Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards
U.S. EPR Subcommittee Meeting

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Monday, February 7, 2011

Work Order No.: NRC-695 Pages 1-317

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

DISCLAIMER

UNITED STATES NUCLEAR REGULATORY COMMISSION'S
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The contents of this transcript of the proceeding of the United States Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards, as reported herein, is a record of the discussions recorded at the meeting.

This transcript has not been reviewed, corrected, and edited, and it may contain inaccuracies.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

+ + + + +

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

U.S. EPR SUBCOMMITTEE MEETING

OPEN SESSION

+ + + + +

MONDAY

FEBRUARY 7, 2011

+ + + + +

ROCKVILLE, MARYLAND

+ + + + +

The Advisory Committee met, at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., Sanjoy Banerjee, Chairman, presiding.

COMMITTEE MEMBERS:

SANJOY BANERJEE, Chairman

JOHN W. STETKAR, Member-at-Large

WILLIAM J. SHACK, Member

NRC STAFF PRESENT:

STEVE BAJOREK, NRO/RES

JASON CARNEAL, NRO

JOSEPH COLACCINO, NRO

RONALD HARRINGTON, NRO/RES

MICHELLE HART, NRO/DSER/RSAC

BILL KROTIUK, NRO/RES

SHANLAI LU, NRO

SHAWN O. MARSHALL, NRO/RES

GETACHEW TESFAYE, NRO

ANTHONY ULSES, NRR

DEREK WIDMAYER, Designated Federal Official

ALSO PRESENT:

PAUL BERGERON, AREVA NP

DOUGLAS BROWNSON, AREVA NP

KENNETH COFFEY, AREVA NP

JERRY HOLM, AREVA NP

KEITH MAUPIN, AREVA NP

MARTIN PARECE, AREVA NP

PEDRO PEREZ, AREVA NP

JACK ROSENTHAL, ISL

ROBERT SALM, AREVA NP

LILIANE SCHOR, AREVA NP

SANDRA SLOAN, AREVA NP

JONATHAN WITTER, AREVA NP

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

1 Adjournment 317

2

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

8:30 a.m.

CHAIRMAN BANERJEE: The meeting will now come to order.

This is a meeting of the Advisory Committee on Reactor Safeguards, U.S. EPR Subcommittee. I am Sanjoy Banerjee, replacing just for this meeting, for the Subcommittee, Dana Powers, who, of course, is irreplaceable -- immiscible and irreplaceable.

(Laughter.)

ACRS members in attendance are Bill Shack, John Stetkar. We are supposed to have Harold Ray and Joy Rempe. So, if they come, we will know.

Derek Widmayer of the ACRS staff is the Designated Federal Official for this meeting.

The purpose of the meeting is to continue our review of the SER with open items for the U.S. EPR Design Certification Document. We will hear presentations on and discuss the Chapter 15 transient and accident analyses, except for Section 15.6.5.

The Subcommittee will hear presentations by and hold discussions with representatives of AREVA NP and the NRC staff and other interested persons regarding these matters.

The Subcommittee will gather relevant information today and plans to take the results of this review, along with the remaining section of Chapter 15 of the U.S. EPR DCD SER with open items reviewed by the Subcommittee in a meeting scheduled in March 2011 to the full Committee at a future full Committee meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in The Federal Register. We have received no requests for members of the public to speak at today's meeting.

A transcript of the meeting is being kept and will be made available, as stated in The Federal Register notice. Therefore, we request that participants in this meeting use the microphones located throughout the meeting room when addressing the Subcommittee. The participants should first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

Copies of the meeting agenda and handouts are available in the back of the meeting room.

A telephone bridge line has been established with the meeting room today, and I understand we will have participants from AREVA on the

line at various times throughout the meeting.

We request that participants on the bridge line identify themselves when they speak and to keep your telephones on mute during the times when you are just listening.

Is the bridge line on, Derek? Do you know?

MR. WIDMAYER: Do you have anybody calling in?

MR. BROWNSON: I expect that there should be a couple of people calling in.

MR. WIDMAYER: I didn't hear anybody so far.

CHAIRMAN BANERJEE: Okay, but it is on.

So, we will now turn to Getachew Tesfaye, the NRO Project Manager, for the review of the U.S. EPR DCD, for your introductory remarks.

MR. TESFAYE: Good morning, Dr. Banerjee and everyone.

My name is Getachew Tesfaye. I am the NRC Project Manager for EPR design certification.

Today and tomorrow we will continue our Phase 3 ACRS presentations of the staff's safety evaluation of all those open items.

For the record, as I routinely do here, I

will go over and summarize the Phase 3 activities.

To date, we have completed the Phase 3 presentation of 11 chapters. We presented Chapter 8, Electric Power, and Chapter 2, Site Characteristics, on November 3rd, 2009, and Chapter 10, Steam Power Conversion System, and Chapter 12, Radiation Protection, on November 19, 2009.

On February 18 and 19, 2010, we presented Chapter 17, Quality Assurance, and portions of Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation.

On March 3rd, 2010, we presented Chapter 4, Reactor, and Chapter 5, Reactor Coolant System and Connected Systems.

On April 6th, 2010, we presented Chapter 11, Radioactive Risk Management, and Chapter 16, Technical Specifications.

On April 8th, we briefed the ACRS full Committee on the seven chapters that were completed through March 2010.

On April 21, we completed the Chapter 19 presentation, and on April 21, 2010, we received a letter from the ACRS full Committee Chairman on the seven chapters that were completed through March 2010. The letter indicated that ACRS has not identified any

issues that merit further discussion.

On May 27th, 2010, the staff submitted its reply to ACRS, and our last activity was last November, November 30th, where we presented Chapter 13, Conduct of Operation.

Today and tomorrow we will present Chapter 15, Transient and Accident Analysis. Please note that for Phase 2 and Phase 3 reviews, the Chapter 15 sections are broken up into two groups.

Today's and tomorrow's staff presentation will focus on Group 1 sections.

Our current schedule calls for completing the Phase 3 presentation of the remaining eight chapters by mid-October 2011.

Mr. Chairman, that completes my prepared statement.

CHAIRMAN BANERJEE: Okay. Shall we turn it over to AREVA then?

MR. TESFAYE: If there are no question, yes, please, go ahead.

CHAIRMAN BANERJEE: Okay. Go ahead.

MS. SLOAN: Thank you. Thank you. We are glad to be back before ACRS and continuing to make progress moving through the design certification review for EPR.

For today, as Getachew indicated, we are here to present Chapter 15, Transient and Accident Analyses, with the exception, as Mr. Tesfaye noted, of 15.6.5, which is the LOCA portion of Chapter 15.

You can see the agenda here. We will start with an overview of unique U.S. EPR design features.

Some of the members will recall that about a year and a half ago we gave an overview of these features. We thought it would be worth going back and maybe reviewing some of the key ones that are of most relevance to the analysis of non-LOCA events for U.S. EPR. And Doug Brownson will present that.

From there, Dr. Witter will talk about the trip functions for U.S. EPR and, then, move into a discussion of the in-core transient methodology. Following that, Mr. Brownson will again pick up and talk about the Chapter 15 evaluation of the non-LOCA and transient analyses. Pedro Perez will then join us to talk about the evaluation of radiological consequences of events.

And finally, at the end of the day, we understand that there may be one or more members of the Subcommittee who had a particular interest in understanding more about the steam generator tube

rupture mitigation strategy for the U.S. EPR. So, we have some time devoted at the end to that discussion, and Mr. Coffey will support that.

With that, I will turn it over to Doug Brownson.

CHAIRMAN BANERJEE: Sandra, can you just give us a little overview of also what is going to happen tomorrow? Or you're not prepared to do that right now?

MS. SLOAN: No, no, no, I'm fine with that.

Tomorrow we will talk about our control rod ejection methodology. Again, I think it was about a year and a half ago we came and gave an overview of that methodology. And so, we are back to review some of that and maybe answer some additional questions on that methodology.

And, then, after that, we will talk about our small break LOCA methodology. We understand there have been some specific questions posed to AREVA regarding potentially some unique phenomena for EPR small break LOCA response. So, we would like an opportunity to address that. So, we will be prepared to do that tomorrow.

CHAIRMAN BANERJEE: But that will mainly

be methodology, not results?

MS. SLOAN: Correct. We are not here to present the small break LOCA results tomorrow. That is captured in 15.6.5.

CHAIRMAN BANERJEE: Which will be later?

MS. SLOAN: Which will be later. But we did understand that the Subcommittee had some particular questions on phenomenology, reflux condensation --

CHAIRMAN BANERJEE: Correct.

MS. SLOAN: -- and potentially countercurrent flow limitations. We would like to talk about those at this opportunity.

CHAIRMAN BANERJEE: Okay. Great.

Go ahead.

MR. BROWNSON: Thank you.

My name is Douglas Brownson. I will be the first speaker here.

Some background on myself, before we get started, I have 27 years' experience in the nuclear industry. I have an undergraduate and master's degree from the Pennsylvania State University.

The first job was working for Commonwealth Edison, where, amongst other things, performed PBR core reloads, followed by 10 years at the Idaho

National Engineering Laboratory performing severe accident research and PRA Level 2 analyses for the U.S. NRC.

I have also worked at Yucca Mountain project. And most recently, I have spent the past five years involved in the performance of Chapter 15, Non-LOCA Analysis, for the U.S. EPR.

I would like to begin today with some background information, as Sandra pointed out. The U.S. EPR is a conventional four-loop PWR proven by decades of design licensing and operating experience. The NSSS volumes of the U.S. EPR are larger compared to existing PWRs. This increases the operator grace period for many of the transients.

Here's just a quick comparison of some of the key parameters.

CHAIRMAN BANERJEE: You mean larger in volume per units of power or?

MR. BROWNSON: No, larger in the RCS volume, pressurizer volume, those NSSS components.

CHAIRMAN BANERJEE: But when it comes to per unit of power, it is about the same or not?

MR. BROWNSON: Well, from the slide here, are you talking about it from a core average linear heat rate or -- well, here, down at the bottom that's

one of the parameters here I am going to go into. The linear power rate is actually a little bit less, but the overall core thermal output and electrical output is larger. We are operating at higher temperature ranges, leading to a higher plant efficiency.

The primary system pressure, operating pressure, is the same as other operating PWRs. We have a much larger reactor coolant flow per loop, and we have a 14-foot core versus a 12-foot core, which results in a lower core average linear heat rate and peak linear heat rate.

MS. SLOAN: I think Marty would like to comment.

MR. BROWNSON: Yes, Marty.

MR. PARECE: Yes, this is Marty Parece.

Is this working?

I am Marty Parece. I am with AREVA. I am the Vice President of Products and Technology. So, I am responsible for the design authority for the unit.

The main points are that, from a power scaling, the primary volume is about the same, but that is about the only thing. The pressurizer volume is much greater on a per-megawatt basis and a per-volume basis, which obviously makes the pressure response slower.

And, then, the secondary inventory, well, it is not on this slide. The secondary inventory is significantly greater per megawatt than a standard four-loop unit. So, on a typical reactor trip with a loss of feedwater, there is approximately 30 minutes of decay heat removal in the secondary steam generators all by themselves.

So, the secondary inventory is significantly larger. The secondary volume is significantly larger.

CHAIRMAN BANERJEE: Per unit of power?

MR. PARECE: Per unit of power, yes.

CHAIRMAN BANERJEE: What about the flow area in the hot leg?

MR. PARECE: Well, the flow area in the hot leg is approximately the same. It is a little larger. It is approximately the same.

CHAIRMAN BANERJEE: Per unit of power.

MR. PARECE: I believe the ID is 33 inches. Per unit of power, it depends on the design, but the per unit of power, the area of the hot leg is a bit smaller.

CHAIRMAN BANERJEE: Okay. So, the hot leg is 33 inches, you said?

MR. PARECE: ID, I believe. And on this

unit, the cold leg is the same diameter as the hot leg. So, that is a difference from the existing four-loop units.

CHAIRMAN BANERJEE: And what is the sort of typically existing four-loop plant? It is 33 inches as well?

MR. PARECE: Approximately. And I think the cold legs are 29 inches typically.

CHAIRMAN BANERJEE: Okay. And your steam generator flow areas on the riser side?

MR. PARECE: I don't remember. I have to do some math in my head.

CHAIRMAN BANERJEE: But can we come back to that?

MR. PARECE: Yes, we'll come back. I will do the math. I will come back to you on it.

CHAIRMAN BANERJEE: Yes, I am very interested in understanding the hot leg flow area per unit of power. So, that is going to be a bit less than the existing PWRs and the flow area on the steam generator side?

MR. PARECE: Yes. If you scaled up a current plant, if you scaled up the hot leg area just by itself on power, you would have something closer to 40 inches, I believe.

CHAIRMAN BANERJEE: Right. Okay. Thanks.

MR. BROWNSON: Any more questions on this slide?

(No response.)

I believe in tomorrow's presentation --

CHAIRMAN BANERJEE: And the loop seals are conventional?

MR. BROWNSON: Yes.

CHAIRMAN BANERJEE: Is there anything special I should know about them?

MR. PARECE: There's nothing special about the loop seals except that we have minimized the size of them as much as practical, the depth of them. So, the depth of the loop seals is less. And the elevation of the core to the center line of the hot leg and cold legs has been lowered to improve the amount of inventory above the core.

CHAIRMAN BANERJEE: Okay. So, we will get into all of this when we talk about LOCA and small breaks and things, right?

MS. SLOAN: There will be a discussion during small break LOCA, guaranteed.

CHAIRMAN BANERJEE: Excruciating detail, I'm sure, yes.

MR. BROWNSON: And I believe during

tomorrow's presentation on SB LOCA methodology a similar slide is being presented with some slightly different information. I think some of the volumes and stuff are on that slide.

Some of the safety-related systems and features is that the U.S. EPR has four train front-line safety systems for the medium head safety injection, the low head safety injection, accumulators, and emergency feedwater. There is one train for each primary or secondary loop.

This redundancy provides additional reliability for the mitigation of events, and it allows for single failures and preventative maintenance of trains while still satisfying event mitigation criteria.

The automatic partial cooldown feature of the steam generators allows the secondary system to drive the primary system pressure down to the medium head safety injection shutoff head in a controlled and rapid manner while maintaining saturation margin. This feature is significant in the event of a loss of coolant accident or steam generator tube rupture.

CHAIRMAN BANERJEE: What's the maximum rate of coolant?

MR. BROWNSON: A hundred and eighty

degrees per hour.

CHAIRMAN BANERJEE: Celsius or?

MR. BROWNSON: Fahrenheit.

CHAIRMAN BANERJEE: Fahrenheit.

MR. SALM: A hundred degrees C.

CHAIRMAN BANERJEE: Right. Yes. Okay. Now that makes more sense.

(Laughter.)

All right. I could do that conversion.

All right. Go ahead.

MR. BROWNSON: The automatic trip of the reactor coolant pump --

CHAIRMAN BANERJEE: Oh, just a point. Are you going to talk a little bit about this cool-down procedure or has that already been done?

MR. BROWNSON: I believe the cool-down procedure is going to be discussed in the steam generator tube mitigation strategy.

CHAIRMAN BANERJEE: Okay.

MR. BROWNSON: The next slide after this slide --

CHAIRMAN BANERJEE: Sure. If you are going to do it later, that's fine.

MR. BROWNSON: Yes. And there will be a little bit of additional information on the next slide

on how we accomplish that.

The automatic trip of the reactor coolant pumps on coincident SIS signal and low delta pressure across the pumps is an important LOCA mitigation feature. This prevents excessive coolant loss through the break and prevention of pump cavitation. This feature also satisfies the post-TMI criteria.

As Dr. Witter will describe later in greater detail, the U.S. EPR has low DNBR and high linear core power density protection system functions. The uniqueness of these trip functions, however, is that they are based on real-time in-core measurements rather than inferring the protection function from ex-core measurements.

The in-containment refueling water storage --

CHAIRMAN BANERJEE: Now, for the control rod ejections, you have ex-core measurements, don't you?

MR. BROWNSON: Yes, we do --

CHAIRMAN BANERJEE: Yes.

MR. BROWNSON: For some of our protection system functions, there are ex-core.

MS. SLOAN: And Dr. Witter will touch in more detail about the trip functions.

CHAIRMAN BANERJEE: Okay.

MR. BROWNSON: Yes.

The in-containment refueling water storage tank provides the water source for medium head and low head emergency core cooling safety injection. This tank is sized such that there is no need to switch to external water sources during the mitigation of an event.

In the event of a large break LOCA, the U.S. EPR has a two-train emergency boration system to provide additional criticality shutdown margin. This is a safety grade system, but is manually actuated.

In order to help operators identify the occurrence of steam generator tube rupture, there is a safety grade alarm provided on high activity in steam lines.

Another unique steam generator tube rupture mitigation feature is the steam generator blowdown/transfer lines. These safety-grade lines allow for the transfer of inventory from the isolated, affected steam generator to a sister steam generator that is also isolated in order to reduce the pressure in the affected steam generator and allow the primary system pressure to decrease to RHR cutting conditions.

Again, that feature will be discussed a

little bit -- no, I guess we are not. So, sorry about that.

Main steam relief operation --

CHAIRMAN BANERJEE: Well, let's do understand the core has a larger diameter than the normal PWRs, the four-loop. Do you have any special devices to mix things coming in at the bottom to improve the mixing?

DR. WITTER: In the lower plenum regions, there is a flow device in the lower plenum regions that help with the flow distribution somewhat in there for some of the mixing. That is part of the lower internals.

CHAIRMAN BANERJEE: And you have tested this?

DR. WITTER: We will let Marty speak to that, but I believe it has been tested at the Juliette facility.

MR. PARECE: Yes, what we found was, when we removed all of the in-core instrumentation from the bottom, those tubes coming up through the bottom caused the flow to be well-defined. And when you remove them, the flow tends to move from quadrant to quadrant. So, we have a flow distributor device at an inlet to the core below the core support plate that

makes that flow stable and well-distributed. And that has been tested at a 1/6th facility called the Juliette facility in France.

MS. SLOAN: Is there a figure of that in the FSAR? Because, if there is, we will pull it and we will provide it to the Subcommittee.

MR. PARECE: Yes, I'm sure that you can see --

CHAIRMAN BANERJEE: I am not right on top of this because I haven't been --

DR. WITTER: And this is Jonathan Witter.

I believe it was mentioned as part of the Chapter 4 presentation material. I think there was an image of the flow distribution device in Chapter 4.

MS. SLOAN: We will find a picture of it, and we will provide that to the Subcommittee.

CHAIRMAN BANERJEE: Okay. And with regard to the borating system, I guess the mixing in the lower plenum is fairly well-documented and discussed somewhere, right? I mean how well things mix. There's no plumes coming in?

MR. PARECE: Well, I am not sure where it is discussed in the FSAR, but that system is a fairly low flowrate with a fairly high concentration. So, clearly, for most of the transients, when the reactor

coolant pumps are running, it comes in in the cold leg and mixes pretty well.

So, I don't know where that is discussed in the FSAR.

MS. SLOAN: Yes, I think we are going to take an action, Dr. Banerjee, to find. If it is in the FSAR, we will point to the section.

CHAIRMAN BANERJEE: Okay.

MS. SLOAN: If it is not in the FSAR, we will see what supporting documentation could be provided.

MR. PARECE: I think it is about, what, 7,000 to 7,700 ppm enriched B10. So, that is equivalent to about 12,000 ppm natural boron.

CHAIRMAN BANERJEE: Weren't you doing some work at Rossendorf on the mixing? I seem to remember seeing an experiment in Dresden.

MR. PARECE: I don't recall. I don't know.

CHAIRMAN BANERJEE: All right.

MS. SLOAN: Yes, it doesn't ring a bell for me.

CHAIRMAN BANERJEE: The Juliette is what you depend on?

MS. SLOAN: Yes. Yes.

CHAIRMAN BANERJEE: Okay.

MR. BROWNSON: Anything else?

(No response.)

Main steam relief train operation. The main steam relief train, there is one of these per steam line. Each MSRT consists of two valves in series, the main steam relief isolation valve that functions as a safety relief valve and the main steam relief valve that maintains the secondary pressure at a prescribed setpoint.

The MSRT provides two functions for the U.S. EPR, pressure relief during overpressure events and a controlled partial cooldown during LOCA and steam generator tube ruptures. The MSRT is sized equivalent to 50 percent of the full load steam capacity.

And in overpressure protection mode, the relief valve opens at a prescribed setpoint of 1385 psia, and the control valve maintains its initial position to allow full relief.

In the partial cooldown, the MSRT starts at the 1385 and regulates the pressure decrease at a secondary rate, as we discussed earlier, of 180 degrees F per hour.

The decrease in secondary pressure drives

primary pressure down to allow timely actuation of the medium head safety injection system, which, again, has a shutoff head of 1405 psi. And for steam generator tube rupture, the setpoint of the MSRT on the affected steam generator is reset high, up to the 1436 psia value, to preclude its actuation in the release of radionuclides.

MEMBER STETKAR: Doug, you mentioned that the steam relief capacity of the MSRT is 50 percent of rated steam flow. Is that the total capacity of all four trains or is it per steam generator?

MR. BROWNSON: Per loop.

MEMBER STETKAR: Per loop?

MR. BROWNSON: Fifty percent per loop.

MEMBER STETKAR: Okay. Thanks.

CHAIRMAN BANERJEE: And you don't take steam through the condensers or anything to depressurize?

MR. BROWNSON: There's a non-safety system that actuates before the MSRTs, the turbine bypass system, which will dump directly into the condensers. That has a total capacity of 50 percent of the full load, of the entire plant full load capacity.

CHAIRMAN BANERJEE: But that is a non-safety system?

MR. BROWNSON: That is a non-safety system, which is not credited for the Chapter 15 events.

CHAIRMAN BANERJEE: Right.

MR. BROWNSON: That finishes the unique features. Jonathan will now talk about the trip functions.

DR. WITTER: Good morning.

My name is Jonathan Witter. And I have spoken once or twice before the Committee, but I will give a little bit of background about myself again.

I have got my bachelor's and master's in nuclear engineering from Rensselaer Polytechnic Institute, 1982, 1983. Qualified EOOW in the Navy nuclear program under GE as the civilian and nuclear plant engineer for training.

Then, spent a stint at the James A. FitzPatrick nuclear power plant in Oswego, New York, as a station nuclear engineer for five years.

Then, went on to school to get my PhD at Massachusetts Institute of Technology, and I did a simulation of control for nuclear rocket control at the time. In 1993, it was going back to Mars, so it was the thing to do to get into space.

Then, after that, I went to the Knolls

Atomic Power Laboratory. I worked in the advanced concepts nuclear design for 13 years with the last two years serving for the Jupiter Icy Moons Orbiter mission, which didn't last very long.

And at that time, I got a little bit -- decided to come back into the commercial world and joined AREVA in 2006 for the U.S. EPR project, and have been working on, whether it has been the fuel performance, thermal hydraulic performance, and core-related design aspects for the U.S. EPR.

Today I will be talking a little bit about the trip functions for the U.S. EPR and, then, a little bit more about the in-core monitoring system and how we verify the change in methodology and the setpoints, to verify that we do not violate the specified allowable fuel design limits during the Chapter 15 events.

Okay. So, the first slide here shows a little depiction of some of the instrumentation that we make use of for reactor trips for the protection system, the typical instrumentation in the hot leg and cold leg for temperatures and pressures, reactor coolant pump speeds, delta pressure taps for flow measurements. We also measure the control drop positions for the control rods, as they provide an

input into some of the safety systems for the in-core trip systems.

And what I mean by the in-cores are, for the SPNDs, they are self-powered neutron detectors where we have an array of in-core detectors within the core region that, instead of just serving as a monitoring system, they are actually used as part of the protection system. So, they monitor and then will also perform a trip function.

And I will have a series of slides discussing how we make use of those.

CHAIRMAN BANERJEE: So, I guess you have discussed these delta P over the elbow taps at some other meetings?

DR. WITTER: I believe so in the Chapter 4 parts of it for the RCS description and the thermal hydraulics.

CHAIRMAN BANERJEE: And nobody asked you any questions about it?

DR. WITTER: Not that I recall.

MS. SLOAN: I don't recall any particular interest.

(Laughter.)

CHAIRMAN BANERJEE: How interesting.

(Laughter.)

What are you claiming for these delta P --

DR. WITTER: It is primarily a measure for the low-low flow reactor trips. Well, I will get into it a little bit with our high core power level monitoring. They don't feed into the core power calculation. They are just used there to essentially predict the flow rate and whether or not we have flow in a loop or low-low flow for the total system to cause a reactor trip on that.

So, there is no real flow measurement being used for an enthalpy-based calculation.

CHAIRMAN BANERJEE: Okay. So, you are using your ultrasonic detector? I noticed you get a .48 percent.

DR. WITTER: Yes, sir.

CHAIRMAN BANERJEE: For your LOCA initial conditions, I assume?

DR. WITTER: Yes. Yes, for the secondary --

CHAIRMAN BANERJEE: We would like to discuss that, of course, in depth for sure.

DR. WITTER: We can discuss that a little bit.

MS. SLOAN: Did you want to do that during the LOCA discussion, Dr. Banerjee?

CHAIRMAN BANERJEE: Oh, okay. We can do it then. That would be good.

MS. SLOAN: We have an open item related to this with the staff. So, we are not prepared necessarily today. I think Marty has a couple of comments, Marty Parece, that he can make, but that is an open item still under resolution with the staff right now.

Marty, did you want to say a couple of words at this point?

CHAIRMAN BANERJEE: You just referred to LOCA, but go ahead.

MR. PARECE: Well, for the heat balance uncertainty itself, we have a fairly rigorous statistical application for the EPR based on what we have done on several Appendix K uprates or MURs using the Caldon equipment.

And we think we know that the .48 percent is bounding, but it includes uncertainty. The biggest uncertainty is the feedwater measurement itself, which is around .3 percent or less. And, then, we have uncertainties in specifically the RTDs for getting temperatures to do enthalpies for the heat balance.

And this flow in these elbows is actually calibrated to the secondary heat balance and the core

power from the in-cores and the ex-cores and a primary heat balance. So, the flow in those elbows ends up getting correlated to the heat balance itself.

CHAIRMAN BANERJEE: Right.

MR. PARECE: So, we are fairly rigorous on that, and the Caldon equipment has been shown to be very accurate in their testing at Alden Labs.

This is the leading-edge-plus flow meter.

CHAIRMAN BANERJEE: I know the machine, yes.

MR. PARECE: Right.

CHAIRMAN BANERJEE: Of course, in the staff SER there is a requirement to do both for typical testing in the lab -- well, there is a parallel path. You can install it and you can test it, but, in general, it has to be tested in a prototypical situation --

MR. PARECE: Right.

CHAIRMAN BANERJEE: -- installed, and then there has to be in situ tests.

MR. PARECE: Correct.

CHAIRMAN BANERJEE: Then, the problem, of course, is the Reynolds number scaling on that because you cannot do the calibration tests of the Reynolds numbers in the plant and the viscosity conditions.

So, while it may have gone through some MURs, this is a different application. And every plant, it has to be tested for every plant, right?

MR. PARECE: I don't think we disagree.

CHAIRMAN BANERJEE: Yes. So, you have a program in place to do the testing and everything or are you going to leave this to the COLAs?

MR. PARECE: Well, this will be part of the startup procedures for the unit.

CHAIRMAN BANERJEE: Right.

MR. PARECE: And getting the first heat balance. So, that part of it will definitely be the plant owner's responsibilities.

CHAIRMAN BANERJEE: Sure. All this testing, calibration.

MR. PARECE: And we haven't written the equipment specifications yet for the flow meters. So, some of this can be or a lot of it, the testing will be flowed down to the manufacturer.

CHAIRMAN BANERJEE: Are you actually specifying how they meet this requirement or leaving it up to the COLA as to what instrument they use? I mean, is this your spec or you are actually specifying the instrument? I'm a little confused.

MR. PARECE: Well, at this point, for this

1 particular instrument with this quality of
2 measurement, there's only one manufacturer. So, I
3 would hate to say that I'm specifying a manufacturer
4 for the owners to purchase, but at this point there's
5 really only one.
6 MS. SLOAN: And, Dr. Banerjee, I would
7 suggest that we carry that -- this is an open item in
8 the LOCA section of the FSAR. We understand and are
9 following ACRS interest in power measurement
10 uncertainty. So, when we get to 15.6.5 for LOCA, we
11 understand we have to come prepared to discuss that
12 measurement.
13 CHAIRMAN BANERJEE: You have seen our
14 Vogtle letter?
15 MS. SLOAN: We have seen the letter to
16 Vogtle, yes, sir.
17 CHAIRMAN BANERJEE: Okay. And you saw the
18 requirement was somewhat different from what you are
19 asking?
20 MS. SLOAN: We understand there is
21 heightened interest.
22 CHAIRMAN BANERJEE: Yes. Okay.
23 DR. WITTER: Okay. I think that it is
24 within this slide.
25 MEMBER STETKAR: Jonathan?

DR. WITTER: Yes, sir?

MEMBER STETKAR: Before you leave it --

DR. WITTER: Sure.

MEMBER STETKAR: -- all of the instrumentation that you show on the slide with the exception of the steam generator pressure measurements, the sensors are located inside the containment, is that true?

DR. WITTER: The containment, that is correct.

MEMBER STETKAR: The steam generator pressure sensors are located outside containment, the safeguards building? Okay. How far from the containment penetration are they? You know, half a meter, 12 meters, 120 meters?

DR. WITTER: We have to take an action to that.

MS. SLOAN: Yes, we will take an action to follow up and find the distance.

MEMBER STETKAR: The reason I am interested, obviously, is when we get into steam line break, you very carefully craft the location of the steam line break based on where those sensors are located. So, I was curious where they are.

DR. WITTER: Yes, the protection system

functions are broken into kind of like three areas: core-related, the RCS-related, and, then, secondary related.

This slide provides an overview of the key reactor trips that are core-related. Where we have the first two bullets, the high core power level and low hot leg saturation margin trips are enthalpy-based calculations based on the conditions of the cold leg and hot leg of each of the RCS loops. And in this case, the high core power level trip is 105 percent with a 10 percent uncertainty associated with that. The low hot leg saturation margin -- we will talk a little bit more about these -- is basically there to protect the calculation of the enthalpy to ensure that we are not outside the bounds of the core relations used to determine the enthalpies.

Then, for the ex-core detector, so we still make use of ex-core power range and intermediate range detectors located on the outside of the reactor vessel, where we have a high neutron fluctuated change for the power range detectors and then a high neutron flux and low doubling time or short period trips for the intermediate range detectors, primarily used in the startup regions for the intermediate range and, then, at power conditions or at all conditions the

high neutron fluctuated change.

And that one is made use of in the control rod ejection, as we model that with our stuff. And we will talk a little bit about that tomorrow in detail for the control rod methodology and the results from 15.4.8.

Then, the next two sets for the linear power density and the low DNBR, these are the ones that are based upon the in-core measurement system where we make use of the readings from the self-powered neutron detectors to determine the linear power densities where we will set limiting conditions for operation and, then, also, have a reactor trip based on a high linear power density. And we use a second max for that to avoid spurious trips.

For low DNBR, we make use of the currents, but, then, make use of them in a different manner to determine a DNBR value for the core. And, then, we have a series of different reactor trips. We have the limited condition for operation which sets our operational limits. Then, we have a symmetric reactor trip, again, based on now a second min to avoid a spurious trip by using the first min.

And, then, if there were conditions in the core that would lead to an asymmetric condition or

reduced accuracy potential of the in-core monitoring system, we have an imbalance in rod drop threshold which will shift the DNBR trip setpoint up to a more conservative value. Then, also, if we have indications of rod drops in more than two of four divisions or meaning more than two rods have dropped into the core, it will shift to a first minimum calculation and also raise the threshold level for that DNBR calculation, again, to accommodate the increased uncertainty or reduced accuracy of the SPNDs when there's potential for an imbalance.

And, then, one other thing that isn't really a reactor trip, but it performs a function, a safety function, to isolate potential dilution events, is that it at a certain level of concentration measured for the boron concentration, if it drops below a certain level, it will isolate the CVCS system to prevent or to isolate the most probable condition potential for dilution. So, it will isolate the coolant volume tank. We will talk a little bit more about that trip in a slide coming up.

But they have that set for power conditions and then standard shutdown conditions and, also, shutdown for refueling conditions when no RCPs are running. And I have got another slide that goes

into a little bit more detail on that.

For the RCS-related functions, these are more of your kind of like standard PWR reactor trips on whether it is low-low RCS flow rate in one loop, low RCS flow rate in more than one loop, low RCP speeds. So, all those are flow-related-type reactor trips.

If we have low hot leg pressure or low pressurizer pressure, so low pressure reactor trips, and then we also have a high pressurizer water level to avoid high levels and protect the RCS system, and, then, also, a high pressurizer pressure as well to protect the RCS boundary conditions.

And, then, also, if there is a high containment pressure indicated, there will also be a reactor trip as a complete backup.

CHAIRMAN BANERJEE: So, on a low power REA, some system-level trip would be required, right?

DR. WITTER: A system-level trip like the ex-core system trip or a low-pressure trip if it is a depressurization event, yes, for rod ejection, yes.

CHAIRMAN BANERJEE: So, which trips would be operational?

DR. WITTER: All these would be operational.

CHAIRMAN BANERJEE: Oh, all of these will, yes.

DR. WITTER: Right. So, for a rod ejection event, you could end up with a low pressurizer level. You could end up with a low pressurizer pressure, a low hot leg pressure, a high fluctuated trip from the low power conditions.

CHAIRMAN BANERJEE: Okay. And for the high power, of course, it would be the ex-core detectors?

DR. WITTER: For the high power, it would be ex-core detectors or potentially, again, if it is a depressurizer event, you will ultimately potentially trip on a low pressure signal and/or if the in-core monitoring system is operational, you may also trip on a low DNBR. It depends on the rod worths and things, and we will talk about that really more tomorrow. But it depends on the initial conditions and the rod worths and how much of a pulse you end up with, whether or not you actually trip on the ex-core system. If it is a worthy enough event, or if it does not trip on that initial ex-core power range detectors, you have to wait for another trip to come in, whether it is the low pressure or the low DNBR or something like that.

CHAIRMAN BANERJEE: You are going to discuss this in detail tomorrow?

DR. WITTER: Yes. Yes. Yes, we will have examples of the limiting events for hot full power and hot zero power tomorrow.

Okay. And, then, for the secondary side, we have the low steam generator pressure, high steam generator pressure, and then we also have low and high level steam generator water level conditions. And, then, there is a trip that forms to protect an excess steam demand potential where it is the high steam generator pressure drop. And, essentially, it is a variable setpoint that monitors the steam generator pressure at the point in time you are at. So, essentially, it serves as a variable power trip, where if the pressure drops in the steam generator 100 pounds less than where you're at, it will trip, as long as the pressure is less than 1088.

So, that this little nomenclature there means that if the rate of change is greater than 29 pounds per minute, it will trip. If it drops more than 100 pounds in the sequence time of the acquisition of the signal, it will trip, as long as the pressure is less than 1088. So, there is a clip there to kind of make it so that you are not going to

trip with higher pressures at that point in time.

Ultimately, you will always have the low steam generator pressure to come in if it is a very rapid steam increase or a steam line break, to mitigate the event there. So, this is kind of like another intermediate reactor trip that can help protect the secondary side in the reactor.

Any questions on that?

MEMBER STETKAR: They need more brackets around things.

CHAIRMAN BANERJEE: Okay. No.

DR. WITTER: Well, we had to put the squiggly lines because we couldn't use the brackets because then it indicates proprietaries.

MEMBER STETKAR: No, no, no. It's "A or B and".

DR. WITTER: Oh, yes. Yes, exactly. Yes, "A or B" --

MEMBER STETKAR: All of those.

DR. WITTER: Yes.

Okay. So, a little bit of detail about the high core power level trip or the enthalpy-based reactor trips. The high core power level calculates the core power in each of the loops based on the cold leg and hot leg temperatures and the hot leg pressure.

So, it determines the enthalpies of the inlet and outlet of each of the loops and determines a core power based on those loop indications.

It uses a constant volumetric flow rate for the loop and determines the flow rate by the density and that fixed flow rate. It is calibrated to also include a time-rated change of the mass inventory of the system. So, there's rates of change of the mass and the enthalpy to conserve the energy of the calculation. And, then, at the time of calibration, it is calibrated to the secondary calorimetric point. So, it will read the same as the secondary calorimetric at the calibration and surveillance points.

If it reaches the 105 percent power level, it will trip off as that, as long as you have two of four of the loops indicating reactor power above the threshold.

Typically, the in-core monitoring system will be the primary reactor trip, but this serves as a backup in case that system is not kicking in, and it preserves the analysis space where we determine the maximum power level that we would assume for the DNBR calculations.

For the low saturation, it just makes use

of the hot leg conditions and determines the margin to saturation, and if it drops below the threshold level, then it will also cause a trip of two of four of the loops, indicate margin less than reactor trip threshold. In this case, it was about 30 BTUs per pound, which I think is about -- well, I had better not venture a guess. I won't guess on the temperature difference there.

And that essentially serves, it can serve as, because it is not fixed to one pressure, it allows you to have that margin for any pressure level that you may be operating at. So, if the pressure drops and your temperature is still high, that saturation margin will drop and you will trip on that low threshold.

CHAIRMAN BANERJEE: So, the flow rates for the low flow rate, is that measured with those elbow pressure taps or how is that done?

DR. WITTER: I believe yes. For the low flow rate and the low-low flow rate, those are used on the delta Ps, and they will trip off on the indicated percent flow for those trip levels for the delta P trips there.

For the high core power level trip here, there is no use of the flow per se in there.

CHAIRMAN BANERJEE: Right. Yes. And these delta Ps are in the cold leg, right?

DR. WITTER: Yes, they are in the cold leg.

CHAIRMAN BANERJEE: And they are distributed around and you have got several taps in the periphery? How do you do that?

DR. WITTER: Yes, I am not exactly sure of the number of taps per loop that are available. It may just be one set per loop, but I am just not sure. I am not sure.

MS. SLOAN: Okay. We will follow up.

CHAIRMAN BANERJEE: Yes.

MS. SLOAN: We will find out.

DR. WITTER: Okay. The next aspect, which is not really a reactor trip function, but it serves as the chemical and volume control system isolation function, that we continuously monitor the boron concentration in the charging line. And, then, if the concentration drops below the threshold, it will isolate the system and isolate the control volume tank because that is the main source of a potential dilution.

Let's see. The setpoints, there are three levels of setpoints. One at power, where it monitors

the concentration in the reactor coolant system by monitoring the injection concentrations and having a calculation of the mass of the system and the concentration within the RCS. At power, it is set for an inlet temperature, and they are updated periodically as a function of the burnup or the boron letdown curve.

For the standard shutdown conditions, here it is a similar measurement of the boron concentration as a function of the injection rate and the concentration calculation. But now it also includes a temperature compensation for the mass of the system as you are cooling down.

So, in this case, the setpoint is actually a function of temperature and periodically updated as a function of burnup. And the curve here shows an example of the setpoint implementation into the system, where at zero gigawatt days where you will have a higher boron concentration required in the core, the boron concentration setpoint is a higher value. And, then, later during the cycle, the concentration setpoint would be lowered down, as the boron letdown curve comes down and as a function of temperature. As you cool down, obviously, you need more boron concentration to maintain your shutdown

margin and avoid criticality.

So, the setpoint measures the cold leg temperature to determine what value is the threshold value to be used, but it also uses that cold leg temperature to determine the mass of the RCS as well, to verify the concentration within the system.

CHAIRMAN BANERJEE: I notice that we don't have any coordinates here because I guess it is proprietary or something.

DR. WITTER: Right. Right. The boron concentration in ppms --

CHAIRMAN BANERJEE: But at least at some point, I imagine it is in the -- but if you could --

DR. WITTER: Right, the setpoint itself would appear in the core op report, and they would be updated for each cycle. At each of the burnup points that you would implement the new setpoints in, you would have the tabular set of the concentrations that you would use as a function of temperature in there.

CHAIRMAN BANERJEE: But this is just a qualitative --

DR. WITTER: This is a qualitative trend of how it would actually look.

CHAIRMAN BANERJEE: Sure. Right.

DR. WITTER: In the Chapter 4

descriptions, the boron letdown curves I think for cycle one had the boron concentrations of around 1,000 ppm natural at the start of the cycle. And so, this setpoint would be somewhere less than that 1,000 ppm level. Typically, it is about half of what that hot operation concentration is.

CHAIRMAN BANERJEE: How widely does that vary typically with fluid temperature?

DR. WITTER: Well, up at this end would be your operating conditions, and at this end would be your refueling temperatures for that part of the curve.

CHAIRMAN BANERJEE: Okay.

DR. WITTER: I would have to look at the calculation. Maybe later today I can get the numbers for that.

CHAIRMAN BANERJEE: And any maldistribution of the boron or anything, does that affect --

DR. WITTER: Well, this is one reason why they have the trip set up as a function of whether or not the RCP pumps are running, which they will assume that there is adequate mixing. But, then, in shutdown, when no RCPs are running, then they will shift the concentration over to the IRWST or the in-

containment refueling water storage tank concentration minimum, based on that. So, that level will be shifted up to a much higher concentration if the RCP pumps aren't running to essentially handle the potential of having slugs moving around.

So, if the RCP pumps are not running, then it will shift to a concentration based off of a threshold from the IRWST.

CHAIRMAN BANERJEE: So, do you do any CFD or how do you sort of look at this phenomena?

DR. WITTER: In the shutdown modes?

CHAIRMAN BANERJEE: When the mixing may be expected to be less effective.

DR. WITTER: Okay. That is not my area of expertise.

MS. SLOAN: You're talking about the boron dilution analysis, Dr. Banerjee? I think that also is still being worked with the staff. So, we still have open issues left with the staff. I don't think we are prepared today to talk about the boron dilution evaluation.

CHAIRMAN BANERJEE: Okay.

MEMBER STETKAR: Let me ask a question about the boron. I know you do have open items, but none of them address the question I had. So, I might

as well get it on the table.

During shutdown, I understand the isolation signals. They close the two VTC suction valves and they close the RHR letdown isolation valve. Those valves and the isolation signals come from only two divisions of the protection signal, the divisions one and four. And in particular, the RHR letdown isolation only comes from division four.

Now, during shutdown, if I have a failure or an excessive removal of boron out in the cleanup loop, which is completely unaffected by the VTC suction valves, a single failure to close that RHR letdown isolation valve would seem to keep that boron dilution going forever. Now not forever, but I want to know what type of analysis you have done to show how much boron dilution I can get during cold shutdown conditions with that single failure to isolate the RHR letdown loop, with the most likely condition being some sort of fault out in your cleanup system because that is where we always had problems with resin beds, for example, not being correctly saturated or people cutting in new resin beds, or something like that.

So, I am curious, and I did not see that type of questioning. The only reason I brought it up is I did not see that type of questioning in the

staff's discussion of the boron dilutions. So, put that on your plate.

MS. SLOAN: Okay. I made a note.

MEMBER STETKAR: Thank you.

DR. WITTER: Okay. Next we move on to the aspects of the in-core monitoring system. This figure depicts making use of the ex-core neutron detectors, as historically has been done, which monitors the periphery of the core. It can provide a core axial power shape.

We have power range detectors in the top half and bottom half of the core, and, then, the intermediate range they are centered about the core center line access.

And historically, other plants may have used the control rod positions to help infer core internal power distribution, but, really, the aspect was you were using the ex-core detectors to give a view into the core.

And, then, they may have had in-core monitors for just surveillancing and providing some feedback of the core power, but not as a protection system function.

By making use of the in-core detector system, now we can take a slice into the core and

actually see the local power distributions at certain locations radially and axially within the core, and we can monitor the local power and we can directly measure the radial and axial power distribution effects as they may change due to any sort of accident that may induce any symmetry or control rods move.

So, this figure shows a power shape with some of the control rods from our controlling bank inserted about one-third of the way up the core. So, you can see these blue regions indicate where the control rods would be located. And you can see the power distribution kind of peak towards the bottom when you insert the rods in that far.

Whereas, the ex-core detectors, you wouldn't really necessarily see that radial distribution inside the core, and now we are able to see that by making use of these three-dimensional array of our 72 self-powered neutron detectors.

And so, now how do we make use of those in-core detectors? The first aspect of this, and some of this has been discussed in the Chapter 4 presentation, but we will spend a little bit more time in some of the detail and how we actually make use of them in the transient analysis for the Chapter 15 events.

So, for the DNBR, for one of the divisions, one of four, each of the 12 strings -- there are 12 strings located regularly throughout the core. And we discuss in Chapter 4 where they are located, coming in off the lances from the top of the core. So, we have bottom penetrations for instrumentation.

So, we have 12 SPND strings, each of which has six elevations of SPNDs located at six different axial elevations. Each of those 12 strings is calibrated to read the hot channel. So, in this figure, the hot channel is indicated by this red line showing the hot assembly or the hot channel.

Each of the 12 strings at the time of calibration, using the aeroball measurement system, and getting the fine core map of the power distribution, determining the hot channel, each one of the strings, then, is set to read that flux shape or that linear heat generation rate of that hot channel. So, if the hot channel is the smooth shape in the fine detail, then the six elevations of SPNDs are calibrated to read that linear heat rate for that hot channel, to reconstruct the limiting DNBR channel.

Those six elevations are used to reconstruct that flux shape at any given time post

calibration between the calibration periods there. At the time of calibration, all of them will read identical. But, then, after that, they are left to their own wiles to monitor the linear heat rates locally, but tied to that hot channel, monitoring that hot channel.

The DNBR is estimated by using that reconstructed power shape, but also using each of the loops. In this case, each division has its own reactor coolant loop. We use the inlet temperature, the flow signal based on the reactor coolant pump speed, and then the pressure of the pressurizer, fed into the DNBR algorithm, which is a single-channel algorithm for the online monitoring.

And, then, it determines a sensed minimum DNBR. It will determine the DNBR at eight different elevations in the channel for each one of those SPND locations, determine the minimum for each of the strings, and then each of those 12 strings are compared for the second minimum DNBR. And that is compared against the reactor trip threshold. If the reactor trip threshold, if the minimum DNBR is less than the threshold, it will cause the reactor to --

CHAIRMAN BANERJEE: So, the flow, the pumps are calibrated against your secondary site flow

or how are the pumps' speed --

DR. WITTER: Right. Right. The pump speed and the volumetric flow rate is calibrated and used as a constant. The inlet temperature and pressure are then used to determine the inlet density, to then provide a mass flow rate or a mass flux for the core signal based on the pump speeds. So, the pump speed serves as the volumetric flow rate adjuster.

CHAIRMAN BANERJEE: So, you said they are divided into quadrants.

DR. WITTER: Quadrants only by loop. Each division handles each of the RCS loops, will provide its own protection system channel calculation of the DNBR. So, each of the loops will provide the inlet signals for the thermal hydraulic boundary conditions.

All 12 SPNDs are used in each one of the four divisions. Then, you will end up with 12 calculations times four of DNB to be then compared against the thresholds in each of the protection system divisions. If you have two of four divisions indicate the DNBR is less than the threshold, then you will have a trip.

CHAIRMAN BANERJEE: So, let me just try to understand the physics of this.

DR. WITTER: Yes.

CHAIRMAN BANERJEE: You have got all this flow coming together in the plenum.

DR. WITTER: Yes.

CHAIRMAN BANERJEE: Somehow it gets distributed. We haven't looked into this yet. And, then, it goes to the channels.

DR. WITTER: Yes.

CHAIRMAN BANERJEE: So, you have a way to figure out how it distributes. Okay. Once it is in these channels, of course, there's all sorts of other things that go on, crossflows and so on. And eventually, you have a way to do the calculation for the hot channel.

So, you have a DNBR algorithm clearly because that can only be done for a channel at your testing facility at KATHY, wherever it is, or Karlstein -- I don't know -- does this stuff. And then, you reconstruct using some sort of table lookup or computer code, or whatever, the flow. So, that flow is an inferred quantity?

DR. WITTER: The flow is an inferred mass inlet, inlet mass flux to the channel. That's right.

CHAIRMAN BANERJEE: Well, there are crossflows, obviously, as well.

DR. WITTER: There are, but that all goes into the correlation and the bias factors that are put into the simulation algorithm. It is the single-channel representation, but there are bias factors and correlations that fit it back to the full crossflow, full channel, open channel models.

So, this all ties back to essentially our LYNXT database and the correlation limits and CHF correlations, which are tied to the test data from Karlstein.

CHAIRMAN BANERJEE: I assume that you have got something that is like COBRA IV or something running.

DR. WITTER: Yes, that's right. Right, COBRA LYNXT, exactly.

CHAIRMAN BANERJEE: LYNXT. And you have some little calculation or a lookup table or something which takes your RCS speeds and reconstructs things and comes up with the flow?

DR. WITTER: Yes.

CHAIRMAN BANERJEE: Now is that all validated? Because the prediction of DNBR obviously depends on the flow as well as, of course, the power and the power shape, and all these things that go into that. But the flow is all the measurements you have

of the RCS pump speeds. Maybe you have the delta Ps around the taps. Okay. At this point, you have got sort of an overall flow in each of these loops, which are all getting mixed up and coming to these channels. And, then, you have to reconstruct.

What is the sort of validation of that that is done?

DR. WITTER: Where that comes in is it really kind of comes into the whole setpoint and transient methodology for the topical report of ANP-10287P, where the buildup of all the uncertainties and the allowances from a hydraulic flow or the flow maldistributions, the uncertainty of the measurements of all the boundary condition inputs, and then, also, just the measurement accuracy or tracking error of the SPNDs themselves. Then, tying the single-channel algorithm to the open channel with bias factors and correction factors as well. All that gets fed into both the setpoint calculation to have a threshold that is high enough or far enough removed from the actual CHF condition, and, then, within the transient methodology, monitoring the sensor lags and inputs from that aspect of it. The two tied together handle, I guess, the uncertainties of not really knowing what that flow is in that one particular channel at the

real-time condition.

So, the setpoints themselves are set up to handle the conditions of all the uncertainties associated with the real plant versus test versus open-channel versus single-channel calculations of the DNB.

CHAIRMAN BANERJEE: So, has that methodology -- you know, I am just coming into this in the middle.

DR. WITTER: Yes, yes.

CHAIRMAN BANERJEE: So, has that methodology -- I presume there is a whole suite of computer codes, including your LYNXT code, and so on, which have probably been accepted or approved at some point. The applicability of those codes to this specific situation with a larger plenum, with a different mixing situation, has that been explored already and discussed?

DR. WITTER: I imagine it is part of the --

MS. SLOAN: Well, let's just be clear. The methodology is specific to EPR. So, it is not a generic, it is not necessarily a generic methodology. There was a special in-core trip setpoint methodology developed for EPR, and that is what Dr. Witter is

referring to, and it has been reviewed by the staff. I am not sure we have come -- did that come up in Chapter 4?

DR. WITTER: A little bit in Chapter 4.

MS. SLOAN: I thought we covered a little bit of this material in Chapter 4. I am not recalling the specific questions that you are asking have been raised before.

CHAIRMAN BANERJEE: Yes, I am simply trying to understand whether the methodology which you are using has been explored and discussed and approved by --

DR. WITTER: It is in the process of approval.

CHAIRMAN BANERJEE: Okay.

DR. WITTER: We don't have the approved topical report.

MS. SLOAN: We have the draft SER.

DR. WITTER: We have the draft SER.

MS. SLOAN: We have a draft SER.

CHAIRMAN BANERJEE: Okay.

MS. SLOAN: And we discussed this in some detail at the Chapter 4 meeting.

CHAIRMAN BANERJEE: But will it come to this Committee under some other chapter, the

Subcommittee, or has the Subcommittee already seen it?

MS. SLOAN: I believe it came in Chapter 4.

CHAIRMAN BANERJEE: So, it is already --

MS. SLOAN: It was folded in in the Chapter 4 review along with some other fuels-related methodologies.

CHAIRMAN BANERJEE: Okay. And so, the issues related to flow distribution, and so on, were covered at that point?

MS. SLOAN: You know, I think the answer is yes. We will have to follow up. I don't remember these specific questions that were asked, Dr. Banerjee. We will have to go back and look at the transcripts.

Okay. Go ahead, Jerry.

MR. HOLM: This is Jerry Holm. I work for AREVA.

The questions of mixing the lower plenum and the flow mixing device, those were all RAIs on Chapter 4.

CHAIRMAN BANERJEE: Okay.

MR. HOLM: They were discussed a little bit in the ACRS meeting on Chapter 4, but more in the background during the staff's review of the --

CHAIRMAN BANERJEE: How to go from 1/6th to full scale and --

MR. HOLM: Yes, all that is really addressed in that topical report, that ANP-10287 --

CHAIRMAN BANERJEE: Okay.

MR. HOLM: -- where we have a draft SER from the staff.

CHAIRMAN BANERJEE: And just to sort of give me an idea, I assume you used CFD tools of some sort to go from 1/6th to full scale for mixing. Or how did you bridge that gap, so that you know what is getting to these channels and things?

MR. HOLM: You know, LYNXT is like a COBRA code.

CHAIRMAN BANERJEE: Right.

MR. HOLM: So, it has got open-channel calculations.

CHAIRMAN BANERJEE: Within the core, I see how you are doing this, of course. Yes.

MR. HOLM: Right.

MS. SLOAN: I think he is asking about the scaling of the flow mixing.

CHAIRMAN BANERJEE: Yes.

MS. SLOAN: I think we are back to an earlier topic that you brought up. If we have these

mixing devices and they have been tested, how was that scaling evaluated for those test facilities to scale that up to applicability to EPR? I think it is a slightly different question.

MR. HOLM: Yes, I think that question was asked also, but we would have to go back and review the RAI response for Chapter 4.

MS. SLOAN: All right.

CHAIRMAN BANERJEE: Okay.

MS. SLOAN: We will go back and check it.

CHAIRMAN BANERJEE: So, the staff asked it and you responded?

MS. SLOAN: We are going to go back and check the questions.

CHAIRMAN BANERJEE: Okay. Because, clearly, that is an important issue, that you have got to get the flow right because your DNBR is going to be fairly --

DR. WITTER: Or to make sure that it falls within the uncertainties that you have accommodated --

CHAIRMAN BANERJEE: Sure. Yes.

DR. WITTER: -- in your setpoint.

CHAIRMAN BANERJEE: Yes. So, if you have quantified the uncertainties, of course, it is there.

DR. WITTER: Right.

CHAIRMAN BANERJEE: All right.

DR. WITTER: Okay.

CHAIRMAN BANERJEE: Yes, you can move --

DR. WITTER: I think that is it with the DNBR part of it.

CHAIRMAN BANERJEE: And the DNBR, just for me to get, you have developed this, I assume, in your facilities in Germany --

MS. SLOAN: Yes.

CHAIRMAN BANERJEE: -- with full-scale, typical --

MS. SLOAN: We did specific testing for this.

CHAIRMAN BANERJEE: Yes.

MS. SLOAN: And there is an approved CHF correlation for use in EPR.

CHAIRMAN BANERJEE: With different flux shapes, and you have explored all that stuff?

DR. WITTER: Right, uniform and cosine.

CHAIRMAN BANERJEE: Yes.

DR. WITTER: Right.

CHAIRMAN BANERJEE: All right.

DR. WITTER: And the 14-foot height test.

CHAIRMAN BANERJEE: With stop peaking and things like that, well, something like that profile

you are showing there?

DR. WITTER: Whatever the standard CHF testing they would do for non-uniform flux shapes is what is in this for the ACH 2 correlation that we are using.

One other aspect about this is that the reactor trip thresholds, depending on the number of failed SPNDs, if one SPND of the string has failed, it will fail the entire string from consideration for DNBR calculations. And so, let's say if one fails, then you are left with 11 strings that are left to monitor the core. And so, the reactor trip thresholds will also shift based on the number of failed detectors up to allowable number of five. And then, if there are more than five, it either puts us into an LCL clock period or, if there are seven or more, then it will cause an automatic reactor trip.

So, there is a limitation on the number of SPNDs that can be failed, and the thresholds will shift depending on the number of failed SPNDs.

CHAIRMAN BANERJEE: Okay.

DR. WITTER: Okay. For the LPD, it is a little bit simpler, a little bit more straightforward. But we make use of the currents coming from the SPNDs, and now we calibrate them in a different

manner.

This just depicts two axial slices of the six that we have. In each of the axial slices, there are the 12 SPNDs. Each one of them will be calibrated to read the local hotspot in that zone, axial zone, of measurement.

And so, in this case, this region here was the hotspot or the hot local linear heat rate. Each of the 12 SPNDs in that elevation will be calibrated to read that value. Each of the six elevations have their own calibration point for the hotspot in its axial zone.

So, each one of the six axial slices will be calibrated to read a particular hotspot value for that zone. This allows, instead of just globally tying all 72 to one spot, now we are tying six elevations to six different hotspots.

It can give us a feel for changing in power distributions from either global power, rod influences, xenon transients, et cetera, or asymmetries induced into the core.

For the high linear power density function, all 72 SPND readings are then ranked and sorted, and then compared against the reactor trip threshold. And again, now we have the second max as

the reactor trip to avoid a spurious trip on the first max. And again, if we have a number of SPNDs that are failed, the reactor trip set point is also lowered in this case to a more conservative value, depending on the number of failed SPNDs.

CHAIRMAN BANERJEE: So, is this methodology exactly the same as, say, the EPRs that you have under construction right now?

DR. WITTER: The setpoints, well, actually, the trip functions are identical, yes.

Okay. As an example of some of the aspects of having the calibrations tied at each axial elevation to one level, and making use of the LPD signals to shift the thresholds for the DNBR reactor trip threshold comparisons, this slide, it is a little busy, but it is an example of, if we calibrated one of the axial slices, in this case the hotspot was at 8.6 kilowatts per foot. All 12 SPNDs in that elevation were reading identically at the time of calibration.

Then, if there was something that happened during the course of operation between calibration periods, in this case if, say, the centered J-09 control rod dropped into the core, we have symmetric pairs of strings. Each of the 12 SPNDs have a symmetric partner. So, there are six pairs of

symmetric locations. In this case, the dashed line indicates the symmetric pair in this case.

Because the J-09 dropping in the center of the core, it is a symmetric event, this wouldn't necessarily pick up that there was an imbalance in the core, although the readings of the SPNDs at the edge of the core are indicating higher than the calibration point, and the center of the core are reading lower, just due to the flux perturbation due to the rod drop.

If the rod drop wasn't picked up, the rod drop 104 divisions, the center rod is in one of the divisions. If that is picked up, the imbalanced threshold would be triggered.

There is no imbalance indicated because all the symmetric pairs are reading identical in this case. And so, then, the DNBR normal symmetric setpoint would have to also accommodate this single rod drop in its setpoint for the conditions to monitor the DNBR. In this case, these are the LPD readings, not the DNBR readings.

If it happened to be an asymmetric event, or in this case if we dropped the NO7 control rod, in this case you can see the flux tilt from the hot orange side, reading about 10 kilowatts per foot, down to the colder side, about 5 kilowatts per foot. In

this case, the imbalance would be more likely triggered based on the calculation of the SPNDs, and it would trigger the DNBR threshold to be raised up. And then, if the DNBR happened to be lower than that higher DNBR threshold, you would have a reactor trip based on that asymmetry.

So, this provides an indication of sort of the real-time monitoring that the SPNDs can provide, making use of the symmetric pairs, but also making use of the maximum readings of each of the LPDs as well.

For asymmetry structures, asymmetric cooldowns or heatups, the asymmetry may not be as dramatic as this, but it will still be able to pick up the readings and any shifts in the power distributions, such that even if the core power remains the same, one part of the core may get hotter and ruin your heat generation rate. And you would want to monitor that for the DNBR and be able to trip if the DNBR thresholds were exceeded.

Okay. So, that provides the overview of the actual trip functions, how we calibrate them. Now we will talk about, well, how do we actually make use of them to verify that during the safety analysis portion of the Chapter 15 that we did not violate any of the SAFDLs.

CHAIRMAN BANERJEE: So, I think if we could plan to finish this section, which is to slide 26, and then we will take a break whenever that comes.

DR. WITTER: Okay.

CHAIRMAN BANERJEE: Okay. Is that okay with you?

MS. SLOAN: I need to step out for just a second. I might ask Mr. Gardner to step in for me.

CHAIRMAN BANERJEE: Sure. All right.

DR. WITTER: Okay. So, we will talk a little bit about our in-core transient methodology. Again, this is described as an overview in Section 15.6.3, I believe, and also in ANP-10287P, which is the setpoint and transient methodology topical report.

Okay. This provides kind of a process map of how we worked through the transient analysis for any one of the events in Chapter 15. So, the normal S-RELAP runs for the monitoring or simulating the plant system responses is the typical historical method of analyzing the plant response to any of the events, whether it is uncontrolled bank withdrawal, an increase in steam flow. An S-RELAP analysis is performed for varying conditions to determine various scenarios of plant conditions and provides bounding conditions which will then be used for the inlet

temperature, the flow rates, and the pressures of the reactor plant and the core power levels, to provide that as a boundary condition into our transient in core simulation algorithms.

Also, upfront work is done in the neutronics area with our PRISM calculations, our PRISM code, to provide power distributions for multitudes of different power distributions and also provide simulated SPND responses to the core for those different power conditions.

CHAIRMAN BANERJEE: Where does your LYNXT code come in?

DR. WITTER: The LYNXT code part comes in as part of the setpoints and part of the bias factor aspects, where the online algorithm aspects are tied back to the LYNXT code through the bias factors as part of the methodology.

CHAIRMAN BANERJEE: So, that ANP-10287P is --

DR. WITTER: Is the heart of everything. Yes.

CHAIRMAN BANERJEE: Yes. This is actually a suite of codes?

DR. WITTER: It is a suite of codes that end up being combined to build up the validation of

the transient in-core simulation of the protection system itself.

So, the protection system isn't part of the S-RELAP model. Now we run it, because S-RELAP also uses the point kinetics to drive the core power, where we need core power shapes for the in-core monitoring system.

So, we take the core power distributions from PRISM. We take the plant bounding conditions from S-RELAP. We take the setpoints, which are determined from a series of ties to LYNXT calculations for the database of the correlations and the power shapes, to determine setpoints for wide varying conditions of pressure temperature flows in the core power distributions to determine the most limiting setpoints for DNBR and LPD to provide the 95/95 setpoint values.

Then, we use those setpoints. Now the process is to validate those setpoints and also verify that we have adequate dynamic compensation or dynamic response of the protection system itself.

So, by simulating the protection system, where as RELAP currently does with a series of control variables and different things for the normal sorts of trips, like low pressurizer pressure, our in-core

simulation algorithms simulate the protection system for the DNBR and the LPD with the various process logic diagrams from the protection system itself to provide -- the protection itself has a series of lead-lag functions. We also handle the instrumentation response delays from the signals being provided from the S-RELAP bounding conditions.

So, this simulates the protection system to the best that we can and, also, handles taking in all the SPND readings, providing the filtering of them, and then determining the DNBR in an online sense using the same online algorithm that the protection system will use to determine a trip time, if there needs to be one, from the bounding conditions during the event.

So, for one particular PRISM power shape for SPND readings for a particular S-RELAP plant event, we then simulate that event, taking those bounding conditions, and proceed along to determine whether or not we predict --

CHAIRMAN BANERJEE: So, the filtering is done in the simulation --

DR. WITTER: Right.

CHAIRMAN BANERJEE: Not in PRISM?

DR. WITTER: Not in PRISM, no. No, PRISM

provides the raw SPND signals for a static condition. Then, those are used, they are scaled based on the power level from those RELAP conditions to provide a response to the core for that.

If an in-core trip is predicted out of the two of four division logic, we will run through a series of these. We will keep track of any reactor trip time that is predicted by this in-core system, whether it is an LPD trip or a DNBR trip. We will just continue the log of those as we sequence through all the various power shapes for one particular S-RELAP event.

CHAIRMAN BANERJEE: What is chi again?

DR. WITTER: That is a quality. There is also a quality trip to verify that the calculation of, again, the enthalpy parameters, and that your quality is within the limits of the DNBR correlation itself. The correlation will have quality limits. So, to ensure that our calculation of the DNBR is adequate or correct, or within allowances, there is a trip on the quality calculation.

CHAIRMAN BANERJEE: Just an enthalpy calculation?

DR. WITTER: Right, it is an ex aequali enthalpy-based calculation. As long as that quality

is within the limits, that validates the DNBR calculation. So, it is a reactor trip. If that quality out of whack or if it is too high, then it invalidates the DNBR calculation and says, well, I can't trust my DNBR calculation; I had better trip the plant.

CHAIRMAN BANERJEE: Really?

DR. WITTER: So, it protects --

CHAIRMAN BANERJEE: So, you don't switch to a different DNBR calculation at that point?

DR. WITTER: No. It will trip the plant on the high quality.

CHAIRMAN BANERJEE: What is the limit on quality?

DR. WITTER: Right now it's --

CHAIRMAN BANERJEE: Unless you can't say it in an open session.

DR. WITTER: I think it is in -- it should be. Well, I had better --

MS. SLOAN: We will look it up. Better to look it up --

DR. WITTER: Right, we will look it up.

MS. SLOAN: -- and be certain than to try to get a --

DR. WITTER: It is pretty high.

CHAIRMAN BANERJEE: Okay.

DR. WITTER: Okay. So, we keep track of all these in-core trips. And within the algorithm here itself, we will start the events at the limiting condition for operation. So, we bring it to the maximum, and then we will run the simulation. Then, we will continue to back off on the linear heat rate or the F delta H, to essentially move ourselves further and further away from the LCO until the point where we reach a maximum DNBR that we want to start the event or a minimum LPD that would start the event.

So, we run through a series of power shapes, initial boundary conditions, initial plant conditions, and run through the sequence for each of the event types and keep track of all these reactor trips.

In some cases, each of those will be checked for the amount of undershoot or overshoot. I will have a figure that shows this a little bit before everybody gets too confused.

But, then, ultimately, for a few of the trips that we predict, we will provide those back into the S-RELAP to have them rerun the event with that trip that we predict for the trip, and then allow them to insert the reactor trip with the scram delays and

the loss of offsite power. So, we also get a loss of flow event coupled to this. We run that through the simulation one more time to verify that the undershoot of our uncompensated or real DNBR does not violate. And I will show you what I mean by that.

CHAIRMAN BANERJEE: How many scenarios do you run through?

DR. WITTER: Many, many, many.

CHAIRMAN BANERJEE: Yes, it seems --

DR. WITTER: So, there is no more one limiting event where the maximum overpower and minimum pressure and come up with a fixed power shape assumed and a maximum F delta H assumed. We run through the whole suite because we are monitoring it online in real time.

CHAIRMAN BANERJEE: So, did, say, AREVA for its PWRs in France, for example, follow a procedure like this in the past?

DR. WITTER: They have followed a similar procedure where they will run through many, many, many, many, many shapes to determine their tracking error. And, then, they will apply a maximum tracking error to the different events and run through to verify that their setpoints are adequate as well.

So, yes.

CHAIRMAN BANERJEE: And this is done online?

DR. WITTER: The online, meaning, yes, we are monitoring the shapes --

CHAIRMAN BANERJEE: Yes.

DR. WITTER: -- online, yes, and comparing them against one setpoint. So, what we are doing here is simulating multiple possible scenarios as part of the safety analysis to cover the range of potential real operating conditions that you might end up at.

CHAIRMAN BANERJEE: I see.

DR. WITTER: So, online it is just constantly monitoring the power shape and the SPND readings, determining linear power density that you are checking, and then determining a DNBR that you are checking. So, it is an online real-time estimate of the DNBR using in-core measurement systems rather than inferring it from some assumed power shapes from the ex-core detectors.

Okay. So, an example of what that whole process map does for one particular event, as provided in the Final Safety Analysis Report in Section 15.1.3 for the increased steam flow event, this shows a typical response for the DNBR for one of the multitude of scenarios that we ran through.

In this case, the dark, solid line is what the transient in-core simulation algorithms would provide with all the compensation, the dynamic compensation, of the lead-lag filters and the instrumentation response for the signals providing the online algorithm in the protection systems. So, this is essentially what your plant computer would be indicating for the DNBR during this event.

In this case, we have our reactor trip threshold setting as the dashed line here, which has reached about six, six and a half seconds into this event. At that point in time, we provided that reactor trip time to the S-RELAP analysis, and they induced that trip at that time. Then, the rods begin to insert after the signal delays. And essentially, at this point, there is enough of a power turn to cause the DNBR to increase again.

What we then also check is what we call the uncompensated minimum DNBR, which is based on a bunch of static snapshots of the plant condition, bounded conditions, using a static snapshot. So, it is not a transient calculation. It is a static calculation of the DNBR using the power shape, reconstructed power shape, and then the inlet conditions, which in this case we used the core exit

pressure instead of the pressurizer pressure. So, that is why there is a difference between the DNBR values here.

And what we do is we ensure that with the post-trip and the loss of offsite power, with a loss of flow, that the undershoot of this DNBR does not drop below the reactor trip setpoint. And that was what we call our SAFDL for these conditions.

So, we verify for each of the events that the uncompensated minimum DNBR signal never drops below the reactor trip setpoint. This was validated for each of the events.

CHAIRMAN BANERJEE: But the reality that you expect is the blue line, isn't it?

DR. WITTER: The reality is the measured.

CHAIRMAN BANERJEE: Yes.

DR. WITTER: The measured is the blue line, right. And that can go below, as long as we trip at that point, and we verify that we don't let the uncompensated drop below the trip setpoint, then we have validated that setpoint. If, for some reason, this red line had dropped below the dashed reactor trip SAFDL line, we would either have to change the setpoint to have a higher setpoint or change the lead-lag function such that the rate of change of this blue

line would be such that it would be indicating the DNBR would be decreasing more rapidly, so that we would, let's say, increase the gain on the signal, such that it would trip earlier, such that the real one never dropped below it because the trip would come in earlier for that.

And this is just one particular case. The most challenging ones were the cases that came in most quickly or had the highest rate of change of the blue line. And so, we had to validate that, that we never let the red line drop below the dashed line for that.

So, that is how the DNBR events, most of the events were handled with the DNBR trip. There were a handful that were handled by the high LPD trips.

In this case, a very similar aspect is done, except for in this case it is just the raw reading that you are compensating or lead-lag filtering again. So, again, the dark blue line is the sensed, measured from the protection system. It reaches a reactor trip threshold for high LPD at around 16 seconds or thereabouts. We provide that trip time to the S-RELAP analysis. They put it in, put in the reactor trip signals with the scram curve for the reactivity insertion rates and, also, the loss

of flow. Then, we get the turn in power.

In this case, the turn happens much more quickly because it is truly tied to the neutronics, and not to a calculation of a delayed thermal hydraulic condition for that.

Let's see. That should be it with the LPD signals.

So, the number of cases that we look at, just for this one uncontrolled bank withdrawal event, all these dots essentially are all the various scram times that we have predicted using our transient in-core algorithm simulators, where the lower left-hand side shows the full suite of S-RELAP runs for beginning of cycle, end of cycle, reactivity conditions, different rates of addition, different magnitudes of addition of reactivity for the control banks.

Then, moving up and down the initial F delta H of the assembly, where we are limited by the tech spec limit of 1.7, and then it would back off, as I mentioned, automatically to continue to reduce it down to a certain level. So, we have variable multiple starting times.

The general trend is that, if you have a lower starting F delta H, you will have a later trip

time. The general trend, also, is the longer trip times or maybe not even a reactor trip based on the in-core system is such that the event was benign enough in the RELAP world that we didn't need the in-core trip. And so, later scram times meant typically lower initial F delta H's or less challenging plant responses to the system.

Zooming into the less-than-30-second timeframe for the reactor scram, you can see there's still a very large number of points that can be used, and we would pick one of these points along here to then provide a representative plot back into the Final Safety Analysis Report to provide one of the plots similar to this for the Chapter 15 events.

So, essentially, we look at several thousand different conditions as a combination of S-RELAP events, power shapes, initial starting conditions, and that we have validated each one of those through the process of the topical report methodology approach, that we have predicted for all of the cases that we predicted in core trips, all of them had valid setpoints and that we did not violate the SAFDLs in any of those cases.

Okay. So, finishing up, as a broad summary of those, each of the events were analyzed in

a spectrum of times for reactor trips, based on the in-core monitoring system, and we have several thousand renditions for each of the events.

The shortest time to trip and also the highest rate in signal change are typically the most adverse for the minimum DNBR undershoot and the LPD overshoot.

For events that were too fast to be covered by the in-core system, the limited conditions for minimum DNBR and LPD are specified to protect those SAFDLs in the events that are too fast, say, for instance, the loss-of-flow event.

For some combinations of 3D power shapes, initial conditions, and magnitudes, there may not have been a trip predicted by the in-core system. In those cases, either the plant system trips or there was no system trips required to protect the plant. Those would just be covered by your LCOs or no need for a trip in those cases because the SAFDLs were not challenged.

CHAIRMAN BANERJEE: Let me ask you, I mean, you are sampling a space of many dimensions here, right? So, you have got many thousands of possible cases, and you have run, you know, you have sampled it in some way. And, clearly, you can't do

everything. So, there are some cases you haven't run. Are you using some statistical methodology to assure yourself that you are within some level of uncertainty due to the fact that you are sampling these thousands of cases, but not everything? Obviously, you can't do that.

So, do you use some statistical methodology to put a little uncertainty on what you might have missed?

DR. WITTER: Right. Within the setpoint determination itself, that is where the vast majority of the statistical treatment is handled. When we are doing this sample set on the transient side of it, it is allowing us to look at more of a spectrum of, instead of just one stated point that might be maximum power, let's say, in a historical sense, we are looking at the spectrum of conditions for pressure temperature flow and power, local power, that can lead to a DNBR or LPD situation.

CHAIRMAN BANERJEE: I presume that gives you some relief. Otherwise, why would you do that?

DR. WITTER: By viewing more of them --

CHAIRMAN BANERJEE: Yes.

DR. WITTER: -- it is providing a better, I guess to say, well, how many shapes do we have to

use to get a 95 confidence in our setpoints --

CHAIRMAN BANERJEE: Right.

DR. WITTER: -- which are 95/95, determined to be a 95/95 --

CHAIRMAN BANERJEE: So, you can cut down conservatism somehow this way, right? The 95/95 and looking at this?

DR. WITTER: By looking at more of the plant conditions over the course of all times of the event and evaluating them, taking the RELAP plant system boundary conditions for all conditions without our trip applied, until we predict a trip would be applied, we see the full spectrum of how the protection system would really sense the DNBR or LPD in the plant itself.

CHAIRMAN BANERJEE: I understand the methodology, but compared to a more prescriptive way of doing this, it allows you probably to set the trips somewhat lower or less. Is that what happens at the end?

DR. WITTER: Let me --

CHAIRMAN BANERJEE: I'm just asking, why do this, I mean?

DR. WITTER: Right, why do this?

CHAIRMAN BANERJEE: Yes. There's

obviously a reason.

DR. WITTER: I guess it's, well, I will see if Keith Maupin can help me out with this.

MR. MAUPIN: Yes. Yes, my name is Keith Maupin.

It sounds like the question is getting at, does the use of a statistical methodology to establish the in-core trip thresholds that is statistical in nature allow you to have less conservative trip thresholds?

CHAIRMAN BANERJEE: That is exactly the question.

MR. MAUPIN: And my thinking on that is that, absolutely, it does. It allows us to, by sampling a large number of situations or input conditions, establish what the 95/95 is for that, rather than to take what has been identified as the most limiting combination of conditions.

CHAIRMAN BANERJEE: Right.

MR. MAUPIN: That is the subject matter in 1037.

CHAIRMAN BANERJEE: So, how much margin? I mean, so you know, if you take something which is different, but let's take a LOCA calculation. If you go into a very prescriptive way of doing this, rather

than sampling of 59 runs or whatever people do --

MR. MAUPIN: Right.

CHAIRMAN BANERJEE: -- it buys you quite a bit, a couple of hundred degrees, actually, in some cases, not in all cases, you know.

MR. MAUPIN: In LOCA, is that what you mean?

CHAIRMAN BANERJEE: Yes. I don't know in your case, but I know of cases where it does, okay, for LOCAs. But what does this do to you in terms of margins? I mean, does it allow you to sort of make less conservative trip limits by a certain fairly significant amount?

MR. MAUPIN: I do understand the question.

CHAIRMAN BANERJEE: Yes.

MR. MAUPIN: I don't know that I know the answer.

MS. SLOAN: Yes, I don't think we know how to answer that.

DR. WITTER: Right, because the concept of margin kind of disappears with --

CHAIRMAN BANERJEE: I realize it disappears.

(Laughter.)

I mean it is a nice way to -- you know,

you are saying 95/95, but, then, you may ask, why not 99/99 or something, you know. It may be that it is because the staff allows 95/95.

DR. WITTER: Right.

CHAIRMAN BANERJEE: Right? That's the answer probably.

(Laughter.)

Well, let me think about this. But I think I have got the picture. So, it does allow you to be less conservative in some ways than a very prescriptive sort of way of doing this?

DR. WITTER: Right, and it allows, the conservatism gets applied into the setpoint itself with the accommodation for all the measurement accuracies or inaccuracies and, then, other accommodations that need to be put in there for even just the algorithm uncertainties and those sorts of things.

CHAIRMAN BANERJEE: Sure. You put in all your uncertainties that you have to.

DR. WITTER: Right. Almost all that stuff now appears into the setpoint itself.

CHAIRMAN BANERJEE: Yes.

MS. SLOAN: And, Jonathan, just to be clear, it is not the first time a statistical DNB

methodology has been applied to a power plant in the U.S. That is not unique --

DR. WITTER: No. Right.

MS. SLOAN: -- in and of itself.

CHAIRMAN BANERJEE: Do you know how many other cases there have been or how widespread this practice is?

MS. SLOAN: We have an approved methodology. I am going to look over at Jerry. We have a statistical DNB methodology generically for operating plants?

MR. HOLM: Yes, this is Jerry Holm.

I think certainly all the Westinghouse and Combustion Engineering design plants we support we use a statistical DNBR methodology.

CHAIRMAN BANERJEE: For your fuel reloads and things?

MR. HOLM: Right. Yes. I am not really familiar with the B&W plants. They may also use it. Marty says, yes, for the B&W plants, we use the statistical DNB.

Now the U.S. EPR perhaps is more extensive. We treat more parameters statistically. But, certainly, on the other plants we are using a statistical DNB methodology where we are combining

quite a few uncertainties, but I think we have extended it for the U.S. EPR to a more extensive with this ANP-10287.

CHAIRMAN BANERJEE: Okay. Well, we use it for LOCAs. We can use it here. Why not? It is the same idea.

(Laughter.)

Okay.

DR. WITTER: Okay. Then, the last two slides are really just meant to provide a suite of the events that were considered and made use of the in-core SPND-based reactor trips. And in this case, I have provided the example plots for the 15.1.3 for the increased steam flow and, then, the 15.4.2 for the uncontrolled bank withdrawal events. But we also considered the trips were also included in for the decrease in feedwater flow, the increase in feed flow, for the inadvertent opening of a steam generator relief valves, for other control misoperations, and, then, for the inadvertent opening of pressurizers. So, increase in steam demands, cooldowns, and things like that. All of those events demonstrated the adequacy of the reactor trips.

For other events, they may have also included the in-core trips, but also may have either

been too fast for the in-core trip system to be picked up, so, therefore, the limiting condition for operation thresholds were used, or the events may have been part of that event suite that did not call for an in-core trip because they were benign enough, or they would have tripped on something else earlier than called for for the DNBR.

So, for instance, the increase in steam flow, the high steam generator pressure drop trip, if it was a large enough increase in steam flow, the pressure drop was quick enough to come in to trip the plant before the DNBR trip would have been required.

For the loss for forced reactor coolant flow, that is the DNBR event that set the LCO. And so, that set the initial LCO, the 2.5 level, to ensure adequate protection, because that event happens too quickly for the DNBR in-core system to pick that up.

The loss of one pump, like a locked rotor or a broken shaft, is bounded by the loss of the full RCP flow.

For the spectrum of postulated rod ejection accidents, we will talk about that tomorrow, but that is a postulated accident and it is covered by the methodology report in 10286P, which we will talk about tomorrow and the results for that event tomorrow

as well. In that case, you are allowed to have fuel failures, but they have to stay less than the limit prescribed by the radiological release.

And I think that is the last slide for my part of the presentation.

CHAIRMAN BANERJEE: Well, I think what we should do probably is take a break now and come back a little bit early. So, we were supposed to be back at 10:40. So, instead of that, let's come back at 10:35.

I think you have got enough time to cover your 59 slides by 2:15 today. So, if needed, we can always shrink lunch a little.

So, with that, we are just going to take a break.

We are off the record.

(Whereupon, the foregoing matter went off the record at 10:20 a.m. and resumed at 10:38 a.m.)

CHAIRMAN BANERJEE: We are going to go back in session.

So, let's hand this back to Sandra. Take it from here.

MS. SLOAN: Okay. I am going to turn it over to Mr. Brownson to talk about the event evaluation in Chapter 15.

MR. BROWNSON: Okay. We just finished the

discussion of the background information. We are now going to proceed with the discussion of the Chapter 15 results.

We will begin with a presentation of the safety analysis results, followed by the discussion of the radiological results by Mr. Perez.

The U.S. EPR non-LOCA methodology is conservative and deterministic methodology, as presented in the codes and methods applicability report for the U.S. EPR, that is ANP-10263PA.

The methodology follows the guidance and requirements as outlined in NUREG-0800, Standard Review Plan, and Regulatory Guide 1.206.

The non-LOCA methodology has been updated for the U.S. EPR to obtain initial fuel conditions from the COPERNIC computer code rather than RODEX2A. One of the advantages of this is that COPERNIC allows for the generation of burnup-dependent fuel characteristics.

The non-LOCA methodology utilizes S-RELAP5 to provide system fluid boundary conditions for external DNBR, fuel centerline melt, and radiological evaluations.

In accordance with regulatory guidance, the U.S. EPR safety analyses do not model non-safety-

related functions unless their operation is detrimental to the event progression. Key parameters, such as protection system setpoints, are biased for conservatism and uncertainty in accordance with Reg. Guide 1.105.

Protection system trip and safety function setpoints are biased in the most challenging direction for an event.

As part of the methodology, the most limiting single failure is accounted for as well as sensitivities performed to consider the effect of loss of offsite power. Operator actions are assumed not to be taken until 30 minutes after event initiation. Limiting radial and axial power distributions are evaluated as part of the fuel performance evaluations. Additionally, when applicable, preventative maintenance of key equipment is assumed in order to challenge plant performance.

Events are evaluated for all operating modes and power ranges from hot zero power to hot full power. Reactivity coefficients are determined at beginning of cycle, most reactive conditions, and end of cycle.

I would like to do a quick review of the Chapter 15 contents. During my presentation, I do not

plan to go into detail on the individual events of each category. I will, however, be presenting a summary of those events highlighted. These represent those events that are considered challenging to the event category acceptance criteria.

15.0.3 is the radiologic consequences. Pedro will be discussing the results of that section after I finish.

15.1, increase in heat removal by a secondary system. We will be talking about the increase in steam flow and the steam system piping failure, decrease in heat removal, loss of external load. Actually, we are not discussing that one. We are discussing turbine trip and inadvertent main steam isolation valve closure and feedwater line break inside and outside containment.

Decrease in RCS flow, we are discussing, the complete loss of forced reactor coolant flow, and reactor coolant pump rotor seizure.

MEMBER STETKAR: Doug?

MR. BROWNSON: Yes?

MEMBER STETKAR: If we have questions about some of the events that you are not going to explicitly highlight, when is the best time --

MR. BROWNSON: Probably during when we are

discussing that event category.

MEMBER STETKAR: With that category? Okay, got it. Thanks.

MR. BROWNSON: Reactivity and power distribution anomalies, we are going to have a summary of the control rod misoperation, rod drop, and the rod ejection analysis.

15.5, increase in RCS inventory, we will discuss both of these events, inadvertent operation of ECCS or EBS, and chemical and volume control system malfunction.

And decrease in RCS inventory, we will discuss the inadvertent opening of the PSRV and steam generator tube rupture. 15.65 will be the topic of a later ACRS meeting.

And, then, we will finish up with a discussion of the anticipated transient without scram.

CHAIRMAN BANERJEE: So, when is this loss-of-coolant meeting going to be, Derek?

MR. WIDMAYER: We are talking about the last week of March.

CHAIRMAN BANERJEE: Okay. That is fixed now?

MS. SLOAN: No, that is the methodology for realistic LOCA. That is the methodology only. To

my knowledge, it is not the LOCA part of the FSAR --

MR. WIDMAYER: Is that right?

MS. SLOAN: -- 15.65.

MR. COLACCINO: This is Joe Colaccino of Projects.

What you have in front of us, Dr. Banerjee, is the first part of Chapter 15 with the exception of 15.65. So, the staff is working towards getting that safety evaluation with open items probably sometime in the late March timeframe.

And I was just talking about that with Joe Donoghue just a little bit ago because I knew that that's where you are heading with some of your answers.

So, probably when we would be presenting that information to the ACRS would probably be in a May slot, if we have that slot available to us. That is just my guess right now, looking at that. We are evaluating that.

Also, recognize, too, that it is 15.65 without the portion on downstream effects.

CHAIRMAN BANERJEE: Okay.

MR. COLACCINO: I hope that helps you.

CHAIRMAN BANERJEE: Yes. Okay. Go ahead.

MR. BROWNSON: Okay. Moving into a

description of the increase in heat removal, the increase in steam flow event includes a spectrum of cases covering the ranges of all six turbine bypass valve openings. Each valve represents 10 percent of full steam load capacity, and that is not per loop; that is total steam load.

As a bypass valve is opened, the turbine control valve opens wider in order to maintain full steam flow to the turbine to generate 100 percent power. The resulting increase in steam flow causes heat transfer in the steam generators to increase and, consequently, reduces primary system temperatures.

Because of the common header prior to the turbine control valves, the resulting steam increase and overcooling effects is realized in all four steam generators. The reduction in primary system fluid temperatures may cause an increase in reactor power due to reactivity feedbacks.

The event is terminated due to reactor trips on load DNBR, high linear power density, or if the steam flow is great enough on high steam generator pressure drop.

For all scenarios evaluated, all acceptance criteria are met.

Moving on to the steam line break event --

MEMBER STETKAR: Doug?

MR. BROWNSON: Yes?

MEMBER STETKAR: Now I am going to start asking you questions.

MR. BROWNSON: Okay.

MEMBER STETKAR: On the increase in steam flow event, what happens if an MSIV fails to close?

MR. BROWNSON: The MSIVs do not close during this event.

MEMBER STETKAR: The flow is not large enough to get you the MSIV closure signal?

MR. BROWNSON: It will if I think all six valves pop open and you get a reactor trip on high pressure drop, steam generator high pressure drop. The MSIV will close.

MEMBER STETKAR: That would certainly close the MSIV.

MR. BROWNSON: Right.

MEMBER STETKAR: That is the MSIV closure signal.

MR. BROWNSON: Right.

MEMBER STETKAR: So, my question is, if this spectrum includes everything up to -- is it up to and including or up to --

MR. BROWNSON: Up to and including, all

six valves.

MEMBER STETKAR: So, my question is, if it is up to and including that, and you would get an MSIV closure signal, what are the consequences if an MSIV fails to close and why hasn't that been analyzed?

MR. BROWNSON: It would have been analyzed as part of the six valve closures. At that point, you would end up with all four of your steam generators bottled up and your pressure would increase due to the heat transfer. You would get the reactor trip on the high pressure drop as well as --

MEMBER STETKAR: I guess what I am asking you is, do you have an analysis that shows that transient with an MSIV failure? Have you analyzed that event?

MR. BROWNSON: Paul Bergeron.

MR. BERGERON: I think there may be a slight --

MS. SLOAN: Paul, can you introduce yourself?

MR. BERGERON: My name is Paul Bergeron, representing AREVA.

And I think I heard you say that you shifted to the increase in feed flow event, correct?

MEMBER STETKAR: No, no. No, this is

increase in steam flow. If, indeed, the spectrum of increase in steam flow goes all the way up to and including all six --

MR. BERGERON: Yes.

MEMBER STETKAR: -- turbine bypass valves opening --

MR. BERGERON: Right.

MEMBER STETKAR: -- which is 60 percent full flow, my question is, have you done an analysis of that limiting event with a single failure of the MSIV to close?

MR. BERGERON: Correct. Yes, we have.

MEMBER STETKAR: You have?

MR. BERGERON: Yes. The spectrum includes all the way up to the -- and if the MSIVs would have --

MEMBER STETKAR: Okay. It wasn't clear to me when I was reading that analysis because it says there is no single failure which will make this event more severe. And it seems like the cooldown, if an MSIV fails to close, would be --

MR. BERGERON: Yes, I understand.

MEMBER STETKAR: -- reasonably substantial, the post-trip cooldown.

MR. BERGERON: Yes. I mean, basically,

this event takes you up to the six turbine valves. If you haven't challenged the MSIVs, the steam line break would be the event where the MSIVs would be challenged. And that spectrum just extends this one for the turbine valves.

MEMBER STETKAR: Except that the steam line break is presumed upstream from the MSIVs. So, you need two MSIVs to fail to get you an overcooling; whereas, here, since this is downstream --

MR. BERGERON: We actually look at steam line breaks on both sides --

MEMBER STETKAR: Okay.

MR. BERGERON: -- of the MSIV.

MEMBER STETKAR: That also wasn't clear from what was in the FSAR.

MR. BERGERON: Okay. Yes. No, we actually look at different locations for the steam line break --

MEMBER STETKAR: Okay.

MR. BERGERON: -- which includes upstream and downstream --

MEMBER STETKAR: Okay. Okay.

MR. BERGERON: -- of the MSIV.

MEMBER STETKAR: Okay.

MR. BERGERON: And the single failure of

the MSIV would be accounted for.

MEMBER STETKAR: And that spectrum is included. Okay.

One of the reasons is I had some problems as I read through the FSAR. There was a discussion in Section 15.0.0.3.8 about what limiting single failures are possible and which are not possible. The MSIVs were not listed in that section as not possible. So, I assumed that they were a possible single failure.

MR. BERGERON: Correct.

MEMBER STETKAR: And, then, I started to think about whether they were included, in which analysis. So, if I have assurance that you have included both the upstream and downstream steam line breaks with the MSIV failure, then I am pretty happy about that.

MR. BROWNSON: The information listed in the FSAR is the single most limiting failure for the event presented in the FSAR, not necessarily the whole range of all failures.

MEMBER STETKAR: Well, that is why -- I am looking at Table 15.0-11, Single Failures Assumed in the Accident Analysis, and for increase in steam flow it says one protection division. Now one protection division will not fail a single MSIV.

MR. BERGERON: Correct.

MR. BROWNSON: Right.

MEMBER STETKAR: And, then, it says, "There is no single failure which will make this event more severe." A failure of a protection system division is inconsequential. The protection system is single failure-proof due to its redundancy.

So, I was led to believe from what is in the FSAR that there wasn't any single failure that could make anything in this full range of these steam flow increase events more severe. And it seems to me like a MSIV failure would.

MR. BERGERON: But the single failure listed in that table is the limiting single failure for that event relative to the criteria of that event. For example, we are looking at -- this is an AOO -- so, we are looking at DNB and linear power density.

So, for that event, and it turns out, I think, for the excess steam demand or the increase in steam flow, it is a 50 percent flow case, not the full case, because in the full case you get a trip right away with the steam generator pressure drop feature.

MEMBER STETKAR: Okay. Okay. Thanks.

MR. BROWNSON: Okay. Moving on to the steam line breaks, we evaluated both pre-reactor trip

and post-reactor trip scenarios. For both scenarios, the spectrum of break sizes were considered, both in containment and outside containment, upstream, downstream of the MSIV, depending on break size and number of protection system functions that activate a reactor trip.

Following the initiation of the steam line break, the four steam generators depressurized through the break due to the steam line common header. However, the individual steam lines are soon isolated due to steam generator high pressure drop, which also initiates a reactor trip.

The limiting single failure for this event is the failure of the MSRCV to close in one of the unaffected steam lines following steam line isolation.

MEMBER STETKAR: Doug, if you look at a steam line break inside containment, I understand you took the steam line break outside because that delays the reactor trip due to the adverse environmental conditions for the steam pressure transmitters. For a steam line break inside the containment, since all steam generators will continue to feed that break through the crossconnect header --

MR. BROWNSON: Okay.

MEMBER STETKAR: I am not a thermal

hydraulics guy.

MR. BROWNSON: Okay.

MEMBER STETKAR: Let me preface that.

MR. BROWNSON: Okay.

MEMBER STETKAR: Pumps and pipes and valves are about all I can do.

But if you have a steam line break inside the containment, all four steam, you know, you will feed, the broken steam generator will feed that break, and the other three will feed it through the crossconnect --

MR. BROWNSON: Up to the point of MSIV closure.

MEMBER STETKAR: Now the question is, is there a substantial delay of the MSIV closure signal due to the location of the pressure transmitters outside the containment? In other words, that line is going to hang up a little bit --

MR. BROWNSON: Right.

MEMBER STETKAR: -- because the other steam generators are still heating it.

MR. BROWNSON: Right.

MEMBER STETKAR: Is the delay for the steam line break inside the containment longer or shorter than the delay that you took for the break

outside with the adverse environmental conditions?

MR. BROWNSON: Depending upon the break, of course, this is going to depend upon the break size and how quickly you are depressurizing the system.

MEMBER STETKAR: Yes.

MR. BROWNSON: I think if it is a large enough break to catch us on -- if it is a large enough break to catch up steam line pressure drop, I don't think -- the location of the break is not going to matter. The pressure drop is going to be rapid enough that it doesn't matter what that distance from that pressure sensor is going to be.

And if the location of the break is near the pressure sensor --

MEMBER STETKAR: Yes.

MR. BROWNSON: That's going to be another question?

(Laughter.)

MEMBER STETKAR: No. I'm sorry. I am not going to fine-tune it that closely.

MR. BROWNSON: Okay. But, yes, if the break is small enough that you are not going to be caught on pressure delta P, one of our other protection system functions would catch us, if it was slow enough, if that pressure decrease was slow enough

that our pressure, steam generator pressure, dropped and the protection system trip did not catch us, more than likely, another function did step in.

MEMBER STETKAR: Well, it would be something like probably steam generator low level.

MR. BROWNSON: Right.

MEMBER STETKAR: But that has also got to be delayed because those sensors are inside the containment, and it will take a while to blow down that steam generator --

MR. BROWNSON: Correct. Correct.

MEMBER STETKAR: -- to a low-level setpoint. So, I was just curious about that relative timing because the accident analysis in Chapter 15 emphasizes that point, about how carefully you selected the break location for the adverse environmental effects for the --

MR. BROWNSON: Right. The hostile environment, yes.

MEMBER STETKAR: -- hostile environment --

MR. BROWNSON: Right.

MEMBER STETKAR: -- for the location of those particular pressure sensors. And I was trying to understand whether, indeed, that is actually the most bounding break location from an event timing

perspective, because it all deals with the delay in the protection system signal.

MR. BROWNSON: Paul, do you have anything to add to clarify that?

MR. BERGERON: No. I think we looked at several locations and accounted for the fact of where the pressure sensors were using the harsh uncertainties.

MEMBER STETKAR: Yes. The question is, you know, a break inside the containment --

MR. BERGERON: Right.

MEMBER STETKAR: -- you don't have to penalize the pressure sensors for the harsh environment.

MR. BERGERON: Right.

MEMBER STETKAR: On the other hand, could there be other things that inserted an additional delay in the signal that is more than compensated by the harsh environment?

MR. BERGERON: I think for the steam system piping failures that the break is big enough that you get a relatively early trip, and that the isolation of the MSIVs that would isolate the --

MEMBER STETKAR: The other three.

MR. BERGERON: -- the other three, it is

not that far behind.

MEMBER STETKAR: Yes. Okay. I mean that is the relative timing that I am talking about.

MR. BERGERON: Right.

MEMBER STETKAR: Yes.

MR. BERGERON: Right. Yes, and we did a spectrum of breaks, so that that spectrum I am sure caught --

MEMBER STETKAR: Spectrum inside and outside?

MR. BERGERON: Yes.

MEMBER STETKAR: Okay.

MR. BROWNSON: Following the initiation of the break, four generators depressurize through the break, common header. Individual steam lines are soon isolated due to steam generator high pressure drop, which also initiates a reactor trip. The limiting single failure for this event is failure of the MSRV to close in one of the unaffected steam generators.

Failure of this control valve in one of the unaffected generators further aggravates the RCS cooldown until the pressure in that steam line drops below the MSIV's closure setpoint.

The post-reactor trip is the most limiting of the two scenarios predicting a return to power for

the hot zero power case of approximately 23 percent rated thermal power. Fuel damage due to cladding strain is predicted to occur, but radiological releases are within the acceptance criteria.

Moving on to the decrease in heat removal from the secondary side, the two events that most challenge the primary and secondary pressure boundaries are the turbine trip and MSIV closure, single MSIV closure. The turbine trip event is the most limiting RCS overpressure event, and the single MSIV closure event is the most limiting secondary overpressure event.

Both events have a most limiting single failure of an MSRT to open. The peak RCS pressure calculated for the turbine trip event is 2785 psia, which is below the acceptance criteria. The peak secondary pressure calculated for the single MSIV closure event is 1567 psia, which, again, is below the acceptance criteria.

In addition to the failure of the MSRT, the single MSIV closure event also assumes a preventative maintenance on the low setpoint MSSV on the steam line of the MSIV closure. This results in a greater challenge to the secondary pressure acceptance criteria.

Accounting for a similar preventative maintenance during the turbine trip evaluation was not necessary, as the secondary pressure is shared between all four steam generator lines, due to the steam line common header, as the MSIVs do not close for the turbine trip event.

Feedwater line break is a postulated accident that helps establish the adequacy of the emergency feedwater capacity. A spectrum of feedwater line breaks and sizes and locations is evaluated. Both peak primary and secondary pressure scenarios are assessed.

For the peak primary system pressure scenario, a single failure of an EFW pump train and preventative maintenance of a second EFW are assumed on two of the unaffected steam generators.

For the peak secondary pressure scenario, a single failure of an MSRT to open is taken on one of the unaffected steam lines.

And on this same line, preventative maintenance of a low setpoint MSIV is assumed. A second preventative maintenance of an EFW pump train is assumed on a second unaffected steam line.

Operator actions are credited at 30 minutes to realign the EFW from the affected steam

generator to one of the steam generators with an inoperable EFW. The operator also is credited with tripping the RCPs to the two loops without EFW.

Again, all acceptance criteria are satisfied for this event.

MEMBER STETKAR: Doug, I am going to stray a bit here, but it is the only place I can ask this.

In the feedwater line break event and the two others that I can't find my notes, but turbine trip, closure of an MSIV, and feedwater pipe break event, one of the limiting single failures was failure of an MSRT to open.

I will go back to the section of the FSAR that I cited early, 15.0.0.3.8. The MSRIVs, the main steam relief isolation valves, are explicitly highlighted in that section as not being considered as single failures in --

MR. BROWNSON: Right.

MEMBER STETKAR: -- the accident analyses because the valve is designed to be single failure-proof.

MR. BROWNSON: Correct.

MEMBER STETKAR: But here it sounds like they were considered as single failures.

MR. BROWNSON: No. From the initial

presentation, the MSRTs consist of two valves in series.

MEMBER STETKAR: Yes.

MR. BROWNSON: It is the MSRCV, the control valve, and the isolation valve. It is the control valve that is assumed to fail.

MEMBER STETKAR: Fail to open?

MR. BROWNSON: Fail to open or fail to --

MEMBER STETKAR: It is normally open. So, it doesn't --

MR. BROWNSON: It fails closed, yes.

MEMBER STETKAR: Oh.

MR. BROWNSON: First, open and then goes closed.

MEMBER STETKAR: But an MSIV can fail to close. That is a single failure.

MR. BROWNSON: Yes, it could stick open, yes.

MEMBER STETKAR: But an MSRIV cannot fail to open? Are both of those statements true? That none of the transient or accident analyses account for failure of an MSRIV isolation valve failure to open? Is that true? Paul?

MR. BERGERON: The MSIV or the MSR, the IV --

MEMBER STETKAR: Main steam relief isolation valve.

(Laughter.)

MR. BERGERON: Yes.

MEMBER STETKAR: The relief isolation valve. We will get away from the acronym.

MR. BERGERON: The added single failure --

MEMBER STETKAR: And the --

MR. BERGERON: Because it is in series with another valve that potentially could fail closed, we have taken the conservative assumption that that relief train would not be available and it would be a single failure of the CV. Because, basically, what happens, the IV would open, and that is single-proof to an overpressure event.

MEMBER STETKAR: Yes.

MR. BERGERON: That would open, and the CV would control the pressure.

MEMBER STETKAR: Yes, normally.

MR. BERGERON: Normally.

MEMBER STETKAR: Right.

MR. BERGERON: We are postulating that that CV, since it is not single failure-proof --

MEMBER STETKAR: Goes closed.

MR. BERGERON: -- is driven shut.

MEMBER STETKAR: Okay.

MR. BERGERON: Now it is a 40-second valve, so that we could probably credit it before it goes fully shut.

MEMBER STETKAR: But you don't?

MR. BERGERON: But we have chosen not to --

MEMBER STETKAR: Okay.

MR. BERGERON: -- in the safety analysis.

MEMBER STETKAR: I understand that then. Is it that carried through for things like small LOCAs?

MR. BERGERON: You mean as far as the IV?

MEMBER STETKAR: The way you explained it, I don't care whether it is the IV failing to open --

MR. BERGERON: Right.

MEMBER STETKAR: -- or the CV closing --

MR. BERGERON: Right.

MEMBER STETKAR: -- because, either way, you don't get steam relief through the control valve.

MR. BERGERON: Yes.

MEMBER STETKAR: Is that failure considered through the full spectrum of --

MR. BERGERON: Of all the events --

MEMBER STETKAR: -- of all the events,

including small LOCA, is what I am particularly interested in?

MR. BERGERON: Yes, I guess Liliane has probably got the answer to that.

MEMBER STETKAR: I know we are not talking about small LOCAs in this particular meeting, but I was trying to understand that statement in the FSAR about what is the scope of presumed single failures within the context of the transients that we are looking at now, and seeing how that translates through all the rest of the analyses. So, that is why I am asking.

Go ahead. You have to speak right into the microphone.

MS. SCHOR: My name is Liliane Schor. I work for AREVA.

For small break LOCA, we did a single failure assuming the MSRCV starts to operate and then failed closed. But the single failure for small break LOCA was EDG failures being more limiting.

MEMBER STETKAR: Does an EDG failure -- okay. Okay. But you did look at the CV --

MS. SCHOR: We did look at the MSRCV closed.

MEMBER STETKAR: Yes. No steam release

through one of the four trains, basically?

MS. SCHOR: Correct. We did.

MEMBER STETKAR: Okay. Thank you.

All right.

MR. BROWNSON: Moving on to the decrease in RCS flow rate events, the complete loss-of-flow event may result from a simultaneous failure of an electrical power supply of the RCPs. The resulting decreasing reactor coolant flow degrades the core heat transfer and reduces DNBR margin.

The protection system logic initiates a reactor trip when two out of four loops generate a low RCP trip signal. By assuming one protection system division has failed and another is removed for preventative maintenance, the reactor will only trip when the low RCP trip signal is reached in all four coolant loops.

This is the limiting delta DNBR event that does not trip on DNB. It establishes the limiting condition for operation on initial DNB.

The RCP rotor seizure event is postulated to be an instantaneous seizure of an RCP rotor. The sudden decrease in reactor coolant flow while the reactor is at power degrades core heat transfer and can lead to fuel damage. A reactor trip signal is

generated when the low-low flow setpoint is reached. Eight percent of the fuel rods are conservatively predicted to undergo DNB-related fuel damage, resulting in radiological doses below the acceptance criteria, which were established assuming a 9.5 percent fuel failure rate.

Reactor and power distribution anomalies. Both single and complete bank RCCA drop events have been analyzed. Rod drop events are initiated by de-energizing an RCCA drive mechanism or by a malfunction associated with a control bank during power operation.

Following a rod drop, there is initial power decrease followed by a power increase due to reactivity feedback and the automatic withdrawal of control banks when the average coolant temperature control function is active.

The DNB limiting condition for operation and the low DNBR reactor trip provide protection against DNB-induced fuel failures. High linear power density limits are not exceeded, resulting in fuel centerline temperatures not exceeding the melting point and fuel clad strain not exceeding 1 percent.

The RCCA ejection accident, which will be discussed in greater detail during tomorrow's meeting, is defined as a postulated rupture of the control rod

drive mechanism housing that results in the complete ejection of an RCCA from the reactor core. The power spike resulting from the rod ejection is quickly countered by reactivity feedbacks when the fuel temperature begins to increase, and may be terminated by reactor trip on high neutron flux, high neutron flux rate, or low neutron flux doubling time.

For these scenarios, for the core evaluation scenario, a spectrum of ejected rod worths and initial power levels from hot zero power to hot full power were performed. All scenarios resulted in satisfaction of acceptance criteria.

For the RCS overpressure scenario, no opening in the RCCS is assumed as the ejected rod worth plugs is assumed to plug the rupture in the housing mechanism. The spectrum of power levels and rod worths were again evaluated. RCS pressure is maintained well below the 120 percent acceptance criteria, and the secondary MSSVs never actuated for this event.

CHAIRMAN BANERJEE: So, the cases where opening is assumed, how big are these openings?

DR. WITTER: This is Jonathan Witter.

The flange housing size diameter was the assumed flow hole leakage size for the

depressurization.

CHAIRMAN BANERJEE: Which is how big a hole?

DR. WITTER: I can let you know in a little bit.

MR. BROWNSON: We will take an action --

MS. SLOAN: Right. We will look it up.

MR. BROWNSON: -- to find out that size.

CHAIRMAN BANERJEE: Go ahead.

MEMBER STETKAR: Doug, I'm sorry, I was writing notes here. Can you go back to the previous slide? Thanks.

The complete loss of forced reactor coolant flow, somewhere -- and I was trying to find my notes and I couldn't find it -- one of the accidents or transients that is evaluated is a loss of non-emergency AC power to station auxiliaries. Or, you know, it is an euphemism for a loss of offsite power.

The analysis says that that event is bounded. There is apparently not a separate analysis done for that event because it is supposedly bounded by a complete loss of forced reactor coolant flow. Obviously, the reactor coolant pumps would not operate under that condition.

However, the single failure for the

complete loss of forced reactor coolant flow is one protection system division with the second one being out of service for maintenance, which is effectively a non-issue.

MR. BERGERON: Right.

MEMBER STETKAR: However, if you are using that accident to also include losses of offsite power, don't we have to care about diesel failures or not? Maybe not for DNBR. In other words, I am worried about using this thing as a surrogate for loss of offsite power and then saying, well, the most limiting failure for this thing is a protection system division. Am I losing an analysis --

MR. BERGERON: Right.

MEMBER STETKAR: -- for loss of offsite power --

MR. BERGERON: Right.

MEMBER STETKAR: -- with a single diesel failure, for example?

MR. BERGERON: Right. I think the loss of AC case, the FSAR says, for the beginning of the event, the loss of flow is --

MEMBER STETKAR: Is limited.

MR. BERGERON: -- is comparable.

MEMBER STETKAR: Yes. Okay.

MR. BERGERON: For the long-term event, it is the loss of normal feed. So, the loss of normal feed would also consider that as a potential single failure. And I think it turned out that worst single failure for that is an EFW train.

MEMBER STETKAR: EFW division, which would be picked up with --

MR. BERGERON: Right. Yes. So, the long-term aspects of the loss of AC would be the loss of normal feed, which we have analyzed specifically, looking with and without pumps. And, of course, the loss of normal feed is more limiting with the RCPs.

MEMBER STETKAR: Yes. Yes. Yes.

MR. BERGERON: So, that is how we cover --

MEMBER STETKAR: Okay. Yes. Okay.

MR. BERGERON: That's how we cover the long-term aspects of the loss of AC.

MEMBER STETKAR: Okay. Okay. Thanks. Thanks.

MS. SLOAN: Jonathan has an answer.

MEMBER STETKAR: Jonathan?

DR. WITTER: This is Jonathan Witter again.

For the flange area size, it is .246-foot diameter. So, about a 3-inch diameter hole. That is

the maximum hole size, if the drive mechanism were completely removed from the flange housing.

CHAIRMAN BANERJEE: That is what the pipe diameter is?

DR. WITTER: Right. Right.

CHAIRMAN BANERJEE: Okay. And that, you will talk about tomorrow, right?

DR. WITTER: Yes. Yes.

CHAIRMAN BANERJEE: Does that give you as big a power pulse in the sense that you probably get some voiding, right, in the region of the rod drop? I mean rod ejection.

DR. WITTER: In actuality, we don't consider the depressurization in the rod worth or the power pulse part of it. The depressurization part plays a role in the DNBR calculation, in the failure sense, for the post-ejection period of the --

CHAIRMAN BANERJEE: But you don't look at the void formation in that region?

DR. WITTER: Only in the sense of however our neutronics calculations would handle that part of it.

CHAIRMAN BANERJEE: Yes, of course. But, for the neutronics, you have got to feed it some of the hydraulics, right?

DR. WITTER: Exactly. Right. The event where the rod is ejected is .1 seconds. So, the impact of the depressurization part of that would happen much later than the pulse part of it. So, it really doesn't play much of a role in the initial power pulse part of it, except for the thermal hydraulics part of the feedback with the moderator feedbacks for it.

So, we will talk more tomorrow about the details of that.

CHAIRMAN BANERJEE: Yes. We will need to explore that, how you are doing it.

DR. WITTER: Okay. Yes.

CHAIRMAN BANERJEE: Fine. Okay.

MR. BROWNSON: Moving on to the increase in RCS inventory events, the inadvertent operation of the ECCS or EBS, of these two cases, the ECCS operation is inconsequential, as the MHSI's shutoff head is well below the operating pressure of the U.S. EPR.

For the EBS, the EBS system is manually-actuated and is designed to inject highly-borated liquid into the RCS against RCS pressure following a design basis event. The system consists of two trains, each with high-pressure, positive displacement

pumps and an EBS tank.

It is postulated that both pumps are started with the corresponding isolation valves simultaneously opening to increase RCS inventory.

CHAIRMAN BANERJEE: So, do you have a larger sort of a feed-and-bleed-type system, as you don't have a high-pressure injection system?

MR. BROWNSON: Well, we have the capability of performing feed and bleed through the PSRVs and the MHSI system or the EBS. For the MHSI, we would have to get the pressure of the system down to the shutoff head, which is 1405, in order to perform feed and bleed.

CHAIRMAN BANERJEE: And are these systems larger than would be the equivalent flows in and out? I'm thinking back to the old Candu days. They had, more or less, the same issues as you have. They didn't have a high-pressure injection system, right? And they did a controlled cooldown. That is what they do. They actually divert the steam into their condensers. There are some water hammer problems, but that is a separate issue we will talk about.

But now they had a much oversized feed-and-bleed system to be able to handle some of it. But maybe you can answer it.

MR. PARECE: This is Marty Parece again, for the stenographer.

The main steam safety valves are both media-operated and the operator can open them as well, the power operator.

MEMBER STETKAR: The pressurizer --

MR. PARECE: The pressurizer safety valves.

MEMBER STETKAR: The pressurizer safety valves.

MR. PARECE: I'm sorry, the pressurizer safety valves.

The pressurizer safety valves are both media- and manually-operated. They are 660,000 pounds per hour for the three valves. If you used all three, you could depressurize the plant below 200 psi in less than 20 minutes because our severe accident valve, one of those is the equivalent of three safety valves. And that is what that analysis shows.

So, yes, we can depressurize the plant to do feed and bleed, and that is one of the credited backup cooling methods in the PRA.

CHAIRMAN BANERJEE: That makes sense.

MR. PARECE: So, they are very big. A comparison is a current three-loop Westinghouse plant

has safety valves that are 420,000 pounds per hour. And those are spring-loaded. And their PORVs are only about 120,000 pounds per hour.

So, we can use these big valves also manually.

CHAIRMAN BANERJEE: Okay.

MR. BROWNSON: Again, it is postulated that both EBS pumps start, and corresponding isolation valves simultaneously opening to increase RCS inventory. The pressurizer level increases until reactor trip occurs. Upon reactor trip, turbine trip occurs and simultaneous loss of offsite power. Loss of offsite power results in RCP coastdown and loss of main feedwater.

There is a single failure of one MSRT to open, resulting in decreased heat transfer from the secondary system and greater heatup and swell to the primary system.

The event is terminated at 30 minutes with operator isolation of the EVS.

Again, all acceptance criteria are satisfied for this event.

The CVCS malfunction event is a malfunction of the pressurizer level control system. The CVCS maintains a constant charging flow and

adjusts the letdown flow to account for volume changes due to RCS temperature variations.

Normally, only one of the two CVS charging pumps are in operation. For this event, we assumed both pumps operate with the letdown system isolated.

This event is terminated by a reactor trip on high pressurizer level. A single failure of an MSRT again is assumed. A single failure of an MSRT to open is again assumed. This, again, results in a decrease in heat transfer, greater heat of the primary system and swell.

Additionally, the non-safety pressurizer heaters and sprays are also assumed to operate as designed. Pressurizer sprays will reduce RCS pressure, delaying actuation of the PSRVs and aggravating system overfill. If actuated, pressurizer heaters will help increase the primary system swell and also aggravate system overfill.

No operator actions are credited in this event to mitigation. However, it is assumed that one of the CVCS pumps are restarted following the loop in order to aggravate the overfill situation.

Again, all the event acceptance criteria are satisfied for this event.

Moving on to the decrease in RCS inventory

events, we will only be discussing two. The two events for discussion today are the inadvertent opening of the PSRV and the steam generator tube rupture.

The inadvertent opening of the pressurizer safety relief valve is defined as a spurious opening of the PSRV that is normally closed. The U.S. EPR has three such valves, as Marty just stated, located at the top of the pressurizer and automatically open to prevent RCS overpressurization. The three valves operate independently. So, a single failure can only affect one PSRV.

The IOPSRV causes a loss of reactor coolant inventory that cannot be offset by the CVCS charging. This causes primary system depressurization and decreases the reactor coolant density. Reactor power is initially controlled through reactivity feedbacks, through the change in moderator density, and the rod position actions of the average coolant temperature control function. The drop in RCS pressure results in a reactor trip and subsequent loss of offsite power.

The limiting single failure for this event is failure of one emergency diesel generator to start which results in the loss of one safety injection

train and EFW train.

A second diesel generator is assumed to be removed due to preventative maintenance resulting in the unavailability of a second SIS and EFW train.

Beginning-of-cycle conditions are predicted to be the most limiting, as this provides for the most positive moderator reactivity feedbacks.

Operator actions are credited 30 minutes into the event to align EFW flow from the two operational EFW trains to the four steam generators. The two operating medium head safety injection trains are adequate to offset RCS inventory loss in the later phase of the event and maintain core cooling throughout the transient.

The event's drop in RCS pressure and initial positive reactivity feedbacks make this event challenging to the DNB acceptance criteria. However, the DNB LCO and low DNBR trip are adequate to protect the fuel, and all acceptance criteria are satisfied.

Moving on to the steam generator tube rupture event, this event is postulated as a double-ended rupture of a single steam generator tube. The main acceptance criteria for this event is to maintain radiological releases below acceptable limits.

A second criteria is to prevent overfill

to the steam generator secondary side to prevent water from entering the steam lines. The steam generator tube rupture is taken to occur in the shortest steam generator tube, near the tube sheet, on the steam generator primary system inlet side. The location maximizes the pressure difference between the primary and secondary system and maximizes the break flow.

The break begins depressurizing the RCS and the level in the pressurizer begins to drop. The non-safety CVCS system flow is credited with operating to maintain pressurizer level and delay reactor trip.

For overfill scenarios, the limiting single failure is the failure of the main feedwater full load control valve, as the steam generator level increases.

For radiological scenarios, the limiting single failure is the failure of the MSRT on the affected steam generator to close.

The limiting overfill and radiological FSAR cases are with loss of offsite power, resulting in the loss of the CVCS, RCPs, and main feedwater pumps. Operator actions, including manual reactor trip, are not taken until 30 minutes after event initiation.

All overfill and radiological acceptance

criteria for this event are satisfied in the FSAR analyses.

CHAIRMAN BANERJEE: Are we going to go into this in more detail at some point?

MS. SLOAN: We will talk about tube rupture mitigation strategy, which is what Ken Coffey will talk about.

CHAIRMAN BANERJEE: Okay.

MS. SLOAN: Do you have a particular question?

CHAIRMAN BANERJEE: No, not right now.

MS. SLOAN: Okay.

MEMBER STETKAR: I do. I'll wait.

CHAIRMAN BANERJEE: Yes, we will wait.

(Laughter.)

MR. COFFEY: If the question pertains more towards the analysis, assumptions made, or systems used in the analysis -- this is Ken Coffey, sorry -- Doug Brownson might be better to answer those. I am going to handle more the mitigation aspects of the events. So, feel free to ask your questions when you feel more appropriate.

MEMBER STETKAR: You are not leaving, are you, Doug?

(Laughter.)

MR. BROWNSON: No.

MEMBER STETKAR: Okay. We'll wait.

MR. BROWNSON: The final section in Chapter 15 that we are going to talk about today is the anticipated transients without scram.

The ATWS occurs when the control rods fail to insert following an anticipated operational occurrence. This occurs as the result of a mechanical blockage of the control rods or an electrical or mechanical failure of the protection system.

A mechanical blockage of the control rods is determined to be an insignificant contributor to the probability of an ATWS, so it is not considered.

In the event of protection system failures, the U.S. EPR is designed with a diverse actuation system, or DAS, that provides for an independent initiation of a reactor trip, turbine trip, and other essential systems and functions required to bring the reactor to a stable condition.

The design of the DAS satisfies the acceptance criteria of 10 CFR 50.62.

MEMBER STETKAR: Doug, I read through the text trying to convince me that mechanical blockage of the control rods is insignificant, but I wasn't convinced.

MR. BROWNSON: Okay.

MEMBER STETKAR: It seemed to be saying that, well, people take 10 to the minus 5s, and 10 to the minus 5s are sort of okay for current operating plants. So, therefore, it is probably less than 10 to the minus 5. So, therefore, it is probably not important.

That doesn't convince me that it is an insignificant contributor to the probability of an ATWS event if everything else, indeed, is much less than that. It could be the most important contributor to an ATWS event.

So, I am not convinced from a probabilistic standpoint. From a deterministic standpoint, I understand what you are doing with ATWS.

MR. BROWNSON: Right.

MEMBER STETKAR: So, this is more a PRA-type question, sort of challenging that assertion in a real -- this is a probabilistic assertion.

MR. BROWNSON: Yes.

MEMBER STETKAR: It is not a deterministic assertion.

MR. BROWNSON: Right.

MEMBER STETKAR: Because your deterministic analysis says the reactor doesn't trip.

I asked that question, by the way, in the PRA. And the PRA does not include mechanical blockage because it is assumed to be an insignificant contributor. We will just leave that as a PRA question.

(Laughter.)

But a more pertinent question for ATWS is I found in Section 15.8.1.4, which discusses the emergency feedwater system, a statement that says, "The U.S. EPR is designed so that flow from the EFW system is not required for the first 30 minutes following an ATWS."

It strikes me as a rather bold statement if the cause for the ATWS is loss of all main feedwater. How can I survive the loss of main feedwater without tripping the reactor for more than 30 minutes, or let's say 30 minutes, with no sources of emergency feedwater?

MR. BROWNSON: Okay. This would have been covered under our D3, was it?

MEMBER STETKAR: The reason I bring it up is --

MR. BERGERON: We have an automatic EFW actuation for D3.

MR. BROWNSON: Okay. So, that was added

probably after --

MR. BERGERON: As part of the diverse actuation system.

MR. BROWNSON: Okay.

MR. BERGERON: I think we also had ATWS included as part of the diverse actuation system automatic EFW.

MEMBER STETKAR: Well, this is just a statement that says it's not required.

MR. BERGERON: Right.

MEMBER STETKAR: That, to me, says it's not required.

MR. PARECE: This is Marty Parece.

That assumes that the DAS operates when the reactor trips. If the reactor trips, then you don't need emergency feedwater for 30 minutes because there is decayed heat removal capability in the steam generators for at least that long.

MEMBER STETKAR: I guess I interpret an ATWS as the reactor did not trip. I understand, if the reactor is tripped, I have enough inventory in the steam generators, but perhaps it is my misinterpretation.

The reason I ask is the staff in the SER says, the staff notes that thermal hydraulic analyses

of an ATWS have not been provided, nor are they required, to numerically demonstrate that in the event of an ATWS, the pressure increase does not exceed ASME service-level speed conditions.

And the SER cites this fact, that you don't need emergency feedwater for 30 minutes during an ATWS event. So, I'm curious about whether we are understanding who is analyzing what, recognizing that an ATWS event is beyond design basis, of course.

MR. PARECE: This is Marty Parece again.

If you follow the daisy chain, it says we don't consider the blockage of all the control rods, physical blockage of all the control rods, to be credible. Then, the DAS, the Diverse Actuation System, operates, and it trips the reactor, and therefore, all the events are mitigated, all the ATWS events are mitigated with the DAS and eventually aux feedwater to provide that coolant.

MEMBER STETKAR: So, presuming that enough rods can always go in somehow --

MR. PARECE: Yes.

MEMBER STETKAR: -- I understand that. Okay. Thanks.

MR. BROWNSON: This concludes my summary of the Chapter 15. If there's no additional

questions, then Mr. Perez will continue on with the radiological results.

MR. PEREZ: Good morning.

My name is Pedro Perez. I have been here before you. Do you want me to do an introduction or can I continue?

CHAIRMAN BANERJEE: Continue.

MR. PEREZ: Thank you.

I will be presenting the radiological consequences of design basis accidents, which are summarized in the FSAR in Section 15.0.3.

Our design basis radiological evaluations are based on the Standard Review Plan, Section 15.0.3, with the additional guidance provided in Section of the SRP 4.2, which provides interim acceptance criteria and guidance for reactivity-initiated events.

Our DBAs were evaluated applying the alternative source term methodology, as documented in Regulatory Guide 1.183. We also took into consideration the Regulatory Issue Summary 2006-04, which was a summary of the experience with implementation of the alternative source term, and the AST radiological acceptance criteria you can find in 10 CFR 50 and 10 CFR 52.

We used regulatory guidance in 1.145 for

the offsite atmospheric dispersion factors, determining those. In Regulatory Guide 1.94, which is ARCON-96, is the control room atmospheric dispersion factors.

The events evaluated are the following.

CHAIRMAN BANERJEE: Sorry. The onsite dispersion factors, that is what happens within your control rooms and everything. You use your own codes there to do the calculation?

MR. PEREZ: No. That code is the NRC's code, ARCON-96.

CHAIRMAN BANERJEE: Okay. Okay. That is an NRC code?

MR. PEREZ: Yes, which is the effluent from the release point to the control room intake, which are very short distances.

CHAIRMAN BANERJEE: And, then, what happens within the control? Don't they have a code called HABIT or something?

MR. PEREZ: That is the code that we do not use, and that is for control room habitability.

CHAIRMAN BANERJEE: Yes.

MR. PEREZ: And for control room habitability, we use our own code, and the code calculates the immersion and the inhalation doses.

CHAIRMAN BANERJEE: Right.

MR. PEREZ: Of course, RADTRAD can also do that calculation, too.

CHAIRMAN BANERJEE: And you used your own code here?

MR. PEREZ: And we also used RADTRAD because we knew the staff was going to use RADTRAD, and we wanted to make sure there were not going to be any differences between codes.

CHAIRMAN BANERJEE: So, HABIT, it is a multi-compartment code, isn't it? It sort of looks at concentration fields within --

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: -- the control rooms and things like that?

MR. PEREZ: Well, the control room envelope becomes one of the compartments.

CHAIRMAN BANERJEE: Yes.

MR. PEREZ: So, you basically release the radioactivity from one compartment to another compartment.

CHAIRMAN BANERJEE: Yes.

MR. PEREZ: If you would, that connection takes into account the dispersion from compartment A to compartment B. So, within either code that you

use, the RADTRAD or our own code, it is basically the same solution.

CHAIRMAN BANERJEE: I guess we will ask the staff whether they did some confirmatory analysis on this when the time comes.

MR. PEREZ: Okay.

CHAIRMAN BANERJEE: All right. Go ahead.

MR. PEREZ: And I know they did do it because we have had RAIs to discuss some differences in what did we assume and the specifics about our analysis.

CHAIRMAN BANERJEE: Okay.

MR. PEREZ: The events have been evaluated following the SRP and following the regulatory guidance in 1.183 as small line breaks outside of the reactor building, steam generator tube rupture, main steam line break outside the reactor building, locked rotor, rod ejection, fuel-handling accident, and the old maximum hypothetical accident, the large-break LOCA, the LOCA.

I will point out that the reactor cooling pump broken shaft and the feedwater line break radiological consequences are bounded by the locked rotor and the main steam line break, respectively, okay, from a radiological perspective.

MEMBER STETKAR: Pedro?

MR. PEREZ: Yes?

MEMBER STETKAR: Let me ask you about the small break outside the reactor building, and shuffle it off to someone else if the question is in a different area.

It is my understanding that you looked at two different lines. One was the CVCS letdown line, which is a 6-inch line, and the other lines you looked at were primary sampling lines, which are quarter-inch lines --

MR. PEREZ: Correct.

MEMBER STETKAR: -- directly connected to the primary system. The conclusion was, I believe, that from a dose perspective, the sampling line, although it is much, much smaller, was the most limiting line.

However, when I read the discussion of the CVCS line, there is a lot of description of the fact that the break would be downstream of the purification loops, which I believe I have lost the quote here, but it sounds like credit may be taken for the filters in the purification loops, removing a substantial amount of the radionuclides and, in particular, iodine from that particular break.

That break requires a manual isolation. So, the letdown line is not isolated automatically. So, therefore, the flow through the break and the release continues for 30 minutes until the operators isolate the line.

My question was, suppose the break in the letdown line occurs upstream from the purification loop, so that the filters and the demineralizers don't remove any of the radionuclides. Would that change the conclusions?

MR. PEREZ: Because it is private, yes.

MEMBER STETKAR: Because that also would require manual isolation.

MR. PEREZ: Right.

MEMBER STETKAR: So, you would have a release from that location for 30 minutes.

MR. PEREZ: Right. And what I need to do is I need to go back and take a look to understand exactly why we chose that location. Because the way you just explained it, you know, of course, that would change the consequences.

MEMBER STETKAR: Yes.

MR. PEREZ: Right now, I don't remember that description. So, in the afternoon I will look and go back to understand how we chose that location

for the CVCC.

MEMBER STETKAR: I have some suspicions, but I wanted to hear it from you. You know, the basic question is, would the location in that letdown, the location in the break in that letdown line, whether it was upstream from the purification loop versus it is my understanding that it was presumed downstream from the purification loop. Would that make enough of a difference in the radiological releases to shift the emphasis from the sampling line break? If it is still bounded by the sampling, I don't care.

MR. PEREZ: Right.

MEMBER STETKAR: There's such a big difference in the size of those lines that I am just curious.

MR. PEREZ: Okay, and I will get back to you.

MEMBER STETKAR: Okay. Thanks.

MR. PEREZ: This slide is a little bit difficult to see, but I wanted to show the relationship between the control room air intakes and the multiple release points on the nuclear island.

There are two control room intakes on this top side of the picture. These are the control room

intake locations. And, then, there are multiple release points, depending on the event we are analyzing. You have the stack, which is one of the prominent release points, but you also have steam generator MSRT release locations which are going to be closer to the control room.

The calculations we performed for the onsite chi over q's, using that ARCON-96 code, basically, looks at the dispersion between each release point and the control room intake. So, you will notice that we have in our Chapter 2 of design certification multiple onsite chi over q's, multiple onsite dispersion factors, because there are multiple potential release points to the control room intake.

The main control room HVAC includes charcoal and HEPA filtration. This picture, this cartoon, shows the HVAC in the filtration configuration. There is a total of 8,800 cfm of air being recirculated. Of that, you have 3,000 cfm going through the charcoal filters, plus 1,000 cfm of fresh air also going through the charcoal filter. So, the total filtration is 4,000 cfm. And this is acting as a kidney filter for the entire control room envelope, which is 8,800 recirculated air. That is how the model of the control room has been formed.

We include 50 cfm of unfiltered in leakage. That 50 includes 10 cfm for ingress and egress, opening the doors of the control room for access.

This unfiltered in leakage can come from basically anywhere, if there is a break inside the plant, engineering safeguard features, ESF leakage. So, all the analyses have that consideration of the 50 cfm.

This cartoon depicts, if you would, the different compartments that we have and how these are interconnected. You first start with the inventory in your reactor core. This happens to be the design basis LOCA, which releases the radionuclides into the reactor building. We apply natural deposition, which we will talk in a minute, meaning you have gravitational settling of the particulates that go into the IRWST. What remains airborne will go into the annulus or can bypass the annulus and go into the safeguard building mechanical area. We credit filtration of the annulus space, and we credit filtration within the mechanical area of the safeguard building.

We also have to consider ESF leakage which has the concentration, the time-dependent

concentration, of the nuclides that have been deposited into the RWST.

The safeguard building ventilation system and the annulus ventilation system exhaust through the stack. We assume a ground-level release from the stack. And from that ground-level release, we calculate chi over q's dispersion factors for the exclusionary boundary, a low population zone, and, of course, from the atmospheric dispersion you have radioactivity entering the control room envelope, where, again, we credit the filters you previously saw.

What I would like to do is present two of the events.

CHAIRMAN BANERJEE: So, the way you showed that diagram, the control room is a different elevation from the base of the stack, right?

MR. PEREZ: This is not to scale.

(Laughter.)

CHAIRMAN BANERJEE: No, no, I am aware of that. But the way you have shown it, is there any change in elevation between the base of the stack and the control room?

MR. PEREZ: There is. Yes, there is. And I can't remember now. It is not much.

CHAIRMAN BANERJEE: Okay. So, in fact, release of the base of the stack is adequate to consider --

MR. PEREZ: Correct. Correct.

CHAIRMAN BANERJEE: -- from the viewpoint of the distance for dispersion and things like that?

MR. PEREZ: Correct. Because there is a regulatory guidance on what can be an elevated release, and it is based on the height of the stack relative to the nearest structure. It has to be two and a half times. We don't meet that.

So, then, you apply a ground-level release for this type of calculation. You look at basically what is a taut-string distance. It takes into account elevation, and you also take into account the wake effects of the building. All this is within the ARCON-96 methodology.

CHAIRMAN BANERJEE: Yes, so you do take credit for the WIGs?

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: Yes. Additional mixing in the WIGs?

MR. PEREZ: Well, is it mixing? It is, yes, dispersion.

CHAIRMAN BANERJEE: Dispersion factors,

yes. Well, it is mixing.

MR. PEREZ: Okay.

CHAIRMAN BANERJEE: Dilution, yes.

MR. PEREZ: So, what we have to do is present two of the radiological analyses and, then, collected the tube rupture analysis and the design basis LOCA analysis.

CHAIRMAN BANERJEE: So, just remind me of how this is done. So, suppose you have a release and you are in a certain type of weather --

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: -- which is the lowest dispersive, low winds. So, is there sort of a prescription on what sort of enhanced dispersion you get in the wakes then, based on that?

MR. PEREZ: Well, when you calculate the atmospheric dispersion factors, you follow pretty prescriptive regulatory guidance, and you take the worth, from the real data you take at a site, you would take all that hourly data, or whatever data you have for the entire year, right?

CHAIRMAN BANERJEE: Sure. You go to your weather.

MR. PEREZ: And you process it all and pick the worst atmospheric conditions, considering

delta T's, considering wind directions, everything. Then, for the accident, you would take the 95th percentile worst weather.

CHAIRMAN BANERJEE: So, we have got your worst weather. So, I know how that is done. Now where do you go from that in terms of each design has some specific geometries as to where the buildings are, where the WIGs are, and where the control room is.

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: So, you have got to connect the dots now, which is, having got the weather, the wind direction, and, of course, it is clear what the dispersion factors should be if you had clear ground.

MR. PEREZ: Correct.

CHAIRMAN BANERJEE: But that is just Briggs or whoever these people are that did this about 50 years ago, right? And now you have got buildings.

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: That is the issue really, because you are taking credit for these buildings now --

MR. PEREZ: Well --

CHAIRMAN BANERJEE: -- in some sense

because you are getting dilution to that in the WIGs. So, I am just looking for how you put the specific geometries --

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: -- for your layout into the calculation.

MR. PEREZ: Okay. The computer code, ARCON-96, was developed by Pacific Northwest, by a gentleman that actually did model wind tunnel modeling, small structures, to develop --

CHAIRMAN BANERJEE: Hopefully, then, in water because the Reynolds numbers would not work otherwise. But let's assume he did similitude. I'm not sure of what he did, but carry on.

MR. PEREZ: No, I didn't do it.

CHAIRMAN BANERJEE: Yes. Well, what this gentleman in Pacific labs did.

MR. PEREZ: From all the testing that he performed, the ARCON-96 methodology includes the shapes --

CHAIRMAN BANERJEE: Ninety-six is the year or --

MR. PEREZ: No, ARCON-96 --

CHAIRMAN BANERJEE: Is the vintage?

MR. PEREZ: The vintage of the code.

CHAIRMAN BANERJEE: Right.

MR. PEREZ: And it is what we use in the industry that I am aware of. There is no other alternative right now that has been accepted.

And you basically put your dimensions, your shapes, and from the release point to that control room intake, it gives you the dispersion.

CHAIRMAN BANERJEE: So, you put in the geometry?

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: And this code is used to get the dispersion factors. So, you use the same thing as the staff in any case?

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: Okay.

MR. PEREZ: So, we are going to discuss two of the events, the steam generator tube rupture, radiological consequences, and the LOCA, the loss-of-coolant accident.

As I mentioned earlier, we are following the NRC Regulatory Guide 1.183 guidance for the alternative source term. Appendix F is the steam generator tube rupture.

You do two cases. One case is called the pre-accident iodine spike case, where you have the

reactor coolant system concentration to the maximum value of 60 times your tech spec. So, in our case, the tech spec is 1 microcurie per gram dose equivalent to iodine-131. So, you initialize the calculation with an RCS concentration of 60, and you perform the tube rupture.

You also perform an accident-induced concurrent iodine spike, and you look at it for eight hours. That means that the iodine spike continues to increase with an appearance rate into the primary system by a factor of 335.

So, you keep increasing this appearance rate of iodine over that eight-hour period, developing a higher concentration of the reactor coolant system. We run the two analyses and see which one gives you the limiting radiological consequence.

Noble gases are set at the tech spec limit of 210 microcuries per gram dose equivalent to I-131.

CHAIRMAN BANERJEE: So, are your geometries about the same as what people did before? I mean, has the applicability of these codes been sort of explored?

MR. PEREZ: Well --

CHAIRMAN BANERJEE: I mean I know it is the only code around, but --

MR. PEREZ: No, no.

CHAIRMAN BANERJEE: -- that doesn't mean it is applicable to everything.

MR. PEREZ: Well, the geometry, you have a cylinder and different cylinders. This is also used for boiling water reactors which are square. The reactor buildings are square.

CHAIRMAN BANERJEE: So, the code has been sort of looked at for applicability in these different geometries?

MR. PEREZ: That is my understanding.

CHAIRMAN BANERJEE: So, we will find out from the staff also.

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: But that is your understanding? But that is your independent understanding?

MR. PEREZ: Correct.

CHAIRMAN BANERJEE: And your geometries are, more or less, the same as where the core has been used before?

MR. PEREZ: More or less, yes.

(Laughter.)

Absolutely, because when you look at -- I mean look at a PWR. A large PWR has --

CHAIRMAN BANERJEE: I mean these Siemens designs look like bunkers to me.

MR. PEREZ: Which?

CHAIRMAN BANERJEE: Your design, essentially.

MR. PEREZ: Okay.

CHAIRMAN BANERJEE: Basically, it is --

(Simultaneous speaking.)

MR. PEREZ: Well, not quite.

CHAIRMAN BANERJEE: It is a bit different? Not quite? Okay.

MR. PEREZ: Not quite. You still have the top of a cylinder with a dome.

CHAIRMAN BANERJEE: Right.

MR. PEREZ: Okay? There are many other type of similar reactors that have interference between the cylinder with auxiliary buildings, waste buildings, et cetera.

So, from the dispersion --

CHAIRMAN BANERJEE: It gives them more dispersion, right? If there are more buildings, they sort of mix things up a little bit more.

MR. PEREZ: Well, it depends, again, where your release point is relative to your intake.

CHAIRMAN BANERJEE: Right. So, I am just

wondering, has the applicability of this code been sort of checked for your sort of geometry?

MR. PEREZ: I believe we have a very simple geometry because, from the base of the stack, as I just showed you on that slide -- and we can go back to it. Oops, I am going the wrong way.

From the base of the stack to this control room intake, you basically go around this circle.

CHAIRMAN BANERJEE: So, you get very little dispersion there?

MR. PEREZ: Yes. It is almost a straight line, except for that little curvature.

CHAIRMAN BANERJEE: So, what is the increase in dispersion due to this?

MR. PEREZ: Distance.

CHAIRMAN BANERJEE: Yes, but distance is, you know -- I mean, if you had nothing, you get certain chi factors --

MR. PEREZ: Okay.

CHAIRMAN BANERJEE: -- chi X, chi Y, chi Z, you know. And now that you have got this thing in the way, then you enhance these factors, depending on what is in the way, right? So, if you have got a wake or a building, then that tends to give you higher dispersions, so you get less material. It is more

dilute --

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: -- by the time it gets to your intakes. And if you have a very uncomplicated path, the dilution is less. The more buildings you have around, things mix up better because of these wakes interacting and stuff like that.

MR. PEREZ: Okay. Now, remember, I say you take the taut-string approach.

CHAIRMAN BANERJEE: Yes.

MR. PEREZ: So, by doing that, you do not take the sort of circumferential distance to gain in a dispersion calculation. So, that is one thing you do.

The benefit of having a structure there to go into that control room intake, I cannot give you an absolute number. But, based on this geometry, I cannot imagine you are getting too much benefit because there is almost --

CHAIRMAN BANERJEE: A straight line, yes.

MR. PEREZ: Yes, a straight line.

CHAIRMAN BANERJEE: And depending on the direction the wind is blowing, if you get a very light wind between those two, of course, and, you know, a cold, clear night with inverted conditions, so nothing disperses --

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: -- Pascal F weather --

MR. PEREZ: Yes. Now, remember, in design certification, there is no site.

CHAIRMAN BANERJEE: Right, right.

MR. PEREZ: Right? So, some of these calculations, chi over q's were selected to allow a co-applicant to fall under.

CHAIRMAN BANERJEE: Sure. You have got an envelope --

MR. PEREZ: Right.

CHAIRMAN BANERJEE: -- which is pretty conservative probably?

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: Yes.

MR. PEREZ: We hope. Exactly. We hope.

CHAIRMAN BANERJEE: Why are you laughing?

MEMBER STETKAR: Well, he said, "We hope." He qualified it.

(Laughter.)

CHAIRMAN BANERJEE: Well, I guess it is up to the COL applicant, whoever is the applicant, to show that. Otherwise, they have to redo it, right?

MR. PEREZ: Well, they will have to perform their own dose calculations --

CHAIRMAN BANERJEE: Yes.

MR. PEREZ: -- and take a departure from the DC.

CHAIRMAN BANERJEE: Sure.

MR. PEREZ: We selected these values, and I think I have another slide to talk about it in a second, but some of these values, like the exclusionary boundary, came out of the EPRI Utility Requirements Document for a half-mile exclusionary boundary, 1.0, 10 to the minus 3 seconds.

And from the COLAs that I have worked with, if you keep it to that half a mile, no problem. If you get tighter, you are going to --

CHAIRMAN BANERJEE: The only reason I am going on about this is that there have been issues in other systems where we have got pretty close to certain limits, but with adequate margins, you know, I am talking about.

MR. PEREZ: But I think, are we speaking about control room habitability?

CHAIRMAN BANERJEE: Yes.

MR. PEREZ: I think we are speaking about unfiltered in leakage to the envelope, like your control room envelope is not tight, and you have to, for example, in operating plants, you have to do

testing of the envelope. And people have found problems. It wasn't really the dispersion factor that was giving them a problem. It was the leak tightness of the control room envelope.

CHAIRMAN BANERJEE: Well, I think there could be some special situations because it might have slightly offsite hazards with, you know, possibility of releases from those hazards, and they are not radiological issues. They are non-radiological issues. So, I stand --

MR. PEREZ: Okay.

CHAIRMAN BANERJEE: It is a different problem, but it is an issue which has come up. So, it depends where the site is.

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: You have got to realign what is transporting, whether it has got chlorine in it, and that --

MR. PEREZ: Chemicals, yes.

CHAIRMAN BANERJEE: Okay. Go ahead.

MR. PEREZ: Okay. So, for the tube rupture, let's see, the initial for all other radionuclides we have the design bases concentration responding 0.25 percent failed fuel fraction.

This, by the way, is the same one that we

did for Chapter 12, Shielding, you know, and the other chapters.

And we also, something that came out of the NRC Regulatory Issue Summary 2000.604, the Steam Unit Releases, which includes the noble gases along with the alkalis, and all other nuclides are assumed to stay within the liquid phase. So, you get into the steam space, the iodines, the noble gases, and the alkalis.

The scenario that was chosen out of many, many cases that were run from the thermal hydraulic side, we looked at four events that maximized the break flow and flashing fraction. And the one that was selected corresponds to a case where the reactor trip is by operator action concurrent with the loss of offsite power at 30 minutes.

And I believe that is very conservative. Because if you have 60 microcuries per gram of iodine or a coincidence spike that is bringing you up, you are going to see that pretty quickly from plant monitoring, air rejectors, steam line monitors. You are going to have plenty of notification. But we will wait 30 minutes and then we will start the action.

MEMBER STETKAR: Pedro, this will bear on a question I will have later. I want to make sure I

understand that nominal scenario that is used. And if you are not the right person to answer, then, hopefully, somebody else is.

As I understand it, in terms of timing, the tube rupture event occurs at T=0, time T equals zero. Both charging pumps are assumed to be available. Letdown is isolated, so that you have maximum makeup capability, and a manual reactor trip, as you have noted, occurs at 30 minutes in combination with simultaneous loss of offsite power. It is assumed that offsite power is lost simultaneous with the reactor trip, and that the operators initiate their manual tube rupture responses 10 minutes later. That's right. Yes, 10 minutes later. And that the main steam relief control valves on the ruptured steam generator sticks open at about the same time. It is actually about a minute later, 2450 seconds as opposed to 2400 seconds. And at that point, you start getting the direct offsite release.

And according to the timeline, at least as I can piece together, the partial cooldown of the intact steam generators is completed at 3600 seconds or about 20 minutes after the operators initiate their actions.

Now, from a release perspective, for the

first 30 minutes, you are actually releasing into the main condenser. Does the radiological analysis account for the dilution within the main condenser? So, effectively, you don't get any releases?

MR. PEREZ: We apply a DF of 100. In reality, the DF is much, much higher.

MEMBER STETKAR: Okay, but you do apply the DF --

MR. PEREZ: Yes.

MEMBER STETKAR: -- for that first 30 minutes?

MR. PEREZ: Yes.

MEMBER STETKAR: What is the duration of the actual unmitigated direct release from the stuck-open valve? It sounds like it is a maximum of about 20 minutes, but it might be only 10 minutes, depending on what your presumptions are about isolating the steam generator and whether or not pressure in that steam generator increases enough so that you -- it is a timing issue of how rapidly they cool down versus increase in pressure in the ruptured steam generator. Can you get up to the safety valve setpoints and still get a little more release?

After 3600 seconds, you know, I don't care how long you take it out, you're not getting any

measurable releases out of that thing. So, the question is, what release duration do you have, unmitigated, direct offsite release duration do you have from this nominal scenario, if I have characterized that correctly?

MR. COFFEY: This is Ken Coffey.

I think Paul Bergeron might have the answer for us.

There is actually, following the failure of the MSRT, the steam generator pressure of the affected steam generator decreases rapidly. So, we actually have an automatic isolation of the MSRIV, the main steam release isolation valve, approximately, I will say, within a couple hundred seconds after the failure of the MSRT.

MEMBER STETKAR: Yes, so you are accounting for that? Okay. So, you only have an actual release duration of two or three, three or four minutes perhaps?

MR. BERGERON: There is a table in the FSAR that gives the sequence of events.

MEMBER STETKAR: Yes, I was trying to find that.

MR. BERGERON: Yes, it is Table 15.6-6.

MEMBER STETKAR: 15.6-6? Thanks. I was

trying to listen and talk and find tables. There it is.

MR. BERGERON: Yes, you get the point where the steam generator reaches the MSRT setpoint.

MEMBER STETKAR: Okay. Yes.

MR. BERGERON: And, then, where the IV isolates.

MEMBER STETKAR: Yes. Yes. Okay. So, your release duration is only two minutes?

MR. BERGERON: Yes, in that order.

MEMBER STETKAR: In terms of measurable-type releases?

MR. BERGERON: Right.

MEMBER STETKAR: Okay.

MR. BERGERON: At least for that period, but then you have releases that continue through --

MEMBER STETKAR: Yes, but those releases, you know, they're --

MR. BERGERON: Those are direct.

MEMBER STETKAR: Hum?

MR. BERGERON: That is a direct release.

MEMBER STETKAR: This direct release --

MR. BERGERON: Right.

MEMBER STETKAR: -- is two minutes?

MR. BERGERON: Right.

MEMBER STETKAR: But the first 30 minutes is filtered through the main condenser.

MR. BERGERON: Correct.

MEMBER STETKAR: Then you get a direct release of a couple of minutes.

MR. BERGERON: And, then, you get a continued release through the intact generators --

MEMBER STETKAR: Right, but those are pretty doggone small.

MR. BERGERON: They are small compared to this one.

MEMBER STETKAR: Yes. You account for leakage. But you also get a DF through the water in the attacked steam generators and all that.

I just wanted to make sure that -- and it is only about two minutes --

MR. BERGERON: Okay.

MEMBER STETKAR: Because later this afternoon, I guess when we get into the mitigation systems, I want to understand whether that nominal scenario, in fact, is limiting from a release perspective compared to another one that I dreamed up. But it is too long to get into it right now.

I just wanted to make sure I understood this one first, and I had even missed the two-minute

isolation. So, thanks.

MR. PEREZ: But I believe that in our analyses we said at 30 minutes a decision is made to trip the reactor --

MEMBER STETKAR: Yes.

MR. PEREZ: -- the operator action.

MEMBER STETKAR: Yes.

MR. PEREZ: An additional 10 minutes to isolate the ruptured steam generator. So, an action has not been performed after the reactor trip for 10 minutes. So, you are releasing through the MSRT.

MEMBER STETKAR: No, that's not actually true because the MSRT doesn't stick open until pressure gets in the ruptured steam generator. It gets to the isolation valve opening setpoint, which, indeed, according to this timeline -- and thanks for the table -- is a minute after that 10 minutes. It is 11 minutes into the event, 10 minutes and 50 seconds after the reactor trip.

MR. BERGERON: I think we are releasing through the intact generators, too, until the affected one is isolated.

MR. PARECE: No. This is Marty Parece again.

The key is the loss of offsite power.

When you trip the reactor and then you get the loss of offsite power, all steaming to the condenser ends.

MEMBER STETKAR: That is correct.

MR. PARECE: And all four steam generators steam to the atmosphere through the MSRTs and remove decayed heat.

MEMBER STETKAR: According to this, not until 11 minutes later.

MR. PARECE: No, it happens, from reactor trip alone, it happens from all four.

MEMBER STETKAR: It says, "Affected steam generator pressure increases to MSRT setpoint. MSRCV fails open in fully open position at 2450 seconds."

MR. PARECE: That's what it says.

MEMBER STETKAR: That's what it says.

MR. PARECE: But I don't see how -- well --

MEMBER STETKAR: That's what it says.

MR. PARECE: I know, but you have to release decayed heat.

MEMBER STETKAR: Yes, you do.

MR. PARECE: So, all four steam generators have to steam on reactor trip. They have to.

MEMBER STETKAR: Well, but --

MR. PARECE: So, during that time, you are

releasing --

MR. COFFEY: I don't know what's happening. One of my questions was, how --

MR. PARECE: Well, it is a good question.

MEMBER STETKAR: It is a minute after the guys start to bottle the thing up that it opens.

MR. PARECE: It is a very good question the way it is written because, physically, you have to get the decayed heat out.

MEMBER STETKAR: Yes. No.

MR. PARECE: So, all four generators in under 10 minutes.

MEMBER STETKAR: Right. Right.

MR. PARECE: Then, the operator decides to isolate, and that's when he finds out that -- that is really when his first chance to find out the valves, if there is a failure in the valve.

MEMBER STETKAR: Yes. Yes. But this seems to imply that the valve doesn't stick open until about, let's say, a minute after they decide that they have got a tube rupture. They know that they have a tube rupture from N16.

CHAIRMAN BANERJEE: It seems that we have --

MEMBER STETKAR: There seems to be a

timing question here.

CHAIRMAN BANERJEE: Yes, a question to be sort of clarified and answered. Maybe we can defer that until --

MS. SLOAN: Yes, I think we can handle that during the tube rupture session.

MR. COFFEY: And I could probably add an element to that discussion as well.

MEMBER STETKAR: Because I dreamed up a different scenario that might have a different timing, but this discussion helps me kind of over lunch to understand this scenario a little bit better, though. So, thanks.

CHAIRMAN BANERJEE: So, Derek, let's ensure that this gets discussed during the afternoon.

MEMBER STETKAR: Yes. Well, they have a presentation on the tube rupture.

CHAIRMAN BANERJEE: Yes. Yes. Sure. So, we are going to follow up on that. Okay.

Go ahead.

MR. PEREZ: And again, this slide may help or not. But atmospheric releases consist of the secondary-side activities, reactor coolant system leakage via the affected steam generator, and the normal leakage via the three intact steam generators.

So, those are assumed to be leaking from the primary to the secondary at the tech spec limit.

And again, you have the condition of the either preexisting iodine spike or the accident-generated. So, you are putting in radioactivity into those steam generators. And there, you cool the plant down. For the duration of the eight hours that I mentioned, we track the inventory in the intact steam generators and release it through the plant cooldown.

So, the early releases are via the condenser until 30 minutes, when the operator trips the plant, and, then, via the main steam relief train. And, then, iodine and alkali depletion due to deposition within the condenser is credited with a DF of 100.

CHAIRMAN BANERJEE: What is the pH in this condenser? Do you have a trapped behavior? I mean you say it is depletion, but I don't know what the pH is.

MR. PEREZ: No, no, no. The depletion is by condensation on the condenser tubes. So, it is not a chemical. It is just a condensation will deposit. It has an enormous surface area.

CHAIRMAN BANERJEE: Yes.

MR. PEREZ: But the pH, I would say, I

don't know, but we do not take credit for any speciation, any iodine speciation.

CHAIRMAN BANERJEE: What do you mean by alkali depletion?

MR. PEREZ: Both the iodines and the alkalines are being removed.

CHAIRMAN BANERJEE: What alkali are we talking about?

MR. PEREZ: The chemicals.

CHAIRMAN BANERJEE: Oh, okay, on the --

MR. PEREZ: So, from the reactor cooling system into the secondary system due to the leakage, transported to the condenser, the condenser is acting like a filter with a DF of 100 for these 30 minutes.

CHAIRMAN BANERJEE: I guess the condenser water you are saying is going to be a pH 7, essentially, or what?

MR. PEREZ: No, what I am saying is the process of condensation removes particulates. It removes radioactivity. Noble gases has a DF of 1. It doesn't do anything. But for the iodine and the alkalines that we are tracking, the condenser has an effective DF.

CHAIRMAN BANERJEE: Yes. I am just wondering where that 100 is coming from. Is that a DF

which is normally taken for things like this?

MR. PEREZ: Well, it has been taken in previous applications.

CHAIRMAN BANERJEE: I notice you reference a NUREG.

MR. PEREZ: Correct.

CHAIRMAN BANERJEE: Yes.

MR. PEREZ: It is much higher. The NUREG will tell you it's 10,000.

CHAIRMAN BANERJEE: Okay.

MR. PEREZ: It is much higher because, again, the big surface area plus the process of condensation does effective cleanup. We used 100.

CHAIRMAN BANERJEE: Dana probably knows this stuff. I don't follow this, but okay. That is where it comes from, just the condensation into a large surface area. Okay.

MR. PEREZ: Right. I mean I can give you an analogy. In firefighting, if you go into a heavily-suited room, you spray that air with water; it drops everything down. It cleans it. So, the effect of that water with the heat just scrubs out that air.

CHAIRMAN BANERJEE: Yes, but it is different. So, into your condenser is coming steam, right?

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: And typically, what is happening is that you get film wires or drop wires condensation. This is film wires on your condenser tubes.

Then, it is not quite the same as spraying a room with water. So, what you are assuming is that iodine then transfers particulates by some mass transfer process from the steam to the liquid. Now I am sure this is treated in the NUREG. I'm just not familiar with it. But it is not the same process as spraying it. Unless there is some analogy, it is more a mass transfer process.

Now, if there is a mass transfer going on, then there has to be a driving force, and the driving force will be based on what is the pH of that water because the solubilities, and everything, are different. It is not quite the same thing, unless I am missing the mechanism.

Does Bill know this thing better than I do?

MR. PEREZ: There is a term. You know, when the steam is being condensed, you are getting the water droplets, if you would.

CHAIRMAN BANERJEE: So, let's take just a

liquid film. Okay? And it is condensing there, right? So, there are different mechanisms. So, the iodine might be also condensing on the film, or whatever. I just don't understand. It is not the same as spraying into a particulate system, or whatever, and then washing it down because this is just steam condensing on some form of a surface.

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: There is a very large surface area. If it is a very large surface area, the mass transfer is good. Accessibility to this cold region is good.

Now, as the steam moves, it brings with it iodine as well, which is, let's say, uniformly dispersed in the steam.

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: So, the mechanism is different, and I just don't know what the mechanism is. I'm just trying to figure it out. I guess experiments show that the decontamination is very high probably?

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: That's good enough right now. Let's carry on, yes.

MR. PARECE: I was going to suggest that

we move ahead --

CHAIRMAN BANERJEE: Yes.

MR. PARECE: -- because this is not specific to EPR.

CHAIRMAN BANERJEE: No.

MR. PEREZ: This is true for all condensers and all plants.

Also, the first papers I remember on this go back to the seventies, I think, by Postma and Tam. There is a NUREG where they tried to start the first models on figuring out how iodine is washed out in steam generators during a tube rupture. And in that, they reference a lot of data, and a lot of that is in the NUREG as well.

So, the real factor is that iodine in its particulate and in its gaseous forms wants to maintain an equilibrium in the vapor phase with the liquid phase.

CHAIRMAN BANERJEE: Yes.

MR. PARECE: So, if you have a lot of liquid with no iodine in it, the iodine runs to the liquid to maintain this relationship --

CHAIRMAN BANERJEE: Partition, yes.

MR. PARECE: -- this factor.

CHAIRMAN BANERJEE: Yes.

MR. PARECE: And it generally varies between 1,000 and 10,000, depending on the pH and the oxygen content. And the guys used 1,000.

So, there is a lot of information out there if we want to go for that.

CHAIRMAN BANERJEE: No, no. I don't particularly want to visit it. I just want to know that that is a secure number.

MR. PEREZ: I am confident that number is secure.

CHAIRMAN BANERJEE: Okay. Carry on.

MR. PARECE: This has been studied for --

CHAIRMAN BANERJEE: Decades.

MR. PARECE: -- at least 40 years now.

CHAIRMAN BANERJEE: Yes.

MR. PEREZ: Okay. The next, we decided to discuss the LOCA, primarily because we are going to use Dr. Powers' model.

(Laughter.)

So, I thought he would be asking --

CHAIRMAN BANERJEE: For this part of it, Powers should be here.

(Laughter.)

MR. PEREZ: Yes. Basically, we start out with --

CHAIRMAN BANERJEE: He can't say anything now.

(Laughter.)

MEMBER SHACK: It never stopped him before.

CHAIRMAN BANERJEE: It never stopped him before.

(Laughter.)

MR. PEREZ: We start with the bounding co-inventory. You have seen this before. We do a parametric set of cases using ORIGIN from 5- to 62-gigawatt based on metric ton in 5-gigawatt days per metric ton steps, looking at enrichments, 2 percent enrichment, 3.5, and 5 percent enrichment. And from there, we select a bounding core inventory.

CHAIRMAN BANERJEE: However, we can question Bob Henry's part of the model.

(Laughter.)

MR. PEREZ: The scenario is basically driven, once again, by Regulatory Guide 1.83. It will be Appendix A.

As you well know, it is a time-based release. The timing is provided by the guidance. We only credit natural deposition within the reactor building. We used the Powers 10 percentile natural

deposition model. That model ends at 22.2 hours, and we wanted it to continue using natural deposition. It would not be proper to keep a constant value at 22.2 because the lambdas, the precision rates, keep going down.

Within the RADTRAD code, you have a choice of using Powers or Henry. We, then, extended the deposition with Henry. And what we did was we ran Henry and Powers together, okay, and then knew we had a good transition to continue with Henry.

Now these lambdas are quite natural deposition and quite small. You continue to have a high airborne concentration, but it is an effective way of reducing your airborne contamination.

We start the accident assuming we are actually purging the containment. Ten seconds of containment purge is assumed.

We have an allowable leakage rate of 0.25 weight percent per day for the first 24 hours, reduced to 50 percent of that value thereafter for the duration of the 30-day calculations we performed for the LPZ and control room.

Any release, any containment leakage release, will be assumed to be through the steam generator three main steam relief train during the

time it takes the annulus ventilation system and the safeguard building mechanical area filters to draw down to subatmospheric dose locations. Until we meet subatmospheric, we do not credit filtration.

CHAIRMAN BANERJEE: How long does that take?

MR. PEREZ: Three hundred and five seconds for the annulus drawdown system and 289 seconds for the safeguard building mechanical area.

CHAIRMAN BANERJEE: What buffers do you use? Trisodium phosphate?

MR. PEREZ: Trisodium phosphate.

CHAIRMAN BANERJEE: Yes.

MR. PEREZ: To buffer.

CHAIRMAN BANERJEE: Yes, for this plant, right?

MR. PEREZ: For this plant, yes.

CHAIRMAN BANERJEE: Okay. And how is that suspended? Does it dissolve?

MR. PEREZ: It dissolves by --

CHAIRMAN BANERJEE: Very quickly? How long does it take?

MR. PEREZ: It's 14,000 pounds in four baskets. It is very hygroscopic. It is absorbs water within hours, within hours.

CHAIRMAN BANERJEE: So, in this period, it is all dissolved or not dissolved? Three hundred and five seconds, nothing has happened, right?

MR. PEREZ: Oh, no, but you are not producing -- remember, you start to develop acids due to radiolysis of water and cabling, and that is time-dependent.

CHAIRMAN BANERJEE: So, your pH is always staying around 7?

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: Throughout? And you have done that calculation?

MR. PEREZ: Yes. Yes, we have done that calculation. That is how we determine 14,000 or so pounds of TSP.

What is happening during this time is the airborne contamination in the reactor building is leaving the building unfiltered. So, it has not even had a chance to make it to the RWST. So, it is an airborne through the penetrations that I showed bypassing --

CHAIRMAN BANERJEE: You just take deposition due to this Powers/Henry model?

MR. PEREZ: Yes.

CHAIRMAN BANERJEE: That's all you are

doing right now?

MR. PEREZ: That is all we are doing, yes.

CHAIRMAN BANERJEE: Okay. And that is roughly what? Not that much?

MR. PEREZ: The lambda, something like 0.2 per hour. Compared to other methods, it is very, very low.

CHAIRMAN BANERJEE: Yes. So, why do you need to take it into account at all? Does it help?

MR. PEREZ: Oh, yes. Yes. Over 30 days, yes.

CHAIRMAN BANERJEE: Well, yes, over 30 days, but in this early phase --

MR. PEREZ: You need to start it sometime.

CHAIRMAN BANERJEE: Okay. So, that is why you are doing it, right?

(Laughter.)

But it doesn't help you very much in the early phase, does it?

MR. PEREZ: No.

CHAIRMAN BANERJEE: No.

MR. PEREZ: No. No. No.

CHAIRMAN BANERJEE: It doesn't do much. Okay.

MR. PEREZ: As I mentioned, after we

finally have the drawdown of these rooms, then we have the release at the base of the stack. Because the alignment of these filters is to exhaust through the stack.

The environmental releases, 10 seconds on filter release during containment purge. Containment leakage via the annulus without holdup or depletion, and then, later, 99 percent exhaust filters, ventilation, filtration for the annulus and the mechanical areas of the safeguard building.

ESF component leakage is via the safeguard building without holdup or depletion. And again, within the safeguard building, there's 99 percent filtration following drawdown.

And there was another presentation that I remember the ACRS asking about. You know, how come the doses are high? I think a question was asked like that, and, basically, I had looked at other design centers, and our doses are comparable. And they should be because they are all using the same methodology. They are also using similar dispersion factors.

So, when you normalize things to power, when you look at a per a megawatt basis, we are not that different. If you are taking a passive system

with the lower deposition, you are going to be using the same models. You are going to be using the same depletion rates.

These are the calculated doses in design certification for the exclusionary boundary, the low-population zone, and the control room. All these doses here have been calculated using data from thermal hydraulic calculations and fuel performance calculations.

These last three -- and Jonathan had alluded to this -- these last three, we actually did it backwards. We calculated what would be the decay time for a fuel-handling accident that would give you 90 percent of the dose acceptance criteria. So, if you divide these numbers, you would realize that all these are 90 percent of the acceptance criteria.

And again, that was done. Part of it was timing of when the calculations were done and, again, to provide a bigger umbrella for the COLA applicants, for the future when these applications come through.

And I believe this is my last slide.

CHAIRMAN BANERJEE: Right. Okay. So, does anybody have any questions, comments? John? Bill?

(No response.)

So, in that case, what we will do is we will take a lunch break now. It was supposed to be from 12:15 to 1:15. What we will do is we will take it from 12:30 to, let's make it, 1:30. I think we can manage to keep it under control. Okay. So, we will start at 1:30 with the steam generator, I guess.

Thank you.

(Whereupon, the foregoing matter went off the record for lunch at 12:29 p.m. and resumed at 1:31 p.m.)

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

1:31 p.m.

CHAIRMAN BANERJEE: We are going to go back into session, and our reporter is ready. So, back in session. Back on the record.

MR. COFFEY: Thank you.

Good afternoon, everyone.

My name is Ken Coffey, an Engineer IV with AREVA in the Engineering Integration Plant Operations Division.

A little background on myself: I received my bachelor's degree from the University of Illinois at Urbana-Champaign in nuclear engineering. I did a master's degree at L'Ecole des Mines de Nancy in Nancy, France in material science.

A little over nine years of engineering experience, the last seven of which have been spent developing shear action guidance and emergency procedure guidance for numerous projects, notably GALL Phase 2. The last two years, I have been working on the U.S. EPR. I am currently the Emergency Procedure Guidance Development Team Lead.

CHAIRMAN BANERJEE: You are the what?

MR. COFFEY: I am the lead for the Emergency Procedure Group.

CHAIRMAN BANERJEE: Okay.

MR. COFFEY: The agenda for what I will present to you today. So, that is an overview of the steam generator tube rupture mitigation strategy and goals. We have broken down the mitigation into four main stages. I will cover those one at a time, and finish with the mitigation comparison, how we would use various systems, notably, with or without offsite power. That is one of the major differences into how one would mitigate a steam generator tube rupture.

I would like to start off by saying that the U.S. EPR, the mitigation of a steam generator tube rupture for the U.S. EPR is very similar to existing four-loop PWRs. We have the ability to mitigate them in essentially the same fashion that the existing fleet does today.

We have been able to incorporate the specificities, as Doug Brownson showed us earlier. Once we reset the MSRT setpoint to the high value of 1436 psia, we can then mitigate an SGTR using ECCS systems and control our CS inventory without the risk of lifting the safety. This does help us greatly to minimize releases in the short-term and, also, prevent the risk of overfill due to high-pressure injection of the ECCS.

We do utilize the extra borating system to provide a safety-related means to borate the RCS. So, the main goals in SGTR mitigation:

First, you want to achieve a controlled state to ensure RCS stability, a controlled state being a state in which the core is subcritical, being removed via the steam generators, and core water inventory is stable. And the RCS makeup compensates for the SGTR leak rate.

Next we move to terminate the releases from the affected steam generator as soon as practical and move to cancel the leak rate from the primary to the secondary, and, then, finally, reach safe shutdown. And I will go into detail on how we do each of those to meet those goals.

So, the four main stages of STGR mitigation:

First, you have to identify that you have a steam generator tube rupture.

Next, you need to shut down the reactor and isolate the steam generator as quickly as possible.

Next, again, you cancel the leak flow from the RCS to the affected steam generator.

And finally, cool down and depressurize

the RCS as well as the affected steam generator to an RHR connection.

So, first, SGTR identification. How do we know we have one? Primary means of SGTR identification are main steam line radiation monitors. As noted earlier, we have four main steam line radiation monitors that are safety-related per train. We also have non-safety-related steam generator blowdown line radiation monitors as well as condenser off-gas rad monitors.

For situations where we could have an SGTR at low power, we could utilize system response to help diagnose the SGTR. We have feed-flow/steam-flow mismatch as well as watching the pressurizer pressure and level decrease.

CHAIRMAN BANERJEE: Just go back one slide.

MR. COFFEY: Yes.

CHAIRMAN BANERJEE: Okay. Thanks.

MR. COFFEY: Okay. So, once we have identified the SGTR, we need to shut down the plant. Dr. Witter showed us earlier this morning all the automatic reactor trip functions. So, I am not going to go back into that detail right now.

What is worth noting is that for double-

ended guillotine steam generator tube rupture, the startup of the second CVCS charging pump does allow us to compensate for the leak. This is why the reactor trip may be manual on one of the credited actions we took in our FSAR analyses.

So, once we have tripped the reactor, then we move quickly to isolate it. Isolation of the affected steam generator involves closure of the main feedwater and emergency feedwater isolation valves that affect the steam generator; isolation of the steam generator blowdown lines; closure of the main steam isolation valve, the affected steam generator, and the reset of the MRST setpoint.

This isolation can be either automatic or manual. There is an automatic signal sent from the protection system to perform this isolation. If we have a partial cooldown signal with a concurrent either steam generator or high steam generator level or high activity in the main steam line, detected by the main steam line radiation monitors.

We also have the ability to do this manually. The manual isolation will be performed by a group command. So, essentially, the operator has one button to push, and all these actions will occur.

So, once we have shut down the plant and

isolated it, we have isolated this extra generator, we need to cancel the leak. Stages 3 and 4 can vary greatly depending on the equipment that we have available to mitigate the transient.

So, what I will start off by presenting here in the initial runthrough of stages 3 and 4 are the systems that we used for an SGTR with concurrent loss of offsite power in our FSAR.

In order to ensure RCS inventory, we need to initiate safety injection if we don't get an automatic safety injection signal. One of the outputs of the safety injection actuation is the initiation of the partial cooldown at 180 degrees F per hour, using only the unaffected steam generators.

We will feed the unaffected steam generators with emergency feedwater, which will be started manually, if necessary. And again, RCS inventory is maintained by the use of the safety injection system.

And at this point, we will allow the RCS pressure to be reduced to that of the affected steam generator. Essentially, we have no high-pressure injection means in the RCS. So, there is nothing keeping the RCS pressure hot. The RCS pressure will eventually fall to that of the shutoff head. And

while the RCS pressure is flowing down, the affected steam generator pressure will tend to rise until they reach a quasi-equilibrium.

CHAIRMAN BANERJEE: So, this 180 fan height is the same as if you didn't have the steam generator bottled up, right?

MR. COFFEY: That's correct.

CHAIRMAN BANERJEE: And that is possible with three steam generators?

MR. COFFEY: Right. That is correct. And, essentially, the cooldown rate, we did see in our analyses that we were able to achieve the 180 degrees per hour. The rate, in and of itself, is not that critical to the transient. Again, as I noted, since the steam generator is already isolated, our releases have been stopped from the affected steam generator. So, if it is 180 or 150, the overall impact is negligible. So, yes, but our analyses have shown that we are able to achieve the 180 degrees F with three steam generators.

CHAIRMAN BANERJEE: Okay. And how long does it take to get down to 870 then?

MR. COFFEY: Partial cooldown lasts about 15 minutes. It lasts approximately 15 minutes. At that point, the secondary side is the side that gets

down to 870 psi. The primary side lags. I don't know exactly --

CHAIRMAN BANERJEE: But 870 is --

MR. COFFEY: Is the steam generator pressure of the unaffected steam generators.

CHAIRMAN BANERJEE: Okay. Right. Go ahead.

MEMBER SHACK: How long until you can do the medium head injection for high head injection?

CHAIRMAN BANERJEE: That's the secondary site, right?

MR. COFFEY: I want to say, I can look into it. We have the detailed results and calculations. I can get that information.

CHAIRMAN BANERJEE: So, 870 psi has a significance? John was saying that it allows the medium pressure injection.

MR. COFFEY: Right. So, theoretically, at the end of partial cooldown, we should have --

MEMBER SHACK: That is enough. Again, I didn't do the math, but on convey designs it is enough subcooling such that pressure in the primary system --

CHAIRMAN BANERJEE: Gets down to --

MEMBER SHACK: -- gets down below whatever the shutoff head of medium head safety injection is.

CHAIRMAN BANERJEE: Which is somewhat above this?

MR. COFFEY: Right. The shutoff head is 1405.

CHAIRMAN BANERJEE: Okay.

MR. COFFEY: So, again, stage 3, we acted to cancel the primary to the secondary leak. Now we need to cool down and depressurize the RCS to RHR connection conditions. A cooldown of the RCS is performed using the MSRTs at the unaffected steam generators at either 45 degrees an hour or 90 degrees F per hour, depending on the number of EBS trains we have in service. One train of EBS allows us to cool down at 45 degrees an hour to 90.

Medium-head safety injection will be managed in order to allow us to maintain adequate subcooling margin, keep RCS inventory in the desired range, at least within the pressurized level indication range, and, also, it will manage to allow for the depressurization of the RCS to the RHR connection condition pressures.

CHAIRMAN BANERJEE: Now, for the unaffected steam generators, do you basically take the steam to the condensers?

MR. COFFEY: No, this will be via the

MSRTs. Right now, this is an SGTR with loss of offsite power. So, the condenser is not available. So, in this case, the --

CHAIRMAN BANERJEE: Loss of volts?

MR. COFFEY: That is correct.

CHAIRMAN BANERJEE: All right. So, you just blow steam?

MR. COFFEY: Just blowing steam out the atmosphere, that is correct.

Okay. And the final depressurization of the RCS will be performed by cycling the PSIVs, again, while trying to maintain within a desired band of some clean margin. And finally, that will allow us to get down to RHR connection pressures and temperatures, 455 psia and 350 degrees.

Okay. As we move forward to develop the detailed emergency procedure guidance, they will be developed to utilize all available systems, obviously. We don't just do SGTR mitigation with loop.

Preference is given to systems that accomplish the task as effectively as possible and as close to normal operating conditions as the situation permits. And again, availability of offsite power is one of the biggest factors in determining what we can use, what systems will be available, and what

mitigation strategies are going to be put in place.

So, here is a side-by-side comparison, again, stage 3, which is our leak cancellation stage of the systems that we would be able to use if we did have offsite power available to us. The RCS inventory could be controlled with CVCS makeup and letdown. RCS pressure would be controlled using sprays and heaters. SG inventory would be feeding the SGs with main feedwater. And obviously, anytime you have SGTR, you want a steam condenser, if it is available.

And again, on the right you saw the systems that we used without offsite power: MHSI and EBS injecting into the RCS, and the RCS pressure controls not nearly as fine as we would have if we did have offsite power available to us. And the EFW is used for the secondary side.

And again, for stage 4, our final depressurization, with offsite power, there is a good chance we will have reactor coolant pumps running. If that is the case, then we have a much different transient. We will have forced circulations through the tubes of the affected steam generator, which acts to cool and essentially depressurize the steam generator, essentially, back-cooling the steam generator with a cooler water.

We continue our cooldown and depressurization just as we did in stage 3, the CVCS providing the makeup; sprays and heaters, chill pressure; MFW, and turbine bypass.

Then, without offsite power, again, the RCS pressure is held up by the pressure of the affected generator, which essentially acts as a secondary pressurizer for the system. So, therefore, we need to depressurize the RCS and the affected generator as well.

Again, the systems used in that case for the MHSI and the EBS on the primary side; EFW and MSRTs on the secondary, with our final depressurization done via the PSRVs.

CHAIRMAN BANERJEE: So, does this just go into natural circulation?

MR. COFFEY: Yes. We do observe natural circulation in the three loops where we continue to steam the steam generators.

CHAIRMAN BANERJEE: And how much is the maximum void fraction that occurs in the system? Is there any? Because you now don't have pumps, right?

MR. COFFEY: Right.

CHAIRMAN BANERJEE: So, does the system void?

MR. COFFEY: I can ask --

MR. SALM: Yes. This is Robert Salm.

You have the affected loop that has been isolated, and the secondary side is full of hot, saturated liquid, and you are cooling down with natural circulation in the three intact loops. The affected loop is idle, and you have that hot water bottled up in there.

So, when you start bringing down the primary side, you are going to tend to flash and void in the upper part of the primary tubes of that affected steam generator.

CHAIRMAN BANERJEE: Only there?

MR. SALM: Only there. You have more than enough makeup with the MHSI. So, it is just in that one loop.

And, then, when you depressurize it, you know, it just sort of flushes the saturated liquid out of there.

MR. COFFEY: That's my final slide, but just to recall three main points: the SGTR mitigation for the U.S. EPR is very similar to existing four-loop BWRs. Our additional safety features help us to minimize releases and prevent overfill. And obviously, our MPGs, our major procedure guides, will

be developed to utilize all available systems.

So, do you have questions?

MEMBER STETKAR: Yes. I understand now, I think, the sequence of events for the nominal tube rupture sequence that is used in the release analysis. I thought about a different sequence of events. And I don't know what all the setpoints are, so I don't have a good sense of the timing. But if you can help me walk through it a bit, maybe I can understand it a bit better.

Suppose you had no charging. Okay? So take a lot of the safety analyses, assume that one train is out for maintenance and you have a single failure in the second train. Since you only have two charging pumps, I could have one charging pump out for maintenance and I will take my single failure to be that other charging pump.

So, at T=0, I have no charging.

MR. COFFEY: Even for this case, let me interject, since the charging is not a safety system, we can just assume -- we had to do a whole spectrum of --

MEMBER STETKAR: Let's do that. Let's do that then. I didn't even take another single failure, but I wanted to understand this.

So, let's do away with charging. However you rationalize it, it is not there.

Under those conditions, you do not have any type of makeup to the primary system, right?

MR. COFFEY: Not at the initial stages.

MEMBER STETKAR: And I would assume, then, that you will have an automatic reactor trip --

MR. COFFEY: Right.

MEMBER STETKAR: -- from what? I'm going to guess low pressurizer pressure, but --

MR. COFFEY: Low pressurizer pressure, low pressurizer level, one of those two.

MEMBER STETKAR: One of those two?

MR. COFFEY: Correct.

MEMBER STETKAR: I don't know when that occurs. I don't know when that occurs in time.

MR. COFFEY: Okay.

MEMBER STETKAR: So, let's assume you have the tube rupture at time T=0.

MR. COFFEY: Right.

MEMBER STETKAR: The question is, when does that automatic trip occur? If I, then, assume that offsite power is lost at the time that the reactor trip occurs, consistent with the manual trip assumption, at that time -- and I will call it "T

reactor trip" -- I now have no secondary steam relief. The main condenser is now isolated. I have no turbine bypass. The pressure starts to go up in my ruptured steam generator.

At some time out, I have a competing concern now. Pressure is going up in my ruptured steam generator. So, it may trigger the MSRT system to open. And I have pressure in the primary system continuing to decrease that is heading me toward an SI. So, I have got those two things going on.

At some time after T reactor trip, I will now pop open the main steam relief train and I will take the single failure of stick open the main steam control valve. Okay? And at some time, I will get an SI which triggers the blowdown steam generators.

As I understand it, the blowdown steam generators will blow down the ruptured steam generator. I don't see anything that will prevent that from happening.

MR. COFFEY: Well, actually, as I point out right here, this automatic SG solution, in this case where we have safety injection, an output of the safety injection is a partial cooldown. Again, when you have partial cooldown in conjunction with either a high activity in the main steam line or a high SG

level, you will get this automatic isolation of the affected generator with a signal sent from the protection systems.

MEMBER STETKAR: Where precisely does that show up in the logic diagrams for main steam release isolation valve isolation? Because the only signal I can find for that is low pressure in the steam generator, at least in the FSAR, Figure 7.3-13. And that is the only one that is referred to in the text of the FSAR telling me how I get that main steam relief train isolated, the MSRIV.

MR. COFFEY: Okay.

MEMBER STETKAR: So, if there are other signals, I have been searching to try to find them. They might in there somewhere. I have tried Chapter 7. I have tried Chapter 6, and I have tried Chapter 10. I can't find anything that talks about high radiation as an input stream. Seven is the one I was looking at mostly, which shows the logics.

But it tends to show only the safety-related logic. So, if the other is a non-safety-related isolation signal, I don't know whether the N16 monitors are considered safety-related or not.

Anyway, the concern here is I am trying to understand a scenario where the normal automatic

response of the plant initiates a reactor trip automatically that cascades into a loss of offsite power and you get a safety injection because I have no makeup to the primary system, which triggers a single to blow down --

MR. COFFEY: Right.

MEMBER STETKAR: -- the ruptures steam generator, and that that condition persists until 30 minutes, at which time I can now start taking credit for operator actions to manually bottle up the ruptured steam generator.

And the question is, are releases under that scenario worse than releases under the nominal scenario that has, for all practical purposes, about a two-minute, depending on the relative timing that we were talking about before lunch, kind of a couple-of-minute release time through that open control valve.

And I was curious whether there is an analysis of that type of scenario done. Because I don't know the relative times. I have no idea when the reactor trip is going to come in. I have no idea when the pressure in the ruptured steam generator is going to get high enough to pop open the relief isolation valve. And I have no idea what time the safety injection is going to occur. I don't do those

kinds of analyses.

Any thoughts?

MR. BERGERON: The case without CVCS --

MEMBER STETKAR: Yes.

MR. BERGERON: -- we have analyzed.

MEMBER STETKAR: You have?

MR. BERGERON: Yes.

MEMBER STETKAR: All right.

MR. BERGERON: We have looked at both with CVCS and without CVCS.

MEMBER STETKAR: I know it talks about it in the FSAR, but there's no results presented for it.

MR. BERGERON: Yes, because that case was less limiting.

MEMBER STETKAR: I would be interested to see --

MR. BERGERON: The timing of the trip is somewhere on the order of 900 seconds.

MEMBER STETKAR: Nine hundred seconds?

MR. BERGERON: I mean you get an automatic trip and --

MEMBER STETKAR: Yes, you think you would get the automatic trip pretty soon.

MR. BERGERON: Right.

MEMBER STETKAR: I was guessing, the

sooner it is, the worse you are --

MR. BERGERON: Right. I mean, as far as if you are going to wait 30 minutes to perform any manual action --

MEMBER STETKAR: Yes.

MR. BERGERON: -- correct.

MEMBER STETKAR: And that is a ground rule.

MR. BERGERON: Right.

MEMBER STETKAR: I mean the operators are hands-off until 30.

MR. BERGERON: Yes. Basically, we are hands-off for 30 minutes.

MEMBER STETKAR: I would be interested to see --

MR. BERGERON: We have looked at both cases.

MEMBER STETKAR: -- the results from that. Maybe, Derek, can you see if we can, if you have got it?

MR. BERGERON: Now what Ken was talking about on these --

MEMBER STETKAR: Unless there is an isolation signal that I couldn't find.

MR. BERGERON: Yes. I mean, if you look

at 15.08, Table 15.08 in the FSAR, there is a list of all the ESF functions. And among those functions are high-steam line activity with other signals that actually perform isolation functions.

MEMBER STETKAR: Table 15?

MR. BERGERON: Now in the analysis we have presented in the FSAR, we have not credited any automatic function that comes from steam line activity. We have only used the steam line activity as an indication that we have a tube rupture --

MEMBER STETKAR: Right.

MR. BERGERON: -- and which generator it is in.

MEMBER STETKAR: Yes.

MR. BERGERON: But there are automatic features on the U.S. EPR that perform isolation functions with a radiation signal in the steam line coincident with other signals. For example, if you have a partial cooldown initiated with the high activity, you would raise the setpoint in the MSRT.

MEMBER STETKAR: Do you have that table? I just pulled up the table. Where in the table is that isolation?

MR. BERGERON: It is on sheet 3 of 4.

MEMBER STETKAR: Sheet 3 of 4?

MR. BERGERON: Yes.

MEMBER STETKAR: Of four? Fifteen --

MR. BERGERON: 15.08.

MEMBER STETKAR: This is --

MS. SLOAN: Which revision do you have, John?

MEMBER STETKAR: Well, that could be the problem.

MR. BERGERON: That could be the problem.

MEMBER STETKAR: That could be the problem. Hold on. Yes, I have Rev. 2.

MS. SLOAN: Okay, yes.

MEMBER STETKAR: Because I only have three sheets in that table. All right.

MR. BERGERON: That function hasn't changed.

MEMBER STETKAR: Because my Table 15.08 only has --

MS. SLOAN: Paul is looking at our snapshot, our internal live version of the FSAR.

MR. BERGERON: Rev. 2 looks the same to me.

MEMBER STETKAR: 15.08, I have three sheets. Let me just make sure I have pulled up the right -- bear with me.

MR. BERGERON: Yes, it is under the category of steam generator isolation.

MEMBER STETKAR: I'm sorry. I have Rev. 1 of the DCD.

MS. SLOAN: Okay. Okay.

MR. BERGERON: That makes a difference.

MEMBER STETKAR: I have Rev. 1. I don't even have Rev. 2. How come we don't have Rev. 2?

MS. SLOAN: Rev. 2 is the latest docketed version.

MR. WIDMAYER: When was that docketed?

MR. TESFAYE: On August 31st.

MS. SLOAN: Yes.

MEMBER STETKAR: Maybe I lost mine.

MEMBER SHACK: I don't have one.

MEMBER STETKAR: If he doesn't have it, we don't have it. I trust Bill. Bill doesn't lose anything.

(Laughter.)

MR. BERGERON: Is that what Rev. 1 looked like?

MEMBER STETKAR: I swear to you, Rev. 1 -- I have got two. I have got Rev. 0 and Rev. 1. I thought they were 1 and 2.

MR. BERGERON: I mean that wasn't a

feature that was added, so it should have been there.

MEMBER STETKAR: The only one I could see was to raise the setpoint on the high steam line, but that is not an isolation.

MR. BERGERON: Right.

MEMBER STETKAR: I mean, you know, your pressure will eventually get to that higher setpoint. Well, that's okay. If we don't --

MR. BERGERON: Yes, the MSIV closure on high steam line with partial cooldown initiated is also an isolation feature.

MEMBER STETKAR: Yes, but that is just MSIV. That is not going to help me with the MSRs, the control valves.

MR. BERGERON: Right. Right. Yes, the setpoint has been reset on the MSRTs, so that is automatic.

MEMBER STETKAR: Yes, but that is just a higher setpoint, but I have nothing --

MR. BERGERON: If you pop it and it sticks open, then you have to wait until you get to the IV setpoint to close.

MEMBER STETKAR: And it is exacerbated by the fact that, if I get an SI, there is another signal coming in trying to drive the whole thing open, also.

MR. BERGERON: Right. Right.

MEMBER STETKAR: I mean, if it is stuck open, it doesn't make any difference. I was trying to get the single failure as just the charging pump failure and let auto take care of opening that valve.

MR. BERGERON: Yes. No, but because the CVCS is not a safety system --

MEMBER STETKAR: You can assume it has failed?

MR. BERGERON: -- we looked at both with and without.

MEMBER STETKAR: I would really like to see --

MR. BERGERON: And it turned out, I mean this was based on offsite doses.

MEMBER STETKAR: Yes.

MR. BERGERON: And it turns out that it was higher because you are keeping the pressure up on the primary and driving the leak, primary to secondary. So, you are transferring more over to the secondary side for it to be released.

MEMBER STETKAR: For the case of record?

MR. BERGERON: For the case, yes.

MR. PARECE: But the thing to remember -- this is Marty Parece -- the thing to remember about

the steam generators when it comes to dose is that they are an iodine battery. So, if you don't steam it for 10 minutes and then you lift the valves for one minute, you might as well have steamed it all 10 minutes. It is the same, the same thing.

MEMBER STETKAR: That might be the --

MR. PARECE: Okay. It acts like a battery because the half-life of the iodine is so long.

MEMBER STETKAR: That might be the answer. I was just curious to see that --

MR. PARECE: So, if you have that drives the leak harder, even if there is some difference in how many minutes the total steaming is, if you have one that drives the leak harder, you drive more iodine into the generator. So, when it does lift, it just vents everything.

MEMBER STETKAR: I have to think about it. Make sure we get Rev. 2, also.

Thank you.

MR. COFFEY: Any other questions?

(No response.)

Thank you very much.

CHAIRMAN BANERJEE: Okay. Thank you.

I guess now we are going to have a change. The staff will come on.

So, we don't visit this book again until tomorrow.

MS. SLOAN: Beg your pardon?

CHAIRMAN BANERJEE: We will reopen this tomorrow.

MS. SLOAN: Okay. Yes.

MR. TESFAYE: Are you ready for us?

CHAIRMAN BANERJEE: Yes, I think we can start.

MR. TESFAYE: Yes, Dr. Banerjee, Jason Carneal is the Chapter PM. He will be coordinating the staff's presentation.

CHAIRMAN BANERJEE: Okay.

MR. CARNEAL: Good afternoon.

My name is Jason Carneal. A little background: I received a bachelor's and master's in engineering science mechanics from Virginia Tech in 2001 and 2004. For four years, before coming to the NRC, I worked at the Naval Surface Warfare Center, Carderock Division, performing hydrodynamic experimentation on hull forms and propulsors.

And since coming to the NRC, I have served as Chapter PM for Chapters 4, 6, 15, and 17 for the U.S. EPR design certification application review.

Today's presentation is going to cover the

staff's Safety Evaluation Report with Open Items for Chapter 15, Group 1, of the U.S. EPR design certification application.

The technical staff that supported this review includes members from the Reactor Systems, Nuclear Performance, and Code Review Branch. They handled the majority of the review. For Section 15.03, the review was handled by members from the Reactor Siting and Accident Consequence Branch and the Component Integrity Branch.

A quick overview of the results of our review: in Section 15.0.1 and 15.0.2, we issued a total of 10 questions, and there are three open items remaining, mostly dealing with the assumptions used in the analysis.

In Section 15.0.3, Radiological Consequences, we asked 38 questions. There were two open items in the SER, but since issuance of the SER, those two open items have been closed.

The remainder of the sections on this table, there are no open items with the exception of 15.4, Reactivity and Power Distribution Anomalies.

Again, no other open items in the remaining sections. All tolled, we have issued 131 RAIs in this review with six open items documented in

the SER and four of those items remain.

Just a quick note that the Phase 2 Safety Evaluation Report with Open Items for Section 15.6.5 was not delivered with Group 1 and, as discussed earlier, we are working to deliver that in the late March or early April timeframe.

A quick description of the open items that remain open in the review:

RAI 415, Question 15-9, is focusing on site-specific EOPs to confirm that the assumptions used in the Safety Analysis remain valid.

We have a question in RAI 311, Question 16-317, that is involved with technical specifications. That is a cross-cutting issue, and we are working with the Technical Specifications Branch on that issue.

We also have an open item concerning the .48 percent total power measurement uncertainty claimed by the applicant, including a description of how it will be verified and confirmed by the COL applicant.

And in Section 15.4.8, we have an open item that was also cross-listed in the Chapter 4 review that tracks the review of Topical Report ANP-10286, the Rod Ejection Accident Methodology

Topical Report.

We have closed two open items in Section 15.00.03. These open items were largely editorial in nature, but it was requesting, the first one was requesting that tables be added to the FSAR to demonstrate that the pH would be acceptable after the 30-day timeframe.

The second one was requesting that the applicant include purity and density values for the commercially-purchased TSP and the FSAR.

Both of these items have been responded to with FSAR markups and RAI responses, and the item is now considered confirmatory by the staff.

And before we move on into Shanlai's presentation, there were a couple of discussion items earlier, Section 15.0.3. The staff did not have any dedicated slides on that section, but we had a discussion topic on confirmatory analysis and use of the ARCON-96 code. So, I would like to open up the floor for specific questions on Section 15.0.3, and we also have Michelle Hart to support this discussion.

MS. HART: I'm Michelle Hart from the Siting and Consequences Branch.

I understand earlier you had a question about whether we had done confirmatory analysis for

the Radiological Consequences Analysis. And, yes, we have. We have used our code RADTRAD, and we used the assumptions and inputs that the applicant gave us as long as they went along with our Regulatory Guide and our SRP, and they all have.

So, we did not model like the deposition and the containment differently or anything like that. We used the values that they used once we determined that there were acceptable, that the models were acceptable.

All of our results came out to show that their results were reasonable and acceptable. And as the applicant had stated, some of their analyses were back-calculated to give them the highest chi over Q that they could, so they could put it on the most sites that they could.

Did you have any further specific questions about that admittedly short overview?

CHAIRMAN BANERJEE: No. I think we just wanted to know that you had done some confirmatory calculations.

MEMBER SHACK: Well, Professor Banerjee was asking about what was done to validate the ARCON-96. You know, what is the range of geometries for which you would consider it valid?

MS. HART: Well, one of the first things to recognize is that that's reviewed more specifically in the context of the meteorological review which is done in Chapter 2. So, I personally had not done that. The reviewer that did that review is not here.

The models are not specific enough, is my understanding, to have to validate them in that sense.

Are there any specific questions?

I do have a meteorologist who can talk in generalities, but cannot talk about EPR reviews.

CHAIRMAN BANERJEE: I guess the question applied to the applicability of the code to this specific situation where it had been, you know, validated in some sense. Why didn't our geometries with, you know, conditions, and so on, to cover this as a subset of validation.

MS. HART: Right. We are talking about the releases from the containment of the air. It is not from outside --

CHAIRMAN BANERJEE: Right.

MS. HART: -- in this specific case, and it is for the control room habitability.

CHAIRMAN BANERJEE: So, from the outside is left for the COLA, whatever comes?

MS. HART: Well, yes, we also have to

recognize that this is for the design --

CHAIRMAN BANERJEE: Site-specific.

MS. HART: -- certification.

CHAIRMAN BANERJEE: Yes.

MS. HART: And so, there is a certain amount of back-calculation that can even be done in that case. So, when you come in as a COL applicant, you would to show specifically the use for your site.

Now is it true that everyone uses ARCON-96, the staff, the industry. For the purposes of the dose analysis in the control room, yes, they used ARCON-96.

My understanding is it is not a particularly detailed model. So that, it is not a 3D model.

CHAIRMAN BANERJEE: No.

MS. HART: It is not like a CFD or anything like that.

CHAIRMAN BANERJEE: That is fine. But it is, therefore, conservative.

MS. HART: Right.

CHAIRMAN BANERJEE: Yes.

MS. HART: Right.

CHAIRMAN BANERJEE: It takes into account some mixing, and so on, but not pushing it to the

limits.

MS. HART: It is not pushing it to the limits.

CHAIRMAN BANERJEE: Right.

MS. HART: Right. And if there were specific questions about ARCON-96, it would have to be, you know, discussed with the developer of the code. Of course, `96 is the year that it came out, approximately, and it was based on an earlier version. These models are nothing specifically new. They have been used in the industry for many, many years.

CHAIRMAN BANERJEE: I think it is a question of where we put our attention, and probably this is not going to be one of them.

MEMBER SHACK: There is enough conservatism built into the --

CHAIRMAN BANERJEE: Yes, that is probably good enough. Yes.

For different types of non-radiological release, of course, that is a whole new ballgame because --

MS. HART: Right. ARCON-96 is not used for those analyses.

CHAIRMAN BANERJEE: Not yet, yes. So, you would use, somebody would use ALOHA or something like

that.

MS. HART: ALOHA has been used. Of course, we have HABIT, and I know that that is a topic that will be discussed with the full Committee --

CHAIRMAN BANERJEE: Right.

MS. HART: -- coming up on the 10th in the context of the summer COL applications. So, specific questions may be more appropriate.

CHAIRMAN BANERJEE: Okay. So, I think we can just defer it to the full Committee meeting --

MS. HART: Okay.

CHAIRMAN BANERJEE: -- right now and take it up there.

MS. HART: Did I answer the questions closely enough?

CHAIRMAN BANERJEE: Yes, I think that is fine for now.

MS. HART: Thank you.

CHAIRMAN BANERJEE: Okay.

MR. CARNEAL: Okay. So, if there are no other questions on that topic, I will turn the presentation over to Shanlai Lu.

CHAIRMAN BANERJEE: Shanlai, of course, escaped from --

MR. LU: From GSI-191.

CHAIRMAN BANERJEE: Yes.

(Laughter.)

Well, I hope he is going to look at GSI-191.

MR. LU: Yes, we are. We are. We are coming back on that one.

CHAIRMAN BANERJEE: All right.

MR. LU: Yes. So, my name is Shanlai Lu. I am the lead reviewer on U.S. EPR from the Reactor Systems Branch.

And in terms of a brief bio, I have been with the agency for 11 years and Research four and a half years, and joined NRR I think back in 2002, and then I worked in Power Uprate. I later worked with Mike Scott on the GSI-191.

Then, I thought the strainer issue could be resolved already back to 2007. So, I decided to join NRO to work on the EPR. So, ever since then, I have been working on the EPR.

So, as Jason mentioned, we issued 131 RAIs and related to these non-LOCA sections. And 121 were issued by the Reactor Systems Branch. So, we did a lot of work in this area.

Then, back to three years ago, we launched the audit to analyze this. A lot of work was done

since then.

So, when we were preparing the slides, developing the presentations, we decided to take an approach to give you first the punch line. What exactly are the results of our Phase 2 review?

Then, we will walk you through a detailed evaluation of the SPND, in-core SPND, using our own codes, which is not intended to validate AREVA's design. It is just for the staff to understand how the SPND works and what is the in-core setpoint methodology, the algorithm, how it really works. So that we can launch RAIs.

So, the purpose of that calculation was to help staff to understand the system and generate useful information, so that we can launch/write RAIs. So, therefore, I am going to go into the first presentation about the punchline, Phase 2 review/results. Okay?

We had a big team working on this because a significant amount of work during the reviews since three years ago. And Chu-yu Liang, actually, he was the staff working in this area, and then he was working on this until September. Last year he retired. So, as the lead, I fill in just to present his presentation.

We have Jack Rosenthal as our consultant; Dave Caraher and Vesselin Palazov. And Jack's Technical Evaluation Report is in such a good quality, when we converted that into the staff Draft SER, OGC just told us there is no need for a Chapter meeting.

CHAIRMAN BANERJEE: My thing seems to be missing some slides.

MR. LU: Okay.

MR. WIDMAYER: You gave me corrected --

MR. LU: Okay, this is just a name.

CHAIRMAN BANERJEE: Yes, it is okay, but --

MR. WIDMAYER: Is that the only thing that is different, is the names?

MR. LU: Just the names. I think we added the names there. Make sure.

MR. CARNEAL: And in the second presentation, there is a table that is modified to add a couple of values.

MR. LU: All right. The next slide, this basically is a summary. So, what have we really reviewed as part of Chapter 15 non-LOCA sections? Basically, it goes through that form of 15.1, increase in heat removal by the secondary systems; 15.2, decrease in heat removal by the secondary systems.

All those were detailed or the descriptions were given by Doug Brownson this morning and very detailed. So, I do not intend to go through every single one of them and give you the examples.

So, the 15.3, decrease in reactor coolant system flow rate, and 15.4 will be presented tomorrow with the rod ejection topical report together, because that is coupled and that is the direct application of the topical report.

And the increase in reactor coolant inventory, 15.5; decrease in reactor coolant inventory, 15.6, and ATWS, 15.8.

And one thing very good and that helped the staff a lot was three years ago when we received a submittal. The submittal was in such good shape and high quality. So, it matched to this SRP very well. So, it saved us a lot of time to review that part.

Okay, let me move on. Okay.

All right. High-level, the acceptance criteria we used to evaluate each event under a different category, and then for AOOs, the events expected to occur during the lifetime of the plant. All those events are expected to happen during the normal operations, some of them in shutdown mode.

And the acceptance criteria here are

documenting the SRP and the pressures in the primary and the secondary systems not exceeding 110 percent of the design value. All those criteria are not unique to EPR. It is all across the board for the operating fleet and all the new reactor designs.

CHAIRMAN BANERJEE: Could you tell me why you have put 3 to 10 as your slide numbers, just to confuse us or?

MR. TESFAYE: The numbering is wrong.

MR. LU: Oh, okay.

CHAIRMAN BANERJEE: Only the numbering is wrong?

MR. TESFAYE: Yes.

CHAIRMAN BANERJEE: Okay. Nothing else?

MR. TESFAYE: There are no missing slides.

CHAIRMAN BANERJEE: There are no missing slides.

MR. LU: Well, we cut and pasted it, that slide. Okay?

CHAIRMAN BANERJEE: So, there were slides in between?

MEMBER SHACK: Well, yours is 3 to 10. Mine is 3 to 9.

(Laughter.)

CHAIRMAN BANERJEE: No, I have a new one

that is reflecting that. Okay. All right.

MR. LU: That is maybe at the last minute we were shuffling the slides and the page number may be off there.

Okay. The second criteria we used is the fuel cladding integrity shall be maintained by no DNBR during the transient.

And talking about DNBR, I think in the morning AREVA mentioned about the in-core trip system to protect most of the transient with DNBR trips and LPD trips. I think that is the methodology and, also, the design became the focus of our review for most of our transient because they did take the credit of the existence of such a system.

For AOOs, the third criteria is they should not generate a postulated accident. Those are standard criteria for all the AOOs.

Okay. Continuing on for the postulated accidents, events are not expected to occur during the lifetime of the plant design. However, it is postulated accidents.

And, then, also, we have three criteria. So, the pressure in primary and secondary systems not exceeding acceptable design limits, considering potential brittle as well as ductile failures. For

example, you could have cold water injection into the vessel and watch the PT curve. It needs to be evaluated there, too.

And the fuel failure may occur, but not result in offsite doses exceeding 10 CFR 100 limits. For example, rod ejection and, then, there is a possibility you have a certain percentage of rods fail. But the dose rate cannot exceed the limit.

Okay. A postulated accident cannot be developed in such a fashion that there is a consequential loss of function of systems needed for mitigation of the event. So, that is related to long-term mitigation there.

Okay. When we got this FSAR, and we found a lot of the information when we launched our audit back three years ago, and we found even more information in AREVA's site over in Lynchburg. So, we launched a one-week audit, and we found that for each transient, as Doug described in the morning, there are at least 40 or 30 different cases.

To answer John's question, in a lot of cases we are actually there, but we cannot just remember details over what is a trait and what is exactly the phenomenon there. There is tons of it because we have several cabinets of the information.

So, we have 21 staff and they stayed there for one week, and the AREVA site had 45 people supporting our review, this part of it.

So, based on that audit, we launched our RAIs and then started a review process, Phase 1 and Phase 2 review, until today.

CHAIRMAN BANERJEE: So, you looked at the methodologies and the codes, and most of these or all of these had already been --

MR. LU: Approved.

CHAIRMAN BANERJEE: -- accepted or approved?

MR. LU: That's right. That's right. We, first, verified our analysis to make sure that all the analysis is documented in the record, based on the methodology approved or reviewed by staff before.

CHAIRMAN BANERJEE: Right. Now how did you determine the applicability to the U.S. EPR to that?

MR. LU: Okay. I think of that as sort of the typical question about that is we take a few limiting cases, a special limited case, and to verify what's the upper-bound and lower-bound parameters, especially some key parameters, but we cannot go over every single correlation, for example, of S-RELAP5,

and then verify every single correlation is valid because that has been done as part of the review of the methodology.

CHAIRMAN BANERJEE: So, S-RELAP was approved before my time. Bill may remember. What was it applicable to? I mean I was a consultant on it.

MEMBER SHACK: Right. It was their large-break realistic LOCA analysis.

CHAIRMAN BANERJEE: But, I mean, it was for the --

MEMBER SHACK: Operating plants, yes.

CHAIRMAN BANERJEE: -- operating plants, right?

MEMBER SHACK: Right.

CHAIRMAN BANERJEE: So, this has some special features, clearly, compared to the operating plants, including not having a high-pressure ECCS system.

MR. LU: Yes.

MEMBER SHACK: So, it is a big change.

MR. LU: Yes. As a matter of fact, I think that we are trying to schedule in March, in the March ACRS meeting, talking about large-break LOCA for the EPR application using S-RELAP5.

But this part, the S-RELAP5 was approved

for transient analysis.

CHAIRMAN BANERJEE: Right.

MR. LU: So, for AOOs and postulated accidents.

CHAIRMAN BANERJEE: And it has also been approved for small-break LOCAs, I take it?

MR. LU: That's right.

CHAIRMAN BANERJEE: Or that is still open right now?

MR. LU: Yes, you're right. And that is the reason I think you asked for the application of that approval for the EPR small-break LOCA analysis, and I think AREVA is going to provide a presentation tomorrow specifically on a certain aspect. It is not the entire methodology.

CHAIRMAN BANERJEE: The staff is still considering the matter?

MR. LU: That specific application is still being considered, but the small-break LOCA methodology using S-RELAP5 was approved as part of the applicability report.

MR. TESFAYE: Yes, there was a topical report.

MR. LU: There was a topical report approved before even I joined NRO by, I think at that

time it was by the Reactor Systems Branch, NRR.

MEMBER SHACK: I mean that is standard for all these, is you have to have an applicability report.

MR. LU: Exactly.

CHAIRMAN BANERJEE: Yes, but applicability, for example, for other systems, we required, for example -- just take an example of TRACE, for example. We reviewed TRACE and the staff, of course, developed it. So, we looked at it, but it was applicable to, say, ESBWR or to some specific system for a certain range of things, not for ATWS, but for --

MR. LU: Right. I think for specific U.S. EPR, the TRACE was not submitted by the applicant as a licensing basis. And we did use TRACE to get some results for us, generated the useful information for us to launch RAIs.

CHAIRMAN BANERJEE: Okay. Well, we just need to get a picture of where we stand.

MR. LU: Okay.

CHAIRMAN BANERJEE: I can see that what you are saying here is for a very right now rather limited use of S-RELAP5 for non-LOCA systems --

MR. LU: That's right.

CHAIRMAN BANERJEE: -- in the way that was sketched this morning, right, to us?

MR. LU: Right.

CHAIRMAN BANERJEE: For using it for the DNBR and the other things?

MR. LU: That's right. That's right.

CHAIRMAN BANERJEE: All right.

MR. LU: That is our basis. I think it is not only S-RELAP5. I think there is LYNXT.

CHAIRMAN BANERJEE: Oh, there is a whole bunch of --

MR. LU: We talk about the codes in there. They use it to generate the curves and, then, put it into the setpoint, I think, an in-core algorithm, burnt into chips, supposedly, down the road.

MR. TESFAYE: Dr. Banerjee, before even the application came in, there was a codes and methods topical report that was submitted by AREVA and reviewed and approved by the staff.

CHAIRMAN BANERJEE: Okay. And this was basically things they had used. It was mainly an applicability report?

MR. TESFAYE: Yes, for EPR.

MR. LU: For EPR, yes. But I think the NRR staff's conclusion is they consider they are

acceptable --

CHAIRMAN BANERJEE: Right.

MR. LU: -- to be used for AOO and postulated accident except LOCA. Small-break LOCA was included.

CHAIRMAN BANERJEE: Was included?

MR. LU: Was included as a part of applicability, but we did do additional review as part of the design cert of 15.6.5. So, by the time after the March presentation, once we are done with that one, we plan to give you a presentation on how we review that part.

CHAIRMAN BANERJEE: Okay. As long as we get a chance to look at it.

MR. LU: That's no problem.

CHAIRMAN BANERJEE: Right. Okay.

MR. LU: All right. The second bullet --

MR. TESFAYE: Excuse me. We can send you, I can point out the Safety Evaluation Report that we wrote on those codes and methods topical report. We can make that available before the next --

MR. LU: Before even the ACRS meeting, yes.

CHAIRMAN BANERJEE: Yes, that would be helpful.

MR. TESFAYE: Okay. I will take action to get you those copies.

MR. LU: Okay. So, the first item is the methodologies and the computer codes. And the second one, we did do the review of limiting conditions of operation which provided a bounding or box for Chapter 15 AOO and the postulated accident analysis. So, we want to make sure what they used in the analysis was consistent with Chapter 15.1. This included the initial conditions and configurations, and all --

CHAIRMAN BANERJEE: So, what was the protocol here, whoever wants to answer it? There are certain things that we felt we want to be able to have comments on because we might have a point of view. So, clearly, for small-break LOCA, we are quite interested in this problem. Maybe for large-break LOCAs, we are not even so interested.

But it can be that we should -- what has been followed with EPR up to now? Because with, say, some of the other certifications we have been involved with, we have reviewed, at the request of the staff, some of these codes before they were accepted or approved, at least some of the most crucial ones, you know.

MR. TESFAYE: There is no set protocol.

CHAIRMAN BANERJEE: Right.

MR. TESFAYE: The procedure doesn't require us to bring topical reports to ACRS. But what we do is, for the initial topical reports, we intended to bring them with the chapters that they were referred to. That would be the case with the codes and methods topical report. I think there was some other presentation, also, that was done by AREVA to discuss the codes and methods topical --

MR. WIDMAYER: November of 2009.

MS. SLOAN: Yes, we have presented to ACRS before, AREVA has, on our codes and methods applicability.

MR. TESFAYE: Jack, do you want to talk?

MR. ROSENTHAL: Yes, please.

CHAIRMAN BANERJEE: You have to identify yourself, Jack. Speak at a microphone.

MR. ROSENTHAL: Jack Rosenthal. I work for ISL as a consultant.

And I just wanted to call out that there is an AREVA codes and methods applicability report for the U.S. EPR, ANP-10263P-A, August 2007. It was both proprietary and it was previously approved. That was the basis for our review work in which the codes and methods were discussed.

An example is that the EPR steam generators have an economizer, and there were questions about the modeling of that economizer in RELAP. And so, you will find in our SER some questions and responses that are applicable.

MR. TESFAYE: Okay. Thank you.

And again, Sandra, he reminded me I think that was when we brought unique features of EPR. We did make a presentation to this Subcommittee. I think it was to the full Committee.

MR. WIDMAYER: No, it was this Subcommittee. November of 2009.

MR. TESFAYE: Yes.

MR. WIDMAYER: Yes. But that also postdated the SER, right? I mean it was just information. You could ask any questions you wanted.

MEMBER SHACK: Well, I mean, the only one that I can remember doing in great detail was the containment response. That one was sort of up in the air, and we have never come back to it.

MR. TESFAYE: We will come back to it when we get --

MEMBER SHACK: I assume we will come back to it for the large-break LOCA, yes.

MR. TESFAYE: Yes, it is coming soon.

CHAIRMAN BANERJEE: Yes, there can be different codes for different things, right, that we want to look at?

MR. TESFAYE: Yes, that was a special presentation.

MEMBER SHACK: That was a special presentation, yes.

MR. TESFAYE: Yes, it was different. But we had unique EPR features presentation made to this Subcommittee, I guess, and as part of that, I think we did discuss the codes and methods topical report. And I don't bring it up every time we discuss every chapter and every section, like we are doing now.

CHAIRMAN BANERJEE: Right. So, I am just trying to get my head around this because we have been running so many of these in parallel, you know, these certifications. I am just looking for some consistency between how we did this.

For example, in some cases, we have to do TRACG applicability. We looked at in some depth, how it treated non-condensables and all this sort of stuff.

So, in the case of the EPR, what you are saying is you have made a general presentation to the Subcommittee about codes and methods. I am just

giving it back to you as I heard it. As each chapter comes up, depending on what codes and methods are being used, you are sort of revisiting these. And the ones which are probably of most interest to us, we would identify and say we want to look in a little bit more detail here. That is the sort of methodology that has been followed here.

MR. TESFAYE: Yes.

CHAIRMAN BANERJEE: Okay.

MR. TESFAYE: And for the topical reports, they have been resubmitted recently. I think we have made separate presentations, like we are doing tomorrow morning on rod ejection topical report. We will bring it up in that section. We are going to do the same thing on mechanical fuel design topical report. So, we are bringing in those topical reports when we discuss the chapters.

CHAIRMAN BANERJEE: Right. And we get a chance to review those, if you wish to, or we can ask you, that we would like to at that point.

MR. TESFAYE: Yes, all those topical reports are incorporated by reference to the FSAR. So, they are of the FSAR.

CHAIRMAN BANERJEE: Yes. I have the rod ejection one, right. Okay. Okay. Thanks.

MR. CARNEAL: And looking ahead for the large-break LOCA, ANP-10278, I think we are having a dedicated meeting ahead of the Subcommittee meeting for Chapter 15, Group 2, which is Section 15.6.5. So, we will have a chance to interact before.

CHAIRMAN BANERJEE: And that will discuss containment response and everything.

MR. CARNEAL: Right.

CHAIRMAN BANERJEE: Okay. Go ahead, Shanlai.

MR. LU: If you want to make more comments, that is fine.

MR. CARNEAL: The containment response is going to be part of the Chapter 6 SER that we are giving because it is covered in a technical report in support of the FSAR. So, that should be coming to you soon.

CHAIRMAN BANERJEE: I'm sure Dana has got everything computerized in his mind.

MR. LU: All right. Continuing on with major review areas, we also reviewed the worst single failure, assuming that each event for all those cases analyzed, and we identified and evaluated whether the single failure was being assumed to give you the most challenging case.

So, all the cases on the category of 15.1, 15.8, all of them, any cases related to the single failure assumption, we evaluated the worst case. And, then, as part of it, we also evaluated whether they considered it with or without offsite power.

Okay. That is very high-level. I will keep going now.

The sequence of the events analyzed, we also evaluated to make sure that they assumed operator actions, are reasonably achievable, and although we have an open item related to the EOPs, and I am going to talk about it right away after this one.

And, then, we also reviewed the analysis, including transient predictions, to assure that the consequence of each analyzed limiting case meet the specific acceptance criteria. Different cases may be limited by DNBR. Some are limited by the pressure. Some are limited by the RPD.

Okay. So, that is the different regulatory requirements there, too. Okay.

So, how we approach this, we launched many, many audits. And, then, as I mentioned right in the beginning, we launched a big audit three years ago, and after that, we reviewed their calculation onsite and, then, we also launched our own

confirmatory calculation using many, many different codes. Some codes are our codes developed by the Office of Research. Some codes, we even use AREVA's codes to change the deck and the performance sensitivity of the cases.

So, in this way, the confirmatory analyses are used to generate useful information, so that we can make sure we ask the right RAIs, instead of many numbers of RAIs.

MEMBER SHACK: Do you have access to all of AREVA's codes? I mean, if you want to --

MR. LU: AREVA is very cooperative. Whenever we ask them for codes, they say, "Okay, here you go. Run it."

And, then, actually, I am going to give you the examples of lots of codes we ran, as RELAP5 we actually ran, both transient and LOCA. Okay?

All right. That is the overall approach. Okay.

CHAIRMAN BANERJEE: Just let me ask AREVA, do you have a formalized V&V process in-house?

MS. SLOAN: Yes. Absolutely. Yes. Rigorous procedures written, and I think the staff in previous reviews has looked at our V&V procedures, absolutely. A whole software configuration management

program to go along with that, yes.

CHAIRMAN BANERJEE: Okay. And you have reviewed that?

MR. LU: Yes. Now, actually, right before we got into the official DCD review, we performed a QA review. As part of it, we reviewed that QA procedure, too. That is back to 2007, I think October 2007, pre-submittal, QA audit.

CHAIRMAN BANERJEE: So, for example, TRACE has a bunch of -- and maybe somebody here can comment on this from staff -- but they have a standard set of, let's say, benchmark data or analyses or whatever. So that every time some change is made in a quote, it has to be, you know, passed through this --

MR. LU: Right.

CHAIRMAN BANERJEE: -- sort of validation process.

MR. LU: Right.

CHAIRMAN BANERJEE: Does AREVA have some equivalent dataset against which they validate codes, like S-RELAP or whatever?

MR. LU: Okay. As part of a large-break LOCA, topical report review and the presentation we are going to give to you. And actually, we will tell you how we reviewed that part specifically. I think

it will be in the March timeframe.

But to specifically answer your question, we did learn a lot through our own TRACE experience, code development experience and validation and verification. And the staff has experience with that process. So, when we reviewed that, we actually reviewed similar process. We tried to identify if we have any questions here, too.

So, as a part of the S-RELAP5 LOCA topical report review, we did have a lot of thorough review of the code versions, updated the QA process, and validation and verification.

CHAIRMAN BANERJEE: And not just large-break LOCA, but --

MR. LU: No. Oh, yes.

CHAIRMAN BANERJEE: But also some small-break and --

MR. LU: Yes. We do. We do that.

CHAIRMAN BANERJEE: -- other things? Okay.

MR. LU: That is our job.

CHAIRMAN BANERJEE: Okay.

MR. LU: All right. Let me talk about the open items we have. We have three open items, and all of them are not unique to this design. It is just a

process, and then once you have the design cert, then there is always certain information cannot be developed as part of a DCD. So, this is not unique to EPR, and all the new designs may have similar issues.

For example, the first one is the open item related to the emergency operating procedures. If you go through the FSAR, 15.0.8, something like that, there is a list of the operator actions credited as part of the AOO analysis.

All those need to be verified through the COL action item and validated, the safety analysis assumptions used in Chapter 15 non-LOCA analysis, too.

The example here is 15.0.0.3.7, lists all those operator actions. And the examples can be feedwater line break, main steam line break, steam generator tube rupture. We just talked about that. And ECCS switchover during LOCA long-term cooling in design, like, for example, from cold leg to hot leg switchover.

So, this is an open item, the EOP.

CHAIRMAN BANERJEE: They are going to do this realignment switchover at some time during the long-term cooling phase?

MR. LU: That is being developed. It is not finalized yet, because it is specifically related

to the downstream effects; it has not been finalized yet. We may be able to talk about it during our presentation on 15.6.5 down the road sometime, but not now.

CHAIRMAN BANERJEE: Okay.

MR. LU: Okay. Let me see. Okay. At this point, there is no specific guidelines for the required verification and validation of EOPs to confirm operator actions credited in the safety analysis are achievable. And therefore, as part of a COL information item, we asked AREVA to provide this and, then, describe how they are going to collect all this information and develop this information as part of COL. And, then, what's the plan?

So, down the road, we would expect that the once COL applicant takes over, we need to work with AREVA to develop these EOPs and to verify the developed EOPs can support the Chapter 15 AOO analysis.

Okay. I think that is the last bullet related to the EOP verification.

Power measurement uncertainties, I think we discussed this morning.

CHAIRMAN BANERJEE: It's open.

MR. LU: Yes, it's open. And I think the

reason is very simple. I think when we saw .48 percent power measurement uncertainties, it was very small. So, we asked, what's the basis? We issued RAIs. And actually, AREVA provided two runs of RAI responses already. The first run was provided as part of a large-break LOCA topical report and told us the ultrasonic flow meter will be used.

And the second round of RAI responses was provided as part of 15.6.5 and told us what's the methodology they used, how did they do the uncertainty analysis, and came out with this number of 0.48. And we reviewed the RAI responses. We found they did have a good methodology which has been approved before by NRR staff.

However, the uncertainty input to that methodology, most of them are assumed; at the DCD stage, everybody would assume all those uncertainties because the equipment has not been ordered or a specification has not been given. So, therefore, there is no way to say, okay, down the road it will be 0.48 uncertainty.

So, from that perspective, we decided to launch this RAI and to request additional information from them to verify what is the mechanism for them to achieve this .48 percent, which they have already told

us in the first round of RAI responses.

CHAIRMAN BANERJEE: But the other thing to remember, Shanlai, here is that, of course, the analysis is the initial conditions for the LOCA, right?

MR. LU: That's right. Actually, not only LOCA, but for all EOOs.

CHAIRMAN BANERJEE: Yes. There are a number of other things.

MR. LU: Yes. That is the most important parameters for determining the starting point.

CHAIRMAN BANERJEE: Yes. And we have been through a very long discussion on this in a very much more applied situation where somebody is building a plant and they want to do this. We should seek consistency in some way because --

MR. LU: That's right.

CHAIRMAN BANERJEE: -- there is a history here --

MR. LU: That's right.

CHAIRMAN BANERJEE: -- which we have been through.

MR. LU: Yes. One of the reasons we issued this RAI was we defined this as an open item because we learned from the AP1000 presentation your

viewpoint, and we are considering it as part of an EPR review, too.

CHAIRMAN BANERJEE: Right.

MR. LU: So, that is the reason we also have this open item, too. So, it is not unique to EPR. It is for the AP1000 and other reactors using ultrasonic flow meters.

CHAIRMAN BANERJEE: Yes, we are not talking about MURs here. We are talking about a new reactor.

MR. LU: Right.

CHAIRMAN BANERJEE: Yes. Okay.

MR. LU: All right. There is a lot of technical specifications defined and reviewed, defined by AREVA and are currently under review by staff, particularly the Specification Branch. However, those technical specifications right now establish a boundary or an initial condition for an EPR AOO analysis or postulated accident analysis.

Therefore, we would like to have AREVA to provide the link between the technical specifications to the Chapter 15 analysis assumptions. That is what this open item is about.

Okay. All right. Those are three open items we have. Of course, right now we are just in

the Phase 2. This is a snapshot of the results we had at this point. Any new information will prompt staff to review or additional new information, where these conclusions may be subject to change.

And since the ACRS has brought up the steam generator tube rupture event, and I think we had a lot of discussion on this already, and to give you a high-level aspect of that one not related to the dose. From the reactor system side, we have been focusing on three aspects, the thermal hydraulic aspect.

No. 1, to maximize break flow from one tube from the primary side to the secondary side, which provides a basis for the dose calculation. Then, we review that part.

We also reviewed a technical specification coolant activity, and, then, the operator actions beyond 30 minutes. And what is the offset power availability or not? And we reviewed that part, too.

We reviewed automatic secondary side partial cooldown, automatic emergency feedwater injection, and manual EBS. And we also particularly with this methodology, with this particular event, we reviewed the S-RELAP5 with its application to this steam generator tube rupture event.

And what we found was, to mitigate this

event, it requires quite significant operator actions, which were tied to one of the open items we have right at the beginning, the EOPs. So, this part of the closure needs to be done as part of COL, even for the tube rupture event.

Okay. So, that is a high-level summary. The punchline of the staff's Phase 2 review on the EPR non-LOCA events.

And, then, at this point, we finished 15.0, 15.1, 15.2, .3, .5, .6, and .8 review. We launched a total of 131 RAIs, and there are four open items remaining. One more from the reactor system will be discussed tomorrow to track the rod ejection topical report.

And we did perform a lot of confirmatory analysis with the support from the Office of Research. We got a lot of help from them. Since even before I started to work on the EPR, they were developing the base system models.

So, those confirmatory analyses did help staff to develop RAIs.

CHAIRMAN BANERJEE: So, did the Office of Research primarily use TRACE for this work?

MR. LU: Okay. Yes, at this point, the Office of Research's product to us is primarily based

on TRACE for AOOs and postulated accidents because it has the 3D kinetics capability, and RELAP5, RELAP I think Mod 3 does not have that 3D kinetics capability. Therefore, we relied on the TRACE part, coupled 3D kinetics, to perform the transient analysis.

CHAIRMAN BANERJEE: So, have we seen a TRACE/PARCS applicability report for EPR? Is Chris Hoxie here?

All right. We will defer that question.

MR. BAJOREK: Sanjoy, this is Steve Bajorek.

CHAIRMAN BANERJEE: Go ahead.

MR. BAJOREK: Yes, we have done a TRACE applicability report. This was put together about the same time we were putting together the model. As part of that, we went through, did like a PIRT, compared what TRACE had been approved for or basically developed for compared to the EPR, looked at the differences in small-break. Large-break I don't recall there was much there.

Then, we accompanied our normal set of what I would call generic assessment with things that focused on phenomena that was more important to the EPR small-break. That report is together. We have completed that work. I don't think it has been

presented to you yet.

CHAIRMAN BANERJEE: Okay, but it is ready?

MR. BAJOREK: It is ready, yes.

CHAIRMAN BANERJEE: Okay.

MR. LU: But from our perspective, not only we are using TRACE, which in other cases we use different codes, too.

CHAIRMAN BANERJEE: Yes, but TRACE and PARCS are sort of the workhorse.

MR. LU: Yes, are unique to provide a 3D kinetics capability.

CHAIRMAN BANERJEE: Yes.

MR. LU: That is the reason we are going to give you the presentation about we use that TRACE/PARCS.

CHAIRMAN BANERJEE: Okay.

MR. LU: Okay? I think that finished this part.

MR. TESFAYE: I would like to make some clarification here. Two of the open items that are listed here, one of them belongs in Chapter 16, the other one in Chapter 4. So, there are only two open items left for Chapter 15, and those two open items are closed in those respective chapters. This will also be closed. So, this is kind of unusual for us to

present open items repeatedly here in this presentation.

MR. LU: Also, we have enough coverage for you.

CHAIRMAN BANERJEE: All right. You have to keep us interested, right?

MR. LU: Okay. All right. Should we proceed for the next one or do you want to take a break?

CHAIRMAN BANERJEE: No, we will start and then we will find the right time to take a break.

MR. LU: Okay.

CHAIRMAN BANERJEE: We finished a break at 1:30, our lunch. So, we can continue for a while.

Now you have to tell us what is up-to-date. I have got so many of these pieces of paper now in front of me.

MR. LU: I think that since we are going to give you the presentation of confirmatory analysis, maybe we will ask the staff who worked on that analysis to come over. Somebody needs to call Tony Ulses. I will go over the overview for now.

As I mentioned right at the beginning of my presentation, we developed two sets of presentation. The intention of this set of

presentations is to give you the information about how staff performed the review and tried to use our system or a system code or analysis code to achieve an understanding of a very complicated design using SPND and trip setpoints. Okay. So, that is the intention.

But right at the beginning, I am going to give you some summary of who really contributed to this part of the work. From the Office of Research, we have Shawn Marshall. He is going to talk about the TRACE model. Bill Krotiuk, and they have continued to support us during the past three years. Steve, he has already provided a lot of input into our review. Ron Harrington, he actually initially developed the TRACE stack, ran the LOCA.

From the Reactor System Branch, NRR, Tony Ulses. When he was over Research, he ran this TRACE coupled with PARCS.

And our consultant, Randy Bells from Oak Ridge; Jose March-Leuba and Dave Caraher supporting us to perform this set of analysis.

Okay. I am going to cover the overall confirmatory analysis approach. I want to take this opportunity to not only just cover the AOOs and the postulated accident; I am going to cover overall related to the reactor system EPR review, what we have

done. I will give you an overall picture there.

First of all, we cannot do every single transient that we saw that AREVA ran, all those different cases, when we performed the audit. So, we developed the criteria so that we can have a selected transient, most limiting case for us to perform the analysis.

No. 1 is really we are focused on the new design features, specifically for AOOs because they relied on the in-core setpoint and methodology, and then, of course, SPNDs to produce the protection system trip. That is one of the focuses.

And, then, we used this tool to identify RAIs related to margin evaluation. If some of the RAIs may sound reasonable, it is good to ask, but it is very hard to answer. And if there is not a whole lot of margin value, then we may not even ask those questions because we can ask tons of questions and make the process unbearable.

And the confirmatory analysis scope is specifically related to EPR. And for AOOs, we performed a loss of feedwater heater, controlled withdrawal using TRACE and PARCS.

For the postulated accident, large-break LOCA, small-break LOCA, rod ejections, we all ran the

cases using our codes, AREVA's codes, and to perform the comparison and evaluate whether the RAIs are valid or not.

And, then, to evaluate specific features of the EPR design ex-core detector, because the EPR uses a heavy metal reflector which really reduced the dose and, then, also, its neutron exposure on the vessel. However, it also reduced the neutron flux out of the ex-core detector. So, we are actually right now in the process to perform an MCNP calculation to evaluate the ex-core detector. And the spent fuel pool design, we performed our own criticality analysis.

CHAIRMAN BANERJEE: The MCNP is being done by Oak Ridge?

MR. LU: Oak Ridge, yes.

CHAIRMAN BANERJEE: Who at Oak Ridge is doing that?

MR. LU: John Wagner's team.

CHAIRMAN BANERJEE: Okay.

MR. LU: Okay. As part of these three, we also ran many cases at this point, and we are in the process to finish this part of the confirmatory analysis and to verify whether we have valid RAIs or the RAI closure can be achieved, based on this

analysis. Sometimes we just run the S-RELAP5 code.

That is the overall scope. And, then, I will give you some detailed information about not only the scope, but also the approach and the tools.

As I mentioned, we ran RELAP5, Mod 3. That is RELAP5. We used AREVA code and then used their input deck, but we changed the model. We ran TRACE, PARCS, and then the TRACE/PARCS were supported by NRC Research when Tony was over there.

And ISL provided the confirmatory analysis support using RELAP5.

We also performed a fuel performance evaluation as an initial condition for LOCA. We used FRAPCON; FRAPTRAN; RODEX-4, AREVA code; and we have NRO staff in-house performed those calculations. And we also have PNNL, our consultant, performed the calculation using FRAPCON.

And in terms of AOOs and postulated accidents evaluation, we used TRACE/PARCS, and Tony is walking in. So, he is going to talk about that very soon.

And the TRITON to generate the cross-sections, and RELAP5, S-RELAP5, too.

And for ex-core neutron flux and the criticality evaluation, we used SCALE 6.0, MCNP, and

done by NRO staff in-house and, also, by Oak Ridge.

CHAIRMAN BANERJEE: Just go back to that. Now, for something like rod ejection, the bunch of codes, Area 1, Area 2 --

MR. LU: Yes. Yes.

CHAIRMAN BANERJEE: For things like rod ejection --

MR. LU: Right.

CHAIRMAN BANERJEE: -- there is a whole suite of codes which AREVA is using, which they have the diagram of in their report.

MR. LU: That's right. They did.

CHAIRMAN BANERJEE: How did you sort of -- did you run those codes or you just ran your own codes to validate that?

MR. LU: Particularly for rod ejection, tomorrow morning we are going to cover that.

CHAIRMAN BANERJEE: Yes.

MR. LU: So, there will be a specific section about the confirmatory analysis related to the rod ejection event.

CHAIRMAN BANERJEE: Okay.

MR. LU: And the answer to that question is short. We just used TRACE/PARCS.

CHAIRMAN BANERJEE: Right. I thought that

would be the answer. So, I just wanted you to say that. Okay. That's it.

MR. LU: Okay. All right.

CHAIRMAN BANERJEE: By the way, one other thing. So, in some of these situations where you might get a power pulse, how do you sort of -- which of your codes did you use to look at the enhanced release of radioactive materials during those pulses?

MR. LU: Specifically related to the material --

CHAIRMAN BANERJEE: Material, yes.

MR. LU: As part of a Reactor System Branch review scope, we have not touched too much upon material except for the fuel cladding, which, for example, we use FRAPTRAN/FRAPCON --

CHAIRMAN BANERJEE: Yes, sure.

MR. LU: -- to perform that part of the analysis, which those two codes were developed by the Office of Research, too, supported by --

CHAIRMAN BANERJEE: Yes, but when you get this power pulse or something, you get more stuff being released from the fuel, right, into the gap regions? And AREVA has some methodology to look at that, I think.

MR. LU: Yes. Right.

CHAIRMAN BANERJEE: Or estimate that.

MR. LU: Yes. But I think the details will be provide tomorrow. I think it is specifically rod ejection. And, then, from my perspective, I think Tony is probably the best person to answer some questions about rod ejection confirmatory analysis or --

CHAIRMAN BANERJEE: We can wait until tomorrow. There's no rush.

MR. LU: Okay.

CHAIRMAN BANERJEE: We have time.

MR. LU: Yes. I think for rod ejections really -- okay. All right. That is the overall scope of the NRC's confirmatory analysis.

CHAIRMAN BANERJEE: Do you have your slides? Just as a general request, do we have the slides that are going to be presented tomorrow?

MR. WIDMAYER: Do you want to look at them?

CHAIRMAN BANERJEE: Yes, I don't have them.

MR. WIDMAYER: Okay, I will get them to you.

CHAIRMAN BANERJEE: So, I will get them. So, we have both the applicant's and the staff's,

right?

MR. WIDMAYER: Yes.

MR. TESFAYE: We also will make a quality check. The numbers are correct.

(Laughter.)

CHAIRMAN BANERJEE: Sorry.

MR. TESFAYE: We will make sure the quality will be good in terms of numbering.

MEMBER SHACK: The V&V program.

(Laughter.)

MR. LU: Okay. So, the next one will be Shawn Marshall. He is going to talk about the TRACE model development for the U.S. EPR, which is the base model developed for not only AOO analysis coupled with PARCS to perform the rod ejection analysis, but it also has been used for large-break LOCA analysis, too.

CHAIRMAN BANERJEE: So, I think there is an issue as to whether we should take this before a little break or not. I am just looking at the slides. This is probably going to take more than half an hour. So, I think what we should do is take a brief break. It is a quarter past 3:00 right now.

So, we will go off the record and then we will come on to this. Okay?

Thank you.

(Whereupon, the foregoing matter went off the record at 3:05 p.m. and resumed at 3:17 p.m.)

CHAIRMAN BANERJEE: All right. So, we are going to go back in session, and, Shawn, it is in your hands now.

MR. LU: He's not Shanlai Lu, but we made the mistake in giving the names for the presenters.

CHAIRMAN BANERJEE: That's right.

MR. LU: So, his name tag was not made. We apologize again.

MR. MARSHALL: Well, my name is Shawn Marshall. I work in the Office of Research in the Reactor Systems Analysis Branch.

I have been with the agency almost nine years, starting as a nuclear safety intern, and spent that entire time in the Office of Research.

I have a bachelor's and master's in mechanical engineering from Tuskegee University.

This afternoon my presentation is on the development of the base TRACE model for the U.S. EPR. My objective for this presentation is to provide you with a brief overview of what is in the model that we in Research developed to support NRO's confirmatory analyses.

In doing that, I will provide some general

descriptions of some of the major components in the model, components on the primary and secondary sides, as well as in the emergency core cooling system.

Then, as proof of the fidelity of the model, I will show some steady-state results that we got from TRACE in comparison with some results that were provided by AREVA.

Now our intent in designing this model was to incorporate as much plant-specific, EPR-specific detail as possible without unduly increasing the runtime of the simulations. And this includes simulating all of the basic operational and protection system functions, all of the primary and secondary system functions of the ECCS system and the reactor protection actuation signals.

The information for building the model was provided primarily by the RELAP5 model that was provided by AREVA and the EPR databank. We used our Symbolic Nuclear Analysis Package, or SNAP, GUI to construct the model.

And in summary, the model contains over 185 hydraulic components, over 400 control system inputs, and over 150 different heat structures to simulate the heat transfer.

In this figure, I am showing the entire

nodalization of the TRACE EPR model. In the center is the reactor vessel immediately surrounded by a series of 1D pipe components which simulate vessel bypass flow. These bypass flows range from flows from the control rod guide tubes, guide assemblies, a flow path through the header reflector, and the downcomer upper head leakage flow path.

Branching out from the vessel are the four independent cooling loops, each with a steam generator, and attached to main steam line piping.

At the very bottom of the figure are the four accumulators and accumulator piping with a series of TRACE fields representing ECCS injection sites.

For the primary system, we used a TRACE 3D VESSEL component to model the reactor pressure vessel. In the background I show how we actually nodalized the vessel.

In the foreground I show a cross-section of the vessel with eight azimuthal sectors, one sector for each hot leg and each cold leg. And there are four radial rings with the three innermost rings representing the core and the fourth representing the downcomer.

Staying with the primary system here, I show the pressurizer with four safety release valves

on top. The vessel has TRACE spray components used to provide pressurizer spray. Not shown are heat structures used to simulate pressurized heaters. Unlike what is stated here, we used the proportional integral differential controllers to control the spray and the heaters to regulate pressure. We used a much simpler control to simulate the letdown and charging of the chemical control system.

Proceeding from the bottom of the pressurizer is a surge line that we have modeled to maintain its length and to maintain the elevation change from the pressurizer to the hot leg. And below the surge line you see one of the independent coolant loops with the hot leg reactor coolant pump, cold leg, and crossover leg.

On the secondary side, as I said, we modeled each steam generator separately. The EPR has thousands of steam generator tubes, but we model the tubes with one effective tube, and we use a multiplier to capture, to simulate the surface area of the thousand tubes in the EPR.

We also use a series of 1D components to model the actual economizer in the EPR, to simulate the hot and cold side, downcomer and boiler regions of the steam generator.

CHAIRMAN BANERJEE: Could you just point out the economizer?

MR. MARSHALL: Here is, well, it is a cold leg downcomer and a hot leg downcomer. Here is a separator, the steam dome.

The coolant is separated and it flows down to the boiler section into the downcomer. It is separated; the recirculated condensate falls from the separator to the boiler region, where it is parsed between the hot side and the cold side of the boiler region. But these are the 1D components that make up the economizer. And the feedwater injection injected into the cold side of the downcomer.

CHAIRMAN BANERJEE: So, in a physical load, not a computational load, what does this thing look like exactly? You have got a riser and --

MR. MARSHALL: Right. It is basically a sleeve that allows you to inject in one region and maintain different temperatures on the secondary side, and it allows you to have, I guess, increased steam pressure and improves the overall thermal efficiency of the steam generator.

CHAIRMAN BANERJEE: So, it is like a part of an annulus or there is a --

MR. MARSHALL: It is.

CHAIRMAN BANERJEE: Okay.

MR. MARSHALL: I wish I had a more accurate representation of it.

CHAIRMAN BANERJEE: Yes. I think it is like part of an annulus and it comes in and it goes down and goes across the tube sheet at the bottom or something, but slow on the secondary side? Where does the feedwater come in?

MR. MARSHALL: Well, the feedwater --

CHAIRMAN BANERJEE: Do we have a picture?

MR. PARECE: We have a picture someplace we can dig up.

CHAIRMAN BANERJEE: Yes.

MR. PARECE: What it is is -- think of your typical wrapper a steam generator has, a wrapper where the bundles are inside. And, then, outside is the shell.

CHAIRMAN BANERJEE: Right.

MR. PARECE: So, the water comes down the downcomer.

CHAIRMAN BANERJEE: Right.

MR. PARECE: Well, on half of it, inside is now wrapper outside the wrapper. And the feedwater has J-nozzles that stick in that half, and it forces the feedwater down on the cold leg side.

So, 90 percent of the water coming up the cold side of the tubes on the secondary is at feedwater temperature and 10 percent is recirculated. So, what that does is it gives you the benefits of pre-heater without all the trouble of the pre-heater. Because in the past pre-heaters put the feedwater in low and cross-flow through the tubes and cause vibration. The water boxes collapse due to water hammer and condensation of steam. In this case, we are just directing the cold water up one side, and there is a divider plate between the two sides.

CHAIRMAN BANERJEE: Yes.

MR. PARECE: These steam generators are in operation right now at the N4s in France for more than the last 10 years. These are a slight upscale. And when I say "slight", the drum is stretched to allow more water, and the diameter is slightly increased to allow one or two hundred more tubes. But it is basically the same generator that is in operation at the N4s in France.

CHAIRMAN BANERJEE: Okay. Thank you.

MR. MARSHALL: On the main steam lines, we also modeled the main steam relief train which consists of the main steam relief isolation valve, the main steam relief control valve.

As far as the emergency core cooling system, as I said, we modeled the four accumulators.

CHAIRMAN BANERJEE: So, that economizer is a sort of new thing for TRACE, right? So, how did you validate TRACE, that it was doing okay with that?

MR. MARSHALL: We looked at it. We didn't see that it presented any new phenomena, I guess was our main conclusion. It is a new apparatus, but the phenomena was what we had seen before.

CHAIRMAN BANERJEE: Okay. Let me think about it. Carry on, yes.

MR. MARSHALL: Okay. Continuing with the ECCS injection system, we also modeled the medium and low head safety injections as well as the cross-connect between the low head loops 1 and 2 and 3 and 4.

Once we had the model well assembled, we did a number of shakedown tests and steady-state calculations as proof that the model could accurately simulate the plant at normal operating conditions.

And in this table, I am showing some TRACE results compared to some results provided by AREVA for these specific parameters. And overall, the results look comparable and served as a good baseline case, I mean a lens through which the subsequent TRACE

calculations could be viewed.

So, with that, I have given you a glimpse into what is in the TRACE EPR model, starting with some of the components on the primary, secondary, and ECCS systems.

Based on the results that we have shown, TRACE results in comparison with some of the AREVA results, we feel that the model is solid and that it can provide useful information to the staff as a base model, and it can be modified to perform more specific calculations such as transients and AOOs.

CHAIRMAN BANERJEE: So, the main difference is just the economizer? It is not a difference, but it is the only thing which is significantly different from a --

MR. MARSHALL: Apart from the size.

CHAIRMAN BANERJEE: Yes, apart from the size, yes.

MR. MARSHALL: They have a reflector that is different. We modeled those flow paths, and it doesn't have high-pressure injection. It has a medium head.

CHAIRMAN BANERJEE: Right.

MR. MARSHALL: Yes. Those were the main differences that we saw.

CHAIRMAN BANERJEE: So, how did you validate TRACE to ensure that it captured the secondary site behavior, so that you could get the cooldown and things like that correctly?

MR. MARSHALL: Well, about three years ago, three or four years ago, we did an extensive evaluation of TRACE looking at comparing TRACE with a number of results from some integral and separate effects tests, ROSA, PKL. We found TRACE to be acceptable for simulating large-break and small-break LOCAs in PWRs and BWRs. That was the foundational assessment.

CHAIRMAN BANERJEE: Right. But with regard to the controlled cooldown --

MR. MARSHALL: Well, I think Steve mentioned --

MR. BAJOREK: This is Steve Bajorek from Research.

Shawn, the other thing that you didn't have enough space for on your summary table is we also went around and we looked at the recirculation flow rate in the steam generator, the feedwater temperature in, the steam temperatures out, conditions around that steam generator circuit, and compared those to information that had been obtained from AREVA.

CHAIRMAN BANERJEE: So, what is the phenomena in the steam generator that is important during a controlled cooldown?

MR. BAJOREK: During a controlled -- I'm sorry?

CHAIRMAN BANERJEE: Cooldown. Cooldown. I mean you are doing all sorts of things like blowing steam. What becomes the governing phenomena?

MR. BAJOREK: Well, I think our most interest is for a small-break where reflux condensation, CCFL, in the hot leg in the steam generator elbow are going to be the phenomena which will have the greatest impact on the transient.

The modeling that Shawn used in setting up the EPR model is very similar to the modeling techniques that we used in tests like BETHSY, ROSA4, and in the applicability report, which you haven't seen yet, some additional ROSA tests, some steam generator tests that had been done at Oregon State, and also the flexset steam generator tests.

CHAIRMAN BANERJEE: All right. The flexset in --

MR. BAJOREK: Flexset. Yes, after they had done the reflood tests, they had modeled a steam generator, 1620 tubes, not a very large scale, a

simulated hot leg, and they completed the circuit and measured the amount of reflux condensation that they would get out of that partial size bundle.

So, what we used it for was a way of looking at flexset, BETHSY, and ROSA, of taking TRACE and trying to show that it was scale-independent, as you took a look at full height, which I think flexset was, to partial scale for Oregon State, full scale and height for ROSA. We tried to look at a wide range of geometries.

And those modeling aspects which were used for those tests were preserved in Shawn's model for the EPR.

Now, for the economizer and what goes on in the feedwater system, delivery of water to the bundle, I think we just looked at that primarily as an additional flow path, but not in the way that it was something that was introducing a lot of new phenomena that would have a major impact during either the large- or the small-breaks.

CHAIRMAN BANERJEE: And nothing new happened? There is a dependence here on the secondary site cooldown to get the pressure down.

MR. BAJOREK: Oh, it has been a while since I have looked at that report, but I believe

there is also one of the ROSA4 tests where they had a simulated rapid cooldown of the steam generator secondary side. That was part of the test matrix that they had done for ROSA4. It wasn't part of one of our initial assessments that was in the assessment report, but I think that was picked up in the applicability report.

CHAIRMAN BANERJEE: So, the secondary site of ROSA4 is pretty one-dimensional because it is just a long, thin thing, right? Whereas, this is a sort of fat, big thing? So, what happens in the secondary site?

MR. BAJOREK: Well, one of the other parts of that evaluation is we took a look at, was it necessary and important to model the steam generator with multiple paths through the primary side? You get some different phenomena that occurs in the short-length tubes, the medium-length and the large-length tubes.

We had set up a model that looked at those multiple paths. Again, I would have to go back and look at the report, but I think our conclusion there was modeling things with one single path was adequate for the various phenomena that went on in the multiple flow path model.

CHAIRMAN BANERJEE: You know, the primary site, I can see that. I mean the height differences or some things, some may be actually naturally circulating and others might actually be refluxing. There are all sorts of things that can happen.

But maybe that doesn't matter, but I am just asking the question as to whether the multi-dimensionality of the secondary side -- the primary side, after all, is tubes. So, some are going to go and some are going to reflux maybe.

On the secondary side, though, it is inherently sort of a three-dimensional system, right?

MR. BAJOREK: Yes. I mean there's really nothing except for the divider plate on the hot and the cold side to prevent mixing from the flow. I guess we didn't do any kind of special modeling features to try to account for that.

MR. KROTIUK: This is Bill Krotiuk.

One thing I could say is that we have modeled, as Steve has said, you know, ROSA and PKL and all that. And when we modeled ROSA, I know quite intimately on that test we modeled them using the same methodology that is explained here.

And when we look at test data for both steady-state or the transient conditions, our results

pretty well match the test results.

CHAIRMAN BANERJEE: Which is what you would expect --

MR. KROTIUK: Right.

CHAIRMAN BANERJEE: -- because they are essentially one-dimensional.

MR. KROTIUK: ROSA has a lot of tubes in --

CHAIRMAN BANERJEE: Yes, but it is really fairly small compared to the real thing.

MR. KROTIUK: Yes, it is small compared to this.

CHAIRMAN BANERJEE: Right. Now EPRI had a program at one point, and I think it was Brian Spalding who was developing a code to take this into account. He called it URSULA for some reason. I'm not sure why. This goes back in history.

But there was strong three-dimensional effects, if I remember.

MR. BAJOREK: Which test was that? That wasn't MB2, was it?

CHAIRMAN BANERJEE: You know, my memory fades, but I can probably find out.

MR. BAJOREK: We would be interested in hearing that, but --

CHAIRMAN BANERJEE: Yes.

MR. BAJOREK: At least in looking at the other tests, I think as Bill pointed out, we seem to have matched the available data --

CHAIRMAN BANERJEE: Yes.

MR. BAJOREK: -- in looking at things like information that you can get from ROSA or MB2, we haven't noticed any large 3D-type behavior.

CHAIRMAN BANERJEE: So, is this sort of the first time we have encountered use of secondary site cooldown to depressurize within the PWR fleet?

MR. BAJOREK: Except if you consider Candu reactors.

CHAIRMAN BANERJEE: Yes. Candu, of course, have been doing it for --

MR. BAJOREK: But this is the first time for PWRs with U-tube-type steam generators.

CHAIRMAN BANERJEE: So, what have we learned from Candu's? The B&W steam generator is there. They must have been doing this for quite a while.

MR. BAJOREK: I'm really not sure.

CHAIRMAN BANERJEE: So, you didn't talk to these guys to find out what they have been doing?

MR. BAJOREK: Not for the Candu systems,

no.

CHAIRMAN BANERJEE: There is somebody who wants to talk from the AREVA side.

MR. SALM: This is Robert Salm. I spent 10 years in Germany with B&W's 205 plant, Mulheim Kaerlich. And Mulheim Kaerlich had the same type of medium head safety injection, four-train system, and cooldown of the secondary side. And we did extensive tests at Alliance Research Center with once-through steam generators using this approach and benchmarked our codes against those tests, and really didn't see anything different than we are seeing here.

CHAIRMAN BANERJEE: This is a pretty more complicated steam generator than the once-through, correct?

MR. SALM: Complicated hardware-wise, but not necessarily phenomenologically. I mean a once-through steam generator is pretty complicated from a phenomena standpoint.

CHAIRMAN BANERJEE: Yes.

MR. SALM: And we had subcooled water coming in the bottom, and then you go up to superheated steam.

CHAIRMAN BANERJEE: Yes. Yes. You had it --

MR. HARRINGTON: Yes, this is Ron Harrington with the Office of Research at NRC.

You still have a question of how we modeled the economizer?

CHAIRMAN BANERJEE: Well, I have a broader question. It is sort of this controlled cooldown here is a critical component of depressuring the system. So, okay. So, now, in addition, there is sort of maybe a complexity, maybe not associated with the economizer. These are pretty big steam generators. So, they are quite multi-dimensional.

Whether that is important or not, I don't know, but I was wondering how the multi-dimensional effects were assessed and how you sort of felt comfortable with taking essentially a one-dimensional representation of that.

MR. HARRINGTON: I can't speak to how it was assessed, but I can show you how we modeled it, how we set up the thing.

CHAIRMAN BANERJEE: Right. He showed us the pictures.

MR. HARRINGTON: It was basically that nodalization along with the way the heat structures are set up to go with that. That is how the economizer was modeled.

CHAIRMAN BANERJEE: Right. So, if you had made a model which looked a little bit like a core or something, you know, for the secondary side, would you get the same answers?

MR. LU: Let me add something here. The intention for the AOO analysis right now is to run the feedwater heater trip and the feedwater temperature reduction cases, and, also, the rod withdrawal.

So, this phenomena you asked for for emergency feedwater injection for these particular cases, we are trying to study. Yes, it may be important for other transients, but specifically for these in-core trips is not that important. Now as the model can reach the heat balance, that should be fine.

CHAIRMAN BANERJEE: Yes. I think what you are saying is, if you limit it to the AOOs, I don't disagree.

MR. LU: Yes.

CHAIRMAN BANERJEE: In fact, I may not disagree for everything else, but I am just saying that for this, yes, but for the case where your controlled cooldown becomes a crucial aspect, and no doubt you can make a rough estimate of this based on the heat balance and stuff.

MR. LU: That's right.

MR. BAJOREK: Dr. Banerjee, I think I remember once where one of the reasons that the steam generators and the safety analysis were modeled in more of a 1D or crude fashion is, when you start to go from steady-state power to decayed heat, they are so massively overdesigned to remove that amount of decayed heat that most people have taken a more simplistic view.

CHAIRMAN BANERJEE: Sure. Yes.

MR. BAJOREK: Even it has much more surface area than it needs. So, there hasn't been a whole lot of work, at least not to my knowledge, on trying to get very detailed in how that energy gets out of the primary system.

CHAIRMAN BANERJEE: Yes, you are absolutely right, Steve. The main thing, though, is if you get into a refluxing mode, and if you are refluxing for, let's say, all the tubes -- let's take a scenario, okay, like that. Then, of course, you are using only a part of it. So, you get maldistributions in your heat load. So, you can see where I am coming from.

MR. BAJOREK: Yes, because I think that some of the ROSA tests, what they showed is you would condense in some of the tubes.

CHAIRMAN BANERJEE: Yes.

MR. BAJOREK: And they would flood and drain while steam would start to flow into some of the other tubes --

CHAIRMAN BANERJEE: Other tubes, yes.

MR. BAJOREK: -- condense, and it would sort of chug with alternating tubes doing the condensation.

CHAIRMAN BANERJEE: And, then, if you are refluxing in all the tubes, then, of course, you have got another set of problems now at the elbow to go into the steam generator plenum.

MR. BAJOREK: Then you have the CFFL --

CHAIRMAN BANERJEE: Yes.

MR. BAJOREK: -- allowing that water to get back, yes.

CHAIRMAN BANERJEE: We will visit it when we come to the small-breaks. So, we will let you go right now, because, otherwise, we will be sitting here all evening.

And there is one additional item which AREVA would like to do which is the Chapter 11 certain points. So, we will try to make room.

MS. SLOAN: Thank you.

CHAIRMAN BANERJEE: So, thanks, Steve, and

thank you, Shawn.

MR. LU: I think this provides the information about how we developed the TRACE thermal hydraulics model, but the two simulated in-core trip setpoints and the algorithm and the trips itself, based on the SPND and the system, we need the 3D kinetics.

CHAIRMAN BANERJEE: Right.

MR. LU: Here is Tony Ulses.

CHAIRMAN BANERJEE: Tony is going to tell us.

MR. ULSES: Let's see if I can sit low enough here so you can see the screen behind me. I'm being challenged.

(Laughter.)

All right. So, good afternoon.

As Shawn said, my name is Anthony Ulses. I was in the Office of Research. Now I am with the Office of Nuclear Reactor Regulation.

And we were tasked with trying to develop a model to allow the Oak Ridge work, which we are going to talk about subsequent to this, to have access to actual information regarding what the in-core nuclear detectors would see, which is what those trip functions are using as their function.

So, obviously, if we had actual plant data, we would use that. But, lacking that, we needed to develop a three-dimensional PARCS model to allow us to do that simulation.

I am going to walk you through in a couple of slides here how we developed that model, and, then, I am going to hand it off to Randy Bells from Oak Ridge who will go through the details of how they actually simulated.

Shanlai is going to do that. Well, I apologize. It is a last-minute change, a last-minute game change here.

So, we started with the LOCA deck that you just heard which was described by Shawn. And, then, we overlaid the multi-dimensional neutronics model on it.

This is a pictorial representation of the model itself. This image here, basically -- does this mouse work? Yes. Oh, there we are.

This here is the layout of the core showing the different fuel types. This model here shows how we linked the individual fuel types to the TRACE radial and, also, azimuthal noding. Because, obviously, we have to feed the information related to the moderator temperature, the moderator soluble

poison back to PARCS. And, then, for each one of the fuel assemblies, on the left-hand side here, we added one actual heat structure to individually model each of the fuel assemblies. So, we have a direct one-to-one relationship there.

And, then, again, this was laid into the TRACE model that you heard about from Shawn a minute ago.

CHAIRMAN BANERJEE: What did you mean by 14-to-1 moderator coupling?

MR. ULSES: Well, that means that, if you estimate the number of assemblies which are included in each of these azimuthal and radial sectors, it is roughly about 14 assemblies in each assembly.

CHAIRMAN BANERJEE: Okay.

MR. ULSES: In each one of these larger nodes that we have from the TRACE model. And that is not exactly the representation. That is sort of an estimate based on looking at how they laid out.

We developed our own cross-sections. We used our SCALE/TRITON model to do that. This is a pictorial representation of one of the representative assemblies.

We looked at all the individual assembly types in the core. We actually depleted them

individually, looked at all the relevant branch cases in order to give us a modeling capability to do calculations full spectrum for all the AOO anticipated situations.

And this is also the model we used to do the RAA simulations, which we will talk about more tomorrow.

So, again, as I said, what we are trying to do here is we are trying to develop representative SPND information which we then use to follow on subsequently to evaluate the effectiveness of the system trips, I guess what you asked.

We looked at two individual transients. One was a bank withdrawal, and the other was a loss of feedwater heating. And we picked the points -- what is key here is that, if I can get back here, the point here, this is the point where the evaluation of the individual algorithm takes place. In other words, this will be talked about later in the subsequent presentation.

But as long as the power in the SPND, I should say the simulated SPND in this case from PARCS -- boy, this thing is giving me a fit -- here is used at the same point as I used it in the algorithm, there is a one-to-one relationship. So, that is the point

of what we are trying to achieve here.

And that is really about it that we have for the model demonstration. Again, we needed to have information from the representative detectors. And again, we would have ideally used actual plant data, but, of course, that doesn't exist.

And, then, we looked at the two transients. One was chosen to give us the radially-asymmetric conditions, in other words, the rod withdrawal, and the loss-of-feedwater transient is basically sort of the global power transient.

CHAIRMAN BANERJEE: You have calibrated the code against rod withdrawals and things, but against other plants, right?

MR. ULSES: Right. We looked at the PARCS code against multiple examples from the operating fleet.

CHAIRMAN BANERJEE: Sure.

MR. ULSES: And I wouldn't expect to see any real differences here in the EPR in the behavior of PARCS.

So, that's all I have. Okay?

CHAIRMAN BANERJEE: I'm glad this thing is going forward, TRACE and PARCS.

MR. LU: Okay. As I mentioned, the reason

we want to perform this assessment, to support it, to fully understand the in-core trips system and, then, how it really generated at the DNBR trips and the LPD trips.

And the EPR, it is so unique. It has a very large-sized core, you know, a large number of bundles, and 14 feet of the axial ends. And the ex-core detector outside could not really sense significant change of the inside of the core region, although you have a significant mandated change. So, that is the reason we wanted to use this one, as Tony mentioned.

We have realistic calculated three-dimensional neutron flux from the initial steady-state to the transient, and which are the two transients that we mention here. Then, we can calculate what is exactly happening with the neutron flux, what can be the simulated SPND responses, and use that neutron flux and the simulated SPND response to calculate the timing of the trip, and see whether at the expected timing of the trip the SPND trains will be able to produce that trip. And if it does, at least we would say, okay, we feel comfortable that that concept might work.

And, then, follow the support of all other

RAIs, and we will be able to evaluate it as part of the AOO analysis. Okay. So, that is the purpose of this part.

And actually, Randy Bells and Jose March-Leuba from Oak Ridge, they did this part of the work, and I was just being told the contractors were supporting staff's presentation. So, I am giving the presentation for them.

Okay. All right.

CHAIRMAN BANERJEE: They aren't here? They decided not to make the trip?

MR. LU: Well, we will see at the end point of what we will show you.

I think, since you asked that question, I want to focus on this number. Okay. For rod withdrawal accident, the expected trip is going to happen at 20 seconds, and that for loss of feedwater heater trip, they expected a trip to happen, expect it to happen at 50 seconds. So, we received other SPND, received that signal.

Okay. All right, a little bit more detailed information. I think Dr. Jonathan Witter, he went through this very detailed in the morning. So, I will just give a summary.

Then, there are 72 in-core and self-

powered neutron detectors which are going to be used to generate the trips, and, then, to generate the DNBR trip and the LPD trip.

There is an online algorithm, continuous running to monitor the hot rod DNBR. Every 100 milliseconds, it performs an online calculation.

And that algorithm cannot be very sophisticated because you require so much computing, and then you cannot perform online and subchannel analysis. Therefore, that part of the algorithm was simplified with a 95/95 uncertainty band around that part.

And there are 12 strings of the DNBR string to calculate for release in the rod. So, therefore, what we want to see is let's see if we have actual can calculate the three-dimensional neutron flux. Those are from a steady-state to the transient. Let's see how the aeroball system would generate the calibration factor using the steady-state, and during the transient what aspect it will give the responses, follow the same algorithm.

CHAIRMAN BANERJEE: So, going back to that slide, you say not enough computer power to perform a detailed sub-channel. What do you mean exactly?

MR. LU: Okay. Yes. What I mean is --

CHAIRMAN BANERJEE: Only online, yes.

MR. LU: -- for the sub-channel analysis, the code, into a trip to perform that on every --

CHAIRMAN BANERJEE: A hundred milliseconds is --

MR. LU: Yes, every 100 milliseconds. It is throughout the entire core. It is almost -- I have not seen that kind of a code. If it works, it would be good.

Okay. So, that is the reason, and they simplify that algorithm.

CHAIRMAN BANERJEE: But this is single failures.

MR. LU: Yes. I think you are right there.

CHAIRMAN BANERJEE: You can probably do it on a graphic --

MR. LU: Yes, but there is also -- remember, you asked a question about a chi. There is a quality trip there.

CHAIRMAN BANERJEE: Yes.

MR. LU: So, once you exceed a certain limit, you are going to still hit the two phase, but because of the correlation, the limitations of the design, they do not want to see that. So, at a

certain point they --

CHAIRMAN BANERJEE: All bets are off.

MR. LU: Exactly.

CHAIRMAN BANERJEE: Okay.

MR. LU: Exactly.

CHAIRMAN BANERJEE: Okay.

MR. LU: All right, that is the background. Then, so what we are doing here, what's the reason for us to select the feedwater heater trip because that is a whole core global power check. We want to see if the entire core has the power going up. Because once you have a feedwater heater trip, the steam generator, you know, you have different temperature going through the reactor coolant system. And once the temperature of the coolant coming into the core, that changes the total power because of moderator feedback. And, then, that is expected to be a core-wide power change. Okay?

Rod withdrawal will give you localized neutron flux shape change. So, you see, if you have this kind of rod withdrawal situation, what is going to be the localized SPND responses?

So, those two transients were selected just to evaluate that. Okay.

The process we followed was Tony ran a

steady-state TRACE coupled with PARCS, and once it reached a steady-state, we used that neutron flux calculated by PARCS, every node -- I forgot how many nodes, maybe 23.

MR. ULSES: Twenty-four.

MR. LU: Twenty-four.

CHAIRMAN BANERJEE: What Tony showed as the map --

MR. LU: Yes, but actually there are 24. So, it is a huge three-dimensional core model. And with that one, we can calculate the actual aeroball system, the neutron flux will be assessed by the aeroball system.

And you use that neutron flux to determine the SPND calibration factor, what we call the C ij parameters. And more detail was actually given this morning.

And we use that one. And actually, Jose March-Leuba did this calculation, figured out this is the C ij number for each of the 72 SPNDs. And once that is fixed, then follow the TRACE three-dimensional power during the transient and calculated the SPND responses using the same algorithm defined by the setpoint and methodology, and using that one to see whether all 72 SPNDs would trigger the trip. That is

the calculation process.

So, what the endpoint is, we valued, calculated it for hot rod what's the DNBR and what's the maximum medium power density.

Okay. Here is the calculated number. Okay. There are 12 strings for DNBR calculation. Each string has six SPND axial directions. And you can see that the critical heat flux ratio is calculated based on we know the power, based on PARCS calculation, and we also applied the approved AREVA's, the CHF correlation for that bundle. And we calculated what is the limiting minimum DNBR.

CHAIRMAN BANERJEE: This is a loss of --

MR. LU: That is a loss-of-feedwater heater event.

CHAIRMAN BANERJEE: Feedwater --

MR. LU: Right.

CHAIRMAN BANERJEE: -- heater?

MR. LU: Yes. I think it was at 50 seconds, right. At 50 seconds, you can see that, except for one string, all other strings predicted the DNBR less than what is actually for the hot bounder what it should be.

Because of that, if there is a trip setpoint set low enough at this point, all 12 strings

except one will start the generator trips.

So, this plot just confirmed that, based on TRACE/PARCS evaluation calculation, and, then, with SPND in-core, self-powered neutron detector, if the algorithm and the hardware works, the trip will be generated.

CHAIRMAN BANERJEE: Where is the setting there?

MR. LU: Okay. This is a snapshot --

CHAIRMAN BANERJEE: Yes.

MR. LU: -- of the calculated DNBR at 50 seconds. So, that is expanded at a time the trip is supposed to come in. Because of all the calculated DNBR, at this point it is lower than the actual one. So, we know that the --

CHAIRMAN BANERJEE: The trip is supposed to come at this point.

MR. LU: Exactly.

CHAIRMAN BANERJEE: Yes. Okay.

MR. LU: Exactly. So, therefore, this algorithm itself we found, at least based on this simulation, appears to work. Of course, to make this algorithm and the in-core trips functional, and it still relies on the hardware, so the Division of Engineering and the Regional INCs are evaluating

whether we can really achieve the signal. However, from the reactor system perspective, the algorithm, we verified it looks like it appeared to work.

Okay. The next one, for rod withdrawal, a similar situation. Actually, this is much better then because all 12 strings show the DNBR are less than the calculated, the minimum DNBR.

And that means, at 20 seconds for this particular event --

CHAIRMAN BANERJEE: This is the 20 seconds, right?

MR. LU: Yes, this is 20 seconds for a rod withdrawal event. Okay.

And, then, there is the calculated SPND power change. During rod withdrawal, for rod withdrawal, we were trying to catch the localized power peaking change. And you can see that each one is the string, a number, and then tells you at time zero it is blue. At 20 seconds, the trip, and the control is supposed to be dropped at that time. And you can see that the purple one, the power change for one of this string, you can see almost doubled. This one is increased, but the rest of the stuff --

CHAIRMAN BANERJEE: Going down.

MR. LU: Yes, going downward. What it

really gives us, the confidence here is, okay, with a significant localized power peaking factor change, the calculated DNBR still will trigger the trips. That verifies the algorithm for us to see whether for AOO application is that okay or not, when they take the credit of this --

CHAIRMAN BANERJEE: So, explain to me how that previous slide with those red and blue lines, the bar chart that you had --

MR. LU: The bar chart?

CHAIRMAN BANERJEE: Yes. That shows me which --

MR. LU: That is the radial power peaking factor.

CHAIRMAN BANERJEE: In D13, DO3 goes up, but --

MR. LU: Yes.

CHAIRMAN BANERJEE: -- how does it tell me that I'm okay?

MR. LU: Okay. What really it gives you, each SPND, certain neutron flux.

CHAIRMAN BANERJEE: Yes.

MR. LU: And this neutron flux was translated to how can it not be exactly the same magnitude. So, there is a calibration factor to

calibrate that neutron, measure in a sense the signal, to translate that to what is the exactly predicted hot rod power density, medium power density there and the neutron flux there.

And, then, the algorithm, based on these responses throughout the entire core -- so, the concept is just like R jump (Phonetic) function. You have the function, and it gives a certain contribution to a certain point. And from a certain point, you can look at every single assembly; the neutron flux level will give you a different weighting for your contribution for that power level at that particular point.

With that one, that demonstrates that this algorithm would function to graph the localized change in the power shape to produce the DNBR and the LPD.

CHAIRMAN BANERJEE: Yes, it sort of shows you what these SPNDs are seeing there.

MR. LU: Yes, the change. Yes.

CHAIRMAN BANERJEE: But it is really the previous graph that shows you that.

MR. LU: Exactly. Yes, yes. Actually, those two, the DNBR trip --

CHAIRMAN BANERJEE: Yes.

MR. LU: -- were calculated. But this

just shows you the power shape change.

CHAIRMAN BANERJEE: Yes.

MR. LU: So, that gives you, gave us a feeling of what is exactly the localized power change. Otherwise, we would use three-dimensional graphics to show the core power change, but that is a little bit too hard. But maybe tomorrow you can see some pictures of it.

CHAIRMAN BANERJEE: Maybe some nice animation.

MR. LU: Yes. I think Tony generated that, right?

CHAIRMAN BANERJEE: If I give a talk to students, I have to show them animation or else they go to sleep.

(Laughter.)

MR. LU: I think we need to provide that one.

CHAIRMAN BANERJEE: Yes, to keep us awake.

MR. LU: But we have to double-check AREVA and make sure that is not proprietary information.

CHAIRMAN BANERJEE: We can always close the session.

(Laughter.)

MR. LU: Okay. All right, let's give you

this LPD confirmatory calculation for loss of feedwater heater event. And you can see the blue line is the actual, the measure calculated with PARCS, actual power shape. And not only just the power shape, but it is the actual megawatts for each node.

And, then, for all 12 strings, what is really calculated there, for each point, what is the calculated medium power density from all 12 strings, contributed to each axial node?

Okay. Then, what it really tells you is, at 50 seconds, the calculated, the algorithm itself would predict higher than actual the power density. Because of that, if the SPND started to trip, it would be conservative.

CHAIRMAN BANERJEE: So, D13 systematically shows --

MR. LU: Yes, D13 is a string of the SPNDs. There is core map there to show what is the range.

CHAIRMAN BANERJEE: Yes.

MR. LU: D13 there.

CHAIRMAN BANERJEE: D13 and P15.

MR. LU: Yes.

CHAIRMAN BANERJEE: Yes.

MR. LU: It is just there are a total of

12 strings. Each string of the SPND, inside the core there is six of them. And, then, with that one, that will give you a number. This plot, it just shows you how the SPND calculates the localized power peaking factor, the localized power density.

Because of that, there are so many numbers of the SPND strings; it predicts the higher power density. The demonstrated algorithm itself is conservative.

CHAIRMAN BANERJEE: Yes.

MR. LU: At least for this particular case, TRACE and PARCS.

The same situation for rod withdrawal event. Most of the strings would predict a higher medium power density. So, all those reach the actual setpoints. That means the actual power density is lower than the setpoints.

CHAIRMAN BANERJEE: Boy, that is a pretty big deviation.

MR. LU: Quite big. Quite big.

So, the way they developed this one, although even this, it is a big, but it still will be sufficient to protect the core, from our understanding. But, of course, it has to be realized by the hardware system itself.

So, what really we learned from here is we felt that the algorithm should work, but the actual configuration, the actual system hardware needs to be tested. So, based on this calculation, we actually requested an additional ITAAC item.

Not only just we are relying on the TRACE/PARCS to give us information, but we use this information to tell us how we are going to pursue this review and close this issue. As part of the ITAAC, we asked AREVA to demonstrate and test the entire system, once it is a completed design and installed, to verify that it can do this job it is supposed to do.

Okay. All right. So, that is the results evaluation. What we found is that the EPR protection system using signals from in-core SPNDs should be able to generate a reactor trip.

And as part of the topical report of setpoint methodology, we have issued a Draft SER. And I think a final will be issued sometime down the road. At this point, we feel comfortable that this algorithm should work.

The conclusion, I just want to give this slide as a conclusion. We used confirmatory analysis to help us generate useful information, so that we can ask the right RAIs and quantify the margins, possible

margins, with certain RAIs, so that we can have targeted RAIs, that we focus on the issues which are significant safety issues, if we needed to evaluate that part.

I think this, the confirmatory analysis, developed by the tools, developed by the Office of Research, TRACE/PARCS, and the analysis done by Tony really helped us to move forward and focus on the right issue and the result. I think that is the conclusion related to the overall staff's review of the Chapter 15 anticipated occurrence of transient, operation occurs, and the postulated accidents, and how we concluded our Phase 2 review with this process to help us out.

And as I mentioned right at the beginning, our review is just Phase 2. We are giving you a status of Phase 2 review conclusions. It is not a final DCD conclusion yet. So, if there are any questions or there is any new information coming, we will process that in the review.

CHAIRMAN BANERJEE: Thank you. That was illuminating.

Bill, do you have any questions? Or John?

MEMBER STETKAR: No, sir.

CHAIRMAN BANERJEE: Okay. So, I think we

are probably finished for today with you, and we will have AREVA back now briefly. But thank you very much. We really appreciate this and we look forward to seeing some subset of you tomorrow on the rod ejection.

So, tomorrow we are going to talk about rod ejection and just be briefed on certain topics by AREVA.

MS. SLOAN: Certain small-break LOCA topics.

CHAIRMAN BANERJEE: Yes. Some topics. But the staff won't say anything.

MR. TESFAYE: Not in the afternoon. In the morning, we will have our --

CHAIRMAN BANERJEE: No, only in the morning.

MR. TESFAYE: Yes.

CHAIRMAN BANERJEE: Yes. Okay.

So, thanks a lot. We will see you tomorrow.

So, it is back to you, Sandra, and we are actually running ahead of schedule, which I think is wonderful. I like that.

PARTICIPANT: This design center, we have been usually running ahead of schedule in the ACRS

meetings. And I don't have an explanation why, except maybe we have done a lot of work in preparation, the staff has.

CHAIRMAN BANERJEE: Right.

PARTICIPANT: So, we all, the NRC staff, will take credit for that.

(Laughter.)

CHAIRMAN BANERJEE: Yes. It is an efficient sort of operation.

So, Pedro, you are going to take off for some distant land, I understand.

MR. PEREZ: Correct.

CHAIRMAN BANERJEE: Tell us what you have to before you do that.

MR. PEREZ: This is Pedro Perez. I'm back, and now it is good afternoon.

Back in April, Getachew mentioned that AREVA took a set of homework questions concerning the radioactive waste cleanup system. The System Engineer at that time, Mr. Craig Schmiesing, went back to the office and researched everything. He is now working in Finland at the Olkiluoto site. So, he can really learn his system.

But he left behind the answers to those questions, and I would like to go over them for the

record. And again, this goes back to April of 2010.

The following questions were asked:

"What is the reliability of the evaporator and centrifuge? How often is the equipment maintained?"

The major equipment of the evaporator is known to exist without major maintenance for roughly two to four years, depending on frequency of use. Minor maintenance, such as pump seal replacement, can be done while the system is online because of the infrequent flow through the system. This regular maintenance helps to prolong the life of the equipment.

MEMBER STETKAR: Pedro, I'm sorry. Could you repeat that? I was looking for -- I know I asked this question --

MR. PEREZ: Yes.

MEMBER STETKAR: -- but I couldn't find it. So, could you repeat that, so I can make a couple of notes here?

CHAIRMAN BANERJEE: You don't have it documented in a slide or something that we can have?

MR. PEREZ: We can do it tonight and have it ready for tomorrow.

CHAIRMAN BANERJEE: Yes, just for the

record.

MS. SLOAN: We can get copies. We will send it to Derek.

CHAIRMAN BANERJEE: Yes. As long as we have it --

MS. SLOAN: This is an internal calc file. We will get the information available, and we will send it to Derek.

CHAIRMAN BANERJEE: I mean it goes into the record in the transcripts, but we may as well have it.

MS. SLOAN: Sure.

CHAIRMAN BANERJEE: Also, it is not my area of expertise. So, I am going to sort of need to give this to Dana.

MR. PEREZ: I'll repeat --

MEMBER STETKAR: Thanks.

MR. PEREZ: -- the answer to that question about the reliability of the evaporator and centrifuge.

And the answer was the major equipment of the evaporator is known to exist without major maintenance for roughly two to four years, depending on the frequency of use. Minor maintenance, such as pump seal replacement, can be done when the system is

online because of the infrequent flow through the system. This regular maintenance helps to prolong the life of the equipment.

And this was obtained by our office in Germany, talking to those users with that equipment.

The second question was: "What is the water content of the sludge coming out of the decanter?"

The water content of the sludge from the centrifuge is approximately 30 percent by volume. If the water content is less than 30 percent, the drum is sealed and brought to the drum storage area. If the water content is greater than 30 percent, the drum is brought to the drum drying station of the solid waste processing system to process prior to it being stored in the drum storage area.

The third question: "What storage capacity volume was generated using the 7.5-year storage value?"

The 7.5-year storage capacity was determined using the volumes documented in the FSAR Tier 2, Table 11.4-1.

"Is there a buried pipe in the run from the containment to the radioactive waste processing building?"

The only system that may have buried pipe is the nuclear island drain and vent system. Any buried piping will be guard-piped with means to detect leakage or in trenches, such that it can be maintained or replaced as necessary. Radiation or leak detection monitors will be placed to aid in locating any leakages in the system.

"What are the hydrogen and oxygen concentrations prior to reaching the gaseous rad waste system?"

The normal expected concentrations of hydrogen and oxygen is dependent upon operational states of the connected system. For example, cooling treatment or cooling the gasification system. While one of these systems is in operation, the hydrogen/oxygen concentrations will vary depending on primary coolant chemistry and plant operational status. For example, typical hydrogen concentration during normal operation, while the coolant treatment system boric acid column is in operation, is approximately 1.5 percent by volume. Typical hydrogen concentration during outage preparation and while cooling the gasification system, the gasifier column is in operation, is approximately 2 percent by volume.

And, then, the final question was: "How

do the air-operated valves in the gaseous waste processing system fail on loss of compressed instrument air? And the main concern is the consequences of valve closure around the recombiner."

Two quick closing pneumatically-actuated valves mounted upstream and downstream of the recombiner provide for isolation of the recombiner. A third quick opening pneumatically-actuated valve allows for bypass of the recombiner. The isolation valves will fail closed on loss of air supply, and the bypass valve will fail open. There is a quick-closing pneumatically-actuated valve installed on the hydrogen and oxygen supply lines. Both valves will fail closed on loss of supply air.

Those are the six.

MEMBER STETKAR: So, if I lose air, I wind up isolating the -- I was writing notes. The hydrogen and oxygen supply lines fail closed on loss of air?

MR. PEREZ: Yes. There is a quick-closing pneumatically-actuated valve installed on the hydrogen and oxygen supply lines. Both valves will fail closed on loss of air supply.

MEMBER STETKAR: Okay. I will have to go. Thanks.

MR. PEREZ: And, then, this morning while

we were talking about the steam generator tube rupture, I took two items with me. One was to confirm how long we were steaming the generator with a tube rupture.

FSAR Tier 2, Table 15.0.24, Revision 2, shows that we steam the generator for 13.8 minutes, 13.8 minutes.

MEMBER STETKAR: Now, okay, and that is the time that you used in your radiological --

MR. PEREZ: Yes.

MEMBER STETKAR: -- calculation? That is the time from reactor trip until the MSR isolation valve is closed, accounting for the one-minute closure time?

MR. PEREZ: Yes. Yes.

MEMBER STETKAR: Okay, but does it make any difference to your radiological calculation whether you are steaming -- no, it shouldn't -- whether you are steaming or whether the control valve has failed open? No, it's flow rate.

MR. PEREZ: It is flow rate.

MEMBER STETKAR: So, it shouldn't.

MR. PEREZ: Right. We track the mass.

MEMBER STETKAR: Okay. I still have a question about it. So, your calculations uses 13.8

minutes for the atmospheric release?

MR. PEREZ: Directly from that generator.

MEMBER STETKAR: Yes. I still have a question for the scenario I developed, whether that 13.8 minutes you used, which is longer than I thought you were using --

MR. PEREZ: Right.

MEMBER STETKAR: - whether that still bounds the release that might come from an earlier automatic reactor trip with normal operation of the safety injection system and the cooldown, later accounting for manual isolation of that steam generator, of the isolation valve on that steam generator.

I can see how the steam generator automatic isolation signal will come in to close the MSIV, but it is not clear to me that that is going to stop the release until 30 minutes. So, if for some reason we start the release more than, let's see, earlier than -- I can't do the math in my head -- 16.2 minutes into the event, you might get a longer release from my scenario than yours.

MR. PEREZ: And your scenario was through --

MEMBER STETKAR: My scenario was no CVCS.

MR. PEREZ: Right.

MEMBER STETKAR: Say no makeup. Let the reactor trip automatically on low pressurizer pressure, whenever that occurs -- I don't know when that occurs. At that time, you lose offsite power, at which time you start steaming.

MR. PEREZ: Okay.

MEMBER STETKAR: And, then, the question is, do you remain steaming from that steam generator until the operators actively close the main steam relief isolation valve? I don't think it is that simple because I think in automatic the thing actually might cycle. You reset the setpoint higher. You will get to the setpoint at the same time you are trying to blow down the steam generator, which should depressurize it. I am assuming it is going to close; the pressure is going to go back up again.

So, I am not quite sure what the transient looks like. But if you start steaming early enough -- it is getting late in the day for me to do the math in my head -- and if you don't positively isolate that steam generator to stop the steaming until 30 minutes plus whatever nominal time is when the operators can actively intervene, if that duration is longer than the 13.8 minutes that you used in your calculation,

the actual offsite releases might be higher.

MR. PEREZ: Yes.

MEMBER STETKAR: That was my concern. I just don't have any sense whatsoever of what the timing of that other scenario looks like, but I think AREVA said you ran that scenario.

MS. SLOAN: Yes.

MEMBER STETKAR: So, it should be available somewhere.

MS. SLOAN: Yes. We will take a follow-up, John.

MEMBER STETKAR: Yes. But thanks. You have at least clarified they used 13.8 minutes instead of 2 minutes.

MR. PEREZ: Right. And, then, the last item that I took was concerning the letdown, whether we credited filtration or not. And we went back to our calculation, and it turns out it really doesn't matter because the water is cooled prior to leaving the reactor building. So, you do not have iodine flashing. You are basically releasing noble gases.

The other event with the sample line that is only a quarter inch, that receives no cooling and you get iodine flashing. That is why that becomes a limiting event.

MEMBER STETKAR: So, in effect, what you are saying is, although at least the SER seems to place some emphasis on the purification system and the filtering, that is essentially irrelevant.

MR. PEREZ: Correct.

MEMBER STETKAR: It is all the temperature of the release.

MR. PEREZ: Correct.

MEMBER STETKAR: Okay.

MR. PEREZ: Correct.

MEMBER STETKAR: Thanks. That makes sense.

MR. PEREZ: Okay. That concludes what I had.

MS. SLOAN: Thank you.

MR. PEREZ: Okay. Thank you very much.

CHAIRMAN BANERJEE: Thank you very much.

I think what I will do now is to ask the members if they have any comments or questions that they would like to make today.

MEMBER STETKAR: No.

CHAIRMAN BANERJEE: Tomorrow I am sure we will.

John, do you have anything?

MEMBER STETKAR: No. Thank you.

CHAIRMAN BANERJEE: So, I think we can probably break and we will reassemble tomorrow morning at 8:30 and take up the rod ejection.

Will you lead off that? Yes, I imagine, yes.

MS. SLOAN: Yes. Yes, AREVA leads, then the staff, and then we go to the small-break LOCA.

CHAIRMAN BANERJEE: Then we go back to the small-break after lunch.

MS. SLOAN: Yes.

CHAIRMAN BANERJEE: Okay. So, with that, we will break.

Thank you very much. They were very good presentations, very informative. Thanks.

(Whereupon, at 4:28 p.m., the foregoing matter was adjourned.)

Presentation to ACRS U.S. EPR Subcommittee Design Certification Application FSAR Tier 2 Chapter 15

(excluding Section 15.6.5)

EPR™ by AREVA

AREVA

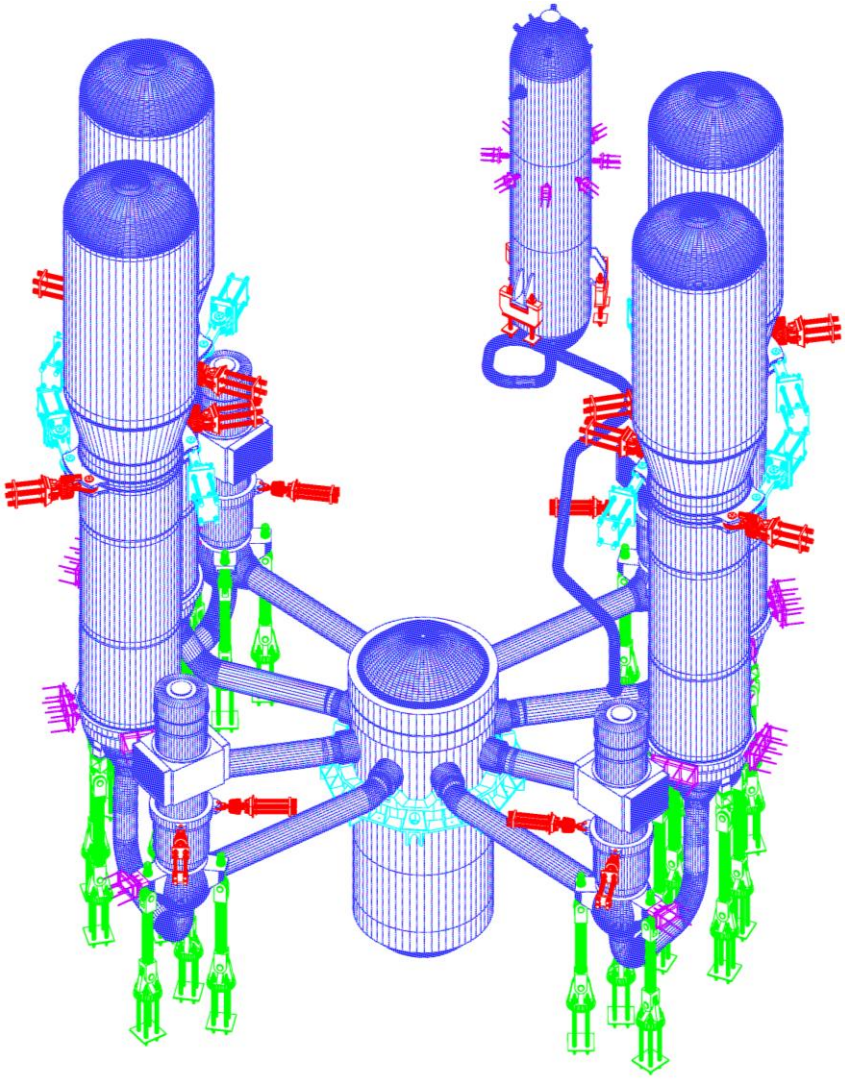
Chapter 15 ACRS Meeting Agenda

▶ Overview of Unique U.S. EPR Features	Douglas Brownson
▶ Trip Functions	Jonathan Witter
▶ Overview of In-Core Transient Methodology	Jonathan Witter
▶ Chapter 15 Event Evaluation	Douglas Brownson
▶ Radiological Evaluation	Pedro Perez
▶ SGTR Mitigation Strategy	Kenneth Coffey

Overview of Unique U.S. EPR Design Features Important to Safety Analysis

Douglas Brownson
Supervisory Engineer, Non-LOCA Analysis

Reactor Coolant System



- **Conventional 4-loop PWR design, proven by decades of design, licensing & operating experience**
- **NSSS component volumes larger compared to existing PWRs; increases operator grace period for many transients and accidents**

Plant Parameter Comparison

Parameter	Current US 4-Loop	Palo Verde	U.S. EPR
Thermal Power, MWth	3411	3998	4590
Electrical Power (Net), MWe	1120	1314	1600
Plant Efficiency, Percent	33	33	35
Number of Fuel Assemblies	193	241	241
Hot Leg Temperature, ^{o}F	616	620	624
Cold Leg Temperature, ^{o}F	559	558	564
Vessel Average Temperature, ^{o}F	588	589	594
Primary System Operating Pressure, psia	2250	2250	2250
Reactor Coolant Flow per Loop, gpm	100,500	111,000	125,000
Average Coolant Flow per Assembly, gpm	2083	1842	2075
Core Average Linear Heat Rate, kW/ft	5.6	5.6	5.2
Peak Linear Heat Rate, kW/ft	14.5	14.6	13.6

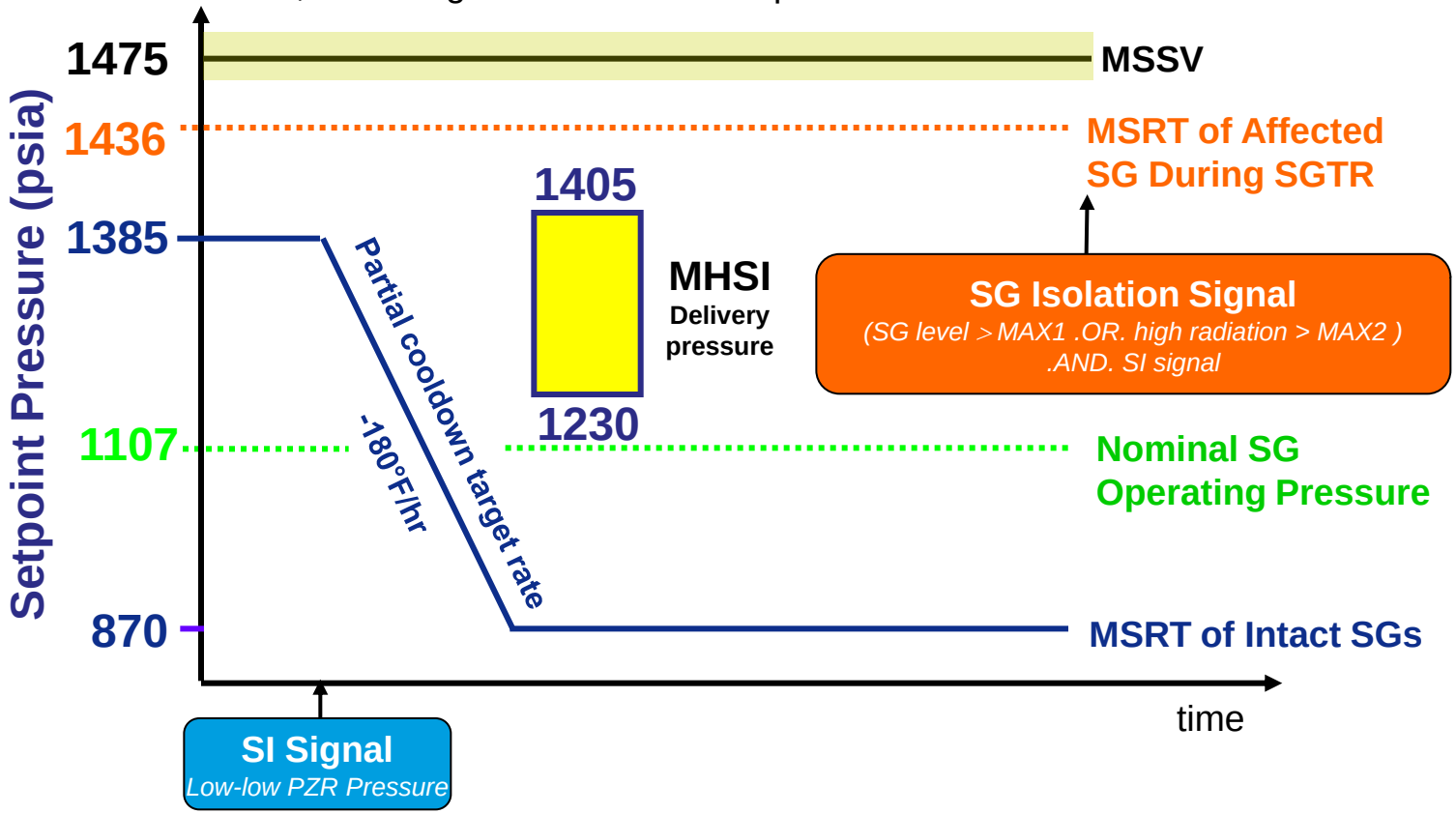
Core design parameters similar to current operating plants

Unique Safety Related Systems and Features

- Four train front line safety systems
- Automatic partial cooldown of steam generators (SGs) on safety injection system (SIS) actuation signal
- Automatic trip of reactor coolant pumps (RCPs) on coincident SIS actuation signal and low delta-pressure across the pumps
- Low DNBR and High Linear Power Density (LPD) trip functions using incore measurements of local core power distributions
- In-Containment Refueling Water Storage Tank (IRWST)
  - Source of emergency core cooling system (ECCS) water
  - No switchover needed to external water source
- Extra Borating System (EBS)
- Safety-related alarm on high activity in steam lines (SGTR)
- Steam generator blowdown / transfer line

Main Steam Relief Train (MSRT) Operation

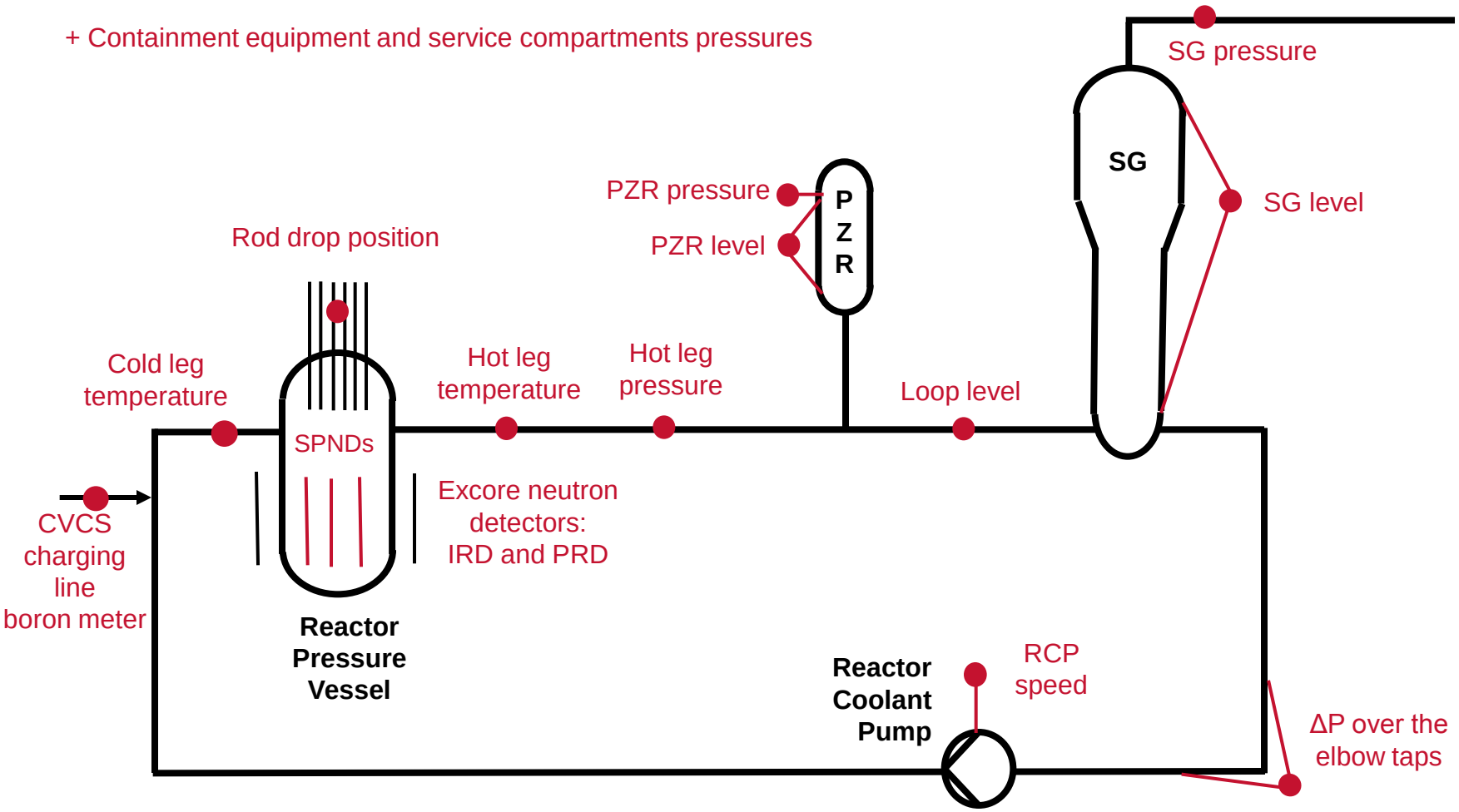
- **Four train, safety-related system (one per SG)**
- **MSRT provides two functions:**
  - **Overpressure protection**
  - **Depressurizes SGs automatically on SIS actuation signal**
    - Depressurization rate equivalent to 180 F/hr (T_{sat})
    - Ensures adequate medium head safety injection (MHSI) flow for SBLOCA & SGTR
    - For SGTR, re-set high in affected SG to prevent relief



U.S. EPR
Trip Functions

Jonathan Witter, Ph.D.
Advisory Engineer, Core Thermal Hydraulics

Protection System Functions - Instrumentation



I&C sensors used in the Protection System

Protection System Functions – Core Related

- High Core Power Level (Loop TH enthalpy based) — 105% (10.2%)
- Low Hot Leg Saturation Margin (Loop TH enthalpy based) — 30 BTU/lbm
- High Neutron Flux Rate of Change (PRD) — 11% (2%)
- High Neutron Flux (IRD) — 25% (10%)
- Low Doubling Time (IRD) — 20 s (±10 s)
- Linear Power Density (SPND)
  - Limiting Condition for Operation — 385/350 W/cm
  - High Linear Power Density RT (2nd Max) — 460 W/cm
- Low DNBR (SPND and Loop TH inputs)
  - Limiting Condition for Operation — 2.50
  - Symmetric RT (2nd Min) — 1.95
  - Imbalance/Rod Drop ¼ Division (1st Min) — 2.10
  - Rod Drop ≥2/4 Division (1st Min) — 3.30
- Anti Boron Dilution CVCS Isolation
  - At power condition — Function of burnup
  - In standard shutdown states — Function of burnup and temperature
  - In shutdown with no RCPs running — 927 ppm (37 a/o B-10)

Protection System Functions – RCS Related

▶ Low-Low RCS Flow Rate (one loop)	54% (4%)
▶ Low RCS Flow Rate (≥two loops)	90% (4%)
▶ Low RCP Speed	93% (1%)
▶ Low Hot Leg Pressure	2005 psia (45 psi)
▶ High Pressurizer Water Level	75% (5.5%)
▶ High Pressurizer Pressure	2414.7 psia (25 psi)
▶ Low Pressurizer Pressure	2005 psia (25 psi)
▶ High Containment Pressure	4.0 psig (± 0.5 psi)

Protection System Functions – Secondary System Related

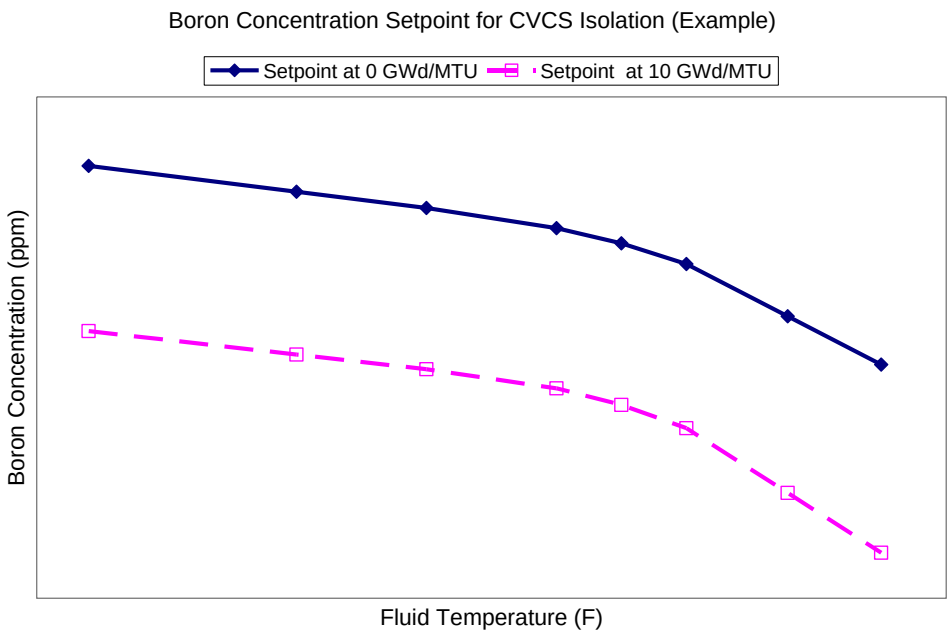
- Low Steam Generator Pressure 724.7 psia (± 30 psi)
- High Steam Generator Pressure 1384.7 psia (± 30 psi)
- Low Steam Generator Level 20% (± 3.5%)
- High Steam Generator Level 69% (± 9.5%)
- High Steam Generator Pressure Drop
  - {(-29 psi/min) .OR. (102 psi < steady state) .AND. (Maximum of 1088 psia)}

Principles of High Core Power Level and Low Saturation Margin Functions

- **High Core Power Level (HCPL)**
  - **Calculates a core power in each loop based on cold and hot leg temperatures and hot leg pressure**
  - **Reactor trip (RT) initiated once 2/4 loops indicate power above the RT threshold**
    - Main purpose is to serve as backup trip for overcooling and RCCA withdrawal events and to limit power to analyses inputs for the incore SPND based trips.
- **Low Saturation Margin**
  - **Calculates an enthalpy margin to saturated coolant conditions in each loop using the temperature and pressure of the hot leg**
  - **RT initiated once 2/4 loops indicate margin less than the RT threshold**
    - Main purpose is to protect the validity of the calculation of the HCPL

Principles of Anti Boron Dilution CVCS Isolation Function

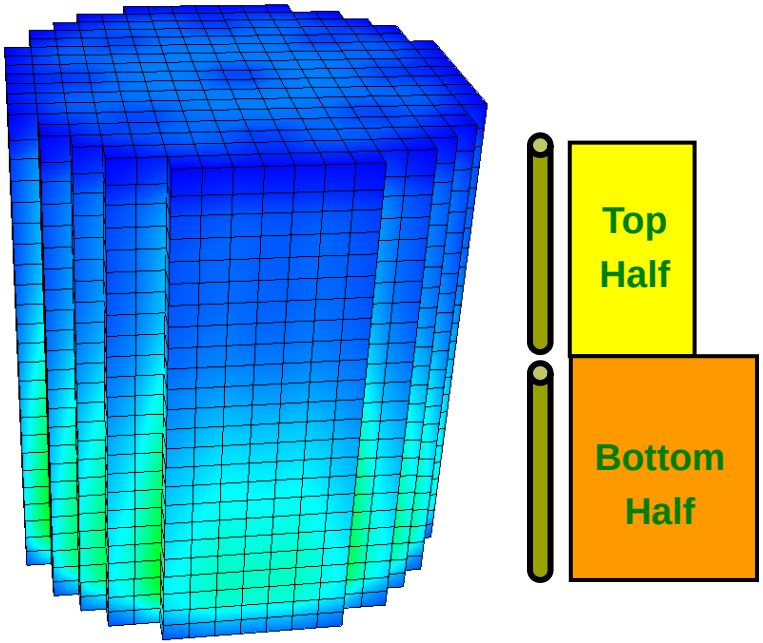
- Continuous boron concentration monitoring on charging line
- Mitigates boron dilution event by isolating the charging pump suction from the volume control tank
- Setpoints
  - At Power is set for an inlet temperature and are updated at periodic burnups
  - In Standard Shutdown is a function of temperature updated at periodic burnups
  - In Shutdown with no RCPs running, fixed value based on IRWST concentration

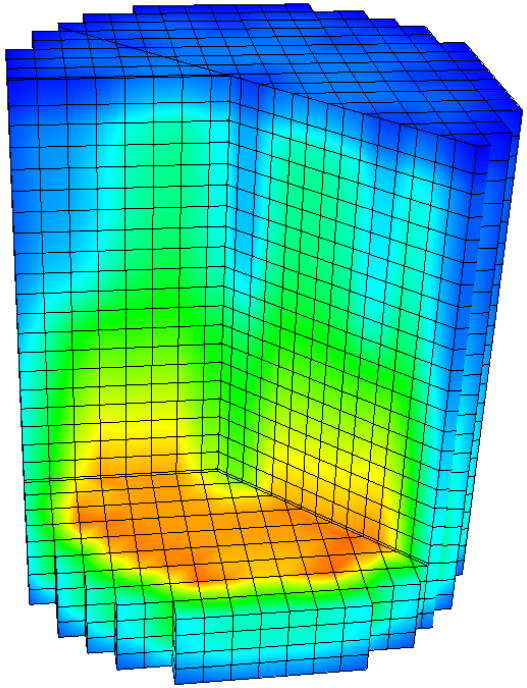


CVCS isolation stops boron dilutions with adequate time to maintain shutdown margin at power and prevent inadvertent criticality in shutdown conditions

Measuring, Instead of Inferring, the 3D Core Power Distribution

- **Excore neutron detectors**
  - **Monitors periphery of core**
  - **Coarse axial power shape resolution**
  - **Use of control rod positions to provide inferred core internal power**



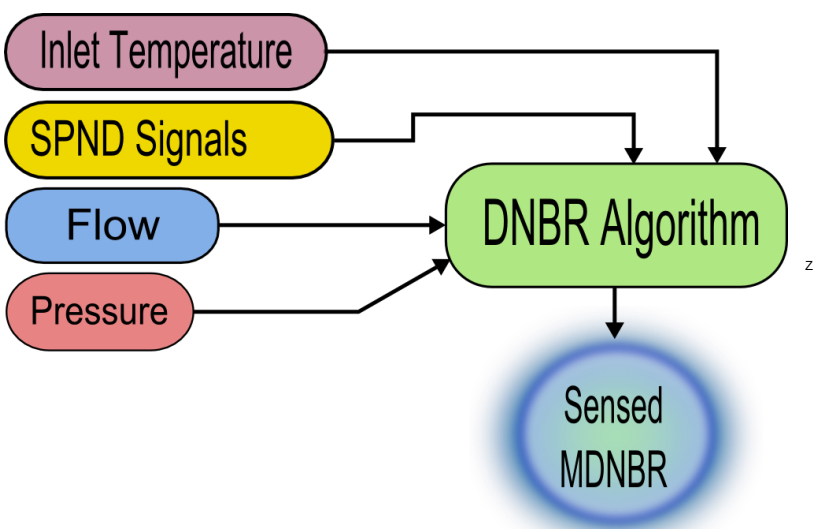


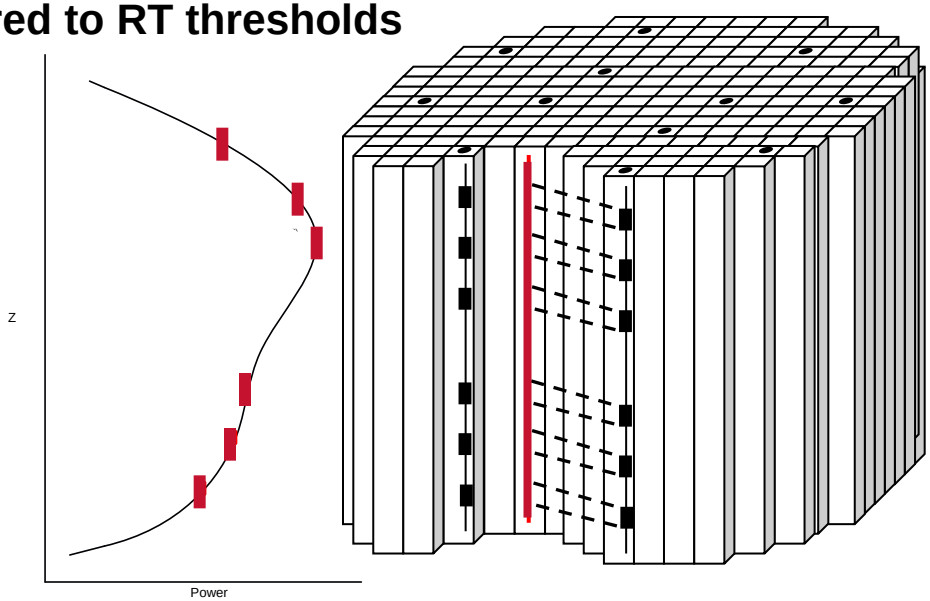
- **Incore detector system**
  - **Monitors local power**
  - **Directly measure radial and axial power distribution effects**

Real time resolution of LPD distribution uses 3D array of the 72 SPND readings

Principles of SPND Calibration for Low DNBR Function

- **DNB Ratio Monitoring (one PS division)**
  - **Each of the 12 SPND strings is calibrated to indicate the LPD axial distribution of the DNBR limiting "hot channel"**
  - **Uses the 6 elevations to reconstruct the power shape/enthalpy rise**
  - **DNBR estimated using reconstructed power distribution and thermal hydraulic boundary conditions from each RCS loop**
  - **12 readings ranked and compared to RT thresholds**

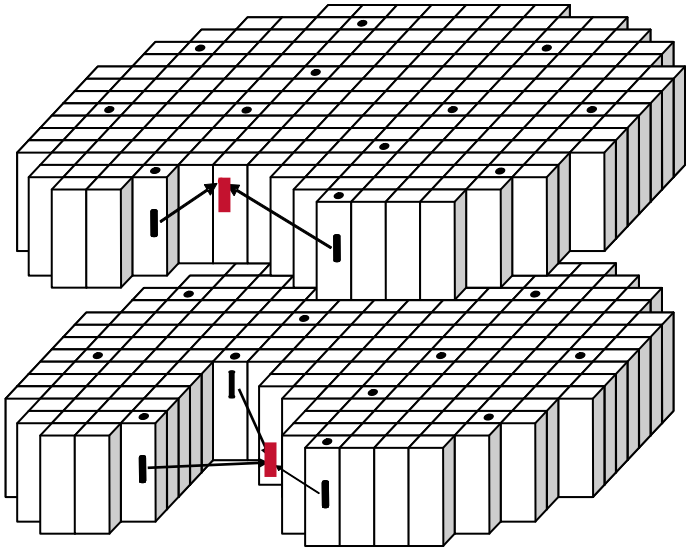




Single calibration gives conservative measure of non-limiting locations. Changes from the calibrated state are seen from the limiting location condition

Principles of SPND Calibration for High Linear Power Density (LPD) Function

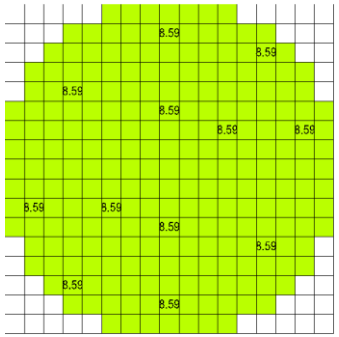
- At the time of calibration with the AMS flux measurement, the "hot spot" in each axial zone is determined
- All 12 SPNDs in an axial elevation are calibrated to read the maximum heating rate in that coverage zone
- Conservative approach since zones away from hot spot will respond as if they were the hot spot
  - Global power, rod influence, xenon transient, etc.
- All SPNDs readings are ranked and compared to the RT threshold
  - RT setpoint is conservatively lowered based on the number of failed SPNDs to cover the reduced accuracy in the resolution of the LPD measurement



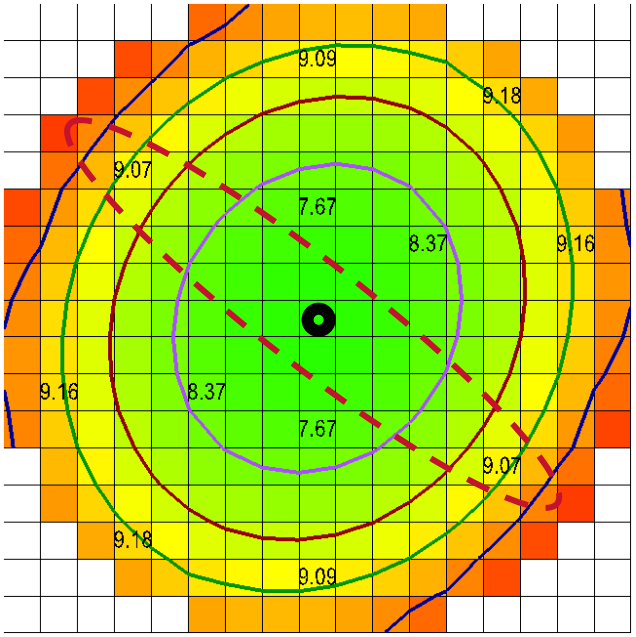
Single point calibration is conservative and more tolerant to SPND failures and facilitates the resolution of power asymmetries

Results of the Ability to Directly Measure Power Shape Changes

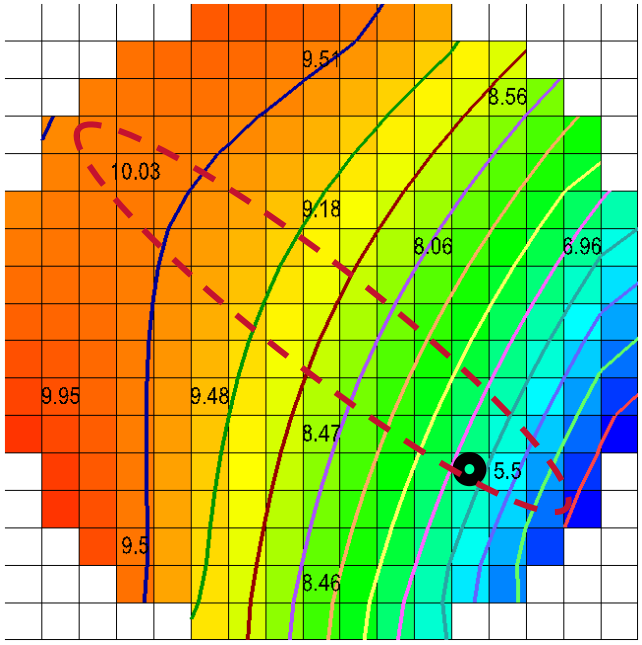
- The array of SPNDs are used to respond to radial and axial shape redistributions regardless of event symmetry
- Can resolve movement of the incore hot spot induced from rods, flow, Xe, etc.
- If imbalance is significant enough, it will trigger use of a more conservative trip threshold for MDNBR Protection (Imbalance/RD(1/4) and RD(2/4) RTs)



Calibration State, 8.6 kW/ft



Rod J-09 Drop - Symmetric

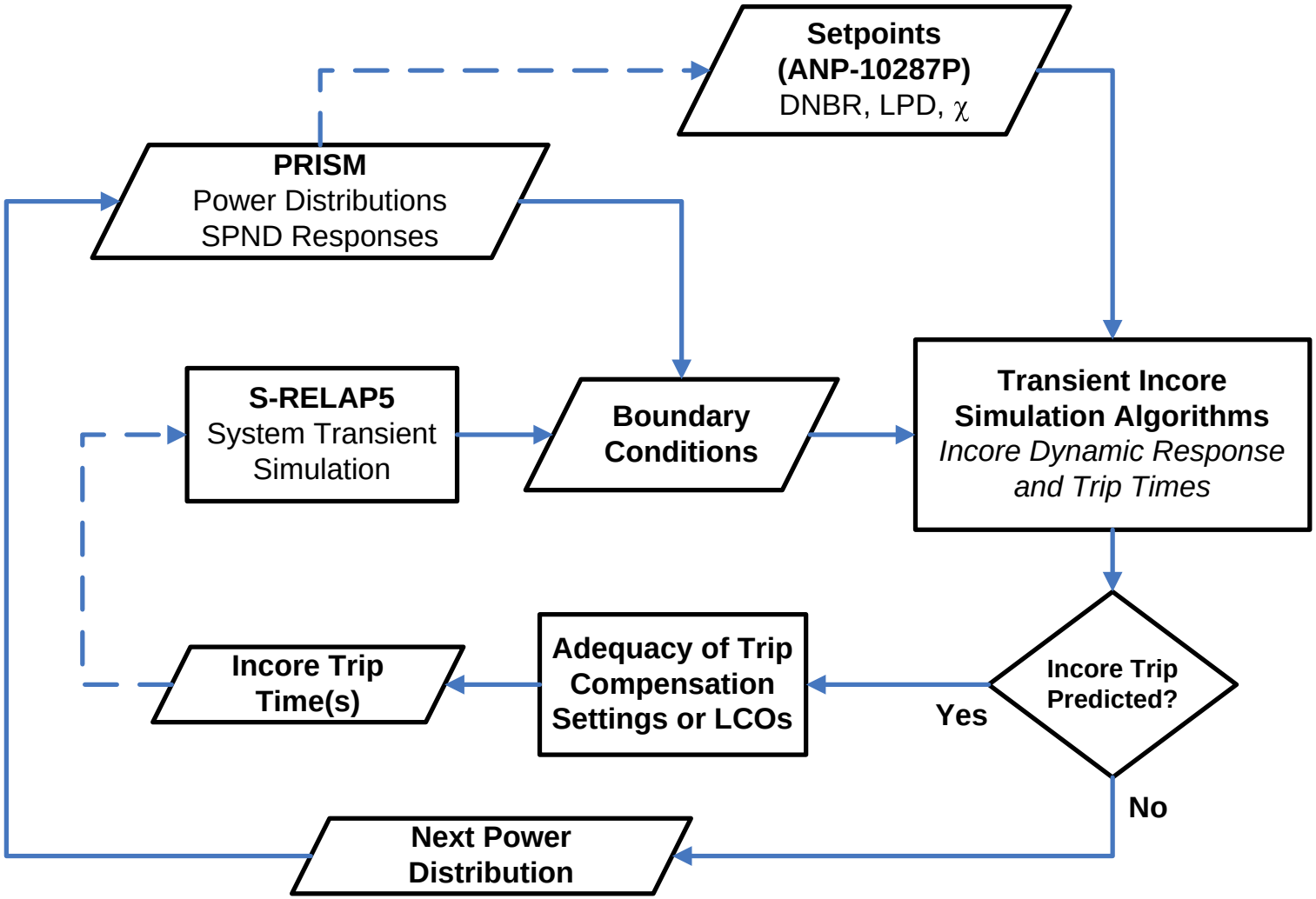


Rod N-07 Drop - Asymmetric

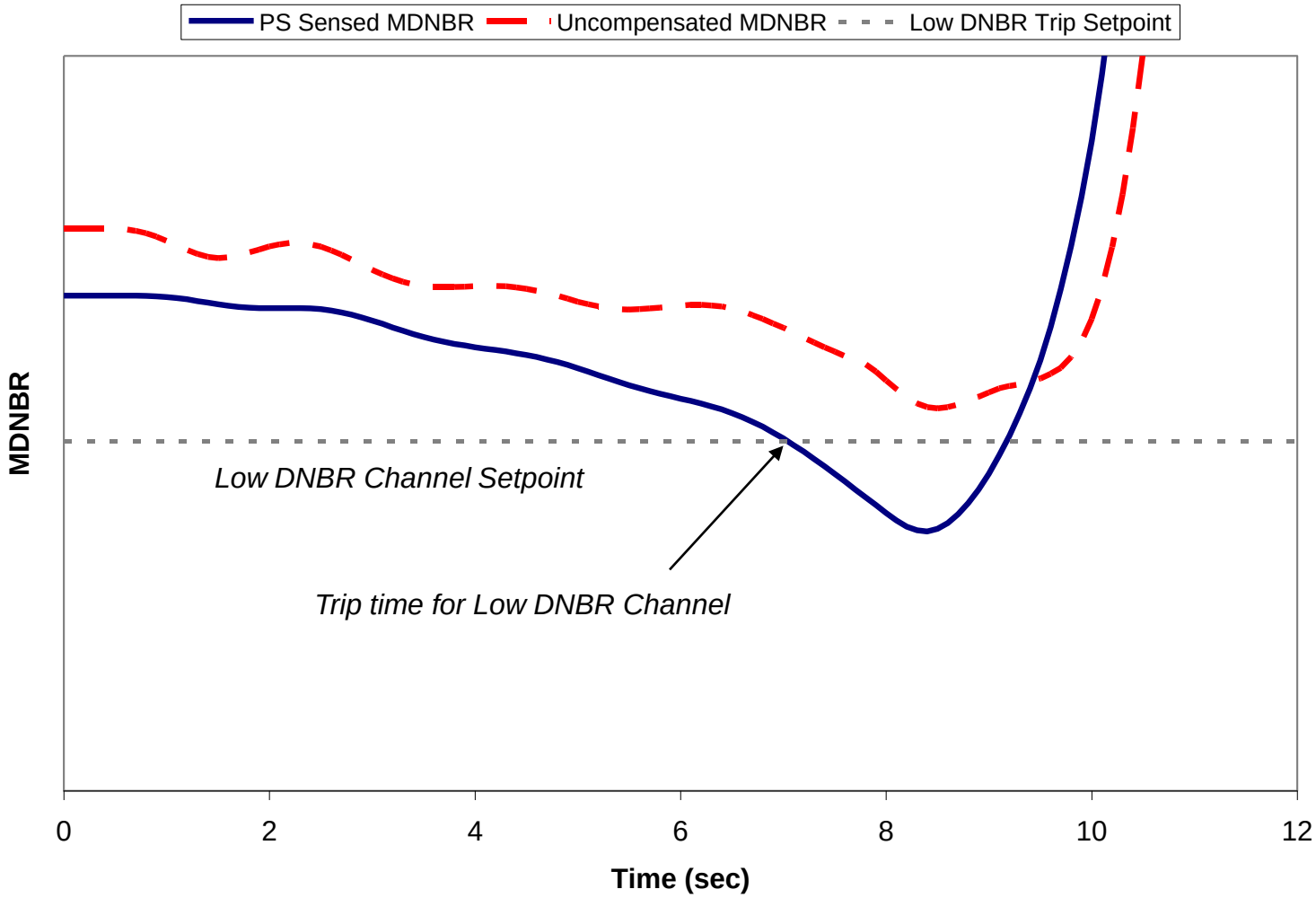
Overview of the U.S. EPR Incore Transient Methodology

Jonathan Witter, Ph.D.
Advisory Engineer, Core Thermal Hydraulics

Incore Monitoring Reactor Trip Evaluation

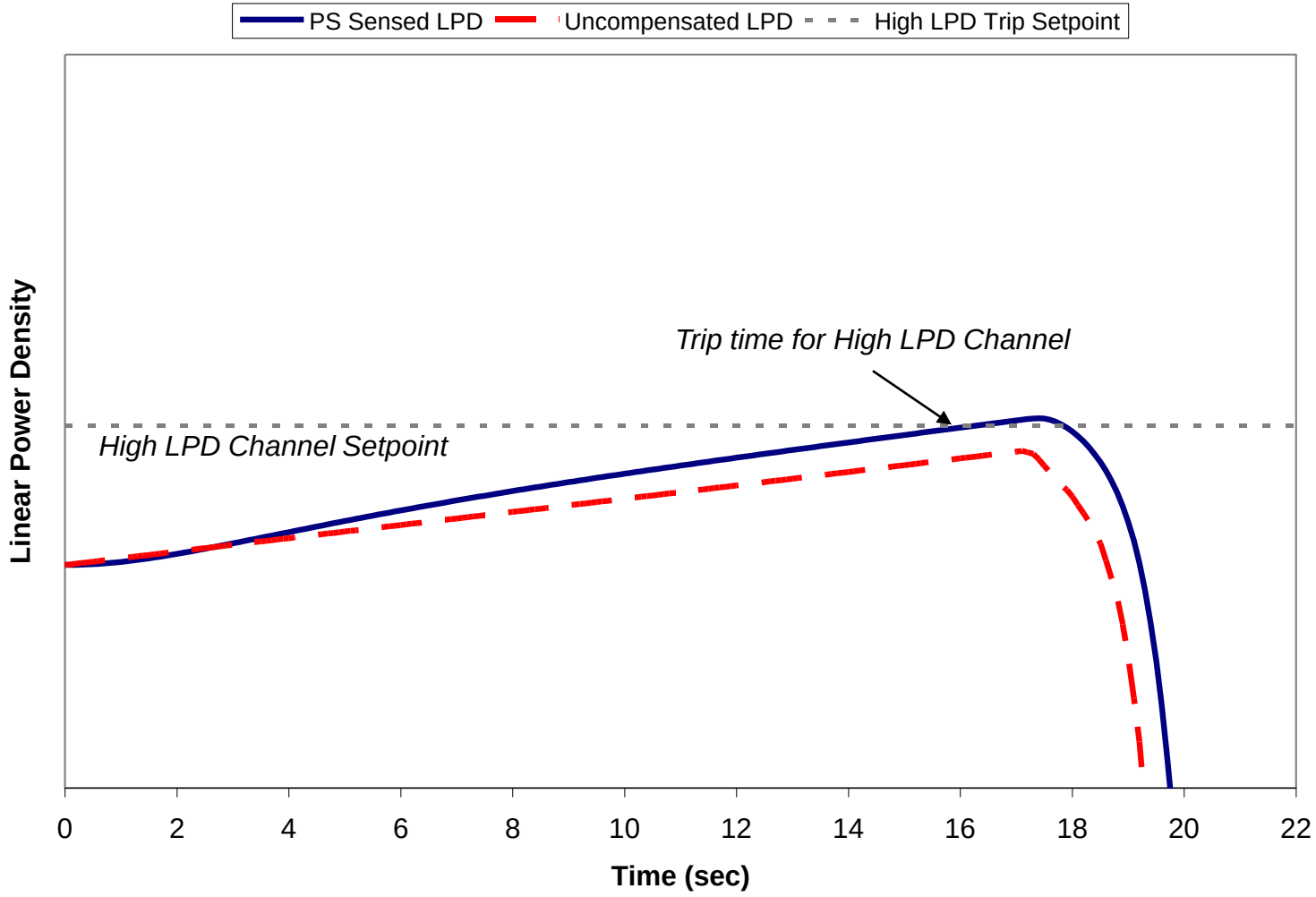


MDNBR Undershoot Assessment from FSAR 15.1.3 Increase in Steam Flow Event



Criterion for success is that the uncompensated static MDNBR does not violate the setpoint

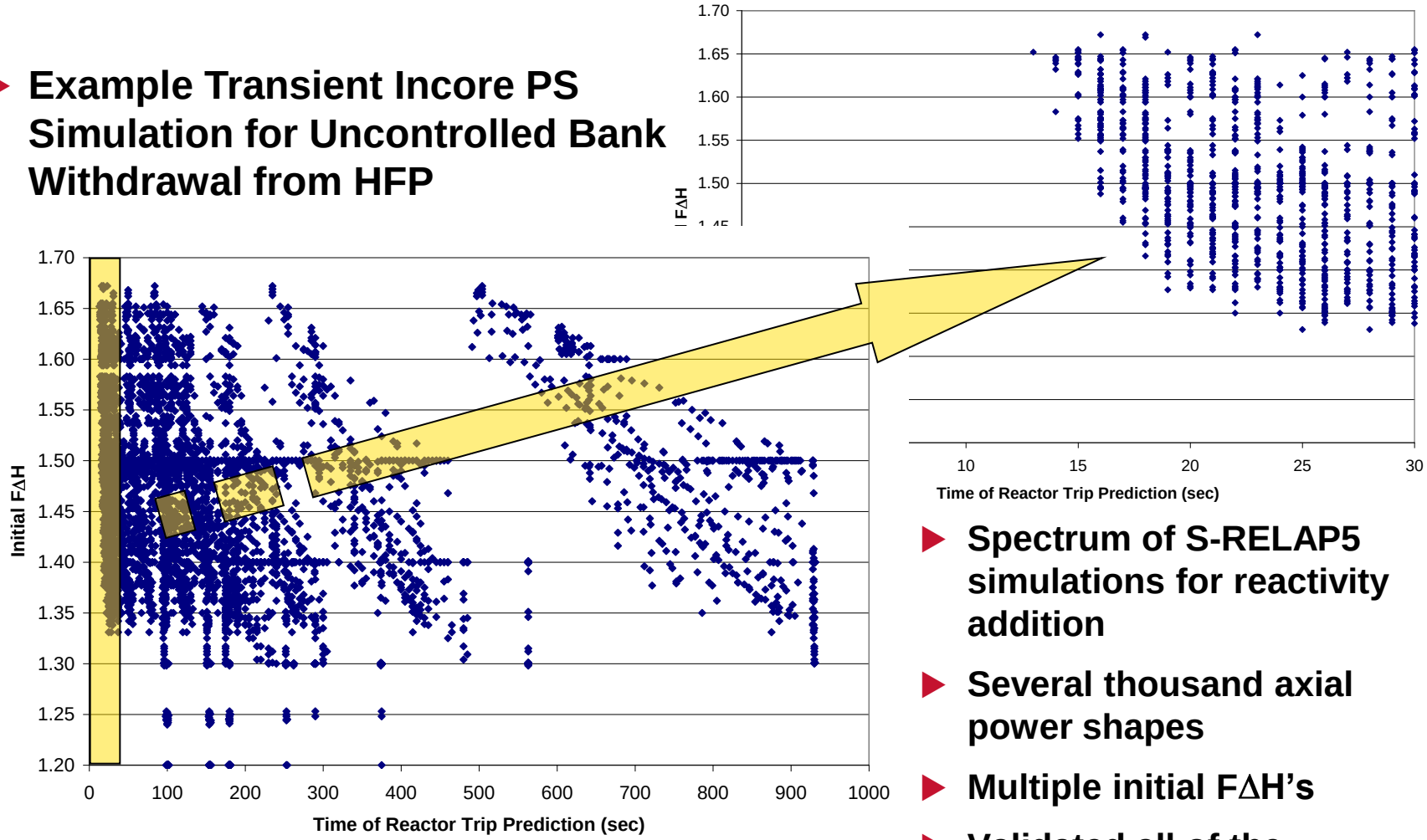
LPD Overshoot Assessment from FSAR 15.4.2 Uncontrolled Bank Withdrawal Event



Criterion for success is that the uncompensated static LPD does not violate the setpoint

Broad Spectrum of Trip Times Demonstrate Protection of SAFDLs and Plant Systems

- **Example Transient Incore PS Simulation for Uncontrolled Bank Withdrawal from HFP**



- **Spectrum of S-RELAP5 simulations for reactivity addition**
- **Several thousand axial power shapes**
- **Multiple initial FΔH's**
- **Validated all of the predicted incore trips**

Broad Spectrum of Trip Times Demonstrate Protection of SAFDLs and Plant

- **Each event analyzed results in a spectrum of times for reactor trip based on incore SPND monitoring (thousands of renditions)**
- **The shortest time to trip and highest rate of signal change are most adverse for MDNBR undershoot and LPD overshoot**
  - **The LCO's for MDNBR and LPD are specified to protect the SAFDL in the events that are too fast or severe to be covered with the incore monitoring system**
- **Some combinations of 3D power shapes, initial conditions, and magnitude of the plant transient result in no trip demanded by the incore system**
  - **For cases not exceeding the incore reactor trip threshold, other plant trips will terminate the event if required**

All Chapter 15 non-LOCA AOO events demonstrate adequate protection

SAFDL Events Protected by the Incore SPND-based Reactor Trips

- SRP 15.1.1 Decrease in feedwater temperature (DNB)
- SRP 15.1.2 Increase in feedwater flow (DNB)
- SRP 15.1.3 Increase in steam flow (DNB/LPD)
- SRP 15.1.4 Inadvertent opening of a steam generator relief or safety valve (DNB)
- SRP 15.4.2 Uncontrolled control rod assembly withdrawal at power (DNB/LPD)
- SRP 15.4.3 Control rod misoperation (DNB/LPD)
- SRP 15.4.6 Inadvertent decrease in boron concentration in RCS (DNB/LPD)
- SRP 15.6.1 Inadvertent opening of a pressurizer relief or safety valve (DNB/LPD)

All events demonstrate the adequacy of the dynamic compensation on the trips

SAFDL Events Protected by Other System Trips and the DNBR or LPD LCOs

- **SRP 15.1.3 Increase in steam flow (DNB/LPD)**
  - ***High SG Pressure Drop***
- **SRP 15.3.1 Loss of forced reactor coolant flow (partial loss) (DNB)**
  - ***Bounded by the full loss of forced RCP flow***
- **SRP 15.3.2 Loss of forced reactor coolant flow (full loss)**
  - ***DNB event that sets the DNBR LCO to ensure adequate protection***
- **SRP 15.4.2 Uncontrolled control rod assembly withdrawal at power (DNB/LPD)**
  - ***High Core Power Level Trip and LCO's***
- **SRP 15.4.1 Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition (DNB/LPD)**
  - ***High Neutron Flux Rate of Change Trip terminates the event***
- **SRP 15.4.8 Spectrum of postulated rod ejection accidents (DNB/LPD)**
  - ***Follows the SRP 15.4.8 and 4.2 Guidance for accident analysis and fuel failure as described in ANP-10286P***

Events that evolve too quickly for incore trips are protected by combination of LCOs and/or other plant trips

U.S. EPR Tier 2 FSAR Non-LOCA Chapter 15 Event Evaluation

(excluding Section 15.6.5)

Douglas Brownson
Supervisory Engineer, Non-LOCA Analysis

Non-LOCA Methodology

- Conservative, deterministic methodology
- Follows the NUREG-0800 Standard Review Plan (SRP) and Regulatory Guide 1.206
- Methodology updated for U.S. EPR to obtain initial fuel conditions from COPERNIC code rather than RODEX2A
- S-RELAP5 provides system fluid boundary conditions for external DNBR, fuel centerline melt and radiological evaluations
- In accordance with regulatory guidelines
  - Non-safety related system functionality modeled only when detrimental to outcome
  - Key parameters, such as protection system setpoints and ECCS performance, are biased for conservatism and uncertainty in accordance with Regulatory Guide 1.105
- Methodology considerations:
  - Single failure
  - Loss-of-offsite power (LOOP)
  - Operator actions (after 30 minutes)
  - Axial and radial power distributions
  - Preventative maintenance

Tier 2 FSAR Chapter 15 Events

- Radiological Consequences of Design Basis Accidents
- 15.1 Increase in Heat Removal by the Secondary System
  - 15.1.1 Decrease in Feedwater Temperature (AOO)
  - 15.1.2 Increase in Feedwater Flow (AOO)
  - 15.1.3 Increase in Steam Flow (AOO)
  - 15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve (AOO)
  - 15.1.5 Steam System Piping Failures Inside and Outside Containment (PA)
- 15.2 Decrease in Heat Removal by the Secondary System
  - 15.2.1 Loss of External Load (AOO)
  - 15.2.2 Turbine Trip (AOO)
  - 15.2.3 Loss of Condenser Vacuum (AOO)
  - 15.2.4 Inadvertent Main Steam Isolation Valve Closure (AOO)
  - 15.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries (AOO)
  - 15.2.7 Loss of Normal Feedwater Flow (AOO)
  - 15.2.8 Feedwater Line Breaks Inside and Outside Containment (PA)
- 15.3 Decrease in RCS Flow Rate
  - 15.3.1 Partial Loss of Forced Reactor Coolant Flow (AOO)
  - 15.3.2 Complete Loss of Forced Reactor Coolant Flow (AOO)
  - 15.3.3 Reactor Coolant Pump Rotor Seizure (PA)
  - 15.3.4 Reactor Coolant Pump Shaft Break (PA)

Tier 2 FSAR Chapter 15 Events (Cont'd)

- **15.4 Reactivity and Power Distribution Anomalies**
  - **15.4.1 Uncontrolled Control Rod Assembly Withdrawal from Subcritical or Low-Power Startup Condition (AOO)**
  - **15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power (AOO)**
  - **15.4.3 Control Rod Misoperation (System Malfunction or Operator Error) (AOO)**
  - **15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (AOO)**
  - **15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (AOO)**
  - **15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (AOO)**
  - **15.4.8 Spectrum of Rod Ejection Accidents in a PWR (PA)**
- **15.5 Increase in RCS Inventory**
  - **15.5.1 Inadvertent Operation of ECCS or EBS (AOO)**
  - **15.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (AOO)**
- **15.6 Decrease in RCS Inventory**
  - **15.6.1 Inadvertent Opening of a Pressurizer Safety Relief Valve (AOO)**
  - **15.6.3 Steam Generator Tube Rupture (PWR) (PA)**
  - **15.6.5 Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary (PA)**
- **15.8 Anticipated Transients Without Scram**

Increase in Heat Removal (15.1)

- **Increase in Steam Flow (AOO)**
  - Includes a spectrum of events covering the range up to all 6 turbine bypass valves opening (60% full flow) at various initial powers
  - Protected by Low DNBR trip
- **Steam System Piping Failures (PA)**
  - Includes a spectrum (break size, initial power, for both pre- and post-RT conditions)
    - Pre-RT
      - FSAR break sizes considered 10%, 50%, and 100% of steam line area
      - RT occurs on High Core Power Level, Low DNBR, or High SG Pressure Drop
      - Timely RT occurs and acceptance criteria satisfied
    - Post-RT
      - FSAR considers spectrum of break sizes and initial power levels including HFP and HZP
      - Limiting case is HZP with a return to power with some fuel damage
      - Offsite doses are within acceptance criteria

Decrease in Heat Removal (15.2)

- **Turbine Trip (AOO)**
  - **Single failure of MSRT to open**
  - **Limiting RCS overpressure event**
    - Peak Calculated RCS Pressure = 2785 psia
    - Acceptance Criteria = 110% Design = 2803 psia
- **Inadvertent Single MSIV Closure (AOO)**
  - **Single failure of MSRT to open; MSSV preventative maintenance**
  - **Limiting Secondary overpressure event**
    - Peak Calculated Steam Generator Pressure = 1567 psia
    - Acceptance Criteria = 110% Design = 1593 psia
- **Feedwater Line Break (PA)**
  - **Design Transient for EFW Capacity**
    - Spectrum of FWLB sizes considered
    - Two EFW pumps to two intact SGs sufficient to satisfy acceptance criteria
    - Assume operator realigns EFW flow from failed SG to an intact SG with an inoperable EFW train at 30 minutes and trips the RCPs to the affect SG and SG loop with the single failure.

Decrease in RCS Flow Rate (15.3)

- **Complete Loss of Forced Reactor Coolant Flow (AOO)**
  - **Single failure of one PS division; preventative maintenance on a second PS division**
  - **RT on low RCP speed signal**
  - **Limiting ΔDNBR event that does not trip on DNB**
  - **Sets LCO on Initial DNBR**
- **Reactor Coolant Pump Rotor Seizure (PA)**
  - **RT on low-low flow in affected loop**
  - **Conservatively predicts 8% of fuel rods undergo DNB-induced failure**
  - **Radiological dose less than 10CFR100 criteria assuming a bounding value of 9.5% fuel failure**

Reactivity and Power Distribution Anomalies (15.4)

- **RCCA Drop (AOO)**
  - **Single RCCA and complete bank drop analyzed**
  - **RCCA drop results in rod withdrawal when average coolant temperature (ACT) control function active**
  - **Low DNBR LCO and RT provides protection for DNB**
  - **HLPD limits are not exceeded**
- **RCCA Ejection (PA)**
  - **Analyzed for core response and RCS overpressure**
  - **Core Analysis**
    - Spectrum of ejected rod worths and initial power levels from HZP to HFP
    - Event acceptance criteria satisfied
  - **Overpressure Analysis**
    - No opening in the RCS assumed
    - RCS pressure well below 120% of design
    - MSSVs do not open

Increase in RCS Inventory (15.5)

- **Inadvertent Operation of ECCS or EBS (AOO)**
  - **ECCS Operation**
    - MHSI shut-off head (1405 psia) below operating pressure (2250 psia)
    - Inconsequential for U. S. EPR
  - **EBS Operation**
    - Single failure of one MSRT to open
    - RT on high pressurizer level
    - EBS manually isolated at 30 minutes
  - **Event acceptance criteria satisfied**
- **CVCS Malfunction (AOO)**
  - **Both charging pumps operate with letdown isolated**
  - **Single failure of one MSRT to open**
  - **RT an isolation of CVCS on high pressurizer level**
  - **Event acceptance criteria satisfied**

Decrease in RCS Inventory (15.6)

- **Inadvertent Opening of a PSRV (AOO)**
  - One PSRV (out of three) inadvertently opens and sticks open
  - Limiting single failure is EDG fails to start (loss of SIS and EFW pumps); additional EDG assumed unavailable due to preventative maintenance (loss of second SIS and EFW pumps)
  - RT on low pressurizer pressure
  - Two MHSI pumps sufficient to offset RCS inventory loss
  - Operator action at 30 minutes credited to align flow from two operational EFW trains to all four SGs
  - Acceptance criteria satisfied and core adequately cooled throughout event
- **Steam Generator Tube Rupture (PA)**
  - Limiting FSAR case is with LOOP
  - Single failure of main feedwater full load control valve assumed
  - Manual RT at 30 min
  - Radiological and overfill acceptance criteria are satisfied

Anticipated Transients Without Scram (15.8)

- An ATWS occurs when the control rods fail to insert following an AOO as a result of:
  - A mechanical blockage of the control rods, or
  - An electrical or mechanical failure of the protection system (PS)
- Mechanical blockage of the control rods is determined to be an insignificant contributor to the probability of an ATWS
- In the event of PS failure an independent, diverse actuation system (DAS) initiates a RT, turbine trip, and other essential safety systems and functions
- The acceptance criteria of 10CFR50.62 are satisfied

U.S. EPR FSAR Section 15.0.3 Radiological Consequences of DBAs

Pedro Perez, P.E.
Supervisory Engineer, Radiological Engineering

Radiological Consequences of DBAs

General

The U.S. EPR design basis accident (DBA) radiological evaluations are based on the following:

- SRP Section 15.0.3, other SRP sections related to specific aspects of a given evaluation, and SRP Section 4.2, *Interim Acceptance Criteria and Guidance for the Reactivity Initiated Events*
- Regulatory Guide 1.183 - alternative source term (AST) methodology
- Regulatory Issue Summary (RIS) 2006-04, *Experience with Implementation of Alternative Source Terms*
- AST radiological acceptance criteria (10CFR50 and 10CFR52)
- Regulatory Guide 1.145 (Offsite atmospheric dispersion factors)
- Regulatory Guide 1.194/ARCON-96 (Onsite dispersion factors)

Radiological Consequences of DBAs (cont'd)

Evaluated DBAs

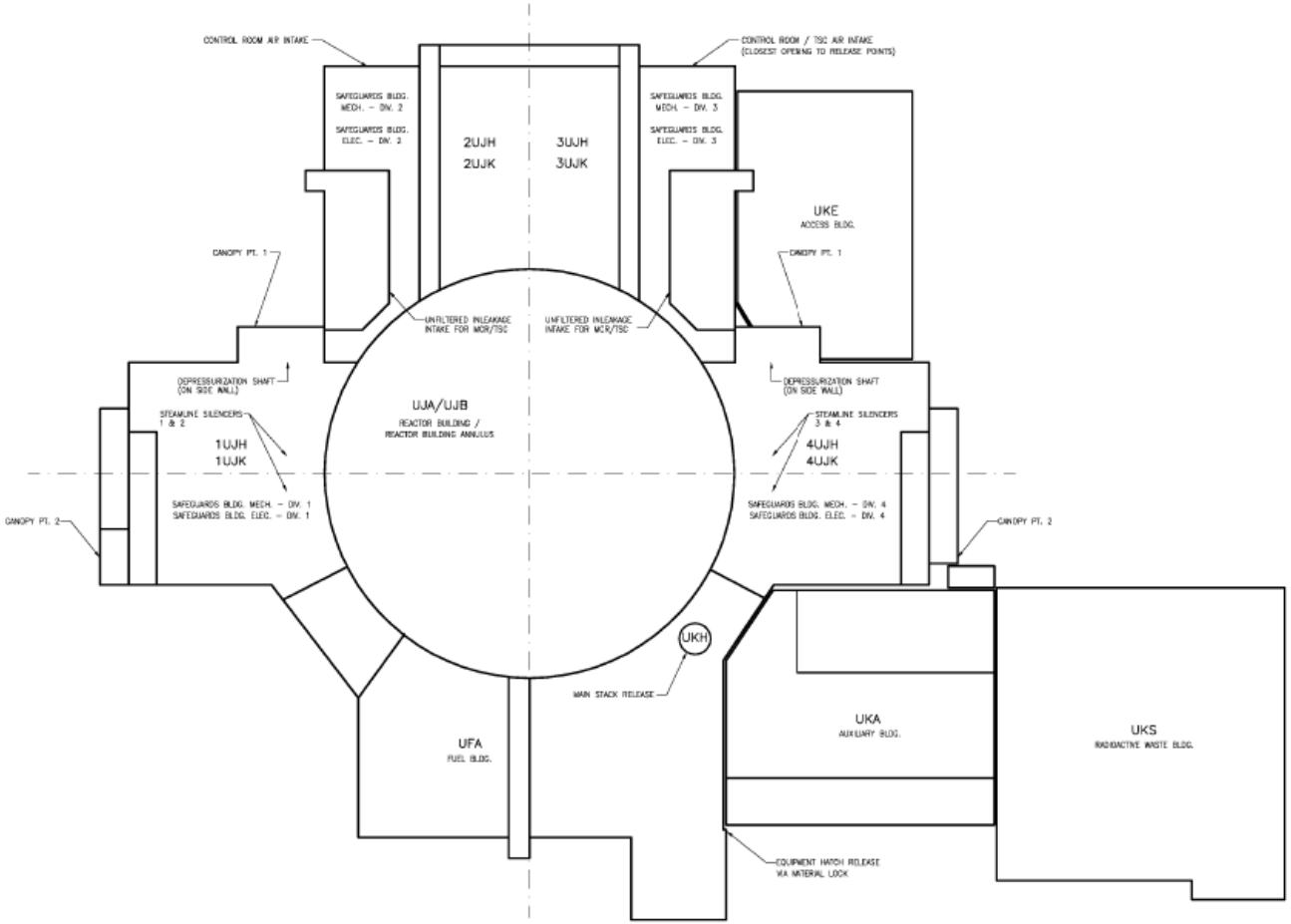
The radiological consequences were evaluated for the following DBAs:

- Small Line Break (SLB) outside of the Reactor Building (RB)
- Steam Generator Tube Rupture (SGTR)
- Main Steam Line Break (MSLB) outside of the RB
- RCP Locked Rotor Accident (LRA)
- Rod Ejection Accident (REA)
- Fuel Handling Accident (FHA)
- Loss of Coolant Accident (LOCA)

The RCP Broken Shaft and the Feedwater Line Break (FWLB) radiological consequences are bounded by the LRA and MSLB, respectively.

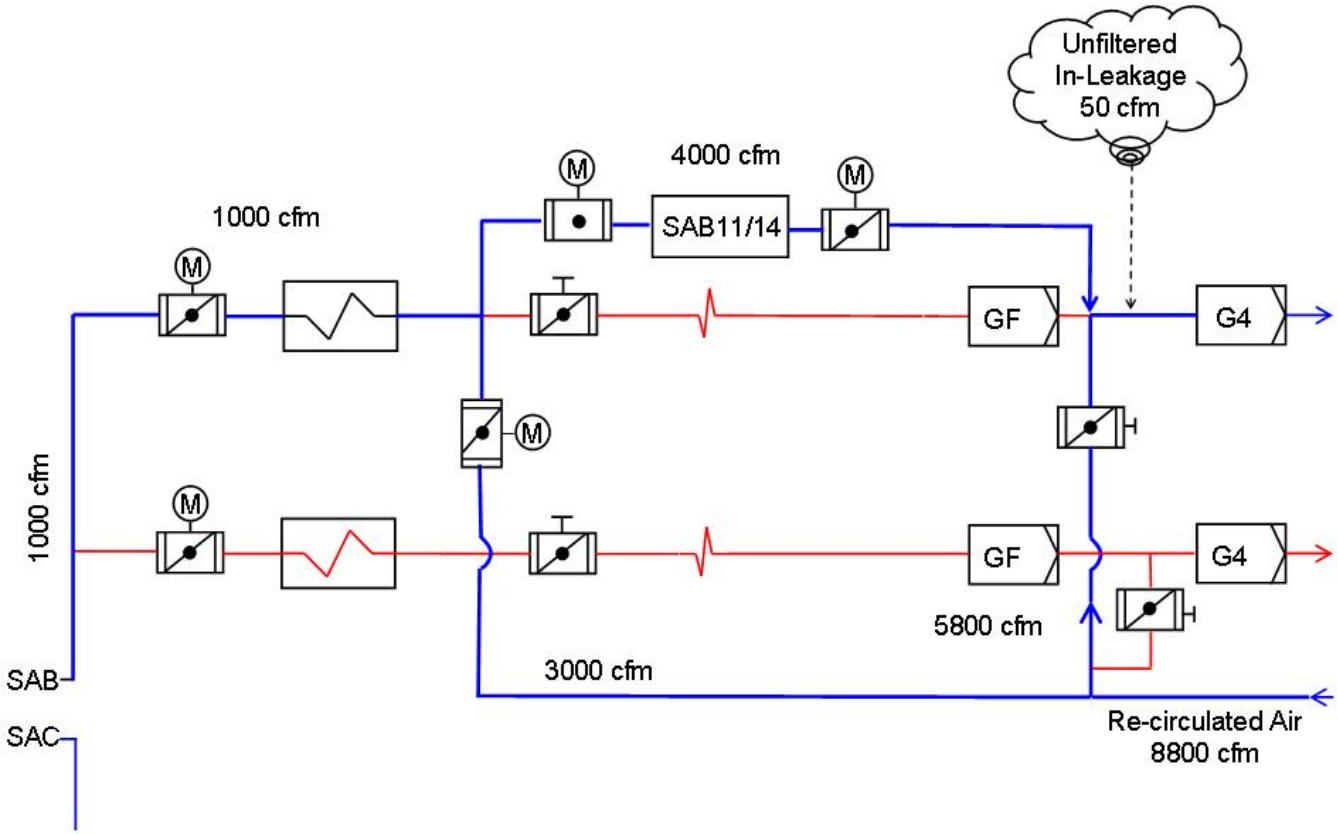
Radiological Consequences of DBAs (cont'd)

- **Release points to atmosphere and main control room (MCR) filtered and unfiltered intakes**



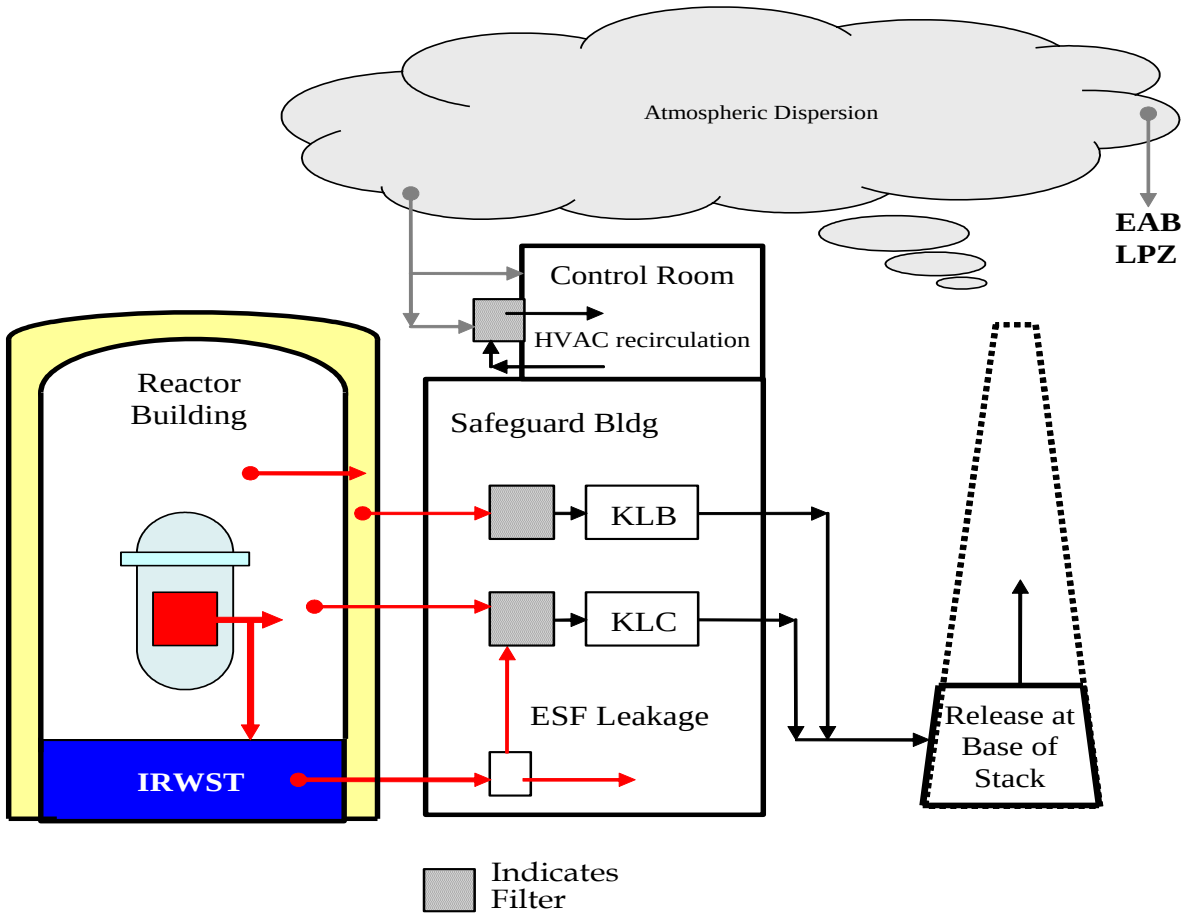
Radiological Consequences of DBAs (cont'd)

- **MCR HVAC Design (99% charcoal filtration system)**



Radiological Consequences of DBAs (cont'd)

- **Example of DBA model – LOCA (after end of annulus and safeguard buildings draw down)**



Scenario for Radiological Evaluation of SGTR

- **SGTR Source Terms (NRC RG 1.183, Appendix F)**
  - Pre-accident iodine spike case. RCS concentration to the maximum value of 60 µCi/gm DE I-131
  - Accident-induced concurrent iodine spike case of 8-hour duration. The iodine spike corresponds to an increase in the design-basis iodine appearance rate into the primary coolant by a factor of 335
  - RCS noble gas concentration corresponds to technical specifications 210 µCi/gm DE Xe-133.
  - All other radionuclide concentrations correspond to the design basis 0.25% failed fuel assumption
  - SGTR release iodines, noble gases and alkalis. All other radionuclides are assumed to remain within the liquid phase (RCS and secondary coolant) per NRC Regulatory Issue Summary 2006-04
- **Scenario**
  - Scenario selected maximizes the break flow and flashing fraction, corresponding to reactor trip by operator action and concurrent LOOP at 30 minutes into the accident, an additional 10 minutes to isolate the ruptured SG

Scenario for Radiological Evaluation of SGTR (cont'd)

- **Environmental Releases**
  - Atmospheric releases consist of the secondary-side activities, RCS leakage via the affected SG, and normal leakage via the other three intact SGs
  - Early releases are via the condenser evacuation system until RT (30 minutes into the accident by operator action), and via the MSRTs thereafter. Iodine and alkali depletion due to deposition within the condenser with a decontamination factor (DF) of 100 is credited (NRC NUREG-1228)

Scenario for Radiological Evaluation of LOCA

- **LOCA Source Terms (NRC RG 1.183, Appendix A)**
  - Bounding core inventory (4612 MWt, 5–62 GWD/MTU, 2-5% U-235)
  - RCS noble gas concentration corresponding to technical specification limits of 210 µCi/gm DE Xe-133, iodines at 1 µCi/gm DE I-131, and alkalis at the design basis 0.25% failed fuel fraction
- **Scenario**
  - Time-phased releases to primary containment; natural deposition of elemental iodines and other particulates (combined Powers/Henry models); no spray effects
  - 10-sec containment purge at start of accident via main stack; 0.25% per day containment leakage for first 24 hrs, 50% reduction thereafter; release adjacent to SG-3 MSRT during 305-sec annulus draw-down time, and via annulus filtration system and stack thereafter
  - ESF component leakage at 4 gpm (to safeguards buildings); release adjacent to SG-3 MSRT during 289-sec draw-down time, and via the safeguards building filtration system and stack thereafter
  - Releases via the main stack are at the base of the stack

Scenario for Radiological Evaluation of LOCA (cont'd)

- **Environmental Releases**
  - 10-second unfiltered release during containment purge; containment leakage via annulus (without holdup or depletion) and 99% exhaust filters
  - ESF component leakage via safeguards buildings (without holdup or depletion) and 99% filters

DBA Radiological Consequences Summary

- **DBA Doses (and Regulatory Limits)**

Design Basis Accident		Offsite TEDE Dose (rem)		MCR Dose (rem)
		EAB	LPZ	
LOCA		12.2 (25)	11.1 (25)	4.0 (5)
SLB outside of Reactor Building		1.8 (2.5)	0.3 (2.5)	0.1 (5)
SGTR	Pre-incident spike	1.1 (25)	0.3 (25)	0.3 (5)
	Coincident spike	0.7 (2.5)	0.5 (2.5)	0.6 (5)
MSLB	Pre-incident spike	0.2 (25)	0.1 (25)	0.5 (5)
	Coincident spike	0.3 (2.5)	0.2 (2.5)	0.7 (5)
	Fuel rod clad failure (3.3%)	5.3 (25)	2.6 (25)	4.5 (5)
	Fuel overheat (0.58%)	5.8 (25)	2.8 (25)	4.5 (5)
LRA (with 9.5% clad failure)		2.3 (2.5)	0.9 (2.5)	1.3 (5)
REA (with 36.7% clad failure)		5.7 (6.3)	3.5 (6.3)	4.3 (5)
FHA (with 34 hr decay)		5.6 (6.3)	1.0 (6.3)	0.5 (5)

U.S. EPR Steam Generator Tube Rupture Mitigation Strategy

Kenneth Coffey,
Engineer IV, Plant Operations

U.S. EPR Steam Generator Tube Rupture Mitigation Strategy Agenda

- **SGTR Mitigation Strategy and Goals**
- **4 Stages of SGTR Mitigation**
  - **Example of Stage 3 and 4 without offsite power**
- **Mitigation Comparison with and without offsite power**

U.S. EPR SGTR Mitigation Strategy

- **Mitigation of SGTR events similar to existing 4- loop PWRs**
- **U.S. EPR Specific Mitigation Features**
  - MHSI shutoff head and resetting MSRT setpoint allows for RCS inventory control without lifting the main steam valves
  - EBS provides safety related boration source
- **Main SGTR Mitigation Goals:**
  - Achieve controlled state to ensure RCS stability
    - The core is sub-critical, decay heat removal via SGs
    - Core water inventory is stable (RCS makeup compensates SGTR leak rate)
  - Terminate releases from affected SG as soon as practical
  - Cancellation of leak flow ($P_{RCS} \approx P_{SGa}$)
  - Reach safe shutdown on residual heat removal (RHR)

U.S. EPR SGTR Mitigation Strategy

- **4 main stages of SGTR mitigation**
  1. Identification of SGTR
  2. Reactor trip and isolation of affected SG
  3. Primary to secondary leak flow cancellation
  4. Cooldown & depressurize RCS and affected SG to allow for RHR connection

Stage 1 – SGTR Identification

- **Radiation monitors:**
  - **Main steam line**
  - **SG blowdown lines**
  - **Condenser off-gas**

- **System response:**
  - **Feed flow-steam flow mismatch (at low power)**
  - **Pressurizer pressure decreasing**
  - **Pressurizer level decreasing**

Stage 2 – Reactor Trip and Affected SG Isolation

- **Reactor Trip**
  - **Manual RT if CVCS is capable of compensating the leak**
- **Affected SG Isolation**
  - **Close MFW and EFW isolation valves of the affected SG**
  - **Verify isolation of SG blowdown lines of affected SG**
  - **Close the MSIV of the affected SG**
  - **Reset the MSRT (isolation and control valves) to the high setpoint for the affected SG**

Stage 3 – Primary to Secondary Leak Flow Cancellation

- **Stage 3 differs depending on the available equipment**
- **With LOOP:**
  - **Cooldown RCS using unaffected SGs**
    - Partial cooldown (180 F/hr to 870 psia)
  - **Supply feedwater to unaffected SGs**
    - EFW
  - **Maintain adequate RCS inventory**
    - Safety Injection System
  - **Allow for the RCS pressure to be reduced to the affected SG pressure**

Stage 4 – Cooldown and Depressurize RCS to RHR Connection Conditions

- **RCS Cooldown with LOOP**
  - **Cooldown the using MSRTs of the unaffected SGs**
  - **Boration for cooldown**
    - 1 EBS train required for 45°F/hr
    - 2 EBS trains required for 90°F/hr
  - **MHSI managed to**
    - Maintain adequate subcooling margin
    - Maintain RCS inventory
    - Allow for depressurization of RCS to RHR connection pressure
- **RCS Depressurization to RHR connection pressure with LOOP**
  - **Cycle PSRVs while maintaining subcooling margin**

LOOP vs. Non-LOOP Mitigation Comparison

- Emergency Procedure Guidelines (EPG) developed to utilize all available systems
- Preference given to systems that accomplish the task as effectively as possible and as close to normal operating conditions as the situation permits
- Availability of off-site power is one of the biggest factors in determining what systems may be used to mitigate an accident

Stage 3 – Systems to Makeup RCS Inventory and Control RCS Pressure

With Off-Site Power

- RCS Inventory Control
  - CVCS
- RCS Pressure Control
  - Normal spray / Auxiliary spray
  - PZR Heaters
- SG Inventory Control
  - MFW
- SG Pressure Control
  - Turbine bypass to condenser

Without Off-Site Power

- RCS Inventory Control
  - MHSI
  - EBS
- RCS Pressure Control
  - RCS pressure imposed by MHSI pump shutoff head
- SG Inventory Control
  - EFW
- SG Pressure Control
  - MSRTs of unaffected SGs

Stage 4 – Multiple Means to Depressurize RCS and Affected SG

With Off-Site Power

- With RCPs running
  - Forced circulation through tubes of affected SG acts to cool and depressurize affected SG
- Continued cooldown and depressurization
  - CVCS
  - Normal / Auxiliary Spray
  - Heaters
  - MFW
  - Turbine Bypass

Without Off-Site Power

- RCPs not running
  - Affected SG pressure holds RCS pressure above RHR connection conditions
- Continued cooldown and depressurization
  - MHSI
  - EBS
  - EFW
  - MSRTs of unaffected SGs
- Final RCS and affected SG depressurization
  - PSRVs

Nomenclature

- ACT — Average Coolant Temperature [Control Function]
- AMS — Aeroball Measurement System
- AO — Axial offset
- AOO — Abnormal Operational Occurrence
- AST — Alternative Source Term
- ATWS — Anticipated Transient Without Scram
- CVCS — Chemical and Volume Control System
- DNB(R) — Departure from Nuclear Boiling (Ratio)
- DAS — Diverse Actuation System
- DBA — Design Bais Accident
- DE — Dose Equivalent
- DF — Decontamination Factor
- EAB — Exclusion area boundary
- EBS — Extra Borating System
- ECCS — Emergency Core Cooling System
- EDG — Emergency Diesel Generator
- EFW — Emergency Feedwater
- ESF — Engineered Safety Function
- FDH — Axially Integrated Radial Power Peaking (Enthalpy Rise) Factor
- FHA — Fuel Handling Accident
- FWLB — Feedwater Line Break
- HCPL — High Core Power Limit
- HFP — Hot Full Power
- (H)LPD — (High) Linear Power Density (kW/ft)
- HVAC — Heating and Ventilation
- HZP — Hot Zero Power
- I&C — Instrumentation and Control
- IRD — Intermediate Range Neutron Detector
- IRWST — In-containment Refueling Water Storage Tank
- LCO — Limiting Condition for Operations
- LOCA — Loss of Coolant Accident
- LOFC — Loss of Flow
- LOOP — Loss of Offsite Power
- LPD — Linear Power Density
- LPZ — Low Population Zone
- LRA — Locked Rotor Accident
- MCR — Main Control Room
- MNDBR — Minimum Departure from Nucleate Boiling Ratio
- MFW — Main Feedwater
- MHSI — Medium Head Injection System
- MSIV — Main Steam Isolation Valve
- MSLB — Main Steam Line Break
- MSRT — Main Steam Relief Train
- MSSV — Main Steam Safety Valve
- NSSS — Nuclear Steam Supply System
- PA — Postulated Accident
- PRD — Power Range Neutron Detector
- PS — Protection System
- PSRV — Pressurizer Safety Relief Valve
- PWR — Pressurizer Water Reactor
- PZR — Pressurizer
- RB — Reactor Building
- RCCA — Rod Cluster Control Assembly
- RCP — Reactor Coolant Pump
- RCS — Reactor Coolant System
- RCSL — Reactor Control, Surveillance, and Limitation
- REA — Rod Ejection Accident
- RD — Rod Drop
- RHR — Residual Heat Removal [System)
- RIS — Regulatory Issue Summary
- RT — Reactor Trip
- RV — Reactor Vessel
- SAFDL — Specified Acceptable Fuel Design Limit
- SBLOCA — Small Break LOCA
- SDM — Shutdown Margin
- SIS — Safety Injection System
- SG — Steam Generator
- SGTR — Steam Generator Tube Rupture
- SLB — Small Line Break
- SPND — Self-Powered Neutron Detector
- SRD — Source Range Neutron Detector
- SRP — Standard Review Plan
- TEDE — Total Effective Dose Equivalent
- TH — Thermal Hydraulic

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

AREVA U.S. EPR Design Certification Application Review

Safety Evaluation Report with Open Items

Chapter 15, Group 1: Transient and Accident Analyses

February 7 and 8, 2011

Staff Review Team

- **Technical Staff**

  - **Shanlai Lu**
    Reactor Systems, Nuclear Performance, and Code Review Branch
  - **Christopher VanWert**
    Reactor Systems, Nuclear Performance, and Code Review Branch
  - **Michelle Hart**
    Reactor Siting and Accident Consequence Branch
  - **Eduardo Sastre**
    Component Integrity Branch

- **Project Managers**

  Getachew Tesfaye

  Jason Carneal

Overview of DCA

SRP Section/Application Section		No. of Questions	Number of OI
15.0.1 and 15.0.2	Radiological Consequence Analysis and Computer Codes Used in Transient and Accident Analysis	10	3
15.0.3	Radiological Consequences of Design Basis Analysis	38	2 (closed)
15.1	Increase in Heat Removal by the Secondary System	25	0
15.2	Decrease in Heat Removal by the Secondary System	25	0
15.3	Decrease in Reactor Coolant System Flow Rate	9	0
15.4	Reactivity and Power Distribution Anomalies	15	1

Overview of DCA (continued)

SRP Section/Application Section		No. of Questions	Number of OI
15.5	Increase in Reactor Coolant Inventory	2	0
15.6.1	Inadvertent Opening of a Pressurizer Safety Relief Valve	3	0
15.6.3	Steam Generator Tube Failure (PWR)	4	0
15.8	Anticipated Transients Without Scram	0	0
Totals		131	6 (4 open items remain)

*The Phase 2 safety evaluation for Section 15.6.5, “Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary,” was not delivered in Group 1 and will not be covered in this presentation.

Description of Open Items

- RAI 415, Question 15-9: Requests that the applicant propose a combined license (COL) action item to require that the COL applicant perform verification and validation of its site-specific EOPs using plant simulators to assure that the operator action time assumed in the safety analyses documented in FSAR Tier 2, Chapter 15 are achievable to assure that the safety analyses of record remain valid.
- RAI 311, Question 16-317: Requests that the applicant provide an evaluation of all the proposed technical specification (TS) values in FSAR Tier 2, Chapter 16 to confirm that all assumptions in FSAR Tier 2, Chapter 15 safety analyses are within the limiting conditions for operation specified in the TS. This question originated in Chapter 16 of the review.
- RAI 432, Question 15.00.02-1: Requests that the applicant provide a description of the mechanism, such as an ITAAC or COL action item, by which the information will be provided to support the 0.48 percent total power measurement uncertainty claimed by the applicant, including a description of how it will be verified and confirmed.
- RAI 344, Question 04.03-28: Tracks topical report ANP-10286P as an open item until the final safety evaluation is produced.

Closed Open Items

- RAI 394, Question 15.00.03-37: Provide tables in the FSAR that demonstrate that the pH would be acceptable after the 30-day time frame with the added amounts of HCl and HNO_3 resulting from radiolytic reactions. **These tables have been included as FSAR markups in an RAI response and this item is considered confirmatory by the staff.**

- RAI 394, Question 15.00.03-38: Requests that the applicant include purity and density values for commercially purchased TSP in FSAR Tier 2, Section 15.0.3. **These values have been included as FSAR markups in an RAI response and this item is considered confirmatory by the staff.**

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

AREVA U.S. EPR Design Certification Application Review

Safety Evaluation Report with Open Items

Chapter 15, Group 1, Sections 15.0.2, 15.1, 15.2, 15.3, 15.5, 15.6, 15.8

February 7, 2011

Staff Review Team

- **NRC Technical Staff**

  Reactor Systems, Nuclear Performance, and Code Review Branch

  Chu-yu Liang (Retired)

  Shanlai Lu

- **Consultants**

  Jack Rosenthal, David Caraher, Vesselin Palazov (ISL)

- **Project Managers**

  Getachew Tesfaye

  Jason Carneal

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Types of events considered in design basis safety analyses
  - Increase in heat removal by the secondary systems (15.1)
  - Decrease in heat removal by the secondary systems (15.2)
  - Decrease in RCS flow rate (15.3)
  - Reactivity and power distribution anomalies (15.4)
    - Will be covered in a separate presentation on February 8, 2011
  - Increase in reactor coolant inventory (15.5)
  - Decrease in reactor coolant inventory (15.6)
  - Anticipated Transient Without Scram (ATWS) (15.8)

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Acceptance criteria for each category of events according to frequency of occurrence (Non-LOCA)

  - Anticipated operational occurrence (AOO) – Events expected to occur during the life time of the plant

    1) Pressures in primary and secondary systems not exceeding 110% of the design value
    2) Fuel cladding integrity shall be maintained by no DNB during the transient
    3) An AOO should not generate a postulated accident

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Acceptance Criteria (continue)

  Postulated accidents (PA)– Events not expected to occur during the life time of the plant design

  1) Pressure in primary and secondary systems not exceeding acceptable design limits, considering potential brittle as well as ductile failures
  2) Fuel failure may occur but not result in offsite doses exceeding 10 CFR 100 limits
  3) A PA should not cause a consequential loss function of systems needed for mitigation of the event

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Major review areas (Non-LOCA)
  - Review of methodologies and computer codes used in safety analyses to assure that they are approved by NRC and applicable to U.S. EPR design
  - Review of LCOs proposed in Chapter 16.1 to assure that they provide bounding conditions to support safe analyses assumptions including initial conditions, plant configurations, etc

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Major review areas (continue)

  - 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 (e.g. Peak pressures, DNB)

  - Review of events analyzed with and without offsite power available. Loss-of-Offsite-Power (LOOP) may occur following a turbine trip

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Major review areas (continue)

  - Review of sequence of events for analyzed event scenario to evaluate the mitigation system and 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 events and all the regulatory requirements

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Site Audits and Confirmatory Calculations

  - Staff conducted various audits of design files at AREVA offices to obtain additional information for detailed staff review

  - Staff performed confirmatory calculations using methods and computer codes available to NRC to establish sufficient confidence to the results of AREVA's analyses.

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Open Items

1) Emergency Operating Procedures (EOPs) verification – The staff request AREVA to add a COL Action Item for verification of plant specific EOPs to validate the safety analyses assumptions.

   - FSAR Section 15.0.0.3.7 states the operator actions are credited in various analyses to mitigate postulated accidents (e.g. FWLB, MSLB, SGTR, and ECCS switchover during LOCA long term cooling).

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Open items – EOP verifications (continue)
  - FSAR Section 13.5 states that a COL applicant will provide site-specific EOPs. However, there was no specific guidelines for the required verification and validation of EOPs to confirm operator actions credited in safety analyses are achievable.
  - Staff requires AREVA propose a COL Action Item in EPR DCD for the above required verifications.

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Power Measurement Uncertainties

The staff request AREVA to provide sufficient information to support its claimed 0.48% power uncertainty assumed in Chapter 15 safety analyses

- FSAR Section 15.0.0.3.1 indicated that the heat balance measurement uncertainty of +/- 22 MWt is applicable to the rated thermal power of 4590 MWt (approximately 0.48% uncertainty) for the maximum core power assumed in Chapter 15 safety analyses.

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Open items – Power measurement uncertainties (continue)
  - The staff request AREVA to provide the following:

a) A description of an mechanism, such as the FSAR and ITAAC and/or COL Action Item by which the information will be provided to support claimed 0.48% power uncertainty.

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Open items – Power measurement uncertainties (continue)

  b) A description of the instrumentation and methodology used for the main feedwater measurement and calorimetric power flow measurement.

  c) A description of more details regarding the methods of treating total uncertainties for the main feedwater flow rate and reactor thermal power, etc.

U.S. EPR Chapter 15 Group 1 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Verification of Technical Specifications

Technical Specifications (TS) supporting safety analysis assumptions – The staff requests AREVA to verify and confirm that the plant TSs are consist with assumptions applied in Chapter 15 safety analyses

- The staff requires AREVA to provide the results of an assessment to confirm that its proposed TS in FSAR Chapter 16 are consistent with the assumptions applied in the FSAR Chapter 15 safety analyses.

U.S. EPR Steam Generator Tube Rupture Event

- Offers direct path to environment
- Thermal-hydraulic analysis provides input to dose analysis
  - Maximize break flow, critical flow, MSRCV assumed stuck open
  - Technical Specification Coolant Activity
  - First 30 minutes no operator action
  - LOOP assumed

U.S. EPR Steam Generator Tube Rupture Event

- Automatic secondary side partial cooldown, automatic EFW, manual EBS
- Approved methods: S-RELAP5
- Meet criteria 10CFR 50.47, GDC 19
- EOP critical to ensure long term cooling to maintain shutdown margin, prevent overfill of steam line and backflow

US-EPR DCA – Chapter 15 Transient and Accident Analyses

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Conclusions

- Staff has completed the Phase 2 review on 15.0, 15.1, 15.2, 15.3, 15.5, 15.6, 15.8
- 131 RAIs were issued on Chapter 15, Group 1
- Four open items remain open in Phase 4
- Confirmatory analysis results supported staff conclusions

Section 15.4 will be presented on February 8, 2011.

- includes open item on ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology Topical Report"

United States Nuclear Regulatory Commission

Protecting People and the Environment

NRC Staff Confirmatory Analysis Supporting Safety Evaluation of U.S. EPR Chapter 15, Group 1

AREVA U.S. EPR Design Certification Application Review

Safety Evaluation Report with Open Items

Chapter 15, Group 1: Transient and Accident Analyses

February 7 and 8, 2011

Staff Review Team

- **NRC Staff**

Reactor Systems Branch, NRO
Shanlai Lu

Reactor Systems Safety Analysis Branch, RES
Shawn Marshall, Willams Krotiuk, Stephen Bajorek
Ronald Harrington

Reactor Systems Branch, NRR
Anthony Ulses

- **Consultants**

Randy Bells, Jose March-Leuba(ORNL), David Caraher (ISL)

Contents

- Overall Confirmatory Analysis Approach
- U.S. EPR System Model Development Using TRACE
- U.S. EPR Core Model Development Using PARCS
- Transient Analysis Results:
  - Feedwater Heater Trip Transient
  - Rod Withdrawal Transient
- Conclusion

Overall Confirmatory Analysis Approach

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Confirmatory Analysis Selection Criteria for Review of U.S. EPR Design Certification, Chapter 15, Group 1
  1. Support staff's review on new design features
  2. Identify RAIs related to margin evaluation

  Goal: Develop proper RAIs and review strategy

- Confirmatory Analysis Scope

  AOOs: Loss of Feedwater Heater, Rod Withdrawal

  Postulated Accident: LBLOCA, SBLOCA, Rod Ejection

  Design Feature: Ex-core Detector, Spent Fuel Pool

  D3: Increase in steam flow, MSIV closure, etc.

NRC Confirmatory Analysis Approach And Tools

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

AREA 1. System Model Development

RELAP5, S-RELAP5 (AREVA)*, TRACE, PARCS,

NRC-RES, ISL

AREA 2. Fuel Performance Analysis

FRAPCON, FRAPTRAN, RODEX-4 (AREVA)*

NRC-NRO, PNNL

AREA 3. AOOs and Postulated Accidents Evaluation

TRACE, PARCS, TRITON, RELAP5, S-RELAP5 (AREVA)*

NRC-RES, ORNL, ISL

AREA 4. Ex-core Neutron Flux and Criticality Evaluation

SCALE 6.0, MCNP

NRC-NRO, ORNL

* AREVA codes run by NRC staff

United States Nuclear Regulatory Commission

Protecting People and the Environment

Development of the TRACE Model for US EPR

Shawn O. Marshall
Office of Nuclear Regulatory Research (RES)
U.S. NRC

February 7, 2011

OBJECTIVE

To provide an overview of the basic TRACE U.S. EPR model developed by RES to support NRO confirmatory analyses.

- Give general descriptions of some of the major component and control systems, to include the primary- and secondary-systems and the Emergency Core Cooling System (ECCS).
- As proof of the models general adequacy, show results from a TRACE steady-state calculation in comparison with initial conditions and steady-state results provided by AREVA.

US EPR TRACE Model

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

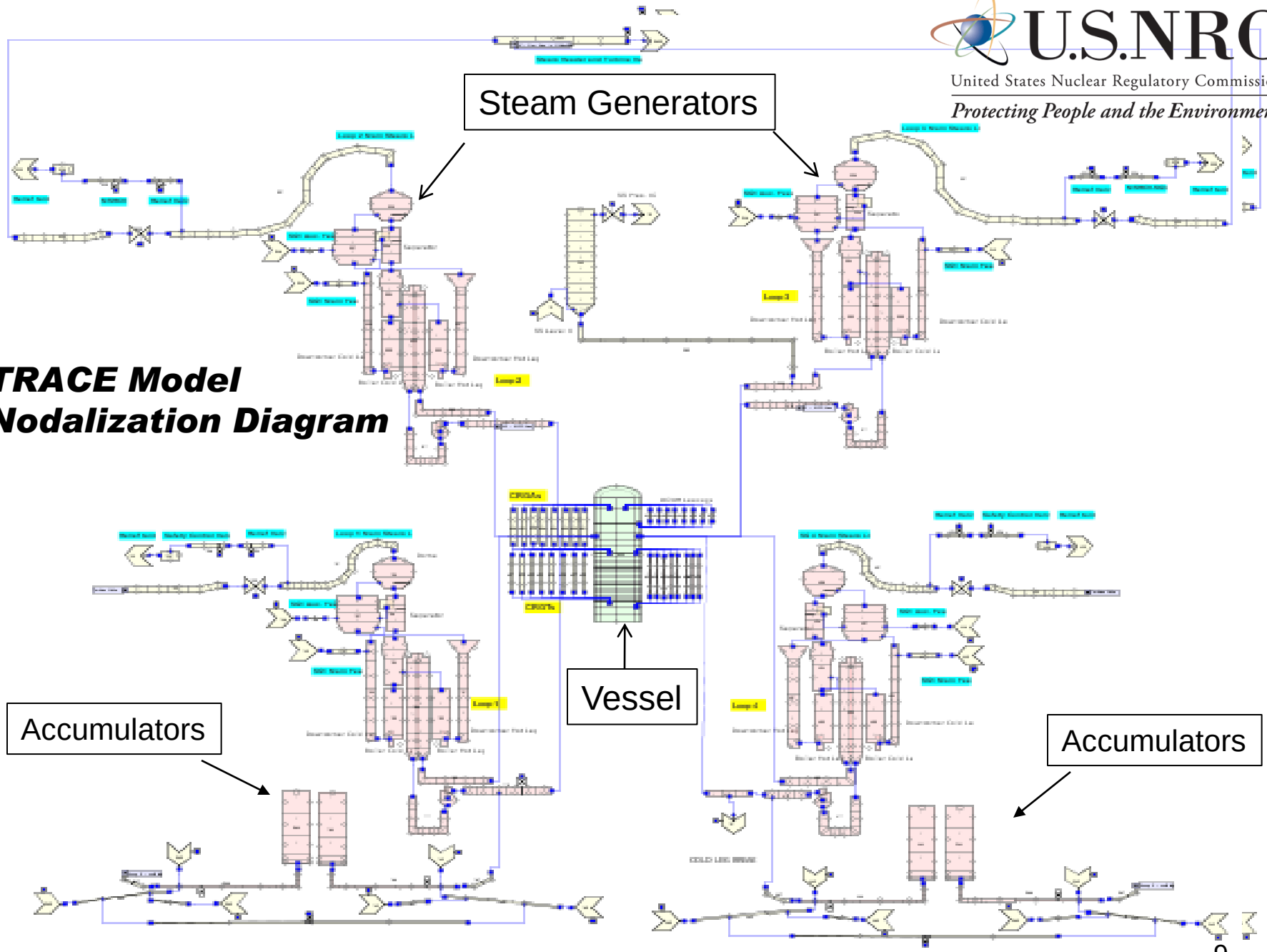
- Model was designed to incorporate as much plant-specific detail as possible without unduly increasing runtime.
  - simulates all basic operational and protection system functions

- Input based on information from SRELAP-5 model and EPR Databank

- Assembled using the Symbolic Nuclear Analysis Package (SNAP)

- Model summary:
  - Contains over 185 hydraulic components,
  - over 400 control- system inputs, and
  - over 150 different heat structures

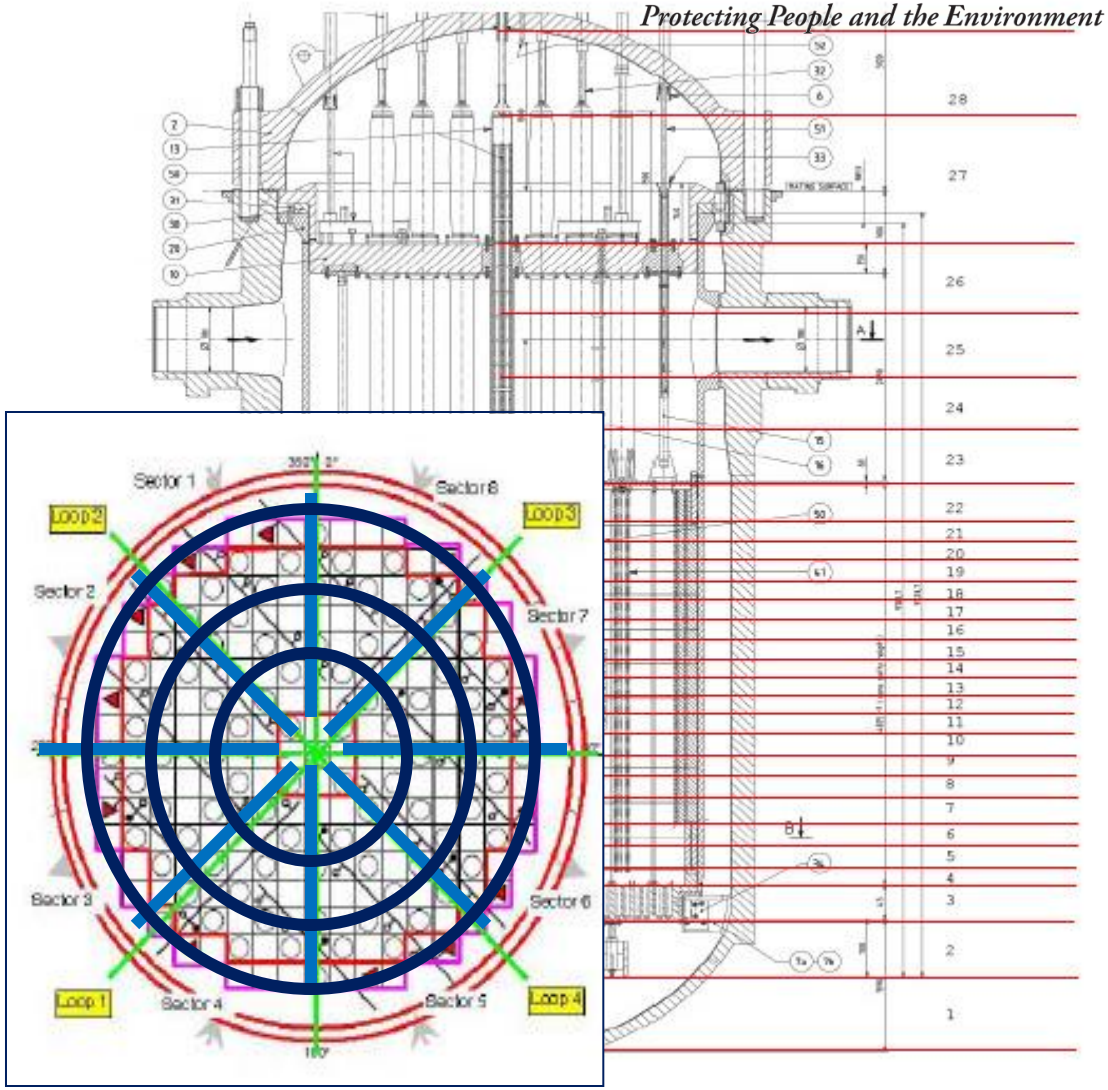
TRACE Model Nodalization Diagram



Primary System

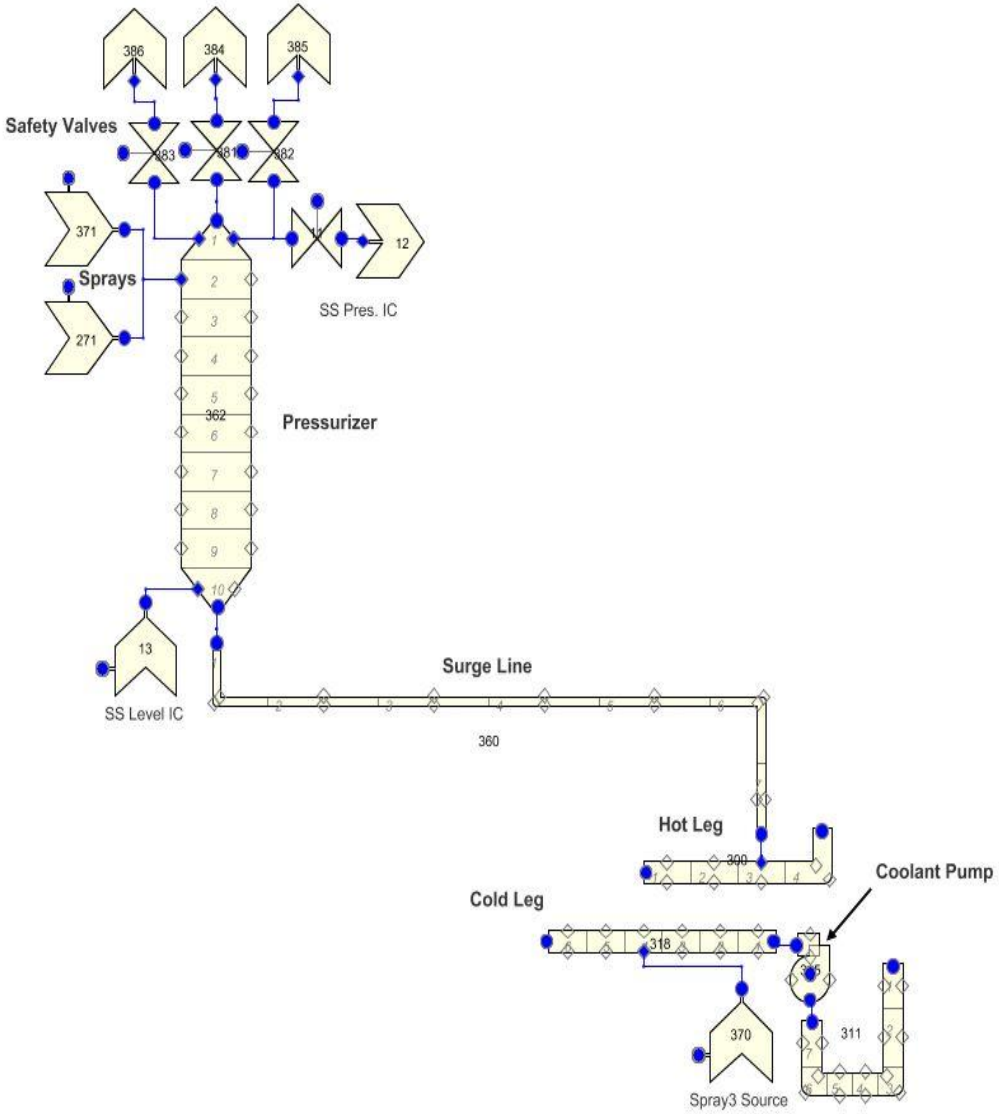
Reactor Vessel

- Modeled using TRACE 3-D VESSEL Component
- There are 8 azimuthal sectors, one sector for each hot leg and each cold leg.
- Vessel modeled with 4 radial rings:
  - Ring 1 contains the innermost 9 fuel assemblies;
  - ring 2 contains other 184 assemblies;
  - ring 3 contains outermost 48 assemblies and the thermal shield, and it extends to outer diameter of the core barrel;
  - ring 4 is the downcomer.



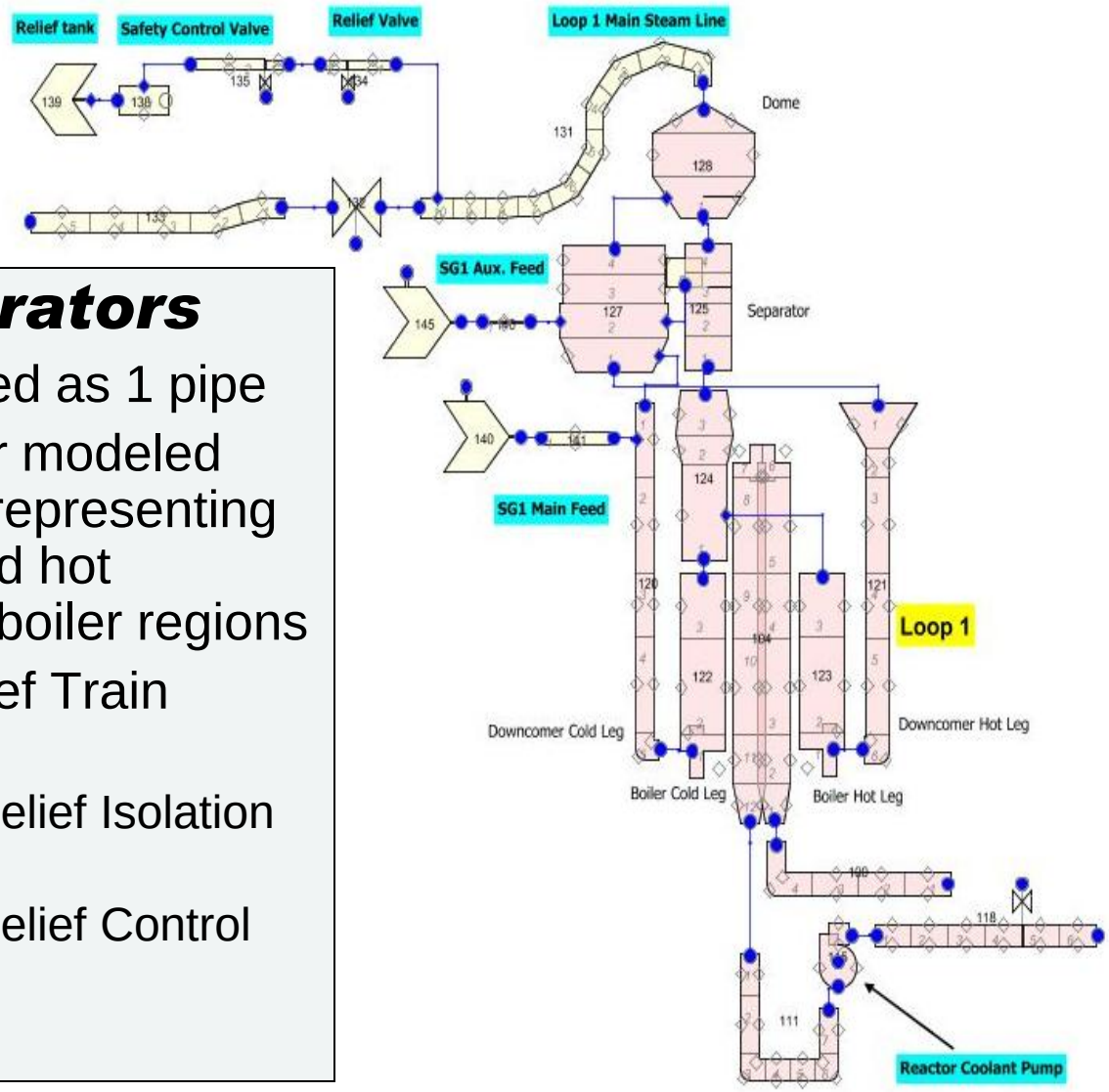
Primary System (Cont.)

- ***Pressurizer***
  - ***3 Safety Relief Valves***
  - ***Pressurizer Heaters***
  - ***Sprays***
    - Proportional-Integral-differential controllers (PIDs) modulate sprayers and heaters to control liquid level
- ***Each of the four coolant loops is modeled independently***
  - Reactor Coolant Pump (RCP)
  - Steam Generator



Secondary System

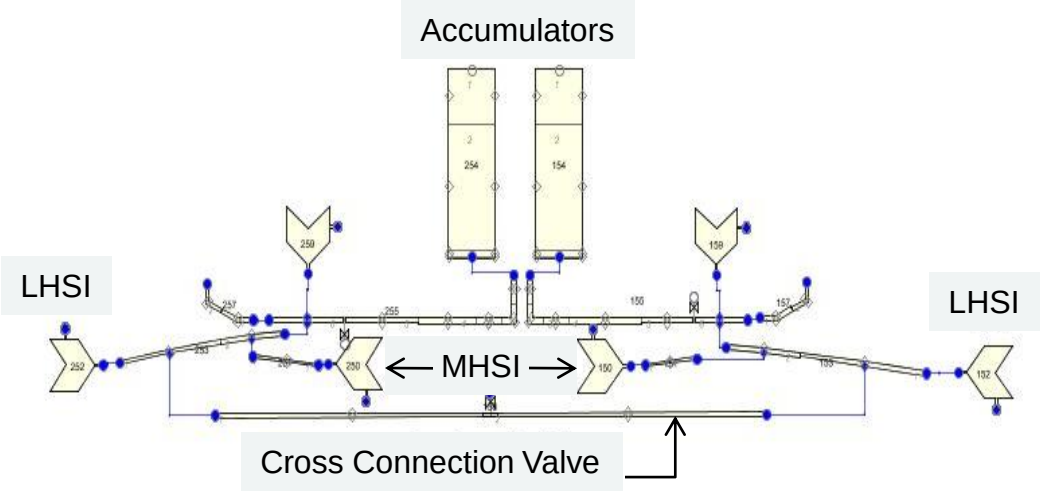
- **4 Steam Generators**
  - SG tubes modeled as 1 pipe
  - Axial economizer modeled using 1-D pipes representing separate cold and hot downcomer and boiler regions
  - Main Steam Relief Train (MSRT)
    - Main Steam Relief Isolation Valves
    - Main Steam Relief Control Valves

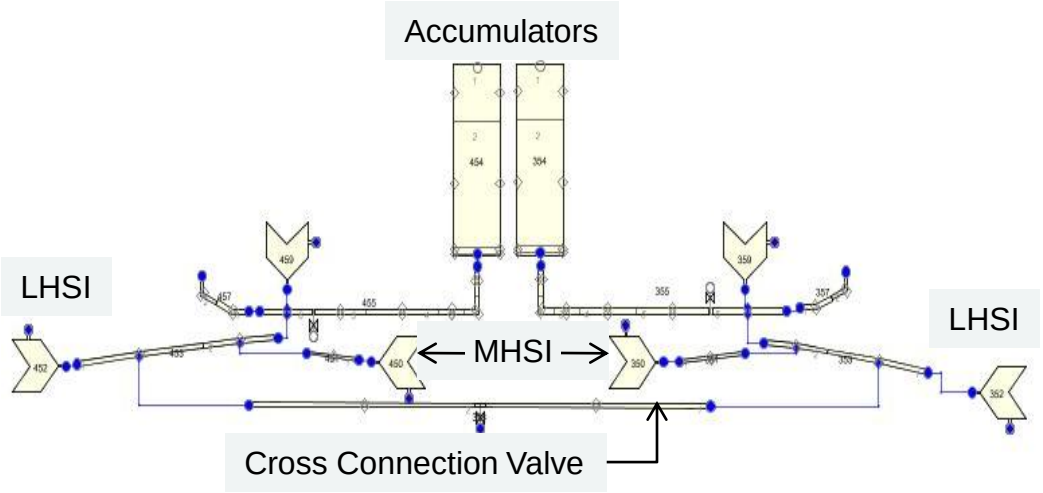


Emergency Core Cooling System

United States Nuclear Regulatory Commission
Protecting People and the Environment

- Independent ECCS trains
  - an accumulator
  - injection sites for the safety systems
    - Low Head Safety Injection (LHSI)
      - Loops 1&2 and 3&4 connected by cross connection lines and valves
    - Medium Head Safety Injection (MHSI)





Steady State Results

Parameter (TRACE Plot Var)	AREVA Results	TRACE Results
Reactor Power, MW (rpower-5150)	4590.0	4590.0
Tavg , K (cb3020)	585.4	585.4
Pressurizer Pressure, MPa (pn-362A02)	15.51	15.51
Pressurizer Level, m (pn-362A02)	6.68	6.68
RCS Flow, kg/s (cb7205)	23182.	22920.
Bypass Flows, %, Total	4.75	4.99
DC to UH (cb7210)	0.4	0.42
CRGT (cb7211)	2.6	2.22
Baffle & DC/UP (cb7212)	2.5	2.35
Upper Head Temp., K	~585	586.
Pump Speed, rad/s (cb400)	124.6	134.1
Pump Head, m	106.4	133.6
Vessel dP, MPa (cb7301)	X	1.46X
SG tubes dP, Pa (cb7304)	X	1.05X

Conclusion

- An overview of the basic TRACE U.S. EPR model developed by RES has been provided, to include general descriptions of the primary- and secondary-systems and the Emergency Core Cooling System (ECCS).
- The TRACE steady-state results were comparable to the AREVA initial conditions and calculated results.
- TRACE model can be used to generate useful information to support staff review.

United States Nuclear Regulatory Commission

Protecting People and the Environment

U.S. EPR Transient Model Development and Analysis Using TRACE/PARCS

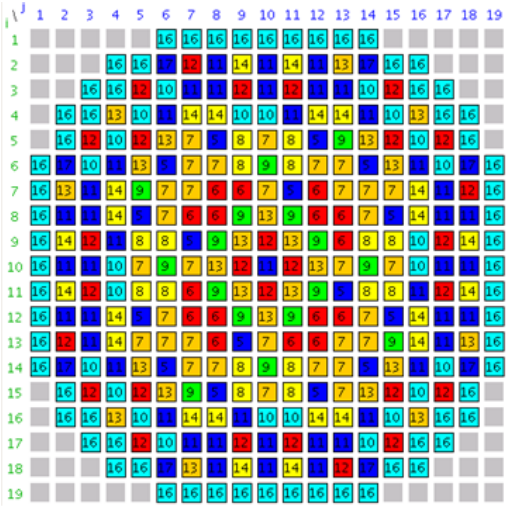
Anthony Ulses

Chief of Reactor System Branch
U.S. NRC / NRR

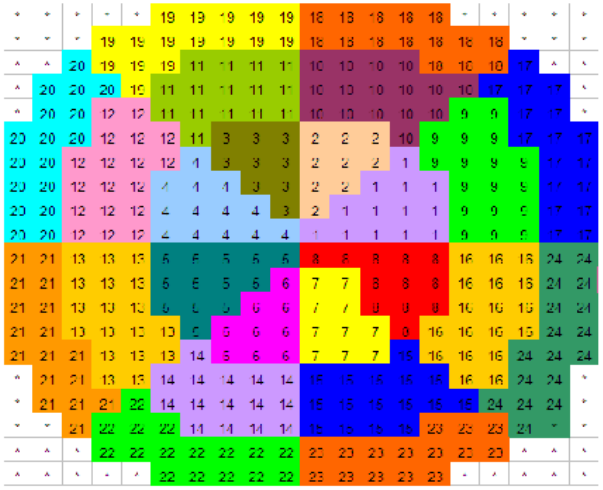
EPR Core Model Development

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Thermal-hydraulic (T/H) Model identical to LOCA deck except fuel assembly modeled as individual heat structure
- Neutronic model is fully 3-D
  - 1 to 1 fuel temperature coupling
  - Approximately 14 to 1 moderator coupling



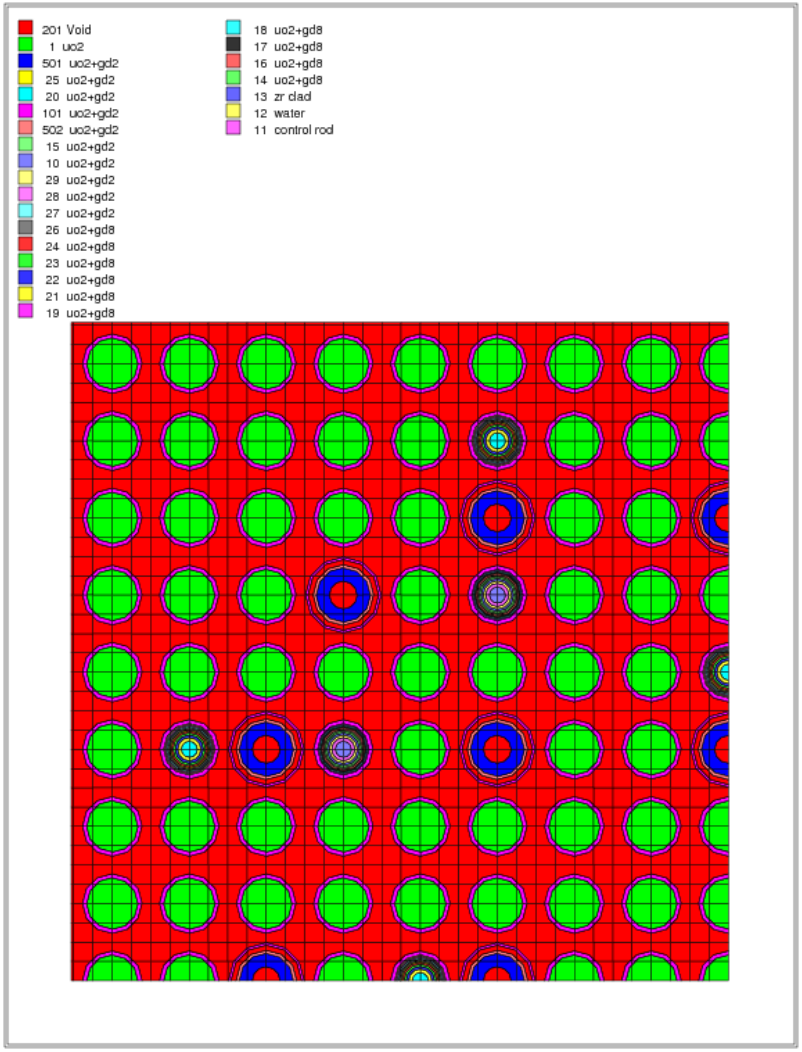
Neutronic Map



Thermal-Hydraulic Map

Cross Section Development

- SCALE/TRITON ¼ Assembly Model
- Parameter Branching Applicable for Hot Zero and Full Power Transients
- Each Fuel Type Modeled



Transient Results

- Used to prepare simulated “data” for subsequent analysis
- Evaluate that the DNBR and LPD trips would conservatively bound the actual plant response
- Two transients chosen
  - Rod Withdrawal Accident
    - Simulated trip at 20 seconds
  - Loss of Feedwater Heater
    - Simulated trip at 50 seconds

Conclusion

- NRC TRACE/PARCS code package has been used to develop EPR 3-D core model based on the cross-section library generated using SCALE/TRITON code package

- Two EPR transients have been simulated and the calculated 3-D neutron flux level throughout the core can be used to evaluate EPR SPND based protection system responses

United States Nuclear Regulatory Commission

Protecting People and the Environment

Evaluation of EPR DNBR and LPD Trip Timing Using TRACE/PARCS Results

Shanlai Lu
NRC/NRO

Randy Bells, Jose March-Leuba
ORNL

February 7, 2011

Background

- EPR relies on 72 in-core self-powered neutron detectors to determine
  - departure from nucleate boiling (DNB), and
  - linear power density (LPD)
- DNBR trip requires an on-line algorithm
  - Must run in ~100 ms in qualified hardware
  - 12 estimates of MDNBR, one from each SPND string
  - Not enough computer power to perform a detailed sub-channel analysis

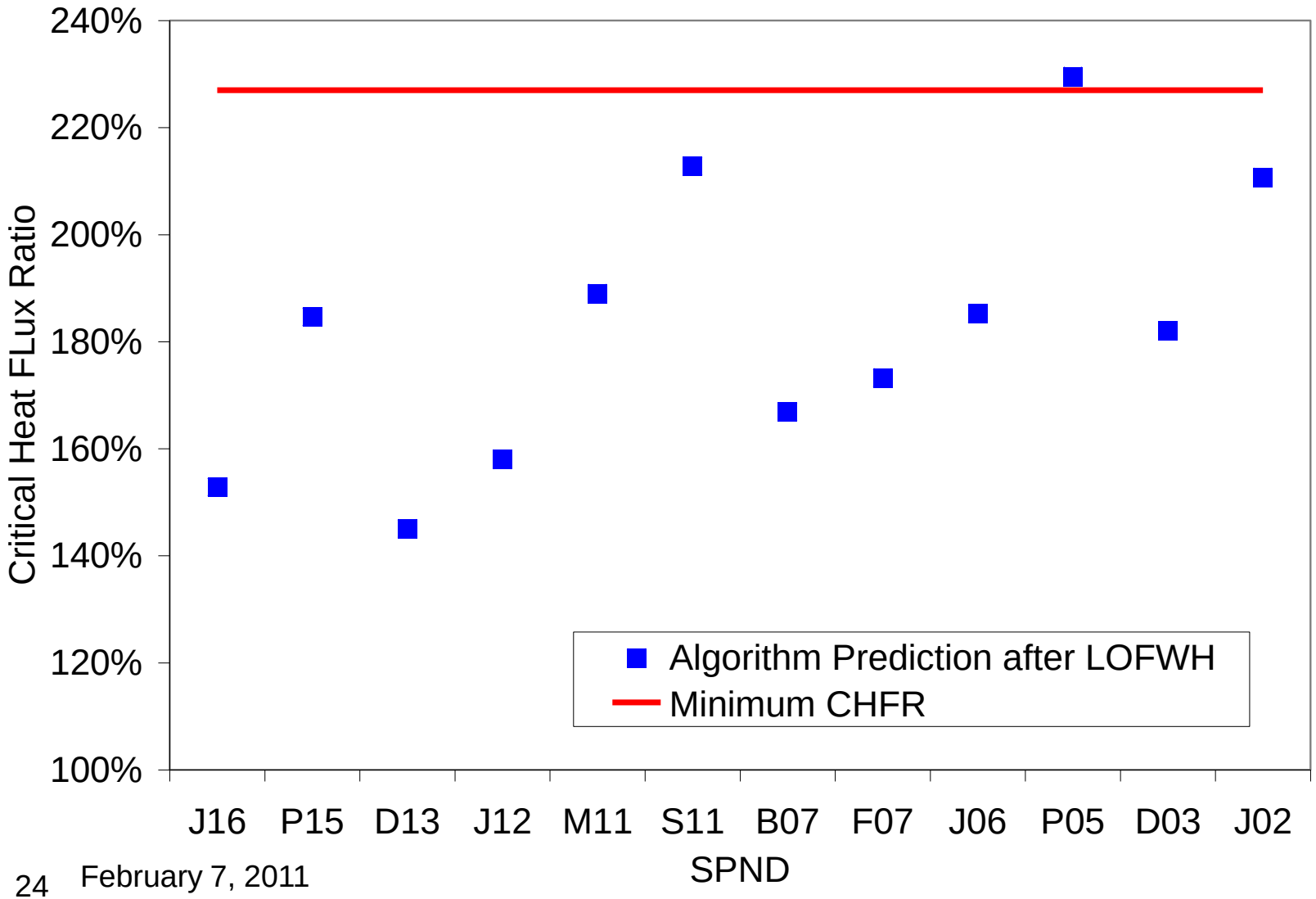
In Core DNBR Calculation Using TRACE Output

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

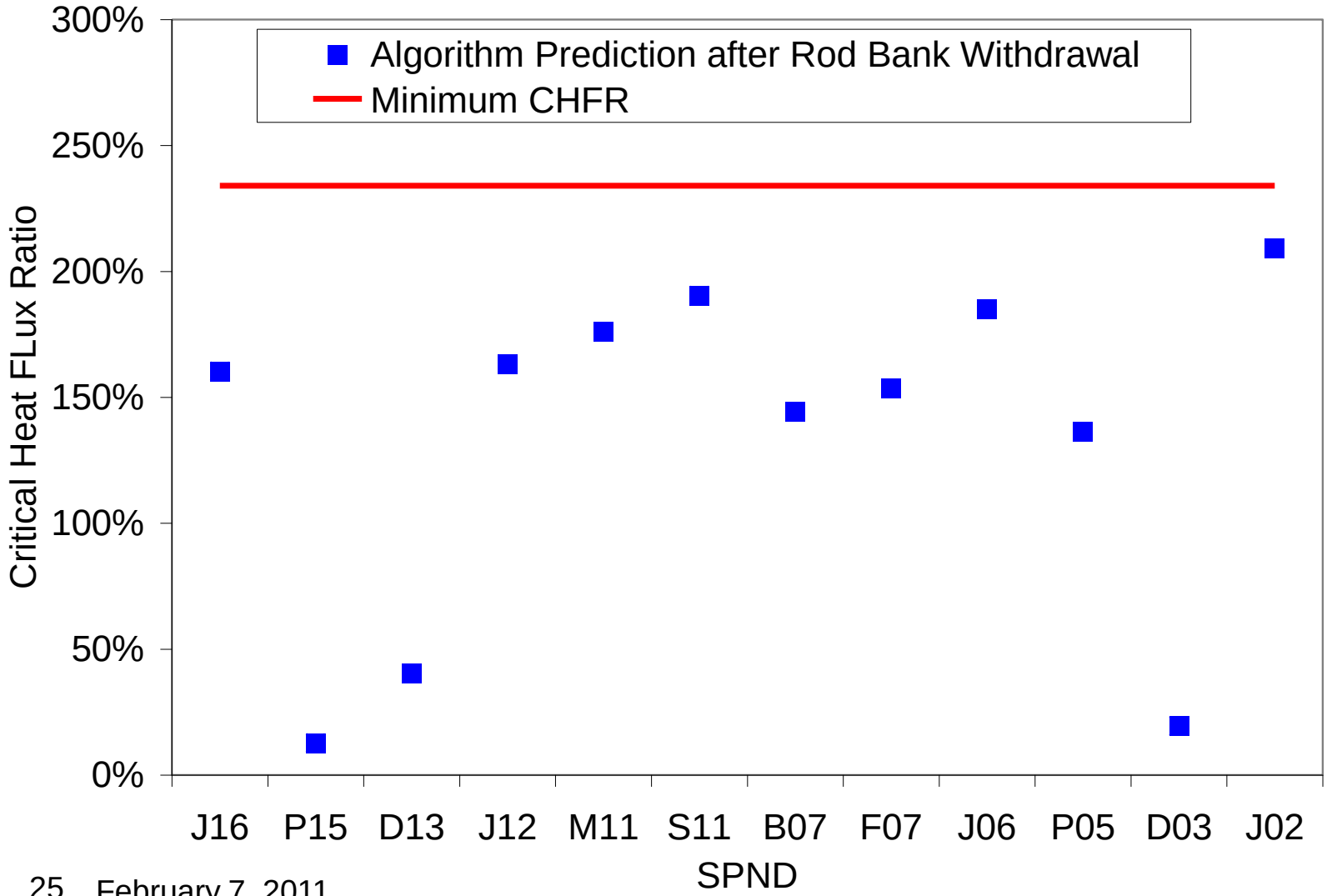
- Two calculations performed with TRACE to simulate SPND and EPR protection system response:
  - FW heater trip: core wide power change
  - Rod withdrawal: localized power peak

- Calculation Process
  - TRACE defines a steady state (3D power distribution)
  - SPNDs calibrated with a simulated AMS measurement
    - C_{ij} parameters for DNBR and LPD are generated
  - TRACE calculates 3D power during transient
  - DNBR & LPD values calculated
    - From real 3D TRACE powers (“real” hot bundle) using the CHF correlation documented in ANP-10269PA
    - From simulated SPND C_{ij}’s (“EPR” 12 hot bundles)

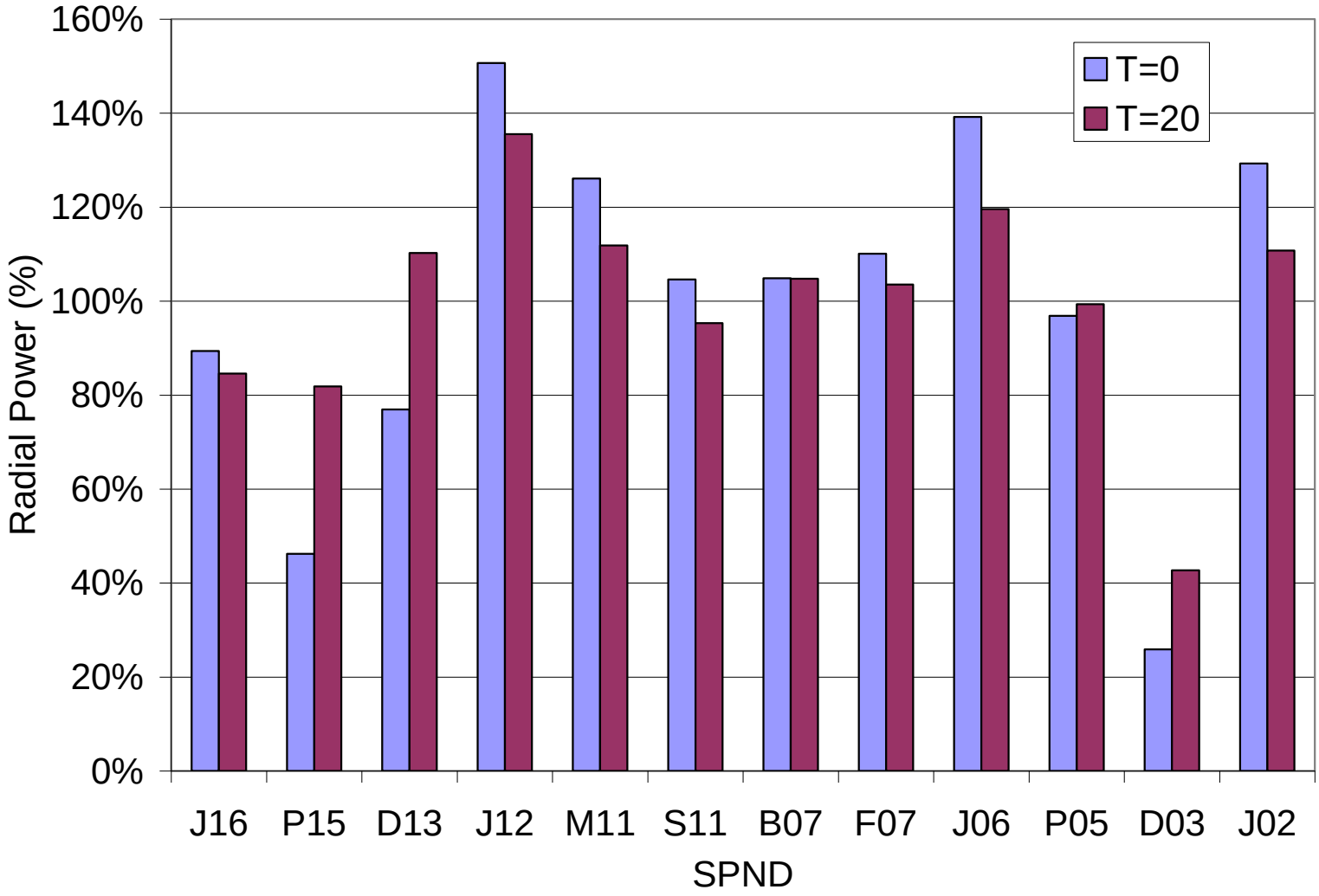
DNBR Confirmatory Calculation (LOFWH Event)



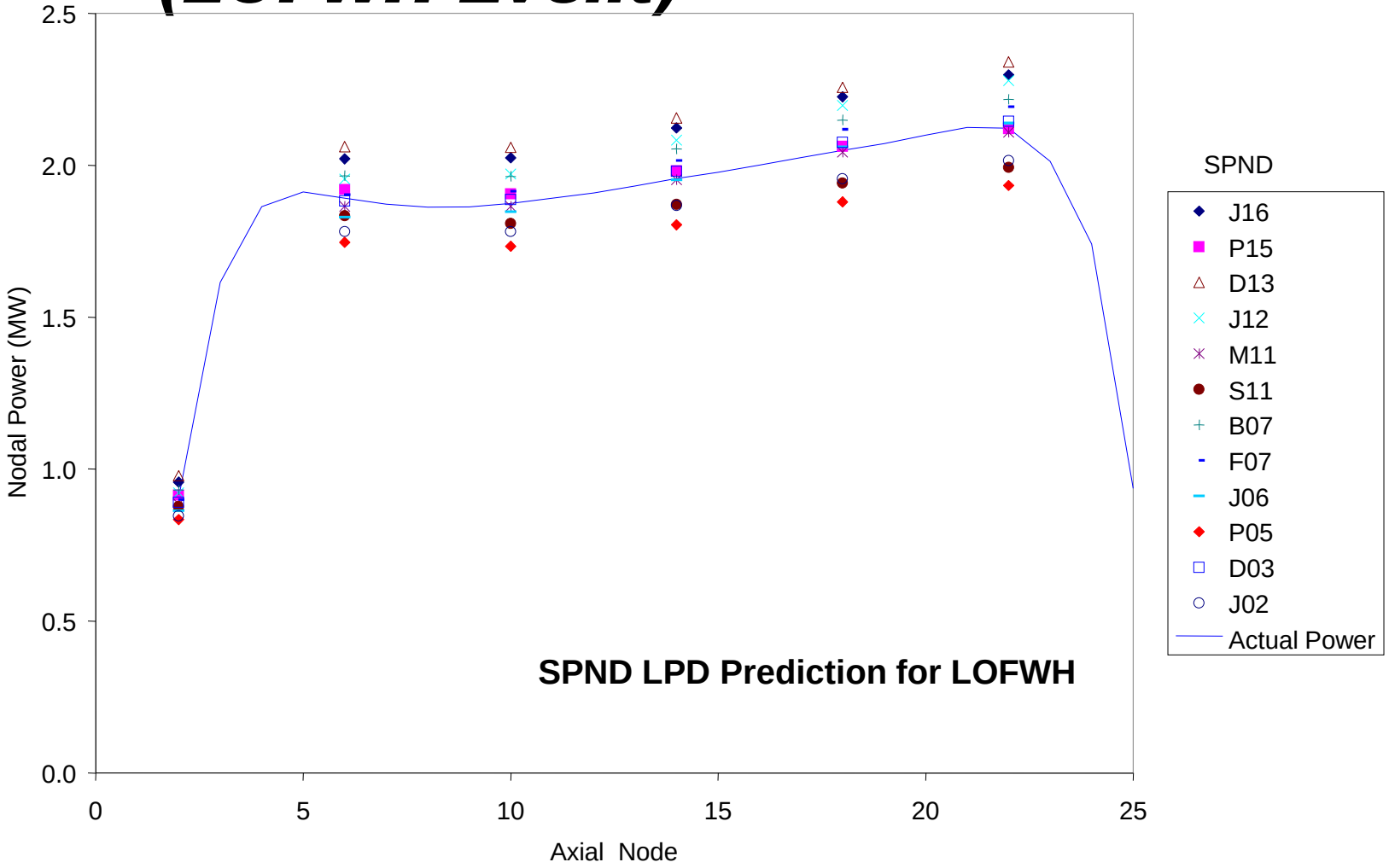
DNBR Confirmatory Calculation (Rod Withdrawal Event)



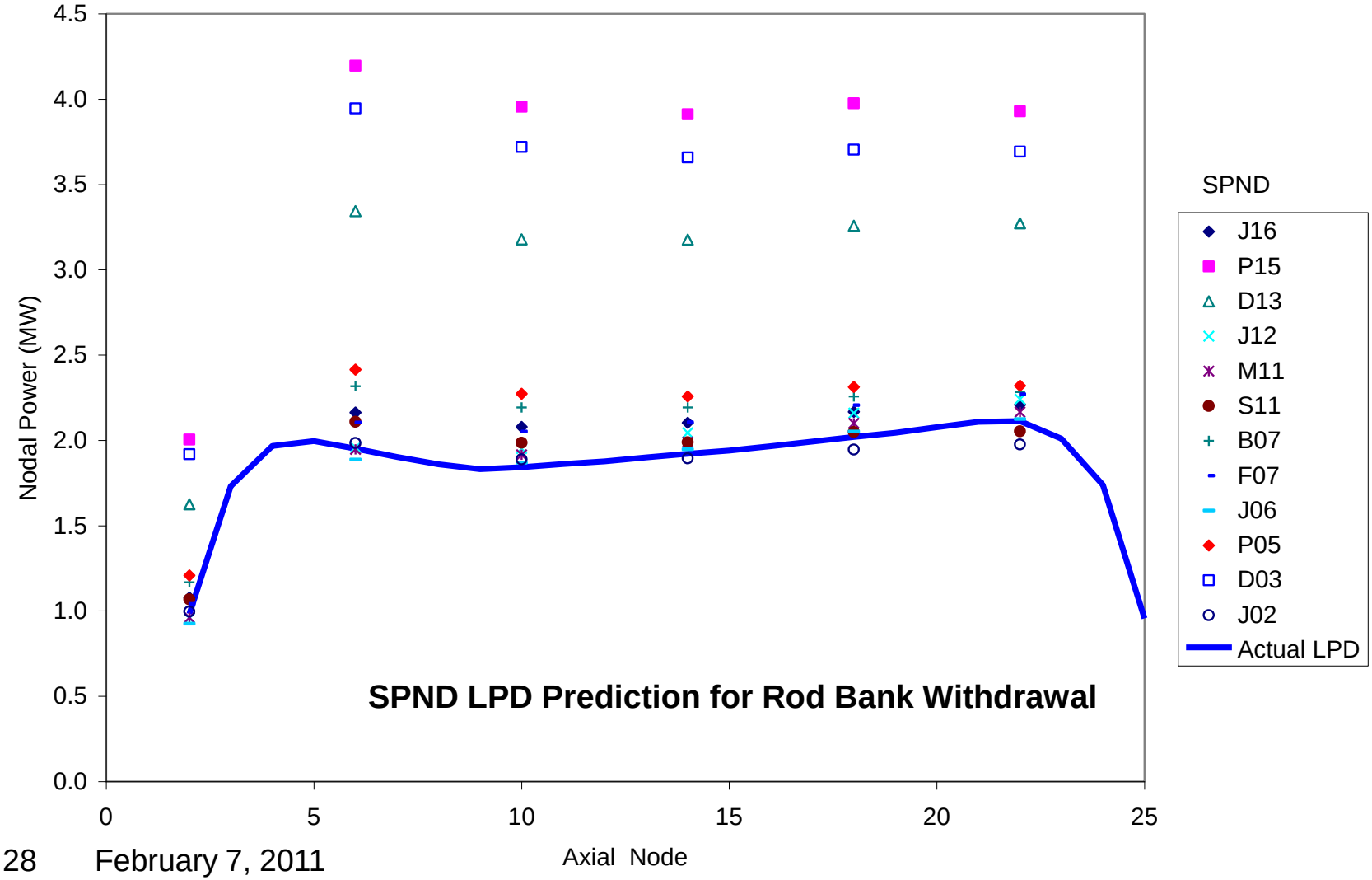
Radial Power Shift (Rod Withdrawal Event)



LPD Confirmatory Calculation (LOFWH Event)



LPD Confirmatory Calculation (Rod Withdrawal Event)



EPR DNBR Calculation Results Evaluation

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Most SPND strings predict lower DNBR and higher LPD values than the real TRACE hot bundle during the transient
- The EPR protection system using signals from in-core SPNDs should be able to generate reactor trips in time during these two transients
- The methodology of predicting on-line DNBR and LPD is shown to be conservative

Conclusions

- Staff has made significant effort to perform independent confirmatory analyses as part of NRC U.S. EPR FSAR review

- The confirmatory analysis results support the staff conclusion documented in Chapter 15, Group 1, Phase 2 SER

Acronyms

- COL – combined license
- CRDM – Control Rod Drive Mechanism
- DNBR – Departure From Nucleate Boiling Ratio
- FSAR – Final Safety Analysis Report
- LPD – Linear Power Density
- SER – Safety Evaluation Report
- RAI – Request for additional information
- SAFDL - Specified Acceptable Fuel Design Limit
- SG – Steam Generator
- SPND – Self Powered Neutron Detector
- TS – Technical Specification