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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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METALLURGY AND REACTOR FUELS SUBCOMMITTEE

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AFTERNOON SESSION

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THURSDAY

DECEMBER 15, 2016

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 1:00 p.m., Joy Rempe, Chairman, presiding.

COMMITTEE MEMBERS:

JOY REMPE, Chair

DENNIS C. BLEY, Member

CHARLES H. BROWN, JR., Member

MICHAEL L. CORRADINI, Member*

WALTER L. KIRCHNER, Member

JOSE MARCH-LEUBA, Member

DANA A. POWERS, Member

PETER C. RICCARDELLA, Member

GORDON R. SKILLMAN, Member

JOHN W. STETKAR, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

WILLIAM SHACK

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER BROWN

ALSO PRESENT:

ALI AZARM, IESS

CHRISTOPHER BOYD, RES

KEVIN COYNE, RES

RAJ IYENGAR, RES

MICHAEL SALAY, RES

SELIM SANCAKTAR, RES

*Present via telephone

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

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

(1:00 p.m.)

CHAIR REMPE: This meeting will now come to order.

MEMBER STETKAR: She didn't turn on --

CHAIR REMPE: The mic I got, shoot. Okay, this meeting will now come to order. This is a meeting of the Metallurgy and Reactor Fuel Subcommittee. I'm Joy Rempe, Chairman of the Subcommittee.

ACRS Members in attendance are Jose March-Leuba, John Stetkar, perhaps Dennis Bley will join us later, Matt Sunseri, Dana Powers, Dick Skillman, Pete Riccardella, and we have Mike Corradini on the bridge line. And we have our consultant Dr. Bill Shack. Christopher Brown of the ACRS staff is the designated federal official for this meeting.

The purpose of this meeting is to receive a briefing on NUREG-2195, Consequential Steam Generator 2 rupture analysis for Westinghouse and Combustion Engineering plants with thermally treated alloy 600 and 690 steam generator tubes. We'll hold presentations from representatives from the Office of Nuclear Regulatory Research.

The Subcommittee will gather information, analysis relevant issues and facts and formulate

proposed positions and actions as appropriate for deliberation by the full Committee.

The rules for participation in today's meeting were announced as part of the notice of this meeting. Previously published in the Federal Register on December 13th, 2016. We have received no written comments or request for time to make oral statements from members of the public regarding 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. Participants should first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

We do have one bridge line established for interested members of the public to listen in. The bridge line number and password were published in the agenda posted on the NRC public website.

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

meeting.

At this time, I'd like to ask everyone in the room to silence your phones and other electronic devices.

Colleagues, are last Subcommittee meeting on this topic was back in April 7th, 2015. And today I've asked the Staff to describe how they've updated the draft version of NUREG-2195 since that time and how they've responded to, or plan to respond, to comments provided by us, during our last subcommittee meeting, and provided by the public.

Also, there's a lot of popping on the line and I'm not quite sure what that's about. Okay, anyway.

Okay, while we're figuring out what's happening, and by the way, we have been joined by Dennis Bley at this time. One of the Members of ACRS.

But today I'd like to give everyone sitting around the table a heads up that I'm going to be asking about the path forward on this effort. In my opinion, the Staff has made good progress on their effort.

However, as with many research efforts, there's been funding and staffing challenges that lead to some decisions that required limiting the scope of this effort. So at the end of the meeting I'm going

to be asking for your thoughts regarding the product that we've reviewed has adequately met the user need for this effort. In addition, the staff has some suggestions to expedite publication of this NUREG and I'd like to hear your thoughts on this topic.

And at this time, we're going to start with the meeting. And I'm going to ask Kevin Coyne of RES to make introductory remarks.

I guess before you start, Kevin, is there an issue we need to think about with respect to this room and the ventilation folks?

PARTICIPANT: It's a good vent, I don't know what's going on.

MEMBER MARCH-LEUBA: Most likely they turned on the auxiliary heat.

MEMBER BLEY: Maybe it's the first time the heats been on.

MEMBER MARCH-LEUBA: Yes, it's actually the heat.

CHAIR REMPE: Okay. So at this time, let's have Kevin just go ahead and start the meeting.

MEMBER BLEY: Just watch for smoke.

CHAIR REMPE: Yes, if anyone sees smoke holler and we'll evacuate.

MR. COYNE: Kevin Coyne, Office of

Research. I'm the acting deputy for the division of risk analysis.

Thank you again for this opportunity to speak to the Subcommittee again. I believe this is the fourth or fifth time, depending on whether you count in a whole committee meeting in the mix that we've had an opportunity to meet with the committee.

One follow-up issue from the steam general action plan that the Staff had spent quite a bit of time pursuing that action plan was closed in late 2009, with the idea that this work was going to be the one piece that was going to be resolved from that plan.

So we've done it in response to a NRR user need. User Need 2010-005. For those that are interested.

And the main part of the request was to come up with a simplified method for addressing Consequential Steam Generator 2 ruptures. Mainly for the purpose of performing reviews for license renewal and for the significance determination process.

It's been a long time since 2010. We've had several conflicts with the work. Staff diversion did more critical projects for the agency. Had been a major obstacle we had to overcome.

So a lot of the Fukushima follow-up work

had impacted our ability to make continual and steady progress on it. But we managed to work through the project pretty well.

We had to de-scope several efforts, as you had mentioned, which ended up being a smaller effort than we had initially envisioned. But necessary in light of some of the conflicts we had to work through.

But I think we've come up with a reasonable approach and a reasonable document of the effort we've done. We've gone through a public comment period, we're prepared to talk about some of the comments received, in addition to the comments we received from ACRS members.

As part of the preparation for the meeting, we did provide an updated version of the report. Updated from the version that went up for public comment. Along with a detailed comment tracking table the staff has been using to resolve and track the various comments.

With that, we look forward to the interchange today. And I'll turn it over to Dr. Raj Iyengar to kick off the remaining.

DR. IYENGAR: Yes, good afternoon. I'm so very pleased to be here. Thank you, Dr. Joy Rempe, ACRS Member and Kevin Coyne and ACRS Staff Christopher

Brown. And our staff as well as the public.

It's been a pleasure to be working on this project for a long time with two challenges. And as Kevin has highlighted, we have been engaged with your Committee for, since inception of this project. And a number of times we've come here, in front of you, to talk about the project.

And you know the evolution of how it started and how it progressed and where we are now. And we've also had a number of meetings with Dr. Rempe and her colleagues of hers.

So I think much of it is not new to you, so you have seen most of it. So I just wanted to highlight a couple of things we have done since the last meeting.

That we have received comments from ACRS Members and we've reviewed them and addressed them in the revised draft NUREG. We've also issued that for public comment and received a number of public comments.

Really quite insightful comments. And we've addressed them as well in the NUREG. And we've also provided responses.

And we revised the NUREG and NUREG-2195. And it's available as a draft right now. And certainly

we would be getting it to you for you in adamant review.

And I think Kevin already went through the evolution of the project, where it started in 2010. And the reason why it was started. It was an offshoot of the steam generator action plan.

And we have had this work scoped three obvious divisions. I'll probably say all three are still here with all the changes that happened.

So very good interaction between the various divisions in the Office of Research as well as NRR. And we've gone to a number of, you know, quite numerous meetings.

And the essential things that you will see here are the thermal hydraulic work and the structures and materials related studies. These are done in-house largely. I think in almost, near 100 percent, in-house work.

And you can see how challenging that would be with all the challenges that staff faces to do many of different things. And a rotation of staff and stuff.

But we persisted that, and thanks to your engagement, encouragement and patience through six years now.

And the PRA work has been contracted out.

And Dr. Ali Azarm was the lead for that. He will be providing some of his work and his insights.

And the part of this is there will be a simplified calculator that was not essentially part of this work but it will be tied to this work.

And during the last four years, you know we've gone through all the challenges we've been to, we've discussed the resources challenges due to Fukushima and many other things. But I think we persisted, our persisted, and got this to good fruition, I think, to a finish line.

And of course, it's left you, you know, you can weigh in on whether this would satisfy, would have satisfied the work user need request.

CHAIR REMPE: So just to be clear --

DR. IYENGAR: Yes.

CHAIR REMPE: -- at one point there was a recommendation, I thought, as part of the user need to issue some subsequent guidance from this work. But at this time, this NUREG is it, right?

DR. IYENGAR: Yes. At this time this NUREG is it, but I think it's coming. Kevin, do you want to weigh in on this?

MR. COYNE: Yes. The work has evolved somewhat from the original user need. We did a

rescoping of it. I believe we briefed the Subcommittee on it several years ago, about trying to streamline the project.

At one point there was more guidance that was envisioned to be developed as part of the NRR user need. At this point, what the office, what NRR would like is that once the report is issued, we would probably develop a section to be inserted into the handbook, which is used to support the significance determination process.

There is already simplified methods to handle inducing generator tube rupture in that handbook, but that section would be updated to include the guidance that we've developed as part of the NUREG.

CHAIR REMPE: Thanks.

MR. COYNE: And that would be the final deliverable that we're --

CHAIR REMPE: Okay. I hadn't realized that that was still the plan. Thank you.

DR. IYENGAR: Thank you. So today's presentation will cover three sections broadly. The thermal hydraulics analysis supporting the PRA, and Mike Salay would be presenting that.

And we received a number of comments on that as well as for other sections, so we'll be

providing our responses and how we have addressed those comments.

And the second one would be a brief comment and response for the structural analysis work that I did as part of this effort.

And then following that there would be a PRA assessment presentation given by Dr. Ali Azarm and Selim Sancaktar.

And each section, because it's taken a long time, I did this work three, four years ago, I forgotten completely about it, and I bet everybody else would have, so what we planned to do is I'll give a little bit of a short background. Just for the benefit of all of you.

And certainly this is not going to be as intensive as we had given earlier, but it will give a little bit of backdrop to what we are talking in terms of discussing the comments.

And I did want to, before I turn on to Mike Salay, I wanted to personally thank Dr. Joy Rempe and especially Dr. Dana Powers. I think both of you persisted.

And I think your encouragement was phenomenal, I think, in our efforts to completing this. And I really thank you for all of that. And it's been

a very interesting work, for me at least. So, Mike, you want to come over here or --

DR. SALAY: No, I'll just -- good afternoon, Mike Salay. I'll talk about the scenario that we're looking at. And again --

MEMBER POWERS: I can't help but interject that Dr. Salay is here on vacation because this is what he considers a vacation.

DR. SALAY: No, actually I'm between vacation.

(Laughter)

MEMBER POWERS: That tells you how miserable his normal life is.

(Laughter)

DR. SALAY: Well, yes, and anyways.

(Laughter)

DR. SALAY: So yes, I'm going to talk about the scenario. The stuff I'm presenting will be what was done, a lot of it was done, what was in this first presentation, I have two presentations, this one is just the overview. And a lot of it is what was done in the steam interaction plan, which I'm not sure when it started, but it was 10, 15 years ago.

So anyways, yes, I'll talk about the description in the analyses, how we combine CFD and

system codes, the experimental basis and then differences between CE and Westinghouse.

So what we did in this work, we focused on CE because that wasn't done before. So we're considering the station blackout sequence.

It's a low probability event. It involves a loss of offsite power, a loss of diesel generators and loss of auxiliary feedwater.

Your reactor inventory boils off, you have fuel -- it boils off, your temperature goes up. And you release fission products.

You have high temperature and pressure. This high temperature and pressure stresses the whole RCS system and your pressures at, essentially your PORV or SRV pressures, and your temperature is rising. So something is going to go. And this is either going to be the steam generator tubes or something else.

Why we differentiate between the tubes and something else is because if the tubes fail you go, well, first I'll go with the RC. If something, other RCS components fail, it dumps into containment where the fissions project can attenuate and then get released, little by little, if there's containment leakage.

However, if a steam generator tube fails,

you're already on the secondary side. And if a valve sticks, or by operator action your secondary side relief valves are open, you have a direct path for your fission products to the environment. So they bypass containments. So it's a containment bypass situation.

MEMBER POWERS: Do they go directly to the environment or do they go to an aux building?

DR. SALAY: Into where?

MEMBER POWERS: To an auxiliary building.

DR. SALAY: Well, auxiliary building.

MEMBER STETKAR: Not if it's an atmospheric relief valve, it's going outside.

MEMBER POWERS: Well, that's what I asked him. Which way is it going?

MEMBER STETKAR: Mostly likely it either goes outside, through an atmospheric relief valve, or it could go to the main condenser and get filter there if the MSIBs are open. Then you get turbine building kind of stuff.

But if the MSIBs are closed or you got a stuck open relief valve it's going outside.

MEMBER POWERS: Well, it kind of makes a difference.

MEMBER STETKAR: It does. It's a PRA scenario.

MEMBER POWERS: When we do the source term to the environment --

MEMBER STETKAR: Exactly.

MEMBER POWERS: -- where it's directly going out, or it's going through the aux building.

MEMBER STETKAR: But again, the PRA that feeds into this should know whether or not it's open to the main condenser or open to the environment.

MEMBER POWERS: That's good, now I want to know.

MEMBER STETKAR: Well, it could be either.

MEMBER BLEY: They have to look at both cases.

MEMBER STETKAR: You have to look at both cases. And depending on how you get into this, one case is more likely than another.

MEMBER POWERS: Okay, which one is more likely?

MEMBER STETKAR: It depends on how you get into it in the plant. I mean, there are many different --

MEMBER BLEY: We're not getting anywhere.

MEMBER STETKAR: This has been characterized as a station blackout and there's no particular reason the MSIBs should go closed under a

station blackout unless your plant has air operated valves that, for example, fail closed. And in which case they might be closed.

So it depends on your plant. If this is a -- but there are many other ways of getting into this that don't involve a station blackout. It's just easier to think about station blackouts.

And those events could involve conditions where a steam generator went dry with a stuck open relief valve and the operators isolated it. But it has an open valve and therefore it's going to go outside. But it's not a station blackout.

So it depends on the input from the scenarios in the plant. As far as which particular release path is more likely during a particular event scenario.

Offsite release. You know, potential offsite release for a source.

MEMBER BLEY: Well, and what you didn't say is even if it goes to the condenser, if it can get -- the non-condensables can come out.

MEMBER STETKAR: The non-condensables can come out because --

MEMBER BLEY: So yes.

MEMBER STETKAR: -- there's a vent path

from the condenser that way.

MEMBER POWERS: And of course, I don't care at all about the non-condensables.

MEMBER STETKAR: That's fine.

MEMBER POWERS: I do care about the particulate and the path they follow makes a difference.

MEMBER STETKAR: And going to the condenser is much different.

MEMBER POWERS: But it still --

MEMBER STETKAR: In neither case will it go to the auxiliary building. I mean it's really hard to get these into something that's called an auxiliary building. Or a reactor building. That kind of thing.

DR. SALAY: As long --

MEMBER POWERS: But you still don't know.

MEMBER STETKAR: That's correct, you don't know.

DR. SALAY: As considered in the calculations, it was considered to go directly into the environment, unless that was adjusted in the PRA.

And the main point I'm trying to make is that if your tubes fail, you can bypass your containment and there's less opportunity for attenuation. And so determining whether steam generator tubes fail or some

other RCS component fails is important when you're looking at your consequences.

So here we'll look at a fast scenario. And this was for Westinghouse calculations during the steam generator action plan.

So you have loss of offsite power, failure of diesels and failure off aux feed. Your secondary starts boiling off.

And you can lose primary inventory through the ports, SRVs. And more so in the Westinghouse and CE through your pump seals.

When your secondary side becomes dry, your pressures starts rising and, again, you start losing primary inventory. But you have a single-phase liquid natural circulation through the system.

After the tube, the top of the tubes is, level goes below that, you break the natural circulation and you lose more and more inventory.

Once you lose enough inventory you're going to have to start having natural circulation, counter-current natural circulation through your tubes goes up.

You have hot gas from the core, go up to the hot leg, go up through the one set of tubes, down through another set of tubes. Mixes some here and some

gases come down here. Your core is heating everything up so this is continually getting hotter and hotter.

Your core uncovers, you oxidize which can, your clad oxides which adds more heat, which accelerates the heat up.

It should be pointed out that this scenario was considering an immediate loss of aux feed. A more realistic scenarios involved later loss that it operates for some time or some operator action keeps water in the system.

And here's one of the Westinghouse, results from the Westinghouse from the previous NUREG. At about 100 minutes there's steam generator dry out. They start getting 3D recirculation effects here and your temperature goes up. You start significantly oxidizing your clad.

And these are the vertical lines, indicate the failure points for the hot leg, the hottest tube and an average tube.

And there are several points of interest in your RCS. You have, of course, the steam generator tubes. Other potential points of favored the hot leg.

The pressurized surge line, your lower head and your instrument tubes. It's not marked on here.

Ultimately this is how much sea leakage you're getting and whether you have a loop seal. Whether water is collected in this loop seal or has been cleared.

The thickness of the walls matter because your hot leg is two and a half inches, your surge line is one and a half inches and your steam generator tubes are quite thin. And because the thermal response time is effected by your thickness.

CHAIR REMPE: Mike, if you don't mind going back for a minute. I looked at your slides and the questions that you're going to be answering today, or comments you're going to be addressing today, and I looked at the handout that was provided to us in advance and I saw the response about the melting temperature comment that I made at the meeting and I guess the Staff's response was, well, it's consistent, that my concern was that the references I had Inconel at a lower melting temperature than Stainless Steel.

And got references from the vendors, from Inconel 600 and Inconel 690, and they're all showing a lower melting temperature. And your response, the Staff's response back was, well, it's consistent with what's in the MELCOR and some training course that some guy from Sandia gives I guess in his handouts.

And that may be good, but I guess my other, the follow-on question is --

DR. SALAY: Reconciled both the --

CHAIR REMPE: Well, does it matter? Because I assume you're doing some sort of structural evaluation and so you would have the tube failed before it melts. But maybe it doesn't matter.

You've got what I consider to be incorrect lumping of melting temperatures for the Stainless Steel and Inconel in your slides and I guess in MELCOR. And so maybe it doesn't matter for this analysis, but I'd like to hear that it doesn't matter.

Does it matter at all that maybe you should be having the lower temperature by at least 100 degrees for Inconel?

DR. SALAY: Well, if you're looking at creep rupture before melting then --

CHAIR REMPE: No.

MEMBER POWERS: Mainly, in general you're right. Inconel does melt at a lower temperature than Stainless Steel. However, it's creep rupture temperatures that --

CHAIR REMPE: That's what I'm asking is does it matter that you've got the, what I consider incorrect temperatures here on this slide. But maybe

someday somebody ought to fix what's in MELCOR or something. If that's the name that it has.

But I'd like to reconcile that it doesn't matter in the analysis.

MEMBER POWERS: I mean, it's the creep temperatures of Inconel.

CHAIR REMPE: Sure.

MEMBER POWERS: I mean, that's why Inconel was invented is it doesn't creep as badly at elevated temperatures as does Stainless Steel.

CHAIR REMPE: And that's what I would like to hear, rather than the response I got back on that.

MEMBER POWERS: I mean, it seems to me that the thing to do is to just look at the tertiary creep rates.

CHAIR REMPE: Yes. And then I assume that the properties for Inconel are obtained with sufficient accuracy. Because I don't have something like that that I can check as easily, but I hope that they are --

DR. SALAY: I just checked a few references to see --

CHAIR REMPE: Okay.

DR. SALAY: -- how they compared.

CHAIR REMPE: Yes.

DR. SALAY: So we have to look at the other references and --

CHAIR REMPE: Okay. I can send Chris, if that would help, the vendor information and things like that.

DR. SALAY: Sure.

CHAIR REMPE: But again, does it affect this analysis. Thanks.

DR. SALAY: All right. And so the scenario you're looking at is the so called high-dry-low scenario where you have high primary site pressure, dry secondary side, and low secondary side pressure.

And there are two flow patterns that are possible for severe acts of natural circulation. And it depends on whether there's water in your loop seal or not. And I'll go over these on the next slide.

So the full-loop natural circulation, water has been cleared from the loop seal. And this loop seal clearing can be affected by several things.

The depth of the pump loop seal, reactor coolant pump seal leakage rate and elevation, primary site pressurization rates and downcomer bypass flows. So the ability of gas to cross from up here to here.

And Westinghouse PWR studies have

indicated that loop seals are more likely to remain blocked with water. So it's important to do this.

And --

MEMBER POWERS: Very recently we had people in here doing a DBA analysis, which is not your station blackout analysis, and they explained, very carefully to us, that within a matter of a few seconds the loop seals were cleared. And they remained clear for about 1,000 seconds.

And so somebody, who will remain nameless, ask them, how do you know that. And they said they'd done all kinds of experiments, in fact, to pursued themselves that these loop seals would clear.

That's all they said. And so I asked them, when did they refill. And they said, well they may, but they didn't carry their analysis out beyond 1,000 seconds. So they didn't seem to know, or care, because they were doing a DBA analysis.

And went on, you say the PWR studies now have indicated that the loop seals are to remained block. What were those?

DR. SALAY: Those were the NUREG/CR-6995, the ones in the steam general action plan. And they could come up with scenarios where it would clear every time, and that was basically if you allowed leakage from

the upper internals to the loop seal. So bypass down here it would always clear.

MEMBER POWERS: So water drops below that downcomer plate then it would clear at that point?

DR. SALAY: Not water level dropping. If you modeled, there's some leakage when you put everything down, there's some gas leakage. If there's enough gas leakage there, the gas coming out the pumps would come from here instead of bubbling through the loop seal. And that's -- so there's another source of gas to leak.

MEMBER POWERS: So on your plot of vulnerable locations, that leakage path there should be arrowed as well?

DR. SALAY: Yes. It's mentioned, but it wasn't -- yes, that's one of the things I was going to look at and characterize as part of the pump seal, as part of the loop seal clearing analysis. Which I ended up not doing because it doesn't really matter that much for CE, and I'll get into that in the second presentation.

MEMBER MARCH-LEUBA: If I remember correctly, the loop seal is clear with an increase in high pressure in the upper plenum. I mean, you generate steam in the upper plenum which then blows the

loop seal away.

DR. SALAY: Well, there are a few different --

MEMBER MARCH-LEUBA: At least --

DR. SALAY: -- there's some where it can go one way or the other way. There's gases coming through if you have the bottom blocked off. There's also --

MEMBER MARCH-LEUBA: No, this was cleared before you melt the core. I mean, while you still have water in the core, you will have high water level in the downcomer, low water, and then you start creating pressure in the upper plenum, which will push the water level inside the core out and clear this monometer.

DR. SALAY: Yes.

MEMBER MARCH-LEUBA: On the same elevation you have it on the loop. And when you get to an elevation which is larger at the loop, the loop clears. So you never get to uncover the core before you clear the loop.

DR. SALAY: But the monometer, if you have the gas leak here, you're getting gas so it doesn't actually close and seal and go over. If you have this leakage between the hot leg and the upper internals, the gas --

MEMBER MARCH-LEUBA: If the cold leg and the hot leg have the same pressure, then it never clears.

DR. SALAY: Yes. And so it also, if you have condensables and you pressurize, that can go down. I mean, you can condense. And so that will --

MEMBER MARCH-LEUBA: Yes. Well, we've been told by the manufacturer, by the vendor that runs these experiments, they have run the calculations and the experiments and the loop clears. Very fast.

Once one loop clears, then you release that pressure. And then the other two or three don't clear. But the first one goes, poof.

And then if it stays clear, because there's so much vapor flow through the cold leg, that this wind, the wind carries whatever water gets in the flat area of the cold leg. So that there is so much wind that it doesn't allow you to backtrack it.

DR. SALAY: Yes, I heard there were -- yes. And so the, I was addressing, to get the point.

The main issue is that you have this high velocity flow through there which allows hot gases to hit the steam generator tube, which opposes the --

MEMBER MARCH-LEUBA: This is still way before the gases are hot.

DR. SALAY: Yes. But I'm saying where -- What happens, why do we care about full-loop natural circulation versus closed-loop?

MEMBER MARCH-LEUBA: So is the presence of the seal relevant to your final conclusions?

DR. SALAY: Well, we're looking only at the closed loop. I was just pointing out the two scenarios.

We didn't even, I mean, I started looking at the loop seal clearing issue, but then we realized it didn't matter for our analysis. And that's one of the things we cut out. And so it didn't really --

MEMBER MARCH-LEUBA: So it doesn't affect the final results? For --

DR. SALAY: For CE it will make some impact, but there's a limited, and that's one of the things I answer in the second, well, I considered --

MEMBER STETKAR: Would it make much of a difference --

DR. SALAY: So basically --

MEMBER STETKAR: -- for Westinghouse plants?

DR. SALAY: For Westinghouse it makes much more difference, and I think it will be clear why after I do the next few slides.

MEMBER MARCH-LEUBA: Okay.

DR. SALAY: And so counter-current natural circulation. If you do have a closed loop seal, so you have a water filled loop seal, and so you're required to have, you can't have flow going through the whole system, and so it ends up, your hot gases have to come up through the hot leg, up through, be cooled in the steam generator but the flow has to go up through some tubes, down through some other tubes, mix here, come back and then come down to the core.

And system code models require external information to insure the consistency. System codes just can't calculate this so you need some other way to calculate it. Either hand calculations and correlations or found out that it's not precise so you have to use CFD codes and then apply the results of the CFD codes and implement them in the system code to get, so that the system code reproduces the behavior in the CFD codes.

MEMBER MARCH-LEUBA: So once you're in non-condensables, I mean you've lost all your water, what's driving the flow back into the vessel?

DR. SALAY: I mean, it's just a natural convection. So it --

MEMBER MARCH-LEUBA: Does it mean it cools

the --

DR. SALAY: -- cools here --

MEMBER MARCH-LEUBA: -- cools a little bit and --

DR. SALAY: -- hot here. And the way pathway is up through here, down through here, back and mixing here.

MEMBER MARCH-LEUBA: A very low driving force.

DR. SALAY: Yes. But --

MEMBER MARCH-LEUBA: I suspect that whatever that flow you assume going through the hot leg will tell you how much heat ends up in the steam generator versus the top of the vessel.

DR. SALAY: Yes. And I'll go over the way it's been characterized and how differences between Westinghouse and see --

MEMBER MARCH-LEUBA: It's going to affect your, I mean, I'm not one to ask a lot of uncertainty on the final results, but it can be a factor of ten off.

DR. SALAY: We did vary parameters quite a lot in some of these parameters and it made a few minutes of difference, in terms of both absolute and relative tailor to timing.

MEMBER MARCH-LEUBA: Well, I haven't seen

your presentation so I'll wait. But I mean the worst thing in my mind is, what's going to break first, the top of the vessel or the steam tube?

(Laughter)

MEMBER POWERS: You've certainly hit the key issue here.

DR. SALAY: Yes. What fails first is key. Well also, that used to be the key. And we sort of -- now, if you have some tubes break but it doesn't depressurize the system, you still look at this continued calculating scenario. And if the hot leg breaks after, that limits how much your fission products can be released.

So that's a consideration now. So there's been some evolution. Oh, Chris has some.

MR. BOYD: I was just going to chime in a little bit. I worked a little bit on this before Mike did.

DR. SALAY: A little? Fifteen, 20 years.

MR. BOYD: I'll make two comments. Is this on?

MEMBER BLEY: Chris, we need your name and where you're from, on the record.

MR. BOYD: Oh, I'm sorry. This is Chris Boyd and I'm from research and I work with Mike.

So the first thing I'll say is we were concerned about the loop seal clearing when we did, I worked a little more on the Westinghouse stuff, and we did a lot of sensitivity studies on what would do that.

There's a significant bypass in a lot of the Westinghouse plants between the upper plenum and the top of the downcomer. In some plants that can be as big as a square foot or so.

Now, we played with that area, and you can shrink that down and then you blow the loop seal. But in values for that, that we thought were plausible, we were well into the case where the loop seal would remain filled with water and not get pushed out, because of that bypass.

There was also a lot of issues with the gap between the hot leg and the vessel and how much leakage you get there. So a lot of that was studied.

And the conclusion was that we don't blow through, at least on the Westinghouse plants, the loop seal.

The other comment I'll make, we did a bunch of sensitivity studies on the other topic Mike was talking about. And it's already slipped my mind what I was going to say so I'll drop that for now.

(Laughter)

MR. BOYD: I apologize. I got onto the one thing and just --

CHAIR REMPE: Just as a follow-up, to further distract you. Well, I heard we're going to have guidance coming out later after this documents done.

Will the guidance be that you need to do a plant specific design or as you've indicated, well, we've come to the conclusion for the Westinghouse designs that certain things can occur?

I know at the end of one of the conclusion chapters that you have some guidance, but it's not specific. And I just am wondering how detailed the guidance will be.

MR. BOYD: On that I think --

CHAIR REMPE: And maybe this goes to mind for --

MR. BOYD: I think somebody was going to talk a little bit about that at the end.

CHAIR REMPE: Okay.

MR. BOYD: The other comment I was going to make was on the hot leg flow. And of course, that effects the energy balance.

We did a lot of, where does the energy go studies with the Westinghouse plants. And we

basically throttled the flow artificially down.

If you throttle it way down, you don't let much energy in. You keep the energy in the vessel and you melt the lower head. But we had to go really far to do that.

But when you change that mass flow, the loop is sort of a single item where you send things out past the hot leg it also goes past the tube. So when we sent more energy out there, both the hot leg and the tube saw more energy. If you send less, both see less.

The relative difference didn't change a whole lot, so the uncertainty is not as, there's a lot of uncertainty there. But it doesn't, it's one of those variables that changes the timing of both.

The relative timing wasn't too significant so we stayed with our best estimate and we ran some sensitivities around that and we're happy with it.

DR. SALAY: And then we also looked at B&W plants but concluded that you wouldn't really expect really significant flows and that they wouldn't be challenged. So these plants have not been part of the recent past decade or two severe accident induced failure studies.

And so the TH analyses that were done during this steam generator action plan, which was

completed about five years ago, they focused on Westinghouse plants. And they looked at a little bit at CE plants but they didn't, CE plants didn't receive the same level of attention. Addition of the hottest tube and other things that the Westinghouse analyses and the steam generator action plan did. And this is documented in NUREG/CR-6995.

So in this work we looked into the CE plants in a little more detail. And for the failure calculations, they used the previous Westinghouse work to look at the risk from Westinghouse.

And so you have to use a system code and CFD code. And again, CFD predicts a spatial flow and system code predicts the transient behavior. And you use the CFD results as input.

And so the results of the system code give you the transient. And it can be combined with those CFD to come up with like a transient spatial temperature distribution.

And CFD calculations were validated against the Westinghouse 1/7th scale experiments. And here you see some of the drawings of the Westinghouse 1/7th scale experiments. And these were used to demonstrate the counter-current flow behavior. They didn't focus on tube integrity but they provided

valuable insights.

And there have been scaling studies performed for this. And they were analyzed in 2001.

And so the calculations, the CFD calculations that were done, they modeled the 1/7th scale. Took the 1/7th model as full scale, but still using the test facility geometry.

And this is Chris' work. And then modeled Westinghouse and compared to the test facility. And then formed many sensitivity studies. Heat transfer, surge line orientation, hydrogen content and tube leakage rates. And for this work, the CE plant design was modeled.

And again, how do you apply the CFD results to a system code? At the top you see CFD results, which by nodalization. And here you have the system code nodalization, which is very course and cannot calculate the speed however.

You have hot flows going up, cold flows going down. And you have this plume that changes shape and time. So you have the temperature distribution going in.

So somehow you have to apply this to the system code. And based on the methods used for the hand calculations, before these were adopted to take the CFD

results and apply them to the system code.

And they have a few parameters. There's the coefficient of discharge, which decides the flow in your hot leg.

Your inlet plenum mixing fraction, which decides how much of your flow mixes, both coming down and going up.

Your hot tube fraction, how much of your tubes is taken up by this hot plume.

Your circulation ratio, how much of the flow goes through the tubes relative to the flow in the hot leg.

And also, the CFD provides a normalized temperature distribution. And surge line split/mixing.

One thing about the choice of these parameters, that have been done for a long time, is that as your temperature rises, the behavior stays relatively constant. And so if you look at early in the temperature rise and late in the temperature rise, the same parameters, the parameters, they kind of fix.

So you can use it to characterize it throughout the whole sequence as it heats up. And I guess here you see a hot temperature distribution.

Okay, in our work we looked at CE plants.

And CE plants differ some from, the CE plant we looked at differs significantly from Westinghouse plants.

The CE plant we looked at had a replacement steam generator that had a very shallow inlet plenum. And also, you have a much lower hot leg length to diameter ratio.

So what effects how hot your tubes, the temperature that your tubes see relative to the hot leg? I mean, you're looking at what fails first, other RCS components, depends on how much mixing occurs as you go down the hot leg to the tubes.

So these lower hot leg length of diameter ratio here. So you have a plume that's about that big, for CE. And it doesn't have to go very far.

And as far as Westinghouse, your plume, your hot plume is there and it has to go four and a half-length of diameter ratio. So there's more opportunity for mixing.

CHAIR REMPE: So, Mike, your Slide 20 says some CE plants have shallower inlet plenum. Do you have any feel --

DR. SALAY: No.

CHAIR REMPE: -- for how many some is? Is it 50 percent --

DR. SALAY: No, this is the one we had

information on so I don't --

CHAIR REMPE: I know now we're talking about an example rather than a representative plant --

DR. SALAY: Yes.

CHAIR REMPE: -- as requested in the user need. And so we've not done --

DR. SALAY: But you sort of have to, not sort of, you have to look at the geometer of an individual plant.

CHAIR REMPE: So the Staff just didn't have time and funding to go and do an inventory and say, well, 1/10th of them are going to be this way?

DR. SALAY: Chris Boyd has --

MR. BOYD: This is Chris Boyd again, I'll just make a comment. We requested geometer for maybe ten or 15 plants. We received five or six.

All of the Westinghouse plants that we received, and this was with replacing the generator. The fear was that a replacement generator could be a boutique design. They really can do whatever they want. Different manufacturers were making them.

In the Westinghouse space, all of the samples that we received looked about the same. At least if you held them at arm's length. I mean you could go up and see half inch differences here and

there, but basically it was the same bowl about the same distance.

In the CE plants, there was some variation, but that distance from the hot leg to the tubes wasn't always significantly closer than the Westinghouse design. And we didn't see anything that was outlandishly different.

But I agree that for any plant, somebody looking at this, they would want to look at the geometer. But in our sample, we didn't see significant, as significant of variation as we might have expected. Given replacements being generated from all sorts of sources.

CHAIR REMPE: It's been awhile since I read through the very lengthy report now, but is that documented in there about, you did request this?

Because I think it's actually a good response back to that question. I don't think I saw it. In fact, the response back to the question was, we're going to change it from representative to example.

But you have done a sampling. And it gives you a little more confidence that what we're seeing is --

(Simultaneously speaking)

MEMBER STETKAR: -- units? Seventy or so.

CHAIR REMPE: Yes.

MEMBER STETKAR: It's a pretty small sample.

CHAIR REMPE: Yes.

MR. BOYD: I like the language where we're being careful and saying it doesn't represent a wide swath. Since when only really only looked at about six sets of drawings. But in our limited statistics it looked pretty good.

CHAIR REMPE: How do you know it doesn't represent a larger, I mean, you've only got five out of the seven.

MEMBER STETKAR: You don't, Joy.

CHAIR REMPE: Yes.

MEMBER STETKAR: But trying to develop guidance based on a very limited set of samples of things that they could easily find can be dangerous.

CHAIR REMPE: Absolutely. But it does, right now it sounds like we've only looked at two and this is what we have developed. And I guess that's what I'm wondering if then could be saying anything else.

But, I mean I agree with you --

MEMBER STETKAR: Haven't heard anything

that doesn't say you need to look your particular plant.

CHAIR REMPE: Yes, I agree.

MEMBER STETKAR: Okay.

CHAIR REMPE: And maybe that's what needs to be in the guidance too. But anyway.

DR. SALAY: Okay. Anyway, this is sort of like the key figure. You have the CE plant again and your Westinghouse plant. You have the temperature of the hottest, the hottest temperature the tube sees is around .9, .95 on the CE plant. And this is normalized temperature.

And whereas in Westinghouse you get about, I think this is higher, around .5. It's a little higher here, but I think there is some meandering of the plume because one other aspect of the CE plant is that the plume comes in normal to the divider plate, whereas in the Model 44 it comes in at an angle so the plume moves around much more.

And so this is -- what happens when you rupture you loop seal is your steam generator tubes can see temperatures way up here. But for CE plants you're already up there. You can't really go much higher.

And that's why it's not a significant, whether the loop seal is clear. I mean, it gets the hottest temperature that your hot leg would seem maybe

a little hotter too. But ultimately already at that very high temperature.

CHAIR REMPE: Out of curiosity, on this you've emphasized that it's a replacement steam generator. Had the original steam generator had different dimensions? Just curiosity.

DR. SALAY: Yes.

CHAIR REMPE: Really?

DR. SALAY: We have a big drawing with, yes, I think they were more like the Westinghouse ones. Chris Boyd has something.

MR. BOYD: The CE plants did have different geometry. They weren't more like the Westinghouse plants, but they did have the close distance. They had a different shape at the bottom and different divider plate.

On the Westinghouse models, a lot of them did have the nice bowl design. Model C, Model D, Model 51, they all look very similar. Replacement and the original.

CHAIR REMPE: Okay.

MR. BOYD: In CE there was a little more variation, but the key components, up at the top, were still pretty similar.

CHAIR REMPE: Okay. Thanks.

DR. SALAY: So again, the issue was, with the standard original failure models, it would predict that your unflawed tubes could rupture before the hot legs. Under certain scenarios. But it's for CE.

And the big concern here was, unlike the rupture of a flawed tube, which you're only likely to have one or two in the hot area, you have all, I mean, a huge clump of tubes could be hot. Are hot. Are at the hottest point.

And so if those happen to fail, you could actually depressurize fast enough to prevent, if they fail fast enough, to prevent other RCS components for failing. Otherwise the RCS components fail and it limits how much you can release.

And so here are other system code considerations. Your pressurizer draining, your surge line orientation, whether it comes in horizontally or vertically.

Your core bypass flow, which we discussed. Your oxidation rate. Your core blockage nodalization, instrument tube failure, it's potential location.

How your primary relief valves behave. We'll discuss some of these. Your inlet plenum mixing recirc.

Heat transfer, tube heat transfer.

Secondary flows, mass flow. Leakage plugging. That's it for this.

I was also going to show animation that some of you have seen before. What the MELCOR results we got with current MELCOR analyses where -- there it is. All right, I'll make it big.

Yes, so here is MELCOR results for the first, the base calculation. Is that the full? Yes, okay.

And again, what you see here is -- and you have system pressures, primary pressure containment, Secondary Loop A. Loop A is the one with pressurizer.

Loop B is the one without the pressurizer. It shows the water levels. Some of the relevant water levels.

It shows what the void fraction is. So if you have a, it sort of indicates foam, it shows your SRVs and PORVs, primary and secondary. Green indicates that they're open.

It will also show, as the time goes on, the system temperatures. It will show some radioactive material in the system. And it will also show a creep rupture.

I mean, the radioactive materials will look like a yellow, green gas. And the creep rupture

in the CE, the components will get transparent, you don't really see it. And then when it fails, the creep rupture necks reaches one it will turn black.

And so when I started the secondary, the general scenario that I described before, is for Westinghouse results. It's going to have here your secondary.

Water is going to boil off and, here I go. And so there are, you have your station blackout, your secondary water is boiling off, your pressure is going down. Your pressurizer empties but it starts to fill again.

And when it went dry the pressure started going back up. And so you develop a bubble here and lose inventory.

And as you go over past, as the water level goes past at the top of the core, the gases start to heat up and the structures start to heat up. So you can tell the structures start getting a little warm and batteries die. And so you're switching to the PORV so the pressure goes up a little more above the core level so it's a DSRV level.

In about six hours a few things are going to happen. You're going to start leaking some gases, the Loop B steam generator fails. The creep ruptures

there and gases are going to go.

There it fails and the system is still heating up. And then the hot leg fails and the accumulators release water into the system. Again, some of you have seen this already, but.

All right, so I'll get onto the second presentation. Just describing the work that we did here and some of the responses to the ACRS and public comments.

Yes, I am repeating a little bit. I wasn't sure if I was just giving this or was also going to give the previous presentation at first.

So for what we did for the TH analyses is provide TH behavior for the clutched engineer plants to be used with the calculator used to calculate flaw failure, calculate tube failure and for element on the calculations.

We also provided some scoping failure calculations and provided some fission product releases. And again, as I mentioned before, system codes can't calculate this so we have to have to use a CFD.

And the test data are in small scales so you can't use those directly either. And if you just scale those up with hand calculations you may not be

confident that you're getting the proper behavior.

So the deck was generated by Sandia National Labs. And it was based on previous RELAP/SCDAP and MELCOR combustion engineering decks. And model approach was derived from that used for the steam general action plan.

NRC calculated the CFD behavior and used a similar approach as for the previous NUREG-1922 work that was done for the Westinghouse plant under the steam general action plan. And this is the reference for the CE calculations.

DR. SHACK: Since that's referred to in public comments and response --

MEMBER STETKAR: Bill, turn your mic on. This is something that you'll have to learn.

DR. SHACK: It would be nice if that was in a more accessible document. You know, you give somebody this reference and it doesn't do them much good.

It's sort of like an, well, I won't make a pejorative statement, but you don't know whether it really supports the argument or not just because you can't get a hold of it. So I would hope it at least would appear as the nuclear engineering design paper, or something like that, if you're not going to make a

NUREG out of it.

The other thing. In the NUREG itself, it would be nice if you hauled Figure 9, the histogram of tube entrance temperatures, out of that paper and put it in the NUREG. Since that's a rather important result that isn't in the paper now.

DR. SALAY: Okay. And the integration method for CFD and system codes used the general method that was applied decades maybe, in 15, maybe 20 years, for a combined CFD and system codes.

And these methods are documented in NUREG-1922 and NUREG/CR-6995. And ECFD methods were validated against Westinghouse 1/7th scale test.

CFR provided target flow parameters for system code and spatial temperature distribution, normalized spatial temperature distribution in tubes.

System code MELCOR was modeled to match the CFD code parameters, provide the overall transient behavior of history and the time-evolution of the spatial temperature distribution.

And we looked at short term station blackouts and long term station blackouts. And what effects that is the timing of auxiliary feedwater failure.

And ultimately, we found that the behavior

was pretty similar, but time shifted. And there is a slower temperature rise. Presumably because the lower decay heat.

We also looked at the different secondary relief valve opening conditions. Immediate, either by operator action or some failure. We looked at a few different failure models.

There were, for each case there was two sets of calculations. A scoping calculation, which calculated failure, and then a calculation where failure, for all components, was disabled so it provided thermal hydraulic histories to be used with finite-element codes. To look at the failure, to analyze failure with finite-element codes and with the C-SGTR calculator.

And I'll go over also a few of the comments and our responses. Both from ACRS and public.

Impact of changing codes, RCP seal leakage, loop seal clearing, uncertainties in thermal hydraulic analyses and a few others.

Both ACRS and the public provided comments on impact of changing codes. Because not only, when going from the steam general action plan to this work, we not only switched plants but also switched codes.

So at the beginning of the work we had to

decide which code should we use. And there were at least one meeting, maybe two, where a bunch of us got together and decided which code should we use.

Should we use RELAP/SCDAP, which was used for the previous steam general action plan work. And so you can directly compare the results. Same code, just different plant. And the hot tube modeling had already been worked out there.

But then again, MELCOR is the main code, severe action code that the NRC uses. And it also calculates the fission product release if you want to look at consequences.

There had been comparisons back in, I think 2004, comparing MELCOR, MAAP and RELAP for this scenario. And then they got similar results. Ultimately we decided on MELCOR.

Dispute that, changing codes was a significant concern. Again, because we were simultaneously changing both the plant and the code.

So the deck developed in the process involved the comparison between MELCOR and the RELAP CE deck to compare the results. And again, as with the 2004 work, similar sequence and timing were obtained for these two analyses. And this is documented in Chapter 4 of the deck development report.

CHAIR REMPE: Um --

MEMBER MARCH-LEUBA: Just out of -- oh, go ahead.

CHAIR REMPE: Oh, you can go first. I'll do last.

MEMBER MARCH-LEUBA: Just out of curiosity, what year was this? I mean, I'm thinking why didn't you consider TRACE?

Why did you not consider TRACE instead of RELAP?

DR. SALAY: TRACE --

MEMBER MARCH-LEUBA: TRACE being the NRC code.

DR. SALAY: Because you're looking at the degradation of the fuel --

MEMBER MARCH-LEUBA: Oh, this is for the full transient?

DR. SALAY: Yes. All the SCDAP/RELAP --

MEMBER MARCH-LEUBA: Oh, okay.

DR. SALAY: -- it's the severe accident part also.

CHAIR REMPE: Bill?

DR. SHACK: Just another comment again. This document doesn't seem to be in ADAMS, so when you refer to the public, to this document for

documentation, they can't get their hands on it.

If it was in the comparison itself, you do the calculation with SCDAP/RELAP, with a stress multiplier of 1, and you do the calculation in MELCOR, with a stress multiplier of 2, which seems like a strange choice if I'm doing a comparison of the two codes to show that they give similar results. Is there a reason that you picked the flawed tube for the MELCOR and an unflawed tube for SCDAP?

DR. SALAY: I thought we used flawed for both. I thought --

DR. SHACK: It says MP, I only know what I read.

DR. SALAY: Yes.

DR. SHACK: It says MP=1 and the 1 and it says MP=2 in the other.

DR. SALAY: Because I know --

DR. SHACK: And then similar, you know, it's a matter of --

DR. SALAY: There is a judgement --

DR. SHACK: There's a judgment as to how similar.

DR. SALAY: Yes.

DR. SHACK: But the different stress multipliers seem to me very bizarre.

DR. SALAY: Yes, the same one should have been used.

CHAIR REMPE: So could you talk a little bit more about what was done with those comparisons?

When I looked at the Sandia report it looked like that you had a special version that you had used to try and do those comparisons, plus it was done with, I forgotten, 1.8.3 or 1.4, which is many years ago.

DR. SALAY: Yes, it was the version that was current at the time. Yes.

CHAIR REMPE: Yes. Okay, nowadays I know that with the SOARCA analyses, between the time they did the Surry analysis for the initial SOARCA report and the uncertainly analysis, they had updated the model for the steam generator, somehow or rather, and changed the nodalization enough that it really extended the time for the sequence to occur. I don't think it changed the temperatures, but it did effect the timing.

And so what version of MELCOR and how can we have confidence with the version that was used, and somehow with this special version that matched SCDAP, give us any insights with respect to truth I guess.

DR. SALAY: Special, and I think it was just a, I don't think they made a special version for

--

CHAIR REMPE: Okay, again, I just know what I read, and maybe I misread. But again, they basically had done some things to make sure they could do a comparison and have similar results. And --

DR. SALAY: I don't think we had the resources for this project to tell the code developers to give us a new code.

CHAIR REMPE: Yes. Okay, so you're saying you just the 1.8.3 right off, I mean --

DR. SALAY: Whatever the version --

CHAIR REMPE: -- yes, version was.

DR. SALAY: -- current to the time. Yes.

CHAIR REMPE: And you did not do any special, I mean, a lot of times, when we do the cross walk for example, with MAAP and MELCOR, there's been some changes in inputs, perhaps the stress multiplier or whatever, but to try and make sure the codes would give similar results. And nothing like, because what I was reading sure sounded similar to that.

DR. SALAY: We didn't have the code developers involved and --

CHAIR REMPE: Yes.

DR. SALAY: -- so we used --

CHAIR REMPE: But the input data.

DR. SALAY: Yes, the input data.

CHAIR REMPE: Yes.

DR. SALAY: Yes, we modified. You had to modify.

CHAIR REMPE: Nodalizations and things like that, to try and give you similar results.

DR. SALAY: They didn't adjust the nodalization to try and --

CHAIR REMPE: Okay.

DR. SALAY: -- match results. They used the nodalization that they did use previously in similar analyses.

CHAIR REMPE: Okay. And then what was the effect of what happened with SOARCA where they went to a different model for the steam generator and they extended things, do you feel comfortable that the results you have are still appropriate?

DR. SALAY: Actually, the way they modeled the steam generator tube did come from SOARCA.

CHAIR REMPE: How could that be with the timings? I mean, the first SOARCA was done about, I would have thought the time you did this and the uncertainty for SOARCA I thought has happened in the last --

DR. SALAY: Uncertainty. I mean, people

work on different decks and ultimately, although they took the geometry from one, the generally approach was --

CHAIR REMPE: The second, the uncertainty SOARCA MELCOR model?

DR. SALAY: I don't think they took the uncertainty SOARCA.

CHAIR REMPE: So you have the older one that would have given you the timing --

DR. SALAY: Yes.

CHAIR REMPE: -- that was later deemed to be less accurate because you went to this improved steam generator model for the uncertainty analysis.

So, again, and timing is important here with respect to failure and things like that. I'm wondering about confidence in results here. If that's a question.

DR. SALAY: Yes. I mean, you can always improve and you always change and you're always going to get a different result if you change stuff.

CHAIR REMPE: Sure. Is there any insights that you might want to say about the results based on the changes that were observed with the SOARCA improved model --

DR. SALAY: Now, you would have to run it

with both.

CHAIR REMPE: Yes.

DR. SALAY: But ultimately what we are looking at is the relative failure timing. And if your results and the heat you're adding is similar and if your results are matching what's coming out of the CFD, you're transporting the heat at the same rate.

CHAIR REMPE: I guess what I'm wondering, again --

DR. SALAY: You should be getting the same results if you do the same, if you're generating the same amount of heat and transferring the same amount of heat.

CHAIR REMPE: Okay. So with the improved SOARCA analysis, for the long term and short term station blackouts, I thought that they saw the timing for peak temperatures and the reactor and the containment to be changed considerably. And I forgot now whether it went out longer or earlier.

But what I'm wondering is if that's going to affect the results that you have now knowing, I mean, is there certain perspective that should be added to these results? Have you considered how more recent changes might affect the results for this analysis?

DR. SALAY: I didn't really look at it.

And sort of worked on what we had an --

CHAIR REMPE: Sure. Is that worth looking at?

DR. SALAY: If you look at how much effort it would take relative to, I mean, because it's completely different to do a MELCOR 3 input. And they were asked one of the reasons why, I mean, once we did choose MELCOR we had to decide, should we use 186 or should we use 3 --

The earlier approach, it was easier to work with the current method than switch to something new. And we were resource limited.

CHAIR REMPE: I understand that. But it just seems like if we, there's no general insights. Improvements to MELCOR would tend to have X, Y and Z effects on the results that we did back with this old version. And it just seems like something that might be a worthwhile insight to think about.

DR. SALAY: We could look at what the differences were and --

CHAIR REMPE: Yes.

DR. SALAY: -- maybe estimate what the differences could be.

CHAIR REMPE: Again, maybe there is nothing that would change significantly, but to just

say, well yes, it's a new version and we have no idea what the old version did doesn't bode well with the reports issued this year.

DR. SALAY: The later versions of 186 and the earlier versions of 3 or 2, markup 2, whichever one --

CHAIR REMPE: Yes.

DR. SALAY: -- I mean ultimately were similar but with different inputs. I mean input structure.

CHAIR REMPE: Right.

DR. SALAY: So the models, under the hood, was a lot of it was the same. And I'd have to look at the specific changes for the --

CHAIR REMPE: Yes. It might be good to take a look at that and just think about, is there anything that might have impacted the results that we're releasing this year, for changes that were made to the code several years ago.

DR. SALAY: And we had a public comment on reactor coolant pump seal leakage for the CE analysis. We used 21 gpm to be consistent with the steam general action plan.

It was pointed out that this may not be correct, you might have a substantially less leakage

for CE. In the comment, they included a calculation which showed a less of a pressure drop. And I'll show a figure.

And so we ran the base case disabling reactor coolant pump seal leakage. We got a similar pressure drop to what the public comment calculation had.

And in there was a delay in the absolute, but not the relative failure timing. And with most TH issues it was time shift, but not qualitative difference.

So here you see system pressures and system temperatures. We're focusing on these two curves. The red one is the original primary system pressure.

And after station blackout, but when your steam generator is still removing heat, the pressure dipped considerably. And you'll see the temperatures rising over here.

It appears the steam generator peak temperatures, structure temperatures base indicates the original calculation. And mod indicates the modified calculations. Purple.

So when we disabled the, we turned the leakage off for the pump seals, the pressure dip was nowhere near as significant. And it just dive-balled.

Seamed to be quite close to what the comment calculation was. And so this was a good catch.

Yes, so now we have results that show kind of bound leakage, seal leakage, between zero and 21. And the amount it impacted, it's failure was a few minutes. And this is where tube fails and hot leg fails.

Loop seal clearing, I mentioned this, several comments received on loop seal clearing. This was looked at in detail for Westinghouse, but apparently, there's other information indicating that perhaps it would clear.

The initial scoping, we did some initial scoping work for CE that built upon the previous steam general action plan work. But this is one of the issues that we cut back on.

Again, because even though it's important for Westinghouse, because you get much hotter gases if the loop seal clear for Westinghouse plants, and it's the geometry that we looked at, you're already as hot as the gas that's entering the hot leg. The temperatures the tube sees are already as hot as the hot leg sees under regular counter-current natural circ, close loop seal natural circulation conditions. So loop seal clearing is not as important.

There's a question about relative decay and oxidation powers. And there was a comment, I think the question originated because the hydrogen showed up in, I think in, I don't know if it was either presentation or report, there was a plot of hydrogen content in the steam generators and the commenter seemed to consider that perhaps the hydrogen was generated earlier and then held up and then transported to the steam generators. And asked, where was it held up.

So as you see here in both the power generation, up top, where green is oxidation power and the hydrogen generation that the hydrogen is actually generated when you see it. And quite later than you see in the Westinghouse analyses.

And then there was a question about oxidation of steel in the RCS. It's something that's typically not modeled in severe accident analysis codes. And MELCOR does model it, steel oxidation in the core.

Our analysis didn't consider it, but since the question arose we went and looked at the reaction rates. And it doesn't seem to be a major effect.

CHAIR REMPE: Just out of curiosity. Even in the core reaching, and this is a curiosity

question, but when it oxidizes, it's near its melting temperature, as you pointed out. And if it happens to be at the right angle, it would immediately flow off exposing more material to oxidation.

It's not a protective layer. And does the evaluation in the core consider that?

DR. SALAY: I think it's just --

CHAIR REMPE: It's a reaction rate that's --

DR. SALAY: Yes. Yes.

CHAIR REMPE: -- acceptably is assuming that it's protected after the oxide forms, right?

DR. SALAY: Yes, I believe so.

CHAIR REMPE: So in real life, again, if it's pointing in the right direction, that's not true because it would continue to expose more surface that would oxidize and I wonder. Again, I appreciate you looking at it and I understand it's not normally considered, but I just thought it was different than what I've seen in real life.

But I think Dana's point about there's hydrogen present in the system that would affect things is something that I have not seen. So anyway, thanks for looking at it.

DR. SALAY: Great. And another question

that was raised in the previous meetings was about uncertainty due to variation in thermal hydraulics.

This was also a concern of the uncertainty that you get from TH behavior. Was a concern upon initial deck generation because the structural parts, they were looking at uncertainty, but they couldn't incorporate the uncertainty from TH.

So when the deck was being developed, they performed the same uniform that uncertainty analysis on a early station, a short term station blackout model. And they sampled parameters and observed the effect on the absolute component failure timing and relative steam generator tube to other RCS component failure timing.

And they looked at a NUREG/CR-6285 and NUREG/CR-6995 in deciding what parameters to consider. And the ones they ultimately considered were discharge coefficients for the primary relief valves, Zirconium oxidation sensitivity coefficients, the mixing parameters, input from the CFD results, steam generator tube, the heat transfers coefficient multipliers, the emissivities for heat transfer and heat transfer from RCS to containment.

And they came up with empirical distributions that had standard deviations of 427

minutes for relative steam generator tube to RCS component failure timing and 10 minutes for absolute timing. And this is also documented in the report.

And yes, I know that's not available in --

CHAIR REMPE: Is it going to be?

DR. SALAY: If someone tells me to work on it or start a contract to work on, yes. Otherwise no.

DR. SHACK: All you have to do is give it an ADAMS number.

CHAIR REMPE: I mean, don't you have a requirement in your tech publications that anything that's referenced has to be available or is that not a requirement?

It's true for journal articles, but --

MEMBER MARCH-LEUBA: It's not only that. Every contractor that provides a deliverable goes into ADAMS. With an ML number. I know all my old technical evaluation reports have an ML number.

It's an internal relevance, but. I think it's a contractual obligation.

DR. SHACK: I mean, if it isn't in there it somehow indicates a lack of confidence in their work.

MEMBER MARCH-LEUBA: No. I mean, you put there, in internal ADAMS you can put a CD with cross sections. It doesn't need to be a report.

DR. SALAY: Yes, I'll verify whether it's in ADAMS. It was a few years ago.

So yes. And so you can look at the spatial temperature distribution in more detail. And because you have a flaw distribution, then if you look at what fraction of tubes you sort of, at the time when I had to provide temperature distributions, we ended up having to provide like a cold average and a hot within the plume. But I think we can do it relatively straightforward.

Temperature distribution gives you based on the inlet. Temperature distribution from CFD. You can come up with a whole surface area temperature distribution. So then you can more precisely MAAP the flaws to that.

And you can look at loop seal clearing. And another issue is water holdup in the steam generator.

Flooding and counter-current flow is being studied, so this should be something we should be able to check. And water has been held up in previous steam generator action plan calculations.

And then Three Mile Island did have a bubble with water in the steam generator, so there was some concern that it may have been non-physical. And

this was something that was looked at during NUREG/CR-6995, the steam generator action plan.

And once that's done you also look at fission product release. During re-flood the calculations get a little unstable so sometimes they crash. So if that happened, we can go back and rerun and try to work to get it to work through that. So that's something that could be done.

And so to conclude, we did some calculations for CE plant for the replacement steam generators. And this provides input to the CFD calculator and finite-element component failure analysis.

Most effects are a result in shift, time shifting of temperature increased curves. And the relative temperature increase rates and relative, primarily the relative failure timing is more important to how much gets released then the absolutely failure time. Although for the absolute of course gives you evacuation time.

And some work was deferred because of limited resources and benefit was not determined to be worth the expense for the project. And received a lot of useful feedback from the ACRS and the public.

CHAIR REMPE: So thank you. Are there any

questions from --

DR. SHACK: Make sure I'm not misinterpreting anything. One of the things you get from this is the hottest tubes are much hotter, in a CE generator, and there's a lot more hot, hot tubes. Is that a correct interpretation?

DR. SALAY: Yes.

DR. SHACK: Yes, okay.

MEMBER STETKAR: How much would that differ if I had a Westinghouse plant that replaced their steam generators with a shallow inlet plenum?

Because you keep characterizing this as a CE steam generator and I think of steam generators as steam generators. Granted the bypass flow is different.

DR. SALAY: There's the two major effects. You have the shorter hot leg for CE, which provides less opportunity for mixing, and then also the amount, the fact that there's a longer distance to the tube sheet, which allows --

MEMBER STETKAR: But if I --

DR. SALAY: So you're somewhere in between --

MEMBER STETKAR: I'm sort of fixed on my hot leg length. But if I change somehow my steam

generator geometry with a replacement steam generator for a Westinghouse plant --

DR. SALAY: I think, well, I mean there's less opportunity -- well, and also the third effect is that it's coming in at an angle. The Westinghouse one, if it came in normal to the plate it might be more symmetrical, waiver less.

So one would expect it if you put this type of geometry steam generator you get less mixing. It sort of seems obvious in that you get somewhere between the Westinghouse, the current Westinghouse and the current CE analysis.

DR. SHACK: I mean, one of the things that seems a little discouraging is in order, you know, because these are not representative, they're examples, there seems to be no shortcut to deciding whether you have a problem or not. You have to do the CE calculations for that particular geometry.

Unless you're willing to live, perhaps with some guidelines as to those distances. But is there anything, I mean, are you envisioning people having to CFD calculations for steam generators?

DR. SALAY: I think it would be better to have a guideline based on distance.

CHAIR REMPE: So do you think the

guidelines will have specific distance or I should wait until the end of the discussion? Because I didn't see --

DR. SHACK: That would seem like a simple solution, but do you think it's sufficient, I guess is the question.

MEMBER STETKAR: So if I had a plant in a different country that was neither steam, was neither Combustion Engineering or Westinghouse and has a particular geometry of its hot leg and its loop seal and its u-tube steam generator, how would I know, if I'm in that other country, whether or not I need to invest in a lot of CFD analysis?

MR. COYNE: Well I can answer that. That would be up to the regulatory authority and --

(Laughter)

MEMBER STETKAR: No, because my regulatory authority hasn't necessarily thought about this yet. But I'm interested in safety of my plant.

DR. SALAY: As a practical --

MEMBER STETKAR: And I'm being serious here --

DR. SALAY: Yes. No.

MEMBER STETKAR: -- I'm not being --

DR. SALAY: As a practical, I mean, the

fact that it comes at, in the CE plant, that it comes in, the plume comes in normal and its short, that gives you the worst short of situation.

MEMBER STETKAR: But that comes back to the sort of, I think where Joy was heading, if there's some general guidelines based on everything that you know about the couple examples that you've looked at, that you could provide to somebody in a more generic sense.

DR. SALAY: I think you could provide in that. And the CE plant sort of does provide the worst case.

MEMBER STETKAR: The geometry that you looked at provides the worst case. It happened to be associated with a CE plant. I'm trying to more generalize that if I have u-tube steam generators and a pressurized water reactor, what elements of the configuration provide cause for concern?

DR. SALAY: It's --

MEMBER STETKAR: And if I'm planning on fixing to replace my steam generators, what am I going to be sensitive to when I make that decision, for example.

DR. SALAY: There's the inlet distance to the tube sheet. And whether it's normal --

DR. SHACK: I mean, I've got a lot of room between 1.5 and 4.5 though.

CHAIR REMPE: So is 2 good? You only did two examples, but do you have any feel where you could say some things a good number? Greater than is --

DR. SALAY: Greater than 4.5 is good.

(Laughter)

DR. SALAY: And also, it's relative also to the plume calculated in CFD too. Because it's length of diameter of the hot plume, which you don't know unless you've done the CFDs.

CHAIR REMPE: Okay --

DR. SALAY: But you could probably have it correlated. Chris Boyd has another comment.

MR. BOYD: This is Chris Boyd again. I'll chime in. This is a severe accident, there is obviously a lot of uncertainty.

And all of this years of thermal hydraulic research, you know, end up in one decision point on this event tree where between zero and one were bounded there. And there's all sorts of other things.

So I think before you put too much emphasis on this, you need to look at the whole picture.

MEMBER STETKAR: But, Chris, don't, I have the perfect PRA. I thought about every possible

scenario where I can get a high-dry-low. Even ones that you haven't even thought about yet, okay.

I'm still looking for how carefully do I need to look for those scenarios. Because if I'm not particularly vulnerable to this condition, because of fundamental features of my plant design and configuration, maybe I don't need to look so hard in my perfect PRA for those scenarios. The ones that you haven't even thought about yet.

And that's a little bit of the direction I'm heading in. Okay. I don't particularly need to spend a lot of attention on seismic events, for example, if I'm in the middle of a swamp.

MR. BOYD: Right.

MEMBER STETKAR: Out in a place that's never had an earthquake before. And in terms of understanding where I need to focus my attention in terms of risk and safety, some general guidance might help.

Not prescriptive guidance, because that's dangerous. But what do I need to look for, as I go out and search for these scenarios, in my perfect risk model?

And start differentiating if I can get it, start differentiating about, do I get a release out

directly to the atmosphere or is it more likely, under some scenarios, to go through the condenser, the main condenser, then the turbine building.

And maybe I don't even care, if I'm not vulnerable to this, if I can have some sort of reasonable confidence.

CHAIR REMPE: Before we let you, if you're done?

MEMBER STETKAR: I'm done.

CHAIR REMPE: Before we let you get off the hook here, Dr. Powers has joined us again. And, Dr. Powers, you've missed some key slides that you've expressed concern about, to me privately, about the loop seal clearing and his comment there about the benefit of work that was differed isn't worth the expense of the project. Do you have any comments you want to clear your mind about at this time?

MEMBER POWERS: You certainly made the case, but you're not terrible concerned about the loop seal clearing in the CE design because of the limited mixing in the lower plenum. Yet kind of the same result.

Whether or not, you get a little more heat flex onto the tubes if you have a clear loop seal, but you get, the point is, you get high temperature gases

there without the mixing. So people have to look at the depth of that lower plenum and the angle of the input. And that's pretty clear from the results you got from your CFD analyses and whatnot.

I think in the documentation of the work, given that you really can't do any more work, but you can't document what you've done apparently, you need to make clear the rather indelicate situation we have with the Westinghouse designs, which make up a big fraction of the plants, that loop seal clearing is, there are ways to get to it and whatnot, and that we don't have a clear, need a much clearer sharper discussion on that.

Otherwise I understand kind of where you're coming from on a loop seal clearing and whatnot. I still don't understand where I'm going to vent.

MEMBER STETKAR: You don't have to, there's a bunch of different places you can. Depending on the scenario.

(Laughter)

MEMBER POWERS: Yes, but people keep asking me what the DF is on the downstream flow pathway and --

MEMBER STETKAR: Anywhere from zero to non-zero.

MEMBER POWERS: No. DFs typically can't go below one. Okay?

MEMBER STETKAR: I said zero to one.

MEMBER POWERS: Oh, okay.

(Laughter)

MEMBER STETKAR: Sorry, never mind. Anywhere from one to more than one.

CHAIR REMPE: So with that, does anyone else have a comment or a concern they want to bring up at this time? We're ahead of schedule, but let's take a break at this time and come back at, jeepers, how about ten till 3:00.

(Whereupon, the above-entitled matter went off the record at 2:36 p.m. and resumed at 2:50 p.m.)

CHAIR REMPE: Okay, let's resume our meeting here. Raj, are you up next?

DR. IYENGAR: Yes. Let me find my slides.

CHAIR REMPE: If it helps, I think we're in Slide 41.

DR. IYENGAR: Okay. Good afternoon again. Thank you. I want to give you a little bit of background on the failure analysis that we did using finite-element method for hot leg pipe in Westinghouse design.

So the purpose was to do a more refined finite-element analysis using the severe accident scenario for Westinghouse that is also done in the simple calculator version that Ali and Selim will be presenting.

And to evaluate the adequacy of the simplified calculator predictions based on more refined finite-element analysis. I'll go through the input of the use.

So our purpose is very small. My work in my presentation is sandwiched between two exhausted presentations. It's very minor in scope.

Primarily in the calculator that Ali will be presented. Ali and Selim use certain equations for predicting the failure of the hot leg and surge line and compare that against the Steam Generator 2 failure to see, in the race between these two, which one will fail first.

So the equations that he uses in his calculator are based from an EPRI report. It's based on a simplified calculation of a cylinder and average temperature through the cross section is used.

And it also uses the so-called Larsen-Miller Parameter. Which is the equivalence of time at temperature for steel to fail on a offsite

structure.

MEMBER POWERS: The original Larsen-Miller Parameter was derived for relatively low temperatures in steel and whatnot. Is here, when you say the Larsen-Miller Parameter, is that just the name that's used but the database under the basis for the parameter, for selecting the parameter values, based on something other than the original database?

DR. IYENGAR: Yes. So we have expanded the database to higher temperatures. I'll just go through that, and I'll have Bill Shack and since then we've done a little bit more.

Yes, the Larsen-Miller Parameter is particularly convenient to use for predicting higher temperatures and the creep condition where the material behaves in a time dependent fashion.

So, we get this data from a database of lots of tests, numerous tests. And in the second equation, for Westinghouse hot leg, you can see the parameter is based on also the effective stress that the pipe would experience.

And once we determined this, and then you can get the time to rupture using this equation. That's what is used in the calculator.

Now the question -- yes?

MEMBER POWERS: I mean the concept behind it is an accumulation of damage to the pipe.

DR. IYENGAR: That's right.

MEMBER POWERS: And I see. So it's just this parametric value changes as a function of the stress?

DR. IYENGAR: Right.

MEMBER POWERS: Okay.

DR. IYENGAR: So you go step-by-step at each given time. You know, this is the temperature, this is the stress, so much is damage and then calculated the damage.

So we, as I mentioned, it's a very simple scope that we have. We wanted to compare the sanity and the accuracy of the calculation using the simplified EPRI equations. Which is using a calculator for the Westinghouse design.

Primarily because the calculator predicts that the, in the case of Westinghouse design, the hot leg could fail sooner than the steam generator tube. So that's of particular importance to do a, kind of an independent check using a more rigorous finite-element analysis.

So I went through all this last time. What we did was we used complete system level model for

Westinghouse design using three-dimensional shell elements.

As you can see, there's a cartoon here which includes a hot leg as a surge line. And the material behavior is assumed to be time dependent.

And also, we take into account instantaneous rate independent plastic strain in calculating the total strength.

So this is more realistic. We have the creep, which is, you know, I'll show you law later in the plasticity, which is the rate independent one.

Now, we also said, well, what if I turn off the creep, the time dependent behavior completely? What happens?

In that case, the material behaves in a rate independent fashion, piecewise-linear, and then it's instantaneous plastic in response. So really there's no stress increase for a given temperature with respect to time.

So that would actually be a case if I used that and calculate that using the stress, calculate the stresses and then use Larsen-Miller Parameter time to rupture. The time to rupture would be longer than if I used both the creep and the plasticity. Obviously.

So that we used to kind of give it an upper

bound off time to failure for the model. Even though it's not realistic.

So just to give us a little bit of a, we didn't do accompanied sensitivity analysis, but we did some hypothetical scenarios, what if.

So in this project, in this work, what we did was we did the, use material data, which is actually documented in Appendix A, which Argonne National Lab, Dr. Saurin Majumdar had done lots of experiments to extend the temperature range of applicability in the data. So it goes up to a 1,000 degrees C, which I think is --

MEMBER POWERS: Yes, that's the step that's really crucial.

DR. IYENGAR: Yes. Right.

MEMBER POWERS: That's very good.

DR. IYENGAR: So we did that. And Dr. Saurin Majumdar is online too if you need any questions specifically addressed to him, he would be willing to respond.

MEMBER POWERS: There's a lot of applications of, early applications, when Larsen-Miller first adopted, the database didn't get anywhere near that 1,000 degrees.

DR. IYENGAR: Yes. Yes.

MEMBER POWERS: That's very good.

DR. IYENGAR: Yes. You know, it's natural because these are not designed to operate at different temperatures, why would you want to expand a lot of, the vendors wouldn't want to do the data if that's not really needed.

MEMBER POWERS: Yes. I mean, that was the problem we always had.

DR. IYENGAR: Right.

MEMBER POWERS: But the general concept of accumulation of damage seems like a very worthwhile thing to pursue.

DR. IYENGAR: Yes.

MEMBER POWERS: Yes, it's pretty nice.

DR. IYENGAR: So in these calculations we used the structural temperatures as initial bond conditions, steady-state conditions. All the input data, as far as the temperature, temporal radiation of temperature and the heat transfer are all obtained from the system, the RELAP code that preceded in the previous case with Chris Boyd and Company had done.

We used that. We also used the time-dependent heat transfer equation as mentioned. We used the upper temperature split. And the circulation that Mike Salay talked about.

So these are all things that are not in the, used in the simplified calculation that Ali will be presenting.

And we also adjusted the heat transfer coefficient spatially because Chris Boyd was adamant about that. Because that's very significant and it's based on some of the work that he and others had done and documented in NUREG-1929. I mean 22, sorry.

And we also modeled the heat loss to the ambience due to the convection and radiation. Which could be significant, right? That could change the time to rupture.

So we did all of these things and ran a thermal-mechanic simulation for short-term SBO.

And here you see the system code at roughly 12,300 seconds. You see the display where you can see that the region where you would see maximum accumulation is in the hot leg away from the nozzle. That's very important. That's a location that you would normally not anticipate.

And then using the Larsen-Miller Parameter, as you can see these equations, and we calculate the failure. And the failure is average to the thickness to determine the failure time.

And so you can determined the failure time

to be 12,300 seconds here. And the failure, it's mapped. The contour is mapped here. The red is at level, I mean, damage is one. This is where the failure will happen.

Now, the system level model, it was very exhausting. It gave us a lot of complications to run than more computationally intensive.

We also wanted to see, what if I put up weld overall and make the thickness of the hot leg longer, because of some mitigation against PWACC, how would that affect the failure time?

So those kind of calculations are kind of difficult to run with these huge system level models. So what we did was we took this region of the hot leg there, hot leg and the nozzle region that you see here, and we modeled that regional alone. It's a sub-model that we used.

And we ran it with the same parameters as the model. And we got about, the results were very similar. Within ten, 15, 20 seconds of failure. So we had some confidence.

And then we did, we used the pipe model to look at what would happen if I applied a weld overlay on top of the hot leg pipe due to mitigation, how would that decrease, I mean increase the failure time.

And you can see here, in this scenario, that the failure time increased only about 70 seconds or so with a weld overlay. The weld overlay is about five to six.

So didn't have a whole lot of effect. But the weld overlay is right here, not here. You know, the nozzle that you would half expect a lot of PWACC cracking.

So I have here, summarized, the various assumptions that I used for calculating the failure time. And what I want to draw your attention, particularly, is to the second one.

That we have a total realistic material behavior of time dependent plus the instantaneous response plasticity. And the weld overlay, there's no weld overlay. The failure time predicted is 12,430 seconds.

And then when I turn off the heat transfer coefficient, I also had a calculation that I've been using, it's there in the report, where you have only plasticity. That only increased over 12,600 second's failure time. Which is not really realistic.

The key point is, the failures times predicated weigh these assumptions here compared very with the 5th percentile failure time estimated by the

calculator, which was 12,800 seconds. The mean failure time, estimated by the calculator, is 12,600 seconds.

We had both, we used the 5th percentile because the last time one of the members thought that is something we need to also highlight. So that's what I have in terms of summary.

So in summary, the hot leg model that we used compared well with the system level model. And we calculated the failure time and it determined to be kind of lower than what the calculator predicts. And so the weld overlay has a very small influence.

The failure time is mainly influenced by, one, the stress redistribution due to the counter-current circulation. And also because of the true thickness variation of temperature.

So that's all I have in terms of the background. If you have any questions, I can wait for that, and then go on to the few comments we received from ACRS members and the public. And our responses.

CHAIR REMPE: I guess go ahead. I don't hear anyone.

DR. IYENGAR: Okay, thank you. So one of the --

MEMBER KIRCHNER: Quick question.

DR. IYENGAR: Yes.

MEMBER KIRCHNER: So I'm thinking of the previous presentation. So what's the impact of these temperatures on other components? Like the pump seals.

DR. IYENGAR: So some of the, I mean I don't know the pump seal, but we did a calculation, I think prior to that Saurin Majumdar did some calculations, which is also documented in this Chapter 4.

All of them take a longer time to fail. Only the hot leg, interesting our design was, a closer or a prior before the steam generator fails.

MEMBER KIRCHNER: Would it change the, over time, the leakage rate out of the seals?

DR. IYENGAR: I don't know, but I don't think we did that calculation. Chris, you remember anything?

MR. BOYD: This is Chris Boyd again from research. I don't know if Mike had the chance to run all those sensitivity studies. In the old NUREG, from a few years ago with Westinghouse, there are some information out there. That's a tricky subject, how those seals operate.

But we ran a whole battery of sensitivity

studies on different leakage rates, starting at different times, based on what information we could gather from experts.

MEMBER STETKAR: Some of this stuff is going to be confirmed because a heck of a lot of the plants are replacing their old seals that had that old temperature model with them as far as time to temperature for seal failure with completely new seal designs.

And I don't think anybody has, I mean, they've looked at the new seal developed models for the new seal designs under the conditions that people have looked at in the past. Station blackout for example.

But not for this kind of issue where you have really evaluated temperatures.

MR. BOYD: Temperatures.

MEMBER STETKAR: Really elevated temperatures.

MEMBER KIRCHNER: I'm just curious because the parametric case that was shared earlier assumed no pump leakage. Seal leakage. And of course, the system pressure stayed up.

But I was thinking that the graph, if you had higher leakage than what assumed as the base case, then the system pressure could drop substantially. Is

that possible?

DR. IYENGAR: I can't answer. Chris, do you want to answer that?

MEMBER KIRCHNER: Which would change the projectory.

MR. BOYD: I would just say that we did look at that, and you can drop the system pressure when the pump seals leak. I see those as just different paths on the event tree.

MEMBER STETKAR: Thank you.

MR. BOYD: And we have to study --

MEMBER KIRCHNER: Yes, I understand that.

MR. BOYD: And he's looking here at the high pressure.

MEMBER STETKAR: The old Westinghouse seal model got you anywhere from a minimum of 21 gpm per pump. That was basically good intact seals up to 480 gpm per pump under the worst possible model for the seal behavior.

The new seals they claim don't leak. I mean --

MEMBER KIRCHNER: Don't leak.

MEMBER STETKAR: Well, that's what they -- but indeed they do model some residual leakage. But it's down in the few gpm. I haven't read that report

yet. We'll have a briefing on it in February.

MEMBER BLEY: It's coming up. But there have been reports that out in the field they haven't worked quite as well --

MEMBER STETKAR: This is with Rev 3, I think, of the seal design.

MEMBER KIRCHNER: I just, to follow-up, I'm just curious how the trajectory of the scenario plays out as this system pressure drops significantly?

CHAIR REMPE: Doesn't that help with the concern about the Generator 2 --

MEMBER STETKAR: Really, really bad reactor coolant pump seals make this much less of a concern.

CHAIR REMPE: Yes.

MEMBER STETKAR: It makes different scenarios more of a concern.

CHAIR REMPE: Right.

MEMBER STETKAR: But that's risk assessment.

MEMBER KIRCHNER: Okay.

MEMBER STETKAR: I mean, that's really, really good reactor coolant pump seals make this more interesting.

MEMBER POWERS: What you're telling me is

risk assessment is bad news. It just depends on where the bad news is coming from.

MEMBER STETKAR: That's why my personality fits it very well.

MEMBER POWERS: That's right.

MEMBER STETKAR: Thank you. Let's continue here.

(Laughter)

DR. IYENGAR: So one of the things I forgot to mention in this many of one scenarios we studied is that you see that the difference in failure time is, it's not very significantly differently, 12,430, which is 560, that's because what happens is things are going very slow. And when the temperature rises fast it rises so fast.

You know, we can use even extensive data. You know would not by so much of time anyway. Because things are happening so fast in such a short time. I wanted to drive that across.

So in the last briefing Dr. Ballinger had a very good remark. I think I had, when I presented these results I had not used these rounded off digits, I used the time to failure as it was predicted by the analysis. In which case you could, for example, I had 12,302 seconds and there was an issue with that. I

understand.

But I think, I just wanted indifference, I did want to mention that we are doing these numerical calculations and they take thousands of time steps. And these time steps are a fraction of seconds or even thousands of seconds.

So when I do a failure time calculation, it's actually encompassing thousand times, I mean hundred time steps say. So for me to average out for one, an average ground off in a different way, the comparison becomes a little bit awkward.

But nevertheless, we understood why he was saying, so we had rounded off more carefully in the final report.

And then there were a few questions from the public. Mainly they wanted to get the references for the ANL test that in Section 5 we refereed to. And we have provided that and included that in the revised draft.

And they also recommended reference literature. And we wanted to mention that I think one point may not have come across well is that in Appendix A we have actually expanded the database for the first time, I think, internationally. In terms of the high temperature data. Creep rupture data for these

materials.

So that probably wasn't coming across well. So we have put in the verbiage in Section 5 to address that.

Now, there was another interesting question. So in our model assumptions, as well as in the calculator, we used 95 percentile Larsen-Miller Parameter. Because it is based on hundreds of tests.

So would the conclusion change if mean values are used? Yes, of course. Certainly it would increase the failure time somewhat.

Now, I do want to emphasize, the increase, I think, we feel that it's not going to be very significant based on what of scenarios we've ran. But it's a good point.

And then there -- yes, any questions?

Then there were some interesting questions also on the calculations for which we have addressed a vast test for some benchmark analysis. We did the benchmark, as Mike Salay had presented for the Westinghouse design.

As far as the finite-element analysis, we didn't actually draw an experiment and compare that failure time against a finite-element analysis. That would have been nice, but its results are intensive,

we had not done that.

But we used, our purpose was to compare that to the simplified calculator version. And I think that's, the purpose was served in what we did.

And then a couple of simple questions about what assumptions we used. Whether we used a stratification of counter flow, and we said yes.

And we, in the case of weld overlay should -- oh, there was one question on whether it would be accounted by the residue of stresses on the weld overlay. It's a very interesting question. But at these temperatures, really all those things wash off. You're not going have any residues, just the same.

The other question related to whether we model MSIP. MSIP doesn't change the thickness of the pipe, it just gives some compressive stress on the surface of the pipe.

And those stresses also would vanish when you start heating. Even before you go to the severe accident scenarios. So that wouldn't make much of an effect. That's our conclusion.

There's one more question related to, well, what if you have a PWACC crack growth? And the time scales are completely different.

PWACC takes a long time. And here we are

talking --

MEMBER STETKAR: There's not a whole lot of water left either.

DR. IYENGAR: Right.

(Laughter)

DR. IYENGAR: So I think all of you have read all those comments and that's about all I have.

CHAIR REMPE: Thank you. Are there any questions or comments before we go to the next section?

(No audible response)

CHAIR REMPE: Then let's move on.

DR. IYENGAR: Thank you. Ali and Selim.

DR. AZARM: Good afternoon. I am Ali Azarm. I presented detailed aspects of CHERPRA back in April, I think it was 2015, and to be consistent with others I am going to give you a very brief overview and then talk about example common resolution and entertain questions and feedback that in the past has been very beneficial to us and I am sure it's going to be the same today.

I already said that I am going to give a summary of the PRA related work. I am going to go through briefly, select an example, and you have heard about Zion, Calvert Cliffs, and I have been asked as a messenger to also talk about path forward. I am just

a messenger.

One thing is important in this presentation when you look at the NUREG we did look at both pressure induced and thermally induced, the creep rupture that you heard, and we had to do that, that was NRR user request and we had to look at both.

But for the sake of this presentation and this brief, you know, presentation we are going to just focus on creep rupture.

So what was the objective of our PRA study underlined that we were asked -- Yes?

MEMBER STETKAR: Just don't turn away to the screen. Use the mouse if you want to highlight something so you talk to the microphone.

DR. AZARM: Oh, all right.

MALE PARTICIPANT: And you can see it either here and --

DR. AZARM: Okay, all right. Thank you.

CHAIR REMPE: It might make it easier, too, if you'll go to presentation mode.

DR. AZARM: Well --

MALE PARTICIPANT: This might be a --

CHAIR REMPE: Is that possible?

If you click on the icon that's the screen on the right --

MEMBER STETKAR: In the red on the right hand side.

CHAIR REMPE: In the red, keep going. There you go, click on that guy.

DR. AZARM: Okay.

CHAIR REMPE: There you go. It's been bugging me and I hadn't say anything.

DR. AZARM: No, we came back to the RCS.

MEMBER STETKAR: It's one of the few -- If you are here for more than about four years it's one of the few skillsets I think that you develop.

DR. AZARM: I think I have --

CHAIR REMPE: That and microphone watching.

DR. AZARM: I have an additional problem, I forgot to bring my glasses, so I have to --

(Laughter)

DR. AZARM: The objective we had, and it is basically what imposed on us by the program and the limitation of the program or restriction of the program or resources of program and what user requests asked, we are developing simplified methodology for quantitative assessment of the risk for C-SGTR and we have to address both thermally induced and pressure induced.

And the underlying word is simplified methodology, so as I am presenting you are going to see that I have made some shortcut and I will tell you the thinking behind it, but we had to meet this goal.

To do that the first thing we have to do is to define the calculational process to estimate the conditional probability of consequential steam generator tube rupture given an accident sequence that challenges the steam generator tubes.

And once we have that calculational process we went through showing how to the thing is going to work. A couple of examples, we did focus on large releases, large early releases. We have used a Westinghouse and a CE plant for doing that.

Okay. What are the requirements for this? The first thing I have to do if I want to do an EPRA, I have to calculate the probability of consequential steam generator tube rupture given a sequence and, for this case, creep rupture that has resulted to a core damage.

What do we need to calculate? What is the C-SGTR? In order to calculate that I have to calculate what is the probability that a steam generator tube fails at a given time after an accident with a certain leak area.

That's the part that we didn't hear before because many of other things that you heard it calculates what is the probability of tube fail. It doesn't look at the flaw, it doesn't look at the leak area, and doesn't look at probability, because in a sense they are not carrying out all this probabilistic calculation in there.

I ever have to do the same thing for hot leg and surge line, and Raj showed one of the slides from the calculation, the equations that is in the calculator, and if you paid attention to that slide it was coming up from an EPRI report and the good thing about it is that you might have noticed it had plus/minus and some error in it, so in a sense it had already uncertainty built into them.

So if I use those equations, because of those uncertainties at a given time, I don't tell you if the hot leg failed or not failed, it's not deterministic, it gives you a probability.

So the first element is to create this calculational process to calculate this C-SGTR. There are two other elements in it. This is given a sequence that has resulted in CD.

The question is that what are those sequences and what are those frequency of them. On

this we were asked to use existing PRA and come up with a way to do that and there are some questions that this may not exist in PRA, it may not identify all the sequences.

And the last item is basically given that I know the frequency of these sequences that I am interested in, and I am going and calculating this consequential steam generator tube rupture probability, how do I decide that this is large enough and early enough to call it LERF.

Okay, you have heard about calculator and this is one of the first pieces that we did back in 2010 and 2011. We haven't yet modified it since then. This is basically a JAVA software program, a very large one.

It's built based on older work that NRC has done in the past, and Raj mentioned Dr. Majumdar, for the last ten, 15 years he published NUREG CRs talking about hot tube fails, what are the underlying equations for failure of the tube under tube rupture, under, you know, pressure induced, et cetera.

Also we take advantage where we didn't have simplified equations from NRC core relations from NRC from what NRC industry provided to EPRI reports, and, again, you saw an example of it.

So what the calculator does is basically

it is a probabilistic failure calculations of tubes, hot leg, and surge line as a function of time after accident.

It also calculates for the steam generator tube the cumulative area as a function at the time that it is being created. And those are the stuff I need in order to define my containment bypass or C-SGTR.

I have to say, okay, I think by this time I have probability of 0.5 to get a steam generator tube rupture that is equivalent to six centimeters square of leakage area. I need to know that, otherwise I can't talk about release, LERF, or anything else.

What goes to this calculator, of course, the first thing that goes in it is the flaws, if the steam generator tubes have flaws, and we needed to establish statistics on the flaws so we can stimulate flaws for any plant at any cycle of their life or use the plant-specific flaw sets for that plant.

We also need to accept all the results that might generate, the TH results, for a steam generator tube temperature, hot leg temperature, hottest tube, average hot tube, and cold tube, and we have to feed those to this calculator.

Now remember when MELCOR goes to calculation it doesn't have a timestamp that is fixed,

it's changing here and then you get these big, big files that if I put that in the calculator it's going to blow it up.

So we have to do some reprocessing to have some sanity check before going to this calculator. We also have as a part of this calculator a library of material properties, like the first strand, Larsen-Miller Parameter, et cetera, as a function of temperature.

And then on top of that we get plant specific information on diameter of a steam generator tube, thickness of the tube, thickness of hot leg material, et cetera.

So we put this in and we basically calculate at the end that as a function of time at any time this is the probability of having a consequential steam generator tube rupture of this area and we do it for five centimeters, six centimeters, et cetera, and this is the probability of having hot leg failure.

And using those two information we do lots of post-processing in order to calculate the probability of containment bypass. Just one information that you might noticed in the report, when they did the MELCOR run for Calvert Cliffs they differentiate the different Loop A and Loop B and you

have created two different sets of temperature trends.

So you have to run the calculator twice and convolve the result outside the calculator. All I am trying to say is that, and this goes to one of the comments, that to use this calculator you can't be novice and you have to do lots of pre and post-processing.

It was not designed, it was relatively very small funding, it was designed as an in-house tool rather than something for outside.

MEMBER MARCH-LEUBA: Ali?

DR. AZARM: Yes?

MEMBER MARCH-LEUBA: Educate me a little bit, okay. When you run the MELCOR or you end up with the deterministic temperature profile at hot leg and of the tube, correct, I mean especially the deterministic?

DR. AZARM: Correct.

MEMBER MARCH-LEUBA: And you assume that neither of those fail in MELCOR because you did not release the pressure? I mean the moment something fails then nothing else will fail because you go off, right?

So you take those profiles in temperature and then put it into your model that tells you what is

the probability of failure?

DR. AZARM: Yes. So what we ask Mike to do, we say suppress your creep rupture failure and other stuff in MELCOR, just give me the time, temperature, pressure, et cetera, and then I feed it to this --

MEMBER MARCH-LEUBA: And then you have a model of your material that tells you that your temperature is 1000 and your --

(Simultaneous speaking)

DR. AZARM: Yes. Yes, and then it calculates, you know --

MEMBER MARCH-LEUBA: Oh, it is 10 percent?

DR. AZARM: Oh, yes. It calculates all the material, property, et cetera, and, you know, you saw the creep rupture equation, that was saying that TR is equal, this will actually integrate over all this little damage.

MEMBER MARCH-LEUBA: Okay, I got it.

DR. AZARM: So there is some complication, and I don't remember them all. This was done, what, like five, six, seven years ago. The important thing that you said, and I was hoping that, because I think in the last meeting somebody asked the question how we handled our certainties of all the material properties, et cetera.

The question is that how are we going to handle the uncertainties of TH. We haven't, we haven't touched it, but I think that question came out a couple of years ago.

And that is really if you have to do that you have to run MELCOR 20 times each time with one realization, run the calculator 20 times.

MEMBER MARCH-LEUBA: More like 200. All you do -- In those 20 runs you change the time at which the temperature reaches, not the temperature.

DR. AZARM: Yes.

MEMBER MARCH-LEUBA: So there the conclusions would not be that different. You have an uncertainty of time when you reach 1000.

DR. AZARM: Yes.

CHAIR REMPE: Before you go on, you talked about that this was developed primarily to be an in-house tool. I know Kevin said at the beginning of the meeting that on a case-by-case basis that it would be released perhaps to the outside.

I believe in some of the questions and answers they talked about that when the public asked for a copy of it and there was some discussion about perhaps your organization wanting to make it

commercially, or some organization wanting to do some commercially-available tool.

How much industry interest, has there just been the one public comment that has requested it? Is there a lot of interest in getting this? I am just curious because of the -- It seems like the response is now, well, we'll think about it.

And maybe you are not the one to ask, maybe it's Kevin, but --

DR. AZARM: No, I am not the one to ask.

CHAIR REMPE: Okay. But I'd like some additional information on what's going on about this. Yes?

DR. SANCAKTAR: Push, okay.

CHAIR REMPE: Yes.

DR. SANCAKTAR: Just push where it says push. Selim Sancaktar. We only got two sets of comments from the public and in one set there was this question and the other one there wasn't.

CHAIR REMPE: Right.

DR. SANCAKTAR: And in principle we are not adverse to making it available to interested parties when we are not equipped to assure that it will run on an operating system of their choice and we cannot

handle questions about it and so on.

So we are not adverse to releasing it but we would like to choose a path that says use it at your own risk and don't use it for licensing.

CHAIR REMPE: Okay. And there has just been one question about it?

DR. SANCAKTAR: One --

(Simultaneous speaking)

CHAIR REMPE: So it's not like there is a pathway of people coming to the door asking questions, because I was puzzled when I saw that two companies were interested in commercializing it.

DR. SANCAKTAR: Right.

CHAIR REMPE: Okay, thank you.

DR. SANCAKTAR: Yes.

MEMBER SUNSERI: Let me ask a follow-on with that. One of the five deliverables in a user request is regulatory tools and guidance for future risk assessments, so does this satisfy that user need?

DR. SANCAKTAR: When we write the guidance in the next stage we will put qualifications on that, but we are going to minimize the use of the calculator because, as Ali pointed out, it requires a lot of pre and post-processing.

Actually, if somebody gave me the funds I have a list of things that I would like to improve and minimize the pre and post-processing because as we used it we got smarter and we are dying to improve it.

However, there is a limit to what we can do unless there is a demand within the organization. But to answer your question, we'd like to discourage people from using it internally unless they are in command of it.

I mean at this point there are only two people who use it, Ali and me, there is nobody else. If we for some reason disappear the tribal knowledge may be no longer available.

MEMBER STETKAR: Selim, since we are talking to the tribe, have you guys used it yet for the Level 3 PRA project for Vogtle? Because we have heard -- We haven't seen, you know, if I switch now to the PRA Subcommittee, we have heard that they are addressing consequential tube ruptures in that model, so are you using it at least in that context?

DR. SANCAKTAR: Yes.

MEMBER STETKAR: Good.

DR. SANCAKTAR: In fact I have a calc note on that and I calculated some input for it and started

--

(Simultaneous speaking)

MEMBER STETKAR: Yes, but --

DR. SANCAKTAR: -- applying the calculator and the methods. However, it's not publically available.

MEMBER STETKAR: No, no, not -- Yes, it's not and we haven't -- I was just curious whether, you know --

DR. SANCAKTAR: Right.

MEMBER STETKAR: Matt had asked the question is it at least being used in-house and we had heard it was and I am glad to hear that it is.

DR. SANCAKTAR: It was a good exercise because it really enables us to see the things that need to be explained and so on.

MEMBER STETKAR: Right. Good, thanks.

CHAIR REMPE: So it may come up at the end, but there is this user need saying tools available for future risk assessments and will the -- Your response is why I want to discourage people from using the calculator unless they know how to use it.

What are the tools, and maybe this is for the end of the day, but what tools are envisioned to

be available for future use based on this study, not just the guidance but the tools?

DR. SANCAKTAR: Well we'll cross that bridge when we write it soon. However, like when we made this calculator initially we are training for about four or five people in NRC and all of them are now unavailable, not even -- Well, maybe one of them is still around.

So the keeping -- This is such a niche subject that keeping it fed with experts is a challenge, so we don't want to impose on the offices that may use it as a requirement in the next stage of things we will suggest.

We will simplify the model so they don't have to use it. They may use it if they choose to, we have the user manual, so --

DR. AZARM: If I may add, I think something we have talked about, we look at this document as a technical basis. The guidance document is something, it's going to be based on this but it's going to have different stuff in it.

Regarding the calculator and the other subject, the PRA guidance, regarding the calculator I think one of the things that we are envisioning is that

we would have sets of runs tabulated that NRR can use and use that as a tool.

And, you know, we tried to make it, you know, simplified, bounding for the use. This is not for the state-of-art PRA, this is more for repeated application, routine application.

CHAIR REMPE: That helps.

MR. COYNE: And to add a little bit more, Kevin Coyne from the Research Staff. So our desire is to make it available to the public it's just a matter of what is the mechanism to do it.

So we have a very active co-distribution process that takes resources to support. This isn't the kind of code, this isn't like SAPHIRE or TRACE or MELCOR that we have a formal distribution mechanism to do.

It's a little tricky even to get the calculator in ADAMS, for example, to keep it archived. You could put it in there, but the ADAMS people don't really like non-document-type files in those kind of things, so we have to work through some of these issues.

Selim had actually drafted several years ago a research information letter, a RIL, that talks about the calculator and the thought at that time was

that the RIL would provide, you know, the context for using the calculator.

IESS and ISL before them had developed a very detailed user manual for the calculator, it is publicly available. So the RIL in combination with the user manual would provide enough background for somebody, a sophisticated user to use the code.

The other issue with the user need is that our customer has evolved since the user need was first written. I am not sure if particularly any of the PRA folks are familiar with Bob Palla who used to be in NRR, he was a fairly sophisticated user of the tool so he wanted a calculation device like the calculator.

I think the needs from NRR have shifted since that period and one of the key tools I think they are looking for now is a simplified method the senior reactor analyst can use for the STP, so this would be something that goes into the RASP handbook.

So something they can use in conjunction with the SAPHIRE Code and the SPAR models for a specific event they are looking at or a condition they are looking at that they can get an estimate for LERF.

So that's a little different use than I think what the user need had first envisioned. So we

are maintaining communication, obviously, with NRR as our customer and we are working with them as this report reaches fruition to try to figure out exactly what they are going to be looking for.

But the calculator is something we want to put -- We want to get into a good place right now. I know it seems a little soft what we are doing with it but we do want to get it to a place where it is available and so others beyond Selim and Ali would be able to use it, so getting the right documentation and putting it into stable format where somebody can grab it from an electronic database or in ADAMS and use it is where we want to be but we have to figure out how to get there.

CHAIR REMPE: Thank you.

DR. SANCAKTAR: One more thing, when this question of distribution came up I actually put this calculator into ADAMS years ago for people to use. So when this came up I checked it out, I tried to use it, somehow it didn't work.

I don't know whether it was because it was --

(Simultaneous speaking)

MEMBER STETKAR: Yes, there is that.

(Simultaneous speaking)

DR. SANCAKTAR: Or was it because ADAMS as things that -- I don't know, that's not my area of specialization, so I --

DR. SHACK: You got a new computer.

DR. SANCAKTAR: Hmm?

DR. SHACK: You got a new computer.

DR. SANCAKTAR: Right.

DR. AZARM: He actually did.

DR. SANCAKTAR: Yes, actually, that was another thing. You think a computer is a computer, operation, operating system. The same Windows at home and here do different things. So, anyway, I had a terrible time.

It took us like three days to figure out, we had to go back to another model. So that's when we realized that we are out of our depth in software distribution.

(Laughter)

DR. AZARM: Okay. For the sake of time, the next slide basically talks about the steps in the risk assessment, at least for what we have done.

It basically says, oh, we are going to identify sequences, we are going to have some TH analyses, we develop flaw set, either plant specific

or simulated, and calculate this conditional core damage.

The first thing is scenarios, and I know this is a very important issue. We know the scenarios we are interested in is the scenarios that the primary is high, one or more steam generator is dry, and the secondary is low.

So in a sense if I have a SPAR model or an existing PRA model, which is a boundary condition for this, we can go to event trees and identify the ones that is primary pressure high, there is no AFW or at least one of the steam generators is dry.

And we always assume, per guidance that we assume the secondary pressure is low. There is enough leakage to MSIB and other stuff that keep the pressure low.

We also noticed when we were doing this pilot application on Calvert Cliffs and Zion it is useful to look at the bending of the Level 2 because they are asking a similar question of about high primary pressure, if a steam generator is dry or not.

For every high primary pressure the rate consists and the steam generator, dry or not, sometimes they are identified, sometimes they don't. So it looks

like a very simple task but if you want to use the existing PRA it is quite involved and needs a guidance.

MEMBER STETKAR: Part of the problem, Ali, is that -- The reason I asked earlier about general guidance in terms of is my plant, because of the fundamental design and configuration of the steam generators, more or less vulnerable to this as many people have not systematically looked for these scenarios because they haven't been taught that they need to look for them.

You can perhaps define them out of existing models, but in many cases existing models have not been structured to particularly look for the case where I have one and only one, let's say, of my steam generators dry and depressurized, because that has never been considered to be important to core damage.

It's important if I have all of them dry, but I don't necessarily need even in that case to think about depressurized, now if you assume they are all depressurized.

So my whole point about structuring a risk assessment to kind of evaluate these conditions if they are particularly important for my plant I may have to structure the front end, the Level 1 part of my risk

assessment, differently compared to the way I structure it today.

DR. AZARM: Right.

MEMBER STETKAR: Some of the way I structure it today will give me these sequences like station blackout or, you know, a steam line break upstream of the MSIBs or a steam line break downstream with MSIB failure, some of those kind of standard ones.

DR. AZARM: Yes.

MEMBER STETKAR: Other ones won't and in some cases it can take a lot of work to restructure those models. So that's why, you know, having just a general sense of do I need to worry about it is really important.

DR. AZARM: We fully agree on that.

MEMBER STETKAR: Yes.

DR. AZARM: You know, you address Level 1 and you address, you know, the one steam generator dry. The issues like some of the scenarios, they don't even ask for AFW.

MEMBER STETKAR: True.

DR. AZARM: It's probable cause with HBI failure. Now you have to go and see what fraction, especially when you go to external event.

MEMBER STETKAR: Right.

DR. AZARM: Then you go to Level 2 and Level 2 has also similar problem. They only ask for dry steam generator if they have a steam generator tube rupture initially.

MEMBER STETKAR: That's right. Yes, that's right.

DR. AZARM: Because they were not worried about swapping, that's it.

MEMBER STETKAR: Yes, that's right.

DR. AZARM: So now you have to -- So, yes, if we want to do this right it's going to impose requirement and guidance for Level 1 and Level 2 and if we have cases that, as you said, the geometry and design that makes them very vulnerable, that type of work is needed.

I do believe, even though I am not involved with, and I don't know if I am -- Anyway, I am going to go ahead and say there is another program that I have no involvement with that they are looking at some of these scenarios. Is that correct, Selim?

DR. SANCAKTAR: Which?

DR. AZARM: That you guys looking under a different program to identify additional scenarios?

MEMBER STETKAR: In principle it may be

the Level 3 PRA but we're getting a little bit off topic here.

DR. AZARM: Yes, yes, yes.

MEMBER STETKAR: It's this -- And whether or not they are is a different issue, so let's --

DR. AZARM: Yes. So now when we went to Zion And Calvert Cliffs based on this process we identified lots of scenarios, now they were dominated by SBOs.

So once you identify these scenarios the question is that you cannot do hundreds of thermal hydraulic analysis. You have to define representative scenarios.

So the two representative scenarios that we are using we are trying to bend all these scenarios either to short SBO or long SBO. Also, I have to say one more thing for the benefit of Dennis and John, we are defining C-SGTR as guillotine break of once cube or more, and there is a reason for that.

If you have, as this is what we understand both from TH or Westinghouse, and see if you have less than one cube it doesn't even challenge the secondary side relief.

So we want to have -- And then the whole

question is that is that a large release and if you are not challenging the secondary side the hot leg could fail, so we are defining a size that challenge the secondary side relief and in a sense for us we always assume that relief is open as a bounding.

I just want you to know the boundary of what we have done so when we define LERF those are the conditions.

Okay, these are from now is easier. We looked at Zion, we looked at Zion and they use RELAP/SCDAP for the thermal hydraulic part of it, for the CEV unit at Calvert Cliffs and MELCOR as the thermal hydraulic part of it.

Just for your information Calvert Cliffs has an IPEEE with a Level 2 PRA, relatively detailed large event tree. Zion had original Zion PSA but also it was a part of NUREG-1150.

Again, for your information there is lots of sensitivity analysis done as a part of TH for both of them.

A few words about the flaw, I just said that one of the input to calculator is a steam generator flaw. You can basically, if you want to do Cycle 15 you know at the beginning of Cycle 15 what was the flaws

in your steam generators.

They do inspection and they have -- So all you have to do is that how many flaws can we generate in the next cycle and then you add the two, so you need to know at each cycle, Cycle 15, 10, or 12, what is the flaw generation rate and you need to know what is the sizes of those flaw, depth element, you need three information, or you can simulate from.

So I don't even know this specific plant, I want to do a simulation based on average industry. So to do that we need to have three statistics, what is the flaw generation rate, what is distribution of the depth, and what is the distribution of length or size of the flaw.

To do that there was no work done. The only work was done in the past was by Gorman but it was for mill and yield (phonetic) or the steam generator that did not apply.

So we tried again back in 2010/2011 with the help of NRC to look at some flaw data and establish the statistics on it. We are quite comfortable with it, but remember this is for average industry.

We cannot use this data to differentiate between good plan, bad plan, and average plan. That

was what was done in Gorman study, you know, ten, 15 years ago. If we want to do that then we need additional data to address plant-to-plant variability.

This is just an example of, you know, I don't want you to focus on it, all it is is that if I use this data and use it as an example for Cycle 15 for a Westinghouse plant I will get 99.8 percent of the flaw generated in that Cycle 15 as flaws as less than 60 percent.

So as you heard if you are dealing with a Westinghouse plant that it going to give you C-SGTR when the flaws are very big, you are talking about 0.02 for a large flaw, a very small probability. That is going to drive your C-SGTR.

MEMBER MARCH-LEUBA: So just to keep me awake, that says that Westinghouse tubes never fail, they have a 0.002 probability of failing?

CHAIR REMPE: No.

DR. AZARM: Under creep rupture that's basically very close numbers we'd get.

MEMBER MARCH-LEUBA: Yes.

DR. AZARM: Now the only way they can fail, and that's why I made the point earlier, this assumes that you have done a very good inspection and no big

flaw was left, probability of detection for a deep flaw is one.

Now if you go to a bad plant, and I quote "bad plant," that the inspection was not effective and you left a large flaw then these numbers can change. This is showing average industry plant. You have to look at the plant-specific stuff.

The next slide basically shows you some of the results. If you look at Westinghouse you see you get -- What?

(Off microphone comment)

DR. AZARM: Oh, I'm sorry. If you look at Westinghouse as you mentioned, you know, 2 percent, if you look at C-SGTR it's about 1.3 ten to the minus two, because Westinghouse is designed only for large flaws to fail, so this result after all these calculations is consistent with that.

When you look at CE you get -- I have to -- When you do MELCOR runs if you assume SRV it's a struggle from very beginning. You depressurize very fast, accumulate all the charge, and you have a totally different scenario.

So if SRV is open from very beginning then you have almost one probability. If SRV is closed you

have 0.22. But that's basically the nutshell of everything that was discussed today that why we think the CE is extreme, or the example of CE is extreme, bad for C-SGTR, and Westinghouse is the other side, it's doing good.

This really has nothing in it. It says, if I look at this, there is four terms in PRA and we get some of them from existing -- The next slide, again, was discussed very detailed.

It basically identifies all the factors that is important to make the probability of consequential steam generator tube rupture worse. Of course, the flaw is the most important one from our viewpoint for plants that have a relatively good design.

Also I want to try to say that the report talks a lot about FLEX and SAMG and how they can help to bring these probabilities down, but frankly we do not do any quantification for neither FLEX equipment nor SAMG mitigation.

Now a reason for it, we are trying to do a state of practice PRA. For the FLEX equipment we don't even know the timing of operation where it's the axiom (phonetic) of sequence timing.

Unless this type of work is done I think incorporating in PRA is difficult. For SAMG, again, we discuss them qualitatively because of some issues regarding the effectiveness of SAMG and operation of the equipment post core damage.

For example, is your core going to open, it's going to jam closed, or is it going to chatter, we don't know. So these are discussed qualitatively but they could be important.

We had three sets of comments and I have identified some examples of them. The comments from ACRS members, comments from PWR Owner's Group, and comments by a very friend of mine, actually, Dr. Fynan. He used to do my consulting, he is now in Korea.

We cannot really say that this comment was PRA because PRA is so integrated with other stuff, like the comment of RCP 21 GPM also applies to PRA.

So we have input it to other comments as well, but also then comments that it's specific to PRA, and I am going to discuss that.

We don't think any of the comments are a showstopper, at least the ones that we got from public. We feel some of them were beyond the scope of this study and we have clarified that and we have written that in

Section 8.2, the certain things we did not address, but the comments that were within the scope we tried to respond.

The first comment was regarding RCP seal leakage. I think this has impact both on TH as well as on the PRA and the question was the 21 GPM comment.

As Mike went through it the 21 GPM does quite a bit affect early on in depressurization but the long term behavior is exactly the same as our base case.

So since all the failures were interested based on a long-term behavior we don't think it impacts the result of the PRA. I have to explain this because there is a comment that why did you guys identify a SIT actuation of 700 PSI for Calvert Cliffs for CE plant. This was all my fault. It's nothing to do with the thermal hydraulic or others.

I have a bunch of tables and the tables was basically saying, okay, our base scenario, our primary pressure is 2250, which is primary relief set point, and if I don't depressurize because of the hole in the steam generator tube rupture how fast is going in my hot leg fail.

Then I did a sensitivity analysis saying that what if I have big holes in my steam generator tubes

such that I am balancing the pressure to the secondary relief, so then I calculated at 400 PSI what is the time between C-SGTR and then we looked at other depressurization, 900, 700, et cetera, in order to get some feeling that what is high pressure.

Unfortunately, when I did 700 I carried the same wording as Westinghouse, called it SIT actuation. So I removed that, it doesn't impact any of our results, it was just a sensitivity analysis.

But, you know, at first we looked at it and we felt we had done something significantly wrong.

The next comment was why don't you guys acknowledge that FLEX and SAMG are going to reduce this, it's just not design of a steam generator, et cetera, and we basically did those changes, made clear in the report that, yes, we believe SAMG, FLEX, EDMG, these are going to be beneficial but we cannot quantify it.

The next comment is an ACRS comment. Basically what is very dear to John and I fully agree with him that how are you going to make sure that you have set of complete sequences for your analyses.

Salim, Dr. Sancaktar, put an appendix out in the report trying to at least look at one of the scenarios and see how significant it is and what I

understand, and is reflected in the comment, they are also going to look and see if they can do a better job using another program.

The next comment, regarding calculator software, I think Dr. Rempe discussed that earlier. There is a concern that, you know, this needs quite a bit pre and post-processing, might be misused and abused.

So they are going to -- Even though it's documented, et cetera, NRC may decide to release it on a case-by-case. Yes?

CHAIR REMPE: Excuse me. I think I didn't quite hear correctly, on the previous slide when you were talking about the last item under resolution did you say that they are going to do some additional work, I thought that we were done?

DR. AZARM: No, no. Please clarify.

DR. SANCAKTAR: Yes, we are done. I actually prepared a 60-page report which is not publicly available. It has proprietary information included in it.

So I tried to systematically go through types of scenarios that I in quotations called and modeled them they are subsumed in existing scenarios

and you have to unravel existing lumped scenarios to see them.

So I kind of went through it systematically for one case, it doesn't prove anything or disprove anything, you know, and in that case it turned out to be rather insignificant.

And I tried to summarize what's in there, in Appendix L, in about three pages. Whether I managed to convey some useful thoughts in there or not is another story, but there is nothing else to be done.

CHAIR REMPE: You are done. Okay, thank you.

DR. AZARM: Okay. This is my job as a messenger, basically what is the thinking today about path forward. NRC, the staff, would like to publish a final NUREG-2195 in 2017.

I think both from resource limitations and the work they have done they feel there is not going to be that much major changes. You know, there is lots of lots of changes that we are doing on it, but no major technical changes in it.

Right now there is a Subcommittee full-day meeting scheduled for May 3rd and Full committee for June 7, 2017, and one option to expedite the publication

of NUREG, because if you wait for this NUREG may not even published in 2017, is that they are requesting to reduce the length of the next Subcommittee meeting or even if it is possible to cancel it and just go with the Full committee meeting, but, again, I am just the messenger.

MR. COYNE: So, Ali, yes. Kevin Coyne from Research. I will bail you out. I actually thought Selim was going to present that slide, but --

DR. AZARM: I told --

MR. COYNE: But that's okay, that's okay.

DR. SANCAKTAR: That's what I thought, too, but since he took the responsibility I didn't want to break his spirit, you know.

CHAIR REMPE: You didn't offer him extra money, huh, okay.

MR. COYNE: So in our communication through Chris Brown we had I think last met as we were just receiving the public comments and we hadn't really had a chance to fully go through them and see what the implications for the report were going to be and so we had this full day meeting scheduled in May.

I think now that we have had the benefit of going through the public comments and going back

through the previous ACRS member comments, and, again, allowing for the feedback we get today, we felt that we were in pretty good shape with the resolution for what we have done to date so we did not envision huge changes in the report going forward into '17 combined with the thought that it would be good to get the report documented.

It has been a 6-year odyssey of getting to the point we are at. As you can tell from the discussion some of the analysis now is three or four years and, you know, it raises its own questions when we go back to, you know, questioning previous versions of MELCOR and that situation is only going to get worse if we hold the report longer.

So that was a key motivation for us to keep moving forward with the publication process. We didn't want to do that while we are continuing our engagement with ACRS though, so that was one thing that we wanted to get feedback from the committee on as far as is there a path that we could be responsive to ACRS issues, but, you know, get to the goal of getting the report published in a quicker timeframe.

CHAIR REMPE: So before we start discussing this I'd like your feedback on there was some

changes mentioned today where there was some suggestions for revising some of the text in the document, do you anticipate doing any of those changes or you really can't because of other reasons I don't know about?

MR. COYNE: So I've been trying to keep notes as we went and highlighting the ones that are key, and I'll have to get with the other staff to make sure that we have a good set. Just to go through them really quickly --

CHAIR REMPE: You don't need to go through them right now, but you are open to some suggestions for some changes?

MR. COYNE: Absolutely, and that's why we are here.

CHAIR REMPE: That's what I wanted here, yes.

MR. COYNE: That's why we are here. And so a lot of what I have heard I -- Personally, I am -- My apologies that you couldn't find some of these documents in the public forum, that wasn't the intent to make a conference paper very difficult to find, because I know how difficult those are to find, so we'll --

DR. SHACK: Appendix M.

MR. COYNE: What's that?

DR. SHACK: Appendix M.

(Laughter)

MR. COYNE: We can certainly provide those to the Committee right away.

DR. SHACK: Well, no, the Committee has them, it's the public and the --

MR. COYNE: Right. So we want to make sure you have them and then the other thing we are going to look at is if we can, if it's in ADAMS but not public can we just switch the flag over and make it publically available.

The Sandia report I think may be a little more problematic, so we'll figure out what to do with that. Just on a point with the NUREGs, and this is a nuance role, if you reference a document in a NUREG it's got to be publically available.

MEMBER STETKAR: That's right.

CHAIR REMPE: That's what I --

MR. COYNE: But if you footnote a document in a NUREG it doesn't have to be publically available.

CHAIR REMPE: Well right now the Sandia report is a reference.

MR. COYNE: Is a reference. So we did look at that, we might have missed one or two, and that wasn't our intent, so we'll look at that to clean that up, and my apologies for issues with that.

MEMBER STETKAR: I forgot, that's true.

CHAIR REMPE: Yes.

MR. COYNE: There were a number of documentation issues that came up, material properties, some of the geometry issues.

CHAIR REMPE: There will be a transcript and so --

MR. COYNE: Right.

CHAIR REMPE: Yes.

MR. COYNE: And so our intent is, particularly for these documentation issues to better clarify what we did and what the limitations of that approach are. I certainly want to get those in the report.

As far as doing additional analysis and additional work that would be a much tougher thing. I know you don't like to hear this, and I don't like to hear it either, we don't have a lot of budget to continue to do any significant thermal hydraulic, structural, or PRA work on the project, so the goal would be more

towards making sure our documentation is very clear on what we didn't do and the limitations associated with what's in the NUREG at this point.

CHAIR REMPE: So for my education talk a little bit about the guidance document that is going to come in the future.

In the updates to this document would there be some possibility to have any glimpses of what one would see in this guidance document in the future at all, or how does that work?

MR. COYNE: So are you familiar -- Well, the RASP handbook, it's a publically available document, but it's used by the agency risk analyst to make sure they are consistent in how they do analyses for say the accident sequence precursor program, NOED support, significance determination process, NDA.3 assessments for our event response.

MEMBER BLEY: Can I interrupt you there?

MR. COYNE: Yes.

MEMBER BLEY: Is it on the public website?

(Multiple yeses)

MEMBER BLEY: Oh, I thought it was only on the in-house website, okay.

MR. COYNE: No, it is. At first it had

been not publically available --

MEMBER BLEY: Yes.

MR. COYNE: -- but about seven or eight years ago we made it public and all updates are public.

MEMBER BLEY: Okay.

MR. COYNE: And I can send a link through Chris so it is available for the Committee members. That has a lot of guidance on various aspects of doing a PRA.

One of them is induced steam generator tube rupture. I believe that is currently in there in a very simplified manner. So one goal is to provide more detail and better technically-based guidance in the RASP handbook for the senior reactor analyst and other analysts to use.

So we would generate an update to that section. We haven't really worked out how long it would be and what it would say.

We definitely got some good feedback from the meeting today, so I think things like covering the geometry, considerations for the steam generator and other things like that would be very good to put into that guidance document.

So at least if we can't give a tool that

can handle any type of steam generator at least we can tell people what type of steam generator the results from this report apply to so they can decide whether it's applicable for the analysis they are doing or not.

And then it would walk through a method that they can extract, very similar to the method that Ali described of using the SPAR model to extract, you know, the high, dry, and low sequences, bending them, counting them, and applying the factors, the four factor formula that Ali went through, for their specific STP analysis, for example.

And so this would be more pertinent for a LERF evaluation rather than, obviously, the CDF which drives a lot of the STP result.

CHAIR REMPE: Again, this is because I want to understand the process, is it going to be something that's done within a year, I know your resources are limited, or five years, because the more that you have in this document the easier it would be to generate the guidance, and if there is a 3-year hiatus it's going to be harder and I just am curious, that's why I am kind of pushing can you put more in the conclusions that would facilitate the guidance development?

MR. COYNE: Yes. So as soon as we are done with the NUREG that actually is one of Selim's next assignments is to generate the RASP handbook update.

CHAIR REMPE: Yes.

MR. COYNE: So ideally, and I think -- Our representative from NRR is here and I think he would be happy to see us get that done next calendar year.

CHAIR REMPE: Okay.

MR. COYNE: And then they have a process they go through with their RASP handbook of -- And it's up to NRR how they do that, whether they make it, you know, for trial use as a draft and then put it in the RASP handbook formally or whether they go through a different process to do that.

But our goal would be to get it to NRR so they could decide how to best use it going forward.

CHAIR REMPE: Okay, thanks. Others have questions or comments?

MEMBER SKILLMAN: Joy, is this your final round?

CHAIR REMPE: No, I was going to do the public, which I'm not sure if there is any public, but I'll see if there is, if there is anyone in the room who feels a desire to make a comment at the last

Subcommittee meeting for ACRS for 2016.

(No audible response)

CHAIR REMPE: Okay. Is anyone from the public out there and if so please speak up now. Well, I think there isn't and so at this point let's do as we always do and as we go through I would really appreciate your input on what the path forward should be.

Are we ready to jump into a Full committee meeting, do we want a Subcommittee meeting before that, and your thoughts about is it good enough to go or is there something really strong you see that you would like to see modified?

Let's start with the guy on the line, just out of curiosity are you still there, Mike, do you want to go first?

MEMBER CORRADINI: Yes. I have been trying to ask questions, I think I was on mute.

CHAIR REMPE: Oh.

MEMBER CORRADINI: Can you hear me now?

CHAIR REMPE: Yes. I am sorry, I thought you were able to talk.

MEMBER CORRADINI: No, I wasn't. Okay, let me ask about Slide 66, just a question for

clarification. Why is it that this extended usage of the turbine aux feedwater system the probability of failure goes up?

DR. AZARM: Is that Mike Corradini?

MEMBER STETKAR: Yes.

DR. AZARM: Yes. Shall I or do you want or thermal hydraulic people?

MALE PARTICIPANT: Yes, please.

DR. AZARM: Mike, it's a couple of things, it's very strange. One is exactly what you are asking and the other one is that why when SRV opened, you know, that we get an hotter temperature, but let's go back to your question.

We had the same question in our mind four or five years ago or so. We went and looked at the Delta T between the hot leg and the hot tube and the hottest tube, they were very comparable.

We looked at a bunch of this stuff. Basically what we find out that the ramp, that when the temperature ramps up, it is much, much faster during the short station blackout when you don't have turbine driven AFW and the rate is slower for the long station blackout.

Okay, so -- And, you know, part of it is

because of in longer station blackout your decay heat is less and you have cooled down. And so the ramp rate is different and you have to think about how we are calculating the creep rupture.

It's not just the Delta T between the hot leg and hottest tube, it's actually if you look at the Larsen-Miller equation it's a function of Absolute T and the effective stress and you integrate that over time.

So it all has to do with that rate of ramping and how that integration over time result and also remember that for the steam generator tube the leak area is slowly changing with time because of the creep, so I can't tell you exactly.

We know the reason is because of the ramp is slower, but it's just the code is giving us this number.

MEMBER CORRADINI: Okay. But so what you are telling me is that you believe the physics of the counterintuitive result?

DR. AZARM: Again, I should -- I didn't have any intuition. I was trying just, I saw something and I was trying to figure out what has caused it.

MEMBER CORRADINI: Okay, because the

reason I am asking the question such as this is if I have FLEX or I have extended turbine aux feedwater I would want to keep the generator cool and if I fail that later in time you are saying the chance of the survival are larger, or the chances of survival smaller.

MEMBER STETKAR: But be careful, FLEX is intended to prevent core damage period.

(Multiple yeses)

MEMBER CORRADINI: Yes, I know that, John, but that's not -- If they get into some sort of degraded state they are going to want to re-put water into the steam generator or keep the aux feed working.

DR. AZARM: Yes. Also, Mike, you know, you are buying so much time that we are not really worried about LERF by that time.

MEMBER CORRADINI: Okay, that actually leads me to my second question. You quoted six square centimeters as an important value, and I can't remember why you said that.

DR. AZARM: Oh. Basically, initially we were trying to look at -- Okay, what are the different concerns we had. One concern was that how much the tube should leak in order to pressurize the steam generator for relief valve to start actuating, so that was the

first concern.

MEMBER CORRADINI: Okay.

DR. AZARM: The second concern was that what should be big enough that if the relief valve is actuating I can call it a big release.

So there was all these different thoughts in our head and we did some back of envelope calculation and looked at, you know, I think that the Westinghouse, Don Fletcher's report, and all of these added up to us to conclude that at least we need to have an area equivalent to one tube failing before we can talk about big release and --

MEMBER STETKAR: That's bigger than one tube.

MEMBER CORRADINI: Yes, that's what I was thinking --

MEMBER STETKAR: Six square centimeters is like an inch diameter tube, so that's bigger than one tube.

DR. AZARM: Yes.

MEMBER STETKAR: I did a back of the envelope and I measured six centimeters --

DR. AZARM: Okay, I now remember, it's twice one tube because -- I'm sorry, I said one tube,

because when you do guillotine break of one tube you get 3 and half from each side, that's why it came to 6-1/2 centimeters.

DR. SANCAKTAR: Yes, the simple answer to this -- This is Selim Sancaktar. The simple answer to this question is for the Westinghouse plant we studied, which is no longer in existence, the tube, if you take the tube area it's three centimeters and a little bit plus, three plus centimeters square, and with the guillotine break you get flow through both.

MEMBER CORRADINI: Okay.

DR. SANCAKTAR: So that's the simple answer to that.

MEMBER CORRADINI: Okay. All right, that's fine. So thank you for those two. I was trying to clarify something earlier I just couldn't get into the conversation.

So a general comment is if, I thought that was Kevin that was saying there is not plans for additional calculations, whether they'd be thermal hydraulic, structural, or et cetera, then I guess it's as good to go as it's going to be.

I do think though that I could see ways to improve it but given the importance of this where it

fits into things I would rather that it be starting to be used to inform the Level 3 PRA that John was talking about than hold it up another year or two. Thank you.

DR. AZARM: Thank you.

DR. SHACK: Just one comment on that though, that assumes the SRV is functional at that point, whereas it's -- You guys don't have a failure model for the SRV that you looked at --

DR. AZARM: At those temperatures -- There is a document out and I can't remember the number of it, at those temperatures post-accident there is a document that talks about operation of the valves.

DR. SHACK: Well, the Surry uncertainly analysis for SOARCA said it's 95 percent chance it's going to be gone.

DR. AZARM: Gone means what? Oh, jammed or it's stuck open?

DR. SHACK: Open.

DR. AZARM: Yes, this document --

DR. SHACK: The question how open is another question, but open --

DR. AZARM: Yes, this -- I agree. This document I looked at and, again, my memory, it basically said it shatters and then most probably it's going to

start but it was only based on one or two tests and they weren't sure. I can develop that document, but --

DR. SHACK: Well, yes, the question whether the Surry SOARCA analysis, but I mean to assume that it works is, you know, I find that even harder to believe.

MR. COYNE: This is Kevin Coyne from the Research Staff. So you may have noticed the awkward body language from the staff when this question came up and we had anticipated that the CE results would come up and it actually has been a source of vigorous debate in many emails over the last three days.

So we do need to get to the bottom of it so we can document it in the report because I think several years ago we had an reasonable answer that satisfied us and that has disappeared into the ether, so we are going to run that down to ground and make sure that -- This is the kind of thing that we would improve in the report to make sure that this kind of thing is explained to the best we can of why the seemingly counterintuitive results appear.

CHAIR REMPE: Yes.

MR. COYNE: But we do believe it to be --

DR. SHACK: To be real.

MR. COYNE: -- a true result, although, you know, for the uncertainties involved they are essentially almost the same number anyway, but we will improve the documentation for that.

CHAIR REMPE: Okay. Thank you. Okay, so let's continue on, and, Charlie, do you want to go next?

MEMBER BROWN: I have no additional information to provide to you. Whether it's a good idea to publish or not I would tend to agree with Mike relative to it based on what they said.

If that's as much as we're going to get then you ought to go ahead and get it out. But since I am not a thermal hydraulic, I am not a PRA guy. I'd leave that judgement to those who are more confident to make it.

CHAIR REMPE: Okay. Jose?

MEMBER MARCH-LEUBA: Yes. I also have no educated opinion on that, but the argument that Mike makes makes sense.

CHAIR REMPE: Okay. John?

MEMBER STETKAR: I have a marginally educated opinion. I agree with Mike, I think it's time to get this out and published. It's not perfect, but, you know, no study ever is, everybody always wants to

do more.

I encourage Kevin or somebody when you do the final washup on the report itself read it from the perspective of this is going to be publically available and will be interpreted as the NRC position on this.

And similar to some of the comments that you have received from us on SOARCA, it is what it is, it's not what it's not, and make sure that people recognize pretty clearly what it is not.

So, you know, going a little too far in terms of drawing conclusions is a dangerous point. If there is some way that you can provide some reasonable insights about, you know, high for lower vulnerabilities but without too much detail I think would be really useful.

I'd hate to see somebody pick this up and say because I have a Westinghouse plant I don't need to worry about this at all. That might be the appropriate conclusion for my plant, it might not be for somebody else's Westinghouse plant, or because I have a CE plant I absolutely do need to worry about it and it's the worst possible thing in the world.

So just, you know, as you do your last read through kind of keep that in mind.

CHAIR REMPE: And these comments I am not hearing anyone saying do we need another Subcommittee meeting.

MEMBER STETKAR: Oh, I don't think we need another Subcommittee meeting.

CHAIR REMPE: Okay. I am seeing consensus around the table. Mike, am I hearing that from you, too, that no additional Subcommittee meeting?

MEMBER CORRADINI: Yes, yes.

CHAIR REMPE: Okay. Dennis?

MEMBER BLEY: I don't disagree with that. You had another scheduled.

CHAIR REMPE: That was just from a guess.

MEMBER BLEY: Yes, and part of the idea was to bring our new members up to speed, but we don't always do that, and the stuff is all here.

I don't have a strongly informed opinion. I agree with what we have heard so far from the others, but not with a lot of substance behind that agreement.

I still need to read more of this stuff and I am a little nervous about a calculator that only two guys know anything about. So I don't know when to deal with that or how to deal with that.

So, yes, I think getting some exercising

on it, getting some more people involved will be good, but that won't get more people involved with the calculator because you are doing the work for at least the next ones that are coming.

CHAIR REMPE: Okay. Matt?

MEMBER SUNSERI: Yes. From my view I am in concurrence with the don't see a need for an additional Subcommittee either, but, you know, I could easily support having this moved to Full Committee and begin the letter writing at an appropriate time based on our workload schedule and the processing of the NUREG and that process, yes.

CHAIR REMPE: Okay. Dana?

MEMBER POWERS: Put out the report, don't need a May Subcommittee meeting. I think we may want to weigh in a little more strongly on these kind of unexpected mechanisms to fail the reactor coolant system.

I personally remain concerned that our primary accident analysis tool is predicting scenarios, especially for station blackout which seems to be the predominant accident of concern lately, is predicting a huge amount of heat going onto the piping system and leading to creep rupture of that piping

system yet there is no evidence of that at TMI and it predicts the same thing for some of the Fukushima accidents and we await with baited breath on some validation of that prediction, and so we may want to weigh in on the relative priority assigned to these kinds of things.

I mean here we have a report, people have a done a lot, a lot of very nice work, but they have done so under terribly constrained financial and resource time conditions and they have spotted many things that they would really love to chase down and yet here is an accident sequence that turns everything into a bypass accident, which since the time of the reactor safety study we have known that that's the severe accident we wouldn't want to see.

So I think we may want to offer an a hand to help the management in its prioritization in that particular area.

CHAIR REMPE: Dick?

MEMBER SKILLMAN: Thank you. A couple things, I was impressed at the discussion about the potential loss of corporate memory for tribal knowledge and I think that needs to be addressed.

You have two gentlemen who are very

familiar with the calculator and with the peculiarities of this specific scenario and it would be important not to lose that.

So if that takes the request for more resources or some emphasis in research to capture that it seems that that would be a very good investment by the agency.

The second thing is I think the guidance document for your calculator is very important and somehow that needs to be codified, and whether it's codified in this version or whether it is codified as some kind of a White Paper, but let's make sure that how to use that calculator and any updates are captured somewhere as you --

CHAIR REMPE: Okay, so let's --

(Simultaneous speaking)

MEMBER SKILLMAN: Excuse me.

CHAIR REMPE: Well, okay, we were told they have a regulatory information letter that was issued along with a user's guide that's been issued and available in ADAMS, the user's guide, and so what else would you want?

MALE PARTICIPANT: Yes.

CHAIR REMPE: They are going to use it and

have tables on --

MEMBER SKILLMAN: What I heard is that updates are continuing or need to be continued and I am thinking about this loss of corporate memory and what might be lost if those updates are not captured.

CHAIR REMPE: Well, again, I think I have heard that you are going to apply it with the one employee and his consultant that know how to use it and come up with tables that are easier for others to use.

So I fear that we are going to lose that calculator basically, and, you know, it is what's going to happen here and I think that's what the agency is doing. Am I --

MR. COYNE: It's a spot on comment. So the user guide and the description of the calculator is in ADAMS and publically available. To be clear, the RIL was a draft, so that still needs to be finalized and issued.

CHAIR REMPE: Oh, okay.

MR. COYNE: So that hasn't been done yet, and so that would be a means that we would provide this to the other offices for use, so that is a definite thing that we need to get done and I very much appreciate the comment and it is something that we are very worried

about.

We had a panic attack when Selim couldn't run it on his computer. So we got through that, but we do need to resolve that issue.

MEMBER SKILLMAN: Those were two of mine, so I've got one more.

CHAIR REMPE: Thanks. Yes?

MEMBER SKILLMAN: I was impressed when you made the point that for as much steam generator tube damage data that we have we probably don't have enough and I know that outage after outage there is a ton of information coming out of BWRs, so if there is a way to capture that either through the IMPO databases or other databases to more expand what would become the database that might make some of these calculations more certain that would be valuable, and I say that knowing how very difficult it is to drag that information out and make it useful.

But the real point was we only have limited set and we know there is information available, if somehow we could capture it we can somehow improve the quality of this, and it seems like that would be a target that would be worth going after. Thank you.

CHAIR REMPE: Okay. Bill?

DR. SHACK: I think the NUREG ought to go out as quickly as possible, again, like other people. You know, it's important to get all this documented, so I am only half kidding about Appendix M for the Boyd paper.

I mean, you know, you really need to have that someplace where people can find it before it gets lost. My biggest concern is that even with the NUREG out the licensees are sort of left in a middle ground.

They don't know whether this is a problem or not, you know, this is an example result, you know. So the Westinghouse guys, maybe if he's got a 4.5 clearance feels pretty good, but, again, you know, am I one of the good guys or am I problem case, and the same with the CE people.

It's that uncertainty that -- You know, John is a little worried that somebody is going to come to a premature conclusion. I am a little worried he just can't figure out where he is and he doesn't know what to do and I am not sure what guidance you are going to give him to tell him what to do.

And that's sort of my biggest hangup at the moment, but I don't see any way to really resolve that. I mean it would be nice to do more calculations and,

you know, you could come up with some, as a result of the calculations you could come up with more specific guidance, but, you know, that doesn't seem likely to happen.

But it is important to capture what we have done and, you know, it's an important piece of work I think.

MEMBER SKILLMAN: Joy, may I have my nickel back just for a second?

CHAIR REMPE: Yes, sir.

MEMBER SKILLMAN: Two things, publish now and no Subcommittee meeting.

CHAIR REMPE: Okay, yes.

MEMBER SKILLMAN: Thank you.

CHAIR REMPE: Okay. So I get to close and I concur, I don't think we need another Subcommittee meeting, but I would really like to emphasize how much I would like to see additional, I know we're limited because of calculations, but some clues of what the guidance document is going to look like and insight so that -- It just seems like that the report is kind of hanging up a little bit and I know you can't do the ultimate guidance document, but just the conclusion section, just some insights about the importance of

geometry, perhaps the need for plant-specific calculations, but just, you know, hone in on the conclusions you can hone in on based on what's in the technical document.

With respect to going to Full Committee I think we would like to see the updated version, that it is cleaned up before we do that. Is that your --

MEMBER STETKAR: Absolutely.

CHAIR REMPE: Yes.

MEMBER STETKAR: I mean the Full Committee should not say issue it before we have it.

CHAIR REMPE: Yes, so let that. So with scheduling it then let's get it at least 30 days before and give us ample time, it's a 500-page document, with Appendix M it's going to probably be longer or something, or nearly 500.

But, anyway, so work with Chris and we'll figure out a time, but thank you, again, for your efforts today, the presentations were helpful.

DR. AZARM: Thank you.

CHAIR REMPE: Oh, and with that I get to close the last Subcommittee meeting of the ACRS for 2016.

(Whereupon, the above-entitled matter

went off the record at 4:35 p.m.)

A Probabilistic Risk Assessment of Consequential SGTR (C-SGTR) for a Westinghouse and a Combustion Engineering Plants

With Thermally-Treated Alloy 600 and 690 Steam Generator Tubes

U.S. NRC/RES, IESS presentation to ACRS Subcommittee

December 15, 2016

Introduction

- Last ACRS meeting on the subject was April 2015
- Since the last meeting:
  - ACRS member comments reviewed and addressed (ML16315A250)
  - Draft NUREG-2195 processed and issued for public comment (ML16134A029) – May 2016
  - Public comments reviewed and addressed (ML16315A251)
  - NUREG-2195 revised (ML16315A253)

Project outline

- Project started in response to NRR-2010-005 User need/work request
- Involved work scope by 3 RES divisions including 4 branches
- T&H and structure/materials related studies were mostly done in-house; PRA work was contracted out
- During its current work period of 6 years, the project competed for resources with other projects, including Fukushima-related ones.

Outline of today's presentation

- Presentation contains the following 3 sections:
  - T&H analyses supporting PRA and also independently assessing severe accident sequence development (RES/DSA)
  - Structural analysis work for assessing failures of "other" (than SG tubes) RCS components (RES/DE)
  - PRA assessment (RES/DRA and IESS) – including SG tube flaw estimates
- Each section will
  - provide a short background for the benefit of new ACRS members
  - discuss comments since April 2015 and responses to technical comments

Severe Accident-Induced Steam Generator Tube Rupture (SGTR)

Thermal Hydraulic Overview of CSGTR

Michael Salay
Christopher Boyd
NRC – Office of Nuclear Regulatory Research

Consequential Steam Generator Tube Rupture (C-SGTR) Subcommittee Briefing

December 15, 2016

Topics

- CSGTR Scenario Description
- TH analyses
- Method (CFD & System Code)
- Experimental Basis
- Differences Between CE and Westinghouse Plants

The Station Blackout

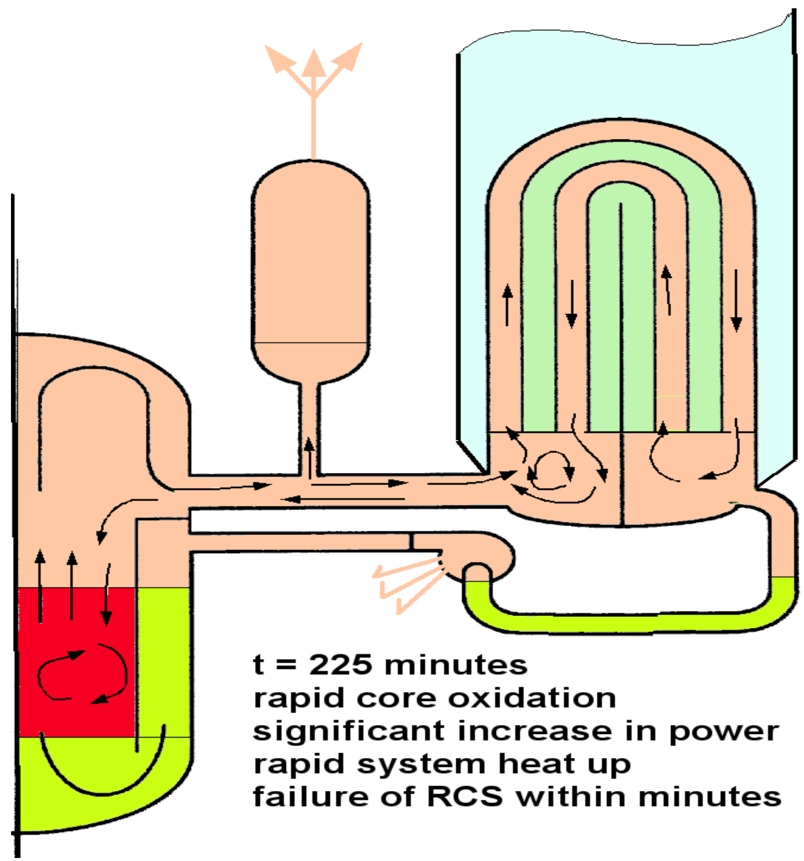
- A low probability station blackout event with immediate or subsequent loss of feed water to the steam generators.
- Reactor inventory boils off resulting in fuel damage and high temperature and high pressure conditions within RCS.
- Failure of the RCS boundary is induced by these conditions.

- If SG tubes fail first, then a flow path is created that bypasses the containment
- Failures of other RCS components (hot leg or surge line), RCS blow down into the containment
- Determining SG tubes failure is important in consequence analysis

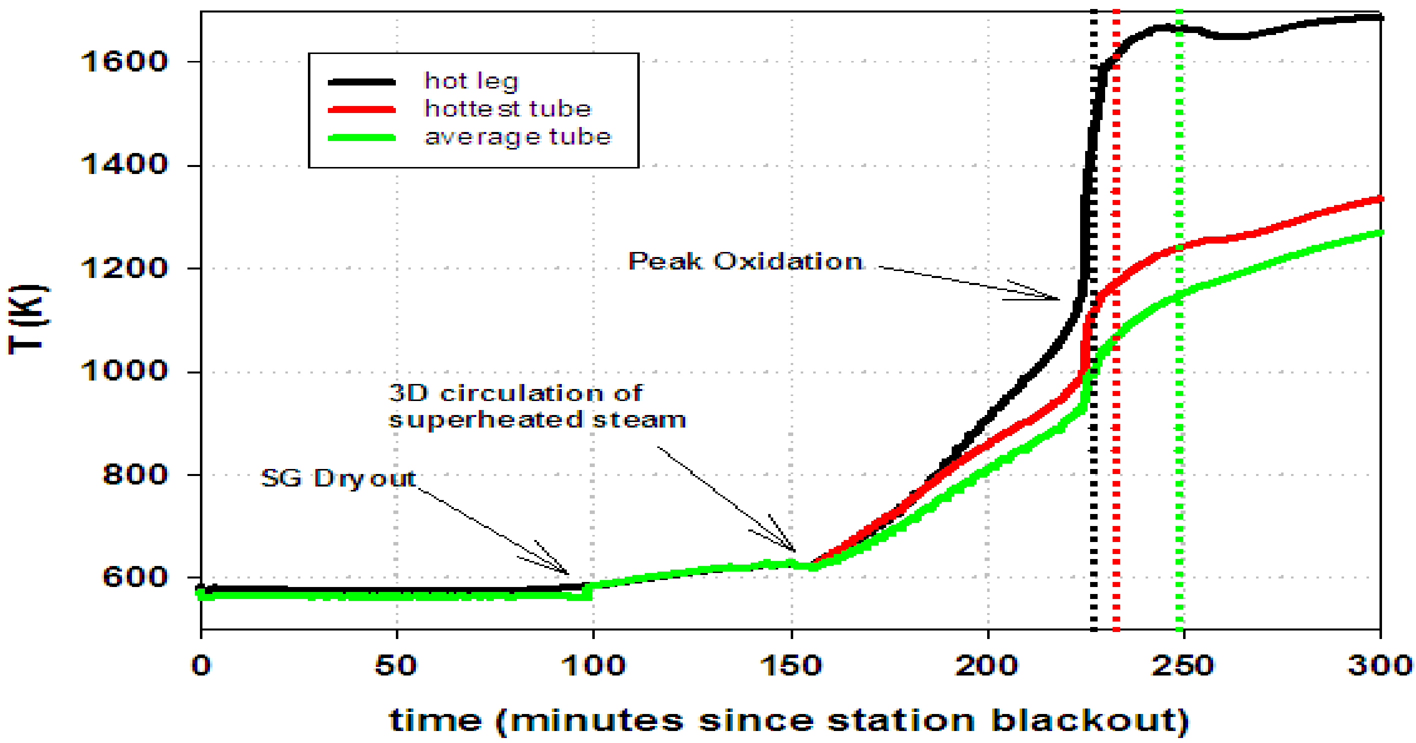
A Fast Scenario RCS failure within 4 hours

- loss of offsite power, failure of diesels, and failure of auxiliary feedwater systems
- primary inventory lost through reactor coolant pump seals. Secondary side boils off
- secondary side dry, primary inventory lost through safety valve cycling and pump seals
- loop natural circulation stops as primary inventory falls in SG tubes.
- natural circulation of superheated steam begins as inventory falls below hot leg. Core and system heat up.
- Core uncovers, core oxidizes and produces significant power, system heat up accelerates and induced failure is predicted for RCS components.

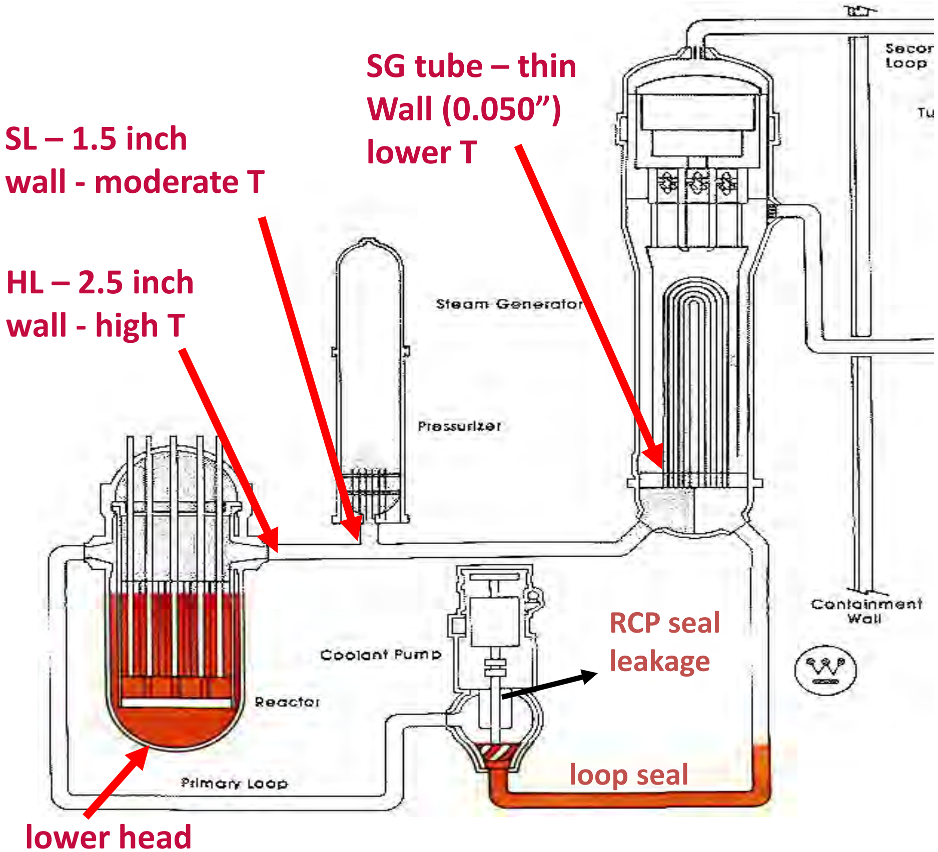
- More likely scenarios involve some auxiliary feedwater or operator actions that significantly delay the failure time.



RCS Structure Temperatures –Fast Scenario



RCS Points of Interest



Considerations

~ T=1475 K, start of rapid Zirc Oxidation

~ T=1725 K, Melting Stainless and Inconel

~ T=2030 K, Melting of Zircaloy-4

~ T>1175 K, tubes fail at system Pressure

- Rapid temperature rise and pressure difference leads to induced failure.
  - failure location affects consequences
- SG tube ruptures provide a path for fission products to bypass containment.
- Wall thickness indicative of thermal response times

High-Dry-Low

Primary Side

<u>High</u> Pressure

* no significant leakage to reduce pressure

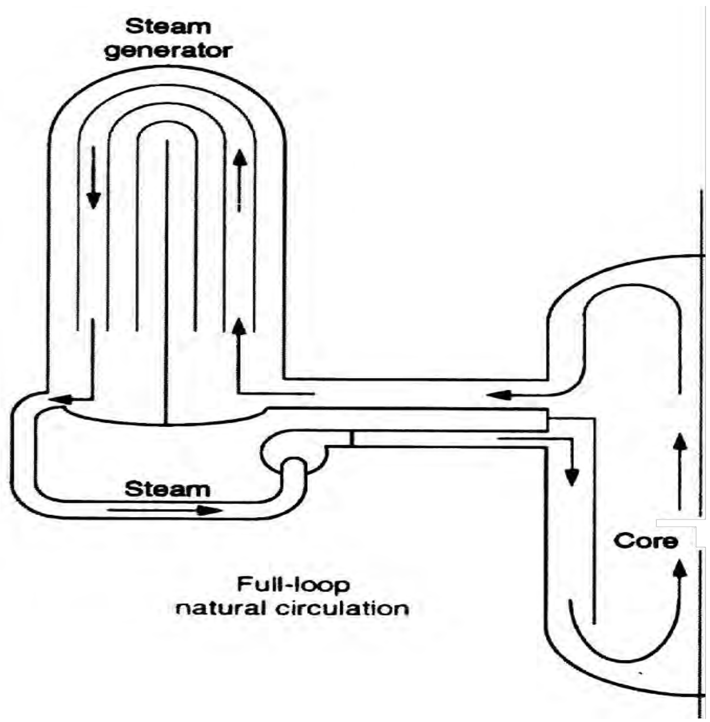
SG tube wall

Secondary Side

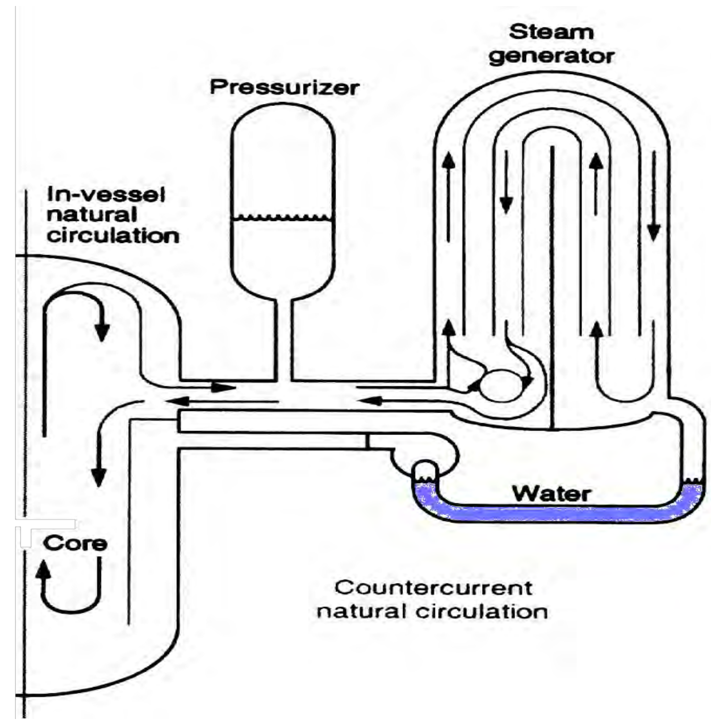
<u>Dry</u>
* Loss of water allows tubes to heat up

<u>Low</u> Pressure
* Secondary side leakage increases pressure difference (i.e. mechanical load on tube wall)

Two Flow Patterns - PWRs with U-Tube SGs



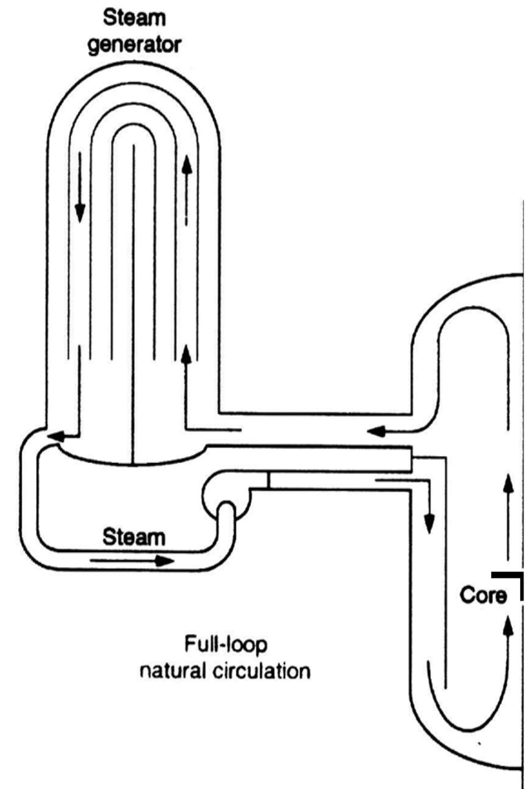
full-loop natural circulation



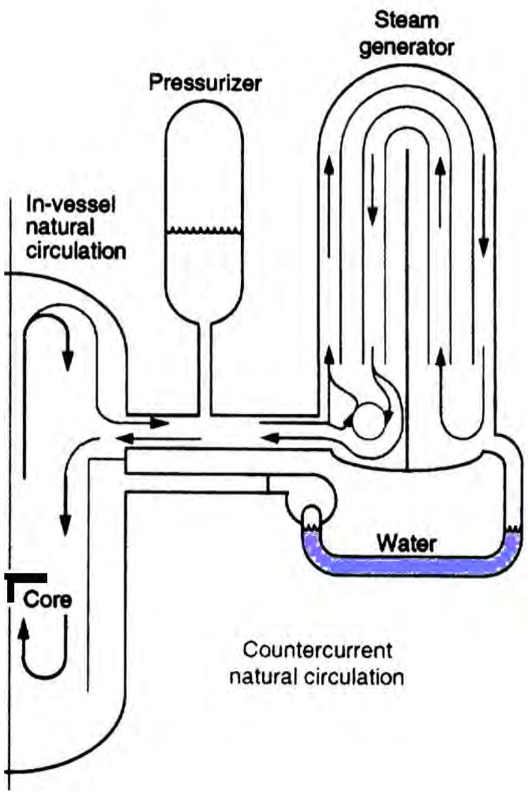
Counter-current natural circulation

Full-Loop Natural Circulation

- Water cleared from the reactor coolant pump loop seal (and lower downcomer).
- Loop seal clearing is affected by:
  - depth of the pump loop seal and water temperature
  - reactor coolant pump seal leakage rate and elevation
  - primary side depressurization rates
  - downcomer bypass flows
- Westinghouse PWR studies have indicated that loop seals are more likely to remain blocked with water.
- Careful modeling and benchmarking is important to build confidence in predictions of loop seal clearing.
- Full loop circulation reduces mixing of the hot gasses that enter the SG tube bundle. A severe thermal challenge.
- System analysis tools such as MELCOR or SCDAP/RELAP5 are used to predict the system flows and heat transfer.



Counter-Current Natural Circulation

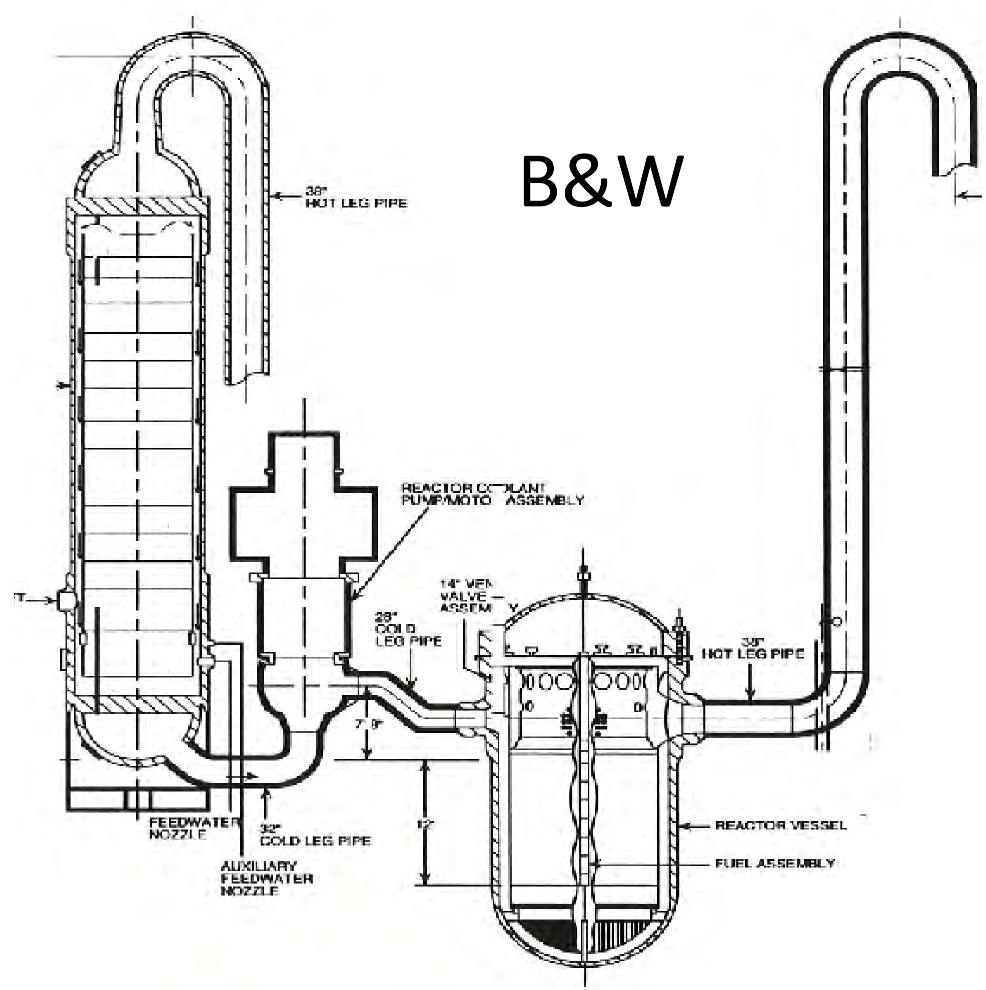


- With the pump loop seal filled with water, a counter-current flow field is established.
  - This flow pattern mixes the hot gases with cooler flows returning from the SG. The thermal challenge to the tubes is reduced but not eliminated.
- System code models require external information to ensure consistency:
  - hot leg flows, mixing, and heat transfer
  - inlet plenum mixing and entrainment
  - pressurizer surge line mixing
  - SG tube bundle flows, temperatures, and distribution
- System codes account for the overall response but are not designed to explicitly predict the three dimensional mixing and entrainment.
  - MELCOR and SCDAP/R5 models are adjusted to ensure consistency with experiments and/or CFD predictions

What about B&W Plants?

- Vigorous natural circulation flows are not expected due to the elevations and design of the hot legs and steam generators.

- These plants have not been part of the recent severe accident induced failure studies.

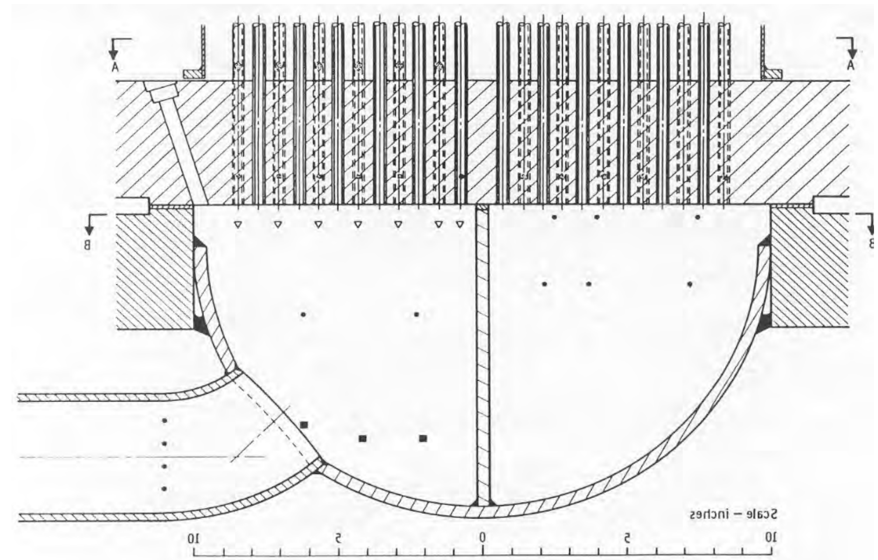


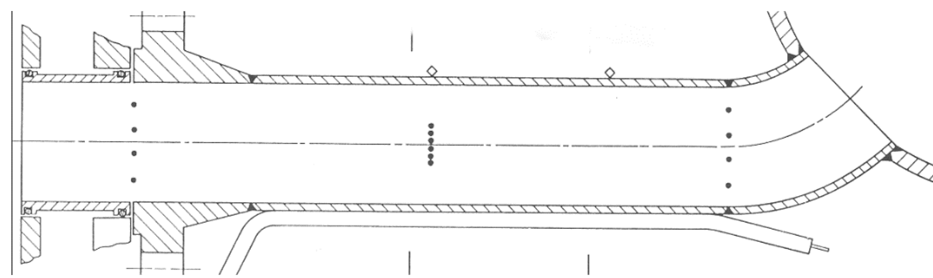
TH Analyses

- Westinghouse TH analyses performed for the Steam Generator Action Plan (SGAP)
  - Documented in NUREG/CR-6995
  - TH analyses for Combustion Engineering (CE) plants did not receive the same level of attention
- Westinghouse and Combustion Engineering TH analyses used for current work
  - TH analyses conducted with CE under CSGTR project
- Use system code and CFD code
  - CFD predicts spatial flow and temperature distributions
  - System code predicts transient behavior
    - Uses CFD results for modeling
    - Results can be combined with those of CFD to obtain a transient spatial temperature distribution
- CFD Validated against Westinghouse 1/7th scale experiments

Westinghouse 1/7th scale tests

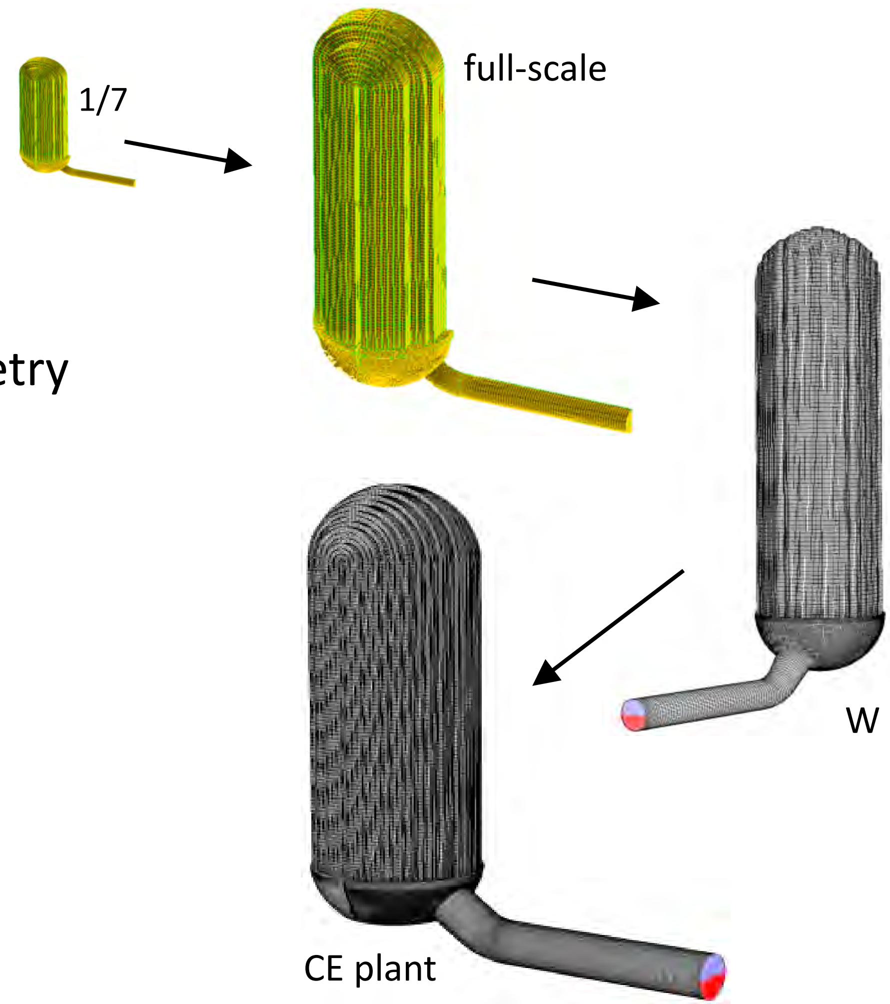
- Demonstrated the counter-current flow path
- Not focused on tube integrity but provide valuable insights
- Many scaling studies demonstrate applicability to full-scale
- Results helped inform modifications made to system codes (SCDAP/RELAP5 or MELCOR) used to study the station blackout scenarios.
- Around 2001, CFD was used to study these tests



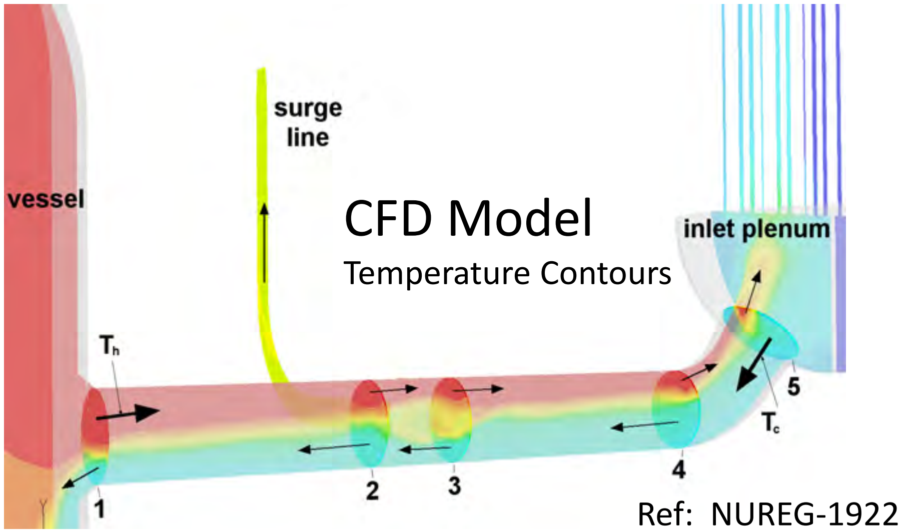


CFD Developments

- Benchmark at 1/7th scale
- Scale-up to full-scale conditions
  - Using test facility geometry
- Prototypical W. Model 44 SG Geometry
  - Compare to test facility
- Sensitivity studies
  - Heat transfer
  - Surge Line orientation
  - Hydrogen Content
  - Tube Leakage rates
- Combustion Engineering Design

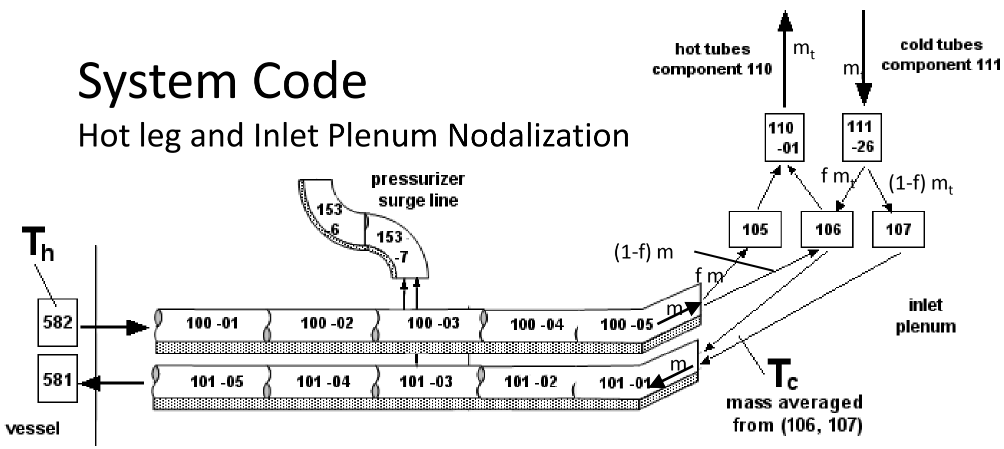


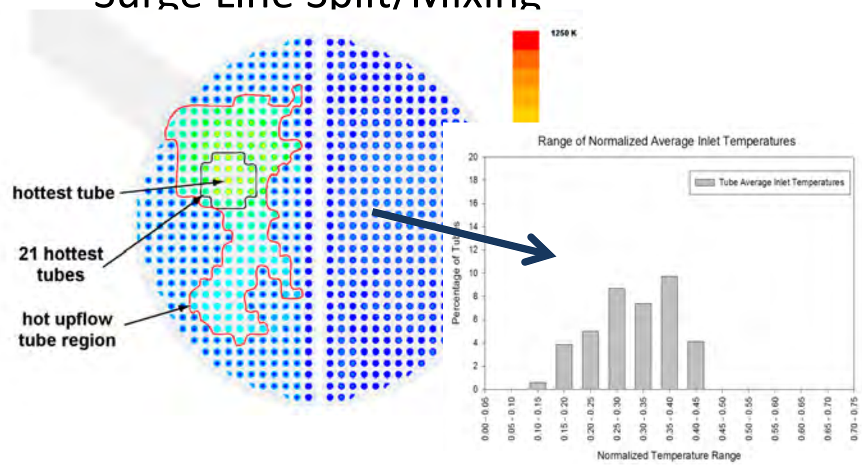
CFD Support Modeling



Ref: NUREG-1922

- Hot Leg Flow Rate - $\mathbf{C_d}$
- Inlet Plenum Mixing - **f**
- SG Tube Bundle Flow and T
  - Hot tube fraction
  - recirc ratio - **r** = m_t / m
- Distribution of Temperatures
  - $\mathbf{T_m}$ - Normalized T
- Surge Line Split/Mixing

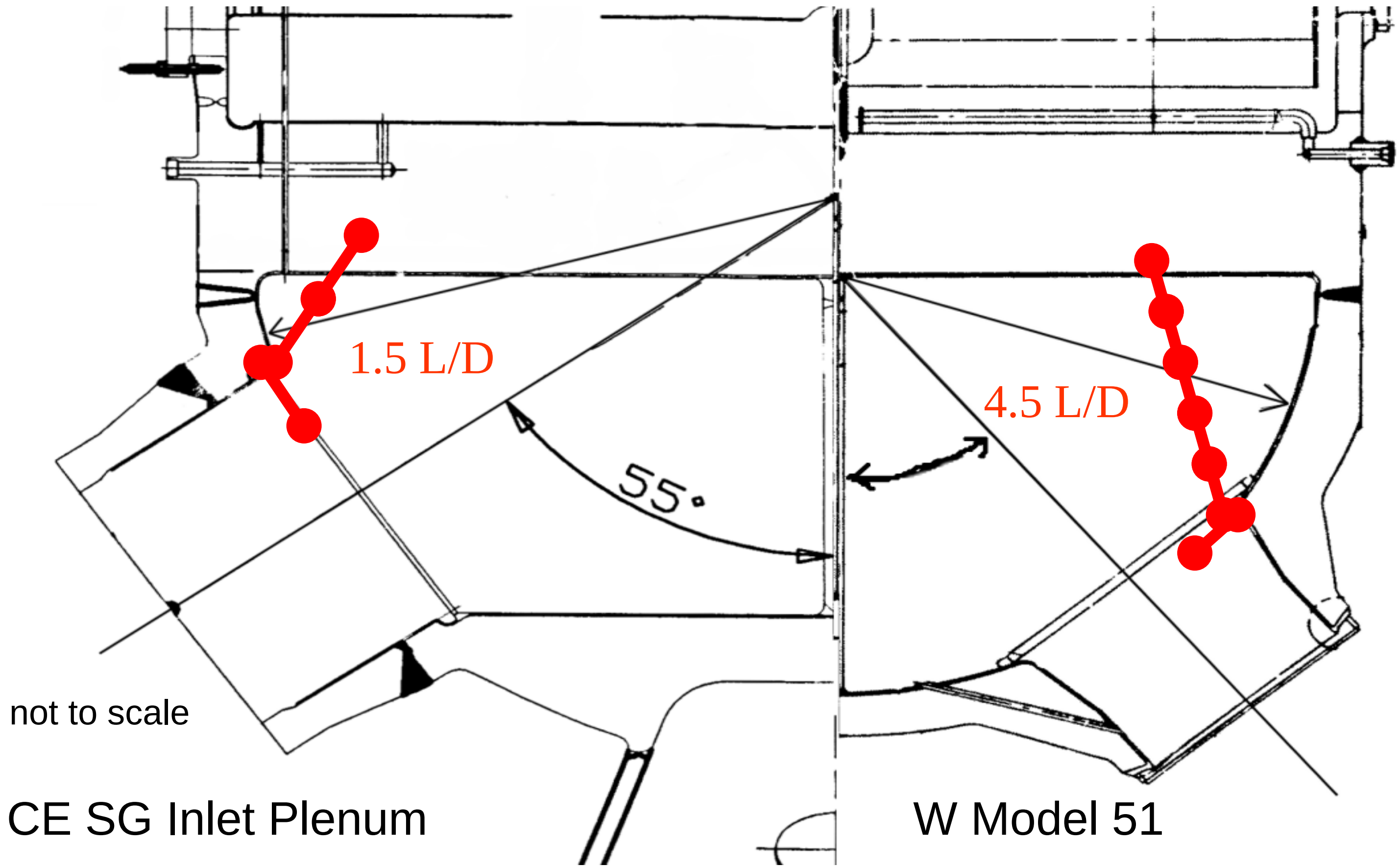




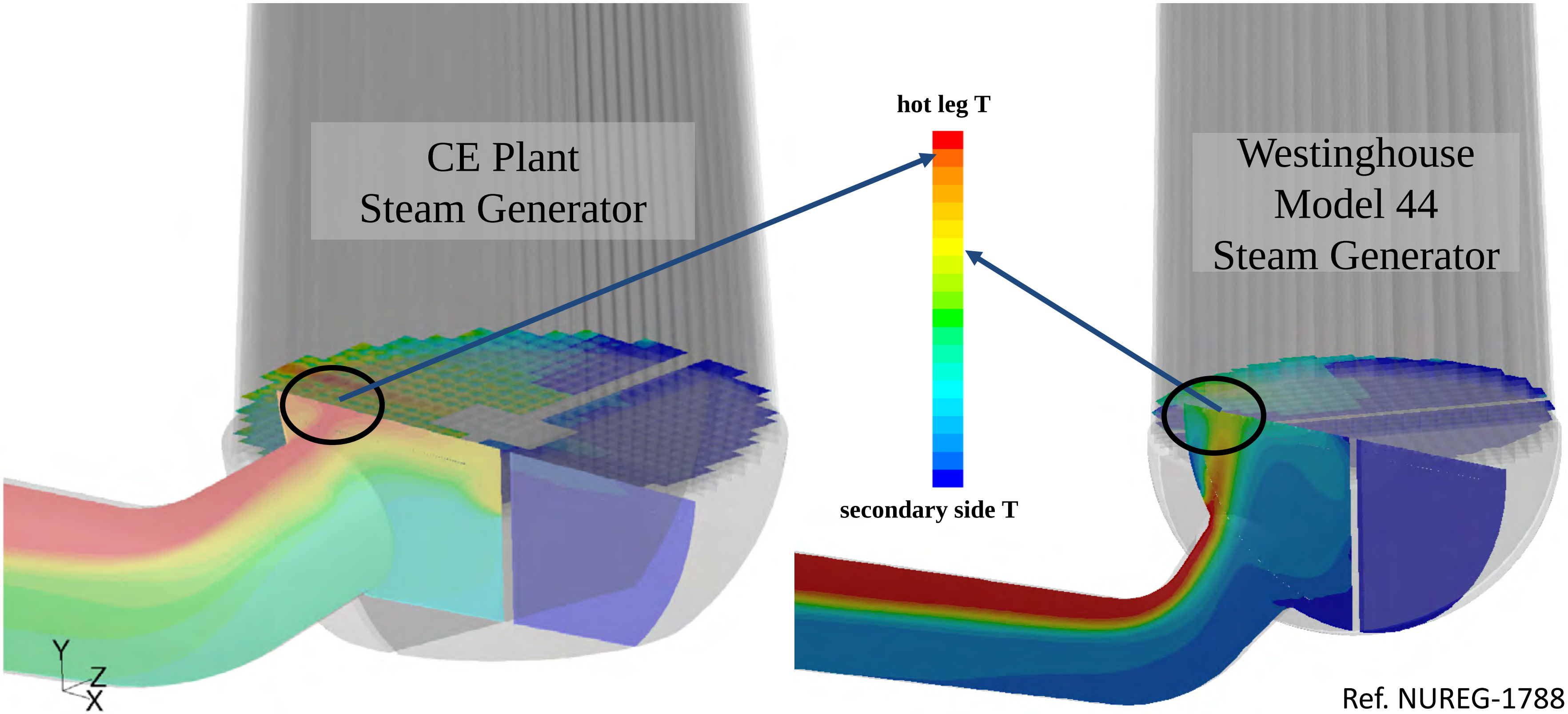
CE SGTR Behavior Differs from Westinghouse Plants

- Less mixing of hot gases before reaching SG tube inlets
  - Lower hot leg Length/Diameter ratio
  - Some CE plants have shallower inlet plena
- In CE SG tubes are exposed to similar gas temperatures as hot legs
- Under certain conditions unflawed tubes could rupture before hot legs
- Unlike for the rupture of a flawed tube, multiple unflawed tubes could potentially reach the failure condition nearly simultaneously resulting in a rupture large enough to depressurize the RCS sufficiently fast to prevent failure of other RCS components.

The CE inlet plenum (compared to W model 51)



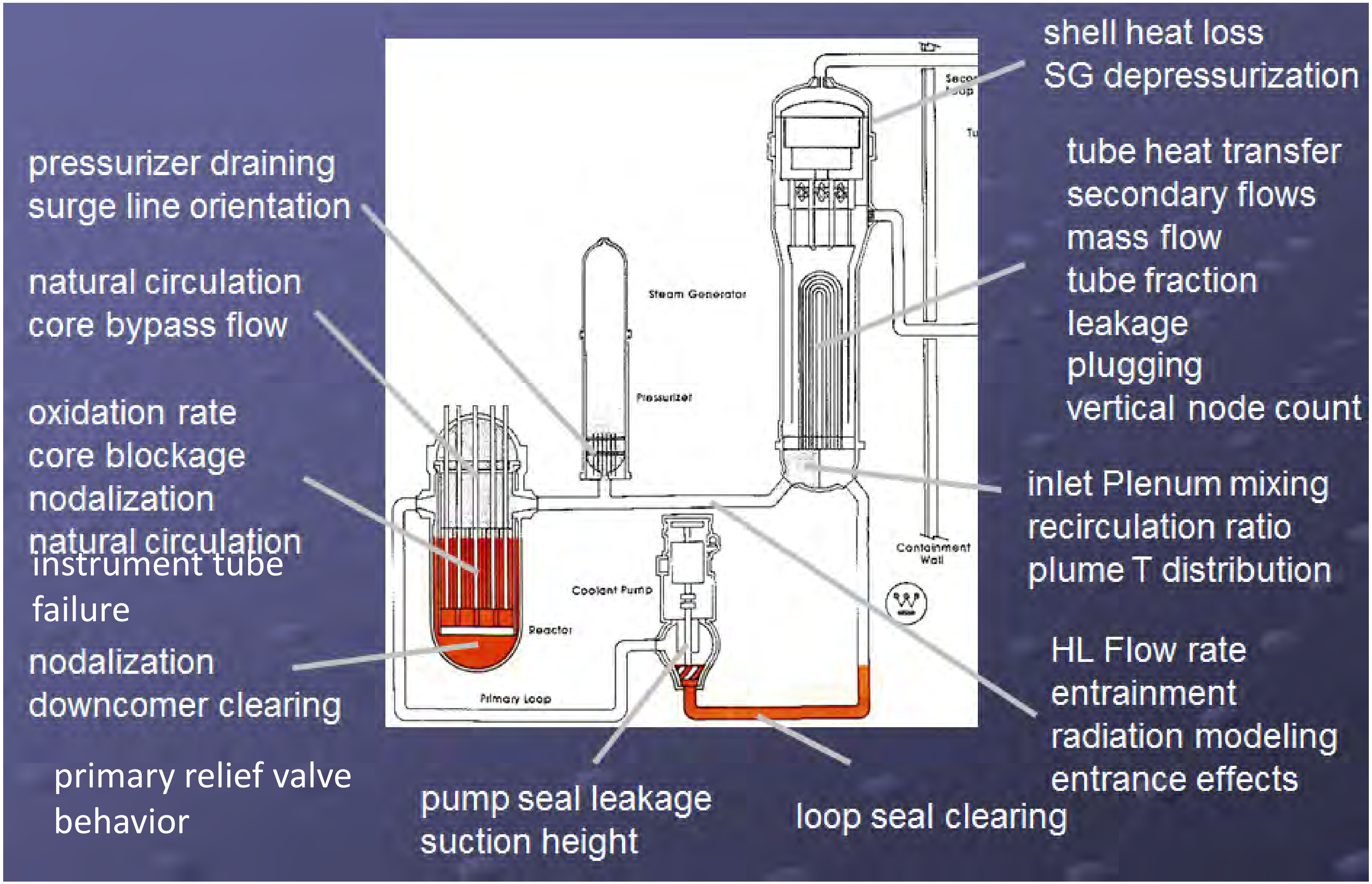
CFD Predictions - Westinghouse and CE (hottest tube region circled)



Ref. NUREG-1788

(temperature contours on vertical centerline plane of hot leg)

Other System Code Considerations



Severe Accident-Induced Steam Generator Tube Rupture (SGTR)

Thermal Hydraulic Analysis and Responses to ACRS and Public comments

Michael Salay
NRC – Office of Nuclear Regulatory Research

Consequential Steam Generator Tube Rupture (C-SGTR) Subcommittee Briefing

December 15, 2016

Objective

- Provide TH for CE analysis to be used with calculator
  - CSGTR Calculator
  - Finite Element calculations
- Provide scoping failure calculations
- Provide FP releases

CE TH Calculations (1/4)

- An individual code cannot practically calculate all relevant TH behavior for thermally induced CSGTR
  - System codes can calculate transient behavior but not transport of heat and material in complex SG flows
  - CFD codes can calculate the complex flows but not practically calculate transient behavior
  - Must use and integrate results of both codes
- Test data limited in scale

CE TH Calculations (2/4): Codes used

- SNL generated Combustion Engineering deck
  - Based on previous RELAP/SCDAP and MELCOR CE decks
  - Model approach derived from NUREG/CR-6995
- NRC-calculated CFD
  - Similar approach for CE plant as that in NUREG-1922 (for W plant)
  - CE CFD model documented in: Boyd, C., "CFD modeling of Severe Accident Natural Circulation Flows in a Combustion Engineering Pressurized-Water Reactor Loop," International Topical Meeting on Advances in Thermal Hydraulics 2016, New Orleans, LA, June 2016

CE TH Calculations (3/4): TH code integration

- Used general TH code integration method applied for decades to CGTR
  - Combination of CFD and system code
  - Methods documented in NUREG-1922, and NUREG/CR-6995
  - CFD methods validated against Westinghouse 1/7th scale tests
- CFD provides
  - Target flow parameters for system code
  - Spatial temperature distribution in tubes
- System code (MELCOR)
  - Modeled to match CFD flow parameters
  - Provides overall transient behavior
  - Time-evolution of CFD-calculated spatial temperature distribution

CE TH calculations (4/4)

- Short term and long term station blackouts
  - Timing of auxiliary feedwater failure
  - Similar, but time shifted, behavior
- Secondary-side relief valve opening
  - Immediate (either per operator action or failure)
  - Different secondary valve failure stick-open models
- Two sets of calculations
  - Scoping calculations that included modeling of tube and component failure
  - Calculations with component failure modeling suppressed for use as input into CSGTR calculator and FE calculations

Comments by ACRS and public

- Impact of changing codes
- Impact of RCP seal leakage
- Loop seal clearing
- Uncertainties in TH analyses
- Others

Impact of changing codes (1/2)

- Both ACRS and public provided comments regarding the impact of changing codes
- Code to use was a major choice at beginning of project
  - RELAP/SCDAP
    - Used for Steam Generator Action Plan (SGAP) work
      - Easy to directly compare results
    - Hot tube modeling and application of CFD already developed
  - MELCOR
    - NRC Severe Accident code
    - Calculate transport and release of fission products

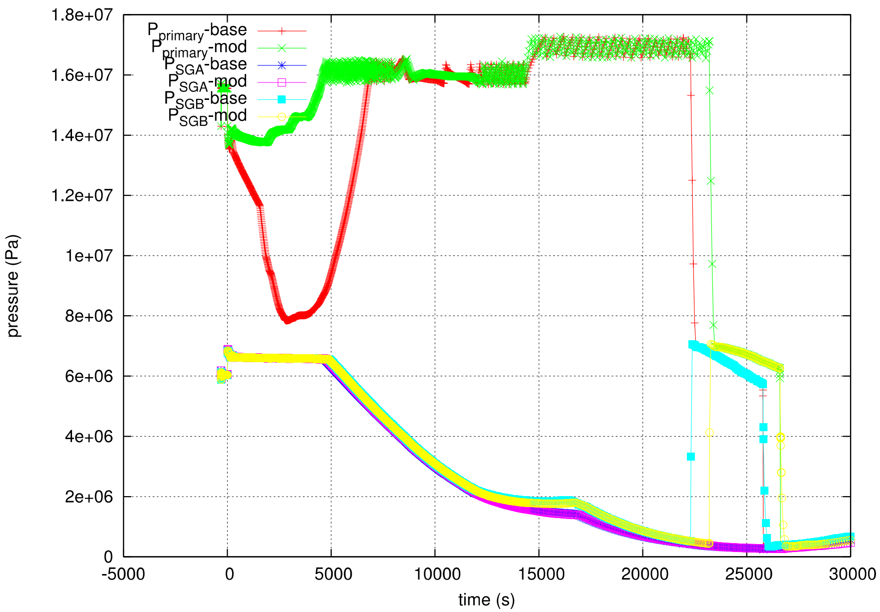
Impact of changing codes (2/2)

- Previous comparisons between codes indicated that MAAP, MELCOR, and RELAP obtain similar results for closed-loop-seal natural circulation
- Decided on MELCOR
  - Changing codes a significant concern
    - Simultaneously changing both plant and code
    - Deck development process involved comparison between MELCOR and equivalent RELAP CE deck
  - Similar sequence and timing obtained with both codes
  - Comparison of results documented in Chapter 4, Comparison to SCDAP/RELAP5, in Sandia Report: D. Louie, et al., *A MELCOR Model of the Calvert Cliffs Two-Loop Pressurized Water Reactor and Containment for the Steam Generator Tube Rupture Scenarios*, Sandia National Laboratories, October 2012

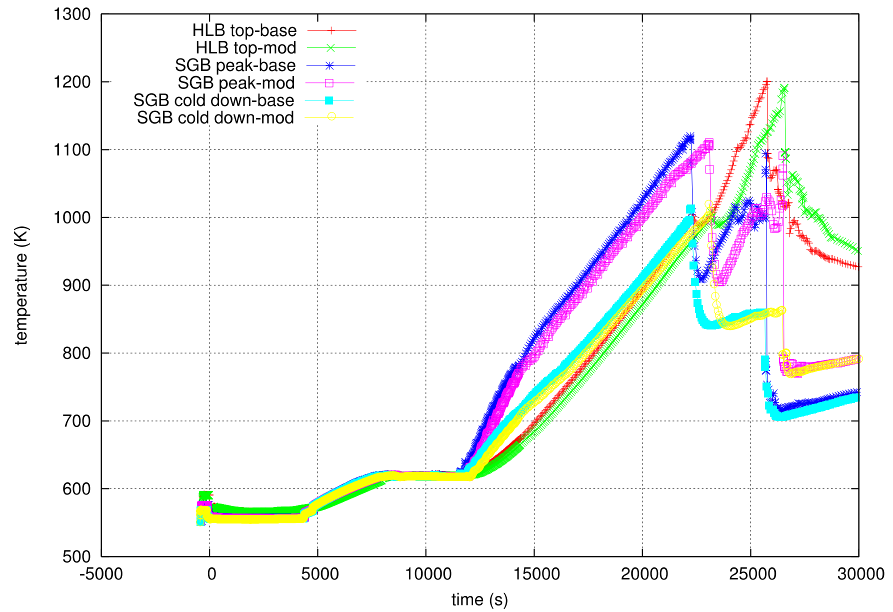
Impact of RCP seal leakage (1/2)

- Public comments regarding RCP seal leakage
  - Used 21 gpm to be consistent with SGAP
  - Public comments indicated less seal leakage for CE
    - Comments included calculation with less of a primary pressure drop
  - Reran base case with no RCP seal leakage
    - Similar pressure drop to public comment calculation
    - Delay in absolute, ***but not relative***, failure timing.
      - As with most TH issues, results in time shift, but not qualitative difference

Impact of RCP seal leakage (2/2)



Effect on system pressures



Effect on Loop B structure temperatures

Loop seal clearing

- Several comments received on loop seal clearing
  - Studied extensively for Westinghouse for SGAP and several mechanisms studied and documented in NUREG/CR-6995
  - Initial scoping work for CE built upon the SGAP analyses
  - Issue not explored fully for CE analysis
    - Loop seal clearing is important for Westinghouse plants because this clearing exposes SG tubes to gases nearly as hot as those in the hot leg
    - For the CE geometry studied, gases entering SG tube bundle are nearly as hot as the gases in the hot leg
      - Loop seal clearing not nearly as important

Other comments

- Relative decay and oxidation powers
  - Comments regarding hydrogen behavior and relative decay and oxidation powers
  - Hydrogen generated when it appeared, not predicted to be held up in system
  - Significant oxidation power only during hydrogen generation
- Oxidation of steel
  - Steel oxidation in RCS components typically not modeled in severe accident analyses
    - MELCOR models steel oxidation in core
    - CSGTR analyses did not consider steel oxidation
    - Reviewed reaction rates
      - Steel oxidation in RCS does not appear to be a major effect

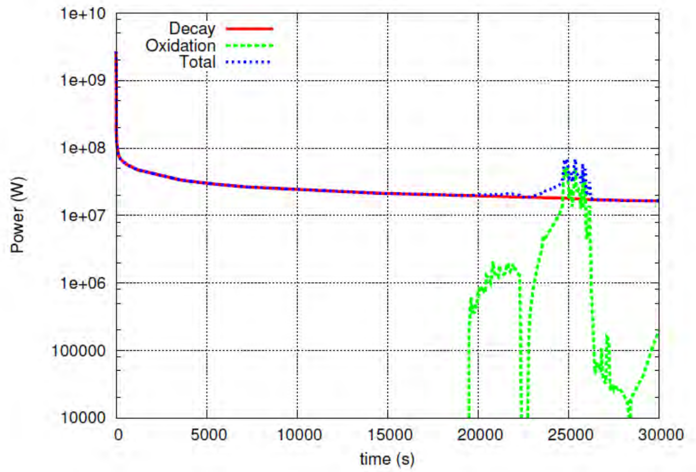


Fig. 8: stsbo decay and oxidation power contribution

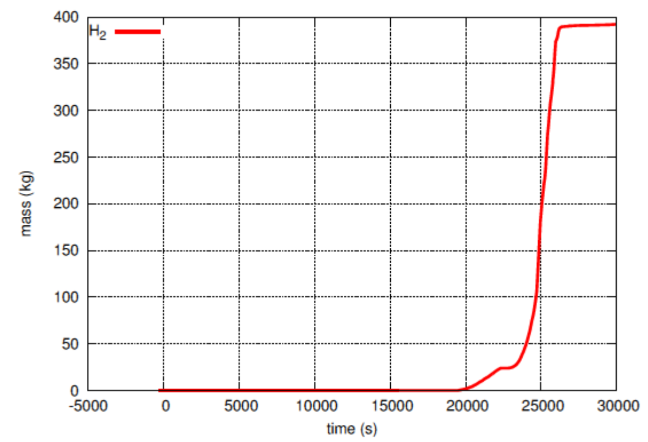


Fig. 7: total hydrogen generation for the stsbo calculation

TH uncertainty (1/2)

- Question about uncertainty due to variation in TH raised in comments
- The impact of uncertainties in TH considered upon initial deck creation
- Performed TH uncertainty analysis on early stsbo model
  - Sampled TH parameters and observed effect on predicted absolute component failure timing and relative SG-tube-to-RCS-component failure timing
  - TH uncertainty analysis parameters chosen based on those in NUREG/CR-6285 and NUREG/CR-6995:
    - PORV and SRV Valve discharge coefficients
    - Zirconium oxidation sensitivity coefficients
    - CFD mixing parameter: coefficient of discharge
    - CFD mixing parameter: recirculation ratio
    - Steam generator tube outer wall heat transfer coefficient multiplier
    - Hot leg wall emissivity
    - RCS-to-Containment heat transfer coefficient multiplier

TH uncertainty (2/2)

- Distribution of failure timings resulting from TH variation uncertainty analysis had standard deviations of approximately:
  - $\pm$420 s (7 min) – relative SG-to-RCS component failure timing
  - $\pm$600 s (10 min) – absolute failure timing
- Uncertainty analysis documented Chapter 5, Uncertainty Analysis, in Sandia Report: Louie, D.L., et al., “A MELCOR Model of the Calvert Cliffs Two Loop Pressurized Water Reactor and Containment for the Steam Generator Tube Rupture Scenarios,” Sandia National Laboratories, October 2012

Possible future TH work

- Interesting but deferred work because of resource limitations
  - More detailed spatial temperature distribution
  - Loop seal clearing
  - Water hold up in SG, flooding / counter-current flow
    - Water also held up in previous SGAP calculations
  - Detailed evaluation of FP release
    - Current focus on TH input, not FP release
      - Didn't rerun cases to solely extract FP release behavior

Conclusions

- MELCOR calculations for a CE plant with replacement SGs provide input to CSGTR calculator and finite-element component failure analysis
- Most effects shift timing of temperature increase curves
  - Temperature rise rates affected to some extent
- Relative temperature increase rates and relative component failure timing between SG tubes and other components more important for releases than absolute failure time
- Some work was deferred because of limited resources
  - Benefit determined to not be worth the expense for the project
- Received and incorporated useful feedback from ACRS and public

Failure Behavior of RCS Components

Raj Iyengar, RES/DE/CIB

presentation to
ACRS Subcommittee

December 15, 2016

Failure Behavior of RCS Components

- Identify, characterize, and model relevant RCS nozzles to assess their potential for failure during a severe accident for Westinghouse

- Evaluate adequacy of simplified C-SGTR Calculator failure time estimates

Failure Estimates used in the C-SGTR Calculator

H**otleg**/surge-line (EPRI-TR-107623-V1)

- Creep equation

  t_R (in hour) = $10^{[P_{LM}/(1.80*T)]-20}$

- Westinghouse hotleg

  $P_{LM} = 1000(41.31 \pm 0.48 - 5.408 \text{ Log}_{10} (\sigma_{ksi}))$

- CE and B&W hotleg

  $P_{LM} = 1000(42.02 \pm 1.09 - 8.477 \text{ Log}_{10} (\sigma_{ksi}))$

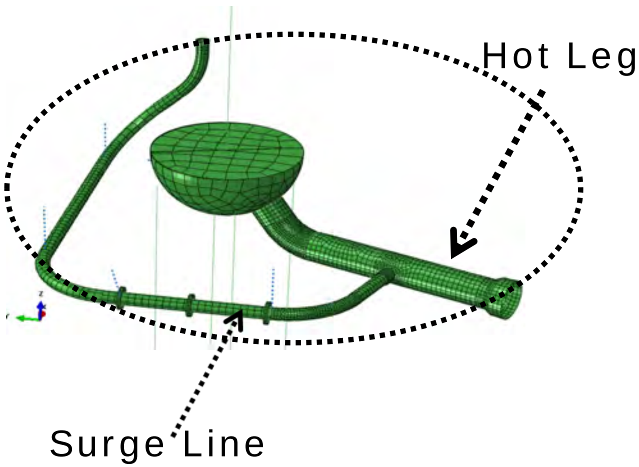
- For Surge line (SS 304)

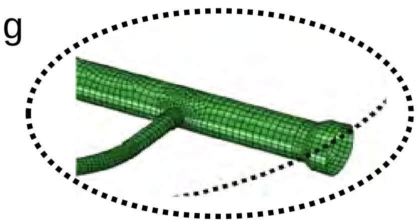
  $P_{LM} = 1000(50.42 \pm 1.25 - 0.833 (\sigma_{ksi}))$

Model Aspects

Finite Element Model

- System-level model for Westinghouse plant – Three-dimensional Shell Elements





- Sub-model of hot-leg used for additional simulations

Material Behavior

- Total strain = elastic + plastic + creep
- Creep Law – time and rate-dependent
- Plasticity Law – rate-independent
  - piecewise-linear stress-stain input from experimental data
- High temperature material data extended by Argonne National Lab (Appendix A)

Analysis Procedure

- HL/SL structural temperatures for initial conditions (steady-state condition)

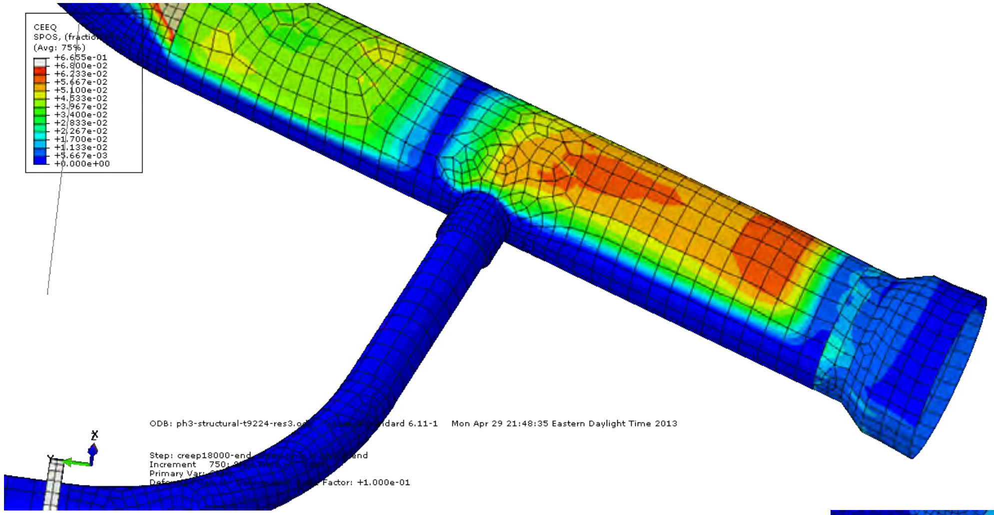
- Time-dependent gas temperatures from system code (RELAP) as a boundary condition
  - Use time-dependent heat transfer coefficient
  - Assume upper and lower temperature split

- Adjust the heat transfer coefficient spatially in the hot-leg region (based on the developing curve provided in NUREG-1922)

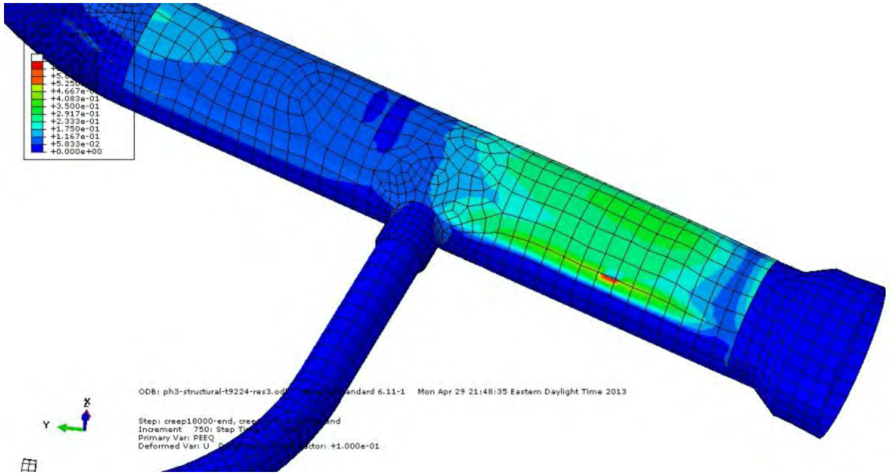
- Model heat loss to the ambience due to convection and radiation

- Run a thermal- mechanical simulation for short-term SBO

Creep and Plastic Strains



Accumulated Plastic Strain



Accumulated Creep Strain

t = 12300 seconds

Damage Prediction

Damage at any material point determined using

Larsen-Miller Parameter (P-LM)

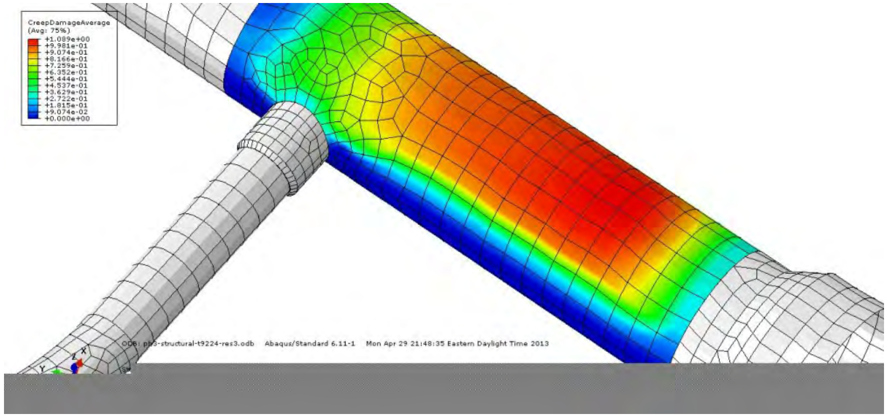
$$P\text{-}LM = A * \text{Log}10(\sigma) + B$$

σ - effective stress; T – temperature

Time to rupture

$$t_r = 10^{(P\text{-}LM/T - C)}$$

A, B, and C - constants



Failure time - 12300 seconds

Damage is averaged through thickness to determine failure time.

Failure Behavior of RCS Components

- System-level model simulations
  - computationally intensive
  - poses issues with convergence
  - Not well-suited for understanding sensitivities to input parameters

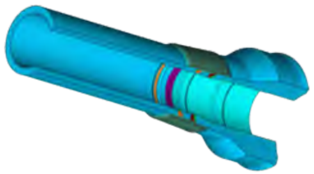
- Failure location in the hot-leg region predicted by the system model

- A sub-model of hot leg and reactor pressure vessel nozzle used for additional simulations

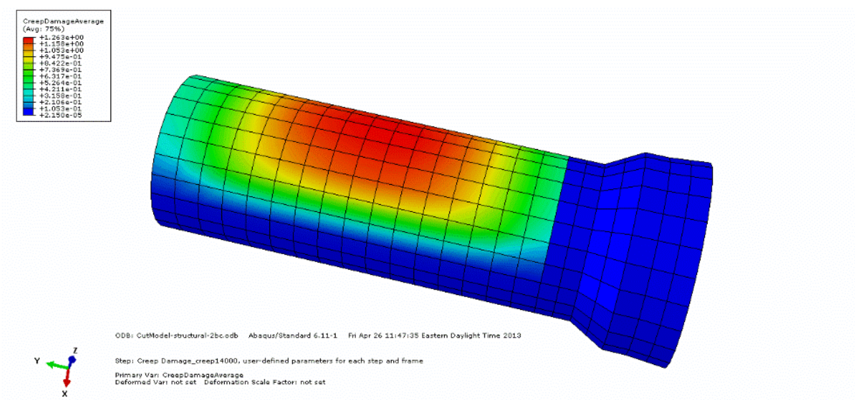
- Results of hot-leg model similar to the system model

Failure Time

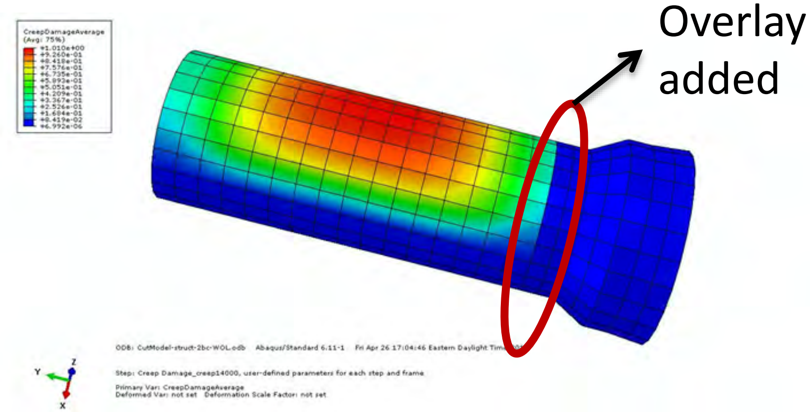
Red - Through Thickness Damage > 1
Blue – Little or No Damage



No Weld Overlay



Weld Overlay



Overlay added

t_r = 12430 secs

t_r = 12500 secs

Failure time increases by 70 seconds with weld overlay
Failure location does not change

Failure Behavior of Hot Leg

SBO with Early Failures of TDAFWs (Westinghouse)

Finite Element Model	Features	Weld Overlay	Failure Time (seconds)
System	Creep and Plasticity: Spatially Adjustment of HTC	No	12300
Hot Leg Model	Creep and Plasticity: Spatially Adjustment of HTC	No	12430
	Creep and Plasticity: Spatially Adjustment of HTC	Yes	12500
	Creep only: Spatially Adjustment of HTC	No	12140
	Creep and Plasticity: HTC not adjusted spatially	No	12560

Hot leg failure time - 12800 seconds
(5th percentile failure time estimated by CSGTR Calculator)

Failure Behavior of RCS Components

Summary

- Hot-leg model yields similar failure location and time compared with the system model (Westinghouse)
- Predicted failure time below the failure time determined by the C-SGTR calculator.
- Weld Overlay has very small influence in failure time and no influence in failure locations
- Failure mainly influenced by temperature and stress redistribution due to counter-current circulation.

Comments/Responses from Previous ACRS Briefing

14	Results in the draft NUREG list significant figures that are not supported by the analysis. Staff should go through the report and revise numbers to reflect accuracy supported by the analysis.	Ballinger	The significant figures are due to the small time steps involved in the finite-element analyses (numerical calculations) to ensure accuracy and precision of the algorithm. Our general principle is to leave the number of significant figures as is, except for reporting the final results. Otherwise, we get occasions where a reader thinks calculations using intermediate results are in error, since they may not match due to round-off. Accordingly, we have removed the significant figures in Table 4.4 (Sec 4.5) summarizing the failure times.

Public Comments/Responses

Section 5.2.1.1.1	Report notes that Argonne National Laboratory (ANL) developed a model for axial part through wall flaws. Please provide reference for the ANL contribution.	All the rupture models that we developed are related to SG tubes with cracks. See for example, NUREG/CR-6575.
Section 5.2.2.1.1	Provide reference for ANL test	See NUREG/CR-6575.
Page 5-5	Recommend that the reference literature on the data for creep rupture be expanded.	As noted in Appendix A, additional testing was conducted at Argonne National Lab through an NRC-funded effort to expand the available database of high temperature (severe accident conditions) creep properties for selected steels and weldments used in the reactor cooling system components. While more data is always better to reduce uncertainties, it is not clear if that would lead to different conclusions.
Page 5-5	Model assumes creep failure based on the 95% L-M creep rupture parameters. Would conclusion be changed if mean values were used.	Yes this would increase the failure time. But the increase will not be significant because high temperatures involved and the rate of temperature increase is quite fast.

Public Comments/Responses

A.5	DETERMINISTIC STRUCTURAL EVALUATION b) Have benchmark studies been performed on the finite element analyses (FEA) and computational fluid dynamics (CFD) tools used for the assessment? c) Section 4.2.1 of NUREG-2195 discusses surge line modeling. Please clarify, are stratification conditions taken into account in the surge line creep failure assessment? The section does not discuss this topic. d) Section 4.3 of NUREG-2195 discusses SG lower head model. Was a divider plate modeled in the FEA for the SG lower head? If not please provide justification. e) Weld overlay analysis in Section 4.4.6.1 of NUREG-2195 should account for the welding residual stresses of the weld overlay process. Are any residual stresses considered in the present analysis? f) Note that some of the PWR reactor vessel nozzle dissimilar metal welds Alloy 82/182 (susceptible PWSCC) have applied the Mechanical Stress Improvement Process (MSIP®1) to redistribute the welding residual stresses and reduce susceptibility to PWSCC. Would this have any impacts on the SGTR evaluation? g) Was PWSCC crack growth considered for Alloy 600/690 tubes? If not, please justify treatment.	b) As discussed in Section 3, a benchmark study by the NRC staff, documented in NUREG 1781, "CFD Analysis of 1/7th Scale Steam Generator Inlet Plenum Mixing during a PWR Severe Accident," demonstrates that CFD predictions can adequately predict the inlet plenum mixing observed in the one-seventh scale tests. The FEA analyses uses material models and parameters based on experiments and are performed using benchmarked commercial code. However, no experiments were performed on the components under the severe accident conditions, considered in the analyses. c) Stratification of counter flow was considered in the analysis. d) Yes, it was modeled (see Fig 4-29). e) The weld residual stresses are not considered in the analysis. Such stresses will relax due to thermal and diffusion creep, as the components experiences such high temperatures. f) The compressive stresses due to MSIP on the surface of the pipe will relax under the temperatures of interest and would not have any impact under the severe accident conditions simulated in the analyses. g) This is not relevant within the time-scale of interest for the simulations considered in this section.

Probabilistic Risk Assessment for Consequential Steam Generator Tube Rupture (C-SGTR)

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December 15, 2016

Presentation objective

- A summary of the PRA-related work
- Selected examples to illustrate the work
- Path forward in 2017

Although the draft NUREG-2195 addresses both the pressure and thermally induced C-SGTR, the summary section of this presentation will focus on the latter failure mechanism.

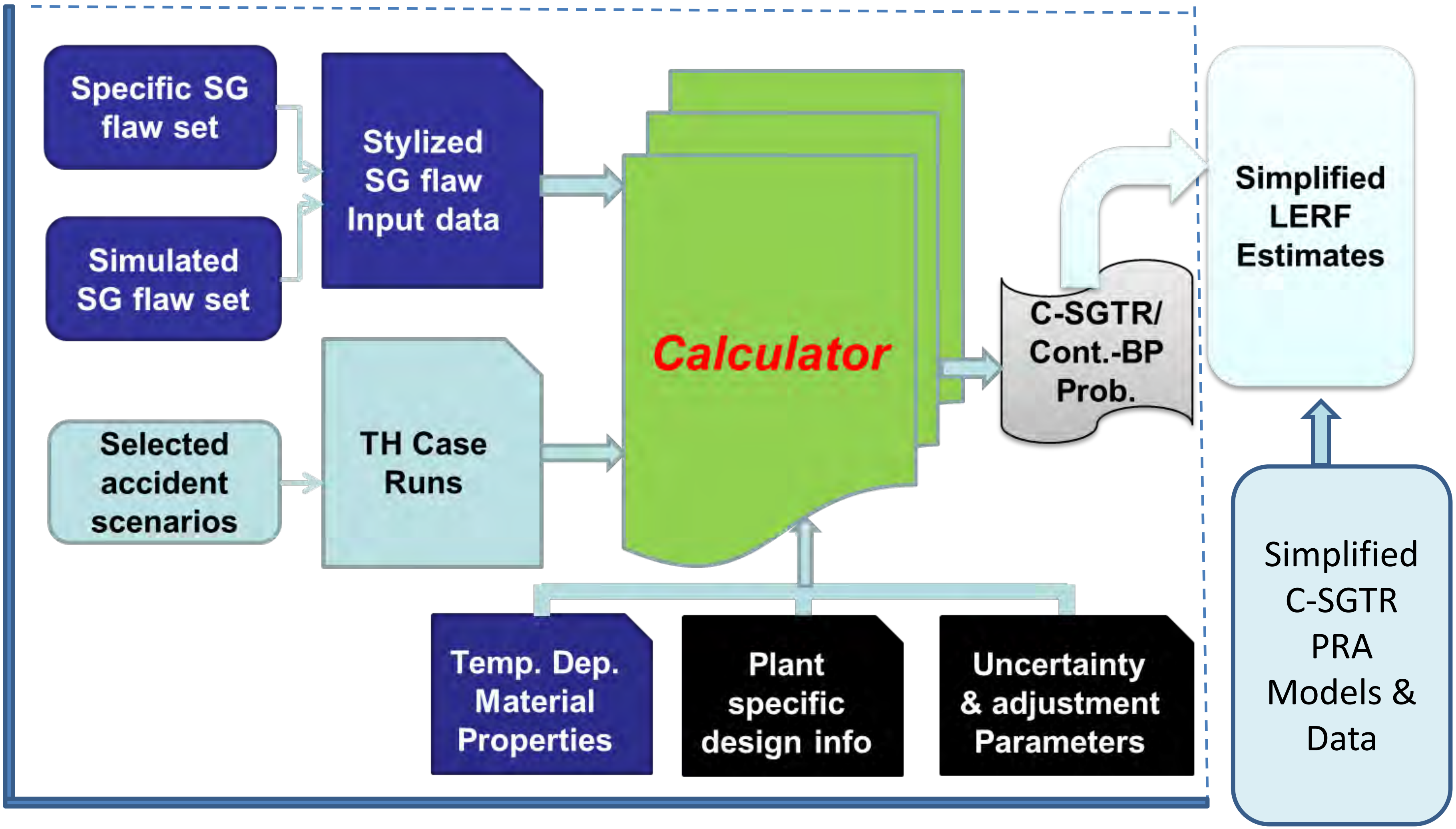
PRA objective

- Objective: develop a simplified methodology for a quantitative assessment of the risk associated with C-SGTR. It includes:
  - **Thermally induced** C-SGTR after the onset of core damage due to high SG tube temperature , and
  - **Pressure induced** C-SGTR before the onset of core damage due to high delta P across SG tube walls.
- For this purpose:
  - Develop a calculational process to estimate the conditional probability of C-SGTR given a accident sequence that challenges the SG tubes (utilizing the C-SGTR Calculator software)
  - Demonstrate use of these probabilities with a simplified PRA method to evaluate risk (e.g. LERF) associated with C-SGTR
    - Demonstrate the method using two PWRs: a Westinghouse and a Combustion Engineering design

Input for conditional C-SGTR probability

- Major input to estimate conditional C-SGTR probability $P(CSGTR|SQ, CD)$
  - Probability of SG tube failures and associated leak rate as a function of time after core damage
  - Probability of hot leg failure, or a large primary leakage including failure of other RCS components as a function of time after core damage
- This probability, combined with frequency of challenging sequences from an underlying PRA model determines containment bypass frequency
- Risk (i.e. LERF) is driven by the timing of the above occurrences

Overview of the PRA process



Risk assessment process steps

1. Identify representative accident sequences
2. Determine T&H characteristics of the sequences
3. Develop flaw set; either plant specific or simulated
4. Calculate conditional C-SGTR probabilities using C-SGTR Calculator
5. Use a simplified PRA model to estimate changes in CDF and LERF as applicable

Representative scenarios for Thermally induced C-SGTR

- PRA scenarios of interest
  - All core damage scenarios that are binned to a high primary pressure, dry SG, and low secondary pressure for the purpose of level 2 PRA evaluation

- Representative C-SGTR scenarios evaluated
  - Short SBO with early failure of TDAFW
  - Long SBO with late failure of TDAFW after batteries depleted

T&H characteristics of the representative sequences

- NUREG/CR-6995 for Zion Nuclear Power Plant (ZNPP) using RELAP/SCDAP

- In-house MELCOR analysis for CCNP (Calvert Cliffs Nuclear Power Plant)

- Several sensitivity case runs for both ZNPP and CCNP to evaluate the robustness of the conclusions

SG tube flaw input

- Plant Specific flaw set from the most recent inspection report
- Simulated flaw set
  – Random simulation of additional number of flaws generated in the cycle using flaw generation rate model
  – Random simulation of flaw sizes using flaw depth and length distributions

Probabilistic flaw model and its parameters

- Flaw data from 47 refueling cycles for Thermally Treated Inconel 600 and 690 (600TT and 690TT) were collected from selected ISI reports
  - Flaw data was manually extracted and compiled into a data base for further analyses
  - The data were binned against operating time, flaw types, and flaw sizes
  - The binned data was used to develop distributions of flaw sizes and flaw generation rate as a function of SG service life
- Flaw generation rate as a function of time (i.e. EFPY: Effective Full Power Years of Operation) using linear regression model
- Flaw Size distribution using Gamma distribution fit for flaw length/arc and flaw depth
- Adjustments were made to achieve better fit in the distribution tail (larger and deeper flaws) at the cost of less accuracy for smaller and shallower flaws.
- Additional flaw data can improve the statistics on large and deep flaws.

Example: probability that a large flaw is created during cycle 15 at a W plant

Flaw Depth Bin	Probability of a Flaw Belonging to Depth Bin
<0.6	~0.998
0.6 – 0.7	1.46E-03
0.7 – 0.8	3.39E-04
0.8 – 0.9	7.70E-05
0.9 – 1.0	small
Total	1.0

- A total of 31 new flaws is estimated to have been generated in Cycle 15 in all 4 SGs. There is a probability of 0.06 that 1 of 31 such flaws will have a depth of 60% and greater.

Output Example: Conditional containment bypass probability*

SBO with Early Failure of TDAFWs (short SBO)		SBO with Failures of TDAFWs after Battery Depletion (Long term SBO)	
CE-690 (with SRV open)	W 600 (690)	CE (with SRV open)	W 600 (690)
0.22 (0.99)	1.31E-02 (8.90E-3)	0.31 (~1)	2.6E-02 (1.8E-2)

$* = P(CSGTR|SQ, CD)$; see next slide.

Simplified LERF PRA model

- LERF estimate can be viewed as a simple 4-factor formula

$$f_{SQ}(LERF) = f(SQ) * P(CD|SQ) \\ * P(CSGTR|SQ, CD) * P(LERF|SQ, CD, CSGTR)$$

- first 2 terms are from the underlying PRA model
- 3rd term is estimated from the current work
- 4th term can be further developed to consider additional factors but can be taken as 1.0 for a simple LERF estimate.

SQ: Accident Sequence

Important factors for C-SGTR

- Number and size of SG flaws
- The temperature difference between hot leg and the hottest and average hot tube
  - Degree of mixing in the SG inlet plenum(deep or shallow SG inlet plenum)
  - Degree of mixing in HL (including HL length and diameter)
  - Pressure drop in HL and SG tubes (i.e., an integral effect)
  - Heat losses through the flow path between vessel and SG
  - Reliability of primary and secondary relief valves post onset of core damage
- Creep rupture resistance and physical sizes of SG tubes and RCS piping
- Assumptions in the underlying PRA model

(such as duration of DC availability including load shedding capabilities; Early SAMG activities; severe accident mitigation measures provided by EDMG and FLEX; including extended and diversified power sources, black start and extended operation of TDAFW without DC)

EDMG= extensive damage mitigation guidelines
FLEX diverse and flexible mitigation capabilities

Status of Comments Resolutions

- All three sets of comments are considered resolved
  - Comments by ACRS members
  - Comments by PWROG
  - Comments by Dr. Douglas Fynan
- Responses were provided and made available and draft NUREG was modified accordingly
  - Some comments were considered beyond the scope of this study
  - Project limitations and recommendations are presented in section 8.2 of the NUREG

Example of Comments Resolutions (RCP seal leakage)

- Question
  - Both PWROG and Fynan commented that the nominal leakage of 21 gpm is not applicable to CE pumps. Furthermore; it was noted that 21 gpm leakage causes early depressurization which may impact the risk evaluation
- Resolution
  - Report was modified to clarify that the leakage of CE pumps are expected to be much smaller than 21 gpm
  - Additional MELCOR simulations clearly showed that the early depressurization has little effect on long term temp/pressure time trend, therefore the relative timing of the failure of SG tubes and RCS components are not expected to changed

Example of Comments Resolutions (SIT Pressure in CCNP)

- Question
  - PWROG indicated that the reference to 700 PSI for SIT discharge is not appropriate for CE plant
- Resolution
  - 700 PSI was referenced by PRA for sensitivity analysis. The sensitivity analysis considered various primary pressure and hence time to hot leg failure after C-SGTR. Several different primary pressure was assumed; e.g. 2250 psi based on SRV set point (base case), 1200 psi based on secondary relief setting, and 700 psi for SIT actuation.
  - All references to 700 psi for SIT actuation for the CE plant sensitivity analyses was removed from the NUREG. This will not impact the PRA results reported in the report.
  - The NUREG was also modified to clarify that SIT activation pressure of 214 psi for CCNP will not reach during C-SGTR scenarios therefore have no affect on C-SGTR timing.

Example of Comments Resolutions (FLEX will reduce C-SGTR risk)

- Question
  - Both PWROG and Fynan indicated that FLEX equipment could be effective in reducing the risk by significantly extending the DC power availability, and the operation of TDAFW which can affect both short and long term SBO scenarios.
  - ACRS suggested that the impact of FLEX on results should also be discussed.
- Resolution
  - Although we generally agree that FLEX equipment could reduce risk, crediting FLEX equipment is not currently a state-of-PRA practice and it was not credited in the study.
  - Modifications were made in several places in the draft NUREG to indicate that the current plants are equipped with additional FLEX equipment which is expected to reduce the risk associated with C-SGTR .

Example of comments resolutions (Completeness of sequences from the underlying PRA model)

- Question
  - ACRS indicated that the draft NUREG states that sequences where C-SGTR can occur can be identified from existing Level 1 PRAs. However PRA information doesn't consider all conditions that could led to thermally-induced SGTR. In addition, assumed values for operator actions don't consider adverse human behavior that may occur during such events. Hence, conclusions about the importance of this event (based on existing PRAs can be misleading). In particular, this may be true for two-train CE plants.
- Resolution
  - The staff agrees that further PRA modeling can be made to identify possible additional sequences of interest. However, the study scope was limited to the potential major sources of C-SGTR challenges using existing state-of-practice PRAs to provide tools to support NRC programs such as the Significance Determination Process.
  - A detailed investigation of the impact of this assumption was done as part of another research study. This additional investigation indicated that "unmodeled" PRA sequences that can lead to High/Dry/Low conditions are a relatively small contributor to total C-SGTR risk. Further detail can be found in Appendix L.

Example of Comments Resolutions (Availability of calculator software)

- Question
  - PWROG inquired about NRC plans to release the calculator software to external stakeholders
- Resolution
  - The C-SGTR Calculator software is not formally supported by the NRC. No resources are currently available to update and support the software or address distribution requests.
  - It should be noted that the use of the Calculator must be coupled with pre- and post-processing of input/output and various judgement calls on the part of the user.
  - Details on key calculator functions are described in the publicly available basis and user guide document (ADAMS ML15054A495)
  - Staff will evaluate distribution of the calculator on a case-by-case basis.

Path Forward in 2017

- Plan to issue NUREG-2195 in 2017
- ACRS Subcommittee and Full Committee meetings scheduled

  - subcommittee - May 3, 2017 (full day)

  - full committee - June 7, 2017
- Staff is not anticipating significant revisions to draft NUREG-2195
- Possible options to expedite publication of NUREG
  – Reduce the length of next Sub Committee meeting and/or schedule sooner
  – Cancel Subcommittee meeting
  – Schedule full Committee meeting sooner