKAR: Put it on your calendar.
18 September 20th.
19 MEMBER REMPE: I will.
20 MEMBER BANERJEE: I'm sure you'll have
21 more questions when the LBLOCA comes up.
22 MR. IKEDA: Okay, so the last topic is
23 in order to simulate the advanced accumulator flow
24 rate, we have developed the advanced accumulator

1 model based on the RELAP5-3D accumulator models.
2 Next slide please.
3 So hereafter, I'd like to give a summary
4 of US APWR small-break LOCA analysis for the Safety
5 Evaluation.
6 M-RELAP5 can treat the LOCA with the
7 break sizes up to one square foot.
8 MEMBER BROWN: Can -- not being a
9 thermal hydraulics guy, they said they've scaled --
10 they've got their scaling bias set up from the
11 halfscale data using the CFD calculations. Is that
12 a common --
13 MEMBER BANERJEE: They will go through
14 that, I guess, later.
15 CHAIR STETKAR: Yes, we've got an
16 October subcommittee.
17 MEMBER BROWN: Yes, okay.
18 CHAIR STETKAR: We are going to hear
19 about the detailed models for the accumulator.
20 MEMBER BANERJEE: No, it's not obvious.
21 MEMBER BROWN: Okay. I just --
22 CHAIR STETKAR: At least that's what's
23 scheduled right now.
24 MEMBER BROWN: I just wanted to know how

1 it was validated. That's --

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

3 MR. SHUKLA: It's a good question.

4 CHAIR STETKAR: It's an excellent

5 question.

6 MEMBER BANERJEE: How do you know

7 everything?

8 MEMBER BROWN: Thank you. Okay, my

9 questions end. You can go on. Thank you.

10 MR. IKEDA: Thank you. So the technical

11 report, MUAP-07025, small break LOCA sensitivity

12 analysis for US APWRs, which is referred to in the

13 US APWR DCD, provides the following sensitivity

14 calibration, like break location, break size, break

15 orientation, plant noding, time step size, offsite

16 power availability, single failure assumption and

17 accumulator flow rate, to check the sensitivity of

18 the flow rate bias for the advanced accumulator.

19 By performing this sensitivity

20 calculation, it is identified that the loop seal PCT

21 occurs during the 7.5-inch cold leg bottom break and

22 that the boiloff PCT occurs in the one square foot,

23 cold leg bottom break.

24 In addition, the condition of the loss

1 of offsite power and two ECCS trains available due
2 to -- the two ECCS is not available, provides the
3 limiting PCT in the safety analysis.
4 CHAIR STETKAR: The sensitivity
5 calculations that you did for the small-break LOCA,
6 I know that you did, I think, one calculation from
7 the pressurizer steam space, and it, I believe it
8 said that the size of that break was about 6.7
9 inches, is what it said in the report, about.
10 How large is the depressurization valve
11 line? Not the safety depressurization valves, but
12 the -- the severe accident depressurization valve
13 line. How large is that pipe?
14 MR. IKEDA: How large --
15 MR. HAMAMOTO: This is Hiroshi Hamamoto,
16 MHI. The size is six-inch.
17 CHAIR STETKAR: It is six-inch?
18 MR. HAMAMOTO: Yes.
19 CHAIR STETKAR: Thank you. So that
20 bounds -- the 6.7 would bound a break in that line.
21 MR. HAMAMOTO: Thank you.
22 MR. IKEDA: This slides provides a
23 representative small-break LOCA result for US APWR
24 one square foot cold leg break, the limiting PCT

1 case. Following the break occurrence, the reactor
2 coolant system pressure rapidly decreases and the
3 natural circulation and loop shield phases are not
4 found clearly.
5 Due to the rapid loss of RCS coolant,
6 the cladding heatup starts around 100 seconds.
7 That's almost the same time the advanced accumulator
8 and high-head injection system start injecting the
9 safety coolant and the PCT is suppressed below 1328
10 degree air.
11 The RCS inventory recovers by safety
12 coolant injection and the core is quenched at
13 approximately 400 seconds. It is noted that the
14 figure for the ECCS flow indicates the total flow
15 rate for the accumulator and HHIS high-head
16 injection system respectively with the unit of pound
17 per second.
18 Okay. I would like to summarize my
19 presentation today. The US APWR small-break LOCA
20 important phenomena have been identified and M-
21 RELAP5 modeling capability has been assessed in
22 conformance to EMDAP.
23 M-RELAP5 is capable of predicting all
24 the high-ranked phenomena of small-break LOCAs, and

1 detailed information will be provided in tomorrow's
2 presentation.
3 And now, M-RELAP5 is currently being
4 applied to small-break LOCA safety analysis for US
5 APWR reported in the design control document,
6 Chapter 15. Thanks for your attention.
7 MEMBER BANERJEE: Have we got that
8 previous slide you showed? This doesn't seem to be
9 -- yes, it doesn't seem to be in my package. Is
10 that just --
11 MEMBER REMPE: They were sent to us
12 electronically.
13 MEMBER SHACK: It's in the electronic
14 version.
15 MEMBER BANERJEE: Ah, it's in the
16 electronic version. So, I don't have it here. Is
17 that -- is that because it's proprietary or what?
18 MS. STEINMAN: No, that would have been
19 a mistake if it's not in there. The intent was that
20 it was to be included.
21 MEMBER BANERJEE: Oh, okay.
22 MEMBER BROWN: There's three or four
23 slides that aren't included in the package here that
24 they went through. So --

1 MEMBER BANERJEE: Make sure that we have
2 a full package.
3 MR. SHUKLA: I'll do that.
4 MEMBER BANERJEE: Because otherwise we
5 get mixed up. So the PCT here is referring -- can
6 you walk us through this just once more?
7 MS. STEINMAN: Each slide, just
8 describe.
9 MR. IKEDA: Okay. These figures show
10 the reactor coolant system pressure and here, the
11 occurrence of the break, and the break size is
12 large, comparably large, so as RCS system pressure
13 rapidly decrease.
14 And in the small-break case, I mean the
15 comparably small-break cases, here the plant
16 transits to the natural circulation. But in this
17 case, in this limiting, one square foot break case,
18 no clear natural circulation exists.
19 CHAIR STETKAR: This is a big small
20 LOCA.
21 MR. IKEDA: Yes.
22 MEMBER BANERJEE: A foot square.
23 CHAIR STETKAR: Yes.
24 MEMBER BANERJEE: I mean it was a big --

1 MR. IKEDA: Yes, and no clear loop seal
2 formation in the clearing. And here is pressure
3 level probably equivalent to the secondary pressure
4 level, and -- but soon the loop seal clear and the
5 RCS pressure decrease again soon.
6 MEMBER BANERJEE: So, how do you clear
7 the loop seal in your -- is it by differential
8 pressure or -- is there a specific module in your
9 code which clears the loop seal or --
10 MR. IKEDA: No specific module.
11 MEMBER BANERJEE: It just clears it by
12 pressure difference or --
13 MR. IKEDA: Yes, and the capability of -
14 - predictability of the code was assessed and
15 validated using the UPTF number 5 test data, and
16 also, using IET, integral effect test obtained in
17 the ROSA test facility.
18 Using the IET test data, we verified our
19 code capability to predict the loop shield clearing.
20 Unfortunately, for this case, no clear loop shield -
21 - I'm sorry, no clear loop shield clearing here.
22 But tomorrow -- tomorrow I will give the example,
23 the transient including the loop shield formation
24 and the clearing, exactly a 7.5-inch break. 7-5-

1 inch break cases provides a limiting loop shield PCT
2 -- I'm sorry, loop shield clearing PCT.
3 MEMBER BANERJEE: So, you allowed the
4 loop seals to clear by just the pressure difference
5 across them?
6 MR. IKEDA: Yes.
7 MEMBER BANERJEE: So, whichever ones
8 will clear, they will clear. It -- you don't, you
9 do not do a conservative calculation which says that
10 once a loop seal has cleared, only one will clear?
11 MR. IKEDA: Exactly --
12 MEMBER BANERJEE: And all of them clear.
13 MR. IKEDA: Exactly no conservative
14 exemption is made for M-RELAP5.
15 MEMBER BANERJEE: And where do you go
16 into refluxing? Do you have a refluxing situation?
17 MR. IKEDA: For the DCD calculation, the
18 reflux condensation mode is not included, because in
19 the US APWR design, the reflux condensation occur
20 when the operator manually decreases the secondary
21 pressure level.
22 MEMBER BANERJEE: So if you don't reduce
23 the secondary pressure, you don't get into reflux
24 condensation?

1 MR. IKEDA: Significant reflux
2 condensation does not occur -- not occur if the
3 manual secondary pressure reduction is activated.
4 MEMBER BANERJEE: If it is not.
5 MR. IKEDA: Not, yes, not.
6 MEMBER BANERJEE: So is that because the
7 secondary system stays at high pressure and may even
8 be a heat source, or --
9 MR. IKEDA: Yes, but --
10 MEMBER BANERJEE: A heat sink ever?
11 MR. IKEDA: Yes, that's right, about
12 there -- better for the --
13 MR. KATAYAMA: I want to say the
14 capacity of HHIS for US APWR is very large, and HHIS
15 can suppress the PCT -- can suppress the PCT enough,
16 and long-term cooling can be accomplished by HHIS.
17 And if operator decided to start
18 shutdown mode, then the secondary pressure level is
19 to be decreased by manually. During the manual
20 operation, the PCT does not reheat up, and even if
21 automatically the secondary pressure level is
22 decreased, I think heat-up does not occur in such a
23 case --
24 MEMBER BANERJEE: So, it would be

1 helpful for us to see, for some of these limiting
2 breaks, what the pressure that you calculate on the
3 secondary side and the primary side is, to
4 understand, you know, the phases of the accident,
5 because if the secondary side pressure becomes lower
6 than the primary side, then it will be a heat sink
7 and obviously the steam generator will act as a
8 condenser, you know, and I'm not sure that I can --
9 CHAIR STETKAR: But I think what he is
10 saying, for this size break, that's not going to
11 happen.
12 MEMBER BANERJEE: Yes, but I'm saying
13 that as you look at the spectrum of breaks, I don't
14 know what the phenomenon looks like. I don't have a
15 good feel for this plant so -- because you have no
16 low pressure injection --
17 CHAIR STETKAR: I think you are asking
18 probably down in the three or four inch size, right?
19 MEMBER BANERJEE: What other size do you
20 have? By the way, the staff will come up, but I
21 wanted to ask you, did the staff do any confirmatory
22 calculations on this?
23 MR. TAKACS: Yes, we did do confirmatory
24 calculations.

1 MEMBER BANERJEE: Over a range or break
2 sizes?
3 MR. TAKACS: That is correct.
4 MEMBER BANERJEE: And will you be
5 presenting some of those?
6 MR. TAKACS: Yes, we will be.
7 MEMBER BANERJEE: Okay. So we will
8 catch up with you on that. Okay.
9 MR. TAKACS: Okay.
10 MR. IKEDA: Do you need additional
11 explanation for the other figure?
12 MEMBER BANERJEE: Why don't you go
13 through it, yes. So --
14 MR. IKEDA: Okay. Okay, this is RCS
15 pressure and ECCS flow rate figure gives advanced
16 accumulator flow rate. In total three advanced
17 accumulator here, so not flow rate, not the flow
18 rate are one ECC advanced accumulator, three
19 accumulator flow rate, and high-head injection flow
20 rate -- and okay.
21 When the RCS pressure decrease below
22 approximately 600 pound, the advanced accumulator is
23 activated and start injection. This is large flow
24 rate model.

1 And the flow rate is comparably lower
2 compared with the case for large-break LOCA cases.
3 But anyway, here the advanced accumulator flow rate
4 model is changed -- to the small flow rate injection
5 model.
6 And here the temporal degrees of RCS
7 pressure appear and check valve is activated and
8 advanced accumulator is tentatively stopped and
9 restart again.
10 And at almost the same time, high-head
11 injection system starts injecting the safety coolant
12 like this.
13 MEMBER BANERJEE: So advanced -- the
14 accumulator stops due to the check valve. That's by
15 a pressure difference.
16 MR. IKEDA: Yes.
17 MEMBER BANERJEE: And then, due to the
18 pressure difference changing at about, what is it,
19 250 or something, it starts to open the check valve
20 automatically?
21 MR. IKEDA: Yes.
22 MEMBER BANERJEE: And it goes on
23 injecting?
24 MR. IKEDA: Yes. In case of the

1 advanced case of the cold leg break. The
2 accumulator flow rate motion is dependent on the
3 break size and the break location.
4 And here is the --
5 MEMBER BANERJEE: So, what is the
6 purpose of that check valve?
7 MR. IKEDA: To protect the accumulator.
8 MEMBER BANERJEE: From inflow?
9 MR. IKEDA: Yes, reverse flow. Reverse
10 flow means from cold leg to the advanced accumulator
11 tank.
12 MEMBER BANERJEE: Okay, let's go to the
13 next slide -- the next panel.
14 MR. IKEDA: Okay, this is the liquid
15 level for the core region and the upper plenum
16 region. As you know, the large scale for the break
17 is comparably larger so the RCS inventory rapidly
18 decrease and also cold leg level is depressed
19 rapidly.
20 And here, the time, ECC injection and
21 advanced accumulator injection starts, and the
22 liquid level certainly recovers like this.
23 And this figure indicates the RCS mass
24 inventory, and similar to the core and upper plenum

1 liquid level, the -- by activating the advanced
2 accumulator and high-head injection, the RCS
3 inventory will recover like this.
4 This is the heat -- figure for the PCT
5 and the cladding heatup. Here at approximately 100
6 seconds following the break, the cladding heatup
7 starts like this, and also, the advanced accumulator
8 injection starts. That's -- heatup keeps, because a
9 large amount of core coolant is lost due to the
10 large break size.
11 And that, the PCT occur along 160
12 seconds and it starts decreasing like this.
13 MEMBER BANERJEE: So if you go to the
14 third plot, 160 seconds coincides to when the -- you
15 start to recover the core -- the liquid level, is
16 that it?
17 MR. IKEDA: Yes.
18 MEMBER BANERJEE: It's about there?
19 MR. IKEDA: Yes. It mostly coincides.
20 MEMBER BANERJEE: What is happening
21 physically to change the slope that much there? I'm
22 just trying to understand. It was -- is there some
23 water coming from somewhere, the accumulator filled
24 something up or what happened?

1 MR. IKEDA: I see. I estimate, I'm
2 sorry, maybe the downcomer filled with the advanced
3 accumulator and HHRS safety coolant. So --
4 MEMBER BANERJEE: It would be useful, at
5 least, you know, for me, to try to understand --
6 MR. IKEDA: Okay.
7 MEMBER BANERJEE: how well the core
8 does. I'm always skeptical about cores.
9 MR. IKEDA: Okay.
10 MEMBER BANERJEE: Because it has to be
11 associated with some physical phenomena. So if you
12 are filling the downcomer it makes sense, okay? So
13 could you check whether that is due to the downcomer
14 filling or not?
15 MR. IKEDA: Okay. The --
16 MEMBER BANERJEE: So actually the
17 maximum in PCT depends on the -- how well you
18 predict this filling, right? Okay.
19 MR. IKEDA: Anyway, the information
20 about the other parameters, I think, are -- all the
21 parameters you need are included in the DCD. But I
22 will try to provide tomorrow.
23 MEMBER BANERJEE: Yes, it's hard for us
24 to decipher, you know, all this stuff, so if you can

1 just give us -- you know, it's much easier if you
2 can lead us through what's happening in the critical
3 case, let's say, the worst case, and associate it
4 with some physical phenomena, like this is filling,
5 that's why the temperature turns around, or whatever
6 is happening.
7 MR. IKEDA: Yes, I see.
8 MEMBER BANERJEE: And then you can see,
9 you know, how well do you predict that filling, and
10 what can be there. Okay. You have quite a bit of
11 margin here, so --
12 MR. IKEDA: Yes.
13 MEMBER BANERJEE: not very sensitive.
14 MR. IKEDA: That's all. Any other
15 comments?
16 MEMBER BANERJEE: And what was the
17 maximum oxidation that you saw?
18 MR. IKEDA: Very low. Less than 0.2
19 percent, less than --
20 MEMBER BANERJEE: Even for the most
21 critical --
22 MR. IKEDA: Yes.
23 MEMBER BANERJEE: Okay. So tomorrow we
24 are getting loaded up with a lot of stuff.

1 CHAIR STETKAR: We may have to start at
2 3 o'clock in the morning. We'll see how we do.
3 We'll see how -- we have to finish at 12 tomorrow.
4 That's a firm time. We'll see how we get through
5 this. This is our first introduction to this topic
6 and it may not be our last.
7 Any other members' questions for MHI?
8 If not, thank you, appreciate it, and we'll ask the
9 staff to come up.
10 MEMBER BANERJEE: John, just a question
11 for you. Are we expected to write an interim letter
12 based on this, or what?
13 CHAIR STETKAR: We'll talk about that.
14 We are writing interim letters. The next one will
15 probably be in October. What we have been doing is
16 bundling together chapters. Unless we determine
17 that there is a fairly substantial issue and we can
18 kind of move up the schedule and priority for those
19 interim letters.
20 The simple answer to your question is
21 yes we will.
22 MEMBER BANERJEE: Okay, yes, but in
23 October?
24 CHAIR STETKAR: In October. And we have

1 some flexibility about those, especially because we
2 won't be hearing about the transient topical report
3 and the advanced accumulator until the October
4 subcommittee meeting. We may not write about the
5 LOCA indices in that October interim letter.
6 MEMBER BANERJEE: But you'll integrate
7 everything --
8 CHAIR STETKAR: Yes. I mean we'll talk
9 a little bit about the schedule later. Mike, it's
10 all yours.
11 MR. TAKACS: Good morning Chairman
12 Stetkar and Subcommittee members. My name is Mike
13 Takacs. I am the chapter PM for transient and
14 accident analysis. I have been involved in this
15 chapter since May of 2008. The last time we had a
16 similar meeting where we gave an informational
17 briefing was back in February, 2009, in Pittsburgh,
18 and that included all the topical report status.
19 Today, the staff is going to provide an
20 overview of the safety evaluation for small-break
21 LOCA methodology.
22 MEMBER BANERJEE: I remember we visited
23 that facility, right?
24 MR. TAKACS: Yes we did. Several of us

1 were there with you.

2 MEMBER BANERJEE: Okay.

3 MR. TAKACS: With me today, some of the

4 NRO staff, the technical staff here, presenters up

5 front. Jeff Schmidt, Andrew Bielen will introduce

6 themselves when they get there.

7 Jeff Ciocco, as you know, is the lead

8 PM. Supporting this whole review throughout the

9 time, the four-year period, Information System

10 Laboratories is here as well, in the audience.

11 Their names are up on the screen.

12 And with that, I am going to turn over

13 the presentation of the technical substance to Jeff

14 Schmidt.

15 MR. SCHMIDT: Hi, I'm Jeff Schmidt, and

16 I was the lead reviewer for the small-break LOCA and

17 I'm also the reactor systems lead for the APWR in

18 general. Today we are going to talk about the

19 small-break LOCA and another person who was key in

20 the review of this is Andy Bielen.

21 A little bit about myself real quick. I

22 was -- I'm a nuclear engineer from Iowa State

23 University, undergrad, and graduate at University of

24 Illinois, spent about three and a half years at

1 Westinghouse, the core design group, moved on to
2 Salem nuclear power plant when they were taking on
3 their own reload methodology, spent about seven
4 years there, then moved on to Palo Verde where I
5 spent the last 11 years, both as a senior engineer
6 there in the reactor -- nuclear analysis and safety
7 analysis group, and then the last six years there as
8 the supervisor of that group. So that's a real
9 quick history on myself. Andy.
10 MR. BIELEN: Hi, I'm Andy Bielen, like
11 Jeff said. I have Bachelors degrees in mathematics
12 and nuclear engineering and a Masters degree in
13 nuclear engineering from Penn State.
14 I came to the NRC in 2008 and spent
15 three years in reactor systems and currently I'm at
16 the University of Michigan under Tom Downar and
17 Annelisa Manera doing my PhD under Research
18 sponsorship.
19 MR. SCHMIDT: Okay, so let's get into
20 the staff review, the overview slide here. As MHI
21 had mentioned, the M-RELAP model is based on the
22 Idaho National Labs RELAP-3D code.
23 The methodology only applies as licensed
24 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.

1 So to fix those up would have caused a
2 schedule impact. We had RELAP-5 MOD3.3 up and
3 running and it was performing well, so to minimize
4 the impact on the overall review schedule, we went
5 with MOD3.3 but we did have both in parallel for a
6 while.
7 MEMBER SHACK: Did you develop your own
8 accumulator model?
9 MR. SCHMIDT: No. It's taken from the
10 Mitsubishi model.
11 MEMBER BANERJEE: Does ISL have TRACE
12 running right now?
13 MR. SCHMIDT: I guess I'm going to have
14 to ask ISL if they do run TRACE, but I don't know --
15 MR. PRELEWICZ: This is Dan Prelewicz
16 from ISL. We have run TRACE for a lot of other
17 cases but we didn't run it for MHI for the US APWR.
18 MR. SCHMIDT: Yes, I know that but I
19 wasn't sure if Sanjoy was asking just for US APWR or
20 just in general. Okay. Next slide.
21 One outstanding issue we had, which we
22 have talked about with Chairman Stetkar, is that the
23 final accumulator bias is still to be determined.
24 We did this review in parallel with the accumulator

1 model. At some point we thought the accumulator
2 model was somewhat resolved and that we could take a
3 scaling bias associated with it, put it in the
4 small-break LOCA model and proceed from there.
5 But as we -- as we proceeded in the
6 accumulator review we found that the scaling bias
7 that was used in this topical report may not be the
8 final value.
9 So there still is some uncertainty
10 associated with the final bias that is going to be
11 accounted due to the fact that there was half-scale
12 testing and that we are trying to come up with a
13 bias that would account for the full scale.
14 MEMBER BANERJEE: So when you say bias,
15 it's some multiplier that reduces the flow?
16 MR. SCHMIDT: Yes, that's correct.
17 MEMBER BANERJEE: And the timing, it
18 stays about the same for everything?
19 MR. SCHMIDT: The bias accounts is --
20 for their time zero.
21 MEMBER BANERJEE: Okay.
22 MR. SCHMIDT: And is always there, it's
23 a constant.
24 MEMBER BANERJEE: Okay. Maybe at some

1 point, if you aren't going to explain it now, when
2 we talk about the accumulator, we'll go through this
3 bias and how it's being set to take account of scale
4 effects.
5 MR. SCHMIDT: Okay. And the path
6 forward is, once the accumulator SE is finalized and
7 we come up with what the staff considers an adequate
8 scaling bias, then we are going to revise the small-
9 break LOCA topical report and revise the SE
10 reflecting the final scaling bias and of course the
11 Chapter 15.6.5 will be revised, showing the final
12 PCT values.
13 But really, the small-break LOCA
14 methodology, we feel, is fine. You know, it's been
15 benchmarked, validated. But the final PCT values
16 are still to be determined based on that accumulator
17 scaling bias.
18 MEMBER BANERJEE: How significant is the
19 reflector on the heat sink -- a heat source, sorry,
20 in this case?
21 MR. SCHMIDT: The reflector on the heat
22 source? It's pretty small compared to the decay
23 heat. We did look at that. It's a very small
24 contributor.

1 MEMBER BANERJEE: But this is a massive
2 reflector.
3 MR. SCHMIDT: Yes, it is. But still,
4 it's a massive core, it's 257 assemblies, 14 feet
5 high. We did --
6 MEMBER BANERJEE: The other thing, two
7 feet higher as well.
8 MR. SCHMIDT: Right, it's got a lot of
9 decay heat.
10 MEMBER BANERJEE: Right. Yes. Okay.
11 MR. SCHMIDT: The decay heat is still
12 down there. Next slide. Mitsubishi covered some of
13 this already so I will go quickly through this, some
14 of the design features of the APWR.
15 MEMBER BROWN: Can I ask a real quick
16 question?
17 MR. SCHMIDT: Sure.
18 MEMBER BROWN: Okay, again, on the
19 accumulator. You say you didn't -- you used their
20 model for the accumulator.
21 MR. SCHMIDT: Right.
22 MEMBER BROWN: Why did you come to the
23 conclusion that you were comfortable with their
24 model as opposed to independently verifying that

1 they had a reasonable model that would predict its
2 performance?
3 MR. SCHMIDT: Well I mean, we are really
4 handling in that in the advanced accumulator topical
5 report, right? The purpose of that report is to
6 come up with something that is satisfactory to the
7 staff in that advanced accumulator topical report.
8 If we find in that report for some
9 reason that the correlations that were developed,
10 were not applicable, we will have to come back and
11 revisit this. But we weren't really -- we really
12 considered that a separate item under the advanced
13 accumulator, and it took this really as input for
14 the small-break LOCA.
15 MEMBER BROWN: Okay, so you -- go ahead.
16 I'm sorry.
17 CHAIR STETKAR: I was going to say,
18 there are obviously some ends that need to be tied
19 together. That's one of the benefits for the way
20 that we have been conducting these reviews, is that
21 this meeting gives us an opportunity to kind of
22 weigh in on any issues, if you will, with the
23 overall methodology.
24 The results, we will not see those

1 results until the final review of the advanced
2 accumulator, the changes to the topical report and
3 the changes to Chapter 15, when we do our final
4 review of, you know, the SER with no open items,
5 which will be some time not this year.
6 MEMBER BANERJEE: Is there a schedule in
7 rough terms even?
8 CHAIR STETKAR: Yes, in rough terms. I
9 don't know what that is, but I'm sure there is one.
10 And I'm sure it will change. So it's sort of a
11 moot point.
12 MEMBER BANERJEE: The question of when
13 all this will come to a head, right, when you have
14 no more open items?
15 MR. SCHMIDT: Right now I think the
16 tentative schedule is March, March next year --
17 MEMBER BANERJEE: So that's fairly soon.
18 MR. SCHMIDT: for the advanced
19 accumulator.
20 MEMBER BANERJEE: Okay.
21 CHAIR STETKAR: That's for the advanced
22 accumulator.
23 MR. SCHMIDT: That's advanced
24 accumulator, right.

1 CHAIR STETKAR: The SER on the advanced
2 accumulator. The SER on the overall design
3 certification with no open items is -- I'm sure the
4 staff, I don't know if they want to comment, I'm
5 sure they have a schedule. It changes periodically.
6 It's probably not next year. Some of the chapters,
7 I suspect, will be next year. But --
8 MS. SNYDER: But they just reissued a
9 scheduling letter and I believe it's some time in
10 2015.
11 CHAIR STETKAR: '15, the final.
12 MEMBER BANERJEE: I was just wondering
13 whether I'll still be here on this committee.
14 CHAIR STETKAR: If we can nail your feet
15 to the floor, you will.
16 (Laughter)
17 MR. SCHMIDT: Okay, all right. Again,
18 you know, it's a very similar design to the
19 Westinghouse four-loop. The main bullet here is the
20 increased high pressure injection flow, safety
21 injection flow, on a megawatt basis and we gave a
22 number here of 2.8 times at the runout conditions,
23 and I'll show a plot later on of just how
24 significant that increased high pressure safety

1 injection flow is.

2 Again, increased heat generators, MHI

3 had already talked about that. They have a larger

4 pressurizer, 14-foot active port, neutron reflector,

5 as we know, and the core power is about 25 percent

6 higher but it still has a lower average kilowatt per

7 feet because of the large core, large number of

8 assemblies and high height.

9 Next slide.

10 The methodology we talked about before,

11 it's basically RELAP-3D, it's run in a 1D mode and

12 Mitsubishi added the Appendix K features and the

13 advanced accumulator model.

14 Again, as they mentioned, they followed

15 Reg Guide 1.203, Transient and Accident Analysis.

16 Again, the acceptance criteria, which we are all

17 familiar with, that's 10 CFR, Appendix K, ECCS

18 Evaluation Models.

19 Staff closely reviewed the 15.6.5, which

20 is the sensitivity analysis, and that's documented

21 in MUAP-07025 and we will talk a little bit about

22 that, actually during this presentation on the break

23 size sensitivity, there is an overlap.

24 MEMBER BANERJEE: So, the additional

1 features mainly were -- the Appendix K features were
2 mainly heat transfer correlations that were made
3 conforming to --
4 MR. SCHMIDT: Whatever made -- yes. For
5 instance the decay heat right, heat transfer
6 correlations, all the thing that make it acceptable
7 for Appendix K. But they had it.
8 MEMBER REMPE: So 3.3 is older. There
9 were a lot of updates to 3D. And so after MHI has
10 modified 3D, and you were trying to do confirmatory
11 calculations with an older code, don't you find
12 corrections that were implemented causing
13 differences, or have they improved 3.3 so that you
14 don't notice any discrepancies?
15 MR. SCHMIDT: I mean, there are
16 differences in the models. One of the questions we
17 did ask was are all the error notices for RELAP-3D
18 accounted for --
19 MEMBER REMPE: In this version of 3.3
20 MR. SCHMIDT: In this version of their
21 3D, which is the M-RELAP version if you will, the
22 Mitsubishi version. And they did address all the
23 known problems between the version that they took
24 from Idaho National Labs and their final M-RELAP --

1 MEMBER REMPE: I'm not making my
2 question very clear.
3 MR. SCHMIDT: Okay.
4 MEMBER REMPE: 3D has corrected things
5 that were in 3.3, just over the years.
6 MR. SCHMIDT: Oh, I see.
7 MEMBER REMPE: What I'm asking is, were
8 the differences in the ISL NRC 3.3 version, were
9 those corrections implemented when you compared it
10 to the Mitsubishi RELAP-3D version?
11 MR. SCHMIDT: Right, right.
12 MEMBER REMPE: And were they?
13 MR. SCHMIDT: I'm going to have to ask
14 Dan, who keeps track of all the changes to MOD3.3.
15 MR. PRELEWICZ: It's Dan Prelewicz from
16 ISL. We have been exchanging error reports for the
17 last several years with INL. So we, if there is
18 something that has been corrected in 3D, we correct
19 it in 3.3 and vice versa.
20 MEMBER REMPE: So it's not just the 3.3
21 that is referenced in the NUREG that I found in your
22 topical report? It's an improved version of 3.3?
23 MR. PRELEWICZ: Well, it's the version
24 of these, the NRC version of 3.3. We -- again, we

1 have been correcting the errors, so there are --

2 MEMBER REMPE: Okay.

3 MR. PRELEWICZ: The differences are --

4 no significant differences in terms of errors.

5 MEMBER REMPE: Okay.

6 MR. PRELEWICZ: The models might be

7 different but any errors that were found in either

8 code, are corrected in both codes.

9 MEMBER REMPE: Okay.

10 MR. SCHMIDT: Thanks, Dan. Okay, next

11 slide. The acceptance criteria, again, just a

12 repeat of what MHI had said, PCT below 2200, local

13 maximum cladding thickness less than 0.17, core wide

14 oxidation less than one percent and then the core

15 remains in a coolable geometry and assurance of

16 long-term decay heat removal.

17 Again, we really focused on M-RELAP and

18 small-break LOCA on the first three bullets.

19 The PIRT process, as MHI described,

20 broken down into the five phases: blowdown phase;

21 natural circulation; loop seal clearance; boiloff;

22 and core recovery. We agreed with those basic five

23 steps or stages of the small-break LOCA.

24 The model assessment cases as a part of

1 the validation were selected to evaluate the high-
2 ranking phenomena for each transient phase. And the
3 staff agreed that the thigh-ranked phenomena had
4 been correctly identified in the PIRT.
5 MEMBER BANERJEE: So, refluxing was not
6 considered high-ranking phenomena?
7 MR. SCHMIDT: I don't remember of the
8 reflux was in the PIRT on a high-ranking phenomenon.
9 I'd have to get back to you. I don't know if that
10 was a high-ranking parameter.
11 We did add assessment cases to address
12 reflux which you actually see on the next slide.
13 MEMBER BANERJEE: Okay.
14 MR. SCHMIDT: But I can't for sure say
15 off the top of my head if it was a --
16 MEMBER BANERJEE: It's just a matter of
17 the design, as to how significant the heat sink
18 stimulator can be.
19 MR. SCHMIDT: Yes, we did looking into
20 the CCFL correlations fairly heavily as part of this
21 review. And you know, we found that the UPTF data
22 for the 15-bar and the 3-bar were conservative for
23 CCFL correlations.
24 MEMBER BANERJEE: Okay, we'll get into

1 that.

2 MR. SCHMIDT: Yes. Yes. The staff did

3 ask questions on the validation test matrix. The

4 staff requested that larger break sizes be

5 evaluated. Originally the Rev 0 of the topical

6 repoirt had a five percent break size, which you

7 know, covered both loop seal and core boiloff, but

8 because of the nature of this plant and its systems,

9 we thought that larger break sizes should also be

10 evaluated, so we requested that 10 percent and 17

11 percent cases for the ROSA be added.

12 Also, the TMI action items requests

13 require that LOFT 3.1 and Semiscale be added as part

14 of the validation matrix so MHI did that.

15 And then we looked at some of the

16 reflood heat transfer as a separate effects test --

17 because of the large-break nature we wanted to make

18 sure what the heat transfer coefficients looked like

19 under the large-break regimes.

20 And through all the validation cases,

21 these are -- what's listed here are just the ones

22 the staff, MHI agreed to add. The M-RELAP

23 demonstrated the ability to represent the test data

24 fairly well.

1 Go ahead. Next slide. One of the
2 unique features of this plant, as we talked about
3 earlier, is the higher -- high pressure injection
4 flow rates. You know, one of the first things that
5 caught our attention was there was no typical heatup
6 in the two to four inch break of a typical PWR, a
7 Westinghouse PWR.
8 Like I said, we asked that validation
9 case be added to address the larger break sizes. We
10 also asked additional cases for the loop seal
11 clearing phenomenon, and then an overall explanation
12 of why we don't see the heatups in the two to four
13 inch range, and why we do see the large heatups in
14 what we consider a fairly large, small break.
15 So we kind of grouped -- had a lot of
16 questions in that regard, both small breaks, loop
17 seals and the large break phenomenon. Next slide.
18 Here is a -- this is a confirmatory
19 analysis where the staff artificially stopped loop
20 seals from clearing. You can see the circles are
21 all loop seals clear, and then as you move up to the
22 squares and the diamonds, we are letting less loop
23 seals clear.
24 And the purpose of this --

1 MEMBER BANERJEE: This is what I was
2 looking for.
3 MR. SCHMIDT: Yes, I thought it might
4 be. What we were trying to gauge here is, because
5 we didn't see the heatup in the lower break ranges
6 that we are used to seeing, we wanted to understand
7 why.
8 So we stopped, we artificially stopped
9 the loop seals from clearing, and even with, you
10 know, the two loop seals, you know, can clear, case,
11 we still didn't see a heatup in the low break sizes.
12 And the real reason for that is the high
13 pressure safety injection flow is just so great it
14 can make up for breaks up to six inches, and that's
15 really why you don't see the two to four inch break
16 in it, but I'll have some followup slides which will
17 further describe this too.
18 But, so this was kind of what --
19 MEMBER BANERJEE: Just the -- that you
20 have more flow is that it?
21 MR. SCHMIDT: Yes, that's it.
22 MEMBER BANERJEE: Because this is very
23 unusual results.
24 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.

1 And you can see the blue line there, the
2 diamond line, you do see the peak that you typically
3 see with the Westinghouse plant in the two to four
4 inch range when the HPI flow is significantly
5 reduced.
6 MEMBER BANERJEE: How many loop seals
7 cleared here?
8 MR. SCHMIDT: Let's see.
9 MEMBER BANERJEE: All of them?
10 MR. SCHMIDT: I don't -- all allowed, I
11 believe. We never -- we didn't artificially control
12 the loop seal clearing in this case.
13 MEMBER BANERJEE: On this one.
14 MR. SCHMIDT: Yes. Right. Only the one
15 where we were trying to understand the break size
16 behavior initially. So this would just be whatever
17 the model represents, right?
18 The other thing we had looked at was the
19 green curve here, which isn't that significant. We
20 were also looking at, as part of he understanding of
21 the break size phenomena, we looked at steam
22 generator size too, because it's bigger than the
23 model on a four-loop Westinghouse plant.
24 So we were looking at both parameters,

1 but, you know, as the graph shows, by and large it's
2 really the -- it's the increased safety injection
3 flow that really makes the transient behave
4 different --
5 MEMBER BANERJEE: But you used steam
6 generator size in the four-inch case, gives you
7 quite a bit lower peak clad temperature, right?
8 MR. SCHMIDT: Quite a bit -- well, I
9 mean that's with -- that's only changing the steam
10 generator. It doesn't have the reduced flow.
11 MEMBER BANERJEE: Oh, okay.
12 MR. SCHMIDT: That's the difference. So
13 they are two separate cases. So the green would be
14 the MHI flow rates with just the reduced steam
15 generator.
16 MEMBER BANERJEE: Yes, so you are
17 getting the effect of the reduced -- sorry, I got
18 mixed up.
19 MR. SCHMIDT: Oh, that's fine. That's
20 just -- I probably didn't do a very good job of --
21 MEMBER BANERJEE: No, but it's there in
22 the caption. I just didn't read it correctly. Go
23 ahead.
24 MR. SCHMIDT: Okay. Next slide. Some

1 of the other issues that the staff found that are in
2 the review was the critical flow model that was
3 being used.
4 MHI was using the Moody correlation and
5 the pull through model and there were issues
6 associated with the modeling there, associated with
7 the stub pipe, which is a connection between the
8 cold leg and the containment basically.
9 I don't want to get too much into the
10 modeling of that, because I think it's considered
11 proprietary, so we can talk more about that
12 tomorrow. But the general gist was the staff was
13 finding, as we varied the stub pipe size and
14 dimensions, that we were seeing vastly different PCT
15 responses, and I have a curve that will show that.
16 And that kind of led the staff to start
17 to question things like the pull through model and
18 the break flow correlation an its uses.
19 So, let's go to the next slide and we'll
20 take a look at -- confirmatory calculations show
21 that the Moody was not consistently applied when two
22 phases were present as required by Appendix K, and
23 there were some issues with the pull through model
24 switching logic that caused numerical oscillations.

1 MHI corrected these issues and
2 effectively eliminated the strong dependence on the
3 PCT response to the actual size of this stub pipe or
4 this phantom pipe.
5 And as MHI described, the bottom leg
6 break became the limiting break, which is consistent
7 with other PWRs. And I think the next slide
8 probably shows the real interesting portion of it.
9 The black line there is this PCT versus
10 time for the limiting case, the one square foot
11 break case, and it's with the two versions of the
12 code -- 1.4, which had some of the issue with the
13 Moody break correlation and the pull through model,
14 and then the final version, which is version 1.6.
15 So, the version 1.6 is the DCD value and
16 you can see that with the stub pipe -- the stub pipe
17 sensitivities, we plotted here the max and the min
18 PCTs.
19 So we looked at a range of different
20 stub pipe sizes, and since it's an imaginary pipe,
21 it can take on a variety of imaginary dimensions.
22 So what we did here is, this is a case where 1.4,
23 which is the green and the red line, you see a huge
24 different in PCT based on the stub pipe dimensions.

1 Now, granted, this is the worst case.
2 But you can see why the staff was concerned and had
3 a lot of followup questions in this.
4 MEMBER SHACK: Is this model something
5 that MHI did to RELAP, or is this something that
6 occurred because you are now using RELAP and forcing
7 it to do Appendix K-type calculations?
8 MR. BIELEN: We should probably --
9 they're very protective of their modeling choices as
10 far as this goes, so I think we should probably
11 defer that to either closed or tomorrow.
12 MR. SCHMIDT: Yes, I think that would be
13 a good idea. I'd probably go to --
14 MEMBER BANERJEE: Are we going to have a
15 closed session, John?
16 CHAIR STETKAR: Yes.
17 MEMBER BANERJEE: Okay. So let's guide
18 this as a --
19 CHAIR STETKAR: We'll see how we go
20 today. I mean, if things -- if we get through
21 material in a timely fashion today, we can close
22 part of the session later this afternoon.
23 MEMBER BANERJEE: It would be
24 interesting to know more about this.

1 CHAIR STETKAR: We don't want to let too
2 many things pile up until tomorrow because of the
3 schedule problems we have.
4 MEMBER BANERJEE: Okay.
5 MR. SCHMIDT: Next. This is a plot the
6 staff put together really showing -- and the red
7 there is the Henry Fauske correlation and the black
8 is the Moody correlation and you can see that the
9 code was switching back and forth between
10 correlations and that caused some of the PCT
11 response you saw versus time. This is just to
12 illustrate that, that it was a switching problem
13 back and forth between the two break flows.
14 The other major issue, or another major
15 issue we had was this --
16 MEMBER BANERJEE: How was this resolved
17 finally?
18 MR. SCHMIDT: It went to Moody only.
19 MEMBER BANERJEE: And for the -- was
20 there a period of subcooled discharge? Did they use
21 Henry Fauske at that point?
22 MR. SCHMIDT: They used whatever was the
23 worst case in the part of the -- during part of the
24 transient. The logic would pick whatever the worst

1 case scenario would be.

2 MEMBER BANERJEE: I'm just trying to

3 imagine. You've got a hole, let's say, in the cold

4 leg. So for a period this is going to be sub-cooled

5 and they, presumably if we are using Henry Fauske, I

6 don't know, maybe we should ask --

7 MR. SCHMIDT: If you go back to the

8 previous slide there, you'll see, let's see --

9 MEMBER BANERJEE: It's not clear what

10 they are doing here.

11 MR. SCHMIDT: There is --

12 MEMBER BANERJEE: The M flow J --

13 MR. SCHMIDT: Oh, actually this is the -

14 - this is the uncorrected version, so you are really

15 asking about the corrected version.

16 MEMBER BANERJEE: So, M flow J is some

17 model of their own?

18 MR. SCHMIDT: No, it's just a variable

19 that we are -- it's kind of a time average variable

20 of what both correlations are doing.

21 MEMBER BANERJEE: Oh, okay.

22 MR. SCHMIDT: Perhaps maybe Dan or ISL

23 can better answer Sanjoy's question on the --

24 MEMBER BANERJEE: So, during sub-cool

1 discharge, what's being used?

2 MR. PRELEWICZ: Not sure I exactly

3 understand the question. There was a switching

4 logic problem where instead of -- whenever the --

5 Appendix K requires whenever it's two-phase, you are

6 supposed to use the Moody model.

7 MEMBER BANERJEE: Right, I'm talking

8 about sub-cooled. What did you use during sub-

9 cooled?

10 MR. PRELEWICZ: Well, the sub-cooled,

11 they can use the Henry Fauske, yes, when it's

12 sub=cooled.

13 MEMBER BANERJEE: Yes, so that's what I

14 was saying. Did they use Henry Fauske during sub-

15 cooled?

16 MR. PRELEWICZ: Yes.

17 MEMBER BANERJEE: And then switched to

18 Moody when it was two-phase.

19 MR. PRELEWICZ: Well, that's what it

20 does now. Before, there were some cases when it was

21 two-phase and it was still using Henry Fauske.

22 MEMBER BANERJEE: Ah, I get the problem

23 now. Okay.

24 MR. PRELEWICZ: And it was switching

1 back and forth --

2 MEMBER BANERJEE: Between the two --

3 MR. PRELEWICZ: during the event.

4 MEMBER BANERJEE: But why was it doing

5 that? Was that just a cooling error or --

6 MR. PRELEWICZ: Well, the switching --

7 the switching logic wasn't based on two phase. It

8 was phased on -- if you have non-equilibrium, there

9 was two phases there. But if you looked, you mixed

10 them all together, you would be sub-cooled. So they

11 had like sub-cooled voids and that sort of thing in

12 there that was causing it to switch back to the

13 Henry Fauske --

14 MEMBER BANERJEE: So it was taking a

15 quality based on averaging between the two phases,

16 and they were in non-equilibrium, so the water was

17 sub-cooled and the steam could be superheated or

18 whatever.

19 MR. PRELEWICZ: Yes, but there was steam

20 there.

21 MEMBER BANERJEE: Yes, okay.

22 MR. PRELEWICZ: And it was supposed to

23 be using Moody but it wasn't.

24 MEMBER BANERJEE: Right.

1 MR. PRELEWICZ: As you can see, it was
2 switching back and forth. Wherever the red, it was
3 using -- on this curve, it was still way down in, it
4 was using the Henry Fauske where it should have been
5 using Moody, should have been all the black color.
6 The red showed up. That means it was using the
7 wrong critical flow correlation.
8 MEMBER BANERJEE: So, rather than using
9 an average quality it should use a void fraction to
10 switch or something?
11 MR. PRELEWICZ: We did a lot of testing.
12 We looked at things like the void fraction,
13 whenever there was any voids we should use Moody. I
14 don't remember exactly how -- what criteria it wound
15 up in. It's now, both codes have a criteria that
16 uses only the Moody. You have the same problem with
17 3.3 --
18 MEMBER BANERJEE: Henry Fauske basically
19 goes to homogeneous, doesn't it, at two-phase flows?
20 I mean, there is no sub-cooling what happens to
21 Henry Fauske? Remind me.
22 MR. PRELEWICZ: When there is no sub-
23 cooling?
24 MEMBER BANERJEE: Yes.

1 MR. PRELEWICZ: Well --

2 MEMBER BANERJEE: You know the Henry

3 Fauske correlation, where does it go? It actually

4 applies, even in two-phase flow, doesn't it?

5 MR. PRELEWICZ: Well, the code was

6 calculating Henry Fauske in two phase, that's

7 correct, and I don't know, it may have been used

8 again improperly. I'm not sure. But it was

9 calculating using Henry Fauske in two phase.

10 MEMBER BANERJEE: It should go -- it

11 should go into two phase as well, but I'll look it

12 up myself. Don't worry. Okay. Go ahead.

13 MR. SCHMIDT: The other issue, as I

14 started to mention, was there was no detailed

15 scaling evaluation in the original Rev 0 of the

16 topical report. There was some comparisons and some

17 verbiage regarding four-loop -- a comparison to a

18 four-loop Westinghouse plant.

19 The staff had questions about the

20 scaling to a four-loop plant because of the size

21 differences, and requested that a quantitative

22 scaling analysis be performed.

23 Initial scaling analysis was limited to

24 a single facility and a single test and did not

1 cover the range of phenomena. MHI provided a final
2 scale and report with multiple ROSA cases.
3 The top-down scaling demonstrated that
4 the relative rankings of the non-dimensional
5 coefficients between the plant and the test were
6 preserved.
7 No new or different system interactions
8 from those exhibited in the test facility, and the
9 same dominant processes and phenomena occur in the
10 US APWR and the integral test experiments that were
11 used for validation.
12 Experimental data are acceptable for
13 validation of the M-RELAP5 code, and there were some
14 minor scaling distortions identified in the top-down
15 analysis were evaluated in the bottom-up --
16 MEMBER BANERJEE: So what were these
17 scaling distortions? Do you remember?
18 MR. SCHMIDT: No, I'm going to turn to
19 Marcos for hopefully remembering what the
20 distortions were.
21 MEMBER BANERJEE: You have to come to
22 the mic, Marcos.
23 MR. ORTIZ: This is Marcos Ortiz from
24 ISL. I don't exactly remember which ones. What I

1 remember that we did was, we looked at the
2 coefficiency questions and made sure that they were
3 in the same magnitude, and the past the couple of
4 more important ones, in the end there were variants,
5 and then we checked those.
6 But they were in the non-important --
7 MEMBER BANERJEE: So, let me try to
8 understand this. We are trying to look at the
9 applicability of the code to this system, which has
10 quite a lot of unusual attributes, for example the
11 accumulators and the lack of a low-pressure
12 injection system
13 I would have expected significant
14 distortions in the top-down scaling because of these
15 -- the flows are so different and everything else.
16 And you know there would be== it's just an
17 unexpected result to think that the distortions are
18 small, in a case where you've got three times as
19 much HP flow.
20 MR. ORTIZ: Well, the accumulator was
21 not in the part that I reviewed. It was -- that was
22 a separate --
23 MEMBER BANERJEE: Yes, the real question
24 to me is do the ROSA integral effect tests provide

1 an adequate database with the accumulators being a
2 purely separate effects test. There is no
3 integrated test done with the accumulators, right?
4 MR. ORTIZ: Yes. That's correct.
5 MEMBER BANERJEE: So I would expect
6 large distortions.
7 MR. ORTIZ: They were not in the
8 analysis that I reviewed, you know, the --
9 MEMBER BANERJEE: But you did it without
10 the accumulators, right?
11 MR. ORTIZ: Well, yes, at such, yes.
12 MEMBER BANERJEE: So to me that has
13 always been an open question and I should say this,
14 that it could be that -- thanks Marcos, but maybe
15 you can answer this. But it can be that we can have
16 a bunch of integral tests, which are the ROSA, or
17 whatever else the integral tests there are, then you
18 have te separate effects tests with the
19 accumulators. We had no integral effects tests with
20 the accumulators. So there seems, at some point, a
21 fairly -- you are in some area of scaling where
22 there isn't a large database available to hang your
23 hat on.
24 It doesn't mean that the results don't

1 hold, but I would expect there would be distortions.
2 That's why I was startled by this.
3 MR. SCHMIDT: In some sense, you know,
4 during the large-flow portion of the accumulator, it
5 acts just like a regular accumulator.
6 MEMBER BANERJEE: Right.
7 MR. SCHMIDT: You know, and in that
8 phase, it's -- and most of these --
9 MEMBER BANERJEE: So you think you can
10 appeal to the database there?
11 MR. SCHMIDT: Right. Right.
12 MEMBER BANERJEE: Okay.
13 MR. SCHMIDT: And most, for instance,
14 like the limiting case, the one square foot break
15 case, you know most of that flow, as MHI pointed
16 out, is during the large-flow phase of the
17 accumulator.
18 So, again, it's acting like a regular
19 accumulator at that point.
20 MEMBER BANERJEE: Yes, the thing that
21 was sort of different, is where you expected to see
22 the peaks in the Westinghouse-type plants, which is
23 the four-inch breaks. You don't' see the breaks.
24 MR. SCHMIDT: Right, right.

1 MEMBER BANERJEE: You see quite
2 different sort of phenomena occurring with regard to
3 PCT. You are getting no peaks and then you are
4 getting something at a larger break size.
5 So -- and you know, the Westinghouse
6 plants have been well scaled against experimental
7 data, the codes have been validated etcetera. It's
8 just that here it seems you are seeing different
9 things, different phenomena.
10 MR. SCHMIDT: I would say it's probably
11 not different phenomena. I would say it's due to a
12 plant design difference that causes the difference
13 in behavior, not the phenomenon, but the plant.
14 MEMBER BANERJEE: Yes, I agree. But the
15 integral effects tests that were used to validate
16 the codes and so on, and show the applicability of
17 the codes to the SBLOCAs for these Westinghouse
18 plants, you know, the concern is, are these set of
19 integral tests sufficient? Should there have been a
20 set down with, you know --
21 MR. SCHMIDT: Right, an integral system
22 basically --
23 MEMBER BANERJEE: Yes, an integral
24 system. It seems to be missing.

1 MR. SCHMIDT: You know, obviously there
2 weren't.
3 MEMBER BANERJEE: Yes.
4 MR. SCHMIDT: And again, I think the
5 only thing I can really say is that in most cases,
6 this accumulator acts like a normal accumulator for
7 these --
8 MEMBER BANERJEE: Well, I can read that,
9 yes.
10 MR. SCHMIDT: break sizes. So I mean,
11 that's really, I think, one of the reasons why we
12 are saying that the ROSA tests are still valid for
13 the APWR.
14 MR. BIELEN: And they also did a ROSA
15 test recently, in 2009, for the 17 percent
16 equivalent break to try to look at APWR-specific --
17 MEMBER BANERJEE: Ah, okay. That's more
18 interesting.
19 MR. SCHMIDT: But it was still with a
20 traditional accumulator --
21 MR. BIELEN: It was still with the --
22 (Simultaneous speaking.)
23 MR. BIELEN: -- but at least you are
24 still looking for the same kind of thermo-hydraulic

1 phenomena that you would expect in the US APWR
2 specifically.
3 MR. BIELEN: But they added the 17
4 percent --
5 MR. SCHMIDT: Right. That matched up
6 with the one square foot limiting break here.
7 MEMBER BANERJEE: Because when you look
8 at that nice curve you showed with the loop seals
9 not being allowed to clear, you know, it's sort of
10 an interesting curve, because --
11 MR. BIELEN: Agreed.
12 MEMBER BANERJEE: Right. Well this is
13 all very interesting, because you are seeing sort of
14 different phenomena because in the small-break
15 business, you know, we look and you are getting this
16 sort of -- but you see a big separation by the
17 number of loops he's clearing as you get to the
18 larger breaks.
19 MR. ORTIZ: I -- the different
20 interactions. The interactions are delayed.
21 There's a little different -- the phenomena are
22 still the same though.
23 MEMBER BANERJEE: I believe the
24 phenomena are the same, but the pie groups are

1 supposed to be measures of well your experiment
2 applies to your full-scale system, right?
3 MR. ORTIZ: Yes. Yes. And they were
4 quite similar, but -- and more importantly, they
5 kept the same relative ranking.
6 MEMBER BANERJEE: Well, if you didn't
7 specifically look at accumulator effects, then --
8 MR. ORTIZ: Well, it's a separate thing,
9 but if the flow was there, the numbers were there,
10 so the parameters were there, and the pies accounted
11 for whatever they --
12 MEMBER BANERJEE: So you took the flows
13 from the accumulators in calculating your pie
14 groups, or how did you do it?
15 MR. ORTIZ: I reviewed what they had in
16 terms of those pie numbers. It was not like our
17 AP600 deal where I did the whole thing. I reviewed
18 what they had.
19 MR. SCHMIDT: But it has the accumulator
20 flow --
21 MR. ORTIZ: It has the numbers in there.
22 MEMBER BANERJEE: It has the numbers.
23 Okay. All right. Well, there is a report on this,
24 right?

1 MR. SCHMIDT: Yes, there is a separate
2 scaling --
3 MEMBER BANERJEE: Scaling report.
4 MR. SCHMIDT: Yes.
5 MEMBER BANERJEE: And we have access to
6 it, clearly.
7 MR. SCHMIDT: Yes.
8 MEMBER BANERJEE: Girija, let me take a
9 quick look at that at some point. Okay. Thanks.
10 It's an MHI report.
11 MR. SCHMIDT: And I looked up while you
12 guys were speaking, one of the distortions was in a
13 volumetric flow out the break, was one of the
14 distortions that was looked at from a bottom-up
15 perspective too.
16 MEMBER BANERJEE: Yes, the distortion
17 that would be here is the break flow to the
18 accumulator flow, or the break flow to the HPI flow.
19 There would have to be some significant distortions
20 there. But I'll take a quick look at it. Okay, go
21 ahead.
22 MR. SCHMIDT: Okay. Next slide please.
23 Conclusions. MHI small-break LOCA methodology is
24 acceptable for calculating PCT, hydrogen generation

1 and fuel cladding surface oxide for the US APWR with
2 the appropriate Appendix K evaluation models.
3 What we listed there are the preliminary
4 or I would say today's PCT values. Again, as we
5 discussed, the accumulator scaling bias is not
6 final, it's not been determined. But there is
7 significant margin to the 2200 here.
8 MEMBER BANERJEE: Unless only one loop
9 seal clears.
10 MR. SCHMIDT: Unless only one loop seal
11 clears. That's right, which for these large breaks,
12 we anticipate --
13 MEMBER BANERJEE: Yes, these are fairly
14 big breaks.
15 MR. SCHMIDT: These are big small
16 breaks, right, because of the high pressure
17 injection. So as MHI discussed, you know, there
18 were no artificial means of controlling the loop
19 seals if you well, so -- and the reason why staff
20 found that acceptable is because of the HPI flow,
21 that limiting breaks are big breaks, and you would
22 expect the loop seals --
23 MEMBER BANERJEE: Yes, I think that's
24 the argument, because the staff, for example, but

1 EPUs really for the small breaks, as required that
2 you keep the loops seals, only let one clear.
3 MR. SCHMIDT: Only want one clear.
4 Right, right. We do not have the requirement here.
5 MEMBER BANERJEE: Yes, but that was --
6 that's in the smaller sizes.
7 MR. SCHMIDT: Right, right.
8 MEMBER BANERJEE: So, we need to think
9 through that logic, what, you know --
10 MR. SCHMIDT: We did consider that quite
11 a bit --
12 MEMBER BANERJEE: Right.
13 MR. SCHMIDT: in the review, and that
14 was one of the reasons why, our first curve, we were
15 examining, you know, as we've artificially closed
16 off loops, we've stopped them from clearing, we
17 wanted to see the response to see if it did push it
18 down to the four-inch and then we were going to
19 reconsider whether we had to add some artefacts to
20 the loop seal clearing.
21 MEMBER BANERJEE: Okay. All right.
22 MR. SCHMIDT: And this is, the final
23 slide here, is just to give an idea again on break
24 spectrum versus the staff's RELAP model. Again you

1 see a very good match up of behavior between the two
2 models, again, this is with their final corrected
3 version 1.6.
4 As you can see, you know, again, you
5 don't see the heatups on the small breaks. You see
6 a small increase in the MHI model on 7.5 inch loop
7 seals clearing break, and then again as you move up
8 in break size the PCT just goes up.
9 MEMBER BANERJEE: And this is with two
10 trains of high pressure?
11 MR. SCHMIDT: Yes. Okay. All right.
12 Well, thank you very much.
13 MEMBER BANERJEE: Interesting.
14 CHAIR STETKAR: Any other questions or
15 comments for the staff? If not, thank you very
16 much. Very good presentations.
17 MR. SCHMIDT: Thank you.
18 CHAIR STETKAR: You guys did really
19 well.
20 MEMBER SHACK: These background slides
21 you have, you didn't go through.
22 MR. TAKACS: Some of them may be
23 proprietary.
24 CHAIR STETKAR: Unless Dr. Shack really

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

2 MEMBER BANERJEE: Well, if we are having

3 a proprietary session, we probably should.

4 CHAIR STETKAR: Well, what I want to do

5 is, let's see how we get through this stuff. I

6 don't want to push too much stuff until tomorrow

7 because otherwise --

8 MEMBER BANERJEE: Well, if we don't have

9 to write a letter, we can always --

10 CHAIR STETKAR: We don't have to write a

11 letter, but I also don't want to -- if any issues

12 come up regarding the proprietary information, I

13 want to make sure we have enough time to get them

14 out on the table, either today or tomorrow, by the

15 time, you know, by the time we finish tomorrow noon

16 time. So --

17 MEMBER BANERJEE: Well, what is

18 happening afternoon, the P&P?

19 CHAIR STETKAR: No, well, at noon we

20 have got PMP and tomorrow afternoon there are two

21 other subcommittee meetings, Seabrook and --

22 MEMBER BANERJEE: Oh, right.

23 CHAIR STETKAR: And flooding. So we

24 can't run into tomorrow afternoon and we can't push

1 the P&P because we are backing up against
2 subcommittee meetings, so we are --
3 MEMBER BANERJEE: Yes sir.
4 CHAIR STETKAR: We are stuck. We
5 originally had this scheduled for two full days and
6 we had to -- I got negotiated back to a day and a
7 half.
8 MEMBER BANERJEE: I'm sure you are a
9 good negotiator John?
10 CHAIR STETKAR: Not too well on this
11 one, but at least I held it to days where you didn't
12 have to come back from Switzerland.
13 (Whereupon, the meeting went off the record at 10:10
14 a.m. and resumed at 10:26 a.m.)
15 CHAIR STETKAR: Okay, let's reconvene
16 and we'll hear from Mitsubishi about the large LOCA
17 methodology. Dr. Banerjee will join us.
18 MR. TERAMAE: Thank you, Chairman. Good
19 morning everyone my name is Tetsuya Teramae. I am
20 working for the Mitsubishi Heavy Industries.
21 This morning I am talking about the
22 large-break LOCA code applicability report for US
23 APWR. MHI has committed to the NRC to supplement
24 its US APWR design certification application.

1 And today's -- this shows that today's
2 presenters, I will be supported today by my co-
3 presenter, Ms. Narutomi and as well as MHI staff.
4 Here are the contents of our
5 presentation today. We will be presenting the
6 LBLOCA code and methodology. I will provide an
7 overview of the topical report and then I'm going to
8 present some of sample plant analysis with analysis
9 and result, with WCOBRA/TRAC M1.0 code. Finally, I
10 will summarize our presentation.
11 So let me start with a brief overview
12 that highlights the main aspects of the topical
13 report. The objective of the topical report is to
14 present a comprehensive assessment of the
15 applicability of WCOBRA/TRAC M1.0 code with ASTRUM
16 methodology to US-APWR LBLOCA analysis.
17 The key topics in the topical reports
18 are, first, the US APWR design features to be
19 evaluated for the code applicability have been
20 identified and discussed.
21 Second, the applicability of the code
22 and the methodology of -- to US APWR has been
23 examined and confirmed, based on the code scaling
24 applicability and uncertainty evaluation

1 methodology.

2 In relationship with the design

3 certification document, design control document,

4 DCD, WCOBRA/TRAC M1.0 with ASTRUM methodology is

5 being applied to the LBLOCA analyses for US APWR

6 design certification. The topical report is

7 referenced in section 15.6.5 of the US APWR DCD.

8 I'd like to begin with slide 6 plus, and

9 then I'm going to go back slide 5 and then slide 4.

10 Sorry for this inconvenience but I think the

11 explanation will be more clear in this order.

12 Let's show page 6 of the slides, it

13 starts applicability of the code and methodology.

14 The applicability of the code and the evaluation

15 methodology to the US APWR has been examined and

16 confirmed based on the code scaling, applicability,

17 and the uncertainty evaluation methodology.

18 That means it has been considered under

19 the following five main viewpoints: for the US APWR

20 PIRT, the modeling requirements for performing

21 LBLOCA calculations for the US APWR were identified

22 and discussed.

23 And at the same time, the features of

24 the US APWR that affect the applicability of the

1 existing PIRT were specified and estimated.

2 In the place of the code applicability

3 determination, the applicability and assessment for

4 features of the US APWR was examined based on the

5 PIRT result.

6 The nodalization for the US APWR has

7 been developed based on the same scheme approved for

8 current three- and four-loop PWRs in the U.S.

9 Then, the demonstration for typical

10 LBLOCA characterization with WCOBRA/TRAC M1.0 has

11 been provided.

12 For the statistical analysis, the

13 uncertainty treatment for new or improved features

14 and other plant-specific parameters of the US APWR

15 are determined.

16 MEMBER REMPE: I have a question on the

17 previous slide. When you talk about he same scheme

18 for nodalization approved for current three- and

19 four-loop PWRs, do you really mean what you did was

20 follow recommendations to come up with your

21 nodalization? Or what -- could you elaborate what

22 you mean by that?

23 (Off the record comments)

24 MR. TERAMAE: Nodalization is originally

1 based on the approved methodology, ASTRUM, developed
2 by the Westinghouse methodologies, and --
3 MEMBER REMPE: So it was an ASTRUM-
4 approved methodology.
5 MR. TERAMAE: Yes. That methodology has
6 nodalization scheme, basically, and we, based on the
7 same scheme nodalization.
8 MEMBER REMPE: Okay. And they gave you
9 guidelines, perhaps, on how big of a volume you
10 might use, or something like that, or how did you
11 adapt it to your plan? What kind of guidance?
12 (Off the record comments)
13 MR. TERAMAE: The original scheme
14 basically depends on the loops, number of loops.
15 For example the four-loop plant, the downcomer is
16 divided from the four divided channel.
17 MEMBER REMPE: So you had, like, maybe
18 in some cases a larger volume but you used the same
19 number of --
20 MR. TERAMAE: Yes.
21 MEMBER REMPE: nodes. For the core
22 height, for example, you might have this 10 or
23 something still, even though the plant has a larger
24 core?

1 MR. TERAMAE: Core height, core height
2 divides based on the grid, portion of the grid, grid
3 spacer. And we refer to grid and a half, grid to
4 grid on the series methodology of that main.
5 MEMBER REMPE: Okay, thanks.
6 MR. TERAMAE: Therefore the longer core
7 are only the ones where stability is increased.
8 MEMBER REMPE: Okay, thank you.
9 MR. TERAMAE: Can I move on?
10 MEMBER REMPE: Yes.
11 MR. TERAMAE: Back to the slide number
12 5. The applicability of WCOBRA/TRAC M1.0 to the US
13 APWR has been evaluated and confirmed especially
14 respective to simulating high-ranked LBLOCA
15 phenomena and new or improved design features, such
16 as: longer core; advanced accumulator; and direct
17 vessel injection; and neutron reflector.
18 This slide lists computer code and
19 uncertainty evaluation methodology, using the best
20 estimate LBLOCA analysis for the US APWR.
21 The WCOBRA/TRAC M1.0 code is MHI's
22 version of WCOBRA/TRAC which incorporates the US
23 APWR design features of the advanced accumulator and
24 neutron reflector, and it calculates the thermo-

1 hydraulic behavior during the LBLOCA.

2 The HOTSPOT code calculates the fuel rod

3 behavior, such as the cladding burst, metal-water

4 reaction, and fuel relocation following burst

5 phenomena, using the same model, same fuel rod model

6 as that using the WCOBRA/TRAC. The code can

7 calculate the effect of the uncertainties.

8 ASTRUM, standing for automated

9 statistical treatment of uncertainty method, is a

10 non-parametric order statistical methodology that

11 does not assume the peak cladding temperature

12 distribution.

13 This statistical method determines that

14 124 cases must be run in order to assure the 95/95

15 level, peak cladding temperature, local maximum

16 oxidation and core-wide oxidation.

17 Next slide will provide some result of

18 the DCD section 15.6.5 reference case as a sample.

19 Major LOCA parameters for WCOBRA/TRAC M1.0 are

20 listed in the table 3.6-5 in the topical report.

21 The value of statistical parameters are

22 basically nominal for the reference cases. As I

23 said before, the nodalization was developed based on

24 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.

1 MEMBER BANERJEE: What scale? What
2 scale was the data?
3 MR. TERAMAE: One third.
4 MEMBER BANERJEE: And did you do any two
5 phase tests?
6 MR. TERAMAE: Two phase data is not
7 obtained MHI, but US APWR plant and current
8 operating Westinghouse plant are all near very
9 similar to the pumps, and very near value of the
10 specific feed.
11 Therefore we -- MHI decided 93A, two-
12 phase data adopted to the US APWR two-phase data.
13 MEMBER BANERJEE: So you used the
14 similar curves to Westinghouse or --
15 MR. TERAMAE: Yes.
16 MEMBER BANERJEE: what did you -- yes.
17 So are your pumps similar? It seems bigger.
18 MR. TERAMAE: Bigger. It was just
19 bigger.
20 MEMBER BANERJEE: Just bigger. And did
21 you, did you assess the size effect on your two-
22 phase flow behavior, the curve pumps?
23 MR. TERAMAE: It's --
24 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?

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?

1 MR. TERAMAE: Yes.

2 MEMBER BANERJEE: So, your peak clad

3 temperature is always coming the second peak, not

4 the first peak? There is two peaks, right? In the

5 blowdown peak and the reflood peak?

6 MR. TERAMAE: Yes.

7 MEMBER BANERJEE: So you are always in

8 the reflood peak?

9 MR. TERAMAE: That depends on the break

10 type or sites.

11 MEMBER BANERJEE: I'm saying large

12 break, usually what do you find?

13 MR. TERAMAE: Usually the reflood PCT is

14 --

15 MEMBER BANERJEE: Controlling? The

16 second peak is bigger than the blowdown peak.

17 MR. TERAMAE: Yes.

18 MEMBER BANERJEE: Okay. But you have a

19 lot of emergency cooling, right? So -- I'm trying

20 to understand the phenomena because of your -- if,

21 if you have a lot of emergency cooling, shouldn't

22 that reduce your reflood peak? Or does it come too

23 late maybe?

24 I don't know, when does your high

1 pressure injection start? Around 100 seconds or
2 what?
3 MR. TERAMAE: Yes. The single failure
4 condition on the offsite power off condition, the
5 high head injection, about 124 seconds.
6 MEMBER BANERJEE: Ah, okay. That
7 explains -- 124 seconds.
8 MR. TERAMAE: Yes.
9 MEMBER BANERJEE: It's delayed a little
10 bit.
11 MR. TERAMAE: Delay time is 100.
12 MEMBER BANERJEE: Okay. That's because
13 of your gas turbine right?
14 MR. TERAMAE: Yes.
15 MEMBER BANERJEE: Okay. So, I'm just
16 looking at it. You don't have the flow anywhere
17 here. It would be useful to know what the different
18 flows are, you know, from which sources they are
19 coming, and how much flow, because your high head
20 injection is 124 seconds.
21 MR. TERAMAE: Yes.
22 MEMBER BANERJEE: So that is on when you
23 get the core quench. But where does the water go?
24 Does it fill the downcomer and get into the core?

1 It would be nice to know where the water sources
2 are, and where they are going, the water, okay?
3 Because this, the plant is a little bit different.
4 Your high head is coming a bit later, right?
5 So -- and when do the accumulators come?
6 MR. TERAMAE: Accumulators come from the
7 cold leg.
8 MEMBER BANERJEE: No, I'm saying what
9 time is the --
10 MR. TERAMAE: Accumulator injection
11 started --
12 MEMBER BANERJEE: Okay, you are showing
13 -- okay.
14 MR. TERAMAE: is 30 minutes -- 30
15 seconds.
16 MEMBER BANERJEE: Okay. Okay. So 10
17 seconds, right? Thirteen seconds. Thirteen, okay.
18 MR. TERAMAE: Thirteen.
19 MEMBER BANERJEE: One-three.
20 MR. TERAMAE: One-three. Thirteen.
21 MS. STEINMAN: It is 13 seconds.
22 MEMBER BANERJEE: Okay, well that also
23 explains, yes.
24 MR. TERAMAE: Modeled results presented

1 tomorrow's session will.

2 MEMBER BANERJEE: Okay. Now, so you are

3 saying that the two-phase pump model is not

4 important. Did you do some calculations of this, to

5 show it's not very important for the loop case?

6 MR. TERAMAE: Loop case, loop case also

7 the short of the two-phase flow, because US APWR is

8 automatic pump trip logic.

9 MEMBER BANERJEE: But you have a pump

10 rundown, right?

11 MR. TERAMAE: Yes.

12 MEMBER BANERJEE: Over how long does the

13 pump run down?

14 MR. TERAMAE: About 20 seconds from the

15 ESF signal, it shoots.

16 MEMBER BANERJEE: But the pump, does it

17 affect the blowdown peak?

18 MR. TERAMAE: Yes.

19 MEMBER BANERJEE: Yes, it does, right?

20 MR. TERAMAE: Yes.

21 MEMBER BANERJEE: So -- but the blowdown

22 peak is not affected by your two-phase flow model in

23 the pump?

24 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

1 Fahrenheit occurs at the elevation 10.4 feet. This
2 time, after this time the advanced accumulator
3 injected is already activated.
4 The whole core is quenched at 220
5 seconds. We can see that these figures that
6 transient during the LBLOCA for US APWR is very
7 similar to the typical kind for PWRs.
8 MEMBER BANERJEE: But your clad PCT
9 temperature is lower. It seems lower to me.
10 MR. TERAMAE: Yes.
11 MEMBER BANERJEE: Why?
12 MR. TERAMAE: This case is a reference
13 case. I used the -- many permit I use the aluminum
14 except that active types on the power relative to
15 parameters.
16 The core power is 100 percent and if
17 there is edge on each assembly, there possibly is a
18 tech spec value. But other parameters use nominal
19 parameter. That is a reason for the low PCT.
20 MEMBER BANERJEE: But in your ASTRUM
21 calculations later, what is the range of the peak
22 clad temperature above this? Is there a number of
23 runs?
24 MR. TERAMAE: There are limiting cases

1 above this case but the range is very --

2 MEMBER BANERJEE: Wide.

3 MR. TERAMAE: wide.

4 MEMBER BANERJEE: Will you speak of this

5 later? Tomorrow. Because we would like to know

6 what effects the reference case, the most, what

7 parameters.

8 MR. TERAMAE: Can I move on? Let me

9 summarize our presentation. The WCOBRA/TRAC code

10 with ASTRUM methodology is used for -- being used

11 for the US APWR LBLOCA analysis.

12 Second, the new or improved US APWR

13 features, such as a longer core, advanced

14 accumulator, direct vessel injection and neutron

15 reflector have been evaluated for the code

16 applicability.

17 Third, the applicability of WCOBRA/TRAC

18 M1.0 code and ASTRUM methodology to US APWR has been

19 examined and confirmed based on the CSAU approach.

20 Fourth, WCOBRA/TRAC M1.0 has been

21 applied to a sample US APWR plant analysis and its

22 capability to simulate the US APWR LBLOCA transient

23 was demonstrated.

24 Last, the US APWR LBLOCA analysis is

1 reported in DCD Section 15.6.5. Thank you very much
2 for your attention.
3 CHAIR STETKAR: Any other questions for
4 MHI at this time? If not, thank you very, very
5 much. Appreciate it. And we'll hear from the
6 staff.
7 MEMBER BANERJEE: So, John, we are
8 putting a lot of things back to tomorrow.
9 CHAIR STETKAR: I'm aware of that.
10 Tomorrow --
11 MEMBER BANERJEE: I mean, maybe it's
12 worth having a closed session today.
13 CHAIR STETKAR: That's, well, we'll see
14 how -- there's a lot of material to cover today
15 also. It's -- becoming a bit concerned about that,
16 so. Let's go through the staff's presentation on
17 this, and before we break for lunch, we'll talk a
18 little bit --
19 MEMBER BANERJEE: Right.
20 CHAIR STETKAR: I think about strategy
21 for the next 24 hours.
22 MEMBER BANERJEE: It's pretty hard to
23 take a two-day meeting and make it into one and a
24 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

1 to provide an overview of the large-break LOCA
2 safety evaluation applicability to the US APWR.
3 This presentation is slightly shorter
4 because the code that was utilized here,
5 WCOBRA/TRAC, the derivative used here for
6 WCOBRA/TRAC M1.0, did not -- was previously approved
7 code, if you will.
8 We have a different cast of reviewers
9 here. With me is Fred Forsaty, the lead reviewer,
10 and as well as the ISL support staff, the names are
11 up there, and of course Jeff Ciocco being the lead
12 PM for this review.
13 And that's it for my opening, and what
14 I'm going to do, Fred, you can take over for the
15 technical side of the review now, the presentation.
16 MR. FORSATY: Thank you. It is still
17 morning, right? Good morning. My name is Fred
18 Forsaty. I work for the agency, NRO. We have come
19 to you previously for similar topic, large-break
20 LOCA topical report for the previous design, so we
21 have some kind of experience doing this review.
22 We, in the initial design review, we had
23 some concern about thermal conductivity and the
24 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

1 other codes that we have in house, and we are
2 convinced at this point that we don't have any
3 thermal conductivity issue.
4 The FINE code is still under review, but
5 as fa as I am concerned, we should not have any
6 thermal conductivity issue that could impact the
7 PCT.
8 As a partner of the review, we looked at
9 the CSAU-related issues that we have had previously,
10 scaling issues. We looked for some detail under
11 WCOBRA/TRAC Mitsubishi version, especially with the
12 features that are unique to the design, especially
13 the accumulator, neutron reflector, and we also
14 looked at the thermal conductivity issue these are
15 the three areas we focused on.
16 MEMBER BANERJEE: So you looked at the
17 reflector of the heat sink as well?
18 MR. FORSATY: To some extent.
19 MEMBER BANERJEE: Heat source, sorry.
20 MR. FORSATY: Yes. Again, I am going to
21 say the same thing. I am going to repeat whatever
22 Jeff said. You know, the reflector does not impact
23 the analysis or doesn't have a major impact on the
24 heat source.

1 MEMBER BANERJEE: Okay.

2 MR. FORSATY: Our confirmatory run was

3 performed by ISL. They supported us. They used

4 RELAP5 and we had the opportunity to run the

5 WCOBRA/TRAC Mitsubishi version, and the idea behind

6 that was to basically have some understanding of the

7 new design, especially the accumulator design,

8 because using that -- the modeling that is in the

9 WCOBRA/TRAC Mitsubishi for accumulator, we took that

10 modeling, we incorporated that to RELAP code, the

11 MOD3.3.

12 And prior to that we ran the WCOBRA/TRAC

13 Mitsubishi version to be comfortable with the code

14 and the new design that is -- it is in the code

15 itself.

16 Again, --

17 MEMBER BANERJEE: So you used both

18 codes?

19 MR. FORSATY: We ran both codes.

20 MEMBER BANERJEE: And HOTSPOT, and what

21 did you do -- what did you use for the fuel code

22 yourself?

23 MR. FORSATY: Okay. That's a good

24 question. We are going to come to that. HOTSPOT is

1 basically --

2 MEMBER BANERJEE: It's their code,

3 right?

4 MR. FORSATY: Yes, it uses the boundary

5 conditions that are given to it by the WCOBRA/TRAC.

6 It takes all the boundary condition and provides --

7 it creates or calculates the fuel response to that,

8 basically. Did I answer your question?

9 MEMBER BANERJEE: Yes, so what we are

10 all concerned about all the time is the stored

11 energy.

12 MR. FORSATY: Of course.

13 MEMBER BANERJEE: In this case, of

14 course, since you are in a reflood peak situation,

15 it's less important than if you were in the blowdown

16 peak.

17 MR. FORSATY: That's right.

18 MEMBER BANERJEE: But if I understand

19 it, HOTSPOT is used to calculate the stored energy,

20 using some gap conductance source --

21 MR. FORSATY: Yes, use a thermal

22 conductivity, gap conductance, the temperature

23 profile and everything else, uses all the heat

24 transfer coefficients that are given by the

1 WCOBRA/TRAC, and it does its neutronic thermal
2 conductivities calculation.
3 MEMBER BANERJEE: Okay, and --
4 MR. FORSATY: It calculates the
5 temperature profile and picks the hot rod
6 temperature and that would be the PCT that would be
7 used for that --
8 MEMBER BANERJEE: Right, now, HOTSPOT
9 incorporates the thermal conductivity out of Haldon?
10 MR. FORSATY: That's right. It has the
11 latest Haldon --
12 MEMBER BANERJEE: Haldon data. Is that
13 your understanding too?
14 MEMBER REMPE: I've not seen anything on
15 HOTSPOT, so I don't --
16 MEMBER BANERJEE: No, but you've read
17 about it, but you've --
18 MEMBER REMPE: I haven't read anything
19 about HOTSPOT. They said they were going to have a
20 meeting later. So I don't know what HOTSPOT does or
21 doesn't do.
22 MEMBER BANERJEE: So you assume right
23 now that it takes into account TCD? We'll look into
24 it later.

1 MR. FORSATY: The big issue previously
2 was thermal conductivity as a function of burnup was
3 modeled only for the first 10, 15 percent of the
4 core life. After a certain time, as a function of
5 burnup, thermal conductivity would not change. They
6 would just keep it constant.
7 So as a result of that, there was no
8 thermal conductivity degradation model previously.
9 But in the FINE code, they have the latest data, so
10 all the latest data is incorporated in calculating
11 the thermal conductivity. You see the thermal
12 conductivity degradation, and they ran some sample
13 cases in-house to justify that, you know, there is
14 no thermal conductivity issue.
15 MEMBER REMPE: So you ran FRAPCON
16 in-house?
17 MR. FORSATY: Yes, we did.
18 MEMBER REMPE: And then did you put some
19 sort of correction into RELAP 3.3 to account for
20 what you learned with the updated FRAPCON?
21 MR. FORSATY: No, we didn't make any
22 correction to that. We just ran the WCOBRA/TRAC and
23 compared its result with what we had in-house
24 available to us, for the latest data.

1 MEMBER BANERJEE: Let me understand, and
2 maybe -- Joy, sorry for interrupting -- but when you
3 put the calculations and did the RELAP5
4 calculations, you used the stored energy that came
5 out of HOTSPOT, or -- because I guess she is trying
6 to understand, which I am trying to, for your
7 confirmatory calculations, the WCT HOTSPOT we
8 understand. That's a -- what did you do for the
9 fuel with RELAP5?
10 MR. FORSATY: I am going to give you my
11 opinion and I am going to ask Dave, the person who
12 ran the code. I believe the information that was in
13 the WCOBRA/TRAC was extracted.
14 We compared it with FRAPCON. And all
15 that information was used to create the database for
16 the RELAP5 to run the analysis. Dave, can you help
17 me out?
18 MR. CARAHER: My name is David Caraher.
19 I work for ISL. And on these calculations that
20 were done fo RELAP5 calculations, then what we did
21 was we would match the center line temperature and
22 we would put in a gap -- vary the gap conductance so
23 we had the same stored energy.
24 We did not do anything where we ran a

1 fuels code. That's -- when we ran WCOBRA/TRAC, we
2 generally looked at the reference case, which
3 doesn't have a HOTSPOT calculation, but we did
4 repeat their calculations for their limiting cases,
5 which means we ran HOTSPOT.
6 But we never did a corresponding
7 transfer of data from RELAP5 to FRAPCON.
8 MEMBER BANERJEE: So, let me give it
9 back to you to see if I understood. For some of the
10 limiting cases, like loop or whatever there were,
11 you used a HOTSPOT, full HOTSPOT calculation or the
12 results of it, with WCOBRA/TRAC to do some
13 confirmatory tests.
14 With the RELAP5, you used the same
15 center line temperatures and gap conductances to get
16 roughly the same -- not exactly, but roughly the
17 same stored energy, and ran those. Is that how I
18 understood it, or did I get it wrong?
19 MR. CARAHER: No, I think you got it
20 correct.
21 MEMBER BANERJEE: Okay. Do you
22 understand what they did, Joy?
23 MEMBER REMPE: I think I understand what
24 they did, but then is it really an audit, an

1 independent audit calculation?

2 MEMBER BANERJEE: Well, it depends what

3 the audit is about, I guess. If it is about the

4 thermal hydraulics, they have got roughly the same

5 stored energy, right? So --

6 MR. CARAHER: Yes, and remember, the way

7 this methodology works is they run a reference case.

8 That reference case does not involve HOTSPOTs.

9 That's where our confirmatory calculations ended as

10 far as comparing RELAP5 and WCOBRA/TRAC results.

11 MEMBER REMPE: Okay.

12 MEMBER BANERJEE: Let's go over that

13 again. The reference case does not involve HOTSPOT.

14 MR. CARAHER: That's correct.

15 MEMBER BANERJEE: But it has to have an

16 estimate of the stored energy, right? How did they

17 arrive at that then?

18 MR. CARAHER: How do they arrive at

19 that? Well, they, I mean, steady state code, with

20 the powers and thermal -- just the thermal

21 properties would give you that when you initialized

22 the code.

23 MEMBER BANERJEE: Yes, but is there

24 degradation of thermal conductivity or not, in that

1 calculation?

2 MR. CARAHER: You'd have to ask

3 Mitsubishi to be sure in that case, but as far as I

4 know, no.

5 MEMBER BANERJEE: Right, this is --

6 MR. CARAHER: It's a, I mean, it's a

7 beginning of life, the calculation, the reference

8 case.

9 MEMBER BANERJEE: Yes, okay. So the

10 reference case is -- that I didn't know, was a

11 beginning of life calculation. Okay. If that's the

12 case, then it may not matter. Right.

13 Does that make sense? I mean, you've

14 got the hottest fuel, is the beginning of life,

15 right? So I suppose that --

16 MEMBER SHACK: But that's true for other

17 people that see an effect of TCD, so presumably it

18 may shift the worst case a little bit.

19 MEMBER BANERJEE: We have to think this

20 through. All right, we will revert to you and we

21 need to -- because it, in some of the calculations

22 we have seen, particularly where the blowdown peak

23 is important, TCD has a big effect on the results,

24 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

1 that much.

2 But again, it's an open item and it

3 would update the SE based on the information we are

4 going to get later on from the applicant.

5 MEMBER BANERJEE: Do you agree with the

6 applicant that the pump model doesn't have much

7 effect on the PCT?

8 MR. FORSATY: It's a good question. We

9 asked a question related to that. They asked them,

10 it's not exactly the same question but probably the

11 outcome would have the same result, they asked them

12 to run the -- first the question, the LOOP, if they

13 should run, be allowed to have LOOP all the time or

14 not.

15 And my recollection is we asked them,

16 some time ago, to run the LOCA runs with LOOP and

17 without LOOP, we didn't see much of an impact in the

18 final result. We looked at that.

19 Again, we just ran through the SE to see

20 if we asked the applicant to run -- have LOOP all

21 the time as a part of the analysis. We couldn't

22 find that. But I remember that we asked that

23 question, and LOOP and without LOOP, didn't have any

24 impact on this. Does that help?

1 MEMBER BANERJEE: Yes.

2 CHAIR STETKAR: Fred, when they ran the

3 no LOOP, they ran it really no LOOP, they didn't --

4 MR. FORSATY: I don't --

5 CHAIR STETKAR: They didn't just look at

6 it through the, whatever other --

7 MR. FORSATY: No, I believe --

8 CHAIR STETKAR: delay times that they

9 put --

10 MR. FORSATY: Yes, yes, they had the

11 similar question in other designs, so that was one

12 of the questions we asked, and they ran it for the

13 whole --

14 CHAIR STETKAR: And they ran -- okay.

15 MR. FORSATY: Okay, next slide. Again,

16 the US APWR large-break LOCA is a best methodology,

17 it uses the ASTRUM methodology. The applicant is

18 using the WCOBRA/TRAC Mitsubishi to do the nuclear

19 sink supply system calculation.

20 As I mentioned before, HOTSPOT is used

21 through the neutronic side of the code. And we

22 asked the applicant, the applicant basically ran 124

23 cases and it's not mentioned here, we had two

24 questions on the cases, the range of the IRWST and

1 the accumulative temperature was not realistic. If
2 I remember right, the range of the temperature for
3 these two systems were somewhere between 50 degree
4 Fahrenheit to 130, and that's not a realistic
5 temperature for the containment at 100 percent
6 power.
7 That -- Mitsubishi corrected that and I
8 believe the new number was 90 degrees to 130. The
9 impact was 30, 40 degree Fahrenheit on PCT only.
10 MEMBER BANERJEE: So, let me ask on this
11 one. We saw that for some cases the check valve
12 comes in, right?
13 MR. FORSATY: For the accumulator?
14 MEMBER BANERJEE: For the accumulator.
15 How sensitive -- the first question is, does it
16 activate before or after the peak in the
17 temperature? The peak , if I remember the reference
18 case, occurred around 100 seconds or so, correct?
19 MR. FORSATY: That's right.
20 MEMBER BANERJEE: And if I remember, the
21 accumulator was injecting very strongly, still, and
22 it was only after that, that when it started to
23 fall, that the check valve came in.
24 But that was for this large-break LOCA

1 or was that for the small-break LOCA?

2 MR. FORSATY: I think the picture you

3 saw was for a small-break LOCA.

4 MEMBER BANERJEE: Right, the picture was

5 for -- but does it give a similar behavior for the

6 large break?

7 MR. FORSATY: Well, it's -- that's a

8 good question. I think that is specific to a small-

9 break LOCA because you don't have that the high

10 pressure drop for a small break compared to a large

11 break, so you would see that closure of the valve.

12 But also for a small break, it doesn't

13 happen for a large break. So it basically acts like

14 a regular accumulator -- it has a good benefit also

15 that some of the water is still trapped in the

16 accumulator, in the baby accumulator after

17 vortexing, so to me, that helps prevent the nitrogen

18 injection --

19 MEMBER BANERJEE: Going in, yes.

20 MR. FORSATY: So I think that's a

21 benefit. And when it comes to small-break LOCA, you

22 know, closure of the valve would take probably 10

23 seconds to open, 10 seconds to close, so that's

24 about 20 seconds, so you are not going to be cycling

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.

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.

1 MR. FORSATY: So again, compared to the
2 previous design, you know, they are running 124
3 cases, not 59 cases. The thermal conductivity,
4 should that not appear to be an issue here, is not
5 an issue here.
6 We --
7 MEMBER BANERJEE: So, 124 is the normal
8 ASTRUM --
9 MR. FORSATY: That's right. That's what
10 it should be. And then the applicant agreed and
11 corrected the temperature range for the accumulator
12 and the IRWST temperature to make sure that the
13 range is reasonable and realistic.
14 Again, these are some of the features of
15 the APWR design. You've got the new accumulator
16 design. And compared to a Westinghouse four-loop
17 plant, we have basically higher or increased HPI
18 flow. We have significantly bigger or higher heat
19 transfer area for the steam generators. It's almost
20 twice as much.
21 We have larger pressurizer for 10 foot
22 fuel. They have a linear portion in there. We have
23 heavy neutron reflectors with the smaller holes that
24 would cool a reflector, and of course, larger core,

1 larger volume, more fuel, more power.

2 Next please.

3 MEMBER BANERJEE: The volumetric power

4 per unit volume is the same more or less, right, or

5 lower?

6 MR. FORSATY: No, it is -- it's not the

7 same. We're going to get to that. We basically --

8 we have that information later. We'll give it to

9 you and if you need more data --

10 MEMBER BANERJEE: Right.

11 MR. FORSATY: This slide, you have seen

12 it many times. The PCT is -- plenty of margin for

13 the PCT, compared to 2200. The -- no issues with

14 the cladding oxidation, and under this LOCA

15 condition, which we do not have any issues with the

16 coolable geometry, there is no deformation of fuel,

17 and assurance of long-term decay heat removal, I

18 think we are going to discuss that as a part of

19 15.65 in the afternoon.

20 We have two open issues, three RAIs, the

21 open issues related to a small-break LOCA

22 criticality, boron dilution, one RAI on that, and

23 then we have boron precipitation, we are questioning

24 that and that's an open issue.

1 MEMBER BANERJEE: Is that -- we always
2 have these issues with how much mixing we allow and
3 things like this. I mean, we can visit it when we
4 talk in the afternoon.
5 MR. FORSATY: Yes, we will -- but
6 actually --
7 MEMBER BANERJEE: Ideally we would want
8 to know the assumptions that are going into this.
9 MR. FORSATY: Yes. This afternoon is,
10 not closed session, but one thing I can tell you,
11 they did some testing on that and mixing, recently,
12 and the result is under review. But they did a
13 complete testing.
14 MEMBER BANERJEE: Did you guys ever get
15 in touch with Len Ward on this?
16 MR. FORSATY: Len Ward? I don't
17 remember. On which issue, on boron precipitation or
18 mixing?
19 MEMBER BANERJEE: Both.
20 MR. FORSATY: We are closely working
21 together again as a part of consistency, we are, you
22 know, if not daily then weekly communications with
23 the NRR, and we are -- on this specific issue, we
24 are working very closely --

1 MEMBER BANERJEE: With NRR.

2 MR. FORSATY: with NRR. They are going

3 to help us to run some confirmatory runs and you

4 probably already know, our branch chiefs, they meet

5 every week, once a week and they discuss all those

6 issues.

7 MEMBER BANERJEE: Okay.

8 MR. FORSATY: Yes, Len is very much

9 involved.

10 MEMBER BANERJEE: Yes, because we have

11 some long-term running concerns about these mixing

12 and --

13 MR. FORSATY: Boron precipitation. If

14 you look at those two RAIs, we put it on the table

15 two years ago, so that's in line with your concern.

16 So --

17 MEMBER BANERJEE: Okay.

18 MR. FORSATY: That will be addressed.

19 So if you can go to the next one, we looked at,

20 again, we put more time on this than we did with the

21 other design. We had more time. We looked at the

22 PIRTs for the blowdown, refill and reflood. We

23 looked at the -- especially the highly ranked

24 variables and again, we didn't see any issue with

1 their choice.

2 MEMBER BANERJEE: So this is absolutely

3 standard, right?

4 MR. FORSATY: That's right, I mean we

5 followed it to the extent possible. Again, as a

6 part of that we basically identified that the

7 generic issue with the temperature range for a few

8 of the systems needed to be corrected and it was

9 correct.

10 MEMBER BANERJEE: They have of course

11 DVI injection in this system, right?

12 MR. FORSATY: Yes.

13 MEMBER BANERJEE: A little bit different

14 from cold leg injection. It didn't raise any issues

15 in the downcomer and -- if I remember they have a

16 design where they try to lead the flow downwards

17 into the downcomer, right?

18 MR. FORSATY: The DVI injection is right

19 -- in case inject the core in the downcomer and,

20 Dave, if you can help answer this question.

21 MR. CARAHER: They have some pads

22 located on the --

23 MEMBER BANERJEE: You have to come to

24 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

1 difference, and try to understand what consideration
2 was given to it, and then maybe it didn't introduce
3 any new issues.
4 MR. FORSATY: I can get back to you on
5 that tomorrow, but --
6 MEMBER BANERJEE: Okay.
7 MR. FORSATY: I forgot at the beginning
8 to give you a little bit of history of my -- but I
9 will do it at the end, how about that?
10 MEMBER BANERJEE: You don't have to.
11 It's okay.
12 MR. FORSATY: Okay. We looked at
13 actually the early -- I'm sorry?
14 MEMBER BANERJEE: The last point is what
15 I was --
16 MR. FORSATY: Yes. Yes. At the early
17 part of this review, we -- the staff discussed the
18 plan on how we are going to do our review, and at
19 that time the TRACE codes didn't have the
20 accumulator model. So what they had to do is, we
21 got the WCOBRA/TRAC Mitsubishi model, we ran some
22 cases to make sure we get familiar with that, and
23 then we borrowed the accumulator model once we were
24 kind of comfortable with that. We used that for the

1 RELAP5 M3.3 and that's the version we used to do our
2 confirmatory runs.
3 And again, the basic differences here in
4 this -- for the WCOBRA/TRAC, and the WCOBRA/TRAC
5 Mitsubishi model is basically the core length which
6 is for 10-foot fuel, should not have that much of an
7 impact but then accumulator design and neutron
8 reflector and DVI injection, they are looked at as a
9 part of our confirmatory runs.
10 As a part of this review, initially,
11 before we do our actual analysis, we identified an
12 issue with the code and that was the gravitiational
13 head was calculated --
14 MEMBER BANERJEE: What was wrong with
15 that?
16 MR. FORSATY: I knew you were going to
17 ask me. The gravitational head is basically a
18 simple calculation, right, rho g, delta g, I think
19 at some point some of the history of the calculation
20 was not reset, and it was not properly calculated.
21 But, again I'm going to call Dave to
22 make sure it was -- confirm what I just told you.
23 We used that -- we communicated that issue with the
24 applicant. The issue was resolved.

1 This is -- my understanding this is
2 something that existed prior to the code WCOBRA/TRAC
3 to be modified to Mitsubishi. I have no guarantee
4 on that. But again, that issue was communicated to
5 applicant. It was resolved and the impact was
6 minor.
7 MR. CARAHER: David Caraher with ISL
8 again. The first thing we did when we had -- took
9 WCOBRA/TRAC and started doing confirmatory analysis,
10 is just do a null calculation where you turn off the
11 core power and the RCPs and see if the -- all the
12 LOOP flows should go to zero. They didn't. So then
13 we just kicked it back to MHI and asked why.
14 And after their analysis, they found
15 that there was, when they did all the momentum
16 source terms for the RCPs, there was term missing or
17 an extra term added that kept the gravity head.
18 MEMBER BANERJEE: So how could you have
19 gravity head if there was no temperature difference?
20 MR. CARAHER: That's the point.
21 MEMBER BANERJEE: Right, but did they --
22 MR. CARAHER: They had a momentum source
23 term that, I mean, they had the DZ, you would think,
24 in this case, right? rho g delta z. The rhos are

1 the same.

2 MEMBER BANERJEE: Could we get a little

3 bit more -- what was the term that was causing this

4 issue?

5 MR. CARAHER: I can't recall, but we

6 have the coding, I mean, we looked and reviewed

7 their coding when they changed it, and how it was

8 changed.

9 MEMBER BANERJEE: No, I know. But was

10 there a term added or what was it -- what was the

11 nature of this term, could you tell? We can ask the

12 applicant.

13 MR. CARAHER: Yes I think it would be

14 better to ask the applicant.

15 MEMBER BANERJEE: Yes, could somebody

16 from the applicant tell us a little bit more about

17 this term which caused flow in the absence of

18 gravity head?

19 MR. SPRENGEL: Yes, this is Ryan

20 Sprengel. Give us just one second to confirm if we

21 can provide that information now or if we will

22 provide it later today.

23 MEMBER BANERJEE: Sure. If it was

24 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

1 condition, if I remember right, it was like 30 or 35
2 BTU per hour square foot, and during the calculation
3 we noticed that number exceeds.
4 So, that error was identified,
5 communicated to the applicant and it was resolved.
6 And by the way, all these errors also have been
7 communicated through NRR. They are very much aware
8 of this and they have taken all the necessary action
9 to make sure that if that exists in the previous
10 approvals, it would be addressed.
11 MEMBER BANERJEE: So, the -- let me try
12 to understand. The reflood heat transfer coefficient
13 could go beyond -- what was the upper limit? Was
14 there -- I'm just trying to recall, is there a set
15 upper limit on this, or what?
16 MR. FORSATY: Yes, there is a set upper
17 limit.
18 MEMBER BANERJEE: Is that an Appendix K
19 limit, or what?
20 MR. FORSATY: No --
21 MEMBER BANERJEE: Or is it =- can you
22 just enlighten me as to what that is --
23 MR. FORSATY: The upper limit is
24 basically -- is an upper limit that after that is

1 just basically you are going to a different phase,
2 it's not applicable to the phase that you are
3 analyzing, and that basically would not be
4 realistic.
5 So that upper limit was set up some time
6 ago, a long time ago, to be -- not to be violated.
7 And then ISL ran the code that discovered that upper
8 limit.
9 MEMBER BANERJEE: Is this in some reg
10 guide or something, that this is going to be an
11 upper limit?
12 MR. FORSATY: It is in some --
13 MR. CARAHER: This is David Caraher, ISL
14 again. The way it was developed is that when
15 Westinghouse was -- started using this methodology,
16 it was found that HOTSPOT could predict numbers that
17 -- the staff asked at that time, well, you can
18 predict numbers that are higher than what we ever
19 see, or may see in Flecht-Seaset tests.
20 So they went, the staff requested
21 Westinghouse to put an upper limit and that upper
22 limit was based upon the best -- the highest value
23 they might see during the reflooding phase, not
24 during, you know, some really insurge phase, some

1 steady state value.

2 MEMBER BANERJEE: So this was in the

3 pre-cooling phase, in the precooling region, above

4 the quench front?

5 MR. CARAHER: Yes.

6 MEMBER BANERJEE: So it was in dispersed

7 flow --

8 MR. CARAHER: Yes, that's right. And so

9 the staff asked, can you give me -- identify a

10 maximum value and Westinghouse went back to the

11 Flecht-Seaset test, and said, okay, we'll fix a

12 maximum value, here it is, here's the number.

13 And now, HOTSPOT will always -- it will

14 always have a number below that until it quenches,

15 okay?

16 MEMBER BANERJEE: Okay. So there's some

17 limit based on experimental data.

18 MR. CARAHER: Yes.

19 MEMBER BANERJEE: Fine, thanks. I

20 wasn't aware of that actually. Okay. Thank you.

21 CHAIR STETKAR: See what happens when

22 you come to these meetings?

23 MEMBER BANERJEE: Yes, you learn

24 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

1 quickly to the steam generator, so --

2 MEMBER BANERJEE: So from the point of

3 view of steam binding, there are no issues that can

4 occur due to this?

5 MR. FORSATY: I do not see any issue.

6 We spent quite a lot of time on that, and --

7 MEMBER BANERJEE: Somebody looked at it

8 carefully?

9 MR. FORSATY: Yes.

10 MEMBER BANERJEE: Right.

11 MR. FORSATY: And you have some backup

12 slides and maybe tomorrow we can show you that to

13 show you that --

14 MEMBER BANERJEE: So let me try to

15 understand this again. The hot leg is 12 percent

16 larger in area, but the power is 30 percent larger.

17 MR. FORSATY: That's correct.

18 MEMBER BANERJEE: What about the steam

19 generator flow area? Is that --

20 MR. FORSATY: The steam generator area -

21 - the area was, I think it's -- Dave, correct me if

22 I'm wrong -- I think it's like almost twice as much

23 area.

24 MEMBER BANERJEE: No, the area may be,

1 but the flow area. Is it twice as much?

2 MR. FORSATY: No.

3 MEMBER BANERJEE: No. What is the ratio

4 of the flow area?

5 MR. FORSATY: It should be 30 percent.

6 MEMBER BANERJEE: Is it 30 percent?

7 Does it go in proportion or -- we are simply trying

8 to find if it's in proportion.

9 MR. DONOGHUE: This is Joe Donoghue.

10 Does Mitsubishi have that information?

11 MR. HAMAMOTO: This is Hiroshi Hamamoto.

12 Sorry, we don't have it now. We need to confirm.

13 MEMBER BANERJEE: Okay, so Mitsubishi

14 agree with the 12 percent, because that was a

15 question that arose -- to them as well, we asked

16 them how much was the hot leg increased compared to

17 the power increase.

18 So, it's come as 12 percent now, so we

19 need to confirm --

20 MR. FORSATY: That's the number I have,

21 12 percent, and the steaming rate for the hot leg

22 velocity is basically 15 percent more than it. The

23 steaming rate is basically a little bit more than

24 the flow area for the hot legs, about 15 percent

1 higher compared to what --

2 MEMBER BANERJEE: Why is the steaming

3 ratio -- why isn't it 30 percent higher based on the

4 power? Is there a reason?

5 MR. FORSATY: Why --

6 MEMBER BANERJEE: The power is 30

7 percent higher.

8 MR. FORSATY: That's right.

9 MEMBER BANERJEE: So why would the

10 steaming rate be only 15 percent higher?

11 MR. FORSATY: I think that's a good

12 question. It should be more on the, you know, the

13 pressure drop differential and --

14 MEMBER BANERJEE: Yes, it probably is

15 related to a lot of things, but you know --

16 MR. FORSATY: And that tells you that

17 the steam generator flow area is probably not 30

18 percent.

19 MEMBER BANERJEE: Yes.

20 MR. FORSATY: So it should be within the

21 range of 15 to 20 percent higher and not 30 percent.

22 CHAIR STETKAR: Why don't we wait? I

23 think MHI has it, you know, as an action item to

24 come back with --

1 MEMBER BANERJEE: Okay.

2 CHAIR STETKAR: with those areas and --

3 MR. FORSATY: But again, when it comes

4 to large-break LOCA, we didn't see --

5 MEMBER BANERJEE: It doesn't matter too

6 much.

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

8 doesn't matter.

9 MEMBER BANERJEE: It's just the steam

10 binding --

11 MR. FORSATY: We don't have that issue.

12 Okay. And so you went through this slide. Next

13 slide please. Overall, our review indicates that in

14 our WCOBRA/TRAC, the reference base, you know, I

15 believe with our confirmatory runs and the

16 differential was maybe 10, 20 degree Fahrenheit.

17 And we did some sensitivity runs, we

18 didn't find any issues that could impact the

19 acceptance of the code. Again, I just want to

20 emphasize the fact that WCOBRA/TRAC is approved. We

21 focused on the differential, on the changes, and we

22 didn't identify any issues that would impact the

23 results. And there's plenty of margin on the PCT.

24 Next slide please.

1 And the conclusion is that the
2 Mitsubishi large-break LOCA methodology is
3 acceptable for calculating peak clad temperature,
4 hydrogen generation, fuel cladding surface oxidation
5 for the US APWR, and the PCT that we have today is
6 close to 1800 degree Fahrenheit, which, again, has a
7 considerable margin, 400 degree.
8 A little background on me. My name is
9 Fred Forsaty. I have been the agency five, six
10 years, worked in the private industry at numerous
11 facilities, at Beaver Valley, Niagara, Mohawk,
12 Yankee Atomic, VC Cook, built the simulator for
13 Arizona Power, I didn't see Jeff there when I was
14 working. I built a simulator for Perry plant and I
15 have been here six years working on the new reactor
16 site.
17 MEMBER BANERJEE: So, thank you. Just
18 wanted to ask you a question about the PCT. I'm
19 always interested in the physical reasons for this,
20 and I assume that what is happening here is they
21 have plenty of HHS, high head safety injection,
22 which is why they shift out of the blowdown peak
23 into the reflood peak, which is why you're getting
24 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 --

1 MR. FORSATY: Also, the good thing is
2 there is no nitrogen injection because in some other
3 designs we made them to take the nitrogen injection
4 out to see what the impact would be on both peaks,
5 to make sure they are as well as possible.
6 And then, again, for large-break LOCA
7 you are basically, accumulator, it's a normal
8 accumulator when it comes to its function.
9 MEMBER BANERJEE: Yes, but really, why
10 does the blowdown peak come so low compared to the
11 reflood peak here, at least the reference case we
12 saw, that they showed us, the blowdown peak was
13 quite a bit lower than the reflood peak.
14 So there has to be a good physical
15 explanation for that.
16 MR. FORSATY: Well, the stored energy
17 has to, you know, get a little bit more time to --
18 MEMBER BANERJEE: Get out for some
19 reason.
20 MR. FORSATY: so that it's free to get
21 out, and so that, and the fact that, you know, they
22 have got decay heat, they have got to store energy
23 and al of these things that would tend to, you know,
24 provide a little bit more time for the second peak

1 to occur.

2 MEMBER BANERJEE: Yes. Well let's walk

3 through the processes, you know, the physics of this

4 later on.

5 MR. FORSATY: To make sure I understand

6 what my actions are, did somebody with you summarize

7 this, or when they come back tomorrow, we know who

8 is going to come back with some answers?

9 CHAIR STETKAR: Let's -- let me first go

10 -- make sure the -- any other members have any

11 questions for the staff on what we just heard? Joy?

12 MEMBER REMPE: Actually, more of a

13 question for you on process. At the beginning of

14 this you discussed the fact, and throughout this

15 morning we have heard that there's several points

16 that are outstanding, about the accumulator model,

17 the steam generators --

18 CHAIR STETKAR: That's, that's -- I was

19 going to get to that. Does anybody have any

20 questions for the staff on what we just heard?

21 Okay. Now, in about 10 minutes -- now I

22 would like to address the process, because I'm not

23 real clear in my own mind where we should plan to go

24 over the next 24 hours.

1 What we do need to accomplish is I want
2 to be sure that we get through MHI's presentation on
3 all of Chapter 15, and we get through the staff's
4 SER on all of Chapter 15. We need to do that.
5 We also need to address the COL, which
6 is short, but we need to do that on Chapter 15. We
7 have time allocated tomorrow morning for closed
8 sessions on both the large and small LOCA
9 methodology.
10 My sense is that we probably do not have
11 enough time to do the types of in-depth analyses we
12 have been discussing this morning. Given the fact
13 that there are still outstanding questions about the
14 performance of the accumulator, the modeling of the
15 accumulator, we will need to revisit the LOCA
16 analysis at some time in the future.
17 We have a subcommittee meeting scheduled
18 in October. Right now it is slotted for two days,
19 so we have the two days allocated to it. The topics
20 of that meeting right now were for the non-LOCA
21 analysis methodology.
22 There's a topical report on that that we
23 are not discussing during this meeting because the
24 SER is not ready yet for that, and information on

1 the CFD models, at least, for the accumulators.
2 I don't have an answer about what --
3 either from Fred's question or Joy's question, what
4 we should plan to discuss tomorrow. I don't know
5 what the most efficient way to spend the remaining
6 time is, regarding the LOCA analysis, and if anybody
7 has any suggestions, I'd like to entertain them.
8 My sense is that we are going to have to
9 revisit the LOCA analysis in more detail in October.
10 By that time there will be better information about
11 the accumulator and people will have more time
12 available to prepare the information for the actual
13 modeling. But --
14 MR. FORSATY: When you say LOCA, you
15 mean small or large or both?
16 CHAIR STETKAR: I didn't distinguish,
17 did I?
18 MEMBER REMPE: It's not just the
19 accumulator. There's the thermal conductivity
20 degradation that's supposed to be reviewed, the
21 steam generator, so it could all affect the plant's
22 ability to transfer heat etcetera.
23 And so I'm just kind of -- I assume
24 someone is tracking this, but at some point you

1 might decide all of these analyses need to be redone
2 or some fraction of them needs to be redone, because
3 of a model not being --
4 CHAIR STETKAR: Well, I think at a
5 minimum, they will need to be redone to develop what
6 the temperatures are --
7 MEMBER REMPE: Right.
8 CHAIR STETKAR: for example. That's
9 certainly true. Whether -- I think the purpose of
10 this exchange, during this day and a half, is to
11 identify whether we have any fundamental concerns
12 about what we hear about the overall methodology
13 that may then factor into those re-analyses.
14 I mean that's my spin on what we are up
15 to, you know, in this day and a half session.
16 MR. FORSATY: When it came to large-
17 break LOCA I think the stored energy was properly
18 modeled, you know. Conservatively, then, based on
19 our previous experiences, I do not see that that
20 would be an issue, and if you want us to come back -
21 - if that would help I can come back to you with
22 some of the questions that is on the table for
23 large-break LOCA.
24 CHAIR STETKAR: Okay. Let's -- and I'm

1 clueless about the topic. And I'm old enough to
2 admit that. Sanjoy and Joy, because you are most
3 actively involved, perhaps you might want to think
4 over lunch and when we come back from lunch, if
5 there are specific kind of focus issues, questions,
6 that we can address in tomorrow morning's session,
7 if we can kind of elaborate those for either the
8 staff, if it's for the staff, or for MHI, so that
9 everybody is clear, you know, what we expect
10 tomorrow morning, and I suspect they will not have
11 enough time to do the detailed types of information
12 that you were asking for, Sanjoy. I think that's
13 something that we will need to revisit, probably in
14 October.
15 But if there are specific issues that
16 you want to -- topics, questions, to make sure that
17 either the staff, if it's for the staff, or for MHI,
18 so they can be prepared to come and give us those
19 answers in a closed session tomorrow.
20 MEMBER BANERJEE: I think that I have a
21 pretty good idea of the -- from these presentations
22 of the differences in phenomena that you encounter
23 compared to the conventional PWRs they are used to.
24 Nonetheless, there are effects which are

1 somewhat unusual that need to be thought about more
2 carefully and the physics behind them clarified.
3 For example, the shifting of the critical small
4 break from the usual four-inch small breaks to the
5 much larger sizes, you know, it's something we need
6 to get our own sort of thoughts around.
7 It may not be that they have to supply
8 more staff. It's simply that we need to do our
9 homework better, you know, on that, and try to
10 understand, and then you have much more pointed
11 questions.
12 So, because of that, for example, a
13 phenomenon like loop seal clearing may have a
14 different importance than it does conventionally in
15 a, let's say, a Westinghouse plant, or similarly,
16 reflux condensation may not be quite as important,
17 so you need to recalibrate your mind in some way and
18 that hasn't happened with my mind yet. So --
19 CHAIR STETKAR: The only reason I bring
20 it up is obviously the MHI in particular, you know,
21 has presentations prepared for tomorrow morning, and
22 unless there are very focused specific issues that
23 we need to absolutely have addressed, you know, I'd
24 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

1 for that, other than a warm fuzzy feeling that they
2 will clear more easily.
3 CHAIR STETKAR: Okay.
4 MEMBER BANERJEE: You know, they'd have
5 to be made more --
6 CHAIR STETKAR: So if MHI could at least
7 try to address that in the discussion of the small
8 LOCA --
9 MEMBER BANERJEE: Yes, because clearly
10 if they don't clear, and you have only one loop seal
11 cleared, you are going to get a pretty high
12 temperature for your larger small breaks.
13 CHAIR STETKAR: So that's one thing for
14 tomorrow, if you can keep, MHI, if you can keep that
15 --
16 MEMBER BANERJEE: So, clearly we are
17 going to go over the rationale for saying why loop
18 seals clear, why they don't clear, what sizes do
19 they clear. You know, this thing has got to be
20 exposed in its gory detail.
21 CHAIR STETKAR: Yes.
22 MEMBER BANERJEE: Okay. And similarly,
23 you know, I can see that refluxing is probably not
24 important here because you are going into larger

1 breaks so you are getting more heat out of the break
2 --
3 CHAIR STETKAR: Well, but everything you
4 are saying is that rationale of what -- why that
5 critical break size is --
6 MEMBER BANERJEE: All of this has to be
7 thought through very carefully.
8 CHAIR STETKAR: Right.
9 MR. SHUKLA: Dr. Banerjee, as a
10 background, when we were planning for this meeting,
11 there was some question about whether these topical
12 reports and their SERs should be presented to us or
13 not, so, because they are subject to be revised
14 based upon the advanced accumulator analysis.
15 MEMBER BANERJEE: Until the accumulators
16 are done --
17 CHAIR STETKAR: Well, but I mean, in
18 some sense that's details of the calculation. It's
19 not the, you know, a fundamental part of the
20 methodology of what they are doing.
21 MEMBER BANERJEE: Unless there's some
22 huge surprise due to this, I don't think we want to
23 rehash what we have already discussed for --
24 CHAIR STETKAR: Yes, but that's the

1 whole point of having, you know, these subcommittee
2 meetings at the time during the review process, is
3 that if there -- if we, as a subcommittee, had some
4 fundamental issue with the methodology, not
5 necessarily the specific results, because we know
6 they are going to change.
7 MEMBER BANERJEE: So we are very, very
8 concerned, and have been always, about the ability
9 of these codes to predict loop seal theory, you
10 know, so that's why the staff has always taken a
11 position to hold them from clearing, and see what
12 the effect is, and in this case, maybe there is more
13 reason to allow it, because the breaks are larger,
14 but we really need to go over this aspect of it in
15 detail.
16 CHAIR STETKAR: So, if nothing else,
17 then, for tomorrow, in terms of -- in terms of
18 organizing MHI's time, if you can at least focus a
19 bit on that issue for the small LOCA discussion, and
20 maybe reorganize your time a little bit to focus a
21 little more on the small LOCA, because I think that
22 is a bigger concern.
23 MEMBER BANERJEE: And I'm also sort of
24 concerned, not concerned, but would like to know

1 why, what the physical reason is, that the blowdown
2 peak is not dominant here --
3 CHAIR STETKAR: That's a large LOCA.
4 MEMBER BANERJEE: which it is in many
5 plants, compared to the reflood peak.
6 CHAIR STETKAR: And MHI, there's
7 something for your large LOCA discussion.
8 MEMBER REMPE: Is it worthwhile to spend
9 a little bit of time looking at -- well I looked at
10 the slides at the closed session tomorrow, and they
11 show nodalization diagrams that they have selected
12 and to discuss briefly those nodalization diagrams
13 and the basis for it?
14 CHAIR STETKAR: I mean, that's something
15 they -- you know --
16 MEMBER REMPE: that the method, it's not
17 a result.
18 CHAIR STETKAR: they are planning to do.
19 What I'm trying to do is identify --
20 MEMBER REMPE: But if we don't get
21 through the whole presentation, that's the part that
22 was of most -- is of interest to me, knowing that
23 the results can change.
24 CHAIR STETKAR: Okay. Okay.

1 MR. FORSATY: So, on the large-break
2 LOCA, to make it clear, any questions on methodology
3 that I can --
4 MEMBER BANERJEE: I don't think on the
5 methodology, really, trying to understand the
6 phenomena.
7 MR. FORSATY: Okay, I take that action,
8 and --
9 MEMBER BANERJEE: I mean that's the
10 phenomenon.
11 MR. FORSATY: I take that action. Any
12 other concerns that --
13 MEMBER BANERJEE: I think the
14 methodology -- we have agreed to this more or less.
15 MR. FORSATY: Okay.
16 MEMBER BANERJEE: You know. This is a
17 little bit larger size and longer core, more
18 reflector, but --
19 MR. FORSATY: I'll do my homework
20 tonight, try to have something for you tomorrow.
21 MEMBER BANERJEE: Yes, and let's make
22 sure that the DVI injection is something that the
23 methodology can handle.
24 MR. FORSATY: And the PIRT can handle.

1 Those are the two actions. Okay. Thank you.

2 CHAIR STETKAR: Anything else, any of

3 the members? If not, thank you and we will recess

4 until one o'clock.

5 (Whereupon, the meeting went off the record at 12:55

6 p.m. and resumed at 1:02 p.m.)

7 A-F-T-E-R-N-O-O-N S-E-S-S-I-O-N

8 (1:02 p.m.)

9 CHAIR STETKAR: Okay, we're back in

10 session. This afternoon we are going to hear about

11 all of the Chapter 15 transient and accident

12 analysis from both MHI and the staff, and we will

13 start off with MHI's presentation.

14 MR. MARUYAMA: Thank you. Good

15 afternoon. My name is Yuta Maruyama. We are

16 representing Mitsubishi Heavy Industries. Today we

17 will be discussing the US APWR DCD Chapter 15

18 transient and accident analysis.

19 We also have several technical experts

20 from MHI here today. The technical experts

21 represent non-LOCA, small-break LOCA, large-break

22 LOCA and dose sections.

23 This slide and the next slide provide a

24 list of acronyms used during our presentation that

1 may be a helpful reference for you today.

2 Chapter 15 consists of the analysis of

3 various anticipated operational occurrence, and

4 postulated accident event. The US APWR design is

5 very similar to the design of current operating PWRs

6 in the US and Japan. Therefore, there are no new

7 AOO or PA events that need to be defined

8 specifically for the US APWR.

9 Here is the summary of the main plant

10 parameters for the US APWR as compared to a typical

11 four-loop plant in the US. As you can see, the US

12 APWR has a larger core thermal power. The US APWR

13 also has a lower average heat rate, which provides

14 larger thermal margins. The US APWR also uses

15 14-foot fuel.

16 Most of the design features of the US

17 APWR are similar to operating PWRs in the US.

18 However, there are a few differences. This table

19 provides a list of the key design features for the

20 US APWR and how these features impact the safety

21 analysis. In most cases, these design features are

22 explicitly modeled in the safety analysis.

23 Chapter 15 is divided into nine sections

24 and one appendix. Each of these sections will be

1 discussed in detail during this presentation.
2 Section 15.0.0 of the DCD contains introductory
3 material. This section describes generic
4 information that is meant to be applicable to all
5 Chapter 15 events.
6 This information includes classification
7 of events as either AOOs or PAs, analysis
8 assumptions for initial conditions, reactor trip and
9 ESF signals, single failures, operator actions,
10 loss of offsite power, and so on.
11 These descriptions are very general.
12 For each event, any information that is event-
13 specific is discussed in the applicable section for
14 that event.
15 There is one open item in the Safety
16 Evaluation Report for section 15.0.0 related to the
17 revision of the reload evaluation methodology
18 technical report.
19 Section 15.0.2 of the DCD concerns the
20 methodology of the safety analysis. MHI's
21 methodology follows the regulatory guidance in SRP
22 Chapter 15. This section also discusses the
23 computer codes used in the safety analysis
24 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.

1 Doses are evaluated for the exclusion
2 area boundary, the outer boundary of the low
3 population zone, the main control room and the
4 technical support center.
5 This section describes the source term
6 assumptions, atmospheric dispersion factors, dose
7 conversion factors and airborne radioactivity
8 removal coefficient.
9 The details of a representative dose
10 analysis will be provided later in this
11 presentation, during the section 15.6 discussion.
12 CHAIR STETKAR: I had a question on the
13 dose analysis general information in 15.0.3, and
14 tell me if it's better to discuss it in 15.6. But
15 let me ask you.
16 There are two questions. One, they are
17 related to controlling the pH in the RWSP water
18 after an event. One question is something that we
19 had from an earlier meeting and that is why is it
20 necessary for the US APWR to operate with a 4,000
21 ppm boron concentration in the RWSP?
22 We asked that question during an earlier
23 meeting and we were told, well MHI would answer that
24 as part of the accident analysis. So I'm asking

1 that question again.

2 It's related -- I'd like to know the

3 answer to that question of why you need that boron,

4 but that is then related to the control of the pH in

5 the water during an event response, and the control

6 of the pH depends on the rate at which the sodium

7 tetraborate, whatever it is, NaTB, is dissolved.

8 In the accident analysis, it seems that

9 you assume that the buffer is dissolved over a 12-

10 hour period -- I believe I read that somewhere --

11 which requires 12 hours' continuous containment

12 spray operation.

13 And I had a question about whether or

14 not that's a realistic assumption considering how

15 your operators may be instructed to really operate

16 the containment sprays.

17 In many plants the operators are

18 instructed to operate the containment sprays only

19 until containment pressure is reduced below the

20 actuation set point and then to turn off sprays, and

21 that's not continuous operation over a 12-hour

22 period.

23 So I'm not sure whether you had the

24 answers to those questions now or whether you may

1 address them in 15.6, but I had those two questions,
2 and they are -- one is related to accident response,
3 but it has a feedback into controlling the pH in the
4 sump, because of the high boron concentration.
5 MR. MARUYAMA: So, but your question is
6 firstly why do we need 4,000 ppm --
7 CHAIR STETKAR: Yes, compared to most
8 other plants that are in the 2,000 to 2,500 ppm
9 range, what particular feature of the US APWR design
10 requires that much, much higher boron concentration.
11 MR. FUJISHIRO: Used to be the
12 Westinghouse plant and also operating plant in Japan
13 had boron concentration is 2,000 ppm.
14 CHAIR STETKAR: Right.
15 MR. FUJISHIRO: However, after the
16 United States and also in Japan eliminated boron
17 injection tank, at that time MHI increased the RWST
18 boron concentration 3,500 ppm or 4,000 ppm. It's -
19 -
20 CHAIR STETKAR: Well, but there are many
21 operating plants in the United States that
22 eliminated the boron injection tank and kept their
23 boron concentration at 2,000 to 2,200, 2,300 ppm,
24 without doubling it.

1 MR. FUJISHIRO: Yes, I know. And it
2 depends on the core characteristics in some plant in
3 Japanese domestic plant, MOX core or high burnup.
4 So we increased RWSP boron concentration.
5 CHAIR STETKAR: Well, what I'm trying to
6 understand is that, you know, it isn't always
7 necessarily true that more of boron is better for
8 everything. And I'm trying to understand whether --
9 whether there's a --
10 MR. FUJISHIRO: There is no such
11 critical issue in the APWR control than -- as
12 concern over the accidental, transient and
13 accidental release is the higher boron concentration
14 can put in the core more negative radioactivity.
15 CHAIR STETKAR: That's true, but it
16 increases the likelihood of boron precipitation with
17 a need for, for a hot leg recirc, and also it
18 increases the acidity in your RWSP, which means you
19 need to add much more buffer at some rate to
20 maintain a mutual or slightly basic, and you are
21 very, very slightly basic.
22 MR. FUJISHIRO: Yes, in Japanese
23 regulations, they are a little bit too strict to
24 getting back some critical after steam line break,

1 so --

2 CHAIR STETKAR: Well, but what I'm

3 interested in -- we are reviewing this design for

4 certification in the United States, and I'm simply

5 asking a technical argument of what's the basis for

6 that 4,000 ppm? In other words, do you need 4,000

7 ppm? If you do not need 4,000 ppm, what is -- what

8 is the minimum concentration that is necessary to

9 mitigate the spectrum of events for this particular

10 design, considering your core design, considering

11 your burnup, you know, all features of this design.

12 And I'm trying to understand that. I'd

13 like to understand that.

14 MR. FUJISHIRO: Steam line break, hot

15 zero power case, it -- I remember 1.5 something,

16 there is still a little bit too margin to safety

17 analysis limited. However, I don't know how much we

18 can make a deduction.

19 Maybe we can, but I don't know.

20 CHAIR STETKAR: Perhaps we could take

21 that as an open item, if you had some notion of what

22 margin you have in there. I mean, even if it was

23 3,000 ppm versus 4,000 ppm --

24 MR. FUJISHIRO: Optimization of RWST?

1 CHAIR STETKAR: Essentially that's what
2 I'm asking about, is if you have looked at that or
3 if you could give us some insight about --
4 recognizing it probably is a steam line break that's
5 your limiting event for boron concentration, but if
6 you could give us some sense of what -- how much
7 margin you do have with the 4,000 ppm, it would
8 help.
9 MR. FUJISHIRO: That is, we need to
10 basically -- we must calculate the DNBR of the steam
11 line break. It's based on the 3D calculation of the
12 core and also using some by power, the MARVEL, ANC
13 and the bypass, then. It takes time, sorry.
14 CHAIR STETKAR: Well, let's see, if you
15 can, perhaps we can get some information.
16 MR. MARUYAMA: Yes, we understand your
17 comment, so then as Fujishiro-san said, we confirm
18 that 4,000 ppm has enough margin in calculation of
19 steam line break analysis, so we don't know how --
20 CHAIR STETKAR: That's right. It's
21 clear that you do have enough margin. The question
22 is, the question is, can you reduce that and by
23 reducing that, provide additional margin for some
24 other functions, and particularly boron

1 precipitation and control of pH in the sump water.
2 MR. SCHMIDT: This is Jeff Schmidt from
3 the staff. One of the reasons maybe for their plant,
4 and I know it's where the plant I came from was, is
5 that the 4,000 ppm is set sometimes for refueling.
6 If you remove your control rods while you still have
7 fuel in the pot, to reach k = 0.95 you may have to
8 have a higher boron concentration.
9 You know, a lot of the Westinghouse
10 plants, for example, keep their control rods
11 disassembled and put in the fuel. So the 4,000 may
12 not be a safety analysis value, but may be a
13 refueling convenience if they remove their control
14 rods and store them next to the plant.
15 CHAIR STETKAR: That would be an answer,
16 if that's the answer. I'm just looking for, I
17 haven't seen it anywhere, and given the margins that
18 I have seen, and all the analyses, it's just not
19 clear to me why they need that much boron.
20 MR. SCHMIDT: Yes, I know for instance,
21 I came from Palo Verde and that was the reason why

1 they have 4,000 ppm still today, is because they --
2 CHAIR STETKAR: Palo Verde runs with
3 4,000? I didn't know that.
4 MR. SCHMIDT: Yes.
5 CHAIR STETKAR: Okay.
6 MR. SCHMIDT: And their k was 0.95.
7 That's what they --
8 CHAIR STETKAR: I didn't know that.
9 Thank you. The other part of my question was let's
10 suppose that 4,000 ppm is, you know, it's certainly
11 the value that's included in the design right now.
12 I believe, if I read the accident analysis, it says
13 that the buffer is dissolved over a 12-hour period,
14 and because the buffer relies on the containment
15 spray flow to flow through the baskets, that that
16 presumes the containment spray flow will be
17 maintained constant for 12 hours, which at least in
18 current operating plants that I'm familiar with, is
19 not consistent with the guidance that the operators
20 are typically given. Guidance that the operators
21 are typically given in emergency operating

1 procedures is to monitor containment pressure, and
2 when containment pressure goes below, either below
3 the initiation set point or with some margin, to
4 either reduce or completely stop containment spray,
5 reset it, and you know, if containment pressure
6 rises again, then to initiate spray.
7 I have no idea how -- I know you do not
8 have emergency operating procedures for the US APWR
9 yet. I read somewhere that emergency response
10 guidelines, at least ERGs were being developed. I
11 don't know whether they are available yet or whether
12 they have that level of detail.
13 But if the analyses are including debt
14 for operating containment spray for 12 hours, and
15 the operators really would not operate the
16 containment spray for 12 hours, I think we'd like to
17 understand how the analyses include credit for full
18 dissolution of the buffer in the sump water, and
19 whether or not those -- the analyses are consistent
20 with what the operator's guidance would actually be.
21 MR. HAMAMOTO: This is Hiroshi Hamamoto.

1 We assumed that 12 hours time. As you see, the
2 figure 6.3-10, and the line under 6.3-10, there is
3 cooperation with containment and the basket
4 configuration.
5 CHAIR STETKAR: Yes.
6 MR. HAMAMOTO: And also, 6.3-10 is a
7 spray that is part of the spray water is filled up
8 containment vessel.
9 CHAIR STETKAR: Yes.
10 MR. HAMAMOTO: We assumed there are
11 minimum spray through, assuming, even if the minimum
12 spray flow fills up containment, 12 hours is
13 sufficient to dissolve all of the energy.
14 We have 23 energy containment and I
15 forgot the detailed numbers, what we assumed that
16 spray minimum flow under time under dissolved time.
17 I confirm.
18 MR. SPRENGEL: Okay. I think we
19 understand your question.
20 CHAIR STETKAR: I was going to say, I
21 understand all of that. I know you only have two

1 spray trains running and you took the minimum spray
2 flow, but my point is that --
3 MR. SPRENGEL: What if you stop.
4 CHAIR STETKAR: Suppose you stop, and in
5 most plants, at least that I'm familiar with, the
6 EOPs tell people to stop.
7 MR. SPRENGEL: And that's zero, right.
8 We'll take that as a question.
9 CHAIR STETKAR: Or half.
10 MR. SPRENGEL: We'll have to follow up
11 on that.
12 CHAIR STETKAR: I mean, it's an
13 integration between the emergency operating
14 procedures and what's in the accident analysis, is
15 really what I'm asking about. Okay. You want to
16 say something?
17 MR. YOSHIDA: Our calculation --
18 CHAIR STETKAR: Make sure -- you have to
19 say your name also, so we have you on the record.
20 MR. YOSHIDA: Oh, sorry.
21 CHAIR STETKAR: That's fine. These

1 microphones are very sensitive, so you can speak
2 here.
3 MR. YOSHIDA: I am Tsuyoshi Yoshida from
4 MHI. In our pH analysis of the site, we don't
5 consider operator manual action. We assume safety
6 injection signal is actuated, spray is actuated and
7 automatically pH is going up.
8 CHAIR STETKAR: I understand that, but
9 in my sense, the accident analyses should also
10 consider -- in this case, not considering the
11 operator action to stop spray as a function of
12 containment pressure may result in optimistic
13 results from your accident analysis.
14 In other words, assuming that spray
15 comes on, minimum spray flow is available, only two
16 trains and so forth, but that it stays on for at
17 least 12 hours, that assumption allows you to
18 dissolve all of the buffer over that 12-hour period,
19 but if that assumption is optimistic with respect to
20 normal operator performance, then it leads to a
21 question about how reasonable is your accident

1 analysis.

2 This is not necessarily an operator

3 error. It is an operator doing what the operator is

4 normally instructed to do, at least from other

5 plants.

6 Now you know, you can write procedures

7 any way you want to write procedures, but --

8 MR. SPRENGEL: Right, but you just want

9 to confirm that the analysis, though, is --

10 CHAIR STETKAR: Is not optimistic.

11 MR. SPRENGEL: Right, is appropriately

12 conservative. So --

13 CHAIR STETKAR: Is appropriately

14 conservative considering expected operator response,

15 because other analyses that you perform, for tube

16 rupture and a few other transients, account for

17 expected operator performance.

18 MR. SPRENGEL: Okay. Okay. We'll

19 follow up on that.

20 CHAIR STETKAR: Okay. Thanks.

21 MR. MARUYAMA: Thank you very much,

1 okay, can I move on?

2 CHAIR STETKAR: Yes.

3 MR. MARUYAMA: Okay, now we are in slide

4 number 14. This table shows all of the Chapter 15

5 events that have been analyzed for radiological

6 consequences. Each event is classified as a

7 postulated accident. The RADTRAD computer code is

8 used in each case. There are no open items for

9 section 15.0.3 now.

10 This table shows the increase in heat

11 removal event in section 15.1. To save time, MHI

12 will not discuss each event in detail. Instead, a

13 representative event in each of the major sections

14 will be presented, as indicated by the red text in

15 the table.

16 The representative events were generally

17 selected as a limiting event in a section. For the

18 increase in heat removal events, MHI has selected

19 the steam system piping failure in section 15.1.5.

20 A steam system piping failure will cause

21 a large increase in steam flow. This will result in

1 a very large decrease in core inlet temperature due
2 to the increased heat transfer in the steam
3 generator.
4 MHI analyzes a lot of -- a hot standby
5 and a hot full power case. For the hot standby
6 case, the largest break size without loss of offsite
7 power is the most limiting scenario.
8 The radioactivity increases until
9 criticality occurs, and the reactor returns to
10 power. Safety injection actuation occurs on a low
11 steam line pressure signal.
12 The borated water from SI reduces
13 reactor power and restores subcriticality. The DNB
14 analysis shows that DNBR remains above the safety
15 analysis limit, and therefore no fuel failure is
16 predicted.
17 Both reactor coolant system and SG
18 pressures are not limiting, because they decrease
19 during this event.
20 For the hot full power case, the most
21 limiting break size is an intermediate break of

1 approximately 0.4 square feet.

2 Since the reactor is already at power,

3 the colder coolant result in an increase in reactor

4 power above the full power. Eventually, reactor

5 trip occurs on overpower delta T for the break size.

6 So reactor trip protects against DNB such that the

7 DNBR remains above the safety analysis limit,

8 throughout the event.

9 Again, RCS and SG pressure are not

10 limiting as they decrease during the event.

11 In addition to these analyses, MHI also

12 uses a steam mass and energy release result in order

13 to perform a containment integrity analysis in

14 Chapter 6.

15 There are no open items for section

16 15.1.

17 This table shows the decrease in heat

18 removal event in section 15.2. As before, the red

19 text indicates the representative event that MHI

20 will discuss in detail.

21 For the decrease in heat removal events,

1 MHI has selected the feedwater system piping break
2 in section 15.2.8.
3 CHAIR STETKAR: Before you do that,
4 because I have some questions about that, the loss
5 of external load analysis that you performed,
6 accounted for loss of offsite power at the time -- I
7 think it's three seconds after the reactor trip. Is
8 that correct?
9 MR. FUJISHIRO: Well, there is -- you
10 mentioned LOOP, loss of offsite power.
11 CHAIR STETKAR: Right.
12 MR. FUJISHIRO: However, this is
13 initiating event is, precisely speaking, loss of
14 external electrical load demand.
15 CHAIR STETKAR: Okay, I understand that.
16 Let me then ask the question that I really want to
17 ask. Have you evaluated an event that is initiated
18 at T time T=0 by a loss of all non-essential AC
19 power?
20 MR. FUJISHIRO: That is loss of offsite
21 power, different initiator, and this one is a --

1 CHAIR STETKAR: No, I understand that.
2 That's --
3 MR. FUJISHIRO: Oh, okay.
4 CHAIR STETKAR: What I'm asking is, I
5 think I do understand what you analyzed with loss of
6 external load, and that is not loss of offsite
7 power. It's loss of -- loss of something out there
8 that will accept the electrical load.
9 What I'm asking is, in any of your
10 analyses, have you evaluated an event that starts at
11 zero with no non-1E AC power, Such that the turbine
12 must then trip, the reactor must then trip, the RCPs
13 start to coast down at T=0, that you don't have any
14 force flow for some period of time. And I am not
15 sure whether you have evaluated that condition.
16 That condition, obviously, can occur. It happens at
17 plants.
18 MR. MARUYAMA: Yes, the RCS, we have a
19 variation that assumed that loss of offsite power at
20 time=0.
21 CHAIR STETKAR: You do.

1 MR. MARUYAMA: So that variation is in
2 section 15.2 that --
3 CHAIR STETKAR: That's part of the LNEP?
4 That's part of LNEP?
5 MR. MARUYAMA: Yes.
6 CHAIR STETKAR: Okay. Okay. I thought
7 it might have been, and I was hoping you were going
8 to say that. I just couldn't' necessarily draw all
9 of the lines together from what was in the DCD. So
10 that is part of LNEP.
11 MR. MARUYAMA: Yes.
12 CHAIR STETKAR: Okay, thank you. And
13 now I'll let you talk about feedwater pipe breaks.
14 MR. MARUYAMA: Okay. So, then, we will
15 talk about 15.2.8, the feedwater piping break. The
16 analysis for the feedwater break assumes that
17 feedwater is lost at time 0 that feedwater pipe
18 break is assumed to occur at the same time as the SG
19 water level reaches -- the low SG water level
20 reaches a reactor trip analytical limit.
21 During the resultant heatup, the

1 pressurizer safety valves and the main steam safety
2 valves open to reveal primary and secondary
3 pressure. As a result, the RCS and SG pressures
4 remain 110 percent of the design pressure.
5 The analytical -- the analysis shows
6 that DNBR decrease but the DNBR remains above the
7 safety analysis limit throughout the event. This
8 event also results in an increase in pressurizer
9 water level. The analysis results shows that the
10 pressurizer does not overfill and therefore no water
11 relief occurs through the pressurizer safety valves.
12 Radiological consequences of this event
13 are not analyzed because they are bounded by the
14 steam line break analysis due to the larger steam
15 release for that event.
16 MEMBER POWERS: I'm curious about your
17 second bullet. Suppose I have a double-ended
18 feedwater line break with steam water -- steam
19 generator water level above the low steam generator
20 water level reactor trip, what happens in that case?
21 (Off the record comments)

1 MR. FUJISHIRO: We have --
2 conservatively at time 0, all steam generator no
3 main feedwater. That the story is if we -- imagine
4 we assume the some intermediate size of feed line
5 break, then all main feedwater will spill out that
6 hole.
7 Then no main feedwater or steam
8 generator. Only main feedwater goes out from break
9 area.
10 MEMBER POWERS: I guess what I'm asking
11 really is suppose that I don't trip on break.
12 MR. FUJISHIRO: It is a very
13 conservative scenario to cover the small or
14 intermediate size of feed line break.
15 MEMBER POWERS: I guess the question I'm
16 asking, is that the conservative --
17 MR. LYNN: I think one of the things is
18 that this is a heatup event, so what we are trying
19 to do is maximize the possibility for heatup. So if
20 you have a feedwater line break that happens at time
21 0, it would obviously affect that faulted SG, and it

1 would drain the water level in that SG very quickly,
2 and you would get a reactor trip, and the intact SGs
3 would be relatively unaffected and their inventory
4 would then be available to help cool down the plant.
5 Now, in the scenario we analyzed, we
6 drain all of our SGs down to that low, low level,
7 and then have a break such that when the reactor
8 trip does occur, all of the SGs, including the
9 unaffected ones, have the least amount of inventory
10 possible that you could postulate, and so therefore
11 you get more of a heatup than the other case --
12 MEMBER POWERS: What I'm asking you, is
13 follow your scenario exactly, but now give me two
14 feet of water in the steam generators so that I
15 don't -- so that I get some time at power before I
16 trip. Where is the worst case in that scenario? Is
17 it in fact the case you have selected, or is it a
18 somewhat higher water level?
19 MR. FUJISHIRO: Guillotine break at time
20 0 guillotine break, the steam generator water level
21 is going down very fast, then maybe after 10 seconds

1 the reactor trip signal will come in.
2 At that time, intact steam generator
3 have lot of water inventory, then long time it is
4 not remitting. Then reducing the break area from
5 routine break, the -- after the steam generator
6 water level is close to intact steam generator,
7 water level goes down. Then more small case, maybe
8 all main feedwater will spill over, then all steam
9 generator going down, same slope and hit the lower
10 steam generator or at the same time, it is the most
11 severe case and after that, the reactor trip, we
12 assume guillotine break.
13 This one is -- covers intermediate
14 break.
15 MR. MARUYAMA: So, we believe this case
16 is most conservative case, compared to the case we
17 assumed that the -- in fact, rupture this overfill -
18 - not overfill -- spillout occurs at the time 0.
19 MR. FUJISHIRO: No, we not to assume the
20 blowdown from rupture of the steam generator before
21 reactor -- because if we assume some flowdown

1 rupture of the steam generator, that steam generator
2 will -- water level going down fast, and the heat to
3 the reactor trip set point earlier, at that time,
4 intact steam generator have more water inventory,
5 and long-time, intact steam generator water
6 inventory will be consumed by decay heat.
7 Rupture of the steam generator sooner or
8 later will dry out, then that doesn't matter. It --
9 our concern is, at that time, at that point, is that
10 intact steam generator with an inventory minimum
11 case is worst case. Then --
12 CHAIR STETKAR: You want to think for a
13 minute? Let me ask him another question if --
14 MEMBER POWERS: Please.
15 CHAIR STETKAR: Give you some thinking
16 time. I had a question also about this event. And
17 that question is whether it is optimistic from an
18 energy release into the containment, the last bullet
19 on your slide says, "The radiological consequences
20 from this event are bounded by a steam line break
21 due to the larger steam release," larger energy

1 release.

2 The -- if you start this with a loss of

3 all main feedwater flow, allow levels to go down to

4 the trip set point and then break the feedwater

5 line, by definition because you started with no main

6 feedwater flow, there is no main feedwater flow.

7 If I go break that line, at time T=0,

8 and I'll give you nominal steam generator levels,

9 you are right, the broken steam generator level will

10 fall more quickly than the other three.

11 You will get a reactor trip signal

12 perhaps a little bit earlier. But I still have main

13 feedwater pumping a lot of energy into the

14 containment through that broken feedwater line, a

15 lot of energy, and that doesn't stop until main

16 feedwater is isolated.

17 Main feedwater isolates automatically

18 on, in this case an ECCS actuation signal, which has

19 some delay time associated with it. I don't know

20 when it comes in.

21 The question I have is, is it really

1 true that a main steam line break deposits more
2 energy into the containment than a main feedwater
3 line break at T 0 with delayed isolation of that
4 main feedwater line?
5 I have seen analyses of other plant
6 designs where indeed the limiting event is the main
7 feedwater line break because of the delayed
8 feedwater isolation, so I was curious, in your
9 particular analysis, you don't need to consider that
10 because by definition, you started with no main
11 feedwtaer, so there is no need to isolate main
12 feedwater. There is zero flow.
13 So I was curious whether you actually
14 looked at that other case to assure yourself that
15 indeed, you are still limited by the steam line
16 break. That's a different -- that's a little bit
17 different topic than what Dana was asking about.
18 MR. FUJISHIRO: Feedline break, main
19 feedwater spill over the containment, however
20 relatively low energy compared to steam release.
21 CHAIR STETKAR: For steam.

1 MR. FUJISHIRO: Yes, the -- MHI
2 currently believes a steam line break is limiting,
3 however we have not evidence right now. Do you have
4 something?
5 MR. MARUYAMA: Yes, actually, the mass
6 and energy release analysis is a form -- described
7 in Chapter 6 and in Chapter 6.
8 CHAIR STETKAR: Yes, I mean we haven't
9 reviewed Chapter 6. That's a -- I try to bounce
10 back and forth a little bit, but you run out of time
11 after a while.
12 (Simultaneous speaking.)
13 CHAIR STETKAR: In Chapter 6 you did?
14 Okay.
15 MR. LYNN: We have both analyses and the
16 steam line break is limiting. The only question,
17 then, that we are trying to determine, is whether
18 the assumptions for the Chapter 6 analysis are the
19 same as this.
20 CHAIR STETKAR: That's -- I couldn't, I
21 couldn't -- I couldn't find that quickly.

1 MR. LYNN: And so I think we may have to
2 look that up and get back to you.
3 CHAIR STETKAR: Okay. Okay. I tried to
4 check. I did go look at Chapter 6, and at least
5 quickly, what I could find there, I couldn't
6 determine exactly what those assumptions on the
7 feedwater line break were. So thanks, if you can
8 confirm that, that would help.
9 MR. MARUYAMA: So we will confirm the
10 assumption in chapters -- difference in the
11 assumptions in Chapter 6 and --
12 CHAIR STETKAR: Right, as long as the
13 Chapter 6 analysis, where you really do address this
14 issue --
15 MR. MARUYAMA: Yes. We think there are
16 some differences but I will check about this.
17 CHAIR STETKAR: Dana, did you have
18 anything to follow up on --
19 MEMBER POWERS: No, I just don't believe
20 they have gotten the most conservative case.
21 CHAIR STETKAR: Yes. So the question is

1 still there, whether some demonstration that indeed
2 that assumption about low initial levels is, is you
3 know, with a trip happening at effectively T 0 when
4 the break occurs, is that the most limiting
5 condition.
6 MEMBER POWERS: The problem is they
7 would get into troubles if they delayed the trip.
8 MR. WOOD: I'm Doug Wood. I have worked
9 with these people on some analyses. I just wanted
10 to point out, for those of you who don't have that
11 event, the feedline break, in front of you right
12 now, the peak -- all the peak parameters, you know,
13 are way out in time.
14 This is really a global energy, you
15 know, balance on the RCS. The event, you know, the
16 worst case parameters for these events peak many,
17 many, many seconds after the reactor trip, so this
18 is really an energy balance event and the -- I think
19 that if you think of it in that way, it's logical to
20 conclude that the worst case situation occurs where
21 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

1 trying to connect the dots on the two. I'm done.
2 CHAIR STETKAR: It's strictly just a
3 question of, even in the short term, how much energy
4 is added.
5 MEMBER BROWN: I understand that part of
6 it. I just --
7 MR. MARUYAMA: Okay, I'll go on to the
8 next page.
9 MR. SPRENGEL: Real quick, Yuta. Does
10 MHI understand the outstanding question?
11 MR. MARUYAMA: Regarding the timing of
12 the --
13 MR. SPRENGEL: I don't know what the
14 question is. That's why I asked.
15 CHAIR STETKAR: Dana, if you could --
16 MEMBER POWERS: Well, I guess just how
17 you perturb this, this thing. What they assume here
18 is an all steam generator comedown to this level set
19 point. They get a break, a double-ended guillotine
20 and instantaneously trip the reactor.
21 CHAIR STETKAR: With some minor time

1 delay.

2 (Simultaneous speaking.)

3 MEMBER POWERS: No, all I'm asking is

4 that we in fact consider suppose you came down to a

5 foot above the low level at a double-ended

6 guillotine type break, and so now that some time

7 delay becomes a little longer time delay, and if it

8 goes long enough, then clearly this is more

9 limiting. But is there an intermediate case that is

10 in fact more limiting than this instantaneous time

11 trip? It's a very simple question. The alternative

12 that's been posed is in fact the other steam

13 generators are not low, okay? And I will admit,

14 this is more conservative than those. But suppose -

15 - I'm just following from their hypothesized

16 situation except saying they didn't quite get that

17 low, so you have a little longer period of power, so

18 you put a little more oomph into the system before

19 you trip.

20 At what point are you most conservative?

21 (Off the record comments)

1 MR. LYNN: So just to clarify, you agree
2 that starting at -- starting the break at time 0
3 with all of the SGs at their nominal levels would be
4 less limiting than what we presented, but you are
5 saying somewhere between here and there might be --
6 MEMBER POWERS: Well, in between the two
7 --
8 MR. LYNN: Okay.
9 MEMBER POWERS: have you looked to see
10 if in fact you found the most limiting case or not?
11 Because you are right, I mean, as long as those
12 other generators are full you have got all inventory
13 to help your cooling down. When does that water
14 stop being your friend and start being part of the
15 pressurization? Maybe you are absolutely correct
16 for all I know, but it would surprise me if you were
17 absolutely correct.
18 I am often surprised. Yes. Well you
19 didn't need to agree so enthusiastically.
20 CHAIR STETKAR: Well if you admit it.
21 MEMBER POWERS: I admit it, but it

1 should be viewed as something of humility on my part
2 and not an obvious truth.
3 CHAIR STETKAR: Humility and power are
4 never used in the same sentence. Do you actually --
5 do you understand the concern? It's not necessarily
6 saying you know, we are not saying start it, you
7 know, from nominal level, because it's clear that --
8 MR. MARUYAMA: So you're asking if there
9 is intermediate conservative case or not. So --
10 CHAIR STETKAR: Could there be some
11 intermediate level that indeed the combination of
12 somewhat reduced heat removal from the other steam
13 generators, with the additional delay time for the
14 reactor trip, are more bounding than what you used.
15 MR. MARUYAMA: You are asking
16 intermediate size break or --
17 CHAIR STETKAR: No, take the big break
18 but start it, instead of -- you started it at this
19 level. If this is the normal level, is there some
20 intermediate level in here that may be more limiting
21 --

1 MR. MARUYAMA: Oh, however, steam
2 generator water level is controlled by feedwater
3 control system and as initial condition, it should
4 be a condition one situation then steam generator
5 water level should be programmed level. We must
6 take account some failure in feedwater control
7 system. It's quite a real case. And it should not
8 be considered as an accidental condition.
9 If we have a feedwater controlled system
10 have some there, then we cannot keep the steam
11 generator water level trip either.
12 MR. SPRENGEL: Okay, we'll take some
13 time to discuss. I think we understand the point,
14 some of us understand the point.
15 CHAIR STETKAR: Okay.
16 MR. SPRENGEL: And we'll discuss
17 internally. We'll move on for now.
18 CHAIR STETKAR: That might be an
19 argument, in terms of this is an accident analysis.
20 MEMBER BROWN: I'm trying to relate from
21 my past to this one and each generator has its own

1 control system, so that three -- when you have a
2 break in one, doesn't mean that the other ones don't
3 try to drive the other one up so that you don't
4 break any of those down to generate a trip.
5 So I don't know how many trips it takes,
6 just one? So it's when the one that has the
7 guillotine break comes down, it's not a coincidence
8 type thing for that, for these -- for this plant,
9 for these designs.
10 CHAIR STETKAR: Low level in any one --
11 MEMBER BROWN: Any one.
12 CHAIR STETKAR: Any one of the four will
13 --
14 MEMBER BROWN: Okay, but I mean, so if
15 they start right at the trip point you don't have
16 time then for the other one to be driven up and then
17 drive yourself into the scenario that Dana is
18 talking about.
19 CHAIR STETKAR: That's why they assumed
20 the loss of feedwater event at time minus or
21 something, if you want to think of it, as some

1 rational way to get all of the levels, all of the
2 levels down.
3 MEMBER BROWN: I just wanted to relate
4 to his comment, relative to the level control
5 systems interaction, and now I understand what the
6 difference is. I thought it was a coincidence, the
7 two of them, and so not understanding that changes
8 the --
9 CHAIR STETKAR: Low level in any one of
10 the four steam generators will do it for you.
11 MEMBER BROWN: Okay, I'm done. Thank
12 you.
13 MR. MARUYAMA: So, actually there are no
14 open items for section 15.2 now. The next table
15 shows the decrease in reactor coolant system flow
16 rate event in section 15.3.
17 For the decrease in RCS flow events, MHI
18 has selected the RCP rotor seizure in section
19 15.3.3. the seizure of one RCP rotor causes a rapid
20 decrease in RCS flow, which results in an increase
21 in RCS temperature and decrease in DNDR.

1 A reactor trip will occur on the low
2 reactor coolant flow signal. Even with the reactor
3 trip, the DNBR decreases below the safety analysis
4 limit. Therefore, MHI calculates number of rods in
5 DNP and this is significantly less than 10 percent.
6 MHI also calculates the peak cladding
7 temperature. It remains below the 2,200 degree
8 limit. RCS and SG pressure both remain below 110
9 percent of the system design pressure.
10 A dose analysis is then performed by
11 assuming 10 percent of fuel rods have failed. This
12 fuel failure bounds the actual analysis results
13 described above.
14 MEMBER POWERS: When you do a dose
15 analysis where 10 percent of rods failed, what do
16 you do? What do you release from the failed rods?
17 MR. FUJISHIRO: Would you ask me again?
18 Ten percent of fuel rods --
19 MEMBER POWERS: You have failed -- 10
20 percent failed rods --
21 MR. FUJISHIRO: Maybe some credit

1 calculations, some things that we assume that were
2 going to fail, and account in radioactivity
3 calculation.
4 MEMBER POWERS: What I'm asking is what
5 is the radionuclide release you associate with 10
6 percent railed rods?
7 MR. YOSHIDA: This is Tsuyoshi Yoshida
8 with MHI. We assume radioactivity from broken fuel
9 rod released into RCS and primary to secondary
10 leakage is assumed, so contaminated tube water leaks
11 to secondary site, and from main steam relief valve
12 contaminated steam is released. That is our
13 assumption.
14 MEMBER POWERS: What radionuclide
15 release do you assume from the rod itself?
16 MR. YOSHIDA: We assume noble gas and
17 iodine and cesium.
18 MEMBER POWERS: Just a gap inventory?
19 MR. YOSHIDA: Pardon?
20 MEMBER POWERS: Just a gap inventory?
21 MR. YOSHIDA: Yes.

1 MEMBER POWERS: Then you take that gap
2 inventory to be what?
3 MR. YOSHIDA: Gap -- pardon?
4 MEMBER POWERS: What fraction of the
5 inventory of radionuclides are in the gap?
6 MR. YOSHIDA: Ah, okay. We assume 10 --
7 based on radioactivity at 1.13, we assumed --
8 MEMBER POWERS: Five percent.
9 MR. YOSHIDA: Yes.
10 MEMBER POWERS: And so you don't leach
11 anything. You don't leach anything out of a bit of
12 fuel. No.
13 MR. MARUYAMA: There are no open items
14 for section 15.3 now. Okay the next table shows the
15 reactivity and power distribution anomaly events in
16 section 15.4.
17 In this case MHI has selected two
18 events. MHI has selected the control rod assembly
19 withdrawal at power event as an AOO to present. MHI
20 has also selected the rod ejection as a PA to
21 present.

1 Rod withdrawal at power insert
2 radioactivity to the core, and therefore increases
3 with the power from its initial value. MHI analyzes
4 different combinations of radioactivity insertion
5 rates, BOC and EOC and initial powers to determine
6 the limiting case or cases.
7 The event is terminated by a reactor
8 trip. The reactor trip that occurs depends on the
9 conditions of the particular case, but it is one of
10 the four trip signals listed here.
11 The DVD provides transient results for
12 two cases -- a high reactivity insertion rate case
13 an a low reactivity insertion rate case. The
14 reactivity rate case is more limiting from a DNBP
15 perspective.
16 For both cases, DNB decreases until
17 reactor trip occurs, but remains above the safety
18 analysis limit such that fuel failure is not
19 predicted.
20 This event is not limiting for either
21 RCS or SG pressure.

1 MEMBER POWERS: What kind of energy
2 deposition do you get in the fuel?
3 MR. MARUYAMA: Excuse me, could you say
4 that again?
5 MEMBER POWERS: What kind of energy
6 deposition to get in the fuel?
7 MR. MARUYAMA: Energy deposition?
8 MEMBER POWERS: Right. You've got a
9 reactivity insertion, you are putting energy
10 suddenly into the fuel, how much do you put in?
11 MR. MARUYAMA: Well, we evaluate the
12 fuel centerline temperature and it is a little bit
13 to increase, but energy deposition is basically the
14 criteria for the reactivity initiating event. That
15 is from zero power.
16 There is no such criteria if you
17 interpret something. There is no. Only we are
18 looking for the fuel melting, not melting. That is
19 -- may I continue now?
20 MEMBER POWERS: You don't get centerline
21 melting?

1 MR. FUJISHIRO: No. The power is very
2 increasing slowly and at maximum power is a little
3 bit overshoot, maybe 120 percent, something the --
4 not so much big energy deposition and fuel
5 centerline temperature well below the melting point.
6 MEMBER BANERJEE: The DNBR calculation
7 is based on full-scale experiments?
8 MR. FUJISHIRO: You mean the DNBR test
9 itself?
10 MEMBER BANERJEE: Yes, well the 14-foot
11 test, did you do?
12 MR. FUJISHIRO: It is already done I
13 think.
14 MEMBER BANERJEE: Finished? It was
15 full-scale? Who can tell us?
16 MR. MARUYAMA: Yes, we did full-scale
17 test.
18 MEMBER BANERJEE: Full scale with --
19 where was it done?
20 MR. MARUYAMA: Where?
21 MEMBER BANERJEE: Full power, full

1 scale? It's a big, big LOOP.
2 MR. SPRENGEL: The DNBR testing that was
3 done, was done in Germany.
4 MEMBER BANERJEE: Okay, at AREVA?
5 MR. SPRENGEL: At an AREVA facility,
6 yes.
7 MEMBER BANERJEE: Because this is a
8 large facility. This was done at full scale, so did
9 you look at transient effects as well?
10 MR. MARUYAMA: Transient effects?
11 MEMBER BANERJEE: Because I guess --
12 MR. SCHMIDT: This is Jeff Schmidt with
13 the NRC. They looked at a spectrum of cases to bound
14 the Chapter 15 events and their cases were
15 confirmatory in nature, using a certain existing --
16 MEMBER BANERJEE: Correlation.
17 MR. SCHMIDT: correlation, yes.
18 MEMBER BANERJEE: Which we can't name.
19 MR. SCHMIDT: Which I cannot name, which
20 shall remain nameless, at least until tomorrow. So
21 --

1 MEMBER BANERJEE: It starts with a B.
2 MR. SCHMIDT: So the transient cases, or
3 the cases cover the transient cases and it was five
4 by five array, 14 feet high.
5 MEMBER BANERJEE: Not a 17 by 17.
6 MR. SCHMIDT: No it was not a 17 by 17.
7 MEMBER BANERJEE: And you are satisfied
8 with this five by five area?
9 MR. SCHMIDT: Yes, it's typical DNBR
10 testing, yes.
11 MEMBER BANERJEE: All right. Will you
12 be speaking to these tests later? I mean, staff
13 will give us some of your views on these AOOs?
14 MR. SCHMIDT: I mean it's covered as
15 part of the VIPRE topical report. That's where the
16 DNBR testing is referenced and VIPRE is referenced
17 in Chapter 15.
18 MEMBER BANERJEE: Okay.
19 MR. SCHMIDT: But you know, mainly the
20 discussion of the actual testing will be done under
21 the VIPRE topical report.

1 MEMBER BANERJEE: Okay, thank you.
2 MR. MARUYAMA: Okay, next page. This
3 page. Page 29. MHI has also chosen rod ejection
4 postulated accident as a representative event for
5 section 15.4. A rod ejection causes a large
6 reactivity insertion, and also large power peaking.
7 The initial power increase is mitigated
8 by the Doppler feedback effect. Reactor trip will
9 also occur, but reactor trip that occurs depends on
10 the condition of the particular case, and can be any
11 one of the four trip signals listed here.
12 MEMBER BANERJEE: For the calculations
13 do you use a spatial kinetics code which is coupled
14 or you decouple the spatial modes from --
15 MR. MARUYAMA: No not coupled.
16 MEMBER BANERJEE: Not coupled. So you
17 do, you have a time and space separate calculations?
18 MR. MARUYAMA: I mean, the time and
19 space -- we use 3D calculation for HOTSPOT case,
20 this is time and space combined. But we mean the --
21 some are calculation and kinetic calculation is not

1 coupled.

2 MEMBER BANERJEE: But your rod ejection

3 is a fully coupled code, right? Or not.

4 MR. FUJISHIRO: He explained our

5 calculation covers 3D, however initial temperature

6 is constant and the pressure is also constant, and

7 internally, there is of course thermal hydraulic and

8 also 3D nuclear calculation, to include 3D

9 calculation.

10 MEMBER BANERJEE: The 3D nuclear

11 calculation is a transient 3D?

12 MR. FUJISHIRO: Yes.

13 MEMBER BANERJEE: The nuclear

14 calculation is fully transient 3D? But it's not

15 coupled to the thermal hydraulics.

16 MR. OGAWA: Not coupled with the fuel

17 calculation, sorry.

18 MEMBER BANERJEE: Not coupled. So let

19 me understand what it is coupled to.

20 (Off the record comments)

21 MR. MARUYAMA: Sorry, I'd like to modify

1 what was said. The fuel calculation on the 3D
2 kinetic calculation is performed at the same side as
3 also, as we explained, the 3D transient calculation,
4 using TWINKLE-M code.
5 MEMBER BANERJEE: Okay. So the fuel
6 calculation, the temperatures, everything, get fed
7 into this 3D neutronic calculation? It's fully
8 coupled. Everything is coupled, temperature as well
9 as the --
10 MR. FUJISHIRO: Fuel temperature and
11 coolant temperature.
12 MR. BIELEN: This is Andy Bielen from
13 the staff. They start with the neutronics
14 calculation, with TWINKLE-M, so they do a full 3D
15 kinetics calculation. Then they make some
16 adjustments to it and put it into VIPRE to do
17 temperature evaluation.
18 MEMBER BANERJEE: And that's fed back or
19 --
20 MR. BIELEN: No. Nonono. It's one way
21 -- one way.

1 MEMBER BANERJEE: Thank you.
2 MR. BIELEN: Sure.
3 MEMBER BANERJEE: It's clear what you
4 did. Thank you.
5 MR. MARUYAMA: Okay, so where am I?
6 Now, so, RCS pressure --
7 MEMBER BANERJEE: Does this give you a
8 conservative --
9 MR. MARUYAMA: Yes.
10 MEMBER BANERJEE: Conservative bound on
11 the effect, right, of the rod reaction?
12 MR. MARUYAMA: You mean the one-way
13 calculation?
14 MEMBER BANERJEE: Yes.
15 MR. MARUYAMA: Yes. If we considered
16 feedback from the thermal hydraulic calculation,
17 that will, you know, reduce the --
18 MEMBER BANERJEE: All right.
19 MR. MARUYAMA: power so that is
20 conservative.
21 MEMBER BANERJEE: Okay.

1 MR. MARUYAMA: The rod is -- so may I
2 continue?
3 MEMBER BANERJEE: Yes.
4 MR. MARUYAMA: Okay. The rod ejection
5 will also cause a breach of the reactor coolant
6 pressure boundary that will result in a decrease in
7 RCS pressure. MHI analyzed several cases in order
8 to consider different combinations of EOC and BOC
9 along with initial power.
10 For the full power cases, BOC and EOC,
11 MHI uses the measured neutron flux detector power
12 considering a single failure to determine when the
13 reactor will trip.
14 For these full power cases, the peak
15 fuel centerline temperature and other fuel pellet
16 enthalpy at the hot spot remain below their
17 respective limits.
18 For the zero power cases, the react will
19 trip on the high neutron flux low setpoint. For
20 these zero power cases, the average fuel pellet
21 enthalpy at the hot spot under hot spot peak fuel

1 enthalpy remains below their respective limits.
2 The zero power cases also look at PCMI.
3 The analysis results show no PCMI failed, no PCMI
4 failed the fuel rods.
5 MHI also performs analysis to determine
6 the number of rods in DNB for the hot full power
7 cases. In the DNB analysis, the limiting case
8 occurs when the reactivity insertion is such that
9 reactor trip on neutron flux does not occur.
10 Instead, reactor trip occurs later,
11 during the RCS depressurization caused by the breach
12 of the reactor coolant pressure boundary. No DNB
13 occurs during the initial prompt power increase.
14 However, DNB eventually decreases below the safety
15 analysis limit during the RCS depressurization.
16 MHI calculates the number of rods in DNB
17 and it is less than 10 percent. MHI performs
18 additional analysis for RCS pressure, thus shows
19 that RCS and SG pressures remain below 110 percent
20 of the design pressures.
21 MEMBER BANERJEE: How do you establish

1 this 10 percent? Do you have a probability
2 distribution curve of some sort? This is right in
3 the tail side, right, of the curve? Like it's just
4 10 percent means if you have a curve like this, it's
5 just at the tail right? So how do you establish
6 this? I know there is a well-established
7 methodology, but are you, what do you --
8 MR. MARUYAMA: Yes, actually for DNB
9 failure calculation, first we calculate the
10 consensus, consensus if there is an HN. Then, on
11 the other hand, we calculate the limiting that
12 reaches the DNBR in transient calculation.
13 So comparing those two results, we will
14 find a fuel that have more -- that have more than
15 limiting, then we assume that those fuels are
16 failed.
17 So such kind of methodology is described
18 in detail in non-LOCA topical report methodology.
19 MEMBER BANERJEE: Because you are in a
20 very small region.
21 CHAIR STETKAR: We'll see that report in

1 October.

2 MEMBER BANERJEE: Okay.

3 MR. LYNN: The relevance of the 10

4 percent is that 10 percent is the number assumed in

5 the dose calculation, so --

6 MEMBER BANERJEE: I know that you have

7 this, but it's a question of how you establish it.

8 That's -- but as John says, we will see it.

9 CHAIR STETKAR: Yes, it's just a problem

10 in terms of timing of our meetings and staff reviews

11 and things that are beyond our meager control at the

12 moment.

13 MEMBER BANERJEE: Okay.

14 CHAIR STETKAR: And that's something

15 that we will not discuss tomorrow, that's not on the

16 table. So --

17 MEMBER BANERJEE: Fine, as long as it's

18 discussed at some point.

19 CHAIR STETKAR: Yes,, it's on our agenda

20 for October.

21 MR. MARUYAMA: Okay, then we will

1 discuss our methodology in detail. Okay. Then in
2 page 32, there are three open items for section
3 15.4.
4 For each of our items, MHI has provided
5 a response to the associated RAI. These responses
6 are currently under industry review.
7 And this table shows the increase in
8 reactor coolant inventory events in section 15.5.
9 In this case only the CVCS malfunction event in
10 section 15.5.2 is applicable to the US APWR.
11 Therefore MHI has selected this event to
12 present in detail. The CVCS malfunction is assumed
13 to be a full open failure of the charging flow
14 control valve.
15 The boron concentration of the injected
16 water from the CVCS is the same as RCS, therefore
17 the main concern is an increase in pressure,
18 pressurized water level that could fill the
19 pressurizer.
20 The US APWR has an automatic CVCS
21 isolation feature that occurs on high pressurized

1 water level. This automatic isolation will
2 terminate the increase in pressure the water level
3 before pressurizer overfill occurs.
4 There are no open items for section
5 15.5. And this table shows the decrease in reactor
6 coolant inventory events in section 15.6. In this
7 case MHI has selected two events. MHI has selected
8 the steam generator tube rupture and LOCA events to
9 present.
10 Both of these events are PAs.
11 The steam generator tube rupture event
12 considers the double-ended rupture of one SG tube.
13 MHI analyzes two cases, one to evaluate doses and
14 the other to determine the margin to SG overfill.
15 In both cases, manual operator actions
16 are necessary to terminate the event and are
17 therefore assumed in the analysis.
18 These manual actions -- these manual
19 actions include reactor trip, isolation of the
20 ruptured SG, cooldown and depressurization of the
21 RCS and termination of SI.

1 This slide provides additional detail
2 for the dose evaluation case. A manual reactor trip
3 is assumed 15 minutes after the tube rupture.
4 Isolation of the ruptured SG is assumed at 20
5 minutes.
6 Then, the operators begin an RCS
7 cooldown at 25 minutes. The remaining operator
8 actions are performed as conditions allow, and are
9 therefore determined by the analysis.
10 The event is not terminated until the
11 primary to secondary leakage finally stops at
12 approximately 70 minutes. The RCS cooldown is
13 performed by releasing steam from only the ruptured
14 SGs.
15 However as an additional failure the
16 main steam release valve of the ruptured SG is
17 assumed to open at the same time, remain stuck open
18 until it is later isolated by operator.
19 CHAIR STETKAR: Is that -- I'm wanting
20 to understand that isolation of the stuck-open
21 relief valve, because in one place I thought it said

1 it was operated -- isolated by the operator. In
2 another place it seemed to say it was isolated
3 automatically.
4 Does the analysis account for operator
5 action or is it an automatic isolation?
6 MR. FUJISHIRO: Main steam relief valve
7 will have motor-operated isolation valves and it
8 close well below the setpoint motor isolation valve
9 is automatically closed.
10 If operator opens as steam relief valve,
11 or automatically open and close steam generator MHO
12 will maybe working something like that, and
13 sometimes stuck open then motor operated isolation
14 valve will automatically close it.
15 What does that -- you're mentioning main
16 steam isolation valve?
17 CHAIR STETKAR: No, no, I'm talking
18 about the -- they are either called -- it's the MSRV
19 isolation valves.
20 MR. FUJISHIRO: Then if MSRV is stuck
21 open --

1 CHAIR STETKAR: Right, and =
2 MR. FUJISHIRO: then --
3 CHAIR STETKAR: and in the analysis,
4 that closed -- the isolation valve for the stuck-
5 open MSRV closes at 1,826 seconds after the
6 initiating event, approximately.
7 Now, in some places, in the DCD, it
8 seems that the operator closes it, and in other
9 places it seems that it closes automatically. What
10 I'm asking is, is that an automatic isolation signal
11 or is that an operator action? In the analysis.
12 MR. LYNN: I believe it's an operator
13 action, because the method of closing it is to close
14 a block valve before the valve.
15 CHAIR STETKAR: I understand, but that
16 block valve also gets an automatic signal from some
17 things, and that's what I'm trying to understand
18 about the analysis versus the real world.
19 So, you're saying in the analysis, it's
20 closed by an operator action?
21 MR. MARUYAMA: Yes.

1 CHAIR STETKAR: Okay, thank you.
2 MEMBER POWERS: What do you assume on
3 the iodine spike?
4 MR. FUJISHIRO: Could you say again?
5 MEMBER POWERS: What do you assume on
6 the iodine spike?
7 MR. FUJISHIRO: Iodine spike didn't halt
8 the steam generator.
9 MR. YOSHIDA: Of course we assume iodine
10 spike two case. One is pre-accident iodine spike
11 case and the second is coincident iodine spike case,
12 based on Regulatory Guide 1.183.
13 MEMBER POWERS: 500-fold increase over
14 the water coolant.
15 CHAIR STETKAR: Make sure, if you are
16 answering something, make sure you are at
17 microphone.
18 MR. YOSHIDA: For iodine spike, it's
19 based on technical specifications. We assume 60
20 microcuries per gram concentration, and a coincident
21 iodine spike case based on Regulatory Guide 1.183,

1 335 times of iodine concentration.
2 CHAIR STETKAR: In your dose evaluation
3 analysis for the tube rupture case, do you include
4 credit for filtering through the main condenser?
5 MR. YOSHIDA: In both analyses, we don't
6 consider main condenser.
7 CHAIR STETKAR: So any releases to the
8 main condenser are treated as a direct release to
9 the atmosphere?
10 MR. YOSHIDA: Yes.
11 CHAIR STETKAR: Everything? Noble
12 gases, everything?
13 MR. YOSHIDA: We assume all the released
14 radioactivity goes through main steam release valve
15 and main steam safety valve. So most go through
16 condenser.
17 CHAIR STETKAR: Well, I guess my
18 confusion is in the DCD there's a table, table
19 15.6.3-3, that says the releases -- this is steam
20 mass -- released to the atmosphere from the ruptured
21 steam generator is 109,000 pounds released to the

1 atmosphere. The release to the main condenser is
2 1,280,000 pounds, and the difference is because of
3 the timing that you have assumed on various
4 automatic and manual signals.
5 So, my question -- and it's clear,
6 there's a plot that shows the release to the
7 atmosphere starting at 900 seconds and ending at
8 1,800 seconds, let's say, and they are released to
9 the atmosphere from the steam -- the ruptured steam
10 generator before 900 seconds.
11 Now my question is, before 900 seconds,
12 you are releasing to the main condenser, from the
13 ruptured steam generator. Does the dose analysis
14 treat those releases during the first 900 seconds as
15 a release to the atmosphere?
16 MR. YOSHIDA: In dose analysis, we
17 assume release that from time -- when SG release
18 occur. Yes. So -- but in safety analysis, I'm not
19 sure, but I guess the result is not -- in safety
20 analysis result, steam release is not occur the time
21 is zero.

1 CHAIR STETKAR: I guess I didn't quite
2 understand.
3 MR. FUJISHIRO: We got to you --
4 CHAIR STETKAR: Let me -- the reason I
5 ask this is that if I follow the time line of the
6 analysis, the time line says that the tube rupture
7 occurs at zero, which is good. I understand that.
8 At two minutes, 120 seconds, it's
9 assumed that the operators discover the tube rupture
10 because of the nitrogen 16 alarms. It's just noted
11 in the analysis.
12 At 10 minutes, they identify the
13 ruptured steam generator and at 15 minutes, they
14 then manually trip the reactor, close the main steam
15 isolation valve and at the same time, a loss of
16 offsite power happens.
17 Then the analysis proceeds, oh, and also
18 at 20 minutes, they isolate emergency feedwater to
19 the ruptured steam generator, so that 15 to 20
20 minute time window is an important time window.
21 Then, from 15 minutes, 900 seconds,

1 until 1,826 seconds, when the MSRV block valve is
2 closed, you have a release to the atmosphere from
3 the ruptured steam generator. That's clear.
4 My question is, if the dose analyses
5 only account for the release between 900 and 1,800
6 seconds, that's one issue. If I'm an operator and
7 at two minutes I see that I have a ruptured steam
8 generator, I will probably trip the reactor very
9 quickly, and if my atmospheric release then starts
10 at two minutes or two minutes and 30 seconds and is
11 not terminated until 30 minutes, that's a much
12 different release than something that starts at 15
13 minutes. In other words I have, you know, something
14 on the order of 13 or -- 12 or 13 more minutes of
15 release.
16 So my question is, does your -- does
17 your dose analysis only account for the 900 to 1,800
18 second release, or does it account for the release
19 from zero?
20 MR. FUJISHIRO: I think that before 900
21 the -- during zero and 900, this period, steam flow

1 is going to the condenser. However, DF is two
2 types, steam generator and condenser, then it should
3 be reflected in dose evaluation.
4 However, we said to dose people total
5 amount of the leaks, and it should be noble gases
6 all the leak is discharged to atmosphere in dose
7 calculation and also we send to those guy the safety
8 valve and relief valve steam amount.
9 However, we don't send to you the zero
10 to 900 total steam flow. However, its effect is
11 small.
12 MR. YOSHIDA: Almost all he said is
13 correct and --
14 MEMBER POWERS: Almost all?
15 (Laughter.)
16 MR. YOSHIDA: In dose analysis, release
17 of steam has much value. So I think this 900
18 seconds steam must -- is included over impact
19 condition.
20 CHAIR STETKAR: Can you repeat that,
21 because I didn't quite understand?

1 MR. YOSHIDA: In dose analysis, we based
2 our result of safety analysis, we -- our input data
3 for dose analysis includes some margin, about 20
4 percent, so this 900 seconds period of release must
5 -- steam loss -- is -- effect of this 900 second
6 steam loss is included in our input condition.
7 MR. FUJISHIRO: He wanted to say the
8 total equivalent is from time zero. The noble gases
9 entire one is calculated in dose calculation.
10 CHAIR STETKAR: From zero. I understand
11 that because I read the noble gases go out through
12 the condenser event. I'm more interested in how you
13 are treating iodine and other things.
14 MR. FUJISHIRO: The iodine --
15 CHAIR STETKAR: You assume one percent
16 failed fuel in all of your leakers and things like
17 that.
18 MR. FUJISHIRO: Iodine is weak, assume
19 the -- none of the -- the contamination factor, 199,
20 then one percent only to atmosphere --
21 CHAIR STETKAR: Because of the main

1 condenser?

2 MR. FUJISHIRO: Yes. Then not only the

3 condenser, but the steam generator, right?

4 CHAIR STETKAR: No, I understand -- the

5 steam generator is the steam generator. It all --

6 I'm not arguing about the steam generator because

7 that's always there. What I am concerned about is

8 are you taking an additional DF for releases, in

9 particular for iodine, through he main condenser

10 during that first 900 seconds?

11 MR. YOSHIDA: No, only steam generator

12 water retention.

13 CHAIR STETKAR: Just steam generator?

14 MR. YOSHIDA: Yes.

15 CHAIR STETKAR: So you are treating from

16 zero -- let me make sure that I understand it -- are

17 -- from zero you are treating all of the releases,

18 not just noble gases but iodine, cesium, anything

19 else, as if it was a direct atmosphere release?

20 As if you were steaming from that --

21 let's say through the main steam safety valve. Is

1 that true?

2 MR. YOSHIDA: Yes, and we assume also

3 flushed break fall, so iodine, including this

4 flushed break fall, we assume they have to be

5 released to the environment. Can you understand my

6 English?

7 CHAIR STETKAR: I think so. And let me

8 just say it back, that you have assumed the iodine

9 is released offsite. Okay. From zero?

10 MR. YOSHIDA: Yes.

11 CHAIR STETKAR: Okay.

12 MR. YOSHIDA: Okay.

13 CHAIR STETKAR: So the timing of the

14 reactor trip and the isolation -- effectively the

15 timing of the reactor trip, what I'm trying to

16 understand is, is the dose analysis affected by the

17 timing of the reactor trip, and the closing of the

18 main steam isolation valve, which changes the

19 release from the condenser to the main steam relief

20 valve that is assumed to stick open at that time,

21 and from what I understand, you are saying no, that

1 time does not affect the dose analysis. Okay, thank
2 you. That helps. Thank you.
3 MR. MARUYAMA: Okay.
4 MR. LYNN: Before we move on, I want to
5 clarify. You asked earlier about the main steam
6 relief valve on the ruptured SG sticking open and
7 how that's closed I said that it was -- you said at
8 1826 seconds, and I said that it was, I believe that
9 it was at an operator action, but I did confirm that
10 the closure of that valve at 1826 is an automatic
11 signal that happens due to low main steam line
12 pressure.
13 CHAIR STETKAR: Thank you.
14 MR. MARUYAMA: The main purpose of this
15 analysis is to determine the primary-to-secondary
16 leakage and the release rate from the ruptured
17 interfaces to use as input in the dose analysis as
18 we discussed.
19 However, the result also shows that DNBR
20 remains above the safety analysis limit, and RCS and
21 SG pressures are within limits.

1 Both large-break LOCA and small-break
2 LOCA event are presented as represented event for
3 section 15.6. The large-break LOCA analysis uses
4 the ASTRUM methodology with WCOBRA/TRAC and HOTSPOT.
5 The event begins with a cold leg pipe
6 break and a coincident loss of offsite power.
7 CHAIR STETKAR: I'm sorry. I was
8 writing, and I -- before we start discussing the
9 LOCA, regarding the steam generator tube rupture, in
10 the Safety Evaluation Report, there is some
11 discussion about operator actions and the amount of
12 time available and the amount of time required to
13 perform the operator actions, and in the SER --
14 you'll see why I asked you in a moment her -- it
15 says that in the MHI response to RAI 808-5921,
16 question 15.06.03-3 -- do this so it's on the record
17 -- and now I've lost my train of thought. Thank you
18 so much.
19 The -- don't you have other things to
20 do?
21 (Laughter.)

1 But you supplied a table that showed the
2 times available for each of the operator actions and
3 the time that was required, and you noted that
4 though the available times and the instructions for
5 the operator actions were consistent with your
6 emergency response guidelines. I don't have the RAI
7 response. We can get that. That's not a problem.
8 My question was, it was also noted in
9 that, in the SER, that you had indicated that the
10 ERGs would be available by the end of December 2011.
11 My question now is are they available? Have you
12 finished the ERGs?
13 MR. LYNN: I guess, to clarify your
14 question, available to whom or --
15 (Laughter.)
16 CHAIR STETKAR: No one except himself.
17 MR. LYNN: I wouldn't give them to Dr.
18 Powers.
19 CHAIR STETKAR: No, I -- okay. No. It
20 is a relevant question.
21 MR. SPRENGEL: I can answer that.

1 CHAIR STETKAR: Yes, thanks.
2 MR. SPRENGEL: They are completed.
3 CHAIR STETKAR: They are completed.
4 MR. SPRENGEL: But they are not going to
5 be submitted to the staff. So --
6 CHAIR STETKAR: And typically they are
7 not submitted to the staff because there is support
8 for the EOPs, which is typically a COL item which is
9 typically after the COL is issued, and I understand
10 that.
11 MR. SPRENGEL: But they are completed,
12 and --
13 CHAIR STETKAR: But they are completed
14 and they are available. Okay. But not for formal
15 review because they are not submitted as part of
16 this licensing.
17 MR. SPRENGEL: That's correct.
18 CHAIR STETKAR: Understand. Thank you.
19 Okay. Thank you. I'm sorry. Now you can talk
20 about LOCAs.
21 MR. MARUYAMA: Okay. Thank you. So,

1 about LOCA, in this stage I was explaining about the
2 large break LOCA. The event begins with cold leg
3 pipe break, and the coincident loss of offsite
4 power.
5 ECCS actuate on low pressurizer pressure
6 but only two of the four SI pumps, minimum
7 safeguards are assumed. The values of peak cladding
8 temperature, LOCA maximum oxidation and core-wide
9 oxidation in the ASTRUM analysis satisfy the 10 CFR
10 50.46 acceptance criteria.
11 The small-break LOCA analysis uses
12 MRELAP-5 based on the Appendix K methodology. The
13 event begins with a cold leg pipe break. Reactor
14 trip occurs on low pressurizer pressure and a loss
15 of offsite power is also assumed at the time ECCS
16 actuates on low pressurizer pressure.
17 But only two of the four SI pumps and
18 minimum safeguards are assumed. The values of PCT,
19 LMO and CWO in M-RELAP5 analysis satisfy the 10 CFR
20 50.46 acceptance criteria.
21 The LOCA event is also analyzed for

1 radiological consequences. A guillotine break is
2 assumed. All fuel rods are assumed to be failed and
3 the primary coolant activity is assumed to be at the
4 t spec limit initially.
5 The radioactivity in the fuel is
6 gradually released into containment and then escapes
7 from containment through specific release pathways.
8 Radioactivity is assumed to be removed from the air
9 by containment spray, natural deposition and the
10 filtering systems.
11 The results meet all of the SRP 15.0.3
12 acceptance criteria.
13 MEMBER POWERS: When you do your spray
14 calculation, what do you assume about your spray?
15 MR. YOSHIDA: Pardon?
16 MEMBER POWERS: I was wondering about
17 droplet size distribution, spray volume, unsprayed
18 volumes.
19 MR. YOSHIDA: Yes, we assumed that 60
20 percent of containment volume is sprayed region and
21 the remaining 40 percent of containment volume is in

1 dose analysis.

2 MEMBER POWERS: And this droplet -- have

3 you picked your spray hitters yet? Your spray

4 nozzles?

5 MR. YOSHIDA: No. I have not picked yet

6 to --

7 MR. HEAMES: This is Terry Heames.

8 Dana, are you talking about do they have the spray

9 nozzles set up for this particular design as opposed

10 to what we used for the dose analysis?

11 MEMBER POWERS: Yes.

12 MR. HEAMES: Okay. We can't answer, I

13 can't answer that one. Do you have a particular

14 spray nozzle that you have chosen to use the US

15 APWR? That's his question.

16 MEMBER POWERS: Well really all I want

17 to know is what the droplet size distribution is

18 that you set, so like, if you knew the nozzle, then

19 I'd probably know the droplet size distribution.

20 MS. STEINMAN: So, at this time, no we

21 have not selected --

1 MEMBER POWERS: So, how do you do a
2 spray decontamination if you don't know the droplet
3 size distribution?
4 MR. YOSHIDA: I use for spray model a --
5 we used SRP model for particulate, and for natural
6 division I used Powers model for --
7 MEMBER POWERS: What the hell does he
8 know?
9 (Laughter.)
10 MR. YOSHIDA: -- for particulate and for
11 natural division, for airborne iodine I also used
12 SRP model.
13 MR. MARUYAMA: There are seven open
14 items for section 15.6. All of the open items are
15 associated with the LOCA event. The first two open
16 items here are associated with the initial review of
17 related technical and topical reports.
18 The third open item, shown here, is
19 related to the RCP seal integrity issue. That is
20 not thoroughly a Chapter 15 item.
21 CHAIR STETKAR: Yes, and we had also in

1 our subcommittee meetings, extensive discussion over
2 that. So we are following that.
3 MR. MARUYAMA: These two items are
4 associated with technical reports for PARAGON/ANC
5 and for GSI-185. For these two open items, MHI has
6 provided a response to the associated RAI. These
7 responses are currently under NRC review.
8 This table shows all of the section 15.7
9 that are analyzed for radioactive release. Some of
10 these events are analyzed separately as a part of
11 Chapter 11. Only the fuel handling accident is
12 specifically analyzed in Chapter 15 using RADTRAD.
13 There are no items for section 15.7.
14 CHAIR STETKAR: Your analysis about the
15 cask drop says it's not physically possible in this
16 design to move the cask over the spent fuel pool.
17 Is that right? I think I recall that.
18 MR. YOSHIDA: As you guessed, our
19 design, we applied for single failure approved
20 claim, so spent fuel cask does not drop out to lower
21 floor and --

1 CHAIR STETKAR: That's a different --
2 I'm not asking you about the probability of dropping
3 it. I'm asking about the possibility of dropping
4 it. They are two different things.
5 MR. YOSHIDA: And then transfer pathway
6 of spent fuel cask does not over the spent fuel.
7 CHAIR STETKAR: Okay. That's what I
8 seem to remember. Thanks. That's a possibility
9 rather than a probability.
10 MR. YOSHIDA: Yes, I think I was asking
11 a possibility question.
12 CHAIR STETKAR: No, if it's not
13 possible, I can handle zero.
14 MR. MARUYAMA: There are no open items
15 for section 50.7. Then this is the section 15.8
16 covers anticipated transients without scram, ATWS.
17 The US APWR has an analog diverse actuation system
18 that is diverse from the digital normal reactor trip
19 system and engineering safety feature.
20 If reactor trip does not occur by the
21 normal reactor trip system, the DAS has an automatic

1 reactor trip that occurs on any of the three signals
2 showing here. The DAS also has automatic SI and
3 emergency feedwater actuation features.
4 MHI has performed defense D3 coping
5 analysis in the technical report, MUAP-07014 to
6 evaluate chapter 15 AOOs with a concurrent common
7 cause failure.
8 The D3 coping analysis of some event
9 creates automatic DS features. In all cases, the
10 acceptance criteria are met.
11 CHAIR STETKAR: Thanks, and just for the
12 record, we have not yet reviewed Chapter 7 of the
13 DCD. We will be doing that review, our first review
14 of that, in November of this year. So I am sure we
15 will be examining the DAS and automatic features at
16 that time.
17 MEMBER BROWN: Yes, I wanted to make
18 one, yes, one piece of input, in that there's only
19 three signals that result with this system. There
20 ought to be, when you present this, or when we
21 discuss this later --

1 CHAIR STETKAR: We'll get to it.
2 MEMBER BROWN: There ought to be a full
3 basis for why those three cover all other possible
4 combinations where other reactor trips are not done
5 and why they cover the waterfront, umbrella-wise.
6 So there ought to be a good basis for that, as to
7 why there's a small microcosm of the reactor trips
8 actually will actuate from the diverse actuation
9 system.
10 CHAIR STETKAR: Take that as a preview
11 for your I&C folks who are not here who are --
12 MEMBER BROWN: It just gets better --
13 CHAIR STETKAR: not here today.
14 MEMBER BROWN: It just gets better,
15 right?
16 MR. MARUYAMA: Thank you very much.
17 MEMBER BROWN: Thank you.
18 MR. MARUYAMA: Okay. There is one other
19 item --
20 MEMBER BROWN: Oh, one other point,
21 okay? On the diverse actuation for engineered

1 safeguards features. And this will help me
2 understand this when you all get to this point,
3 those are typically fail as -- ESF systems are
4 typically fail as is, as opposed to fail to actuate.
5 I presume they are the same in your plant as they
6 are in other ones, I'm making that assumption. And
7 John is shaking his head at that, so he obviously
8 knows.
9 How -- one of the other questions I will
10 be asking is how, if they are fail as-is, what is
11 the backup if in fact you really did need an ESF
12 actuation, but because of a failure in your
13 automatic normal systems as well as a failure in
14 here, don't allow that system to actuate. How do
15 you handle those?
16 So that's just a heads-up on part of
17 this interface between the two, particularly in the
18 automatic, normally automatic actuation systems,
19 because they are fail as-is. Now all of a sudden
20 you need it, uh-oh, it didn't actuate, now what are
21 the operator response times and other monitoring

1 things that may allow operator action to take, and
2 is that good enough from your safety -- from your
3 ECCS and other type systems? So, just a question.
4 CHAIR STETKAR: The way I look at this
5 is, we are obviously going to have discussions about
6 both the normal protection system and the diverse
7 systems.
8 If those discussions identify any areas
9 of concern, they might feed back into the accident
10 analyses if they are not resolved. I mean, but for
11 the moment, until we have had the opportunity to
12 kind of finish those reviews, it's -- we don't know.
13 I mean, you have done, for the accident
14 -- right now, you have done for the accident
15 analysis what you need to do.
16 MR. MARUYAMA: Thank you for the
17 comment. I really appreciate your comments. And so
18 the next slide is the last slide of this
19 presentation, explaining Appendix 15A in Chapter 15,
20 which provides additional details on the models and
21 parameters used for radiological consequences

1 analysis.

2 This detail includes description of the

3 RADTRAD code, description of the model for the

4 analysis, and associated input parameters,

5 assumptions for offsite doses, and assumptions and

6 models -- model descriptions for the main

7 contributing dose.

8 There are no open items for Appendix

9 15A.

10 MEMBER POWERS: When you do your

11 analysis on the main control room dose, what's your

12 unfiltered in-leakage?

13 MR. YOSHIDA: For dose analysis

14 initially, we assume filtered intake and unfiltered

15 in-leakage.

16 MEMBER POWERS: And how much is the

17 unfiltered in-leakage?

18 MR. YOSHIDA: Filtered intake, intake

19 flow is 1200 cfm and unfiltered in-leakage flow,

20 it's 120 cfm.

21 MEMBER POWERS: 120 cfm. Substantial.

1 Biggest problem is how you ensure that that stays
2 for all time, but that's a licensee problem.
3 CHAIR STETKAR: That's a -- that's
4 keeping the people from coming in and out of the
5 doors.
6 MEMBER POWERS: Or keeping the
7 maintenance guy from not putting a hole in the
8 building.
9 CHAIR STETKAR: Well, that too. There
10 are some ways to be able to do that.
11 MEMBER POWERS: Well, some plants
12 haven't found it.
13 CHAIR STETKAR: I know. I know.
14 MEMBER POWERS: Thank you.
15 MR. MARUYAMA: Okay, this is the end of
16 this presentation. So thank you for your attention.
17 We will welcome your comments or questions.
18 CHAIR STETKAR: Any members have
19 anything else for MHI? If not, thank you very much.
20 You covered a lot of material very efficiently, and
21 I really appreciate that. I thought it was a very

1 good presentation, and we will recess until 3:20.
2 (Whereupon, the above-entitled matter went off the
3 record at 3:02 p.m. and resumed at 3:20
4 p.m.)
5 CHAIR STETKAR: We are back in session.
6 Mike, it's your show again.
7 MR. TAKACS: Okay. Good afternoon
8 Chairman Stetkar and ACRS members. My name is Mike
9 Takacs. I am the chapter PM for this Chapter 15.
10 Now we are moving into the DCD overview of the
11 safety evaluation. Unlike the topicals, this will
12 have open items and that's where we will focus the
13 presentation on this afternoon.
14 The first slide here of course mentions
15 the reviewers that are involved. You'll see Jeff
16 Schmidt and Fred Forsaty who will be presenting
17 along here. And of course the same staff members
18 from the Information System Laboratories are here as
19 well in the audience.
20 And with that we are going to go ahead
21 and move into the technical side of the

1 presentation. Jeff?

2 MR. SCHMIDT: Thanks, Mike and I'm Jeff

3 Schmidt from the NRC staff. Just going to go over

4 kind of the major review areas and then the as Mike

5 said, cover the open items that we have.

6 You know, we reviewed, obviously, the

7 plant characteristics and initial conditions

8 assuring that they are appropriate, conservative,

9 and that the acceptance criteria are met for each

10 one of the Chapter 15 accidents; reviewed the worst

11 single failures to make sure that they are

12 appropriately chosen and analyzed with respect to,

13 again, the acceptance criteria; loss of offsite

14 power was also reviewed as part of each Chapter 15

15 accident and the review of long-term cooling. Next

16 slide.

17 Review of sequence of events for each

18 one of the scenarios to evaluate the mitigating

19 systems, instrument responses, and assumed operator

20 actions if there were any; assure that the

21 transients are supported by the design of the plant

1 systems and future EOPs. Like you said, we haven't
2 seen the EOPs and it sounds like we won't see the
3 EOPs so --
4 Review of the analyses include transient
5 predictions to assure that the consequences of each
6 analyzed limiting case meet the specific criteria
7 again and all the regulatory requirements.
8 So we are going to list the open items
9 here.
10 CHAIR STETKAR: Jeff, I wanted to ask
11 you, this is kind of a procedural issue. I have a
12 few questions that I wanted to ask the staff, but
13 essentially none of them are related to the open
14 items. They are related to other issues.
15 Is it -- would you prefer to go through
16 the open items first and kind of get through that
17 part of the presentation and then we'll handle the
18 questions? Okay, let's do it that way then.
19 MR. SCHMIDT: The first open item we
20 have is the general reload methodology topical
21 report, with the staff's interactions with MHI there

1 were, I'd say, a fair number of changes to the
2 Chapter 15 accident analysis, and the staff is just
3 keeping this one open to do another review of the
4 reload methodology to make sure it's consistent with
5 the Chapter 15 changes, really, and that's why we
6 have it open.
7 We are going to make another pass
8 through it to make sure it's consistent with the
9 changes we have made in Chapter 15, which like I
10 said were fairly significant.
11 So the second open item there is the
12 non-LOCA methodology. That we are going to come I
13 guess in October, right Mike, with the non-LOCA
14 topical report?
15 MR. TAKACS: That's correct, yes.
16 MR. SCHMIDT: I believe that's with the
17 OGC?
18 MR. TAKACS: It will be today.
19 MR. SCHMIDT: It will be today. Okay.
20 So that's with the OGC. It's, right now it's out of
21 the staff's hands with the OGC.

1 Thermal design is also with management
2 for signatures, which is the VIPRE topical report,
3 there's some thermal analysis, so a lot of these are
4 just keeping track of open items that support
5 Chapter 15, like the non-LOCA topical and the VIPRE
6 topical report.
7 Next slide. We talked about large break
8 still being open because of the accumulator issues.
9 That's of course is an open item in Chapter 15.6-5.
10 The same with small break LOCA and then on 15.8,
11 which is the ATWS, the staff is still going through
12 the coping analysis for the Chapter 15 events that
13 assume DAS operation and common cause failures. Our
14 input will be provided to the Chapter 7 folks.
15 Again, 15.4-4, as MHI mentioned, there
16 is still an open item on control rod misalignment or
17 misoperation. I'm still looking to see whether I'm
18 comfortable that all the reasonable misalignments
19 have been covered. So there's still an open item
20 with that.
21 Startup of an inactive loop, still has

1 questions about lower mode operation, where in modes
2 1 and 2, RCPs are all on, you know, there's no
3 additional ones to start.
4 Lower modes, you can still have one RCP
5 and you can have another one start, staff is still
6 questioning whether there's adequate evaluation for
7 the lower mode operation.
8 And then boron dilutions in modes 4 and
9 5, with RCPs running, that's still an open issue.
10 We are still working on some wording in tech specs
11 that the initial wording on the latest review
12 implied that there was -- there could be some
13 dilution when no RCP is running. The staff is a
14 little uncomfortable with that because of mixture
15 issues so we are still reviewing the wording on the
16 tech specs change for RCPs in the lower mode
17 operation for mixing of boron.
18 Again, this is just a tracker one, also,
19 15.6-5 is a sump strainer that affects, could
20 potentially affect the LOCA analysis as far as the
21 SI pumps, so we are leaving that as an open item.

1 Accumulator topical report we talked
2 about a couple of times at this point, we are still
3 waiting to finalize the accumulator topical report.
4 The main issue there is the scaling bias.
5 Open item 15.6-3, which is a TMI action
6 item. Again, this is kind of a tracking item as far
7 as 15.6-5 is concerned. This has to do with the
8 number 2 RCP seals and how well it would last under
9 station blackout. We are working with DE and the
10 Chapter 9 and Chapter 8 folks on that.
11 Again, this is not primarily handled
12 here but it is brought up in 15.6-5.
13 Our first long term cooling issue, which
14 is just another open item, a tracking item for the
15 ANC code, which is the physics code, advanced nodal
16 code. It's Chapter 4.3. We will check if 4.3 is
17 completed and it's going through, I believe, OGC.
18 Again there's no real issues -- technical issues
19 there, it's just going through the internal process.
20 And I'm going to turn over to Fred now
21 for the long term -- the rest of the long-term

1 cooling issues.

2 MR. FORSATY: Thank you. Again, this is

3 Fred Forsaty. I am the engineer responsible for the

4 long-term cooling. Here we have two topics that are

5 open, three RAIs.

6 The first topic has to do with the GSI-

7 185, small-break LOCA, recriticality mixing, boron

8 dilution. They just completed -- Mitsubishi just

9 completed a test. They have the result of the test

10 back. We are going through that. It's under

11 review.

12 That's for boron dilution, then we have

13 two RAIs on boron precipitation, 93 and 100. 93 is

14 basically -- the question is simple, explain to us

15 at what time boron would precipitate and what's the

16 switchover time.

17 Their analysis and their response to the

18 RAI basically are based on the BACCUS test result

19 and the -- we are doing some confirmatory run on

20 that to understand, you know, their four and a half

21 switchover time.

1 And then RAI-100 is basically RAI-03
2 with debris. You are asking to let us know what the
3 boron precipitation is in case you have a debris in
4 the lower plenum.
5 And that again, once we complete our
6 review of 93 and understand the proper -- if you
7 accept the boron precipitation, and its switchover
8 time, then we would look at that and its impact on
9 the debris on the same two issues. So --
10 MEMBER BANERJEE: Are you doing any
11 analysis yourself or are you just waiting for their
12 response right now?
13 MR. FORSATY: A combination of both. On
14 boron precipitation, we are doing some confirmatory
15 runs. Mitsubishi is helping us to give us the
16 boundary condition, the information that we need.
17 The confirmatory runs should be probably complete
18 within a month or two at the latest.
19 And once we finish that, I think we can
20 have a better feel of what we need to do with the
21 presence of debris in RAI-100.

1 MEMBER BANERJEE: So remind me of what
2 the current switchover time is?
3 MR. FORSATY: Four and a half hours.
4 MEMBER BANERJEE: And is that -- are you
5 reviewing that as well to see if it --
6 MR. FORSATY: No, we are just -- asking
7 questions to ensure that that four and half hours
8 time is accurate.
9 MR. SPRENGEL: Small correction, the
10 switchover time right now is four hours.
11 MR. FORSATY: Four hours, I'm sorry,
12 four hours. I think the --
13 MEMBER BANERJEE: So we just have to
14 wait to see what happens here.
15 MR. FORSATY: I'm sorry?
16 MEMBER BANERJEE: We just have to wait
17 to see what happens here.
18 MR. FORSATY: Yes, it should be we need
19 a month or so, and we should complete our
20 confirmatory run hopefully by then.
21 MEMBER BANERJEE: Have they made any

1 assumption about the amount of lower plenum mixing
2 for this already?
3 MR. FORSATY: Yes, they have some
4 assumptions --
5 MEMBER BANERJEE: After lower plenum?
6 MR. FORSATY: Some, and some of the
7 BACCUS test result is also issued as a part of the
8 response. The -- I think some of the information
9 that I have to discuss probably is better to discuss
10 it tomorrow.
11 MEMBER BANERJEE: For Friday? Okay.
12 Okay. Somebody will be here from the staff tomorrow
13 to --
14 MR. FORSATY: I'm sorry?
15 MEMBER BANERJEE: Somebody will be here
16 from the staff to tell us --
17 MR. FORSATY: I'll be here too.
18 MEMBER BANERJEE: give us a little bit
19 of --
20 MR. FORSATY: Yes, I will discuss with
21 you all the information you need. I think I'm done

1 here.

2 MR. TAKACS: And that's the end of the

3 open items.

4 CHAIR STETKAR: That was good.

5 MEMBER BANERJEE: Everything is

6 proprietary.

7 CHAIR STETKAR: Now we're getting not so

8 good. I have a few questions. First one, and I'll

9 -- these are not in any particular order, so I'll

10 just walk through them -- as I understand it, MHI

11 revised their pressurizer high-level tech spec so

12 that now the maximum level during plant power

13 operation is 60 percent I think.

14 And as I understand it -- correct me if

15 I'm wrong -- if they have a feedwater line break

16 with level at that maximum level, the pressurizer

17 will go water solid and they will have water relief

18 through the pressurizer safety valves. Is that

19 correct?

20 MR. TAKACS: That's correct.

21 CHAIR STETKAR: Okay, and safety valves

1 are not qualified for water relief, is that correct?
2 MR. TAKACS: That's correct.
3 CHAIR STETKAR: Okay. Now, as I read
4 the SER, you are okay with that. I mean, the
5 analysis that MHI, the analysis of record in the DCD
6 is initiated at normal pressurizer level plus
7 uncertainty.
8 MR. TAKACS: Plus uncertainty, right.
9 CHAIR STETKAR: But it's a substantially
10 lower level than the 60 percent, and they show no --
11 they show small but margin --
12 MR. TAKACS: Yes, right.
13 CHAIR STETKAR: to filling the
14 pressurizer at that level.
15 MR. TAKACS: That's correct.
16 CHAIR STETKAR: In the SER, section
17 15.0.0-4, you say that the staff agrees that water
18 relief is allowed for postulated accidents, and you
19 find that the analysis is acceptable.
20 Now, the question I have, is do you
21 allow water relief of a postulated accident because

1 LOCAs are postulated accidents so it's okay for a
2 feedwater line break to also transition to a LOCA?
3 MR. TAKACS: Well, in the case where it
4 starts at nominal plus uncertainty, it doesn't,
5 right?
6 CHAIR STETKAR: I understand that but
7 everything else, everything else is done --
8 MR. TAKACS: From the tech spec value --
9 CHAIR STETKAR: From the tech spec
10 value.
11 MR. TAKACS: Right. Yes. Effectively,
12 what we were trying to do there, is -- I don't know
13 if you are familiar with the older Westinghouse tech
14 spec, where it was around 92 percent level. What we
15 were trying to do is say that's an unreasonable
16 level to be at, because there is no margin even for
17 AOOs to occur.
18 CHAIR STETKAR: Sure right.
19 MR. TAKACS: So we brought it down for
20 AOOs that you have got to survive to your tech spec
21 level.

1 CHAIR STETKAR: Okay.

2 MR. TAKACS: For postulated accidents,
3 you are right, we started at the nominal plus
4 uncertainty and said okay, well, if you -- if -- you
5 know the idea is not to pass water. If you do at
6 the tech spec value and you do pass water, then yes,
7 you have a LOCA that's effectively been analyzed.

8 CHAIR STETKAR: And it's okay to have a
9 feedwater line break transition to a LOCA?

10 MR. TAKACS: In this case, we said yes.
11 I mean it's not unusual, for instance another
12 example would be Palo Verde has that nominal plus
13 uncertainty for their --

14 CHAIR STETKAR: Okay that's what I
15 wanted to ask, is, because I'm personally less
16 familiar with design basis accident analysis rules.

17 MR. TAKACS: Right.

18 CHAIR STETKAR: If -- there are other
19 currently-licensed plants that could have that same
20 condition.

21 MR. TAKACS: That's correct.

1 CHAIR STETKAR: Okay, right. That's --
2 MR. TAKACS: Well, actually if you want
3 to really look at it, a lot of licensees allow it to
4 be at 92 percent.
5 CHAIR STETKAR: Which, yes --.
6 MR. TAKACS: Clearly can't survive
7 anything.
8 CHAIR STETKAR: Which you clearly can't
9 survive anything. So --
10 MR. TAKACS: Right. So our, our -- our
11 goal is to --
12 CHAIR STETKAR: Yes, that -- that's
13 enough. That helps. That helps me. Let me write a
14 note here, only because I -- my memory span is about
15 15 seconds these days.
16 All right. One question and I had, and
17 this is a nit but I hate things that are not
18 accurate, and perhaps it is, but I need to be
19 educated if it is. In many places in the DCD, and
20 consequently in the SER, there are statements
21 regarding a single failure of the reactor trip

1 system, and you know, one of the quotes that I
2 excerpted was, "A single failure was assumed in one
3 train of the reactor trip system, but this has no
4 impact on the safety analysis because any one of the
5 remaining trains is adequate to provide the trip
6 function."
7 As best as I can understand the reactor
8 trip system, that statement is not logically
9 correct, because the tech specs, for example for
10 instrumentational, require that you have three
11 channels of instrumentation available, three, if one
12 of them is removed from service then you must trip
13 that one, which leaves you two.
14 MR. TAKACS: Right.
15 CHAIR STETKAR: If you have a single
16 failure of one of those -- what I'm getting to is
17 you can have one out for maintenance that is not
18 tripped, and a single failure, and one more will not
19 get you a trip function. You need, you need two
20 more trains of censors and if you look at the logic
21 that -- the train logic to the reactor, I'll call

1 them reactor trip breakers, the same is true.
2 So it's not clear to me, and I know -- I
3 know this was excerpted directly from the DCDs so
4 it may be more of a question to MHI. It doesn't
5 make any difference in the analyses, because you can
6 still survive a single failure, but characterizing
7 that single failure with the notion that any one of
8 the remaining things is good enough, is, I don't
9 think, quite accurate. So you may want to re-look
10 at --
11 MR. TAKACS: Right, yes, I think you're
12 right.
13 CHAIR STETKAR: at that in particular.
14 MR. TAKACS: We need to revisit that.
15 CHAIR STETKAR: I don't think it
16 changes, and I tried to think about it every time I
17 bumped into it, and I don't think it changes any of
18 the analyses, I don't think it changes your
19 conclusions about whether reactor trip is the -- you
20 know, whether you assigned the single failure to
21 reactor trip or to a different function. But you

1 may want to look at that. As I said, that's a --
2 it's a nit but --
3 MR. TAKACS: I think you're right.
4 CHAIR STETKAR: But giving the wrong
5 impression about the available redundancy in things
6 is something that we should be careful about.
7 MR. TAKACS: Right, I agree.
8 CHAIR STETKAR: Okay. Oh, this -- I
9 tried to follow the story about when loss of offsite
10 power is a assigned throughout the spectrum of
11 transient events, and -- transients and LOCAs -- and
12 it kind of is triggered at different times for
13 different transients.
14 MR. TAKACS: Right.
15 CHAIR STETKAR: For most transients, I
16 think it's three seconds after the reactor trip
17 occurs --
18 MR. TAKACS: Right.
19 CHAIR STETKAR: And there might be
20 different rationales for why those -- what the three
21 seconds, and I don't want to get into that. It's

1 kind if three -- you've got four circulations for
2 like three seconds, after the trip.
3 You did ask several questions, as I
4 understand it, for the turbine trip transient
5 analysis with specific regard to the timing of the
6 loss of offsite power and I think I followed that
7 whole story. I -- you seemed to probe very well in
8 that area.
9 The conclusion in the SER is that, it
10 says, "The analysis results show that while turbine
11 trip is the most limiting Chapter 15 event with
12 respect to primary system pressure, no acceptance
13 limits are exceeded."
14 Now, all of this is kind of derived from
15 RAIs and things that I don't have immediate access
16 to. I always tell people, and I don't want immediate
17 access to it, because I can't read the things that I
18 have immediate access to yet.
19 So, don't view that as a request. Then
20 when I look at -- I'm trying to read my notes here -
21 - then when I look at the results for the main

1 feedwater line break, there's a statement in the SER
2 that says this is the most limiting Chapter 15 event
3 with respect to primary pressure.
4 So I have two, I have a turbine trip
5 with the updated analysis of a consequential loss
6 of offsite power and there was some timing analysis
7 that apparently went through to identify the most
8 conservative time for loss of offsite power. I
9 don't know what that -- when that is, whether it's -
10 - I just don't know what that time is.
11 But apparently some optimization was
12 done and the conclusion is that that turbine trip
13 with that, if I can call it an optimized loss of
14 offsite power is the most limiting for primary
15 pressure in one part of the SER, and in another
16 part, there is the feedwater line break with some
17 consequential timing of loss of offsite power.
18 Do you happen to know which -- I mean,
19 are they really lose? I --
20 MR. SCHMIDT: I don't remember off the
21 top of my head.

1 CHAIR STETKAR: I kind of lose the sense
2 of how the plant is running for each of these
3 things.
4 MR. SCHMIDT: Michelle is here to help
5 me.
6 CHAIR STETKAR: Okay.
7 MR. SCHMIDT: Because I think we ended
8 up with turbine trip was the answer, right? So
9 Michelle is helping me here, and she is saying it's
10 turbine trip is the limiting case. She was just
11 mentioning that she thought that one was maybe like
12 a PA and one was a --
13 CHAIR STETKAR: Okay, that may very well
14 be, but it just says most limiting Chapter 15 --
15 MR. SCHMIDT: Yes, okay.
16 CHAIR STETKAR: -- which kind of lumps
17 everything together but --
18 MR. SCHMIDT: Yes, that doesn't really
19 break it out, right.
20 CHAIR STETKAR: If you do -- okay. If
21 you do separate them --

1 MR. SCHMIDT: So we've got to make that
2 clear.
3 CHAIR STETKAR: Yes.
4 MR. SCHMIDT: Okay.
5 CHAIR STETKAR: All right. I was just
6 trying to get my -- I do have sort of an intuitive
7 feel, I don't have an intuitive feel for LOCA
8 response on this plant. But transient response I
9 kind of have a feel for, and --
10 MR. SCHMIDT: Yes, I know we had a lot
11 of questions on turbine trip, and turbine trip
12 became a more limiting event --
13 CHAIR STETKAR: Okay, and I could --
14 when I finally worked through all of that, I could
15 kind of understand why.
16 MR. SCHMIDT: Right.
17 CHAIR STETKAR: And then got all of that
18 sent and then I went to the feedwater and I said
19 well wait a minute --
20 MR. SCHMIDT: You went to the feedwater
21 and it said the same.

1 CHAIR STETKAR: I thought this other --
2 okay.
3 MEMBER BANERJEE: How close does it get
4 to the pressure --
5 MR. SCHMIDT: Michelle, what was the
6 values? They were pretty close. What were the feed
7 values ===
8 MEMBER BANERJEE: 2200 or something?
9 MS. HAYES: 2635 and 2616.
10 CHAIR STETKAR: They're fairly close.
11 MR. SCHMIDT: Yes, they're fairly close,
12 yes.
13 CHAIR STETKAR: I mean, you know,
14 percentage, they are fairly close, absolutely.
15 MEMBER BANERJEE: But turbine trip is
16 limiting, right?
17 CHAIR STETKAR: Turbine trip, as I
18 understand it, an optimized time for when you lost
19 offsite power. The plant -- and I still don't quite
20 understand why it's so sensitive to keeping forced
21 flow for a short period of time. There seems to be

1 a lot of stuff -- a lot of attention to that.
2 MR. SCHMIDT: think about that. There
3 was a lot of attention paid to that. Andy, do you
4 remember why? No, okay.
5 CHAIR STETKAR: Okay.
6 MR. SCHMIDT: I'll have to get back to
7 you. I don't remember why. There was a reason.
8 CHAIR STETKAR: I mean it's kind of an
9 interesting curiosity, but it's turbine trip with
10 loss of the forced flow at just the right point to
11 optimize the pressure transient, and I actually
12 don't know what that time delay is.
13 MR. SCHMIDT: We'll get back to you on
14 that. I just have to review it again.
15 CHAIR STETKAR: I mean I haven't read --
16 it's probably buried in all of those responses. I
17 just haven't read them nor do I want to.
18 But if you could find it --
19 MR. SCHMIDT: We'll give you a summary
20 tomorrow.
21 CHAIR STETKAR: That would be good.

1 Thanks. Now, and here's another case where I don't
2 know if it makes any difference, so please tell me
3 whether it does. There was quite a bit of
4 discussion about whether or not the uncontrolled
5 withdrawal of a single rod cluster control assembly,
6 RCCA, is an AOO or a PA.
7 And apparently you asked questions about
8 it, MHI classified it as a postulated accident.
9 They apparently did a failure modes and effects
10 analysis, did some fault tree work, did some
11 quantitative analysis to show that the frequency --
12 it would require more than a single failure and that
13 the frequency was less than frequencies of other
14 postulated accidents, and you essentially said,
15 well, you have the latitude of throwing it into one
16 box or another box, and you agreed that it could be
17 thrown into the postulated accident box.
18 Functionally, does it make any
19 difference whether it's a --
20 MR. SCHMIDT: What do you mean by
21 functionally?

1 CHAIR STETKAR: In terms of the -- in
2 terms of the consequence analysis for design basis
3 accident analysis.
4 MR. SCHMIDT: You know, you know, if you
5 make their conservative assumptions for single rod
6 withdrawal, you end up needing to go to a PA because
7 you failed fuel.
8 CHAIR STETKAR: Okay.
9 MR. SCHMIDT: If you use, say, a best
10 estimate neutronics code that's, you know, space
11 time dependent, I don't think you get to that
12 failure point. But they -- they took a simplified
13 method and said look, I am going to use a simplified
14 method, I'm going to end up with a PA and I'm going
15 to do a dose analysis on it.
16 And we accepted that because it took
17 more than one single failure to get there.
18 CHAIR STETKAR: Okay. Okay. I think I
19 understand that rationale.
20 MEMBER BANERJEE: So what is the
21 simplified method that they used?

1 MR. SCHMIDT: They used just a large
2 reactivity withdrawal rate, overly large reactivity
3 withdrawal rate. I guess I don't want to go too
4 much into the details until tomorrow, but --
5 CHAIR STETKAR: Okay, let's talk about
6 it tomorrow then a little bit.
7 MR. SCHMIDT: Okay.
8 CHAIR STETKAR: The question I had, is -
9 - and I've looked at a few plants and just arguing
10 that it takes more than one failure doesn't satisfy
11 me because a lot of plants take more than one
12 failure
13 and arguing about whether or not a
14 single human error could do it, well, I've done
15 human reliability analyses and you could argue
16 whether or not that's the case for any other plant,
17 as you can argue about, you know, numerical
18 estimates of how frequently this kind of thing
19 occurs.
20 So I'm more concerned about whether or
21 not there is an actual difference in the design of

1 this plant and its core compared to other plants
2 that makes it more vulnerable to withdrawal --
3 uncontrolled withdrawal of a single rod assembly
4 compared to other plants.
5 MR. SCHMIDT: You know, I think MHI
6 could speak better to that, but you know, in my
7 opinion no, there is probably nothing different in
8 their design versus anybody else's.
9 You know, it takes an operator action to
10 go into an individual, select mode for an RCCA, and
11 you usually only do that if you have a dropped rod
12 and I don't even know many operators who will even
13 go recover a dropped rod.
14 So you know, you have to have that
15 single failure to occur, somebody purposely going in
16 there and going in to individual select to control
17 an individual RCCA cluster.
18 CHAIR STETKAR: Yes, but given that, it
19 is one of the design basis accident --
20 MR. SCHMIDT: And historically you know,
21 it has been evaluated as a PA. The SRP allows you

1 to go, either with a PA or an AOO. Again, if you
2 use a more detailed neutronics model and more best
3 estimate type calculations, even if you say best
4 estimate with uncertainties, reasonable
5 uncertainties for your physics model, you know, you
6 are going to end up with an AOO.
7 CHAIR STETKAR: Okay.
8 MR. SCHMIDT: It's just that
9 historically, another vendor has also analyzed this
10 as a postulated accident.
11 CHAIR STETKAR: Okay. That at least,
12 that perspective, as I said, I'm personally kind of
13 less familiar with design basis accident analysis,
14 so that helps.
15 MR. SCHMIDT: Right, you know, the SRP
16 does give you latitude to go both ways on this.
17 CHAIR STETKAR: Yes. Okay. Boron
18 dilution events, and I know you have an open item
19 regarding dilution with the reactor coolant pumps
20 running, I'm going to drop back to the events that
21 were analyzed.

1 In the events that were analyzed in
2 modes 3, 4 and 5, the analyses show time windows for
3 the operator to terminate those events in modes 3
4 and 5 of 16 minutes and mode 4 at shutdown of 19
5 minutes.
6 And apparently the conclusion was made
7 by MHI, and that's the reason I didn't ask them
8 about it was, the conclusion was made that because
9 16 minutes is longer than the 15 minute criterion in
10 SRP section 15.4.6, and obviously 19 minutes is even
11 much longer than that, therefore everything was
12 acceptable.
13 What's the technical basis for a
14 universal 15 minute acceptance criterion for
15 operator response that applies to every single plant
16 in the world with every single scenario and the
17 equipment configuration.
18 MR. SCHMIDT: I have to answer that it's
19 just the SRP criteria that it's probably more of a
20 human factors response time. They really have, you
21 know, significantly more time. I mean, the 16

1 minutes is effectively back-calculated to give
2 really effectively the core designers the maximum
3 margin they can for boron concentrations.
4 So in essence, they really have a lot
5 more time than the 16 minutes would imply. There
6 are other alarms that are -- that could be credited
7 earlier that would give them an indication of a
8 boron dilution event occurring.
9 So, in essence it says 16 minutes, but
10 it's really a back calculation of 16 minutes.
11 CHAIR STETKAR: Yes, I got that sense.
12 I mean I didn't look at the --
13 MR. SCHMIDT: Okay. It wasn't clear
14 initially to us.
15 CHAIR STETKAR: No, it -- well I mean
16 reading the DCD and then reading the SER, I kind of
17 got the sense that it was calculated such that if
18 these conditions were available, you know, did
19 apply, then indeed we could show that we
20 miraculously had 15 minutes and 17 seconds or
21 something like that.

1 MR. SCHMIDT: Right. Right.
2 CHAIR STETKAR: But where -- I guess we
3 just then rely on the human factors engineering to
4 determine whether or not it's, for this particular
5 design, feasible to accomplish the actions within
6 the time window that the -- now this is the formal
7 accident analysis, regardless of what you say about
8 how much additional time might be available in a
9 best estimate analysis.
10 This is indeed the basis for certifying
11 this design, so that in principle, I guess the COL
12 applicant would be required to demonstrate that
13 indeed they can meet that. Is that correct?
14 MR. SCHMIDT: Right. That -- right.
15 CHAIR STETKAR: Are there ITAAC, I mean
16 other than the generic -- other than the generic you
17 shall perform verification validation studies on all
18 of your -- all is not the correct word -- on a
19 sample of your human factors engineering tasks,
20 which is kind of a generic requirement, are there
21 specific ITAAC that are developed for showing those

1 -- the feasibility, demonstrating the feasibility of
2 the operator actions, with a margin for any operator
3 actions that are explicitly included in the design
4 basis accident analyses? You follow my line of
5 questioning?
6 MR. SCHMIDT: Right, no, there's no
7 specific, for example boron dilution ITAAC.
8 MR. FORSATY: But normally, in a
9 situation like this, I think what we -- the action
10 we take is to create a COLA item, and then that
11 would be kind of tracked.
12 CHAIR STETKAR: Well, that's what I was
13 going to, I mean essentially it's -- I understand
14 it's not done during the design certification. It's
15 done -- essentially the COL applicant, when they
16 perform their human factors engineering task
17 analyses, you know, the selection of the scenarios
18 that they use, essentially for their V&V of the
19 human factors engineering, in one sense, there could
20 be a COLA item that says, you know, this has to be
21 one of the ones that you select because it has

1 implications on the accident analysis in the
2 certified design.
3 MR. FORSATY: Yes, but that would be
4 creating a lot of these, like, for the containment
5 cleanliness programs, and you know that the
6 applicant or the licensee agrees to keep the debris
7 to such a level, and as a result of that, they have
8 got to conform to some of the --
9 CHAIR STETKAR: But that one's a little
10 bit different. That's sort of a status of the
11 plant. This in particular would affect the
12 population of scenarios that the COL applicant, or
13 at that time you'd be the COL holder because they
14 always do this stuff after the COL is issued.
15 MR. DONOGHUE: Mr. Stetkar, this is Joe
16 Donoghue. I think there's a general ITAAC in part of
17 the review, for Chapter 18 I think it is, on human
18 factors, but I want to get back to an answer on
19 anything specific to this event or other assumptions
20 for different events, all right?
21 CHAIR STETKAR: That's -- okay -- okay.

1 Because we -- okay thanks Joe.

2 MR. DONOGHUE: I mean I guess unless we

3 could do it today.

4 CHAIR STETKAR: Right, it isn't -- well,

5 it's just a matter of whether or not the staff in

6 their review, I you identify, for example, in your

7 Chapter 15 analyses, critical operator actions.

8 For example, this boron dilution is one,

9 steam generator tube rupture is another, whether or

10 not there is a vehicle for the staff to essentially

11 put a hook into the COLA, into the COL applicant

12 that says, well, because in Chapter 18 it's just

13 sort of a generic you'll go out and select a bunch

14 of scenarios and try to make sure that they are

15 important scenarios but there is no specification

16 about which ones you select.

17 MR. DONOGHUE: Yes, I don't recall us

18 specifically making that connection here.

19 CHAIR STETKAR: Good.

20 MS. HAYES: This is Michelle Hayes and

21 15.6.3, with those operator actions we tied it back

1 to the ITAAC.

2 CHAIR STETKAR: I saw that. That's why

3 I asked about this one.

4 MS. HAYES: that applies to all operator

5 actions for Chapter 15.

6 CHAIR STETKAR: Oh, all?

7 MS. HAYES: Yes.

8 CHAIR STETKAR: Oh okay. I must have

9 misinterpreted -- I thought it was --

10 MS. HAYES: assumed in the safety

11 analysis.

12 CHAIR STETKAR: I thought that only

13 applied to the tube rupture.

14 MS. HAYES: No.

15 CHAIR STETKAR: Since it was in -- okay.

16 I must have misinterpreted that then. Thanks.

17 That helps. That -- and that's in 15.6.3, you said?

18 MR. BIELEN: Yes.

19 CHAIR STETKAR: Okay. With that, that

20 is all that I had. Any other members have anything

21 for the staff? If not --

1 MEMBER BROWN: Just one question,
2 relative to your comment on the operator action of
3 15 minutes, how does that relate to the other reg
4 guides that talk about 30 minutes for operator
5 reaction to accident response and have some
6 justification for that, whereas there's none for --
7 I mean, we had this discussion on time available on
8 --
9 CHAIR STETKAR: You're looking at me as
10 if I would have an answer.
11 (Laughter.)
12 CHAIR STETKAR: It is a valid question.
13 There are many -- that's why I tend to fall back on
14 if an applicant takes credit for 15 minutes or 16
15 minutes or 19 minutes or 27 minutes or whatever
16 minutes in their accident analysis, there should be
17 justification somewhere that indeed the actions that
18 they are taking credit for can be accomplished --
19 can be accomplished, that they are feasible within
20 that time window, with some margin or some
21 confidence of reliability, because I've given up on

1 whether or not it should be 30 minutes or 15 minutes
2 or some arbitrarily assigned number that differ
3 depending on where you might find those numbers,
4 Charlie.
5 That's kind of my take and that's why I
6 was asking about whether, since the boron dilution,
7 you know, analysis of record basically takes credit
8 for operator response within 16 minutes' or 19
9 minutes' work, you know, in principle for some of
10 the other scenarios that they looked at was 61
11 minutes I think and 73 minutes.
12 I mean, the same questions in principle
13 would apply for those, that there be some, at least,
14 requirement for the people who developed the
15 training and the procedures and do the task analyses
16 for the human factors engineering, to demonstrate
17 that indeed people can do that, and if they can't
18 demonstrate that, that then says well, there's an
19 issue, because the certified design says that they
20 will do that.
21 MR. FORSATY: I think that as a part of

1 licensing the operation, some of these items are
2 built into the training, and I don't know with this
3 specific one, but I have seen some of these --
4 CHAIR STETKAR: But in -- the nice thing
5 about the process that we have now is there is a --
6 as opposed to the previous licensing process there
7 is a greater focus on human factors engineering, you
8 know, and before you kind of relied on training and
9 generic procedures and things like that.
10 Here, here it is an element of getting
11 the plant licensed and a lot of the demonstration is
12 left to post-COL issuance once the procedures are in
13 place and the training programs are developed.
14 But without kind of reminders to close
15 those loops, to put the hooks in there to say that
16 there are indeed some scenarios that do have some
17 implications on the licensing basis for the plant.
18 They can easily fall in the cracks, especially
19 things like boron dilution.
20 Tube ruptures are things -- people tend
21 to look at tube ruptures but boron dilutions, not so

1 much, and especially boron dilutions from low power,
2 you know, shutdown conditions that people tend not
3 to put a lot of care into, or at least traditionally
4 have not.
5 MR. SHUKLA: Yes, I think there is an
6 explanation for 30 minutes in the digital I&C ISG
7 33.
8 CHAIR STETKAR: We've run into that in
9 the past and the digital I&C world has been evolving
10 more toward the kind of, what I'd call a
11 performance-based assessment.
12 MEMBER BROWN: Well, yes I have no --
13 even there, there was a 30 minutes. If you can't
14 make a decision, you default to a certain thing, but
15 there's a time available, time required, analysis
16 required and that comes under human factors. But I
17 just polled these numbers in the analysis basis,
18 they are all -- 15 minutes is thrown around all over
19 the place, because I just took the 15 minutes and
20 just ploughed through the entire now and there's
21 lots of 15 minutes in there.

1 CHAIR STETKAR: There are, aren't there?
2 MEMBER BROWN: With no basis at all for
3 any of them, other than we make this assumption.
4 CHAIR STETKAR: Well, they're in the
5 Standard Review Plan.
6 MEMBER BROWN: I didn't say I agreed
7 with that. I just said they're all over the place,
8 okay and they're in direct contradiction to standard
9 other assumptions that us I&C guys make.
10 CHAIR STETKAR: Thanks.
11 MEMBER BROWN: You don't want me to talk
12 anymore, is that what you are trying to say?
13 CHAIR STETKAR: Pretty much.
14 MEMBER BROWN: Okay.
15 (Laughter.)
16 CHAIR STETKAR: I think we got it. I'll
17 go back and look at that 15.6.3, because as I said,
18 I saw that and I probably misinterpreted it as only
19 related to the tube rupture.
20 MR. DONOGHUE: Michelle just pointed out
21 that in 15.0.0-4 there is a discussion about

1 operator actions and it links to a COL and then the
2 ITAAC that she just -- 15.6.3, so it is --
3 CHAIR STETKAR: It does -- okay, I, I
4 apparently missed that link. So thanks. Anything
5 else for the staff? If not, thank you very, very
6 much. Now, we are suddenly -- suddenly have a
7 wealth of time on our hands and because so many
8 things were postponed to the ubiquitous tomorrow,
9 I've been told that the staff is prepared to discuss
10 the Comanche Peak information regarding Chapter 15.
11 Is that correct, Steven?
12 MR. MONARQUE: Yes, this is Steven
13 Monarque with New Reactors. Yes, we are prepared to
14 go ahead with that and discuss our safety
15 evaluation. And I would ask Luminant if they are
16 ready to discuss their slides.
17 CHAIR STETKAR: Is Luminant -- I hate to
18 sandbag you, but you have been sitting there so
19 quietly, and --
20 MR. WOODLAN: Yes Luminant is ready.
21 CHAIR STETKAR: Okay great. Let's see

1 if we can get that out of the way. I am assuming it
2 will be fairly painless to everyone concerned and it
3 will give us a little bit more time tomorrow.
4 There was some discussion about whether
5 even to have the poor folks from Comanche Peak come
6 all the way up here, but I just love seeing people
7 fly on airplanes and -- yes, and in some sense
8 actually to check off our boxes it's important for
9 you to at least present on the record what you have
10 done, what you do have, and the same is for the
11 staff. So Don, I guess this is finally your show.
12 MR. MONARQUE: If I could go ahead?
13 CHAIR STETKAR: Steven, I'm sorry. I
14 didn't want to --
15 MR. MONARQUE: Good afternoon. My name
16 is Steven Monarque. I am the lead PM for the
17 Comanche Peak COL application. I want to thank the
18 committee members for giving us this opportunity to
19 present the safety evaluation as well as giving
20 Luminant an opportunity to present their analysis to
21 the subcommittee today.

1 As you all know, today we present five
2 chapters to the subcommittee, and with that I will
3 go ahead and introduce Luminant.
4 MR. WOODLAN: My name is Don Woodlan. I
5 am the licensing manager with the new built project
6 for Luminant Power and I am here to talk, as Steve
7 said, about Chapter 15 as it relates to the Comanche
8 Peak FSAR and our SER.
9 Here's the -- the administrative slides
10 are longer than the meeting slides. The agenda, we
11 will do an introduction which I have kind of already
12 done, a subsection discussion and then a summary.
13 Chapter 15, I think all of you are
14 aware, is pretty much incorporated by reference.
15 This is standard plan information. We took no
16 departures from the work that was done by MHI for
17 Chapter 15.
18 The primary information that is in
19 Chapter 15 related to the Comanche Peak design is
20 really Chapter 2 information, which is the chi over
21 q values, and that is discussed in Chapter 2. I'm

1 sure it will be addressed in detail in the Chapter 2
2 SER. It is in our FSAR.
3 And the key is that the standard plant
4 values bound the Comanche Peak values so that all
5 the analyses that are done, especially with the
6 doses, remain applicable for Comanche Peak.
7 And we have no contentions pending with
8 respect to Chapter 15. In fact, our ASLB has been
9 closed.
10 CHAIR STETKAR: Don, where is your
11 emergency operations facility going to be for
12 Comanche Peak units?
13 MR. WOODLAN: It's on site. It's maybe
14 a quarter of a mile from the actual plant. It'll be
15 the same facility we use for units 1 and 2.
16 CHAIR STETKAR: Do you evaluate in your
17 COLA dose assessments for the EOF or is it just
18 limited to the main control room and the TSC?
19 MR. WOODLAN: No, I'm pretty sure -- let
20 me look real quick -- but I'm pretty sure it
21 includes the -- I don't have it here. It's

1 primarily the TSC and the control room, the EAD and
2 then the distant food generation areas.
3 CHAIR STETKAR: But since your EOF is
4 onsite, it's somewhere between the TSC and the
5 distant food generation?
6 MR. WOODLAN: Yes, it's outside the EAD,
7 so EAD values would bound it.
8 CHAIR STETKAR: Okay. Okay. Okay. And
9 it is outside the EAD?
10 MR. WOODLAN: Yes.
11 CHAIR STETKAR: Okay. Okay. Thanks.
12 MR. WOODLAN: Next slide. Don't go into
13 any details on the sections, because as I said up
14 front, without departures we are incorporating by
15 reference all the Chapter 15 sections. Last slide.
16 As I said, the key values that are in
17 the FSAR are the chi over q values. We are bounded
18 by the standard plant and we have no outstanding
19 issues in the SER.
20 And that concludes my presentation.
21 CHAIR STETKAR: Well, you did get a

1 question, so --

2 (Laughter.)

3 CHAIR STETKAR: Any of the members have

4 any questions for MHI?

5 MEMBER POWERS: For Luminant.

6 CHAIR STETKAR: I'm sorry, Luminant.

7 MEMBER POWERS: He wanted the questions

8 for MHI. It was okay with him.

9 CHAIR STETKAR: Yes, because they

10 incorporated everything -- I'm sorry. My mind is

11 turning to oatmeal. If not, thank you very much.

12 It was a brief and succinct --

13 MEMBER BROWN: And illuminating.

14 CHAIR STETKAR: I didn't want to say

15 that. I was, I was --

16 MEMBER POWERS: You have no self-control

17 whatsoever, do you?

18 MEMBER BROWN: No.

19 CHAIR STETKAR: Mike? It's yours.

20 MR. TAKACS: It's my turn? Okay, good

21 afternoon again, it's Mike Takacs, the chapter PM

1 for the reference COLA, Chapter 15, the safety
2 evaluation with open items.
3 As you heard earlier, all the sections
4 with the exception of one, all the sections are IBR
5 section -- before I bounce ahead let me point out
6 Michelle Hart is the main reviewer of this one
7 particular section that is not IBR. It is section
8 15.03 of the DBA, radiological consequences section
9 and you have probably seen that information that's
10 about two pages in the safety evaluation of the
11 total of seven pages.
12 So if there are any questions, this is
13 the end of my presentation.
14 (Laughter.)
15 MR. SHUKLA: And you were saving this
16 for tomorrow?
17 MR. TAKACS: I was going to offer to do
18 it today but I -- it's done.
19 CHAIR STETKAR: Anything from any of the
20 members? Thank you, also for keeping us on
21 schedule. It's, as I said, it's -- at some level it

1 sounds silly, but as part of our function to make
2 sure that indeed there is the opportunity for people
3 to ask questions if they have them, and make sure
4 that we have the presentation on the record so that
5 there is some record that we have at least discussed
6 the matter, and we have done that.
7 With that, I have -- Ryan is looking at
8 me as if there might be something that MHI wants to
9 contribute or --
10 MR. SPRENGEL: We can try to fill the
11 time.
12 CHAIR STETKAR: Let me, let me first do
13 something that I need to do. Well, were you going
14 to try to answer some of the questions from this
15 morning, or what did you want, what were you
16 proposing?
17 MR. SPRENGEL: Or consider -- we would
18 consider starting the presentations today, because I
19 think actually some of the presentations --
20 CHAIR STETKAR: Closing the meeting and
21 start --

1 MR. SPRENGEL: Yes.

2 CHAIR STETKAR: Okay. If -- if we are
3 going to consider that, let me do something first,
4 while the meeting is still open, and that is, are
5 there any members of the public here, Girija, can
6 you open up the bridge line if there is anybody out
7 there? Anybody from the public here who has any
8 comments?

9 Seeing none, I am going to wait a couple
10 of seconds and open up the bridge line to see if
11 there is anyone out there who wants to make any
12 comments or ask any questions. And then we can
13 consider closing the meeting so that I don't have to
14 go back and reopen it.

15 MR. SPRENGEL: I don't know if any of
16 the public are on the phone. So --

17 CHAIR STETKAR: I know we had a bridge
18 line open and there is always the opportunity for
19 someone to be out there, so I'll just wait to see if
20 there is.

21 MR. SHUKLA: The line is open.

1 CHAIR STETKAR: Okay. I've been told
2 the bridge line is open. If there is someone out
3 there listening in, would you do me a favor and just
4 make some sound so that I know that the bridge line
5 is open and that there is someone out there?
6 PARTICIPANT: Hello
7 CHAIR STETKAR: Thank you very much.
8 That at least confirms that we are open. Now, is
9 there -- are there any members of the public out
10 there who wish to make any comments or add anything
11 to the record?
12 Hearing none I will assume that we don't
13 have any other comments and we will close the bridge
14 line again, please. And now, members and MHI, it's
15 4:20. We do have a little time left on our agenda
16 for this afternoon. MHI has offered to start
17 presenting some of the proprietary information.
18 I'm certainly willing to entertain that.
19 I have absolutely no personal life whatsoever.
20 What are your constraints? And I know
21 you are not supposed to have any constraints, but I

1 try to be humane about these things.
2 COURT REPORTER:. I have a personal
3 life, but I'll give it up for the ACRS.
4 (Laughter.)
5 MEMBER BROWN: She's much more
6 accommodating than some of us are.
7 CHAIR STETKAR: So let's assume that we
8 can run for about an hour.
9 MEMBER REMPE: Is Sanjoy still here?
10 CHAIR STETKAR: And Sanjoy isn't here.
11 I'd like to find --
12 PARTICIPANT: Where is he?
13 CHAIR STETKAR: I don't know. It's not
14 my day to watch him. We should try to find Sanjoy
15 because he obviously would have an interest in this.
16 I would certainly, I think, I think any amount of
17 material that we can get in, we should probably try
18 to do that. I'd like to find Sanjoy and then we
19 will need to close the meeting.
20 Presuming that we can find Sanjoy, we
21 need to make sure that there is no one in the room

1 from MHI or the staff who are not privy to the
2 information. I have a little delay here while we
3 try to find Dr. Banerjee because without him I fear
4 that this exercise might not be worthwhile
5 MR. SPRENGEL: And would any portion of
6 the meeting be open tomorrow?
7 CHAIR STETKAR: At the -- I mean, we can
8 open and close meetings at will. At the moment, we
9 were not -- our agenda for tomorrow showed closed
10 presentations by MHI, and then opening the meeting
11 for the Comanche Peak discussion, but since we have
12 achieved that, right now I don't see any reason for
13 the meeting --
14 MR. SPRENGEL: My only concern is
15 discussion on planning, specifically the full
16 Committee.
17 CHAIR STETKAR: We can open it for that.
18 MR. SPRENGEL: Okay.
19 CHAIR STETKAR: If need be.
20 MR. SHUKLA: So, small-break first?
21 CHAIR STETKAR: Yes, I'd really like to

1 find Dr. Banerjee.

2 (Whereupon, the above-entitled matter

3 went off the record at 4:21 p.m. and resumed at 4:26

4 p.m.) CHAIR STETKAR: We're back in session. This

5 is an open session, but we are now going to close

6 the meeting to discuss MHI proprietary information,

7 and with that we will also close the bridge line so

8 that we do not share any of this information

9 publicly.

10 So I'll wait back to hear to make sure

11 the bridge line is closed. Again I'll ask the staff

12 and MHI to confirm that everyone who is in the room

13 is okay to be here, and I just want to make sure

14 that we hear back that the bridge line is closed.

15 (Whereupon, the open session was concluded at 4:28

16 p.m.)

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Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: ACRS USAPWR Subcommittee

Docket Number: n/a

Location: Rockville, Maryland

Date: July 10, 2012

Work Order No.: NRC-1738 Pages 1-15

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4 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

5 (ACRS)

6 + + + + +

7 U.S. APWR SUBCOMMITTEE

8 OPEN SESSION

9 + + + + +

10 TUESDAY

11 JULY 10, 2012

12 + + + + +

13 ROCKVILLE, MARYLAND

14 + + + + +

15 The Subcommittee met at the Nuclear

16 Regulatory Commission, Two White Flint North, Room

17 T2B1, 11545 Rockville Pike, at 8:30 a.m., John W.

18 Stetkar, Chairman, presiding.

19 SUBCOMMITTEE MEMBERS:

20 ***JOHN W. STETKAR, Chairman***

21 ***SANJOY BANERJEE, Member***

22 ***CHARLES H. BROWN, JR. Member***

23 ***JOY REMPE, Member***

24 ***WILLIAM J. SHACK, Member***

1

2 NRC STAFF PRESENT:

3 GIRIJA SHUKLA, Designated Federal Official

4 ANDREW BIELEN, RES

5 JOHN BUDZYNSKI, NRO

6 JEFF CIOCCO, NRO

7 JOSEPH DONOGHUE, NRO

8 FRED FORSATY, NRO

9 MICHELLE HART, NRO

10 MICHELLE HAYES, NRO

11 RALPH LANDRY, NRO

12 STEPHEN MONARQUE, NRR

13 NGOLA OTTO, NRO

14 EDUARDO SASTRE, NRO

15 JEFF SCHMIDT, NRO

16 AMY SNYDER, NRO

17

18 ALSO PRESENT:

19 ROBERT BEATON, ISL

20 DAVID CARAHER, ISL

21 JOHN CONLY, Luminant Power

22 CLIFF DAVIS, INL

23 HIROSHI FUJISHIRO, MHI

24 YUKO FUJITA, MNES

1 ED HAOREGO, INL
2 HIDEAKI IKEDA, MHI
3 MASAAKI KATAYAMA, MHI
4 SCOTT KIFFER, MNES
5 ATSUSHI KUMAKI, MNES
6 HIROSHI HAMAMOTO, MHI
7 TERRY HEAMES, Alion
8 KEVIN LYNN, MNES
9 YUTA MARUYAMA, MNES
10 REIKO NARUTOMI, MHI
11 JUNTO OGAWA, MHI
12 MARCOS ORTIZ, ISL
13 KEITH PAULSON, MNES
14 DAN PRELEWICZ, ISL
15 RYAN SPRENGEL, ISL
16 REBECCA L. STEINMAN, MNES
17 RURIKO TAKAMASHI, MNES
18 TETSUYA TERAMAE, MHI
19 EDWARD D. THROM, ISL
20 DOUG WOOD, MHI
21 DON WOODLAN, Luminant Power
22 TSUYOSHI YOSHIDA, MHI
23
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1 P R O C E E D I N G S

2 8:35 a.m.

3 CHAIR STETKAR: Okay. The meeting will
4 come to order. This is the second day of the
5 subcommittee meeting for MHI Chapter 15 U.S. APWR.
6 For the record, members in attendance are Sanjoy
7 Banerjee, Joy Rempe, Charlie Brown, and Bill Shack.
8 And the Designated Federal Official is Girija
9 Shukla.

10 As I mentioned yesterday, what we'll do
11 this morning is, we're in open session right now,
12 but I'm going to close this session, unless, MHI, do
13 you have any responses to questions to put on the
14 open record? What I was planning to do was to close
15 the session for the proprietary information and then
16 reopen it at the end so we can discuss upcoming
17 meeting issues and things like that. If you do have
18 anything, answers to questions from yesterday that
19 are part of the public record, we can do that in the
20 open session. Otherwise, we'll discuss them in the
21 closed session.

22 Staff have anything to say? Nothing?
23 Okay. With that, I would like to close the meeting
24 so we can discuss, continue the discussion of the

1 proprietary information. I'll ask the staff and MHI
2 to confirm that everybody in here is appropriate for
3 the information. And with that, we will close the
4 record.
5 (Whereupon, the foregoing matter went
6 off the record at 8:37 a.m. and went back on
7 the record at 11:43 a.m.)
8 CHAIR STETKAR: We're now in open
9 session, and, as we normally do, I'd like to go
10 around the table now that we're kind of at the end
11 of our day and a half session here to see if any of
12 the members have any additional comments or
13 questions, and I'll start with you, Joy.
14 MEMBER REMPE: Oh, I guess I'd like to
15 reiterate something we brought up yesterday that,
16 again, there's a lot of outstanding issues. And as
17 you are well aware, the fact that, in addition to
18 the outstanding issues with the steam generators,
19 thermal conductivity degradation, the accumulator
20 model, and some of those things will impact the
21 thermohydraulic analyses, and so we'll have to see
22 it repeated. In addition, there were issues brought
23 up by Dr. Banerjee, who's not here today or at this
24 point in the closed session, and so I would like to

1 see those issues addressed and staff comparisons of
2 some of the parameters besides peak cladding
3 temperature to see how well they've been addressed.
4 So are you going to have a meeting later to re-look
5 at the thermohydraulic issues?
6 CHAIR STETKAR: It's clear that we're
7 not done with this topic. How we bring it to
8 closure is something we're going to have to discuss,
9 you know, offline. We can't decide that today.
10 Sanjoy had a couple of ideas, and I'm hoping he'll
11 walk back into the room here in the next five
12 minutes or so. But we need to get to closure on
13 some of these, at least to a better understanding
14 of, as Sanjoy put it, the physics. We don't need
15 precision in the numbers as long as we understand
16 what's actually going on in the models.
17 MEMBER REMPE: It's reasonable and
18 predicted consistently.
19 CHAIR STETKAR: Yes, that's right.
20 Charlie?
21 MEMBER BROWN: Only to reiterate my past
22 stuff from the reactor safety part of it, not this
23 particular part --
24 CHAIR STETKAR: Sure.

1 MEMBER BROWN: -- that's dealt with here
2 relative to what they have to do, but we've still
3 got its basis and how the narrowing down of what's
4 automatic and what's manual and the basis for that
5 and all that kind of stuff. So that deals with the
6 coping --
7 CHAIR STETKAR: We'll beat that to death
8 in Chapter 7, I'm sure.
9 MEMBER BROWN: Yes.
10 CHAIR STETKAR: Dr. Shack?
11 MEMBER SHACK: Well, my biggest
12 confusion relates to this break flow modeling. I'd
13 like to know when it's conservative enough, and I
14 didn't see any criteria for determining that and it
15 sort of left me, you know, it sort of looked like I
16 could get a lot of answers in various ways. That
17 just sort of needs to be resolved.
18 CHAIR STETKAR: Again, I look at that --
19 yes, and I agree. I look at that as -- I don't
20 like the word conservative. Understand where we are
21 on the spectrum of reality would be a decent goal.
22 MEMBER SHACK: What constitutes an
23 acceptable break flow model --
24 CHAIR STETKAR: Yes.

1 MEMBER SHACK: -- and why.

2 CHAIR STETKAR: And why. Dr. Banerjee
3 is not here. He's obviously had a lot of questions
4 and comments, so I think they'll stand by
5 themselves. And I would just echo from what you've
6 heard here. I think that we've learned a lot in the
7 last day and a half in terms of, especially for the
8 LOCA responses and how they're modeled. We haven't
9 talked too much about the transients. I feel
10 reasonably comfortable about the transients. We had
11 a few questions yesterday but not an awful lot.

12 I think that we will need to have
13 additional discussions about, in particular, the
14 LOCA modeling so that we do understand a little bit
15 better the physics, how your models are behaving,
16 what particular assumptions are most sensitive in
17 your modeling of the performance, whether or not
18 your models can be driven, in a sense, to give very
19 different results so that we better understand kind
20 of the sensitivity of those models. And from the
21 staff's perspective, if you can shed any light, you
22 know, on your confirmatory analyses so that we can
23 better understand where the differences are, if
24 there are differences, and the reasons for those

1 differences, I think that would also help to kind of
2 give us confidence about the overall conclusion.
3 And with that, I would like to thank you
4 very, very much. I think we've had a very, very
5 good discussion. I always appreciate these meetings
6 because you do come well prepared and you have a lot
7 of back-up material, and I really appreciate all of
8 the hours that were probably put in last night
9 looking at information to answer a few of our
10 questions. So thank you very much for the
11 presentations.
12 Now, with that, and I have not
13 forgotten, Ryan, I think we need to discuss offline
14 among MHI, the staff, and our staff how we come to
15 closure on the particular issues of the Chapter 15
16 analyses. And I don't think it's worth trying to
17 make any suppositions about how we're going to do
18 that right now. We do have another two-day meeting
19 scheduled in the middle of October. I think it's
20 the 18th and 19th; don't hold me to the dates. In
21 particular, to look at the non-LOCA methodology
22 topical report and the advanced accumulator CFD
23 modeling. That two-day meeting will give us another
24 opportunity to at least revisit some of these

1 issues. We can't afford more than two days,
2 unfortunately, in our schedule, but we may want to
3 think about how best to use that meeting time. And
4 if there's anything else that we can do, perhaps, in
5 the interim, we'll talk about that more offline.
6 Now, we do, I believe, right at the
7 moment have a full committee presentation. I think
8 it's for October's schedule. Is that right, Ryan?
9 You tend to know these things better than I do. And
10 we wanted to discuss a little bit about what the
11 topics would be for that full committee meeting, so
12 that everyone has a chance to prepare.
13 And as you're aware, what we typically
14 do as a full committee is write an interim letter
15 just to inform the staff and the Commission of where
16 we are in our review of the entire issue, both the
17 design certification and the COLA process and, if
18 necessary during those interim letters, to identify
19 any particular issues that we feel merit special
20 attention so that they can be addressed before we
21 have the final review.
22 And since our last interim letter, which
23 was issued in September, a year ago, September 2011,
24 we've had subcommittee meetings that have covered

1 design certification Chapter 9 on auxiliary systems.
2 We've had this meeting now today, and I'll talk
3 about that in a second. And for, in particular, the
4 COL process, we've now completed Chapters 5, 8, 10,
5 11, 12, and, again, this meeting on Chapter 15. We
6 have a meeting scheduled in September on Chapter 4
7 for both the DCD and the COL. That's nominally
8 before our October meeting. It's probably not a
9 good topic for the full committee to address, you
10 know, two weeks later, so I'm not going to include
11 that.
12 My suggestion is that, in the full
13 committee meeting, we cover primarily the COL. The
14 previous letter was entirely design certification.
15 What I'd like to do is cover kind of a status update
16 on the COL, fairly simple chapters but it's
17 important to let people know if we have any
18 concerns. So it would be 5, 8, 10, 11, and 12 on
19 the COL. And only because it's something that I
20 don't want to dangle, we'll include also Chapter 9
21 on the design certification. It's going to be a bit
22 of a split because we won't get to Chapter 9 on the
23 COL for some time yet, but it's a big chapter and
24 there's a lot of systems in there and it's

1 worthwhile. So if that seems reasonable to both the
2 staff --
3 MR. MONARQUE: Chairman, what about
4 Chapter 15?
5 CHAIR STETKAR: Yes. Come to the mike
6 because we're still on the record here.
7 MR. MONARQUE: Sorry, Chairman. What
8 about Chapter 15? Are you going to defer that for
9 another --
10 CHAIR STETKAR: I think, I think it's
11 prudent to defer it. I think that there are enough,
12 that the problem is that we'll have the next
13 subcommittee meeting where we have a chance to
14 address Chapter 15 type issues would be after that
15 full committee meeting. And my sense is that there
16 are still enough questions in the minds of the
17 subcommittee that it might not be fair to bring
18 Chapter 15 to the full committee in October.
19 If we had a time slot before October to
20 have another subcommittee meeting, I'd feel
21 differently. But I just, I don't think it would be
22 fair to either MHI or the staff to bring Chapter 15
23 to the full committee in October. I just don't. So
24 we'll have to push it -- we can always schedule

1 another interim letter closer to the end of the
2 year, December time frame for example. If we want
3 to pick up, if it's possible, Chapter 15 and Chapter
4 4 and issues that might come up on Chapter 7 in
5 November for example, we have that opportunity in
6 December for another interim letter. We don't
7 necessarily have to schedule them, you know, six
8 months to a year apart.
9 MR. MONARQUE: I think, wasn't the
10 general rule, I think, you wanted to include at
11 least six chapters?
12 CHAIR STETKAR: Well, the general rules,
13 I was trying to bundle chapters together. We don't
14 want to issue a letter, you know, with only two or
15 three chapters, or if there was some compelling
16 reason, if we identified some major concern, to have
17 a full committee discussion of that concern and
18 issue an interim letter, you know, kind of as a
19 focus topic. In my opinion, we haven't identified
20 anything like that yet to date, so that's why I've
21 just been kind of bundling things together in larger
22 groups just so we can manage our time and not bother
23 everybody with, you know, several letters.
24 So let's try to do that, if it sounds

1 okay to you and MHI, and we'll plan for that. And I
2 think probably, we can discuss it again offline,
3 probably tentatively plan for another interim
4 letter. The problem is we don't have a meeting in
5 January, a full committee meeting in January. So it
6 would either be December or February. We don't
7 normally have a -- thank you, Dr. Shack -- normally
8 a full committee meeting in January, but kind of
9 December or February to pick up probably Chapter 15
10 and most likely Chapter 4 and perhaps 7. Those are
11 three heavy-hitter chapters, and I know you don't
12 want to delay too long on getting feedback.
13 MR. MONARQUE: Do you want to talk
14 offline on how we're going to move forward with 15 -
15 -
16 CHAIR STETKAR: Yes, absolutely offline.
17 MR. MONARQUE: Okay.
18 CHAIR STETKAR: Yes, yes, yes. Any
19 other questions, MHI, staff, subcommittee members?
20 (No response.)
21 CHAIR STETKAR: And, again, I thank you
22 all, MHI and the staff. I think it's been a really
23 productive day and a half. I'm really sorry that we
24 had to cut it short from our originally-scheduled

1 two days, but we have four subcommittee meetings
2 going on today and the time pressures are just
3 horrendous. So thank you all again. And with 15
4 seconds to spare, we are adjourned.
5 (Whereupon, the foregoing matter was
6 concluded at 11:58 a.m.)
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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)

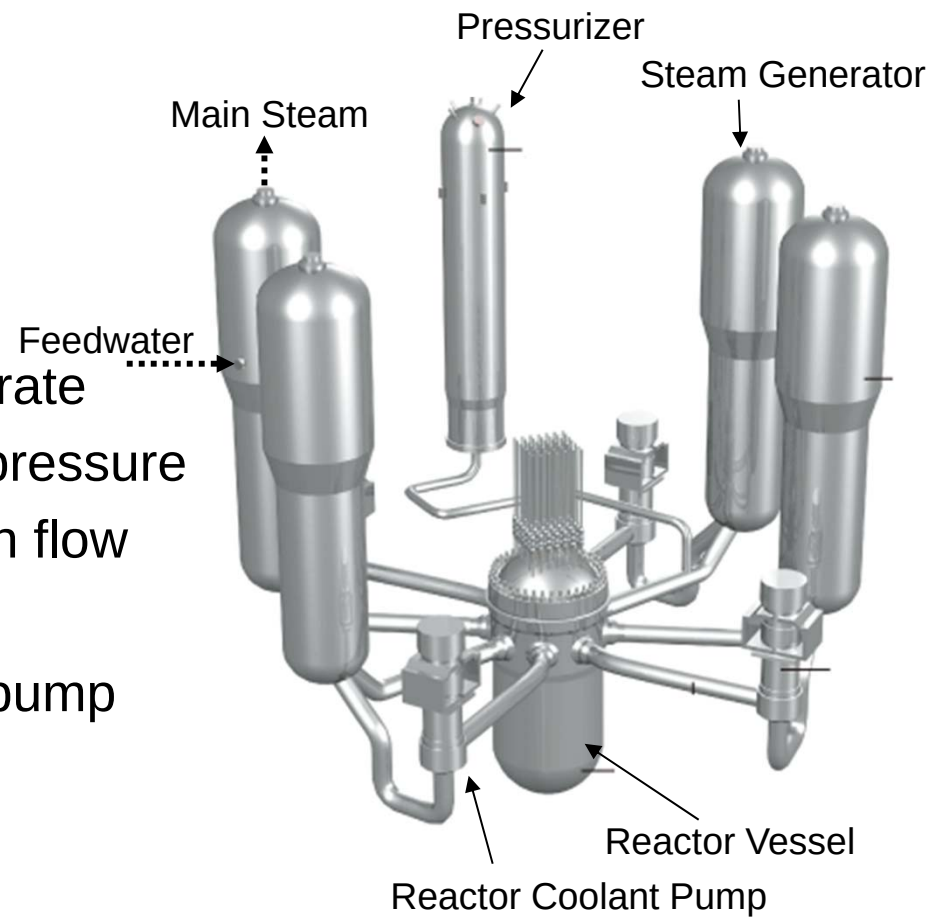
  Deputy Manager
  Safeguard System Engineering Section
  Reactor Safety Engineering Department
  Mitsubishi Heavy Industries, Ltd.

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

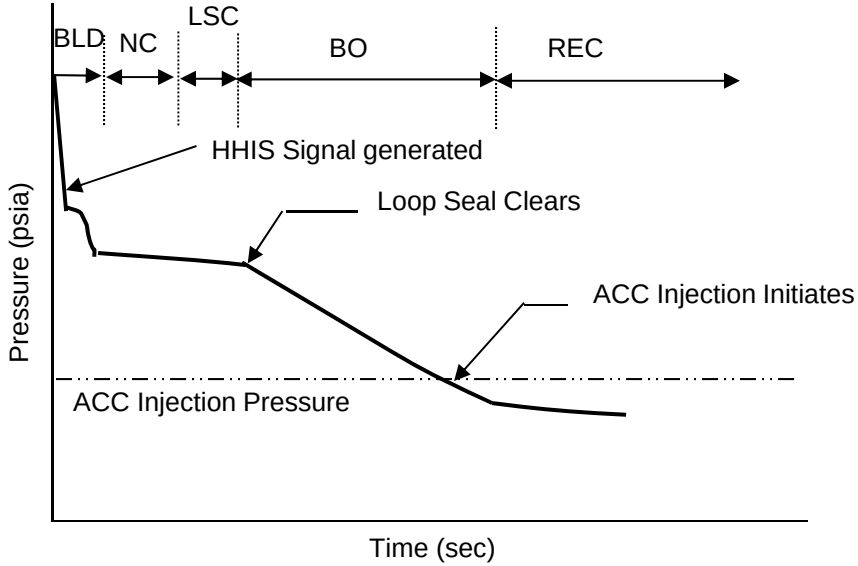
US-APWR Summary

- 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

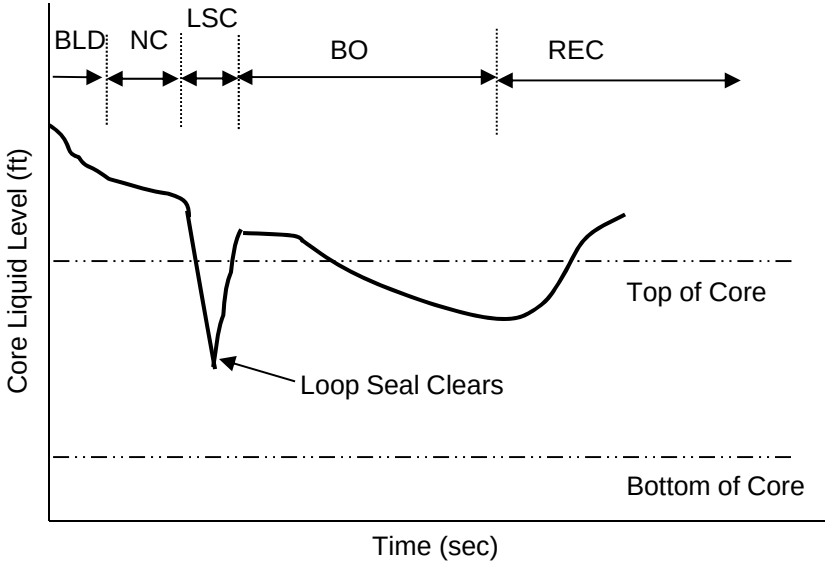


US-APWR SBLOCA

- 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

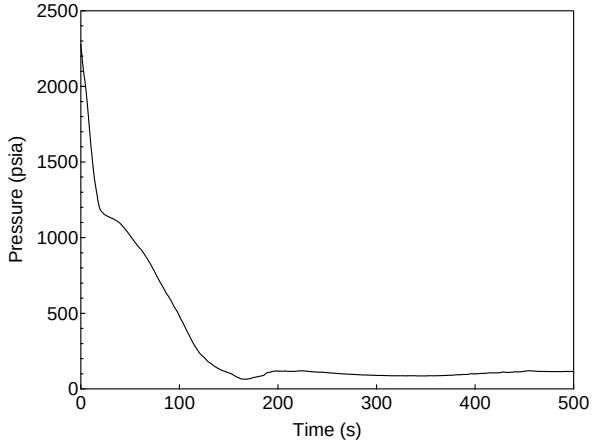
- Advanced Accumulator Model
  - ✓ Accumulator flow damper model based on ½-scale test data is newly implemented with the scaling bias determined from CFD calculations

US-APWR SBLOCA Summary

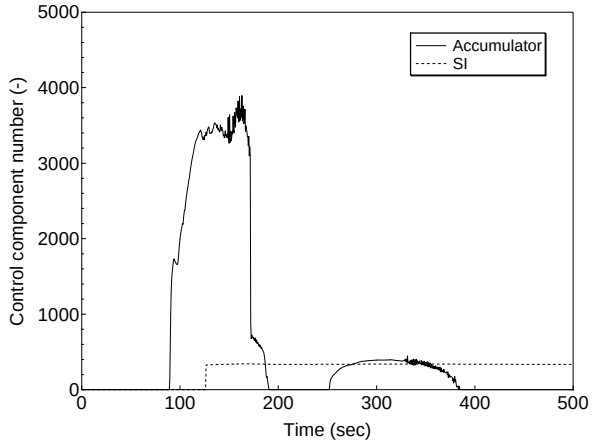
- Spectrum of Postulated Piping Breaks
  - ✓ Design-basis SBLOCAs with break sizes up to 1-ft^2 (cold leg)
  - ✓ MUAP-07025 "*Small Break LOCA Sensitivity Analyses for US-APWR*" provides the following sensitivity calculations
    - Break location
    - Break size
    - Break orientation
    - Plant noding
    - 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)

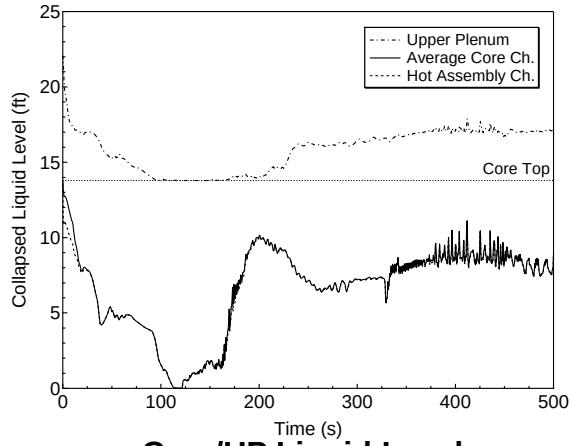
- 1-ft^2 Cold Leg Break
  - ✓ Rapid depressurization and PCT occurring during BO (PCT=1328°F)
  - ✓ US-APWR ECCS flow sufficient to suppress PCT



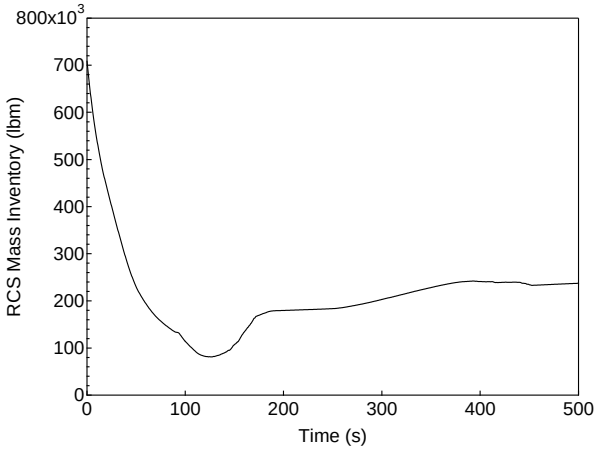
RCS Pressure



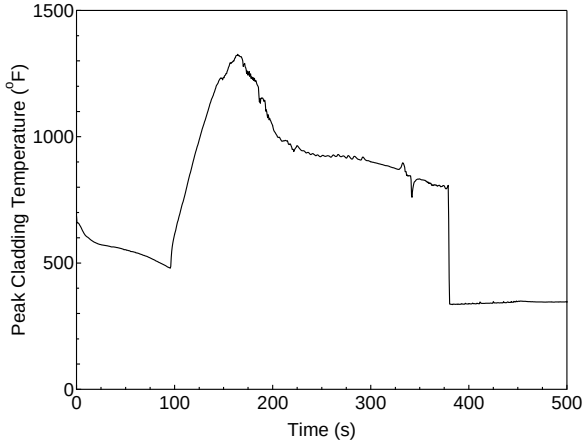
ECC Flow Rate



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

United States Nuclear Regulatory Commission
Protecting People and the Environment

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  - **Marcos Ortiz**
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- **Contractor Confirmatory Analysis Support**
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    Information System Laboratories
  - **Robert Beaton**
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  - **David Caraher**
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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

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- 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

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- 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

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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

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- 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)

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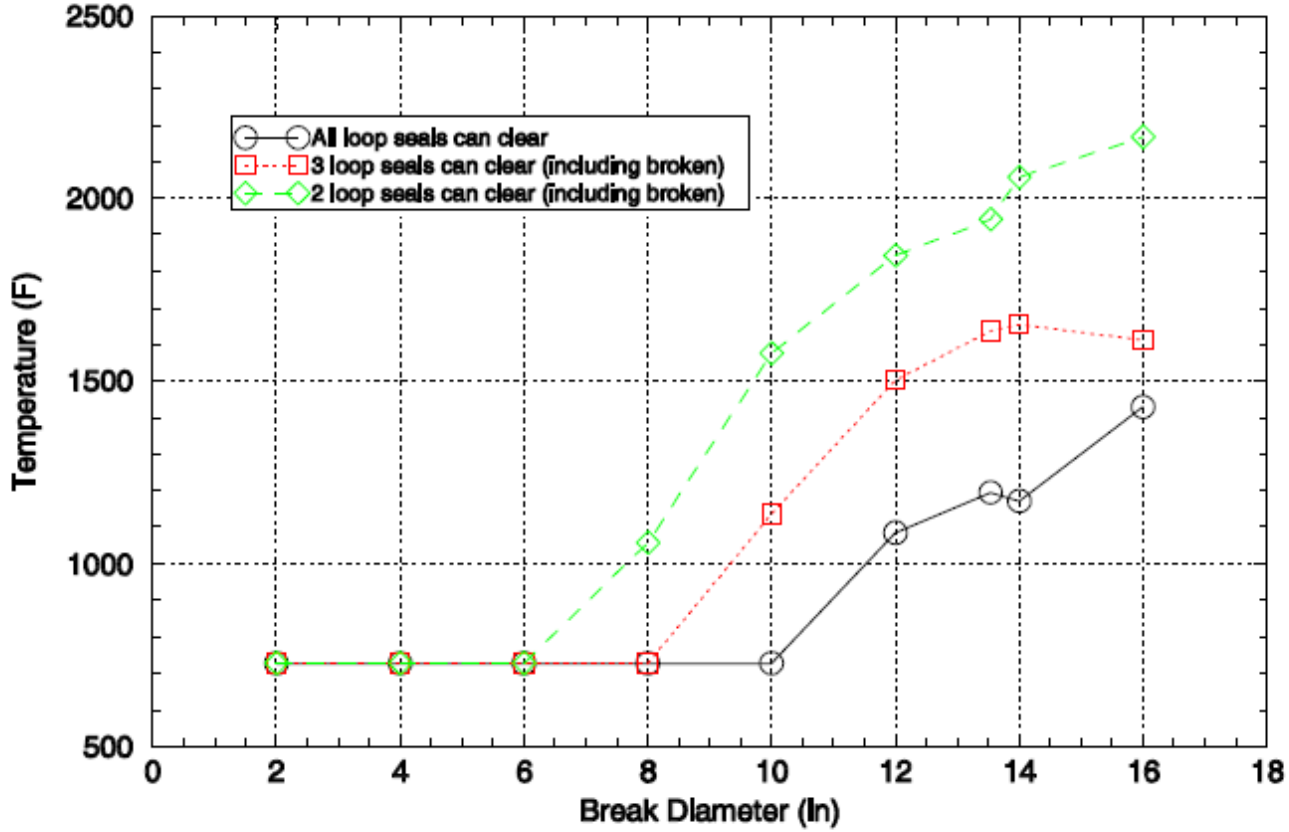
PCT as a Function of Break Diameter and Number of Loop Seals Permitted to Clear



SBLOCA RESPONSE
(Slide 2/3)

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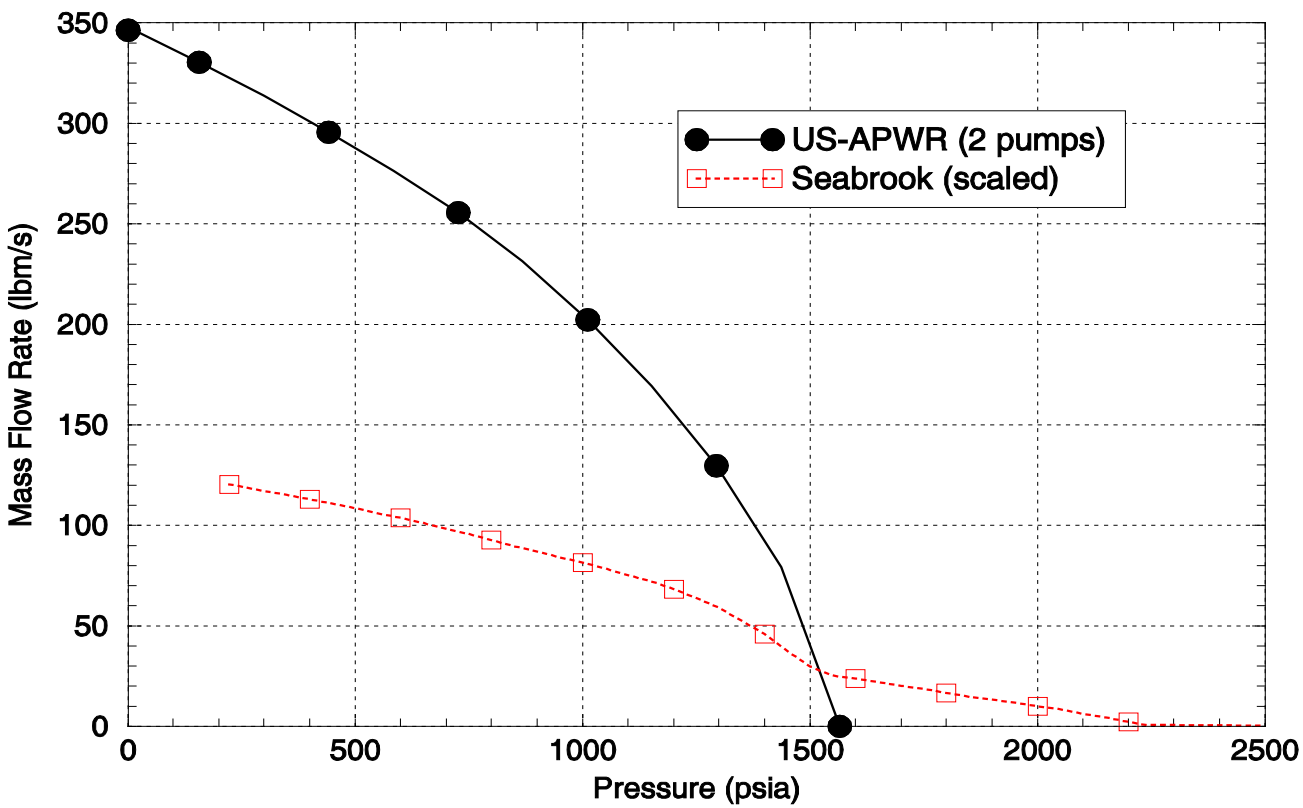
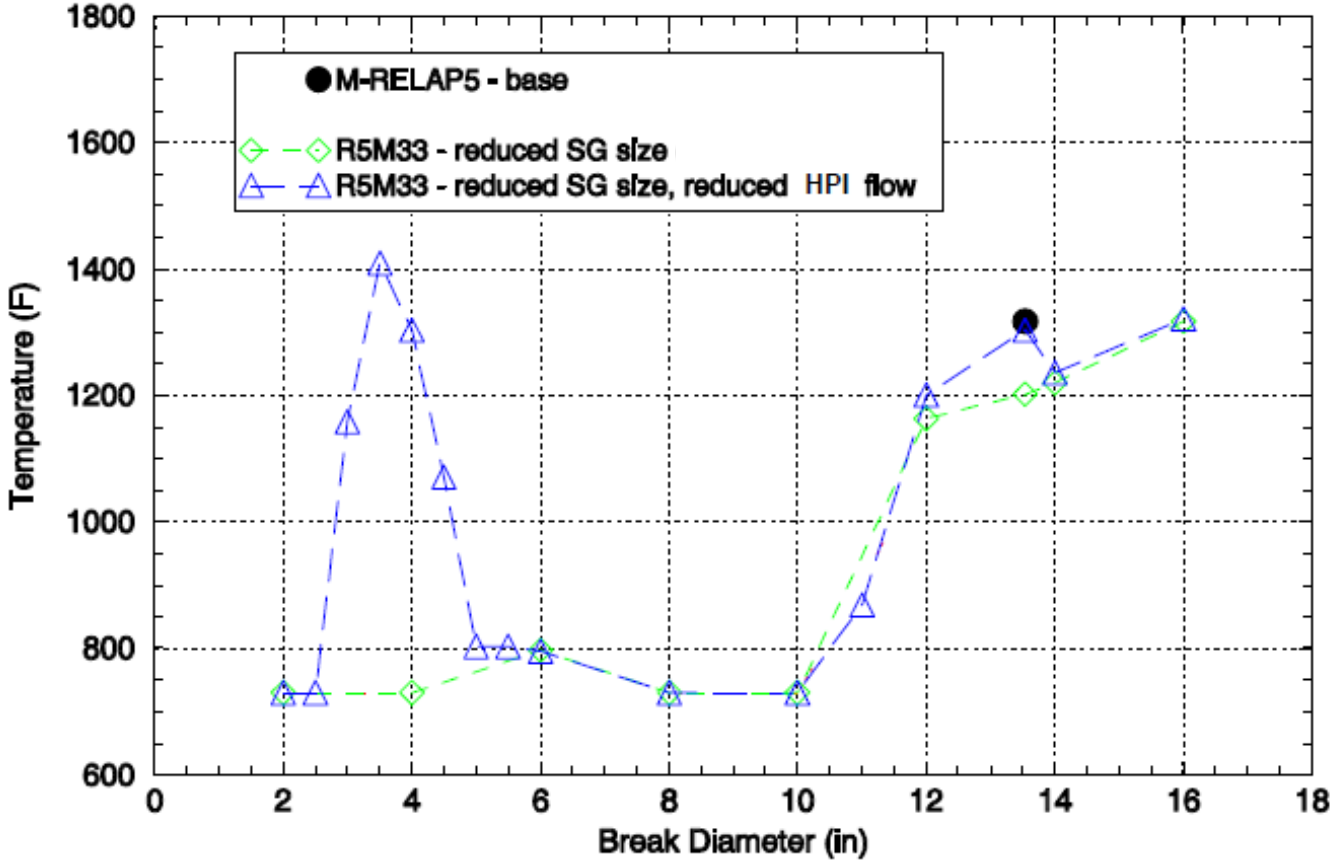
Total HPI Flow US-APWR and Core-Power Scaled Conventional PWR



SBLOCA RESPONSE
(Slide 3/3)



Critical Flow Model

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- 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)

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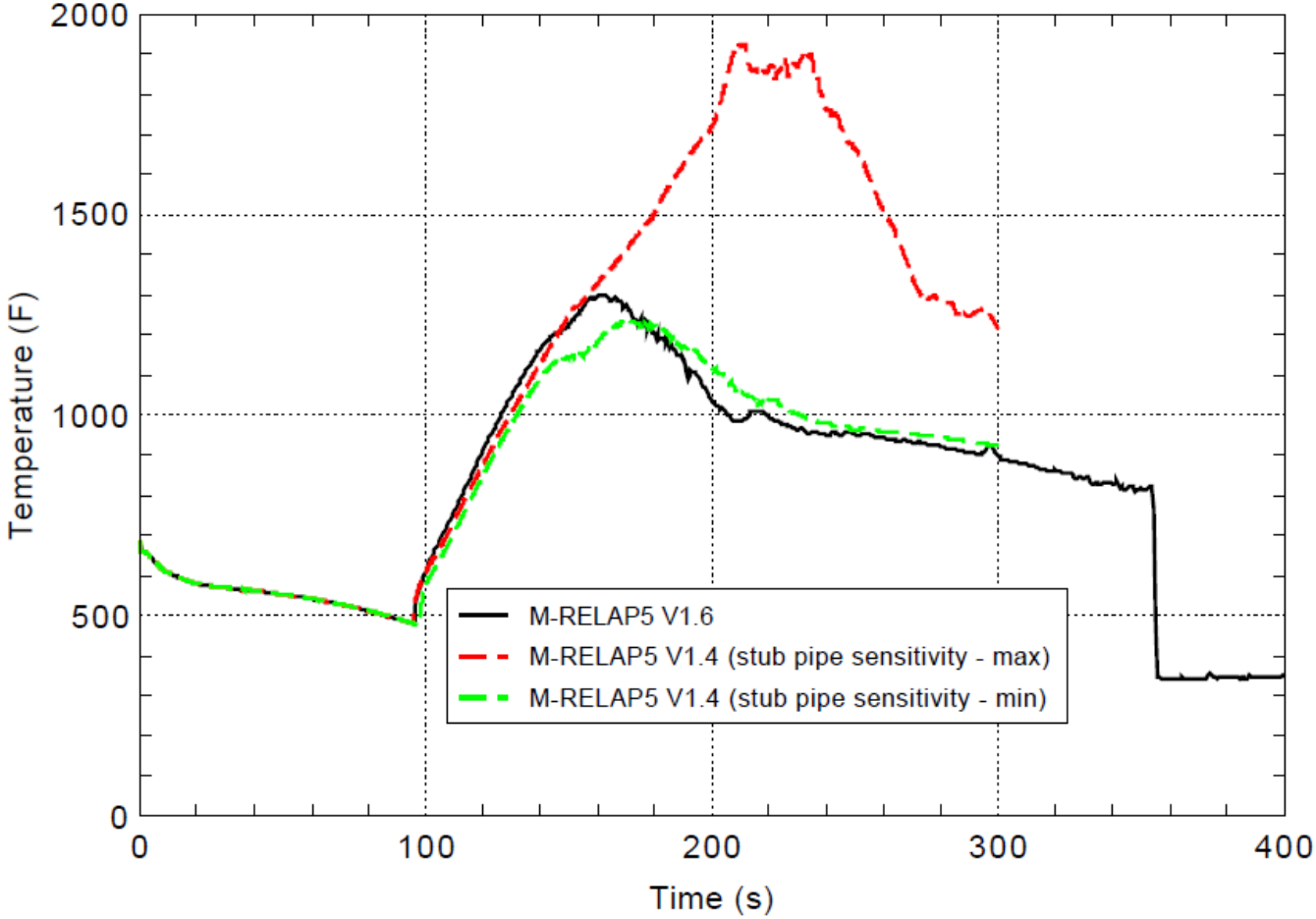
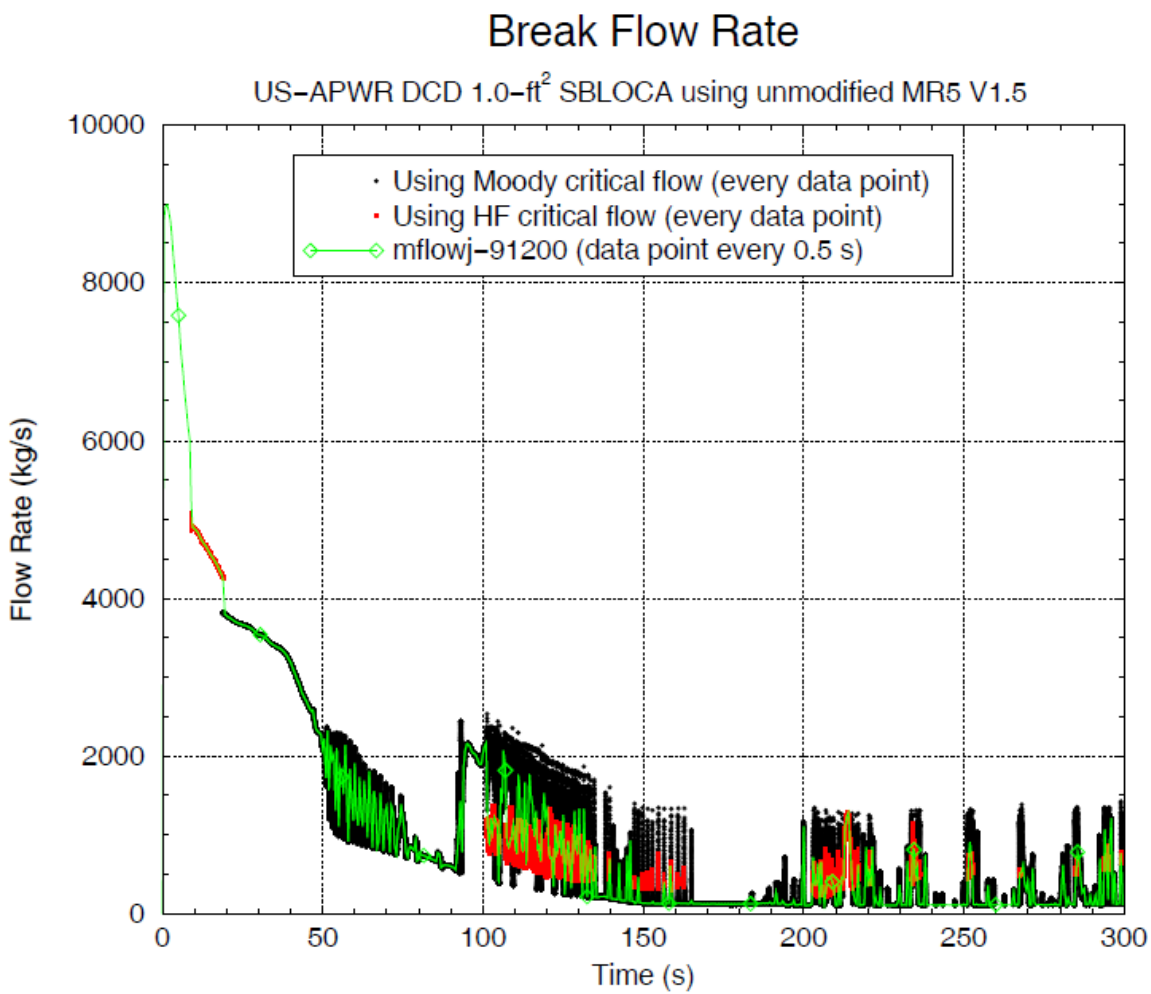
PCT Response with M-RELAP5 Versions 1.4 and 1.6



Critical Flow Model
(Slide 2 of 2)



Scaling

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- 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)

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- 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

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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

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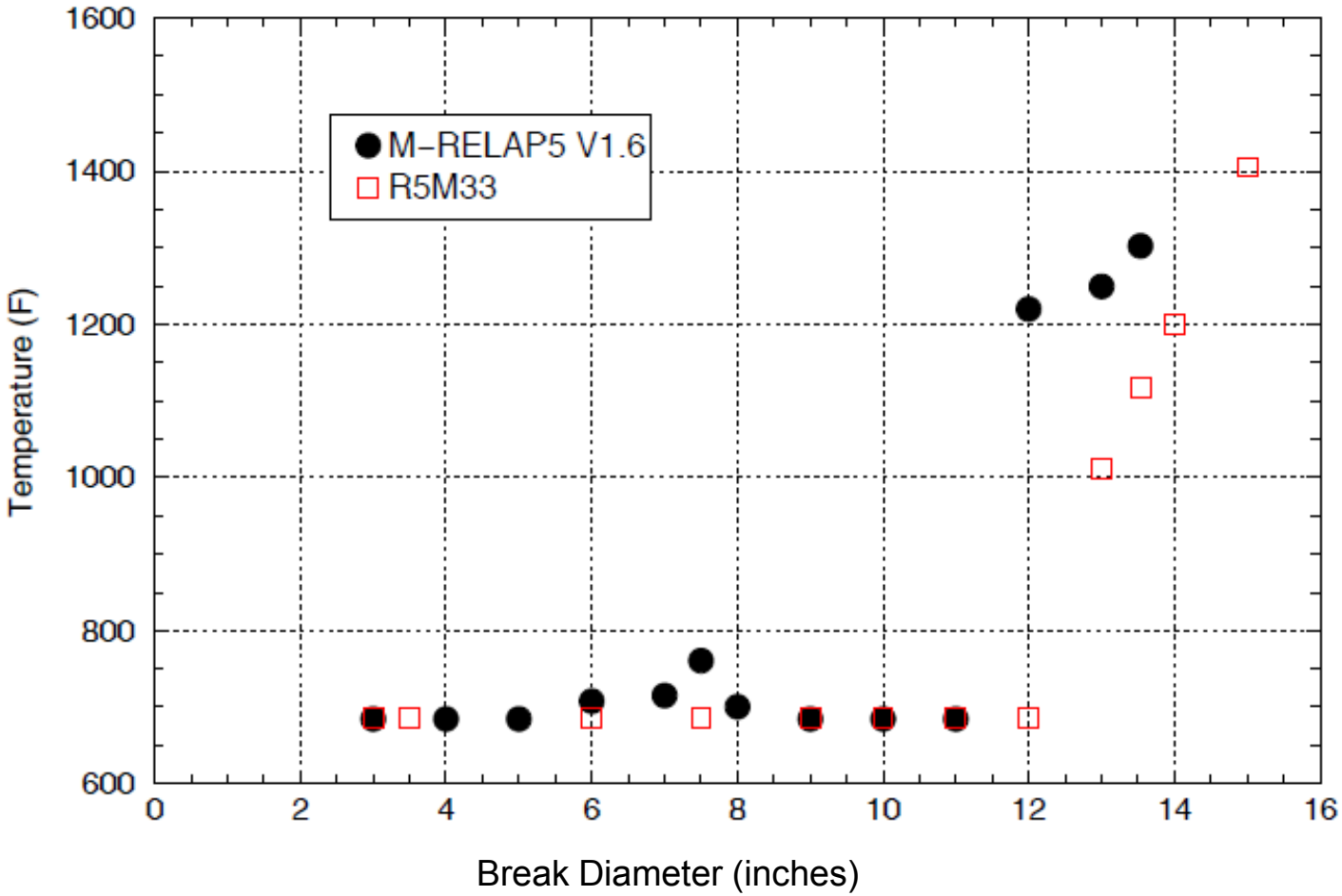
PCT Break Spectrum for M-RELAP5 and RELAP5/MOD 3.3



CCFL (Slide 2 of 3)

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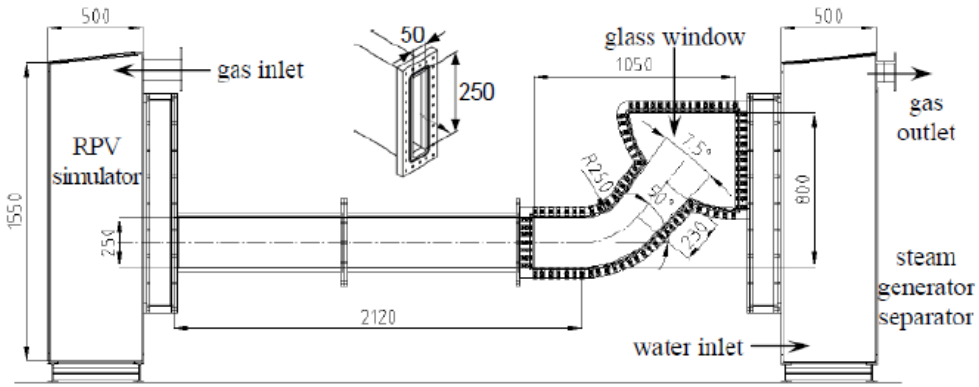
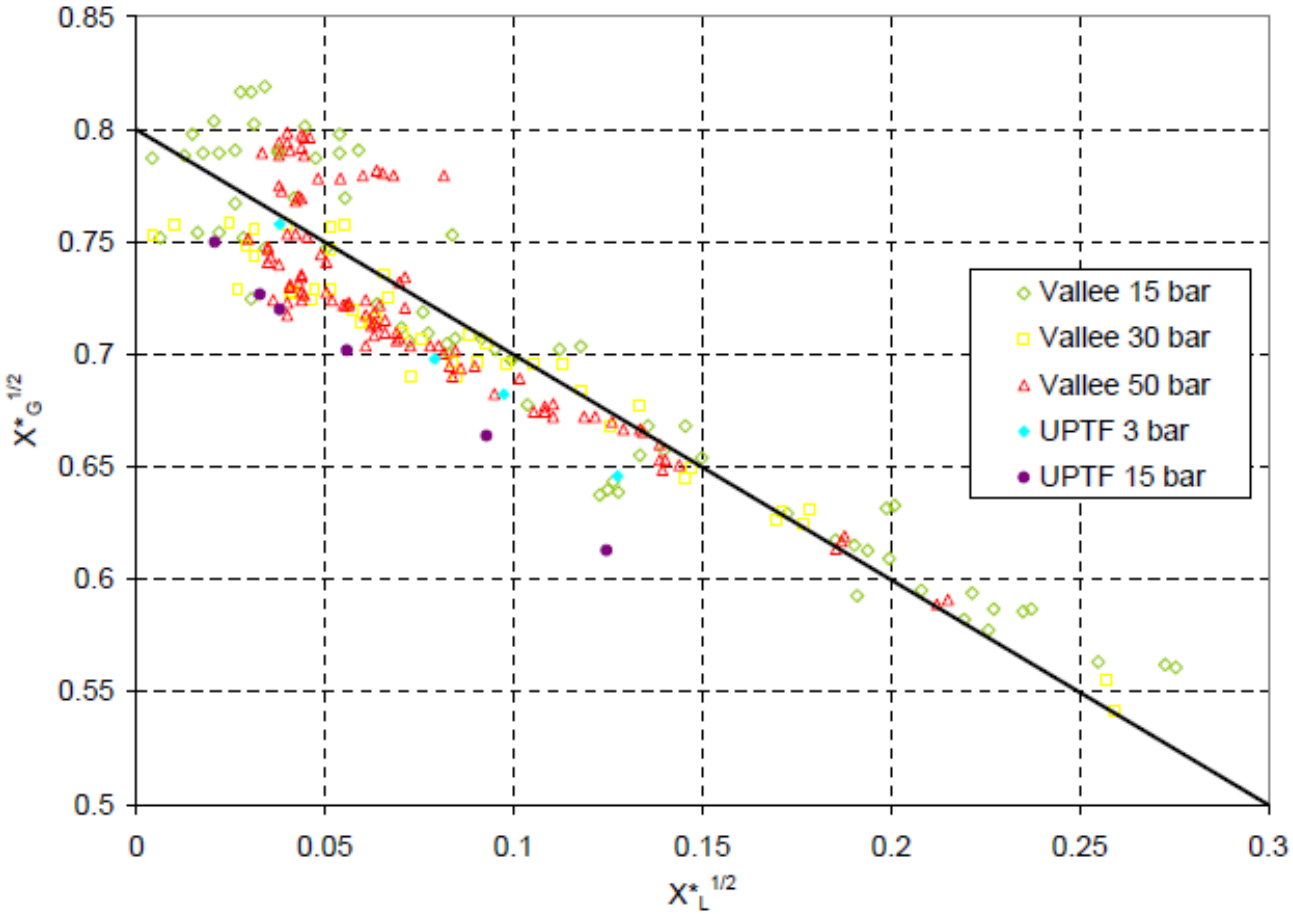
- Dresden (Vallee) conducted experiments to supply suitable data for CFD code validation.



- 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

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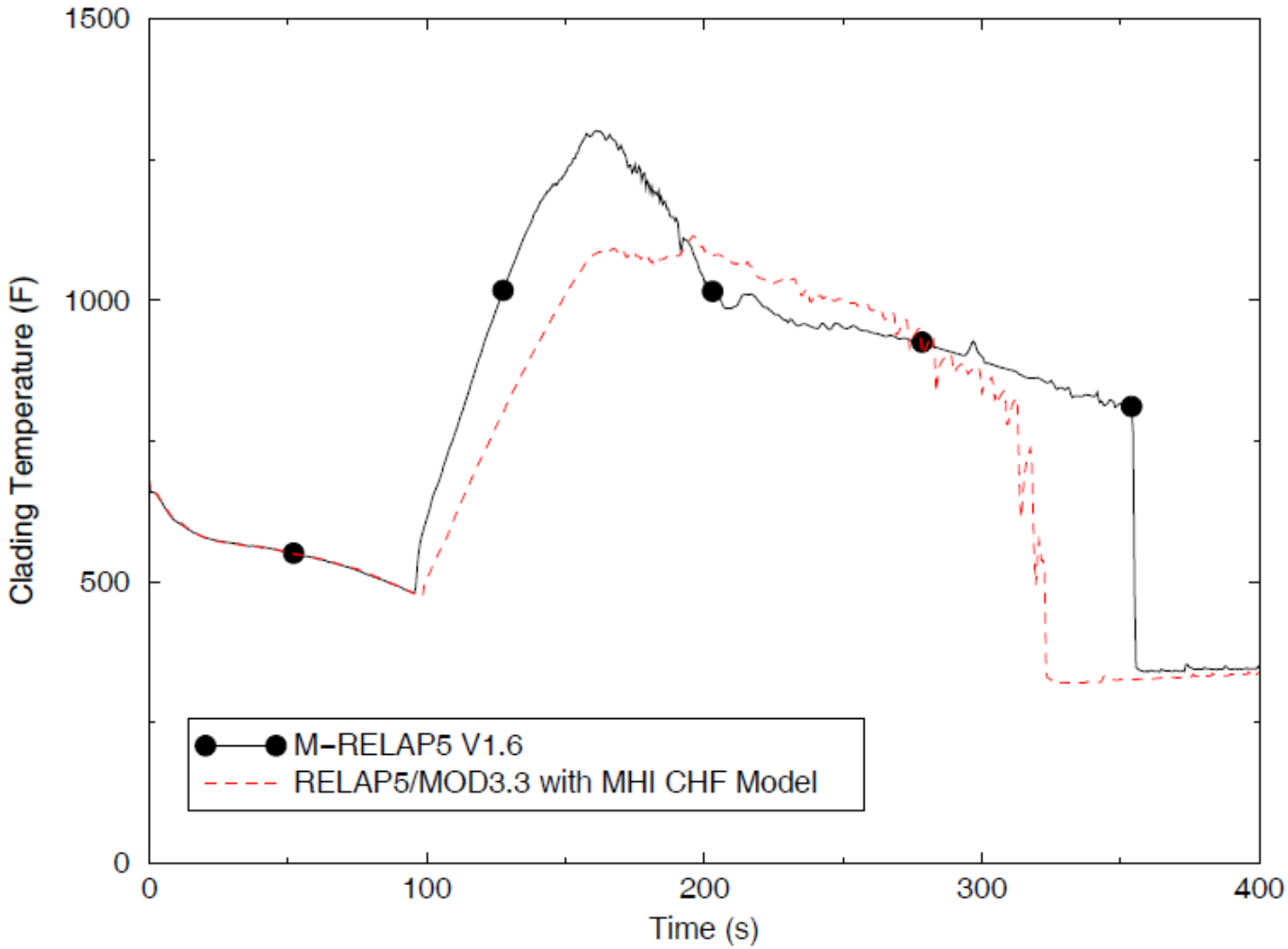
Comparison of M-RELAP5 and RELAP5/MOD 3.3 for Limiting Break



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.

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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)

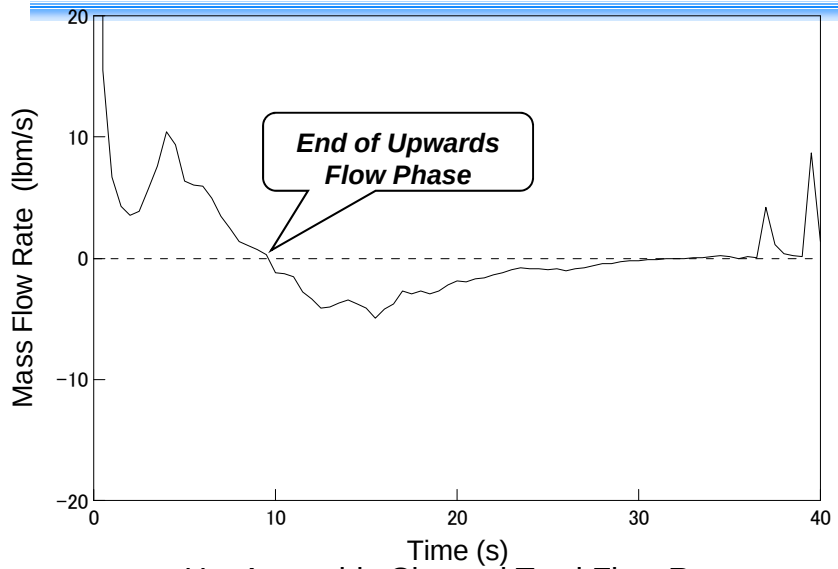
LBLOCA Code and Methodology (3/3)

- 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

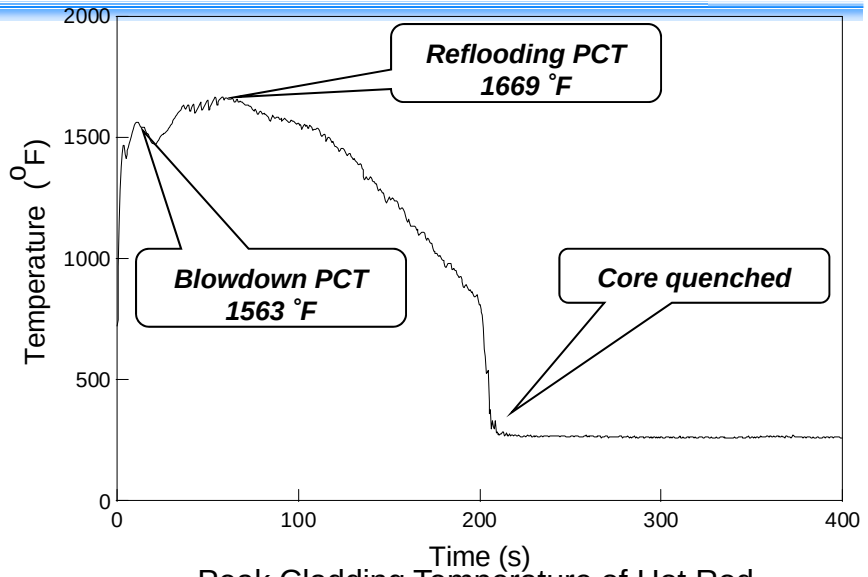
Sample Plant Analysis (1/2) –Reference Case-

- 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

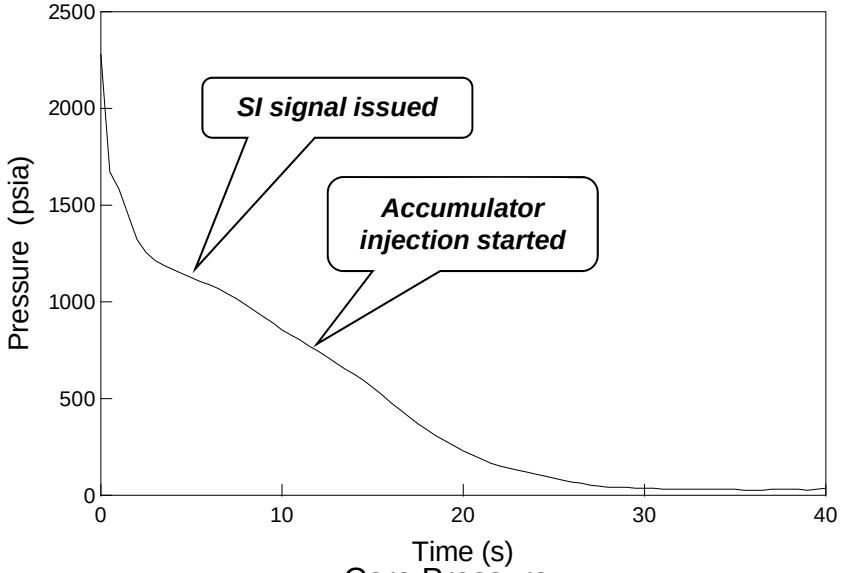
Sample Plant Analysis (2/2) –Reference Case-



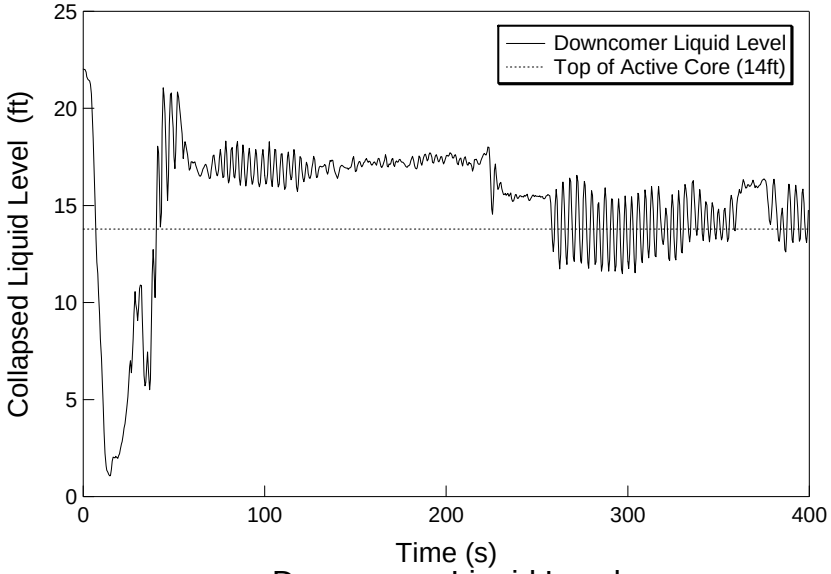
Hot Assembly Channel Total Flow Rate



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

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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

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Technical Staff

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Project Managers

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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

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- 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

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- 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

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- 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

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- 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

- 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

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- 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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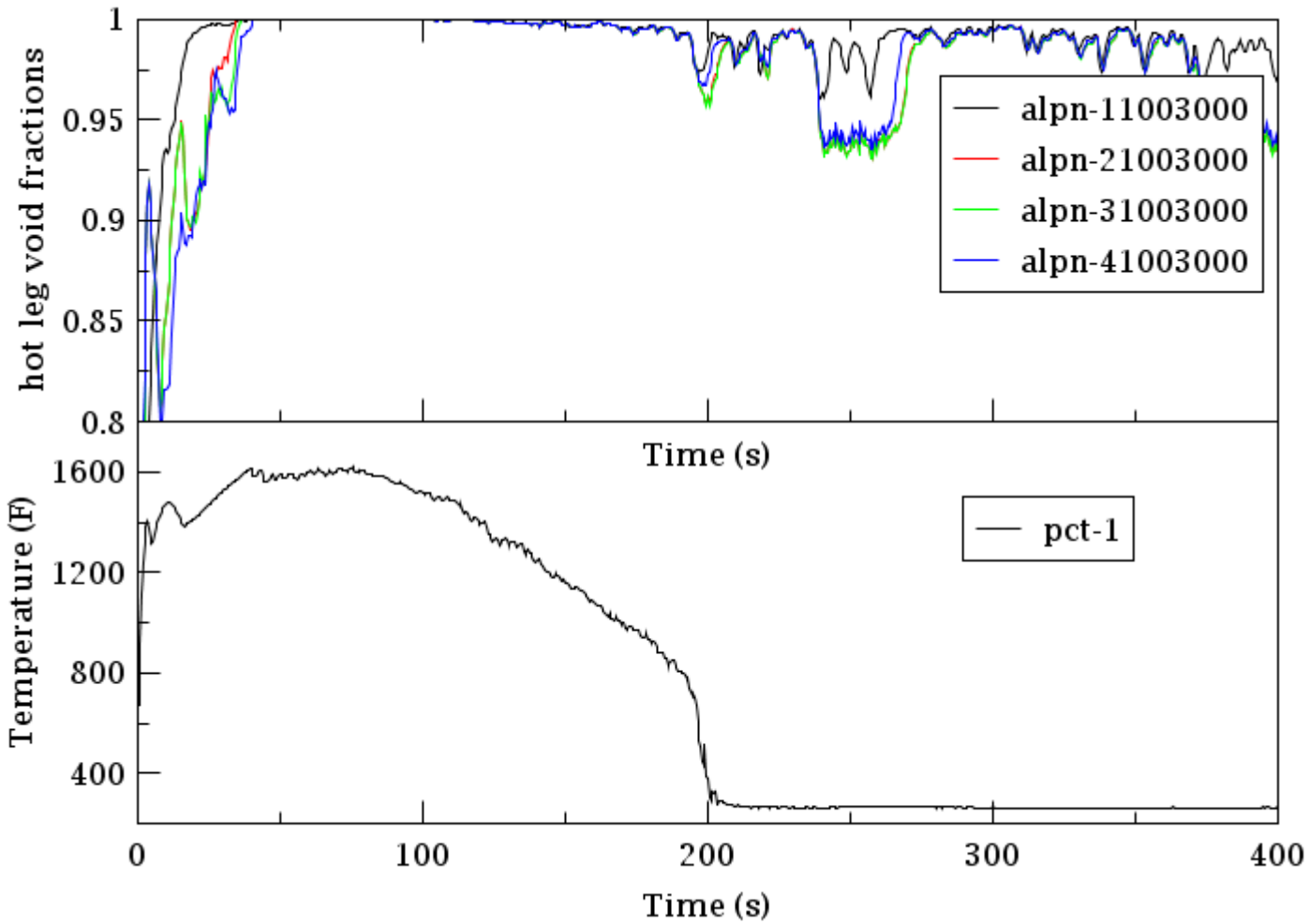
- There is no holdup of liquid in hot legs until after PCT



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

15.0.0 Transient and Accident Analyses (Introduction)

- **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**

➢ 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.

- **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

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.**

- **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)**

15.1 Increase in Heat Removal by the Secondary System

- **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

- **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	--

15.3 Decrease in Reactor Coolant System Flow Rate

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)

15.3 Decrease in Reactor Coolant System Flow Rate

- **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

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

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

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**

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

➢ 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

- **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)

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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

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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

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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.

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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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Presentation to the ACRS Subcommittee

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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