UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

June 25, 2015

MEMORANDUM TO: ACRS Members

FROM: Christopher L. Brown, Senior Staff Engineer **/RA/**
Technical Support Branch, ACRS

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS METALLURGY AND REACTOR FUELS SUBCOMMITTEE MEETING, APRIL 7, 2015 - ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on June 25, 2015, as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc w/o Attachment: E. Hackett
M. Banks

NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

June 25, 2015

MEMORANDUM TO: Christopher L. Brown, Senior Staff Engineer
Technical Support Branch, ACRS

FROM: Joy Rempe, Subcommittee Chairman for C-SGTR
Metallurgy and Reactor Fuels Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS METALLURGY AND REACTOR FUELS SUBCOMMITTEE MEETING, APRIL 7, 2015 - ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on April 7, 2015, are an accurate record of the proceedings for that meeting.

/RA/ 06/ 25 /2015

Joy Rempe, Chairman, Date
Metallurgy and Reactor Fuels
Subcommittee for C-SGTR

Certified on: June 25, 2015
Certified by: Joy Rempe

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE ACRS METALLURGY AND REACTOR FUELS SUBCOMMITTEE MEETING
April 7, 2015
ROCKVILLE, MD

The ACRS Materials, Metallurgy, and Reactor Fuels Subcommittee held a meeting on April 7, 2015, in T2B1, 11545 Rockville Pike, Rockville, MD. The meeting convened at 8:30 p.m. and adjourned at 5:03 p.m.

The meeting was opened to the public.

No written comments were received from members of the public.

ATTENDEES

ACRS Members/Staff

Joy Rempe, Subcommittee Chairman for C-SGTR	Gordon Skillman, Member
Charlie Brown, Member	Sanjoy Banerjee, Member
Ron Ballinger, Member	Dennis Bley, Member
Dana Powers, Member	John Stetkar, Member
Michael Corradini, Member (via phone)	

Christopher Brown, Designated Federal Official

NRC Staff

Mica Baquera	Kathryn Brock	Christopher Boyd	Kevin Coyne
Istavan Frankl	Evelyn Gettys	Raj Iynegar	Richard Lee
Emmett Murphy	Michael Salay	Jason Schaperow	R. Schneider
Antonios Zoulis			

Industry

Ali Azarm, IESS
Roy Linthicum, PWROG

SUMMARY

The briefing described a multi-discipline effort led by NRC/RES on Consequential Steam Generator Tube Rupture (SGTR). C-SGTR is defined by a leakage area greater than a threshold value occurring before a large vent path in RCS is established; either due to failure of RCS components or intentional depressurization. The staff (and their contractors) have documented their analysis and results in a NUREG entitled, “Consequential SGTR Analysis for Westinghouse (W) and Combustion Engineering (CE) Plants with Thermally-Treated Alloy 600 and 690 Steam Generator Tubes.” System analysis tools; such as, MELCOR and SCDAP/RELAP5 are used to predict the system flows and heat transfer. The briefing topics included the flowing: 1) C-SGTR scenario description, 2) TH analyses, 3) computational fluid dynamics (CFD), 4) experimental basis, 5) differences between CE and W plants, and 6) CE MELCOR analyses.

The meeting transcript is attached and contains an accurate description of each matter discussed during the meeting. The presentation slides and handouts used during the meeting are attached to the transcript.

SIGNIFICANT ISSUES	
Issue (s)	Reference Pages in Transcript
1. Introduction by Dr. Kevin Coyne and a brief discussion of the staff's scope of work. Preliminary questions were raised by the members that included deliverables, schedule, and public comments. Members also raised a concern relating to the lack of documentation for the CE CFD and MELCOR analyses. See transcript for further discussion of this matter.	7-26 230
2. The fast scenario beginning on slide 4 was discussed by RES staff. Subcommittee members asked questions pertaining assumptions related to relief valve (e.g., SRV and PORV) performance when exposed to temperatures expected during a severe accident. See transcript for further discussion of subcommittee member's questions and how the staff responded.	28-33
3. It was mentioned that the draft NUREG states that C-SGTR can be identified from existing Level 1 PRAs. PRA information doesn't consider conditions that could lead to thermally-induced SGTR. Member Stetkar interjected that clarity on this the misconception is needed. See transcript for detailed explanation from member Stetkar.	35-36
4. Staff discussion on reactor coolant system temperatures from the W calculations. Members observed that melting temperature assumptions for Inconel and stainless steel presented by the staff appear high. Additional discussion by members Rempe, Ballinger and Powers on oxidation of stainless steel. See transcript for details on this matter.	41-46 237-238
5. Discussion between members and staff on staff slide 11 which shows a counter-current natural circulation and the rate of mixing in the inlet plenum. The pressurizer liquid level was discussed for the analysis. See transcript for additional discussion.	51-55
6. Discussion on CFD CE sensitivity calculations performed. As was raised many times during the briefing, questions and concerns from members on the need for readily available reports and the lack of proper documentation. See transcript for discussion on this matter.	63-65
7. Discussion of B&W Plants. Staff stated that these plants have not been part of the recent severe accident induced failure studies because the vigorous natural circulation flows wouldn't be expected in these plants. Key design features of the B&W fleet	65-67

were pointed out during the briefing. See transcript for additional features discussed about the B&W fleet.	
8. Clarification discussion concerning replacement steam generators for W plants similarities to CE steam generators. See transcript for further on this clarification.	77-78, 147-148
9. The approach taken for the hottest tube analysis for CE plants was discussed by staff. It was noted by member Rempe that the W analyses relied on NUREG 1922; but the CE approach relied on the expert opinion of several NRC staffers. Refer to transcript for detailed discussion of this matter.	83-91
10. Staff discussed hot leg-surge line model aspects and analysis procedures. Failure times and thermal-mechanic simulation were discussed. Questions were raised by SC members on failure time uncertainties and temperature distribution. See transcript for how the questions were answered by NRC staff and its contractor.	107-133
11. Discussion on the C-SGTR calculator. Questions, concerns and comments from the subcommittee that the calculator does not correctly account for the uncertainties. See transcript for further discussion of this topic.	133-138
12. Discussion on updating flaw distribution and questions concerning the flaw data (probability versus flaw depth distribution) in the ISL report. See transcript for further discussion.	149-154
13. Additional discussion on flaw distribution for Thermally Treated Inconel 600 and 690 (600TT and 690TT). Members discussed the limitations in the NUREG as compared to the ISL letter report. See transcript for further discussion.	156-173
14. It was stated that the calculator was developed to support the C-SGTR PRA analyses. The calculator estimates the C-SGTR probability for a specific accident scenario and for a set of flaw in the steam generator tubes. Material data for thermally-treated Inconel 600 and 690 (TT600 and TT690) are used for SG tubes modeled. Questions from members Rempe and Bley on PRA, uncertainties, and MELCOR analyses. See transcript for questions raised and how staff and its contractors responded.	183-198
15. Discussion on sources of uncertainties that the calculator tries to handle; e.g., modeling and measurement uncertainties were discussed. Questions were asked by Members Bley, Powers, and Skillman. See transcript for questions raised and how staff and its contractors responded.	199-206
16. Dr. Coyne summarized how the calculator works.	212-214
17. Discussion on the probabilistic risk analysis of C-SGTR for severe accidents and DBA accidents and PRAs. Members Stetkar and Bley asked questions about PRA scenarios. See transcript for further details.	215-229
18. Discussion on the seven steps for a simple PRA for LERF estimation. W results were discussed. Questions were raised by Member Stetkar on uncertainties in temperature and time.	231-249

Discussion of the CE results using MELCOR. Subcommittee Chairman Rempe raised related questions concerning differences in the MELCOR analysis versus a SCDAP core heat up analysis. See transcript for further details on these subject matters.	
19. Discussion on a single flaw of various depths and determining the probability of SGTR before RCS failure. See transcript for further discussion.	249-252
20. Discussions on estimated containment bypass probability for CE and W plants. See transcript for further discussion.	252-257
21. Discussion on simplified LERF model and summary of the results. In particular, summary of pressure induced C-SGTR for W plant and CE plant. See transcript for further discussion.	257-270
22. PRA summary and insights were discussed by the staff. Comment and discussion by member Stetkar concerning are the PRA models capturing all scenarios. See transcript for further discussion.	270-280
23. Office of Nuclear Reactor Regulations perspectives on the draft NUREG and future plans. See transcript for further discussion.	283-292
24. Public comments on draft NUREG. See transcript for further discussion.	297-299
25. Subcommittee comments. See transcript for further discussion.	300-309

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards
Metallurgy and Reactor Fuels Subcommittee

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Tuesday, April 7, 2015

Work Order No.: NRC-1503 Pages 1-313

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

METALLURGY & REACTOR FUELS SUBCOMMITTEE

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TUESDAY

APRIL 7, 2015

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., Joy Rempe, Chair, presiding.

COMMITTEE MEMBERS:

JOY REMPE, Chair

RONALD G. BALLINGER, Member

SANJOY BANERJEE, Member

DENNIS C. BLEY, Member

MICHAEL L. CORRADINI, Member*

DANA A. POWERS, Member

MICHAEL T. RYAN, Member

GORDON R. SKILLMAN, Member

JOHN W. STETKAR, Member

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER L. BROWN

ALSO PRESENT:

EDWIN M. HACKETT, Executive Director, ACRS

ALI AZARM, IESS

MICA BAQUERA, RES

KATHRYN BROCK, RES

CHRISTOPHER F. BOYD, RES

KEVIN COYNE, RES

ISTVAN FRANKL, RES

EVELYN GETTYS, RES

ACE HOFFMAN*

RAJ IYENGAR, RES

KEN KARWOSKI, NRR

RICHARD LEE, RES

ROY LINTHICUM, PWROG

EMMETT MURPHY, NRR

MICHAEL SALAY, RES

SELIM SANCAKTAR, RES

JASON SCHAPEROW, NRO

RAY SCHNEIDER, Westinghouse

ANTONIOS ZOULIS, NRR

*Present via telephone

T-A-B-L-E O-F C-O-N-T-E-N-T-S

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

(8:30 a.m.)

CHAIR REMPE: Okay. This meeting will now come to order. This is a meeting of the Materials and Reactor Fuel Subcommittee of the ACRS. I am Joy Rempe, Chairman of today's subcommittee. Subcommittees in attendance are Sanjoy Banerjee, Dick Skillman, Dana Powers, John Stetkar, Ron Ballinger. In addition we have Mike Corradini on the phone. And we may be joined by Dennis Bley later in the morning.

MEMBER POWERS: Is that Mike Corradini from the losing University of Wisconson?

CHAIR REMPE: I believe that's the phrase.

MEMBER POWERS: Ah, okay.

CHAIR REMPE: The purpose of this meeting is to receive a briefing from the staff in the Office of Nuclear Reactor, excuse Nuclear Regulatory Research, and the Office of Nuclear Reactor Regulation on consequential steam generator tube rupture.

Specifically we'll hear about information in a draft NUREG entitled Consequential Steam Generator Tube Rupture Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes.

The subcommittee will gather information and analyze relevant issues and facts, and formulate those positions and actions as appropriate for deliberation by the full committee. At this time the full committee briefing is scheduled to occur on June 10th. Christopher Brown is the designated federal official for this meeting.

The rules for participation to be in today's meeting have been announced as part of the notice of this meeting previously published in the Federal Register on March 31st, 2015. A transcript of the meeting is being kept, and will be made available as stated in the Federal Register notice.

It's requested that speakers first identify themselves, and speak with sufficient clarity and volume so that they can be readily heard. Also, please silence all phones or other widgets that make noise.

We have not received any comments or requests for members of the public to make oral statements or written comments. A bridge line has been set up for public participation, which will occur at the end of the meeting.

So let's now proceed with the meeting. And I'd like to start by calling upon Dr. Kevin Coyne, the Branch Chief RES, to introduce the speakers and give us a short introduction.

DR. COYNE: Okay. Thank you, Dr. Rempe. Again, Kevin Coyne from the Office of Nuclear Regulatory Research. Thanks again for the opportunity to brief to the joint subcommittee. Raj Iyengar will go through a quick summary of the previous interactions we've had. But we've had several interactions with the full committee and subcommittees on this work.

Just to go through briefly some of the history. We had the steam generator action plan that ran well over a decade. And it closed in 2009. One of the items that was left remaining for the steam generator action plan was some resolution on some consequential steam generator tube rupture issues.

A decision was made in 2009 to close the steam generator action plan, with the rest of the remaining work to go under our normal work process. That included things like doing more detailed work for the Combustion Engineering type designs. And developing a simple but realistic PRA method, no small challenge, to support severe accident licensing reviews from a PRA perspective, and also support the reactor oversight process.

So it's a complex project. It's had the involvement of all three research divisions, thermal-hydraulics, computational fluid dynamics, operating experience for flaws and flaw distribution analysis, structural work for the reactor coolant system, and obviously PRA work.

In addition, since 2010 we've had some diversion of resources, I think it would be safe to say, with the Fukushima event, dealing with some budget of challenges with sequestration. And this work has always been considered important. But it hasn't always been our highest priority in research. But we've made steady progress over the last five years.

But that has caused us to extend the schedule for the work a bit longer than I think we originally anticipated when we first put this on. But the good news is the technical work is now complete. We have a first draft report that's been provided to the subcommittee.

Our focus has been on getting the technical work right. So we recognize there is some smoothing and some editing details in the report that still need to be worked out. But we want to get the technical information down correct, and any comments from the committee incorporated in the report before go through that process.

So looking ahead our plan is to get

feedback from the committee. We plan to incorporate that into the draft report. Then we'll prep the report for issuance as a draft NUREG for public review and comment. We hope to get that published as a draft NUREG by the end of the year, with a target of publishing the report as a final in 2016. And with that --

CHAIR REMPE: Since you brought up schedule, let me see if I can understand, just to make sure. You are planning to, if there's -- For example, if there were a lot of technical changes you are planning to incorporate those changes, and then by the end of the fiscal year or the calendar year you will issue, update the report and issue it for public comment after you incorporate our comments?

DR. COYNE: Correct.

CHAIR REMPE: But you would, it's currently scheduled to have a full committee meeting in June right now. Is there any flexibility? I mean, maybe there won't be a lot of technical comments. But if there were, would there be interest in maybe having a second subcommittee meeting? Or are you pretty fixed, no, I want just one committee meeting? Or, what's your idea?

DR. COYNE: I guess we'd defer to the preference of the committee on that as to whether you wanted us to come back, say after we finish the public comment period, to --

CHAIR REMPE: Before, if there were some substantial changes.

DR. COYNE: Right. Before we issue it as final. We could work out those details as we got closer to that date.

MEMBER CORRADINI: So, may I ask a question about full review?

CHAIR REMPE: Sure.

MEMBER CORRADINI: Okay. I guess my question is, and I wanted Kevin to correct me if I misunderstand him. But already there's a process under, to use some analysis for consequential steam generator tube rupture. And this work by RES is essentially reducing the (coughing) in an already known process that is used in risk assessment. Is that a correct assumption?

DR. COYNE: Well, yes and no. The part we'll decide today goes through some additional efforts for prioritizing steam generator flaws, and coming up with better distributions.

There's been previous work on characterization of flaws, which we'll discuss later, as a very key input into the PRA analysis to have the distribution of flaws, and the depth of those flaws correct.

So there's been more work done recently as a part of this study, to develop the flaw distributions for alloy 690 steam generator tubes. The earlier work focused on alloy 600 tubes. We also have some better tools available.

We'll talk about the steam generator calculator, which really is just a fancy way to bring together what we know about steam generator tube crack growth and creep rupture failure, to combine that with the thermal-hydraulic analysis, and do a better prediction of severe accident induced containment bypass events.

So I think, and others here can correct me if I'm wrong. But I think the essence of how we approach consequential steam generator tube rupture is essentially unchanged. But some of the details and the technical inputs have been refined.

Additionally, Dr. Mike Salay will talk about additional thermal-hydraulic work that was done for Combustion Engineering plants, which was an input that we did not have previously.

MEMBER CORRADINI: So, but then, I guess the way you describe it, if I were to characterize it what you're really saying is you have a process, you're using process to include a risk assessment, and now you're improving the process.

MEMBER STETKAR: Hey, Mike, you're breaking up.

MEMBER CORRADINI: Well, I can barely hear from papers rattling. But let me try it again.

MEMBER STETKAR: No, you're breaking up.

MEMBER CORRADINI: Well --

MEMBER STETKAR: Coming this way.

MEMBER POWERS: You're better, Mike, as you get closer to the microphone.

MEMBER CORRADINI: Okay. Well then, let me try --

CHAIR REMPE: That's it.

MEMBER POWERS: Much better.

MEMBER CORRADINI: Is that good enough now?

CHAIR REMPE: That's much better.

MEMBER CORRADINI: Okay, fine. So, let me just make sure I'm clear. There's a current process in use. The process is being improved relative to certain alloys and flaw distribution. Also the process is being improved considering combustion steam generators. But by and large the process is being improved. So I would appreciate it if you go through the process by --

MEMBER STETKAR: You're breaking --

MEMBER CORRADINI: -- and how that changed relative to the current analysis. And where, if they're increased or decreased.

CHAIR REMPE: Mike, you're not on a cell phone, are you?

MEMBER CORRADINI: Yes.

MEMBER STETKAR: Yes, well --

(Simultaneous speaking)

MEMBER STETKAR: Do the land line, please.

MEMBER POWERS: Because we can understand about a third of what you're -- Well, we can't understand anything.

CHAIR REMPE: But I think that you said you would like them to emphasize differences in the new process versus the old one. Isn't that what you said, Mike?

MEMBER CORRADINI: Yes. Can you hear me now better?

CHAIR REMPE: Yes, yes.

MEMBER CORRADINI: Okay, fine. So, I guess what I'm trying to get at is, as I looked through the documents, and I now have the NUREG. I'm trying to make sure the staff clearly identifies what's new, what's not new, where the uncertainty is reduced, where things remain the same.

CHAIR REMPE: Got it.

DR. COYNE: Kevin Coyne from the Office of Research. I think we understand. And another point that I failed to mention is there's a previous NUREG on this topic, NUREG-1570, that many of the members may be familiar with that provided a process for looking at consequential steam generator tube rupture from a PRA context.

So I think the best way to view the current work that we're going to discuss today is an extension of some of those methods, and an improvement in some of those methods.

So, I think the overall point is correct. There are existing methods. But this work has refined them and brought them up to date with some of the modern materials that were being used in the steam generators.

MEMBER CORRADINI: Okay. Thank you.

CHAIR REMPE: So I have another question about Schedule 2. The user needs talked about regulatory guidance and updated handbook section for the risk assessment standardization project, and all these other things.

Those are going to be after the end of the year, or after you do the public comment? And then you'll start doing those types of interactions? Or, I know there's a slide at the very end that talks about a different type of process. But what's your vision on that, just so we know up front.

DR. COYNE: Yes. Some of those details don't really work out. We'd like to refresh our user needs at least every five years, if not sooner. So, this one is getting to the five year point. In fact, it may have already passed the five year point just barely.

In discussions with NRR the preliminary agreement we have is, once this report is issued then we're going to revisit the user need, and better focus the topics on what deliverables NRR would need to implement the process. And there would be a follow-on. We'll essentially close the existing user need, and then potentially generate a new user need to focus on development of those regulatory tools.

CHAIR REMPE: Okay. Any more preliminary questions? Thank you. Sorry, but it's good to get these things out up front. Raj.

DR. IYENGAR: Good morning to all of you. I'm pretty happy and honored to be here to present some of our almost nearing conclusion efforts on the consequential steam generator tube rupture program.

As we had this preliminary discussion you could see that was perhaps the harbinger of what's to come in terms of lively discussion throughout the day. You will encounter several presentations from our staff which will outline and disclose some of the most, the more important sophisticated analysis that we have done, both in terms of thermal-hydraulics, as well as in terms of structural analysis, which will offer credence to the simplified calculator calculations that you will see here in the afternoon.

Just as a background, I think Kevin covered most of the background behind this effort. I just wanted to mention a couple of things in addition, is that we have been repeatedly engaging the ACRS members, both informally as well as through subcommittee briefings, and full committee briefings, through the past four years.

So, we've had the benefit of their feed, their comments and feedback, and advice and insights into the product that you probably are reviewing. So, that's been extremely helpful. So this is not a total surprise to most of you, I believe.

I think Dr. Powers had been a driving force behind this effort from the initial get go. And Dr. Rempe has been extremely engaged and very helpful, and offered key advice and insights for this effort.

So, without much ado, I just wanted to make one personal remark for the record. Any project of this complexity and interdisciplinary nature will always have some issues and problems. But it requires a singular driving force to bring it to conclusion.

In that regard I want to acknowledge the effort of Dr. Kevin Coyne, who has been very helpful and persistent, undeterred, throughout the number of other events and overriding priorities, mainly Fukushima.

And in addition to all the major significant projects he and his branch staff do. So, I really am pleased. And without his singular effort we would not be here today at this. Thank you very much.

(Simultaneous speaking)

CHAIR REMPE: I have this one question too. Where the, is this is too preliminary to try and use it yet. So there's been no thoughts of even trying to use any of the insights. Or have any of the insights from this work been used in the Vogtle III Level 3 PRA, for example?

DR. COYNE: Kevin Coyne from research. We have applied the methodology to the Vogtle project, so it can be -- You want to step to the microphone and add something to that?

MALE PARTICIPANT: Yes.

DR. COYNE: Yes. Enough said. So we have --

CHAIR REMPE: We're starting to use it already.

DR. COYNE: Yes. We've used the calculator and the flaw distributions with the thermal-hydraulic analysis that we've done for Vogtle, and applied the same methodology that will be described today for the Level 3 project.

CHAIR REMPE: Okay. Thanks.

DR. SALAY: Mike Salay. And I'll talk a little bit about the sequence and some of the thermal-hydraulic work that's been done. I'm going to go over the scenario description, the thermal-hydraulic analysis, how we're going about solving the problem, a little bit of the experimental basis for the CFD, some differences between CE and Combustion Engineering plants. And then I'll show some of the results of some of the Combustion Engineering Analyses.

CHAIR REMPE: Mike, can you go back a little bit? I'm sorry. First of all, the ARTIS. I looked through your few graphs in advance. And I appreciate you guys them to me so I could. But the ARTIS input was not at all used in this program.

DR. SALAY: We didn't focus on releases.

CHAIR REMPE: Right.

DR. SALAY: So we ended up not calculating releases, so we can -- Well, we did calculate it partially. But it was never part of -- We didn't end up going --

We were initially intending to come up with releases as part of the guidance. But we didn't go there. So we never got to the point where we never added the ARDS stuff.

CHAIR REMPE: And just checked to see it in the work. It doesn't matter. But it actually was, again, this is before my time on ACRS. But it was an action item for the SGAP effort. It was closed out because the staff agreed to use it in this effort. And just what, the rules of the game here, what does that mean that it has not been done? Will it be done later? Or does that matter that it --

DR. SALAY: Well, if you're actually using the releases for something, yes, it would be good to calculate the retention.

CHAIR REMPE: So it's going to be to put it in at some point later into MELCOR?

MEMBER POWERS: Well, it is in MELCOR.

CHAIR REMPE: It is? But it just has not been used.

DR. SALAY: This was done before it was in there. So it's --

MEMBER POWERS: Well, if you're asking if the ARDS' results were in MELCOR, yes. For the SOARCA effort, I mean, I did it. I took the ARDS' results and generated for them a model that they had implemented into MELCOR.

CHAIR REMPE: Okay.

MEMBER POWERS: And what they -- I did a Stokes number scaling on it, and they chose not to take that Stokes number scaling into MELCOR for reasons you'll have to ask them.

But basically it's, the model consists of a decontamination and by the quakes within the steam generator, and the decontamination by the separator and dryer at the top.

CHAIR REMPE: And I believe someone reported, mentioned that you thought that the use of ARDS' input would reduce the releases by a factor of five.

MEMBER POWERS: About.

CHAIR REMPE: So, if you were going to do something like the Vogtle III PRA, you'd have it in MELCOR, and you'd have that reduction, but you'd also use the calculator. So, the fact that it wasn't done as part of this --

MEMBER STETKAR: By the way --

CHAIR REMPE: -- it not a problem.

MEMBER STETKAR: For the record it's not Vogtle III, it's the Level 3 PRA for Vogtle Unit I.

CHAIR REMPE: Yes, okay. I'm sorry.

MEMBER STETKAR: It's just really important.

CHAIR REMPE: Yes, sorry.

(Simultaneous speaking)

CHAIR REMPE: But anyway, so it has, the methodology can accommodate the ARDS' work now. Is that a true statement?

MEMBER POWERS: Let me also be clear. It depends on where your break is, how much reduction --

MEMBER STETKAR: Yes.

CHAIR REMPE: Right.

MEMBER POWERS: I do, the original 1150 kinds of analyses were kind of independent of break location. But the model that's in MELCOR now, very definitely depend on where the break occurs.

And based on our empirical data we are generally recommending a uniform distribution for the location of break within a tube. That's based on historical data that does not have to do with these ally 600, 690 tubes.

CHAIR REMPE: And then the last thing I wanted to bring up on this slide is, for the record, there was earlier documents that reported the earliest ISL SCDAP analyses. There were documents that talked about the CFD work for the Westinghouse plant. But when you talk about the CE CFD work and the CD MELCOR analyses there are no standalone documents?

DR. SALAY: There are standalone documents. I thought we provided those.

DR. COYNE: Yes. There's NUREG reports for the Westinghouse analysis.

CHAIR REMPE: For the Westinghouse. But CE --

DR. SALAY: No. CE, Sandia did some stuff, and I did some stuff.

MEMBER CORRADINI: Let me get that clear. Those were provided to us? I guess, I'm looking, and I don't, I'm not aware of that.

DR. SALAY: Were they? Were they not?

CHAIR REMPE: And I, actually the draft NUREG I reviewed didn't have any references. And that's why I'm asking that question. I mean, it talks about an older CE plant analysis.

But it -- So, you made some changes and it reports those, but not in the level of detail where you could review them. And so, are there standalone documents? And they should be referenced in the draft NUREG if there are.

DR. SALAY: Well, yes. I wrote something. It was in draft form. It was decided not to be included in as a reference.

MEMBER CORRADINI: So, just to be clear, we do or don't have those auxiliary documents if we want to look at details?

MEMBER STETKAR: We don't.

CHAIR REMPE: Okay. That probably will come up again. I just wanted to make sure. And so, was there a reason why they weren't even referenced in the document?

DR. COYNE: I'll ask Dr. Lee to address that, if you could?

DR. LEE: About what?

DR. COYNE: The standalone references for the CE work that was done for the study.

DR. LEE: And why do you need the standalone?

CHAIR REMPE: Well, there's a lot of details for the Westinghouse analyses that we can review. And then it talks about the CE work, and it doesn't have that level of detail. And I am, was just wondering why it wasn't there. And then I hear that there are references, we just didn't want to reference them. And I'm just surprised.

DR. LEE: No. I think maybe we should go through this whole entire presentation then, so you can see whether the thermal-hydraulics analysis relative to other analyses in this whole entire process. And see how important that we need to keep on refining the thermal-hydraulics analysis.

CHAIR REMPE: So, you're saying that the documents are drafted. They're being --

DR. LEE: I didn't draft anything. If Mike had draft something -- What we need to, what we have agreed to do for this report is that report whatever we needed to support the closing of what Kevin needs to do for this issue. Okay.

CHAIR REMPE: Well --

DR. LEE: We don't have enough time to do also, do more things that that we can do. Because Mike has many other activities he has to do. So that's the reason.

CHAIR REMPE: Well, throughout this document, like, you talked about the flaw distribution, those ISL documents that are in the system, and we can, the NUREG relies on information that's already documented other places.

And so, if you're a curious reviewer you can go to those other documents. In the case of the CE MELCOR analysis and the CFD analysis for the CE plant, there wasn't even any citation to references. And so, it makes it a little more difficult for the reviewer.

DR. LEE: Are you saying that within the document itself there's not enough details in there to --

CHAIR REMPE: I thought that way.

DR. LEE: -- make some conclusion on it?

CHAIR REMPE: I thought that way.

DR. LEE: What do you think is going to change?

CHAIR REMPE: Mike, I think you felt that way too, right?

MEMBER CORRADINI: Well, I guess I think

we want to listen to what Mike has to say. But I just wanted to make sure that everything that you want to present on the CE plants is, ought to be in the document. So, if we're looking for more we should speak up when we get to that point in the presentation. Is that correct?

CHAIR REMPE: Sounds good to me.

DR. LEE: Correct.

CHAIR REMPE: Okay. Sorry. You can go ahead with the -- DR. SALAY: So second, I guess second slide. Yes. So the reason that there's an interest in this project is, the main reason is that it could lead to a scare that is called a bypass, a containment bypass.

And this is where your fission products, instead of being attenuated and reduced in containment, they go, they bypass the containment and can reach the environment now. And the scenario that it happens is a station blackout, where you lose your diesel generators.

And then so then what happens is, the reactor inventory boils off. You have a release of fission products. Your pressure stays at your PRV or SRV setpoint, and your temperature goes up. And sooner or later something is going to fail.

And what fails depends, effects whether you have a bypass or not. If your tubes fail, your steam generator tubes fail first, the fission products can go into the secondary side, and then potentially to the environment.

However, if any other component fails, the RCS blows down into containment. So determining whether one of the steam generator tubes fails relative to other RCS components is important in determining the consequences.

So here's a fast, demonstration of a fast scenario that was based on Westinghouse analyses. So you have a loss of offsite power, failure of diesels and failure of auxiliary feedwater systems. And then the primary inventory gets lost through the reactor coolant pump seals. The secondary side boils off. And the secondary side dries out.

Your primary inventory, you start losing your primary inventory through your safety valves. And you continue to lose stuff through pump seals. Your loop natural circulation stops as the primary inventory falls in the steam generator tubes. And the natural circulation of superheated steam begins as inventory falls below the hot leg.

So the core and system heat up. Core

uncovers, core oxidizes and produces a lot of power. The system heats up and accelerates. And then you get a failure somewhere.

And this is a scenario from Westinghouse where your turbine driven auxiliary feedwater was not operational. It's more likely that some feedwater, some actions would get some water in the secondary side to prevent that from happening.

MEMBER BANERJEE: Let's go --

DR. SALAY: Sorry.

MEMBER BANERJEE: So the scenario, this is a scenario, but there would be failures which would occur, right?

DR. SALAY: Yes.

(Simultaneous speaking)

DR. SALAY: It's either the tubes or something else. If something else fails, the surge line, the hot leg. You could have --

MEMBER BANERJEE: It depends on the timing, as we've seen in the past, right?

DR. SALAY: Yes. And even, this is actually initially made for Westinghouse. Yes. So if your tubes fail first -- Well, let's go -- If your RCS, other RCS components fail first you --

MEMBER BANERJEE: Then you're home free,

right?

DR. SALAY: Then you're fine, yes. If your tubes fail first, it depends. If enough, if one tube fails your pressure stays high. And it's likely that it will remain high enough for other components to fail, and then depressurize.

And so you, which reduces the driving force to the, for fission products to the environment. However, if a lot of tubes fail you could depressurize sufficiently to prevent another RCS component from failing by --

MEMBER BANERJEE: So the calc, this is not necessarily the motivation. But when you look at this scenario, the timing of these failures becomes --

DR. SALAY: Yes, the timing. You want to know which fails first.

MEMBER BANERJEE: Yes.

DR. SALAY: And if one or two tubes fail first does your pressure, when, how long after will your, will another RCS component fail?

MEMBER CORRADINI: So, if Sanjoy's done I had a question, a couple of questions just for clarification. So, the assumptions are of course loss of off site power, failure of diesels, failure of aux feed. And also, there is no estimate of cycling the

PORV. And it's essentially leakage. And then depressurization that way?

DR. SALAY: Well, there's assumptions that PORV operates until batteries fail. And that's typically assumed to occur four hours in. And so, at that point the PORV no longer opens. And it goes up to the SRV pressure.

MEMBER CORRADINI: And then there's continued cycling.

DR. SALAY: Yes.

MEMBER CORRADINI: So, where I'm getting at eventually is, in the analyses of the Fukushima reconstruction it was assumed or decided that somewhere between some sort of hot leg failure or SRV failure, due to continued cycling at high temperature.

And I'm curious, in the RES analysis for this, was there a re-investigation of SRV cycling or PORV cycling that essentially led to it failing open, or failing with leakage that would depressurize?

DR. SALAY: We didn't look into that. I don't know if the rest of the --

MEMBER POWERS: Since that's a boiler, and this is a PWR --

DR. SALAY: Yes.

MEMBER POWERS: It's my impression that if you fail to open the PORV you're still pressurized for a very long time.

DR. SALAY: Oh.

MEMBER POWERS: And that, I mean, as you come down in pressure eventually you're going to drop the accumulators. And that's going to take you right back up in pressure. So you're not out of the woods. I mean, it's just hard to depressurize.

MEMBER CORRADINI: Well, I guess I'm asking, so at least in this analysis that was not reconsidered at all?

DR. SALAY: No.

DR. COYNE: Mike, if I could add. What Dr. Salay's going though is an example scenario to highlight some of the pertinent points with a severe accident induced steam generator tube rupture. Station blackout is certainly one of the scenarios we look at.

In general, the PRA method will pull out any scenarios that leave you with high RCS pressure, and dry steam generator conditions, which one of our assumptions is essentially a dry steam generator condition would lead to a low pressure on the secondary side.

MEMBER CORRADINI: And so that was my next question. So, can you just re-talk through? Because you need three things, high, dry and low. And I'm trying to understand the conditions on the secondary side that would lead you to low pressure.

DR. SALAY: What was the question? You wanted -- Well, the scenarios are, there's been a, the historical cause is an assumed leak, about an inch or half inch diameter equivalent.

And also, if a secondary valve sticks open somehow. Or, third option would be if the operator decides to depressurize so he can get some water in. But then somehow the supply of water fails.

MEMBER CORRADINI: Okay. All right. So, to frame it in another way, you picked the select set of conditions to give you the worst delta P, and holding at high pressure on the primary side? So there's a whole population --

DR. SALAY: A lot --

MEMBER CORRADINI: -- of other possibilities.

DR. SALAY: Oh, absolutely, yes. So there's a lot of things that happen. But if the -- But it's under these circumstances that you can get the bypass. Whereas, in the others --

MEMBER CORRADINI: Sure. So then let me

ask the final question. What reference should I go to, to see an event tree that leads me to all the other pathways, so I understand this pathway relative to all the other pathways? Is there a reference you'd suggest I read?

DR. SALAY: I'll defer to the PRA guys.

MEMBER STETKAR: The answer is, no you can't. Because PRAs do not examine this phenomenon.

MEMBER CORRADINI: Okay.

MEMBER STETKAR: We'll get to that.

MEMBER CORRADINI: Thank you.

DR. LEE: Can I make a comment? When we published the NUREG-1570 back in March 1998, there's a whole section of discussion between CE and Westinghouse, and so forth. There are detailed discussion here.

Subsequent to that we published another report SCDAB Rev 5 of the steam generator. Is INL report, published in June 1998. That discussed a lot more about analysis.

But at that time just remember, when we do the CE like the old analysis, we did not change any of the mixing fractions. We were using the Westinghouse type. For those three parameters, in terms about the hot leg, countercurrent flow, the inlet plenum mixing, the ratio of the backflow, and so forth.

There are three parameters key to this same tube rupture. But the values we used at that time is a Westinghouse base. The one that Mike's doing is based on the CE one, which is informed by the CFD, okay.

So, if you want to look at all the variation about the CE plant versus Westinghouse, I would suggest that you go back to read the 1570. NUREG/CR tells you all sort of things different between CE and Westinghouse.

CHAIR REMPE: So, you're saying --

MEMBER CORRADINI: Thank you, Richard.

CHAIR REMPE: -- instead of --

DR. LEE: Read that one. And you can also read the SCDAB Rev 5 report. This has all the plan analysis, including CE, the end of your plan, Westinghouse, all in here.

CHAIR REMPE: But the INL report was using SCDAB, whereas this is a MELCOR analysis.

DR. LEE: But the thing is there's SCDAB Rev 5 and MELCOR are, MELCOR catch up with it later. So it's a similar type thing. Are you expecting something different between MELCOR and this?

DR. COYNE: If I could add a clarification to -- This is Kevin Coyne from the research staff. PRAs do look at consequential steam generator tube rupture. That's why we look at it for the local Level 3 PRA

MEMBER STETKAR: They don't -- We'll get into it later. They do not correctly evaluate the challenge frequency.

DR. COYNE: Okay. So one of the ground rules that NRR really wanted with this method is to be able to have a method that allowed them to make estimates so severe accident due to steam generator tube rupture, even if the PRA did not explicitly consider it.

So that was one of our going in mattering conditions, that this method had to be able to be appended to an existing PRA that took you through a Level 1 analysis to coordinate that frequency.

So one of the key first steps in the PRA analysis is to identify all the scenarios that lead to the high-dry-low conditions that Mike is going to get to in about four slides.

MEMBER STETKAR: If he ever gets there.

DR. COYNE: Mike is giving a typical example of how you would get to these high-dry-low conditions. The PRA will give you thousands, hundreds of data sets that gets you to this condition. So, this is just a typical one to illustrate the key points.

But again, one of the key points wasn't to have a method that would fully integrate this into a PRA. It was to have a method that could be appended to an existing PRA analysis, to get some insights on severe accident tube rupture.

MEMBER STETKAR: And, Kevin, I think that's a very good, especially for the record, a very good and fair characterization of this effort. I just wanted to make sure that when we tread into the notion of the comprehensiveness of this, in the world of assessing risk from this particular condition, we're on a bit more, and in my opinion quite more shaky ground.

Because this has used a few stylized scenarios. And quite honestly, PRAs, I have yet to see a PRA that actually investigates the total frequency of these high-dry-low conditions. And we can get into that later.

But I don't want to interrupt too much, because it's important to understand the physics and the metallurgy, and all of that kind of stuff first. Because that's the real crux of this whole effort here.

DR. LEE: And one of the things I'd like to inform about, to address Joy. You brought up about the benchmarking between these codes. Since this study was done, I think back in about ten years ago we asked Karen Yurib to do some analysis, between the Zion plant, using SCDAB Rev 5, MELCOR, and then also MACCS, which EPRI did it.

And compared for this severe tube rupture, and see how they compare. So they all seem to be about in the same ball park. There was some efforts in benchmarking the three different codes to see how they performed under this accident scenario.

MEMBER SKILLMAN: Mike, let me ask this question, please. What is the assumption about the main steam isolation valves?

DR. SALAY: Not sure. And I assume they're closed.

(Off microphone comment)

MEMBER STETKAR: No. Why would they be closed? There's no reason for them to be closed. They would be fully open, unless I, John Stetkar, the operator, closed them.

MEMBER POWERS: Bingo.

MEMBER STETKAR: There's absolutely no reason for the MSIVs to be open in this particular scenario.

MEMBER POWERS: Closed.

MEMBER STETKAR: Or closed. I'm sorry. To be closed. They will be fully open.

MEMBER BALLINGER: Doesn't it say somewhere in there that there's a .5 square inch --

MEMBER STETKAR: That's -- See, they get around it, because they assume that any of these conditions depressurize the secondary side. That's why it's important --

MEMBER BALLINGER: A .5 square inch --

MEMBER STETKAR: Right. Now, I don't know, and I'm hoping we get to it. There is a notion that that is large enough to get you to the high-dry-low.

MEMBER BALLINGER: Yes.

MEMBER STETKAR: But after the tube fails it's not enough to keep pressure low in the secondary side. And I don't know what the implications of that is. I don't know if you're going to address.

DR. SALAY: Yes, there is --

MEMBER STETKAR: When you get to it, that covers it.

DR. SALAY: All right. And so, these are some of the RCS structure temperatures from the Westinghouse calc. Just to, it looks like you have the dry out, then the temperature starts increasing. You have, start getting the circulation of superheated steam. And it heats up, the system heats up.

The hot leg for Westinghouse is significantly hotter than either the average tube or the hottest tube. And when oxidation, heat oxidation occurs the temperature jumps pretty fast. And there are a few points of interest. Again --

MEMBER POWERS: Have we wrestled successfully with the fact that we didn't see these kinds of temperatures in the hot leg of TMI?

DR. SALAY: Yes. But don't know what temperatures we had, readings we had. So I haven't seen it compared to Three Mile Island.

MEMBER CORRADINI: But I guess, Dana, another way of asking Dana's question is, did any of the measured value at TMI peg out, so that we at least say that they were higher than X, or higher than Y? I know that's the case in the core. But I don't remember anything in the hot legs.

MEMBER POWERS: We kind of know what the temperatures were in the upper core structures at TMI.

DR. COYNE: Mike, correct me if I'm wrong, but TMI was a BW plant --

MEMBER POWERS: It is.

DR. COYNE: It doesn't, would not behave the same way as a Westinghouse or CE plant under these circumstances.

MEMBER POWERS: But we would expect that if we saw high temperatures coming off the core that we would see them in the hot leg, regardless of the manufacturer of the plant, wouldn't we? And at TMI we just didn't.

Now, you're explanation of this as a different type of plant, fair enough. I can't argue with you there. But it would seem to be a calibration that just comes to mind.

DR. SALAY: That definitely makes sense to compare against -- And I do mention that the B&W, that we don't expect to see large recirc flows.

DR. LEE: Remember that the B&W plant is a candy cane design.

DR. SALAY: All right.

CHAIR REMPE: Before we leave that, and again, I'm sorry. I have read the document. But I'm just a little confused on the details. But I thought when you went to the MELCOR analysis you added this capability to do the hottest tube temperature and the average tube temperature in the steam generator. And that was not something that was in the SCDAP analysis that was done years ago.

DR. SALAY: That was in the SCDAP RELAP last analysis that was done years ago. But it was not in the CE analysis that was done years ago.

CHAIR REMPE: Okay.

DR. SALAY: That's one of the things we had to add.

CHAIR REMPE: Okay. That's -- I guess my brain's not functioning right now. Okay.

DR. SALAY: So this is --

CHAIR REMPE: So that's why I was puzzled. I was going, I thought that was just the MELCOR. But okay. Got it.

DR. SALAY: And again, your RCS points of interest where things can fail. You have the lower head, the hot leg, surge line, the tubes, and the seal leakage, and whether your loop seal is coming in, has water in it or not. And I mentioned that.

CHAIR REMPE: So, before you leave that slide. Melting temperature for stainless steel and Inconel. That actually seems a bit high. It's an alloy, and some references might go down to 1650 to 1725k, something like that, right.

And then I noticed you put like oxidation of zircaloy, which is in the core. But also you can have steam oxidation of stainless steel, for example.

Has anyone thought about what would happen in a steam environment for the stainless steel with the carbon steel from the hot leg, and thought about how that might affect thermal properties? Or if the properties, did you have properties for these other structures that, before you get to the steam generator tubes?

DR. SALAY: I'm sure someone's thought about it. But it was not part of the analysis. We're just doing a thermal-hydraulic and what's in --

CHAIR REMPE: Okay. So yes. And I know it has to have been thought of before. But if you start looking at when you're in a steam environment, references have like 540 to 600. So you'll start seeing stainless steel starting to have some changes.

MEMBER POWERS: There's an absolutely brilliant paper, exhaustive in its scholarship --

CHAIR REMPE: Must be by that Dr. Powers guy.

MEMBER POWERS: -- entitled, "Stainless Steel Oxidation, the Forgotten Source of Hydrogen", that explores that in massive and quantitative detail.

MEMBER BALLINGER: Did this exhaustive scholarly document consider the Connecticut Yankee fuel cladding steam oxidation studies that were done about a century ago?

MEMBER POWERS: Trust me. This

exhaustive work was written so far after Connecticut Yankee had stainless steel cladding that we would probably overlook that --

MEMBER BALLINGER: Probably --

MEMBER POWERS: -- very active --

MEMBER BALLINGER: -- should do a scholarly addenda.

MEMBER POWERS: Probably would not, in looking at this either.

MEMBER BALLINGER: Because the stainless steel swelled up like a dead fish.

MEMBER POWERS: Well --

MEMBER BALLINGER: -- and everything when it got real hot.

MEMBER POWERS: When they get really hot stainless steel does foam.

MEMBER BALLINGER: Oh, yes.

MEMBER POWERS: And oxidizes in the steam.

MEMBER BALLINGER: This was amazing, what they saw.

MEMBER POWERS: And in fact, when they pulled the upper core internals off TMI, and turned to that, you can actually see how the stainless steel foamed during oxidation. It's pretty well-known. In fact, when you oxidize stainless steel at any time at high temperatures it foams.

The problem I think, Joy, is that by the time you get into these lines, and things like that, your hydrogen to steam ratio is such that you're really not getting any steam, it's pure oxidation.

CHAIR REMPE: You'd think that. But I just want to, I mean, I know --

(Simultaneous speaking)

MEMBER POWERS: But see, you're running steam on stainless steel. What they're actually bringing up here is a mixture of steam and hydrogen. And you can suppress stainless steel oxidation with just a little bit of hydrogen pretty dramatically.

CHAIR REMPE: Okay. Would you not even see any sort, so you don't think you're going to see any sort of changes in the --

MEMBER POWERS: Oh --

CHAIR REMPE: I mean, it --

MEMBER POWERS: The stainless steel very definitely oxidizes. But the oxide is fascinating. And just a little bit of hydrogen, you can, it's really quite dramatic with stainless steel.

CHAIR REMPE: What about thermal connectivity changes for the rest? I mean, things like --

MEMBER POWERS: Well, it depends on where you're asking about thermal conductivity. Thermal conductivity in the steel body itself isn't changed. The surface layer changes.

CHAIR REMPE: Yes. So, would it insulate?

MEMBER POWERS: It will insulate itself somewhat.

CHAIR REMPE: So, expert opinion, without any calculations, dictates you don't think it's worth looking at, anything at all?

MEMBER POWERS: Not the first thing I would look at.

CHAIR REMPE: Okay.

MEMBER POWERS: Not the last thing either. But not the first thing I would look at.

CHAIR REMPE: But we're spending a lot looking about these steam generators too. So they're Inconel way down the line.

MEMBER POWERS: Yes, I think --

CHAIR REMPE: And I'm just wondering when you first come out of the hot leg --

MEMBER POWERS: I think --

CHAIR REMPE: I think we're forgetting something --

MEMBER POWERS: I think we'll eventually get to it. But not at a very stringent pace. In the counter-current it's the mixing in the lower plenum that leads to crucial issues.

CHAIR REMPE: Okay.

DR. SALAY: All right. And again, there's a big difference in the thickness of the boundaries between severe accident conditions, and containment or marginal conditions, or site by site conditions.

The hot leg, three inch. And the surge line is, has 1.5 inch thickness, rather than the steam generator tube which has the secondary site condition is just five hundredths of an inch. So there's a big difference in heat capacity, and how fast it responds.

And again, you've heard this mentioned before, the scenario we're looking at is a high-dry-low. On the high primary pressure, dry secondary and low secondary pressure.

MEMBER STETKAR: So, Mike, on your previous slide, instead of saying there's 1,000 pounds on the secondary side of the tube, do you mean there's zero pounds?

DR. SALAY: Well, before. I mean, this is when it's higher. Yes, it can be very low, yes.

MEMBER STETKAR: So, once you have the low pressure, the pressure we're looking at --

DR. SALAY: Yes. You have a --

MEMBER STETKAR: Okay.

DR. SALAY: Very low. And so you're assuming that there's, for the high, for the primary side that there's no significant leakage. And for the secondary side you are assuming that there is leakage, or some source of pressure loss.

So again, there are two flow patterns that can occur in severe accident flows. And the difference between the two is whether the loop seal has water in it or not. And if you don't have water in the loop seal, you have natural recirculation.

Whereas, if you have water in the loop seal you have to have, the only way the hot gasses from the core have to go through the hot leg as colder gas is returning from the steam generator, are coming back the other way. And so I'll go into these in a little more detail.

CHAIR REMPE: So, go back to that picture, please. What is the status of the expert opinion now on the potential for loop seal clearing in Westinghouse plants and in CE plants? Because I looked through like NUREG-6995, and this draft NUREG. And I wasn't sure what the current expert opinion is on this topic.

DR. SALAY: From my understanding it's that generally it's not expected to clear. But --

CHAIR REMPE: For Westinghouse? Let's start with Westinghouse, and then go to CE, okay.

DR. SALAY: For Westinghouse. And we, again, for CE we didn't look into it in that much detail. And didn't perform multiple analyses to look to try and feel out what would occur.

MEMBER POWERS: Mike, it does clear. Doesn't it just reform?

MEMBER CORRADINI: Why would it reform?

MEMBER POWERS: Condensation.

DR. SALAY: Well, but I thought it was a race on the leakage. That's what I thought where Joy was going. I'm curious about the leakage to the pump seals. Because it's what you're losing versus what you're gaining.

DR. SALAY: You are condensing some. But I don't know if it's enough to --

MEMBER POWERS: If you clear the loop seal you don't blow all the water out. So it doesn't take very much condensation to reform.

MEMBER CORRADINI: Well, but it is a race. So, but you're, just to answer Joy's question. I was going to ask a similar one, is that, it wasn't looked at for CE. And from the Westinghouse standpoint there's nothing new there. So, for all intents and purposes your impression is it won't clear?

DR. SALAY: That was my understanding. I didn't focus on that issue. And I didn't re-read the NUREG-6995 looking at that. So, from my --

DR. LEE: The NUREG-1570, we have look into ANO-2 versus Surry. The ANL-2 have a very shallow loop seal. So if the loop seal clear, so you will establish the full natural circulation by this way.

If you have a pump seal which is very high you will also tend to clear the loop seal too. So it's very plant specific. So you cannot just lump all the same plant, and say they all behave that way. You really need to look at the configuration of the loop seal versus the rest of the piping. They are not all the same.

DR. SALAY: One of the big factor assumptions that affects whether you clear or not is how much leakage you get between the cold leg and the upper head, or the core region.

If, and from my understanding from different tests in the Westinghouse analyses, if you assume there was no leakage it would clear every time.

Whereas, if you assume there was some leakage of gas through this region it would generally not.

CHAIR REMPE: So, just for cleanup in the report, you might make sure that the statements are consistent throughout it about the Westinghouse plants on whatever the topic is.

And I think the report's pretty clear, saying there's just no analysis for the CE plants. And if that's what you want to go with, that's what it is. Now, did you say earlier you did do some analyses of it for the CE plants, as part of this effort, and we haven't seen it?

DR. SALAY: I looked at it, and looked at how I would go about doing analyses, analysis. But didn't actually go through the calculations. It was decided to focus on other --

CHAIR REMPE: Okay. Thank you.

DR. SALAY: Again, for the full loop natural circulation, water's cleared from the loop, core loop seal. And from NUREG-6995 the loop seal clearing is affected by the depth of the pump loop seal and water temperature, reactor core pump seal leakage in rate and elevation, primary side depressurization rates, and downcomer bypass flow. These Westinghouse studies indicated that loop seals are more likely to remain blocked with water.

MEMBER BANERJEE: Excuse me. When, the previous slide that you showed, the counter-current situation with the loop seal. Do we have capability in our "system" to handle a single phase counter-current flow like that?

DR. SALAY: You're saying in the hot leg?

MEMBER BANERJEE: In that hot leg, yes.

DR. SALAY: Yes. I will get to that. That's a little later. That's what this main project is about, the analyses. So --

MEMBER BANERJEE: Right. It's not obvious.

DR. SALAY: No, the codes can't do it. So you need to use CFD to calculate it, or some other method. And then implement two pathways, and have the --

MEMBER BANERJEE: I don't know if the codes can't do it, the comp codes can't do it.

DR. SALAY: Well the codes by themself, they just don't handle the sheer --

MEMBER BANERJEE: Yes. That's for stratification with counter-current. So, okay. So you'll come to that later.

MEMBER SKILLMAN: And, Mike, before you

change. Just as a matter of discipline you showed the pressurizer with a level. I'm assuming it's dead empty.

DR. SALAY: Actually, in some of the analyses it, for some reason the surge line didn't drain. And it actually stayed, yes. The reason the level's in there is just forgotten.

So it's assumed to be empty. But when we did run analyses in some of them it seemed like it drained but somehow filled back up. I don't know if it's from condensation.

MEMBER SKILLMAN: Really.

CHAIR REMPE: Calculations.

MEMBER CORRADINI: That doesn't make any sense.

MEMBER SKILLMAN: Yes. That doesn't -- Because, you know, I would expect based on what happened at TMI, there's a transfer from the pressurizer to that loop seal. And that, and there's a condensing function continuing. And I would think that that is part of the riddle of what keeps that loop seal in place. But I think the pressurizer has to be empty.

DR. SALAY: That's one of the things it was calculating. So I didn't get a chance to explore why that was occurring either. And then the careful monitoring and benchmarking is important to get competence and to gain confidence in whether you're clearing or not.

Because from what I understand from the sensitivity studies in the previous analyses, if you have your nodalization different, it also can affect what exactly occurs.

So, full loop circulation reduces, you don't get a lot of mixing in full loop recirculation, so you get very hot gasses reaching the tubes. And this challenges the tubes very severely. And the expectation is that they will fail.

CHAIR REMPE: So, you said that it's important to do nodalization studies. And that was done and reported in the older NUREG for the SCDAP analyses.

DR. SALAY: Yes.

CHAIR REMPE: Was that done in your recent calculations for MELCOR?

DR. SALAY: No. We didn't focus -- No. We didn't look at it in detail.

CHAIR REMPE: Okay.

DR. SALAY: And you have full recirculation flows, system analyses tools, such as MELCOR, SCDAP, or use those to calculate the flows.

For counter-current natural circulation the system behaves a little bit differently, since it can't go, flow through the whole system.

It has to go up through the, to the steam generator, up to the core, along the hot leg, up the steam generator, back through the steam generator to the inlet plenum, where it can mix. And then the colder gasses go down along the hot leg, and back to the core to heat up.

So there's a lot more mixing that occurs between the hot gasses and colder gasses, than with the full loop natural circulation. So the tubes are not as challenged. And as Sanjoy Banerjee mentioned, it's, or asked about the codes' capability to do this, they can't, they're not set up to handle this counter-current flow.

And so what is typically done is that, they also can't handle the, provide the two temperature distribution. And that affects where it fails, and whether a flaw is likely to be --

MEMBER BANERJEE: You can handle the, I would think you could handle the flows in the steam generator by sort of modeling in more detail the bundle, right?

DR. SALAY: Yes, you can get that.

MEMBER BANERJEE: That behavior in the core.

DR. SALAY: Yes. But the --

MEMBER BANERJEE: But the counter-current you can't at the moment?

DR. SALAY: Well, it's just 1D and friction. And so you put two 1D together, and then you somehow couple it, either with a model, or with a model that's either directly calibrated against experiments, or a model that's calculated as a CFD.

So you run the CFD calculations to give you help, your rate of counter-current flow, and your rate of mixing in the inlet plenum. And you adjust some parameters in the codes to match the CFD results. And the CFD results found scale with temperature. And so --

MEMBER BANERJEE: Because the reason I'm asking is that I suppose the stratification is one of the key phenomena that you have to capture, right, in order to look at the failure of the hot leg.

DR. SALAY: Yes.

MEMBER BANERJEE: So you almost have to do this entirely by CFD essentially, with the boundary conditions being set by the code, right?

DR. SALAY: Yes.

MEMBER BANERJEE: I'm just looking at the methodology.

DR. SALAY: Yes, no. Yes, this is how -- Yes, with CFD you get the, you sort of get a set state sort of result. But you can't do the whole transient, because it's just too much work. And this can do the transient, but can't get the capture of that. So you'd run the CFD to get the counter-current --

MEMBER BANERJEE: So will you in the future go through the methodology as to precisely how you're sort of coupling the, let's say the system codes, the 1D codes to the CFD, and iterating between the --

DR. SALAY: I won't describe it in that much detail. But it's, ultimately you have I think three parameters. You have how much --

MEMBER BANERJEE: If you're going to do this later you don't have to go through it now.

DR. SALAY: I don't think I go through it in that much detail. So I want, so I just do a brief overview saying you use the CFD. But so, CFD provides you this counter-current amount.

You have this Froude based relationship with a parameter that you can tune to match that. And then you, CFD also tells you how much mixing you get, and what. It also tells you how many tubes are in the hot plume that reach.

Because you're not, your plume comes in, it spreads out, and comes into the tube sheet. But it tells, the CFD tells you how many tubes are actually in the hot leg, and what the distribution of temperature entry does to those.

MEMBER BANERJEE: Yes. I think we've seen those before.

DR. SALAY: Yes. You've seen them but --

MEMBER BANERJEE: And they oscillate in the plume, and all sorts of things up --

DR. SALAY: Yes. So there's the three parameters. There's the flow in the amount of mixing. How much --

MEMBER BANERJEE: Yes.

DR. SALAY: Your mass flow in the hot leg. Your mixing fraction and your recirc ratio. The recirc ratio helps mass go through the steam generator tubes, mass flow rate through steam generator tubes relative, divided by the mass going through the hot leg.

MEMBER BANERJEE: I think I can see what you are doing. But in terms of, you're taking a slice, right, in time? Some point in time which is set on some boundary condition --

DR. SALAY: Yes. But it's --

MEMBER BANERJEE: -- from the system codes, right? In other words you have -- You can't do the full calculation with the CFD. So you said something, you said that, backflow conditions, you know --

DR. SALAY: Right. You set the CFD code, and you run the CFD code at different temperatures. And it's shown that these parameters, the mixing fraction and the mass flow, and the recirc ratio, they generally are constant at the different temperature scale.

So you can have confidence that it's, you can use these parameters as inputs into the system code. And that as you, as the accident progresses that it, that was generally the behavior that you'd get from the CFD.

MEMBER BANERJEE: I don't quite see that. Because as the accident progresses, you don't know the phenomena yet, backflow. But I imagine the amount of hot gas coming out of the core will increase, right?

DR. SALAY: Yes.

MEMBER BANERJEE: And it will go and, go through this complex path. And some of it will run back as you've shown. But it's exactly what's going to happen. But the proportions will change with time, right, when it goes back?

DR. SALAY: From my understanding what's been observed and what was done also determined in the earlier analyses was that it generally doesn't change the amount of tubes that are hot, the mixing ratio and mixing fraction, and these parameters.

They generally stay the same, which, and these parameters are sufficient to characterize that they change, they stay the same as you go up in temperature, and the accident progresses.

MEMBER BANERJEE: Interesting. Are you going to show some of those results here?

DR. SALAY: They've been shown before --

(Simultaneous speaking)

MEMBER BANERJEE: But I don't remember them exactly. Was it sort of a fairly robust result?

DR. SALAY: In the CFD analyses they looked at one temperature. And then at high temperatures they looked at snapshots in time. And then the parameters were pretty similar.

MEMBER BANERJEE: Okay.

DR. SALAY: It's a --

CHAIR REMPE: I'd like to see a variation in the hot tube fraction, for example, by sending the -- Basically when you set that hot tube fraction into

MELCOR SCDAP you're making some engineering judgment on what value to pick, based on how much there was oscillation in the CFD analysis, right?

DR. SALAY: Well, there was more fluctuation in Westinghouse than in CE, because in CE the -- Well, in Westinghouse the hot leg comes in at an angle to the tube sheet. Not the tube sheet, to the separator that divides the inlet plenum and the outlet plenum.

So there was a little more fluctuation. Whereas the CE it's perpendicular. And so there's less fluctuation. But yes. In the parameters in the hot tube fraction distribution did, the temperature distribution in entering the tubes did account for some fluctuation in --

MEMBER BANERJEE: How sensitive, excuse me, Joy, are the results to getting this wrong? What are the uncertainties here? Suppose things change by a factor of two, you got the proportions wrong, what would --

DR. SALAY: Well, it makes a difference between -- Yes, because, it makes a difference between which component fails first.

MEMBER BANERJEE: Yes. I think that's the issue, right.

DR. SALAY: And because, actually, initially had we had the parameters -- We did some adjustments of that and actually did change it. I think it was less than a factor of two. And it completely made a difference between the tubes failing first and RCS components failing first. So, it is very sensitive to these flows.

MEMBER CORRADINI: So but, if I might ask Sanjoy's question differently? If you focus only on the mixing of the counter-current flow I can imagine this mixing co-efficient, or whatever you call it, you insert into the system codes is a large uncertainty. But I would expect the initial boundary conditions that get you the high, dry and low are larger uncertainties.

DR. SALAY: Yes. I'm sure there are lots of uncertainties in those conditions. But are you saying in the sequence itself?

MEMBER CORRADINI: Yes.

DR. SALAY: Yes. I mean, of course, but of course the part that I'm looking at is, what if the part that I'm starting off is, what if you're at start off and get these conditions? So, I'm in --

MEMBER CORRADINI: Yes. I understand, Mike. I'm completely with you. I just wanted to put it in context. I guess I actually wanted to go back to something that Dana asked. And I'm sure you didn't consider it, but I'm curious about that observation. Because I guess I'm not as an expert as Dana is in oxidation.

Is that, if I started having severe oxidation of the upper core internals, would that change the flow path and the flow resistance, that I actually could alter this flow path?

Dana made this point about steel oxidation observation. Would it affect the flow path or the flow circulation path that you're assuming in your calculation?

DR. SALAY: Yes. I don't know what, how much of a size change there would be if it oxidizes.

MEMBER CORRADINI: Okay. All right.

MEMBER POWERS: I surely don't know, Mike. But what I have seen, certainly from the TMI upper grid, locally you change the flow pathway. But there's an awful lot of room up in the upper plenum for it to reform this kind of picture that Mike's portrayed out here.

When you transfer to the boiler I think that you might have a more dramatic effect from swelling and foaming of the stainless steel oxides.

MEMBER CORRADINI: Okay.

MEMBER POWERS: But I assuredly don't

know.

MEMBER CORRADINI: I'm just trying to understand. Because I think Sanjoy's question, which is starting down this path is, this is given that you kind of have to flip between a CFD calculation, and then tune the system code calculation. I can understand where this is a major uncertainty. That's why I was just trying to ask these questions.

MEMBER POWERS: Well, I don't know that that's the major uncertainty. But it's certainly a challenging part of the calculation. There's no question of that.

MEMBER CORRADINI: Well, given, you know -- Yes, okay.

MEMBER BANERJEE: But you have the sensitivities done some? I mean, you've done this sort of parametrically?

DR. SALAY: Well, we kind of look to, the parametrics were done in the previous analyses. We didn't do many sensitivities.

MEMBER BANERJEE: But it was done previously, yes? We wrote a letter on this, right. Bill Shack was the -- I'm trying to remember way back in history. I think Shack was the chairman, wasn't he, of the subcommittee at that time? Do you remember, Dana? I was there attending innocuously, but not the chair.

MEMBER POWERS: You're never innocuous.

MEMBER BANERJEE: But Shack I think was the chairman.

MEMBER POWERS: He truly is never innocuous.

CHAIR REMPE: CFD sensitivities right?

MEMBER BANERJEE: Yes.

CHAIR REMPE: There were some CFD CE sensitivities done previously? No, right?

DR. SALAY: I'm sure Chris did, looked at a few different things.

CHAIR REMPE: But it's not documented previously. It was only done for this effort, right?

DR. SALAY: Yes, I mean, it --

CHAIR REMPE: And we don't have those.

MEMBER CORRADINI: We have all the ACRS letters that were given to us.

MEMBER BANERJEE: Oh sure, yes, we have the ACRS letters. I just don't recall. But I guess this timing was the issue always, right, the uncertainty and the timing?

DR. SALAY: Yes.

MEMBER BANERJEE: Because in one case if

you got massive failure of the steam generator tubes it led to a very different situation from some RCS component failing --

DR. SALAY: Yes.

MEMBER BANERJEE: -- within containment, which was the hot leg really.

DR. SALAY: But if one tube fails, again, the system pressure is still high.

MEMBER BANERJEE: Yes. That's different, yes. It's one or two tubes. It doesn't --

MEMBER CORRADINI: But I think, Sanjoy, to answer your question, it was the October 22nd letter in 2009 --

MEMBER BANERJEE: Right.

MEMBER CORRADINI: -- where it was our final, our most recent set of suggestions for this program, going to a user need only basis.

MEMBER BANERJEE: Okay. I'll re-read that letter.

DR. SALAY: And again, for Babcock & Wilcox plants, from the pervious studies it was, I'm not sure they were CFD, but they came to the conclusion that vigorous natural circulation flows wouldn't be expected in these plants, because of the way they're configured. And so, they were not part of the severe accident induced failure studies.

MEMBER SKILLMAN: I think there's another detail. And that is that the B&W claims had that vent valve in the upper plenum to actually allow the hot leg and the cold leg to equalize, so you don't get a loop fill.

MEMBER CORRADINI: Say that again, Dick? I didn't appreciate that.

MEMBER SKILLMAN: The B&W plants have vent valves. There are four large 14 inch gravity operated valves that equalize between the downcomer and the inner plenum. That is a very key design feature of the B&W fleet, of all the plants that were built.

MEMBER CORRADINI: So that does what again? I'm sorry.

MEMBER SKILLMAN: It ensures that your downcomer is always available. If you go back to the --

MEMBER CORRADINI: Well, I'm looking at Slide 14 to understand what you're saying.

MEMBER SKILLMAN: Okay. If you see right in the, see all those holes, all those round things at the top of the upper internals?

MEMBER CORRADINI: Yes.

MEMBER SKILLMAN: Four of those are 14

inch diameter check valves, gravity operated. And they assure that if there's delta P between the inner and outer portion of the steel on the plenum, that's the core support assembly, that those equalize.

MEMBER CORRADINI: So essentially if there is too much pressure on the hot side it pushes into the cold side.

MEMBER SKILLMAN: That's correct.

MEMBER CORRADINI: Okay. Yes.

DR. SALAY: All right. Thermal-hydraulic analyses for this project used both Westinghouse and Combustion Engineering TH analyses as input to the tube failure calculator, and the finite element modeling.

The Westinghouse calculations as you're, have been, were performed fort the steam generator action plan before. And it's documented in NUREG-6995. The TH analyses at the time did not receive the same level of attention.

So that's why we were looking at it, and performing some additional TH analyses for CE, Combustion Engineering plants under this project. And we're using those CFD and system codes. CFD is used to provide multiple parameters for the system code.

And it can also be used, the temperature distribution can be used to calculate the flaws. The CFD approach is validated using Westinghouse 1/7 scale tests. And here you see some diagrams of the tests. These tests are reduced scale, electrically heated core vessel and two steam generators.

It focused on temperature measurements and transport of heat. You can see in the different diagrams where temperature readings were taken. These tests demonstrated the counter-current flow path. They were focused on tube integrity, but knowing that the flow pattern provides lots of insights for that.

Scaling studies were, many scaling studies were performed to demonstrate that these were applicable to full scale plants. And information from these experiments have been used for system codes, and to study station blackout scenarios. The analyses, the CFD analyses for these were done around 2001.

And the CFD analyses started with the 1/7 scale. We did the 1/7 scale for the first scale, then went to the Westinghouse, and then went to in this analysis CE plants. And several sensitivities were performed on heat transfer, surge line orientation, hydrogen content and potential tube leakage rates.

MEMBER POWERS: These particular tasks, of course, were revolutionary when they were done. And insightful, and just really spectacular result. And they are a true demonstration of the well-known adage that no good deed goes unpunished.

Because they raised a large number of questions that the tests never were intended to address. Does the equipment for doing these tests still exist, do you know?

DR. SALAY: I have no idea. They have it. But the Europeans were planning on performing some similar experiments with the hot, much more resolution in, with their temperature measurements using newer methods. I think some of that's been scaled back, and they're looking for smaller PHD --

MEMBER CORRADINI: Mike, there's just --

MEMBER POWERS: Yes. The tests were done. They revealed things that people had speculated about. And of course it was dramatic. A complete change of thinking when they occurred. And of course now you, then you knew more. And so you started asking more questions.

And the test got criticized for a lot of things that they never intended to address, because they didn't know about them at the time. And one wonders if such experimentation ought not be continued.

DR. LEE: Actually, Dana, remember a few

years ago we went with you to PSI. And they proposed that they going to do some testing when you have six it varies. The inlet coming into the plenum, the inlet coming on stimulator, the hot leg coming in pipe. You can varies by the closer to the sheet.

MEMBER STETKAR: Yes, tube sheet.

DR. SALAY: Right.

DR. LEE: Or further away. But they found out that the cost of doing such testing becomes prohibitively expensive for PSI. So they had abandoned that experiment. I thought they will propose it. But they have abandoned it. So, I think we will want to bring it back up to the French. And maybe they can --

MEMBER STETKAR: I was going to say, AREVA's not, or EEF or AREVA's not doing anything in this area.

DR. LEE: I don't know. But the thing is that it disappeared from the radar screen. And now they're proposing something. And recently I think Chris Boyd and I look at it. And they said it's more like a force flow thing.

So it was really not a natural circulation. So I don't quite know what you can get out of it. So, I think the purpose is that we bring it back to the attention of someone and --

MEMBER POWERS: Well, the reason I bring it up is --

DR. LEE: -- can really do something about it.

MEMBER POWERS: -- the, you know, we're about to embark on discussion of the recric program. There's a hole. And I'm wondering if this isn't something that we should, maybe with Mike's help we could highlight --

DR. LEE: And Chris.

MEMBER POWERS: And absolutely with Chris. To highlight as a need found to understand, especially as we make bigger use of CFD. You need the ability to calibrate, and to validate that CFD capabilities in these fairly complex situations.

And at the same time get information on how this rather subtle points in these design, in fact, behave during accidents might be very worthwhile for us to consider.

MEMBER BANERJEE: Are there international programs directed this way?

DR. LEE: I think when the European Union call for proposals, the different organization can suggest what they can look into. And this is one of the areas they can look into.

Because this is a, you know, steam tube rupture is a low probability, but a high risk type of event. So the Europeans are always interested in reducing the source term into alignment, especially for the EDF or the plants of BWR.

MEMBER BANERJEE: But there is no program in place right --

DR. LEE: Right now we don't have seen one. But a few years ago, yes, I was thinking the need to propose such a program. And we think this will be very useful because that will allow for the better benchmarking for the CFD. And then we can use that one in terms --

MEMBER BANERJEE: Were they proposing it for what was a standup facility? Or was it completely different?

DR. LEE: I don't know what the name of it is. But it is the same place. So they would have a plant there. But in the bottom there's more space so they can adjust that inlet coming in, relatively speaking of, from test to test. So if they can look at making some location of it in different country. The angle coming in, and the distance from the tube plate.

MEMBER BANERJEE: I can see that they have facilities which could be used.

DR. LEE: But it become cost. Cost is a problem. Now, with Switzerland planning to phase out nuclear power, right. So, PSI is having problems getting resources to fund such a project, or to take on such project. So it is better to bring in the others. We should do that.

MEMBER BANERJEE: It's a good point. These were very nice experiments. I remember them.

MEMBER POWERS: They were very revealing. And they changed, had a qualitative impact on our thinking, which is just bad. I mean, it's bad because it proves how dumb you were before, and whatnot. But they also had the impact of, as we can see now, raising lots of questions that we never had before.

CHAIR REMPE: So maybe before this comes to the full committee it would be good to ask someone from Westinghouse if the facility still exists also? Or are you thinking just let's go to Europe, Dana?

MEMBER POWERS: Well, you know, I ask about the facility. My own view is, old facilities usually are well designed for old experiments.

And now that we have such marvelous computation capabilities we ought to be able to design a better experiment. Of course, when you design a better experiment it oftentimes becomes cost prohibitive. It might not see some sort of a balancing. I mean, I'm not the one to decide whether old, reuse old experiments, or --

CHAIR REMPE: But it would be nice to know if it even exists still, or if it's available.

MEMBER POWERS: I had spent, I will admit, a little while looking in the details of that experiment. And I think we could design a better experiment now. I meant one of us be able to design a better experiment.

DR. SALAY: All right. And this shows some of the results of scale-up from the 1/7 scale to full scale. And this is one scale, SF6 to full scale severe accident conditions for generator steam.

On the top right you see the temperature distribution. Chris, scale and the tubes. Okay. The temperature distribution for the two analyses. And this shows the scale from zero. Well, it doesn't go to one. But zero represents the cold side of the steam generator tubes.

And one represents the temperature entry in the hot leg. So it's normalized, as it's based on the temperatures coming in. I mean, that's how you can apply it to different temperatures. So, as the similar solutions were obtained for both analyses, when heat transfer rates are scaled the full scale showed a slightly lower temperature distribution entering the tubes.

Yes, the heat transfer rate under severe accident conditions are different, however. And the geometry isn't similar. And they were identified as an area that needed further looking into. The recirc ratio, the mass flow in the tubes, the mass flow of the hot leg is increased a little bit when they went from 1/7 scale to full scale.

And the mixing fraction, the amount of the, I think the, I can't remember, the hot leg that gets mixed with gas returning from the steam generator tubes went up also a little bit. Oops, wrong way.

And now I'm moving to the CSGTR behavior. And it differs a little bit from Westinghouse plants. This is primarily because there's less opportunity for mixing for hot gas before it reaches the steam generator tube inlet.

There are a few different reasons. There's a shorter lower hot leg length to diameter ratio. So there's less opportunity for mixing there. And some CE plants have shallow inlet plena.

MEMBER BANERJEE: So, did you have to

repeat this CFD calculations with these shallow plena?

DR. SALAY: Yes, yes. The CFD calculations were done specifically --

(Simultaneous speaking)

MEMBER BANERJEE: They're very geometry sensitive?

DR. SALAY: Yes, you have to do it specifically. And a third aspect probably that affects the mixing is that the hot leg comes in normal to the sheet, separating the two plena.

MEMBER BANERJEE: So we haven't seen these, even these calculations, right, the CFD ones? Have we?

DR. SALAY: I think in a few rounds I'm sure they were presented in --

DR. LEE: I thought three years ago I saw it.

MEMBER BANERJEE: Maybe in --

CHAIR REMPE: But not documented I think is the issue here.

MEMBER BANERJEE: I don't always attend these, so I must have missed it. I should have known to see it.

CHAIR REMPE: There's a report. It mentions that the replacement steam generators for the

Westinghouse plants are becoming more like CE plants. And so, if you go through the next couple of slides could you kind of characterize?

I mean, you've talked about this separator plate that's characteristic of the Westinghouse plants. But what are the replacements looking like now?

DR. SALAY: I don't know.

CHAIR REMPE: Oh, okay.

(Simultaneous speaking)

MEMBER BANERJEE: -- geometry sensitive, right.

CHAIR REMPE: Yes, but --

(Simultaneous speaking)

CHAIR REMPE: So we're getting into what may be the --

DR. BOYD: This is Chris -- Is this on? This is Chris Boyd from research. I had that same question. And we were concerned because the only one we really had back ten years ago was Calvert Cliffs because they were just doing a change. And the inlet plenum was significantly different.

So I got the impression that we can do whatever we want, I guess, in designing inlet plena for the replacements. So we did ask for a survey of

Westinghouse replacements. And we asked for maybe ten or 12. We ended up getting about four.

And in the Westinghouse space the ones that we saw were nearly identical, you know, if you held them at arm's length maybe a half inch difference. But there wasn't any significant variations like we saw with the CE plant.

So, with, so the impression that the Westinghouse's are different and flatter, and more like the CE is not anything that we found.

CHAIR REMPE: Okay.

DR. BOYD: That was a hypothetical concern. Because apparently replacement steam generator design can take some liberties on the original design.

CHAIR REMPE: Okay. So perhaps maybe a cleanup of the report. That ought to be --

DR. BOYD: Right.

CHAIR REMPE: -- an item. Thank you.

DR. SALAY: So, with these differences the tubes see hotter temperatures relative to the hot leg, than they do in Westinghouse plants. And because of this it's, under certain conditions the tube, even unflawed tubes using standard failure models could be predicted to rupture before hot legs.

And unlike before the flawed tube, a lot of tubes, if you're looking at unflawed tubes would hit the failure point at about the same time. And so, you could get a big failure, where potentially you could depressurize enough to not, to prevent subsequent RCS failure.

And here shows a CE inlet plenum. This was actually a replacement for Calvert Cliffs, compared to Westinghouse. If you look at the plume it has about, it's about that wide there. And you only have 1.5 length to diameter ratios before it hits the tube sheet.

Whereas, in the Westinghouse you have about four and a half-length to diameter ratios, which provides much more opportunity for mixing. So here, some CFD results for both the Westinghouse steam generator and CE steam generator. And you're looking at the normalized temperature fields.

Again, zero is the cold side of the steam generator, and one, the hottest temperature is the temperature that the hot leg sees. And you can see the temperature entering the tube sheet, it's substantially hotter for CE than for Westinghouse.

And so here, this figure should, details a bunch of things to look at and consider, the pressurizer draining, your natural circulation bypass flow, your oxidation rate core blockage, pump seal leakage, your nodalization, your downcomer clearance.

That's also necessary for full loop natural circulation. Your lower head can't be full of water. Because otherwise it blocks the flow. Of course, loop seal clearing, that's been mentioned. How much mixing you get in the hot leg.

How much mixing you get in the inlet plenum. And tube heat transfer and secondary flows, and tube sheet nodalization. So for MELCOR and CE commercial unit calculation it's Sandia National Labs generated Combustion Engineering deck. That was based on previous RELAP and MELCOR decks.

Go a little bit over. The addition of the hottest tube. Estimate of the tube temperature profile. I don't actually talk about the secondary relief valve opening criteria. But there are a few different criteria.

MEMBER BANERJEE: The reason you moved from RELAP, away from RELAP was what?

DR. LEE: Because we stopped the development of SCDAP Rev 5 and INL.

MEMBER BANERJEE: Oh, so it was --

DR. LEE: -- we put out completely INL and the SCDAP Rev development.

CHAIR REMPE: Well actually, the NUREG that has the SCDAP calculations was done by ISL, not INL. And so --

DR. LEE: But previously INL did a very sensitive --

CHAIR REMPE: Right.

DR. LEE: -- SCDAP Rev 5.

CHAIR REMPE: Yes, but --

DR. LEE: Looking at everything can come under the sun on heat transfer, everything. So is a very comprehensive study at that time.

MEMBER BANERJEE: So, the reason you pulled out was --

DR. LEE: Because Bechtel Company took over the running of INL. And there is conflict of interest problem. And the General Counsel have us pull out.

MEMBER BANERJEE: Because that's what I was wondering, why you would do this. Okay. Maybe these are political reasons --

DR. SALAY: And there is also, I have nothing.

CHAIR REMPE: Well, the report said it was because that you wanted to do the radiation releases too, is the official document statement is why you switched to MELCOR.

DR. SALAY: Yes. We're moving to MELCOR. And we did want to calculate fission particle releases. And if we can do it in the future that's, and the opportunity is there, the deck's there, and --

MEMBER BANERJEE: And MELCOR was adapted to do these sort of detailed thermal-hydraulic calculations in the RCS and secondary side? As you showed in the previous slide --

DR. SALAY: Yes.

MEMBER BANERJEE: -- that's a --

DR. SALAY: Yes. It uses the CFD analyses as input to -- So you have run a few cases to calibrate it so that you're matching the results.

MEMBER BANERJEE: But it can capture those complicated phenomena like loop seal blocking, clearing --

DR. SALAY: Yes. Same thing.

MEMBER BANERJEE: Yes. Okay.

DR. LEE: After we pull out from SCDAP Rev 5 we stopped using it for analysis. It took us about ten years or so to bring in parity between SCDAP Rev 5 and MELCOR incidents and analysis.

MEMBER BANERJEE: Okay. Thank you.

DR. LEE: A long time.

MEMBER BANERJEE: Yes. It's not obvious that it --

DR. LEE: Yes.

CHAIR REMPE: Before you --

DR. LEE: It's a long history. Sorry.

CHAIR REMPE: It's okay. Everybody's -- I'd like to talk a little bit more about the hottest tube implementation that was done for MELCOR. When it was done for the SCDAP stuff I guess they relied on --

You have a prediction for the hottest tube. And I guess a long time ago they had a figure of 17 in NUREG-1922, where they tried to say, okay, this is the hottest tube. And we're going to apply it.

And it's not like this hot tube fraction that's coming from the CFD stuff. They apply it to a certain number of tubes and say, okay, so the one tube failing, there's ten percent, or whatever they want to say percent. And it's I think expert opinion.

DR. SALAY: Yes. It was expert opinion. A few different people said, well we could detect, and it could be 20 in --

CHAIR REMPE: Okay. What did you do for the MELCOR analysis.

DR. SALAY: Twenty.

CHAIR REMPE: Okay. And was that the

same? I mean, that wasn't the same as what was done for the SCDAP analysis a long time ago.

DR. SALAY: In the SCDAP analysis it was done for Westinghouse. Westinghouse the tubes don't get hot enough to -- So that unflawed with the even approach --

CHAIR REMPE: Okay.

DR. SALAY: -- in failure temperature.

CHAIR REMPE: Okay. So this might be an important expert opinion. And I believe it was the NRC experts --

DR. SALAY: YEs.

CHAIR REMPE: -- that came up with this 20 percent. But it's not really documented that well, I thought, in the report. I mean, there's a couple of sentences, and that's about it.

DR. SALAY: Yes. I just heard people talking about it too. I haven't --

CHAIR REMPE: Okay.

DR. SALAY: Again, for the addition of the hottest tube you need to capture the tube temperature that's in, that's shown in the CFD calculations. So it's, you use this to get your peak temperature. And you have your average temperature.

And you can use that to scale your spatial distribution, so this provides sort of a time dependent anchor to the spatial distribution. And a few methods were tried, just used post processing method. But there were doubts that it was, I mean, just calculate based on the results.

And there were doubts that this adequately captured the behavior. So then tried to match the method to the one that was used for SCDAP RELAP. And it had some stability issues.

And then finally went with this method where you actually removed heat at the inlet, and then added the same amount of heat back after, when it came, where the flow comes into that outlet plenum. And so, you're not adding energy, but you're changing the temperature of the tube, so, to match the CFD. And so --

MEMBER CORRADINI: So this was an, that's the one you settled on is an arbitrary addition of a +Q in your cartoon?

DR. SALAY: Well, not -- This was a method that was used previously. And it's what matches with the CFD, yes. And so, you need some way to do the increase of temperature. And you can't really capture it other than --

MEMBER CORRADINI: I just want to make

sure I understand. So you settled on the final, the heat addition method to --

DR. SALAY: Yes.

MEMBER CORRADINI: Okay. And then to get back to Joy's question, by judgment you added this over a group of how many tubes?

DR. SALAY: Well, there's only one tube that you calculate. But you're assuming that it represents 20 when it fails. And --

MEMBER CORRADINI: Okay. Okay.

DR. SALAY: I think it doesn't really make that much difference once it's big enough to depressurize in the system.

CHAIR REMPE: How many smaller -- Like if you went to ten would you have a big enough hole?

DR. SALAY: I didn't --

CHAIR REMPE: There's no sensitivities on it then?

DR. SALAY: Yes. And then for approximation of the tube temperature field, so you then, from the previous addition of hottest tube you have these two temperatures. And you can apply the temperature distribution from CFD.

There was an approximate, it was approximately by parabolic shape that goes all the way to the top of the tube sheet. And so you can come up with a time variant temperature distribution that you can then test against, test your flaws against to see if --

MEMBER BANERJEE: The situation if I remember was the plume waves around, right?

DR. SALAY: It waves around more in Westinghouse. But then, this is sort of meant to be bounding a little bit. So just tried a simple approximation that can be used to calculate.

MEMBER BANERJEE: But the --

DR. SALAY: What fraction.

MEMBER BANERJEE: -- material is exposed to --

DR. SALAY: It varies.

MEMBER BANERJEE: Yes. And fluctuating temperature field.

DR. SALAY: And when this was done the temperature distribution was accounted for the fluctuations and things within 19, NUREG-1922 I think. And so, I know there were discussions back then about what's appropriate to use the peak at one point, or should you average it to account, just vary, have it sort of a time average value at any one point. And I think they did the latter.

But from what my understanding is that the CE analyses, it didn't fluctuate to the same extent. Because part of that fluctuation was that it was coming in at an angle.

MEMBER BANERJEE: Okay.

DR. SALAY: So, and here are some results. And this is for no auxiliary feedwater. The turbine driven auxiliary feedwater is assumed to fail. And again, the small secondary leak --

MEMBER CORRADINI: So, can I ask a question about the pressure plot, so I just understand it? So, I interpret that up to about 14,000 seconds it's the PORV operating, and then you poop out of the PORV. And then --

DR. SALAY: And then you're going to get --

MEMBER CORRADINI: -- the PORV starts operating.

DR. SALAY: Yes. That's correct.

MEMBER CORRADINI: Okay. So we're sitting there for essentially, 7,000, so essentially six hours, unless I missed -- From about 14,000 seconds to about 23,000 seconds. And I'm just basically cycling the SRV.

DR. SALAY: Yes.

MEMBER CORRADINI: Okay. And the temperatures out of the SRV would be essentially the temperatures coming out of the hot leg, yes?

DR. SALAY: Yes, they're hot.

MEMBER CORRADINI: Okay. So what side group is worried about, or at lease evaluating the fact that I would essentially do the same thing here that I would do in a BWR, and essentially fail this thing open, just due to the operation of this valve at extremely high temperatures? Because if I'm going to start worrying about uncertainties, that's the one I would look at.

DR. LEE: Mike, there are on data on SRV failure under severe accident conditions.

MEMBER CORRADINI: Okay.

DR. LEE: That I know of.

MEMBER CORRADINI: Okay.

DR. LEE: United States not at war.

MEMBER CORRADINI: Say it again, Richard, I'm sorry.

DR. LEE: There are no experiment data on SRV or PORV behavior under severe accident condition.

MEMBER CORRADINI: Okay.

DR. LEE: These are called assumption. How many times you cycle it, right. Mike, am I correct?

MEMBER STETKAR: Mike, in principle --

MEMBER CORRADINI: That is correct.

MEMBER STETKAR: -- it ought to be addressed in the PRA model for, you know, the Level 2 PRA model, in principle.

MEMBER CORRADINI: Okay. The only reason I asked --

MEMBER STETKAR: Whether it is or not is a different topic. But in principle that's where that would come in.

MEMBER POWERS: Yes. The difficulty is that you don't have data --

MEMBER STETKAR: Right.

MEMBER POWERS: -- for severe accident conditions.

MEMBER STETKAR: Right.

MEMBER CORRADINI: So, that's fine. I figured that was the case. I just wanted to make, get it on the record. The second question is I guess maybe to Chris Boyd.

And I'm assuming CFD can't do this. But I'm very curious about, when the SRV opens does it change the flow pattern inside the counter-current flow?

MEMBER STETKAR: It does.

MEMBER CORRADINI: I assume it does.

DR. BOYD: Hi, Mike. This is Chris Boyd. Yes, of course that would completely stop the counter-current flow. And we looked at that. And then, once it shuts it reestablishes fairly quickly.

All of the metal materials are sort of pre-heated, and hot gasses sort of seem to quickly re-establish that flow pattern. But there is a delay there, and a little snag, and everything, when you open the PORVs.

MEMBER CORRADINI: Okay. But it reestablishes quickly?

DR. BOYD: Relatively quickly, yes, it reestablishes.

MEMBER CORRADINI: Okay. Thank you.

DR. SALAY: All right. So again, on the top right you see the pressure plot. And it's the hot legs, the two second areas, and the containment's on there at the bottom. And the lower plot you see the secondary water level, and then in the steam generators.

So, when everything shuts down you lose power. The cooling from the steam generators continue to cool the system. And this brings the pressure down. But you're boiling off. And so, this keeps the secondary pressure pretty level.

At some point later you've lost enough inventory that you can't keep up with the heat production, and the pressure, and the primary pressure starts to increase up to the PRV set point. And I'll switch to the second part where -- So it stays there.

The PRV's cycling, and eventually going to cover the core. Start to heat up. As Mike Corradini mentioned, the batteries run out, and they switch to the SRV. Eventually you hit the point where the tubes fail.

And this depressurizes the primary to the safe, as the pressurizer. So the pressure's equilibrate. And it wasn't the high enough pressure to have the secondary SRVs go. And the temperature --

MEMBER STETKAR: First, that would be different if the secondary SRVs were open.

DR. SALAY: Yes.

MEMBER STETKAR: Okay.

DR. SALAY: Yes. I mean, I did some of those analyses too. And so then the temperature continues to rise. And the pressure's high enough to end up with a failure of the hot legs. And the system depressurizes.

And so, this is, the rest of the results were calculated, you know, for all this. It's being done primarily in the calculator. It was also done in MELCOR as a screening calc. And this shows some of the results.

The top one is, the top right figure is the creep rupture index for different components, the steam generator tubes, unflawed steam generator tubes, hot leg As, the surge line, and zero. This is an index that goes from zero to one, where one represents damage.

When this index reaches one the component's considered to have failed. So then it, the loop B hottest tubes fail first. And then followed sometime after by the loop B hot leg. On the lower right you see a similar block.

But for different steam generator tubes with different creep, not creep rupture, it sees different -- I can't remember what it's called. Different stress multipliers, which represent different flaws in tubes. And so --

MEMBER CORRADINI: So just, Mike, just so I understand, the lower right is the same calculation. But now you've manipulated the stress risers, because you have a different damaged, or pre-damaged, or pre-flawed tube?

DR. SALAY: Yes. It's the exact same

calc. And it was actually, it was evaluated, it evaluated all these different stress risers --

MEMBER CORRADINI: Okay.

DR. SALAY: -- just to get a little map in case, to see what happens.

MEMBER CORRADINI: Okay. So same thermal-hydraulic conditions, but different structural boundaries, or structural initial conditions of the tubes?

DR. SALAY: Well, these -- Yes. Different assumed failure of tube. Different assumed tube conditions, or tube flaws.

MEMBER CORRADINI: Okay. Thank you.

DR. SALAY: And yes. You know, it was the exact same calculation. It calculated all these things at the same time.

MEMBER CORRADINI: Thank you. Thank you.

DR. SALAY: All right. And then some variations were run. Here you see in Slide 31 you see the same behavior on if you open, if the operator opens the valves, the secondary deforms when the accident first occurs. And so you're down to atmospheric in secondary pretty rapidly.

And this also, blowdown cools the, cools and lowers the primary pressure. But it comes back up.

And it fails significantly earlier. But also, due to long term station blackout, where the turbine driven auxiliary feedwater operates, it's considered operating until the battery dies.

And then that's also shown in the bottom right. And along with the short term station blackout, which is also above. And if you look at it, it looks, they look pretty similar, but just time shifted. Although the lower the K power it seems that it's, heats up a little slower. But otherwise it's pretty similar behavior.

And some conclusions are that Combustion Engineering's steam generator tube rupture brings consequential steam generator tube rupture behavior differs a little bit from Westinghouse plants because of less mixing. Because this Combustion Engineering steam generator tubes are thoroughly stressed relative to hot legs, in comparison to Westinghouse plants.

And so a greater likelihood the tubes will fails relative, earlier, relative to Westinghouse plants. And despite I showed, the fact that I showed a calculation where the tubes failed first in most of the analyses, the RCS components failed first.

MEMBER CORRADINI: So, may I ask a

question here, Mike?

DR. SALAY: Sure.

MEMBER CORRADINI: As you connect back up to the draft NUREG. And again, I'm jumping to an end state. But I just want to unwrap it. Right underneath the table in the summary for the draft NUREG, the statement is given that for the CE plants the conditional probability is .22 and .31, versus essentially a factor of ten lower for the Westinghouse plant.

Is that directly a function of the mixing that you were speaking of, and the difference in mixing? Are there other things that later we're going to learn about that contribute to that?

DR. SALAY: Well, the mixing is, really affects how hot a temperature your tubes see.

MEMBER CORRADINI: Right.

DR. SALAY: So if --

MEMBER CORRADINI: I understand.

DR. SALAY: So if you get a lot of mixing your tubes don't see that much, that high of a temperature.

MEMBER CORRADINI: Okay. But to say, I want to say it very dramatically. The factor of ten or 20 difference in the summarized probability is primarily a function of this mixing phenomena?

MEMBER STETKAR: Or is it because the hot leg fails earlier?

DR. SALAY: Well, it's sort of the same thing. Because it's, you're saying the same thing as one. Both heat up. But if you get a lot of mixing one heats up faster than the other. And so, yes. I'd say mixing is the --

MEMBER STETKAR: But if you had more Westinghouse scenarios, where you had a completely depressurized steam generator empty at time T zero, open secondary relief valve. Would those numerical factors change your conditional probabilities? They would. Because the hot leg wouldn't fail then. Right?

DR. SALAY: For Westinghouse.

MEMBER STETKAR: For Westinghouse.

DR. SALAY: That's --

MEMBER CORRADINI: So that, so I guess John has unwrapped it more than I have. I assumed mixing was the dominant difference between the two.

MEMBER STETKAR: Well, I think, as Mike said, I'm not a thermal-hydraulics guy. Mike Corradini, you know that. It's --

MEMBER CORRADINI: And you do that bloody electrical engineer on the Board.

MEMBER STETKAR: Yes. Well, no.

DR. SALAY: Yes, but at Westinghouse you need appreciable flaws for it to, for the tube to fail before the RCS.

MEMBER CORRADINI: But I guess what I'm asking, from a flaw standpoint, I can't imagine one steam generator's flaw distribution is different than another. But maybe I'm missing that. That's what I was trying to get at.

MEMBER STETKAR: Yes. And my question was, I mean, your statement on, wherever it is, a couple of slides back, where you did the case where the operator opened the secondary relief valves actively.

The conclusion was much faster, stronger depressurization, earlier tube rupture, which would mean the conditional probability of the tube rupture is much higher in those scenarios, right? Because the hot leg doesn't get a chance to fail. Is that correct? Or am I misinterpreting something.

DR. SALAY: That's again with, that's with these, the scoping calculations, with, where I'm assuming that the 20 fails. These were analyzed in more details with final analyses, and the flaw calculator. So these were just screening calcs. And

I don't know if this specific scenario was --

MEMBER STETKAR: Let me try something. Let me see if I can ask it differently. Suppose the world worked as if every single scenario matched this condition, every single for a Westinghouse plant, such that the secondary side of the steam generator was completely open to the atmosphere, and it was dry.

Would the conditional probabilities that Mike Corradini referred to up front for Westinghouse plants, that factor of ten lower than a CE plant, would those conditional probabilities be the same if the Westinghouse world always worked this way?

DR. SALAY: If the Westinghouse --

MEMBER STETKAR: If every scenario always had, this is Westinghouse plant now, always had the secondary side of the steam generator completely depressurized and dry.

DR. SALAY: Yes.

MEMBER STETKAR: Would those, would that factor of ten lower conditional probability for the Westinghouse plants apply? Or if you want to think of it an absolute sense, would that ten to the minus two conditional probability apply? Or would it be higher?

DR. COYNE: I'm not sure that that's a fair question for you. Because it's going into what the calculator's doing, and the correlations you used for creep failure of tubes. I think a fairer question to you is --

DR. SALAY: And it's also Westinghouse.

DR. COYNE: Right. If the quick depressurization of the secondary side appreciably changes the ratio of, or the relative temperature distribution that the steam generator tubes are seeing.

In other words, does opening the secondary relief valve to rapidly depressurize the secondary appreciably change the mixing coefficients that you would have used for the system level code analysis?

DR. SALAY: What I think is that the Westinghouse -- Yes, and I didn't analyze Westinghouse. So I'm not quite sure how it is. But this is, in the Westinghouse you're getting one or two tubes, because you needed the flaws. You're not getting this depressurization from when the tubes fail. So if the tubes fail then, or if flawed they fail earlier. So --

MEMBER CORRADINI: Mike, I guess, I think I understand what John's asking. Let me try another way of asking it. And I don't know who clarified it. But I think the clarification made sense.

DR. SALAY: Yes.

MEMBER CORRADINI: If all the initial and boundary conditions were the same between the Westinghouse and the CE plant, when it's depressurized all that timing is the same, all the initial positions are the same.

Is it the mix, is it the geometrical differences that cause a mixing difference, that evolved you to a difference in those probabilities? Or is there other things we need to hear about later that add to it? That's what I'm trying to get at.

DR. SALAY: Well, to my understanding it's the mixing that affects the temperature difference between the two. Then you have the material properties for the failure. And it's those two in combination are what decides what fails first.

MEMBER CORRADINI: Okay. Thank you.

DR. COYNE: And we're going to get into more of this as we go through the day. But our presumption has been you're not going to see relative temperatures as high as CE in the Westinghouse plant with those kind of conditions you're describing.

But the relative temperatures for Westinghouse tend to be, and I'll look to Chris Boyd or even Ali Azarm to bail me out here. More in the .5 range, whereas you're seeing relative temperatures more in the .8 or .9 range for the CE, given the mixing coefficients.

DR. SALAY: Yes. Okay. If that's what you're asking, you have the relative temperatures are based on the flow pattern. Yes. So you have the secondary and, I mean, you're already assuming the low pressure on the primary.

So it, on the secondary. So your, the analyses already, this -- What was I going to say? Yes. What he just said. Yes. The temperatures, the normalized temperatures are still going to be substantially lower for Westinghouse.

MEMBER CORRADINI: But this is all driven again by geometry --

DR. SALAY: By the geometry, yes.

Yes. Well, yes, the, in the geometry and the length to diameter ratio of your hot leg.

MEMBER CORRADINI: Right. Okay. Thank you.

CHAIR REMPE: It's the geometry, and that's the hot tube fraction. But then you applied the hottest tube based on expert opinion. And I guess I kind of --

DR. SALAY: How many tubes failed, based on expert opinion? The hottest tube is based on the

CFD result analyses.

CHAIR REMPE: Right. And so I'm just kind of -- There's two parts to it kind of though. The size also. Again, you said earlier, well you got enough that you depressurized. So it's not that critical that you used expert opinion to apply that hottest temperature to just the 20 tubes.

DR. SALAY: Yes. There's something -- Could you repeat the question?

CHAIR REMPE: Okay. So the hot tube fraction is based on the CFD analysis.

DR. SALAY: Yes.

CHAIR REMPE: Whether you do .41 or .20, to 2.5 for the CE plant.

DR. SALAY: And the temperature of the hottest tube --

CHAIR REMPE: And that helps you get the temperature of the hottest tube.

DR. SALAY: Yes.

CHAIR REMPE: And that is all geometry based. But then, how big the hole is from the tubes failing --

DR. SALAY: That's completely, yes --

CHAIR REMPE: Expert opinion. And whereas in the Westinghouse analyses there was --

DR. SALAY: Yes. That's -- Yes --

CHAIR REMPE: Was it based on something more than expert opinion back in that earlier NUREG?

DR. SALAY: Well, in the Westinghouse only the flawed tubes failed. So you only get one or two. So you don't --

CHAIR REMPE: So it didn't matter? It was the pristine, the expert opinion. Or the --

DR. SALAY: Yes. It's only if in the condition where unflawed tubes fail where you have this expert opinion going in --

CHAIR REMPE: Okay.

DR. SALAY: -- multiples. And so it's -- Otherwise it's just a flaw. And I don't know.

CHAIR REMPE: I think I've got it.

DR. COYNE: Mike, to clarify, for the thermal-hydraulic ones that you were using to support the calculator calculation, you're suppressing the tube failures. You have just done other sensitivity studies where MELCOR can do this damage fraction, and you get a sense of when tubes fail.

But we actually calculate that fraction, for the purposes of the conditional probability using the calculator. With the thermal-hydraulic parameters as the input to the calculator. He can just do some side calculations in MELCOR that kind of give you a screen level impression of how the system's going to behave.

DR. SALAY: They're screening calcs, yes.

CHAIR REMPE: Did you ever do like comparisons with what the calculator predicted versus what the screening calcs predicted? And did they come out fairly easy, or consistent?

DR. AZARM: Ali Azarm, consultant to NRC. A couple of things I just want to clarify. Mike was focusing on the hottest tube. But when we actually look at the average hot tube and assign flaw to that, they also fail in the CE plant.

So it's not really driven just by this assumption of 20 hottest tubes. So I want to distract away from total focus on the hottest tube. And you will hear that there is some perhaps divergence of opinion regarding the weak area of it. So that's one of the things that I want to clarify.

Regarding your second question, that have we really checked this magnification factor, or the index versus actual calc on those. No. Actually it's not done. I looked at it one time when I knew they used this factor of two, damage factor of two.

MEMBER BALLINGER: That's this MP.

DR. AZARM: MP factor, yes, that is correct, right. That's what he's talking when he talks about index. This is a qualitative way of saying, okay, if I am at MP of two and, you know, I can calculate one deterministic number.

Now, calculator is very much driven by what TH delta, as temperature and pressure. Because that's one of the very important input to the calculator. That is a specific scenario.

But also, calculator looks at many other uncertainty sources, fraction mechanic models, material properties as a function of temperature, et cetera. So yes, that's a big factor. But it's not 100 percent.

CHAIR REMPE: Thank you.

DR. COYNE: And to add to what Dr. Azarm had said, the stress multiplier factor has been used in the past for these kind of screening studies. But it is very difficult to link the specific stress multiplier to any given plant.

You know the saying, you know, Springfield Unit 1 has the express multiplier of 1.75. It's very difficult to make that leap. But what we can do is we can have characteristic flaw distribution over operating experience for that actual steam generator to see what flaws exist.

So moving the question to what flaws exist in the tubes, as they're characterized by length and depth, and those kind of parameters that we'll talk about this afternoon, is an easier leap to get to a plant analysis, than using the stress multiplier, which was really was intended as sort of a screening level vehicle with the MELCOR analysis.

CHAIR REMPE: Okay. Thank you. Okay, so it's -- If you're done, Mike. And we were supposed to be done at 10:30 a.m. So we're really close to being on schedule. So let's take a 15 minute break and come back at ten of ten.

(Whereupon, the above-entitled matter went off the record at 10:35 a.m. and resumed at 10:51 a.m.)

DR. IYENGAR: Morning again. We had an excellent discussion. And Mike Salay had presetted everything so I don't need to talk much. The topic you're going to hear now is a side topic, a little bit more focused and not an integral report of calculator. But it provides credence to the simplified model used in a calculator.

Inadvertently we had left out a couple of slides. And Christopher Brown has distributed two slides which actually will be the first two in my presentation. And there are some extra copies as well in the back of the slides. Just wanted to make sure that all of you have that.

The primary purpose of the analysis that we did was to verify whether the simplified model used in the calculator for RCS hot leg piping, whether that's adequate for the Westinghouse plant to arrive at the conclusion that there are hot leg fails before the steam generator tube.

It's very important because the calculator, as you will see later, uses a very simple access electric model of hot leg pipes and uses uniform temperature, there's no gradient across the space. And it's a very simple model and uses the model to determine the time to rupture.

So the natural question would be is is this good enough because you're arriving at a conclusion which is very important saying that the RCS pipe will fail before the steam generator tube, hence there will be an abortment of bypass.

So this is very important. So in order to validate that, there were a number of studies done prior to when I started working on this in 2010. AML had done some initial modeling studies of all the components in the RCS piping system and used the input from RELAP SCDAP for the worst case scenario for the thermohydraulic and pressure distributions, both spatially as well as temporally.

And they determine as you will see in the next slide, they determined a number of things they investigated, we have detail everything in Chapter 4 of the draft NUREG. They looked at hot leg surge line and the primary manway to see, especially particularly near the welds in the manway to see if something, either that would melt or creep or default and show, cause some depressurization that way.

And looked at best estimates to determine, to detect, temperature detectors near the elbow region of the hot leg piping and see if that would fail and looked at multiple scenarios.

And it was determined through the study that hot leg in fact reaches, experiences the highest temperature and in fact is more potential for failure there compared to the other components. So as a result of that, we focused primarily on the RCS, I mean hot leg piping in the studies that I will be talking about in a few minutes.

Before that, the other purpose was to look at if hot leg piping fails, what would happen if you had a weld overlay near the nozzle region because as a mitigating effect against PWACCL primary water stress corrosion cracking, weld overlays are used.

And if you did use a weld overlay, that's going to make the region thicker, and would that influence relocation as was time for failure. That was also a consideration we had in our studies.

So mainly these are the two things, really simple, looked at detail finite element calculations of hot leg piping. Initially we started with what I call a system model, the entire hot leg and surge line region.

And we conducted a finite element analysis using three dimensional shell elements. And we assumed a material behavior to be elastic and also having two types of permanent deformation which is the plastic which would be a time independent permanent deformation of the material as well as creep would be a time dependent permanent deformation of the material because it's fairly important at high temperatures, these materials exhibit creep phenomena.

And I gave the creep law, we used a very simple creep law which had time and rate dependent, and the plasticity was rated. It's not needed, you know, this is just an assumption and it's a very reasonable assumption.

You could have more sophisticated constituted models for the high temperature. But sophistication will be of no use if you don't have a lot of experimental data to support the sophisticated constants of a model.

So in order to, see that's one problem. We don't have a lot of three dimensional experiments that could be used as data points to help the constant of the model. So with what we had, we had used this and I think this turned out to be pretty good.

And during the course of the study prior to 2009 that ANL conducted, they had used some data which was available from literature, and that was particularly not suitable for these temperatures because the data available in the literature didn't go up to even 1,000 degrees Centigrade.

So we subsequently did some additional testing which documented in the chapter, in the Appendix A that we have more expanded data. I think that's the first time we have data available in open literature internationally. So we had the benefit of that. We used that in our studies to inform the failure times.

MEMBER SKILLMAN: Raj, before you change, how did you account for the differences in the supports and restraints from the different designs?

DR. IYENGAR: We focused mainly at Westinghouse plant. We had in our model we had used, it's detailed in Chapter 4, we had used, in the modeling they had used the gravitation effect, the weight of the whole system.

And we used the support system for the surge line, the failings and everything, and for the steam generator, mainly that region we use all the support.

They did support, you could do that, you know, you could take a point and a finite element known here and then connect it to a different point and have it as a rigid support or elastic support. We allowed for those things to allow the flexibility as well.

MEMBER SKILLMAN: Thank you.

DR. IYENGAR: So those are all detailed here, all the number of supports. And we also did, we did couple of calculations of this surface area throughout the complex models.

And it turned out that this model is a very difficult one. It takes a lot of time to, and we had convergences issue especially when the temperatures go up really rapidly.

And so in order to do the weld overlay studies, we thought because of the hot leg being the worst point for failure, we used a submodel as I'll be describing later, the hot leg alone.

And that actually provided us an opportunity to look at several scenarios to get some kind of sensitivity. Not a thorough sensitivity studies but to, you will see those results in a moment.

So as far as our analysis goes, this is a coupled thermal-mechanical path for analysis. What makes this more realistic than the simplified model used in the calculator is that we used the actual temperature and pressure distribution from RELAP SCDAP both spatially as well as time.

It's very important that spatially you see as you had seen that Mike Salay had presented that there's a variation in temperature in the upper portion of the hot leg versus the lower portion. We accounted for that.

And that turned out to be quite significant I think in these new calculations. We also looked at the heat transfer coefficient. We adjusted that spatially in the hot leg region. And based on the developing curve in 1922, and Mike Salay talked a little bit about that too.

So we used all those input we got from that RELAP SCDAP for the Westinghouse model and fed it into the finite element model to get the temperature distribution for this and used the temperature distribution to run a mechanical simulation to determine the, ultimately the failure time.

So in the mechanic distribution, as I mentioned earlier, we took the gravity load where we assumed the body forces for the entire hot leg and surge line region and then used a point of mass forces for the steam generator itself and the manway.

And I did mention to you earlier that we used high temperature properties, especially for 316 stainless steel. This is the first time that we used this, and we got that from some of the tests done at ANL.

And then we ran a thermal-mechanic simulation, and the thermal-mechanic simulation gives you the amount of creep, how it was through time, and it also tells you the plasticity which is instantaneous as well as we used those to determine the damage. And I will give you briefly how the damage is determined.

All these are documented. What you see are the stress strain curve for the plasticity assumption which is time independent. And then for the creep we used a simple amount in creep law as you see here. And Q with epsilon dot, a creep is a creep rate and A is a constant depending on the material. And Q is the so called effective stress.

And T is the time. In this case, we didn't used a time hardening. You certainly could use a time hardening as well which will, you know, slightly change the material behavior, creep behavior. But for these materials, the time hardening is very small. So we have not used that, and the end was zero.

MEMBER BANERJEE: And so the time dependence was taken out of the creep?

DR. IYENGAR: No, no, no. There are two methods of time dependent. The time dependence is inherent here, inherent here, epsilon dot creep.

MEMBER BANERJEE: No, I mean the time dependence --

DR. IYENGAR: All this does is --

MEMBER BANERJEE: Hardens the material.

DR. IYENGAR: -- hardens the material. That's different, right? That's --

MEMBER BANERJEE: They've taken that out.

DR. IYENGAR: Yes. Normally for these materials, you know, the end was very close to zero for many materials. There are some materials which, you know, you don't use in reactor pressure vessels. But you could have just significant time hardening.

MEMBER BANERJEE: What's the time scales involved here? How long are the processes going on for the -- I haven't got a clear picture of is it a 1,000 seconds?

DR. IYENGAR: Well, the failure time as you will see later on when I press in is 12,000 seconds or so. But really, that's not the time scale of real interest to us.

MEMBER BANERJEE: It's only when it's hot, right?

DR. IYENGAR: In terms of creep, it's less than an hour because until 9,000 seconds, it's fairly stable state and then shoots up.

MALE PARTICIPANT: And then it shoots up, yes.

DR. IYENGAR: And in fact, where it shoots up, the last 200, 300 seconds for the scenario reviews, that's where things are exciting. That's when all the action happens. And --

(Simultaneous speaking)

MEMBER BANERJEE: What is the temperature range there? Is it going up to, it goes very quickly up, right, once the core starts to --

DR. IYENGAR: Right.

MEMBER BANERJEE: -- oxidize rapidly.

DR. IYENGAR: So after it goes to 9,000 it's not that high. And then it starts going up. But then it ramps up further after about 10,000, 11,000 seconds it ramps up quite fast. It goes from, I don't remember the exact numbers, I don't want to, it's probably from about 950 C to about 1,200 C. That's a very rapid rise.

And I don't want to spoil the final conclusion, but since you asked me this question I'll tell you anyway. It's because of that fact that Westinghouse, the RCS likely fails before the steam generator tube in our calculations because the last 300 or 400 seconds, the temperature rises so high, so sharp in the hot leg region that it really doesn't matter.

You'll see, you can make 150 different assumptions, whether if you can say the heat transfer is either insulated completely, the hot leg piping, or we get it not insulated. It really doesn't matter because your failure times --

(Simultaneous speaking)

MEMBER BANERJEE: So I have to look at this quite physically. The way I would explain it to myself would be that the hot gas and steam coming out of the core doesn't have a chance to mix very much, you know, as it goes down the hot leg it's going to mix with the cool stream that's coming back. Right?

Now what happens then is that the hot leg then gets exposed to this really hot stuff before it mixes with the cold stuff that's coming back, if I think of it that way. And then if you impinge this hot material on the steam generator tubes, probably the steam generator tube will fail first, right?

But because it's mixed in, it doesn't have this really hot material hitting the steam generator tubes, at least implicitly that's how I would explain it to myself.

DR. IYENGAR: I think Mike probably is the best person to answer that question. It seems plausible.

MEMBER BANERJEE: All this other stuff is correct, but I mean physically, that's what's happening, isn't it? I mean, really. And as you get your, I think Chris Boyd was telling me as the hot leg becomes bigger, you know, the core stuff is not mixing that much. So if that impinges on the tube, then you're exposing it to a hotter temperature.

MEMBER POWERS: I think you get a combination of things. It's hot stuff and it's not wandering around among the tubes.

MEMBER BANERJEE: Also, yes.

MEMBER POWERS: So that you're putting a lot of hot stuff on, but it's always on the same tubes.

MEMBER BANERJEE: Yes, because if you have a shallower plenum, that would happen, right?

MEMBER POWERS: Yes.

MEMBER BANERJEE: You would get this, like, a jet heating this thing.

MEMBER POWERS: So it's the combination of the two that gets you in trouble really quickly.

MEMBER BANERJEE: Yes. So on the Westinghouse things you have, is this, are we in closed session, open session, I don't know.

CHAIR REMPE: Open.

MEMBER BANERJEE: Okay. Anyway, the plenum is much deeper. So you've got this --

MEMBER POWERS: Larger opportunity for the return flow to mix and --

MEMBER BANERJEE: Yes, mixing and all that sort of stuff.

MEMBER POWERS: And it tends to wander around among the tubes so that you get, you heat a tube up and then it'll move to a cooler tube and that tube has a chance to cool --

MEMBER BANERJEE: Because it behaves like a plume?

MEMBER POWERS: That's right.

MEMBER BANERJEE: And you've got cross flow. So I think physically, you can see what's happening.

MEMBER POWERS: Yes.

MEMBER BANERJEE: I mean, all this CFD and all is great and --

(Simultaneous speaking)

MEMBER BANERJEE: -- we have to understand what's going on.

MEMBER POWERS: Yes, the physical understanding. But I mean, the two are tightly coupled with each other. I mean, you understand them when you calculate and then you understand more.

MEMBER BANERJEE: The shallower plenum will give you a problem --

(Simultaneous speaking)

MEMBER POWERS: And we seem to be, like, the EPR seems to have a pretty shallow plenum. We're seeing shallower plenums that maybe weren't plausible to what we want.

MEMBER BANERJEE: Right. Okay.

DR. IYENGAR: I did mention the heat flux and heat transfer coefficient adjustment, and this is the developing curve, it's like that. This is closer to the hot leg here. And as it goes away, it changes according to this curve, and we got this curve in the late 1922.

So all of these were used, and we applied it to the various elements across the model. So that's why this was a little bit, a very difficult exercise. It's just very strenuous to model those things.

What you see here is at around 12,000 seconds, like I said, you know, things are fairly stable and nice for 9,000 seconds after that, the action starts. So when I say 12,000 seconds, it's the accumulation of more like 2,500 seconds or so.

You do see two things here. What I have, what I show you are the accumulated creep strain, the effective creep strain on the left hand side, top. What you immediately see that the top surface has more creep train than the bottom surface.

There's no surprise because we used the countercurrent circulation flow there. The top side is hotter and the bottom side is a little bit colder. Not that cold, but relatively speaking, colder.

And you also have the accumulated plastic strain. And I want to bring your attention to the location where these strains are higher. It's in the top portion of the hot leg but away from the nozzle, the RPD nozzle.

And that's fairly important. I think in terms of dimension, I'm not sure this is probably what --

MEMBER STETKAR: You have to stay by the microphone. You can use the mouse.

DR. IYENGAR: This one?

MEMBER BALLINGER: What's the difference between the accumulated creep strain and accumulated plastic strain?

DR. IYENGAR: The plastic strain is the instantaneous response of accumulation. That's the model we assume that the strain is a combination of the elastic strain and instantaneous time independent plastic strain.

MEMBER BALLINGER: Okay, so that's what that is?

DR. IYENGAR: That's what that is. And then you have the time dependent creep strain, that's what that is. So the total plastic strain, if you will, will be combination based.

MEMBER BALLINGER: Okay, okay.

DR. IYENGAR: It's just a simple modeling thing they use for high temperature materials, again mentioned earlier, is a way to combine these methods strain to strain, right? And it's probably what you're thinking.

MEMBER BALLINGER: Well, I mean, this is probably primary creep versus secondary creep.

DR. IYENGAR: No, no.

MEMBER BALLINGER: No?

DR. IYENGAR: No, I won't make that distinction here. Primary creep is also --

MEMBER BALLINGER: Instantaneous, though. Relatively.

DR. IYENGAR: Relatively instantaneous but still time dependent. It's just slow, right? If you bear with me, let me go back to my --

MEMBER BALLINGER: It's just a fine point, it's not --

DR. IYENGAR: Well, it's an important point. So here you have, I had prepared a couple of slides for the last briefing at the request of Dr. Powers on a quick overview of high temperature feed, so I might as well use this, take some mileage out of it.

So when you expose the temperature of these materials to high temperature and some stress, I have to learn how to use this now.

MEMBER BALLINGER: I understand what it is.

DR. IYENGAR: So as soon as you apply the stress, there's an instantaneous, that's your plant time independent plastic strain and elastic strain combination. And then you have primary creep which is time dependent, and then you have a secondary creep which is what we using is fairly steady state.

And the tertiary creep is significant. We haven't modeled that. That's one of the things that I wanted to mention. The temperatures that they're gaining when the simulated accident, really speaking the material is in the tertiary creep region.

But we really don't have a model, we don't have data for that. If anything, that will push the failure times that we show to a lower number.

MEMBER BALLINGER: You can use stress rupture data in there, right?

DR. IYENGAR: Yes. So up there we're using stress rupture data for --

MEMBER BALLINGER: Yes.

MEMBER POWERS: Yes, when we get into tertiary creep, you're in a wild region where materials behave in a non-linear fashion.

DR. IYENGAR: Yes.

MEMBER POWERS: And so --

MEMBER BALLINGER: You're also in a lot of trouble.

MEMBER POWERS: You're in a world of hurt. I mean, but it's not only a function of the material, it's also a function of the geometry. And so it gets very complicated, I mean, you get very complicated models because you shouldn't be using the material there.

DR. IYENGAR: Yes. So in one of the backup slides, we show this using Ashby's deformation map plot where you have, for the stainless steel you have the temperature and the shear stress.

And this shows the region where the steel would exhibit different kinds of behavior, whether it's plastic, whether it's creep. We actually had this remarkable vision of try and put --

(Simultaneous speaking)

DR. IYENGAR: -- one graph. Save a lot of money on it by the way.

MEMBER POWERS: Yes, but it confuses the hell out of everybody that looks at it.

(Simultaneous speaking)

DR. IYENGAR: But the point made here is

these steels, 316 steel as in this blue oval indicate, those are the normal operating conditions. What we're dealing with is on the red side for this. Because these materials are not designed to operate at these raises, you don't have data. You have data, all your data isn't all there.

So that's one of the complexities we have. And today I think there was another complexity that was brought in by Dr. Joy Rempe which we of course did not consider is the oxidation effect on these at these high temperatures.

And I think that's also pretty significant, we don't have a clear understanding on how that will affect the creep failures because that's fairly important as well.

Okay, I have to go back. Sorry, I thought I would just take your time. We show the creep. And then the way we calculate damage is we use the Larsen-Miller Parameter which is a parameter we determined from the initial creep experiments, fresh rupture experiments.

And for the materials that we considered in the study, we have used the most recent data that was done, you know, from ANL. And it's documented in Appendix A. So we used that and get the Larson-Miller

Parameter and determine a time to rupture given stress state and the temperature.

So you could see that at the failure time for this system is the hot leg region set on 12,302 seconds.

MEMBER BALLINGER: See, I see a failure time of 12,302 seconds. Is it 11,000, 13,000, 5,000, 15,000? Or is it 12,302?

DR. IYENGAR: Excellent question. The failure time here is what we did is we did an average to the thickness. So we use time stamps as we go along. We use time stamps of even less than, you know, one second time stamps in this region.

And what I found was when I say 12,302, you know, I like to say it could be 12,250.

MEMBER BALLINGER: You think it's that good?

DR. IYENGAR: These are all approximate. But you know, it's better approximations of the time stamps.

MEMBER BALLINGER: I mean, it's easily a factor of two or three in creep rate on these things. The Larsen-Miller Parameter can vary by 50 percent. So I'm just curious as to really what kind of voracity do you have on these?

DR. IYENGAR: I mean, we have used a lot of experimental data, get the Larsen-Miller Parameter and there could be a variation on that as well. The time stamps that we used in the final element model was very refined and small.

So what I found was these average failure time, average to the thickness was the failure starts at the inner surface of the hot leg and it propagates. It propagates rather quickly within two seconds in the model.

So your question is well taken. But I mean, given the input that I have, given the Larsen-Miller Parameter that I assume, then the failures are very rapid. You could see that with different assumptions I made that the failure times are bounded within, you know, less than 100 seconds if I turn off some things.

So, I mean, it doesn't quite answer your question because you asked the question is more based on the uncertainty of the material data that I used. But if you give me the data, I can tell you comfortably that with various assumptions I made, I'm not getting a whole lot of difference in the failure time.

MEMBER BALLINGER: The bad news is the temperature is high, the good news is the temperature's high --

DR. IYENGAR: Yes.

MEMBER BALLINGER: -- because if the temperature's really high, it doesn't really matter.

DR. IYENGAR: Yes. That's what I'm coming to. So this is --

MEMBER POWERS: I think that's, I mean, I think that's the $64 question is we know what happens when the temperature gets very high. The question is does it get very high.

MEMBER BALLINGER: Yes, that was another question.

MEMBER POWERS: You know, and the TMI accident, we didn't seem to get very hot. Now we have a speculation that in the Fukushima reactor accidents, we did get hot. And so it will be very interesting when we have a chance to look to see if in fact we got very hot.

MEMBER BALLINGER: That boundary between the fusional creep and that's where you get a lot of uncertainty, a lot of scatter in the data. Very high temperature, you fail --

MEMBER POWERS: Very high temperatures, everything goes to hell on you. Very low temperatures, everything's fine. This question of what happens with this hot plume of gas coming up from the core and whether it, how much of the heat loses in transit is one of the features of uncertainty in accidents that we're kind of stuck with right now.

And we do see different modeling, for instance, in MAP where we don't get such high temperatures up in the primary piping system. But it's not one that you can resolve by better analyses or filter analyses.

You really need to melt down more reactors. That's just all the answer is, we just got to melt down more reactors.

(Simultaneous speaking)

CHAIR REMPE: -- if I take the quote out of context.

MEMBER BALLINGER: Do it in Iran?

(Laughter)

DR. IYENGAR: So for further studies, we used a smaller hot leg region to model. This is the model region. And we use the same temperature and pressure distribution both time wise and spatially.

And using the one without weld overlay, you see on the left hand side, the time to failure calculate set a little bit higher than the system model, presumably because a lot of the gravity loads were not used here.

The failure time increases like I say. Then we used a weld overlay on the nodal region. The weld overlay was about nine inches wide. And that's based on some prescription of 1.5 square of R over T on minority.

And not surprisingly, it didn't change the main point when a failure occurs because I was a little bit far away from the nozzle line D in valve region. And it did increase the time slightly and the location doesn't change.

CHAIR REMPE: I mean, we've been a little bit tongue in cheek here, but the difference between 12,428 and 12,500, we really need to get a handle on what the uncertainty is on these numbers, some kind of estimate because I don't look at that as any different at all. In fact I'm not even sure the two is significant.

And so it's just we just got to get a handle on an estimate of what the upper and lower bounds are on these numbers or something.

DR. IYENGAR: Point well taken.

MEMBER POWERS: I will suspect that your systems level modeling can adjust that by 1,000 seconds.

MEMBER BALLINGER: 1,000 seconds sounds okay. But I don't know --

MEMBER POWERS: If you set up the systems level modeling and I set up the systems level modeling, and Joy set up the systems level modeling, we would probably have greater than 1,000 seconds difference from that calculated number.

MEMBER BALLINGER: In a calculated number.

MEMBER POWERS: Well, they're all calculated.

MEMBER BALLINGER: Yes, yes.

MEMBER POWERS: I mean, just how you model the system, how you nodalize it, how many nodes you take and things like that can have that kind of change, this far away from a main actual in the core.

DR. IYENGAR: Very true. But I mean, I wanted to bring your attention back to the purpose of this exercise was to see how the simplified model using the calculator performs, that if there were more sophisticated model.

Yes, I do agree with uncertainties that you have pointed out. But that was not the major focus of this exercise. Was a little bit less, I mean, more focused on comparing how good or how confident are we of using the simplified model.

And with this hot leg model, you know, I did many different things. I turned off creep completely, I turned off plasticity completely. Yes?

MEMBER STETKAR: Let me interject here. I wanted to wait until you got to this slide. You said the purpose of this is to provide some guidance for this simplified calculator. Okay, we have a calculator.

You're proposing that that calculator be used in probabilistic risk assessments that evaluate the likelihood that we have detrimental effects on public from large releases. I see your sub number one here. It says well, we have an uncertainty distribution in this little calculator that we use.

There's no uncertainty there. That is not an uncertainty distribution. If you're proposing to use that uncertainty distribution for this little calculator, I will submit do not use the calculator in probabilistic risk assessment. There's no uncertainty in those parameters.

MEMBER BALLINGER: That's where I was going, actually. That's where I was going.

MEMBER STETKAR: So therefore, your simple calculator is fundamentally flawed because it does not correctly account for the uncertainties.

Uncertainties are everything in risk assessment.

If we don't look at uncertainties, the convolution of uncertainties and timing and understand that effect correctly, we're going to have the wrong answer. Period. It's the wrong answer.

So understanding the uncertainties and characterizing them appropriately is everything in risk assessment.

MEMBER STETKAR: The difference between the CE and the Westinghouse plant is a factor of ten, right? I'm at that order, a factor of ten and I just asked the question is the factor of ten actually --

MEMBER STETKAR: But my point, a lot of, where they're eventually going to get to is the conditional probability of a large early release depends on this relative timing. The relative, the uncertainty and the relative timing is everything.

If you are absolutely certain that one thing occurs 5.77362 minutes before the second one, you're wrong.

MEMBER BALLINGER: If you were to take a pipe, scalable stainless steel pipe and pressurize it and run it at that temperature and do basically a stress rupture test on the thing, there would be a factor of two because when you get into tertiary creep and those kinds of things, a little flaw here, a little flaw there starts a propagating defect. And so you really don't know. So I think there's a factor of two there to start with.

MEMBER STETKAR: I don't know. But my point is that this is being characterized as the uncertainty distribution in the little calculator that's supposed to be used to support probabilistic risk assessment, it is wrong.

MEMBER BALLINGER: The only way we survive in the metallurgy bit is to plot everything on long log paper.

MEMBER STETKAR: Kevin's desperate. He wants to say something.

DR. COYNE: Kevin Coyne, Office of Research. I am desperate to say something. So the Staff is certainly not willing to concede the point that the calculator is fundamentally flawed until at least the point that we get a chance to brief the Committee on what's in the calculator.

So I think you'll see there's more treatment of uncertainties than, you know, is indicated on Raj's graph here and Ali Azarm and I we will all go through some of that. And the comments may very well be valid, but we'll --

(Simultaneous speaking)

DR. COYNE: -- prefer to wait until we've had a chance to brief you on it before we come to that conclusion.

MEMBER STETKAR: I'm just concerned when I hear people say, well the whole purpose of this was to support the calculator and then you see uncertainty distributions that have no uncertainty.

DR. COYNE: So to back up from the purpose, and I'll turn it over to Ali Azarm in a second, but the purpose of Raj's analysis was to provide with some confidence that the more simplified correlations we're using in the calculator to estimate the time of the hot leg failure and pressurized resurge line failure are not, either not conservative or wildly conservative.

So there was an early recognition when this work started that we're using sort of a one dimensional treatment of the temperature and the hot leg failure.

Raj used the abacus code to try to bring some three dimensionality into it, some more sophisticated modeling to give us some point estimates of when he is predicting hot leg failure given the same thermohydraulic conditions using more sophisticated tools for us to compare against what the calculator would give.

So, you know, all these points about the uncertainty of the numbers or lack thereof in Raj's analysis is true. I think Raj's main point on the weld overlay versus non weld overlay is that yes, those are indeed almost the same number.

It wasn't trying to make a distinction in 72 seconds. It was trying to show that there wasn't an appreciable effect with welder relays which was an initial concern of ours that this massive hunk of metal put over the hot leg nozzle would somehow delay the time to hot leg failure such that we had a more significant concern with the steam generator tube failures.

Turns out, the more detailed analysis is indicating that that isn't as an extreme effect as we thought. But again, this was a touchdown, a benchmark point to look at the results of the calculator to give us some confidence that the correlations we're using aren't non-conservative or wildly conservative. And so that's what this figure is trying to show.

MEMBER STETKAR: Your point is certainly well taken there, Kevin. However, all that does is delay the question that I have about how does the calculator treat the uncertainty because I didn't dig into the calculator.

How does it treat the uncertainties

because this uncertainty, this 90 percent confidence interval that it's within 300 seconds, that we understand it that well strikes me as being wolfedly narrow.

So there must be something in the calculator then that does something to expand that uncertainty. So you know, and if we're going to talk about the calculator later, we can do that.

DR. AZARM: Ali Azarm again. Dr. Stetkar, I just want to little bit clarify. The calculator uses the EPRI model for Larson-Miller. And the EPRI model has coefficients, and in front of each coefficient you have the tolerance or the variance for example.

It says that the factor in front of this is 1.1 plus/minus 0.25 which is treated as the normal distribution. Now why the calculator uncertainty in this case was so small? Because when the calculator fail the hot leg, you were at the very high temperature.

And as Dr. Dana Powers was saying, when you are in that region, you are going to fail and your variance is very small. If you would have looked at lower probability stuff at the lower temperature, it's much larger spread.

Anyway, we will talk about it enough. And we hear your comments at that.

MEMBER BALLINGER: But as a practical matter, publishing something that goes out with an M number with a number 12,302 in it just doesn't make any sense. I mean, that's --

(Simultaneous speaking)

MEMBER POWERS: You could right the number down.

MEMBER BALLINGER: I know, but -- okay.

MEMBER POWERS: You know what the uncertainty is, I know what the uncertainty is. I mean, that's the number they get. What's wrong with that? You can see what he does in his percentiles. He's courting them out to numbers.

DR. IYENGAR: We did enough, I come from the combination mechanics background. And when you put these time increments, the code and gets a number. We just repeat what the code tells us.

We don't round off. So if it gives you 12,302, I probably could make it 12,300 seconds. But that's not what the code had given me. So that's all. This is maybe being more honest to what the code predicts rather than trying to, you know, take --

(Simultaneous speaking)

MEMBER BALLINGER: We have code fixation.

(Simultaneous speaking)

DR. IYENGAR: Well, I also want to tell you one point is that I've said many things that we do not understand and don't have data, and the regimes for this temperature.

And assumptions we made here are actually, as I told you, will provide failure times little bit higher than what it would be if you knew everything about how 316 fails at that temperature because you would have a tertiary creep, you would have some oxidation effects.

These are all the push the failure time to a lower time. So I think that also has to be put in perspective when you talk about these things. Any other question? Yes? Thank you.

DR. COYNE: You had a last slide.

DR. IYENGAR: Oh, I have a last slide?

CHAIR REMPE: You do have another slide, yes.

DR. IYENGAR: I thought it was, I think like I said, I think we have gone through all these conclusions. Weld overlay had a small, very small insurance on the location of failure time. And I think what I thought was the more complex analysis predicted times which are little bit lower than what the calculator had predicted. And we had quite a bit of discussion on that point. So if there are any other questions?

CHAIR REMPE: Okay. Because we're going to switch topics next, and it's getting close to the lunch hour, I would suggest we go ahead and take a break for lunch. And we're supposed to come back here at 12:45. Okay? We're off the record then.

(Whereupon, the above-entitled matter went off the record at 11:38 a.m. and resumed at 12:45 p.m.)

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

(12:45 p.m.)

CHAIR REMPE: Okay, so we're ready to resume this meeting and we're going to hear about the steam generator tube flaw distribution characterization effort.

DR. AZARM: That is correct.

CHAIR REMPE: For the record, please introduce yourself too.

DR. AZARM: This is Ali Azarm. I'm a consultant to research. I supported them early on when I was an employee of Information System Laboratory, ISL, and now currently I'm an independent consultant.

I am going to present three sets of slides today or three presentation. The first one is going to be on some work we did four or five years ago in updating flaw distribution.

Then that follows the overview of calculator, and I heard this morning there might be some interest and question regarding that software.

And finally I'm going to present a brief summary of probabilistic risk assessment simplified models that we develop for NRC.

In the first presentation, Mr. Mica Baquera --

MR. BAQUERA: Mica Baquera.

DR. AZARM: -- yes, is the co-author from research so any hard questions I'm going to transfer to him.

As you know by now, there is a software called calculator or steam generator calculator and this software gets bunch of input and one of the input it needs is a sample of flaw, 1,500 flaws with a certain depth, certain length.

So we needed to develop statistics to generate the samples needed for the calculator. At first, initially, we looked at to see what is available rather than trying to undertake this task.

The previous work we found and, you know, was the work by Gorman, NUREG/CR-6521. This basically has the statistics provided, summary statistic provided for characterization of cracks for Inconel 600, which was susceptible to stress corrosion cracking.

Now, given that we know almost every plant in U.S. changes steam generator and now they are using the thermally treated Inconel 600 and 690, which are not really as susceptible to crack but are more susceptible to wear and volumetric type of flaws, the use of Gorman report and the statistics cannot be justified.

So as result, NRC initiated the small task to update the flaw distributions. This work was contracted to ISL so that's why you have a report from ISL on that.

So the first thing we have to talk about, where the data is coming from. Licensee generally reports in the PDF format, hard copy. Information of the detected steam generator flaws at each in-service inspection are usually done during refueling outage.

The information includes the location, depth and length and the voltage associated with the detected flaws. These reports also include number of flaws that are plugged in each cycle.

But given availability of these reports, NRC reviewed a large set of these reports, NRC research, and with the understanding that they wanted to have a manageable set of them for the consultant to go through and extract the data and put in the database and try to evaluate it. And they wanted to select plants that has enough cycles and of different types such that we get a good scope of type of flaws that exist out there.

This data -- okay. So what was reported was given to ISL. It was Surry Unit 1 and 2, which is a 600 thermally treated tube materials.

It shows number of cycles to be 26 cycles. That needs a little bit explanation. It was actually 12 ISI cycle in each unit and then each unit had one mid-cycle inspection, so each unit had 13 cycles so together was 26 cycles. So we have 26 PDF reports. That's what it means.

It doesn't mean it means 26 refueling outage. It doesn't mean it's 26 equivalent full power years of operation. For example, in case of Surry, the total full power operation or EFP month was 268.5 or about 22 EFPY.

But this slide basically show you that we had Surry as a representative of Inconel 600 for Westinghouse and then we have, which had lots of cycle in it so allows us to look at the change of number of flaw generated as a function of the age of the steam generator and the cycle flaws.

And for the C-E plant, or Inconel 690 for both C-E and Westinghouse we had bunch of different plants but with less number of cycles.

CHAIR REMPE: So these plants were made by different vendors, right? Like wasn't St. Lucie made by B&W and I don't know who made the other ones but they're not, how different are the designs under the 690s that are C-E or are Westinghouse plants?

DR. AZARM: We used to have a slide that actually was showing the model number and design of each of the steam generator, but to your question, I want either Mica or Ken Karwoski, if he's still here. They are much more knowledgeable than I am in this regard.

MEMBER BALLINGER: It's not clear that the steam generator supplier for the replacement was actually the original --

CHAIR REMPE: Definitely not --

(Simultaneous speaking)

MR. KARWOSKI: This is Ken Karwoski from the staff. You're correct. A lot of these, the NSSS vendor is different than the steam generator replacement manufacturer.

In the case of St. Lucie 1 and 2, those are C-E designed plants. St. Lucie 1 has an AREVA or, I'm sorry, B&W Canada steam generator. St. Lucie 2 has an AREVA steam generator.

CHAIR REMPE: I remember that one.

MR. KARWOSKI: In terms of similarities of the steam generators, most of the steam generators in the combustion engineering designed PWRs, they tend to be larger. They usually have, typically have two steam generators rather than three or four on the Westinghouse as well as two steam generators, so much larger than the --

CHAIR REMPE: Within a category like the C-E ones, how different are the designs? I remember the issues with St. Lucie Unit 2 versus Unit 1 that occurred and so what I'm trying to get to is just because it's a C-E one and it has a replacement one, that the plant, that the steam generator might be quite different.

MR. KARWOSKI: Yes, the steam generator, yes, can be quite different. There's a lot of similarities but they can be quite different.

In the case of St. Lucie 1 versus 2, the issue at St. Lucie 2 they've attributed to fabrication or manufacturing as you may recall. But, you know, there are similarities. Like in terms of the original St. Lucie 1 and 2 steam generators, I believe they were identical.

And so the replacements, they'll have some different features. You know, one might have a lattice grid tube support structure and the other might have a tube support plate so there can be some minor differences but, for the most part, they're fairly similar.

CHAIR REMPE: But then the inspection results might be quite different just because it was put in a different plant or it might have a bit different in design, perhaps the water chemistry or something like that. I'm not sure as we go further how good it is to be combining results from different plants.

MALE PARTICIPANT: I was going to -- I'm sorry.

MR. KARWOSKI: There can be differences in the performance between one vendor and the next and the performance of one type of steam generator at one unit versus another for the various reasons that you cited.

CHAIR REMPE: Thank you.

MR. BAQUERA: Yes. Just to your point, there's multiple different parameters that can vary on how a steam generator is going to perform, you know, licensee performance, how well they're monitoring things, how often they, you know, go in and inspect, you know.

So there's going to be a variety of different functions here so that's why we wanted to take, you know, some plants that were performing very well, some plants that had steam generators that weren't performing quite as well and use that kind of as a bounding sample for the calculator model.

CHAIR REMPE: Thank you.

DR. AZARM: Okay, when you look at these

PDF files, depending on the design of the steam generator and the vendor, you know, the C-E plants versus Westinghouse, they use different naming for the types of flaws they find.

So and I give you an example of what we have seen and the question is that as you will see later in this calculator software we can only model two things, a volumetric or wear flaw or the crack.

So we have to group them, yes. Like if somebody wants to track the performance of each model of a steam generator separately, needs much more data and should not mix and match.

But for the purpose of the calculator software and consistent with the models and calculator software, we have to group them to two categories, cracked and wear, and this basically shows how that grouping was done.

What we tried to do rather than give you tables of data and, you know, confuse the issue, I tried to -- and, you know, some of the data is missing. You know, some reports may not have documentation of small, shallow type of flaws and some others do.

What I tried to do, I tried to graphically show you what is in this database. In this graph, I basically did empirical or cumulative distribution. I don't know if I have to explain it but basically what it is is that, you know, if I look at all the data I have, the recorded depth of the flaw, and look at the probability that I exceed that depth.

For example, the probability that I exceed a depth of 20 percent might be 0.1. So what I did, I tried to present what the data is in the database and aggregate over everything we see in the form of a graph.

And the point I wanted to make is couple of things. One is that usually they plug a tube that sees a flaw greater than 30 percent. I know the criteria is 40 but they are conservative. They sometimes plug at 35 percent. Sometimes might even plug area if they think the growth of the flaw is unstable.

So what you will see, somewhere around 30/40 percent, kind of the shape of the graph, keep changing because those tubes that are plugged and those plugs that have flaws two sizes are, in a sense are removed. By plugging them, those flaws are not going to grow bigger and show up in other ones.

The other thing I wanted to emphasize here is that, yes, we did see some flaws that they were big, 85 percent depth. So even though you have a plugging limit of 40 percent within the next cycle until you do your inspection, either the smaller flaw grow much bigger or a new flaw is going to be generated that is much deeper. So, yes, there are bigger flaws, deeper flaws, with much less probability.

CHAIR REMPE: So before you leave this plot -- and I'm going to have to acknowledge that some of my questions come from a consultant who's not able to join us today. Dr. Shack mentioned the fact that he really liked this plot, but it doesn't appear in your letter report or the draft NUREG.

And so I, but yet if we go to the letter report, and maybe this is a hard question to answer when it's not here, but there's a Figure 10 in the letter report. And could you, for the uneducated person, try and convey how you went from this information to Figure 10 in your letter report?

DR. AZARM: Can I see the Figure 10? What is Figure 10 in letter report? I'm sorry. These things --

DR. COYNE: Ali, on the laptop there's a reference. There's a folder with references and you hopefully should be able to find the letter report on that reference list.

CHAIR REMPE: It's a probability versus flaw depth distribution and it probably is good to have access to the letter report because that comment about this is a great figure but I didn't see it in the letter report or the main NUREG or the draft NUREG is going to come up a lot today.

And if you wanted me to shut up, just ask me a technical question about my question and I'll have to say I don't know. See if we can go through it because I think some of his points are worth mentioning here.

DR. COYNE: The ISL flaw report?

CHAIR REMPE: Yes. It's a letter report in the flaw database. I can give you the ML number if that helps you.

DR. AZARM: No, I think I find it. It's --

CHAIR REMPE: Yes, I think you're getting there.

DR. AZARM: I am getting there.

OPERATOR: Please pardon the interruption. Your conference contains less than three participants at this time. If you would like to continue, press Star 1 now or the conference will be terminated.

DR. AZARM: Okay. I think what this figure is --

CHAIR REMPE: Is this related to that

other figure you had up there?

DR. AZARM: I do believe so.

CHAIR REMPE: Yes.

DR. AZARM: Supposedly this is the --

MALE PARTICIPANT: It's one minus.

DR. AZARM: Yes, one minus.

CHAIR REMPE: Is it really one minus? Because I actually took --

DR. AZARM: Yes, yes.

CHAIR REMPE: -- a couple of data points and I didn't see it was a direct one and I discussed it with Shack and of course, again, you've got this empirical gamma distribution --

DR. AZARM: Yes.

CHAIR REMPE: -- rather than the direct data --

DR. AZARM: Yes.

CHAIR REMPE: -- which he kind of wished that if you, he would really have preferred to see some direct data showing --

DR. AZARM: Yes, he --

CHAIR REMPE: -- how it compared too because he's got a distribution here.

DR. AZARM: Yes. Correct, and you will see. We will come back and discuss this because we have some note of cautious when we use the fitted gamma distribution. And this shows the gamma distribution trying to fit the actual raw data. The figure I was showing in the slide --

CHAIR REMPE: It's the raw --

DR. AZARM: -- is actual raw data.

CHAIR REMPE: And so his point is, boy, it would be good to show the raw data on there.

DR. AZARM: Yes.

CHAIR REMPE: And he really thought it was a great figure and he was glad to see you put it in the presentation.

DR. AZARM: Yes. I appreciate that. Thank you. Now, I have to go back where I was.

CHAIR REMPE: Sorry.

DR. AZARM: That's okay. Yes, I am. I'm there. Okay. Yes, whenever we fit the distribution, we kind of smear some stuff and, yes, it is always good to look at raw data and I wish, you know, to --

MEMBER POWERS: Is there a basis --

DR. AZARM: -- repeat that report.

MEMBER POWERS: Is there a basis for assuming a gamma distribution?

DR. AZARM: Frankly, no. Good question. The only reason gamma was used, because that's what

Gorman used.

MEMBER POWERS: It kind of looks like a gamma.

DR. AZARM: Yes, yes. That's true.

MEMBER POWERS: I mean, that's the basis but, you know, you would think that if ever there was something that ought to have a heavy tail it would be flaws as a distribution.

Yes, I mean, the advantage of gamma is you got, it's the minimum number of parameters you can possibly take and it's a nice high-entropy distribution.

In fact, I think, isn't the gamma the one that maximizes entropy under constraint of the mean? I think it is.

DR. AZARM: I think it is if --

(Simultaneous speaking)

MEMBER POWERS: It's one of those that, I mean, it's a reasonably high distribution, entropy distribution.

MEMBER BALLINGER: But these are flaws of all types.

DR. AZARM: Correct.

MEMBER BALLINGER: So there's stuff built into here that wear is different than stress corrosion cracking, is different than loose parts and stuff like that. That's everything.

DR. AZARM: That's everything thrown together. It just --

MEMBER BALLINGER: So there's no distribution of loose parts flaws.

DR. AZARM: Yes, but loose part is also cause a flaw that is categorized now under wear. So I can go into the database, very simply do the same graph for cracks and for wear but I cannot go to better resolution than that on this.

Now, the next slide also shows you raw data. This is for the length. Again, we are looking at wear or crack in a very ideal manner that has a depth, a length and, as you know, that is idealized but this is what is reported for the length distribution.

CHAIR REMPE: Before you leave that one and, again, it didn't appear in the NUREG, draft NUREG or the ISL report, but there was a Table 4 in your ISL report.

And in that table there's only one flaw that's 4.4 centimeters and all the rest are less than 2.1 centimeters, and so what's the real data in this empirical distribution and why did it get to be so long?

DR. AZARM: Basically I think this is

coming directly -- Let me just little bit explain this.

CHAIR REMPE: Please do, yes.

DR. AZARM: This directly is coming from the database. There's no manipulation, nothing done. The problem that you see with this distribution right at the tail at 1.5 inches or 3.8 centimeter, I can=t draw it any more. Basically after 1.5 inch is actual data and, again, Ken, correct me if I am, you know.

MR. KARWOSKI: No. That's --

DR. AZARM: Then we have bunch of data. Thirty-nine percent of the data is exactly 1.5 inches, 3.81 centimeter. It's like a sharp drop. That's because of all those flaws was by tube support plate and they might have just put that thickness of tube support plate and they didn't measure it exactly or the flaw.

Then there was one or two flaw, I do not exactly remember, that was in the tube sheet and those were bigger. So I didn't draw that end of the distribution. I will only do a gamma fit, they have smoothed it out.

But the real story of the data is that we have a sharp drop at 1.5 inches, which is all the flaws and tube support. Now, we don't really have actual length for them. And then we have couple of larger flaws that was at tube sheet.

CHAIR REMPE: But, again, if this is the exceedance probability in this plot, it looks like that you have 40 percent or something greater than 3.75 centimeters.

MEMBER POWERS: Yes, well, that's what he's saying, is --

DR. AZARM: That's correct. That is correct.

MEMBER POWERS: When they find a flaw in the tube sheet, they just call it --

MALE PARTICIPANT: All messed up.

MEMBER POWERS: -- the width of the tube sheet because they don't know any better.

MALE PARTICIPANT: Yes.

CHAIR REMPE: But in the database, you only had one that was greater than 4.4 centimeters in that Table 2, or Table 4. I'm sorry.

MEMBER POWERS: No, here it says 39 percent --

(Simultaneous speaking)

CHAIR REMPE: If I go to Table 4 in your report, which is Page 19 out of 33 of your report --

MEMBER BALLINGER: 19 or 16?

CHAIR REMPE: Well, it depends if you look at the report number or the pages in the PDF.

DR. AZARM: It is Table 4, correct.

CHAIR REMPE: Yes. Okay, so I see one flaw that's bigger than 2.1 centimeters and that's that one 4.4 one, right?

DR. AZARM: Yes. That 4.4 one, I am pretty sure that is the one that was on the tube sheet.

CHAIR REMPE: So then there's one.

DR. AZARM: There is one very big one on tube sheet.

CHAIR REMPE: Right.

DR. AZARM: And there could have been lots of 3.81 after 2.1 centimeter that they didn't capture.

CHAIR REMPE: That they didn't capture and so that's why you've made it so many.

DR. AZARM: That is.

CHAIR REMPE: Okay.

DR. AZARM: But, you know, what I did there, I didn't use this table. I just inquire database to dump thing and draw it for me. So what you see in that graph is actually what is in database. Now, they may not have used the whole thing for the statistical analysis because they have bunch of false data that they might have truncated out.

CHAIR REMPE: Okay.

DR. AZARM: Okay. Now, let's see.

(Off microphone discussion)

DR. AZARM: Okay, so if this is the data, what do you want to do with this data? What is this model that we want to create? First of all, we want to be able that if you are at a cycle which is EFPY 15 --

MALE PARTICIPANT: Excuse me.

DR. AZARM: -- and you want to predict how many flaws am I going to generate or am I going to see in my next inspection? Well, that's one statistic we want to create, the flow generation rate.

And we try to create that statistics for 600 thermally treated, 690 thermally treated and we had for 600 -- For 690 we didn't see any crack but for 600 we did see some cracks so we tried to generate the flaw generation rate and I --

MEMBER BALLINGER: When you say 600 thermally treated you saw, I know of only one case where there were cracks at 600 thermally treated and that was an error due to an error in processing the tubes.

That has nothing to do with stress corrosion cracking. They were stress corrosion cracks but they were not generated during operation. They were generated because the tubing itself was not processed properly.

DR. AZARM: Well, that's a more detailed question. I do believe the database has circumferential and axial is called single SCI, I think single crack indication. Now, was it a true crack, but that's --

MEMBER BALLINGER: Was it all Seabrook?

CHAIR REMPE: Seabrook wasn't in our database then.

DR. AZARM: Seabrook is not --

MEMBER BALLINGER: The reports though, the sort of references that go back and forth.

CHAIR REMPE: He had a table here earlier and what they did --

MEMBER BANERJEE: Yes, he had one. He had St., what is it, St. Lucie?

DR. AZARM: No, Surry.

(Simultaneous speaking)

DR. AZARM: Should have had it because, you know, again, this is in the database.

(Off microphone discussion)

DR. AZARM: Just I want to make a note. I just want to be sure. To our knowledge the 690, the oldest plant is North Anna that has 19 EFPY and has shown no crack. So we have no crack with 690, so we can't generate any flaw rate generation for it.

So one part of the model is that to be able for each steam generator at any age to predict how many flaws am I going to see in the next cycle?

The second part of the model is that what is the characteristic of this flaw? What is their length? What is their depth, et cetera?

Again, we can split the hair and create many categories, you know. Basically we have been able with the limited data we have to define the flaw length distribution for wears and cracks.

And as you said and I agree, there's very few cracks in the database. They could not distinguish that much between 600 thermally treated and 690 thermally treated with the exception that there is no crack with 690 and you will see the result of that.

They were able to do something with the circumferential flaw length. Again, this is for cracks only, for 600 thermally treated, a small amount of data.

And they have done fits of the flaw depth distribution for both wear and crack and, again, they use one distribution for 600 thermally treated, 690 thermally treated.

Again, under a larger database, all these things can be separated and we can do much better resolution. I'm just reporting of what has been done.

What is the model? The flaw generation rate increase as a function of the steam generator service life, and that function was shown to be linearly increasing. So it's simple regression model using Excel, so there's no fancy statistics here.

So that's the way that they calculate it and measure and then they fit it to a regression and then they calculate it equivalent to flaw generation rate.

For the flaw length and depth events they tried to use gamma distribution. Again, no fancy fitting. You know, we could have taken large number of distribution, do a goodness of fit, et cetera.

MEMBER POWERS: But the length and depth, they have to be correlated. Length and depth have to be correlated. It has to be a bivariate --

DR. AZARM: Correct. Correct. It has not been done. Correct. I do agree with that. That's known to be independent and they have been fit with the gamma distribution. The parameters were estimated, again, very simply by matching the first two moments.

Okay, I'm going to read actually from my notes. That way we don't confuse you guys. This data summarizes the results of the statistical analysis. For thermally treated Inconel 600, the rate of wear flaw generation per tube is a linearly increasing function of the service life measured by EFPY and shown by k, and I think that's the very first equation.

So it basically says that k is basically lambda as a function of EFPKY is linearly increasing with an intercept at the coefficient.

The result also show little or no dependence on the service life for crack generation. That might be the result of very small data they had.

Actually I have written, this was surprising. I'm reading my notes. It is generally postulated that once crack are generated due to SCC, their number increase fast over service life, but we don't see that.

Gamma distributions were generally fitted for the flaw depth and sizes. Again, the size of crack flaws are only applicable to thermally treated 600 tubes. The distribution for the sizes of the wear are aggregated over both 600 and 690, so that's what this table is showing you and I think it's in the report also.

CHAIR REMPE: Well, it depends on what report you look at.

MALE PARTICIPANT: Yes.

CHAIR REMPE: It agrees with what's in the draft NUREG, but if I go to your letter report, there is a figure, or a coefficient of a half that appears in the first term on the right of that very, the volumetric flow rate and, again, it depends where I look in the report and so I was just wondering is the half supposed to be there or not, first of all?

DR. AZARM: Okay, I will give you an explanation based on my memory collection. When you have linearly increasing hazard rate, it's not that easy to estimate it from the data.

So what you define, you define the function of a cumulative probability distribution which I forgot, is log of PF minus one, something.

And that function is exactly like a hazard rate except there's a coefficient of one half in front of the coefficient. So that is not hazard rate. That's what was fitted and then hazard rate removes that one half from it.

CHAIR REMPE: Oh, okay.

DR. AZARM: So there is a mistake. If that's, you know, the factor of one half is coming because of the function they create to fit.

CHAIR REMPE: Okay. Could I also ask you

to bring up that letter report again and --

DR. AZARM: Of course.

CHAIR REMPE: -- look at Figure 4? It's at Page 10 of 33 in the letter report. It's at the bottom. It's also Page 7 of the actual, but it's Figure 4.

And this goes -- that, right there. And that's why I was asking the questions earlier about is it really appropriate to combine the data for these different plants into one distribution? As we heard earlier, there's a lot of parameters that affect how these steam generators run.

And again, our expert said that he was wondering if it wouldn't have been smarter to have come up with curves for each plant and then have an average one rather than combining the data for the various plants.

And I believe then he referenced Figure 8 of this report and pointed out that when you get to the volumetric/wear flaws, if you go to Figure 8, you can see there's a, it's really hard to justify that that's a good curve fit to represent the data. Oh, you passed it.

MALE PARTICIPANT: No. There it is.

CHAIR REMPE: Yes. And again, I guess I'd like to hear your thoughts on it, an expert at this.

DR. AZARM: I will tell you what is my viewpoint looking at these. So when you look at this, and I think this is that side k that you see there.

CHAIR REMPE: Yes.

DR. AZARM: That empirical is log of one minus PF, et cetera. That is like a linear but has a one-half coefficient in it. Let me see if I can explain.

First thing first, let's assume that I'm at 15 cycle and I am trying to predict number of flaws by the next cycle, so I have a time period I predict and I have a lambda and this lambda is changing over time.

So even if lambda was constant and I ask you how many flaws do you expect to see by the next cycle, you basically say is a Poisson distribution and it could be 10, it could be 20, it could be 25 and is a distribution. It's not one number. So even for a constant failure rate, if I get data from ten different plants, I should see a big spread.

Just the consequence of assuming a Poisson or a generic Poisson when we have a linear increasing hazard rate, are the numbers that we are going to observe. But there is a probabilistic variation already in the Poisson that is going to create the spread.

On top of that, you are correct, there's a plant-to-plant variation, so that means the lambda itself -- if lambda was constant, even if I precisely predicted lambda I would still see a spread in actual data, but if lambda itself is changing because of plant-to-plant variation, you have an added spread.

Now, to do that, creating two levels of uncertainties or variation in your models, that is a good thing to do, is a sophisticated statistic. Again, I am tempted to say wasn't in the scope of this type of a study.

I think I can do those type of study. I need more data, because as you saw in Figure 4, if I separate the data, I don't have enough data to predict anything.

So the question was that given what we have, what is the best we can do and the simplest thing we can do, understanding that what we predict is we predict that you will have 100 flaws, it could be 150 to 200 or it could be 50. It's going to be variation in it. But that said, that's my opinion of explaining that.

MEMBER BLEY: Well, I'm staring at this

thinking if I draw that line and call that a model I'm making things up. About the only thing this seems good for is what you were hinting at and that is if you draw some kind of an upper bound up there you'll be able to say I'm probably less than this with some confidence. That's not a bad thing to be able to do. But the hint that the line is meaningful, just doesn't seem reasonable to me.

MEMBER BALLINGER: Most of this data, this is all wear data and it is very steam generator specific, plant to plant, within a plant, steam generator specific. So that probably dominates everything, I mean, aka, can you spell San Onofre? I mean --

DR. AZARM: I hear you and, again --

MEMBER BALLINGER: Let me ask a really dumb question. If you considered that these steam generator tubes were flawless, in other words you didn't have to worry about this, and you got hot enough, if you just treated them as unflawed and just did stress rupture because you get the very high temperatures, does it make any difference?

DR. AZARM: Okay. We are talking about at least assumptions in the calculator software itself. But I will answer you at least and you will hear elaboration of that.

If we have a tube that is flawed, even it's a very shallow flaw, you either feel it because of you are pressurizing it or you feel it due to creep rupture. The size of length of the flaw is going to be used to calculate the leak area. That's what our model is, you know, so just think about it.

If you don't have the flaw and you are talking about steam tube, of course you can calculate, oh, under pressure induce I can tell you exact thing. You get a big rupture. It's this and that. But for the creep rupture, I can't tell you because if you have a microscopic flaw that is going to go first.

MEMBER BALLINGER: Yes, that's what I mean.

(Simultaneous speaking)

MEMBER BALLINGER: If you get rid of all these flaws and just run a stress rupture test, there's going to be small imperfections in the tube.

DR. AZARM: Correct.

MEMBER BALLINGER: And when you get up to those temperatures, that's where things start.

DR. AZARM: Correct.

MEMBER BALLINGER: And does it make any difference on the time at which you get, on predicted time of failure? Have these or not?

DR. AZARM: You will see hopefully -- there's one message I want to get across, that for Westinghouse plants, okay, it becomes more messy than that.

For Westinghouse we looked at, as a PRA part, both the pressure-induced and the creep rupture, so we looked at accidents like ATWS or main steam line break. That has nothing to do with the very high temperature in creep rupture. And then we looked at the severe accident that has to do with the high temperature and creep rupture.

MEMBER BALLINGER: I would combine pressure induced and creep rupture.

MALE PARTICIPANT: Just assume there's no flaws at all.

DR. AZARM: No. For Westinghouse plant you needed to have -- if you have even 50 percent flaw, you cannot fail it under pressure.

MEMBER BALLINGER: So it's creep rupture then?

DR. AZARM: Creep rupture but then the question of hot leg come, but let's get to PRA.

MEMBER BALLINGER: When does pressure-induced become creep -- anyway, okay.

DR. AZARM: I can tell you that somebody told us start as a temperature of 600 as a threshold but that's, I think, the PRA report and calculator report question.

MEMBER BALLINGER: Okay, so below that temperature it's a classic failure?

DR. AZARM: Correct.

MEMBER BALLINGER: It's not time dependent then?

DR. AZARM: No. No.

MEMBER BALLINGER: Oh, okay.

DR. AZARM: And the calculator model has both of them and it switch. Once it goes above 600, we used to have 800 but we were told 600, we switch between the two models. Very good questions.

Again, this is a very small scope effort just to generate some meaningful flaw distributions to do the PRA. And perhaps more work is needed and more, I don't want to call it sophisticated statistics but at least consideration that even if we fit a line, the intercept might have a upper bound and lower bound and gives us a range.

MALE PARTICIPANT: We're lucky that John Stetkar is not here.

CHAIR REMPE: I think it would behoove the major draft to acknowledge some of the limitations, more of the study and to present a few of the plots that are in this presentation or in the draft or the letter report.

MEMBER BLEY: That seems pretty reasonable, yes.

MEMBER CORRADINI: Yes, this is Mike. I've been listening to all this wonderful material stuff, but I think Joy's last comment is very important, that the draft, before it goes out, needs to properly characterize this. Otherwise, it will cause confusion.

DR. AZARM: Yes.

MEMBER BALLINGER: Because you're calculating LERF at some point.

DR. AZARM: Yes.

MEMBER BALLINGER: Right.

DR. AZARM: Okay, these are just illustration type of graph. They took four steam generator, each steam generator with 3,300 tube and they look at the consequence of this linear hazard rate or linear failure rate.

Like if I am in the 20th cycle and I have 600 or 690, how many flaws am I going to generate in that? So it shows you that at least based on the data we have, looks like 690 performs better.

MEMBER BALLINGER: Just if you step back and look at this, the enhancement factor on just failures due to flaws, stress corrosion cracking, whatever we call it, for 690 over 600 is supposed to be at least a factor of 20. I mean, one of these reports say. That's one of the criteria for using 690. I think it's 20, something like --

DR. AZARM: For crack.

MEMBER BALLINGER: Yes, for cracking.

DR. AZARM: That's correct.

MEMBER BALLINGER: But this shows only a factor of 800.

DR. AZARM: Yes, this is mainly dominated --

MEMBER BALLINGER: What's that?

DR. AZARM: This is mainly dominated by the number of wears.

MEMBER BALLINGER: By wear?

DR. AZARM: Yes.

MEMBER BALLINGER: Yes.

DR. AZARM: By volumetric flaws. I don't think crack is a big contributor to this.

MEMBER BALLINGER: This then is actually pretty informative, 690 is more subject to wear than

600 is, more susceptible to wear.

DR. AZARM: Yes. Yes. Okay, the most important flaw parameter that specifies failure resistance of a tube is the flaw depth. This graph shows the fitted and empirical cumulative distribution. Now, the lower tail --

MEMBER BLEY: And this fits the one you said is just two parameter fit?

DR. AZARM: Two parameter fit. Now, the lower tail, this area, of empirical distribution is affected by the error associated with measuring small flaws.

The reason I'm making that point is that with small flaws and shallow flaws what we notice in the small database we have, we had one plant who reported lots of flaws but they were very shallow, five percent, three percent depth, two percent depth, where the other did not.

And knowing that the small flaws has a bigger error in them, there is a possibility that, one, we are not accurate in lower tail, even in empirical distribution.

The high end tail which I'm talking about here, the high end of the tail is also affected by the tubes removed due to plugging practice since these flaws will not be available for further growth to larger flaw in the next cycle. So we also expect that the end tail on the other side to behave a little bit different.

If I had to do this again, I would have actually divide, would have divided the data to three regions. Region less than 15 percent depth, region between 15 to 35 percent depth and region of 35 percent depth and above.

And there are other reasons for it. For example, in the high end region our probability of detection is very high.

MEMBER BALLINGER: Above 30 percent they would have plugged it, right?

DR. AZARM: Yes, but new ones can be generated. That's why we get big ones. So I would have done three regions. I would have generated three flaw generation rate and I would have generated three gamma distribution. That way I would assure that each region is properly fitted.

MEMBER BLEY: As it is, you kind of fiddled this to above 95th.

DR. AZARM: Yes. As it is, now we have -- what happened is that when we wanted to do the Westinghouse and Westinghouse was totally dominated by high tail end of the distribution, we were getting underestimate because of just what I said.

So what we did, and you see that in PRA report and is very confusing, we tried to shift the gamma distribution to get a better fit at the high end and that's what is shown in the next graph.

The blue one is actually empirical one, the jagged line, the red one is the gamma distribution and the green one is the shifted one that does, because, you know, in that area, a small error is going to make a big difference. There's a big difference between 0.01 and 0.001. So I think you see that in the PRA document, that we shift that. I just wanted to tell you why we did that.

So, again, if we would have known this stuff at the beginning, that, you know, we are going to be very exact at certain part of distribution and certain part of the data is going to be needed, perhaps we would have done different types of approach for this part.

Now, the title of this slide is wrong. This is not 10 steam generators. What I tried to do, I tried to say if I take 500 reactor year and try to see how many flaws I'm going to see in 500 reactor year and apply my distribution to it, what is all the different size I see?

So I'm going to read from my note and then we can -- This graph shows the distribution of flaw length for the 10,000 detected flaw. Assuming 32 new flaw per reactor unit at each refueling inspection, 10,000 flaw would translate to about 312 refueling cycle or about 500 reactor years. So the title is not correct. It's not ten steam generator. It's 500 reactor years= worth of data.

This graph is based on shifted distribution using PRA study. It shows that we expect 4 flaws to be greater than 70 percent depth with 3 of them between 70 to 80 percent and one flaw between 80 to 90 percent. Why is that important? Because it also shows 2,730 flaw out of 10,000 or 27 percent have depth between 15 to 25 percent.

But the important thing is that if we assume that the regular steam generator tube rupture during normal operation can happen if I have a flaw deeper than 90 percent. If I have a very deep flaw, that even the 1,000 psi difference can cause my steam generator tube rupture. This data basically tells you you will have probability that's 1 over 500 or two to the minus three for the new replaced steam generator data.

So it kind of is saying that I'm

calibrating this, even what we have seen, with the steam generator tube rupture during normal operation and it doesn't look as bad.

This is the same thing for the length distribution, and if you don't mind, I won't go through it but it's the same information.

In conclusion, in summary this study generated some statistics that can be used for generating the flaw samples for running the C-SGTR calculator.

It is the first study of its kind for providing insight to and estimate the performance of replaced steam generators with Inconel 600 and 690. It is a starting point and provides insightful observations.

So I think this is the first type of work in this area and I think it gives us at least we can generate flaw samples to run our calculator and, yes, we can do much better but it does the job for right now.

CHAIR REMPE: Thank you.

DR. AZARM: Thank you.

MEMBER POWERS: Just interject a note that because of some previous discussion we had on the selection of the gamma distribution for doing this, I think it's a good selection in the face of having mechanisms generating flaws because none of those mechanisms do you understand very well so for any one of them you might at best know the mean size of the flaw from that.

Then if you maximize entropy for knowing just the main flaw, you come up with the gamma distribution but you come up with an exponential distribution. If you sum the effects of all those mechanisms, you end up with the gamma distribution for the sum.

So I fully support the idea of using a gamma distribution for this because it's, I mean, it's very consistent, has an internal consistency even in the face of multiple mechanisms creating flaws, so just kick that in for whatever it's worth.

MEMBER BALLINGER: I keep coming back to not quite understanding. These deep flaws and stuff, they would have, those tubes would have been plugged.

DR. AZARM: Basically --

MEMBER BALLINGER: Where am I wrong there? Thirty percent volumetric or more, 40 percent, they would have plugged the tube. And so any, and by the way, the rules are any circumferential defect automatically gets plugged so doesn't that shut off the distribution at some point or are you just assuming that the tubes are -- I don't know.

DR. AZARM: Oh, the tube. Let me at least explain my best understanding and Ken or Mica can help you. I am at end of this cycle. I am going to go look at all my tubes with flaws. And, yes, you are right, if I see anything 40 percent I'm going to plug it.

MEMBER BALLINGER: Right.

DR. AZARM: Correct? So if I have done my job properly, which we assume we did, when I start my new cycle, there shouldn't be any tube with 40 percent flaw. Now I am trying to predict next cycle or I want to go to the next cycle and I'm going to do my inspection.

During this cycle, I can have a loose part creating a wear flaw, deep wear flaw and I'm going to see a 50 percent flaw. It does not create --

MEMBER BALLINGER: So it's wear, it's a, because you're not going to get -- okay.

DR. AZARM: Okay? So also there could have been a flaw here 29 percent which I didn't plug and, yes, my growth estimate of that flaw says 70 percent chance this is never going to grow beyond 31 percent but there is still 20 percent, 30 percent chance that it does.

MEMBER BALLINGER: For cracks or wear

maybe, but in the oddball event of a loose part or something you can't --

DR. AZARM: Yes, yes.

MEMBER BALLINGER: -- say anything.

DR. AZARM: So it is a combination. Even though you are plugging them, when you look at next cycle you will see some new flaws as deep and there could be some old flaws that was not as deep now became deep.

Now, we have the capability from the reports to differentiate between those two even though right now we haven't done it, but we have the location of each of the tube and the flaw, how many inches it is above or below the TSP so we can say this big flaw you saw is as a result of the growth of a flaw that you didn't plug or this is a new flaw event.

MEMBER BALLINGER: I mean, I can understand the sort of random event, like a loose part or something, but the way these steam generators are operated, there's a statistical distribution of, statistical number of tubes depending on the ongoing issues of the steam generator that defines the inspection interval.

And if all of a sudden they see a larger increase or it goes outside, they change the distribution, increase the number of tubes that are inspected through a mid-cycle outage. So there's an operational part here that would tend to me to think to reduce the number of actual flaws that may get you.

DR. AZARM: We fully agree. That's why --

MEMBER BALLINGER: Okay. That's not reflected in these probabilities.

DR. AZARM: That's why I was worried about my gamma tail. That's what I was saying. Because of these big flaws are removed, I know it's going to affect my gamma tail. It actually has affected the shape of the flaw solution because you remove them.

Shall I go to the next presentation?

CHAIR REMPE: I think so. Are we ready for the next one, guys?

MALE PARTICIPANT: Yes.

CHAIR REMPE: Okay, please do.

DR. AZARM: Now we are in the calculator. To estimate the consequential steam generator tube rupture probability for a specific accident scenario and for a set of flaw in the steam generator tubes, the staff had its own software written in Java language.

This software use the fracture mechanic models developed by NRC and industry for steam generator tube failures, hot leg and surge line. That guy is happy to leave.

(Laughter)

MALE PARTICIPANT: I thought you guys were -- I'm glad you had lunch because I thought you guys were going to eat with us.

(Laughter)

DR. AZARM: In this short presentation, we are trying to give you an overview and please feel free, ask questions because I don't think, you know, the presentation is written at a very high level but I'm ready to discuss any detailed question you ask.

The main objective of writing this calculator was to support the C-SGTR PRA. NRC did have some spreadsheet before this project to do some of the calculations similar to this, but it was frankly very difficult to trace what was done and to modify it and update it. That's why the decision was made to write a documented software in language that everybody can modify and use.

This calculator is therefore probabilistic in nature. It captures the failure time and leak size of a steam generator tube with different type and sizes of flaw for both thermally treated Inconel 600 and 690.

The software also has built-in creep rupture models for hot leg and surge line failure for generic, Westinghouse and C-E plant.

This slide basically shows an overview structure of the software. We either can put a preexisting set of flaw, like somebody just does inspection and gets 150 flaws and knows the sizes. They can create an input or we can simulate or give him sample from the flaw statistics. We need the thermal hydraulic response of an accident that we want to analyze, and you heard Mike talking about that.

And what we expect from calculator is to estimate the probability distribution of the leakage area of steam generator tube rupture and the probability of RCS failure as a function of accident time.

The calculator performs these calculations for two types of accident, a class of severe accident, post-core damage with dry steam generator and high primary and low secondary pressures when the dominant failure mechanism is creep rupture.

It also evaluates C-SGTR for a class of so-called DBA accidents, prior to core damage, with significantly higher than normal pressure across the steam generator tubes. In this case the dominant failure mechanism is the plastic failure or pressure induced.

The information generated by the calculator then is used in PRA models to estimate the core damage probability and LERF.

The calculator is also equipped with a library of material properties for fracture mechanic analysis and library of all uncertainty parameters that we have used. And as I go through, I will mention to you what are the different types of uncertainty parameters we have.

The next one, it just shows you that this calculator exists. When you run it, this is the first page of calculator. It asks for plant name. It knows for each plant name the number of tubes, the size of the tubes, the material of the tubes and ask you for the accident TH values and it also asks you for critical area that you consider as a C-SGTR.

Next slide simply shows the four major fracture mechanic models of the calculator. These are pressure-induced failure and leak area model for flawed steam generator tubes, creep rupture failure and leak area model for steam generator tubes and finally creep rupture model for hot leg and surge line.

So this is basically the four models we have in calculator. And as we said, most of the models, the first two models is all based on past NRC work. The last two for hot leg and surge line coming from EPRI report.

Okay, this I think repeats what I just said. It basically says all the fracture mechanic models for flawed tube is NRC work. All the creep rupture of hot leg and surge line is EPRI model.

Material properties, we need them as a function of temperature. NRC did have 600 but did not have 690 thermally treated. We basically got those information from several sources of open literature because for each of the material, you know, if you look at ultimate strain, we not only want that as a function of temperature but also look at different sources who come up. What are the uncertainties associated with this?

So if you are at 700 degree temperature and look at XKSI is the mean and what is the error around it. So all these material properties also have uncertainties defined too.

CHAIR REMPE: How high a temperature did you find for the 690? How high a testing temperature did they go to? I assume it was lower than what you could find for the 600 because I think INL used to have it, at least 1,100 C, from the TMI days.

DR. AZARM: I don't know the answer to your question but we have that documented very detailed and we looked at, like, three different sources and we looked at actually one source on precious metal in the wear in addition.

CHAIR REMPE: Well, let me ask it in a way that won't require specific temperatures. Did you have data that encompassed the failure range where you were predicting --

DR. AZARM: Yes.

CHAIR REMPE: -- or were you extrapolating with the data?

DR. AZARM: No, we had the data, sufficient data that, you know, predicted the creep rupture failures we wanted. We did not ask for analysis. That's for pressure induced.

For Larson-Miller creep rupture parameters, again Inconel 600 we have a beautiful equations, et cetera, but for 690 we had to go and rely on open literature.

Again, we have documented what we have found and, again, we have accounted for the uncertainties to the best we saw.

Hot leg and surge line, these are coming from EPRI. Again, they are Larson-Miller type of equations. And as I was trying to say this morning, we can verify it. We haven't verified it. That's why we asked Dr. Iyengar to do independent calculations.

But the equation gives you the coefficients of the empirical model, Larson-Miller, has uncertainties or tolerances associated with that.

And we have taken those tolerances and assumed is a variance of a normal distribution and we have made samples from them and -- We have heard a lot about the sequence input, but what the calculation needs is the fraction of hot tubes and their average temperature, the fraction of cold tubes.

This software does not just look at hot tube. Also looks at cold tube because sometimes a difference between cold tube and hot tube is, like, 100 degree, 150 degree, and if you have a larger flowing cold tube, it can go first. So the calculator tries to look at both of them.

We also need to have the fraction of the hottest tube and you guys discussed this question of 20 versus 100 and it's your judgment but we basically get this input from the thermal hydraulic guys.

We want the primary and secondary pressure because we have to calculate the stresses on the hot leg and surge line and the tubes, and we also need to have the hot leg temperature and surge line temperature to be provided to us.

MEMBER BLEY: How did you decide how big those groups ought to be, the number of hot tubes and the number of cold tubes? There must be an iterative process because number of feed ruptures --

DR. AZARM: I will ask Mike to help, you know, with me but basically think about, at least from my viewpoint, when these things goes through a steam generator, exactly think about fire.

You have a plume and there's a variation of temperature and the hottest tube to me is, they call it group hottest and then they have average hot and average cold.

Now, they have done some CFD calculations. So far they have some idea that, you know, if I think about this as a plume on the top 25 percent is called hottest. I don't know.

(Simultaneous speaking)

MEMBER BLEY: Let me ask it a slightly different way. What I'm interested in is you must have really been thinking of the PRA as you do this, how many tubes do I need for this to be really significant, and done something like that and maybe iterated on it or did you just arbitrarily take some raucous group that's within 100 degrees of each other or something?

DR. SALAY: I don't know how he -- This is Mike Salay. Well, ultimately it came, the temperature distribution came from CFD and they had a hot region and --

MEMBER BLEY: I'm not quite saying right what I'm after.

(Simultaneous speaking)

CHAIR REMPE: Can I interrupt?

MEMBER BLEY: If you took the very hottest tube -- in a second. If you took the very hottest tube and used that, that would be one piece of information but you'd only get one tube, the troublesome one or being vulnerable.

If you took, you know, some arbitrary Number 11, the average temperature would now be lower so it wouldn't be as severe but you'd have more tubes. How did you decide how to do that selection? I don't --

CHAIR REMPE: I think this morning you were missing. We discussed this a bit.

MEMBER BLEY: Oh, you went through that?

CHAIR REMPE: And so let me summarize what we had --

MEMBER BLEY: Sorry, I --

CHAIR REMPE: -- for the Westinghouse

ones. That's why I --

MEMBER BLEY: Okay.

CHAIR REMPE: -- keep wanting to interrupt you. But for the Westinghouse ones, there was this document or this plot that was in the older NUREG.

But for the C-E ones, and there's a couple of sentences in the draft NUREG that they had some experts from NRC decide to use 20 and then there was some additional discussion that as long as it's big enough it doesn't matter.

Am I summarizing what I took away from this morning? So this is a good education for me if I missed something.

DR. SALAY: And I was talking for the, referring to our calcs and when we did the failures and I can't remember exactly how Ali, I mean, split it up in hottest and, or did you get that from us or --

DR. AZARM: I got that from your report, yes.

DR. SALAY: Oh, from --

DR. AZARM: What we basically did, I think we went to your report.

DR. SALAY: Again, my memory is not --

MEMBER BLEY: I'll go to the transcript.

CHAIR REMPE: Well, again, it's good to

discuss it further because maybe I'm not understanding but I'll have to write a letter at some point perhaps and it's good for me to understand it. It was expert opinion. It wasn't a formal NRC interaction. It was --

DR. SALAY: And then Ali used this -- if you use that 20, then that was just, well, how many tubes are going to be the hottest and said about that many and that it's -- we were going to, it would have been nice to look into it further but we didn't, so.

DR. AZARM: And I just want to add something. Perhaps me and Dennis think a little bit similar from PRA words. I am just wondering if your question relates why we are deciding what is the critical size of the break, is it one tube, two tube, ten tubes? Is that what --

MEMBER BLEY: Well, it's related to that but, see, that's what -- I'll say, as I tried to before, if you took the one hottest tube, then you would be more vulnerable I suspect but you'd have less chance of having the flaw in there.

DR. AZARM: Correct.

MEMBER BLEY: If you took a slightly bigger group, you would have a lower average temperature, you wouldn't be as vulnerable, you'd have more chance of a flaw.

This is eventually going to get to when we talk about the PRA how did you treat the uncertainty having to do with this part of the problem?

DR. AZARM: Yes, we will talk about that. Just very briefly I'm going to tell you and we will talk about it.

Basically I think we have, just think about it. We have 100 hots and we have 2,000 average hot and 1,000 average cold and we have 150 flaws. Random we throw them on these tubes, correct, and then see how it fails.

MEMBER BLEY: Given the kind of model we talked about a few minutes ago, yes.

DR. AZARM: So that is what is done in PRA.

MEMBER BLEY: But there is some uncertainty there that maybe things in different temperature regimes would over that, since last refueling might be more likely to have a problem or not.

So what I'm eventually going to wonder about is given all of these things that you could have done if you had all the money in the world, do we have a conservative result? Have we addressed the uncertainty in all of these various things or not? And I don't expect the answer here but eventually I want to --

DR. AZARM: Okay, we will discuss that to the best we can.

CHAIR REMPE: Because we didn't bring it up this morning and I hope Mike's still back there or someone can, when you picked the number of hot tubes to be 20, I think there was some discussion about this was big enough to get a large enough size.

But I don't recall the report ever mentioning any interaction about MELCOR, because you picked 20, you have a lower peak temperature. Did that type of, was that something that MELCOR was considered but --

DR. SALAY: No, it didn't. We just, I mean, there was actually, you simulated, it was just one tube for, in the MELCOR analyses you have one tube that's hot and one tube that's average temperature but it represents multiple tubes and the hot tubes taken to represent 20 and just --

CHAIR REMPE: So there was no interaction. If you picked 30, you wouldn't have had a lower hot temperature?

DR. SALAY: No.

CHAIR REMPE: Yes.

DR. SALAY: But ultimately you are

characterizing the CFD, the distribution of hot tubes going in, so you know how many tubes, I mean, show the distribution for Westinghouse. It was never evaluated for C-E.

But, I mean, so the hottest and the medium give you sort of an anchor that you can put the whole distribution on and so you can actually, I mean, the information is there if you take the time to extract it.

MEMBER BLEY: And is that kind of what was done? You took the hottest one and used that as the temperature for the hot group when we get later views in it somewhere else?

DR. SALAY: No, we gave them the hot temperature and the medium temperature profiles and then I can't remember exactly how --

DR. AZARM: Average cold, average hot tube and the hottest tube.

DR. SALAY: And hottest tube. And then how many tubes were in the plume? I remember that information we gave and now it's in the whole thing and I guess you're using the 20 that we used for the hottest or --

DR. AZARM: I cannot swear.

DR. SALAY: Yes, I can't remember.

DR. AZARM: You know, the number that goes around my head is 100. Don't ask me why, but I think because we needed a number of tubes exposed to hottest for us to throw our flaws at it randomly.

I don't know. My memory says 100 but I am aware of the 20 number that you are discussing also, that that was what failed, but we will get to that --

MEMBER BLEY: Okay.

DR. AZARM: -- in the PRA report. Okay, what is the calculator output? Put simply, when we run this calculator with this input, it has to tell you as a function of time accidents, what is your leak area and it has to give you that in a probabilistic manner.

We can't tell you the leak area at ten minutes is two centimeters squared. It has to tell you what -- it tells you at least what is the 5 percentile, 25 percentile, 50, 75 and 95 percentile. So it gives you a distribution of leak rate as a function of time.

Also it gives you the probability distribution for the time that we expect the hot leg or surge line fails or, in a sense, at each given time says I think there's only 5 percent probability that hot leg fails and sometimes later it tells you it's 25 percent probability.

So it indicates out to the best it can

including all the uncertainties, probabilities for the failures and leak area of a steam generator tube rupture.

So as was discussed this morning, uncertainties are very important in these calculators and, unfortunately, that's the one we had very hard to find and determine.

These are criteria basically showing you the leak area as a function of accident time and, you know, they are showing you the mean, the 5 percent, 25 percent, et cetera, so you get some appreciation of where that this calculates, the calculator calculates as uncertainties.

So if the critical area is 6 centimeters squared, then you will know that there's 5 percent chance that you get that by 14200 seconds. So that's the type of output the calculator generates.

MEMBER BALLINGER: So a leak area going down with time?

DR. AZARM: No, it shouldn't. It's a manifestation of the curve fit. It should go up all the time. Because once, you know, you opened up a crack or you failed the tube to certain size, it's not going to shrink back.

MEMBER BALLINGER: Okay. That's the way

I would look at it.

DR. AZARM: Yes, yes. No, now it, you know, they have done Excel fit and looks like it's going down but, no, it basically, what basically showing, one big flaw might have failed early on and grow and then waits for a little while and now the tube with the big flaw fails. So it's just, it's like a stair or steps. Goes up as more tubes fail and then it goes, so. Now, we --

MEMBER BALLINGER: I'm just struck by the axis. All of this really happens over a 100-second interval.

DR. AZARM: Yes. Yes, yes, you are right.

MEMBER BALLINGER: Go from the, plus or --

DR. AZARM: When I show you the TH result, really nothing interesting is happening until you start having Zirco oxidation, exothermic reaction, rapid heat-up and by the time that happens, in couple of hundred seconds, in less than ten --

MALE PARTICIPANT: Temperature goes up pretty fast.

MEMBER BALLINGER: Well, I understand the temperature goes up.

DR. AZARM: And that's when everything fails.

MEMBER BALLINGER: But I guess we come back to the thing we talked about this morning. When does that point occur? I mean, how is it, you know, everything happens over a 100-second span over 14,000 or 14,000 and change seconds.

DR. AZARM: Yes. When I looked at this, everything is happening within ten minutes, let's put it that way. Now, within, you know, 600 seconds and actually the very rapid rise over, is going to happen within couple of hundred seconds. That's why when we talked about the why is the hot leg uncertainty so small I was saying because there's a very rapid rise of temperature.

If I would run the same model at 700 degrees centigrade constant, you will see a very big uncertainty. But when the temperature is so dynamically increasing, it doesn't allow a big uncertainty. It just squish everything together. These are illustration. We are going to see some actual results later on.

MEMBER BALLINGER: So I'll stop.

DR. AZARM: The other slide basically shows that we sometimes use graphs that talks about RCS survival and survival of a steam generator tube rupture which is one minus three probability. There's a reason for that. Because all these things goes through a post-processing before enters PRA.

Now, I want to very fast talk about different sources of uncertainty. I am not claiming that we did a great job with them. We had to do many shortcuts, but at least this identify what are the major sources of uncertainties that the calculator tries to handle.

The first one is the modeling uncertainties. All fracture mechanic models has to have uncertainty. Even though some of the NRC reports gave us an equation, empirical equation with no uncertainty, we tried to go back to the actual test data and add an error term in the lump form of uncertainty.

MEMBER BLEY: Did you do that analytically or was that a judgement call?

DR. AZARM: No, we basically calculated the variance versus predicted and fitted to a normal. The problem was that, and this is very important, the problem is that everybody adds error term as plus epsilon.

In many of these models, the error term is predicted minus observed is one plus epsilon multiplied by predicted, is multiplicative which usually they don't do.

So we saw some of the multiplicative ones so we put them in a lump but either we use as a summation or multiplicative. The data sometimes was varied, so it's crude but we tried to do that approach.

Measurement uncertainties is a second type of uncertainty we dealt with. We used it in material properties, so when you look at the report and you see long modules, you will see that it's just not giving you the curve as a function of temperature. It gives you lower bound and upper bound.

MEMBER POWERS: When you include measurement uncertainties, do you attempt to handle failure to detect?

DR. AZARM: Failure to detect flaws?

MEMBER POWERS: Yes. I look at a piece of metal and I say there's seven flaws but there are really eight because I just don't see one.

DR. AZARM: It is a very good question. I will tell you of what was our position and our thought process and the gentleman left. This is a question --

MEMBER POWERS: Yes, I know. He got away.

DR. AZARM: I will tell you. All the statistic you do is undetected stuff. If somebody says I can do a statistic on the stuff, that was not detected. So all the statistics that you saw in the flaw is stuff that is detected. Hot leg accounts for the probability of detection.

Usually you do it in a controlled environment and saying, okay, for this type of flaw, this type of flaw the probability of detection is only 50 percent. With this bigger flaw that is crack, the probability of detection is almost 99 percent. You create a curve. And then external to your statistics, you put them together. So the probability of detection and the statistics of flaw has to be joined at the end.

Now, we understood that that's going to be our approach but we have never accounted for curve in our report because we didn't have the right S curve to use for this flaw stuff, but that was our thought process.

Numerical errors in estimation. Again, we did not treat that formally as you know. For example, if I want to estimate a 95 percentile of the distribution, how many samples do I get? If I have 50 different variables, how many samples do I get? If I do numerical integration, which we do quite often in Larson-Miller calculations, how good is our integration routine? How smart is our time spent? It was not done formally.

We did some MCAD calculations and compare rate, et cetera, and we just have the decision to use the calculator stuff. So that's the third type of uncertainty.

The fourth type of uncertainty is that we discussed this morning. First off, you know, one is that different plant have different flaw distributions and there's plant-to-plant variability. So I have to run all these many cases and then combine them outside the code in order to get an appreciation of uncertainty.

Also when we say short-term or long-term station blackout, there is no defined scenario called short-term station blackout or long-term station blackout. If you look at it, sometimes they do fast cooling. Sometimes they do, you know, slow cooling. The status of RCP seal leakage is different. So all of these things we group sometimes and we analyze and we try to use a bounding analysis for it.

But those can be, you know, depending on what is your approach, this last type of uncertainty has to be accounted for. Again, we didn't do the last type of uncertainty formally. We tried to the extent possible this bounding analysis.

MEMBER SKILLMAN: Back on your measurement uncertainties, that's back a slide, how do you account for the differences in the discovery by the rotating pancake coils and the bobbins from plant to plant with different gains and different blocking types and different manufacturers of those bobbins?

DR. AZARM: We didn't. You know, the measurement, uncertainty of flaw size that here we have, we have a placeholder. I think we are using three percent right now.

But we fully understand that it varies based on the steam generator tube design, based on different eddy current testing and rotating coil. We know it's different but we are not going to that level of resolution right now. We have a default value and we are putting three percent for it.

MEMBER SKILLMAN: A three percent uncertainty?

DR. AZARM: Yes. So if you are telling me something is ten percent deep, it's a 9.7 to 10.3 right now.

MEMBER SKILLMAN: Why is the three percent a good uncertainty?

DR. AZARM: We didn't have any input from anybody at that time so we put just a placeholder and that's the number that is, right now is default. But yes, I truly understand and I don't have, I understand the reason you are asking that question but I do not have the knowledge to give you the answer.

MEMBER SKILLMAN: Okay, thank you.

DR. AZARM: You're welcome. Another important thing. I think a lot of talk was given this morning about uncertainties but you have to not look at uncertainty magnitude but also look at the impact of uncertainty.

And I just put a graph here to show my point. This is basically a Larson-Miller parameter. As you know, you know, if I am over 600 or 700 degrees C, I shouldn't use the previous model and I have to use Larson-Miller to make sure that I'm not unnecessarily conservative.

And when we fit this in a log scale, it looked very good and actually when you look at, you know, at the uncertainty is showing up like a little bit 0.44 next to the intercept parameter and that's the fit. That's the fit of the regression.

But remember, this is now creep rupture so you are going to take ten, raise to that power and then you want to integrate, make it equal to one.

So that very little uncertainty that looks like a great fit, if you go with a constant temperature, it can create a factor of ten difference in the time of failure.

And you guys were real appreciative of that in discussing it this morning and we fully understand that. Again, some of these uncertainties was the squish because of the behavior of TH. How am I doing on time?

CHAIR REMPE: You're a little bit over but I know you only have a couple more slides so let's let you finish and then we'll take our break then.

DR. AZARM: Okay. I think this slide shows an actual analysis of one loop. Just for your information, in Westinghouse plant, they did not differentiate it when they did the TH analysis which when looked at pressurize and look without pressurize it.

In C plan the TH or the temperature of the hot leg tube, et cetera, depends if you are in the loop with the pressurizer or you are not in the loop with the pressurizer.

This slide shows an actual analysis of one loop, loop without pressurizer of a C plan. In this analysis the criteria for C-SGTR leak area for the plant is set at six centimeters square or for one loop it's set at three centimeters squared.

The result clearly shows that there is a competition or race between the RCS failure and the

C-SGTR with leak area greater than three centimeters squared. Therefore, containment bypass probability will be relatively high, it=s expected to be around 0.5. It basically talks about when we look at --

CHAIR REMPE: You should use the mouse and point with the screen because the transcriber can't get your comments.

DR. AZARM: When we look at the probability that RCS survive, let's take a number, 46,500. The probability that RCS survive is 0.55 so the probability of RCS failing is 0.45.

And you will see that the probability of C-SGTR to just exceed the three centimeter squared, that criteria, is around the same thing. So at that point I have a probability -- No. I have to actually do conversion and integrate over time. I'm just making it very simple. So there is a probability about 0.25 just at that point.

So when you have a C-SGTR probability that is high, usually you see these two curves are very close to each other across of each other. And, you know, when they are separated, then the probabilities become low. Actual numbers that goes to PRA, they come from point-by-point integration of incremental probability so there's another routine in addition to calculator that does that post-processing calculations.

This is more confusing graph. It basically now is trying to superimpose the 5 percent, 25 percent, 50 percent leak area distribution over the RCS and try to tell you that at any point that you can calculate the probability of RCS failure you can actually calculate the distribution of leak area.

So that way you can, if you change your leak area criteria, you can calculate with the post-process the new number. So those are the type of stuff we can do with the calculator.

Okay, my last slide. We have a calculator. We do some verification and validation on it. We basically, many of the empirical distribution, they also have both testing as well as results published. We try to compare them with them. Many of them we also did it on MCAD on this input platform and compared the result.

And we asked Dr. Majumdar of Argonne National Laboratory to review it and commented and we tried to resolve his comment.

Be fair to him, when he reviewed the calculator, was in draft format and didn't have all the whistles and bells on it that it has today. So he reviewed a version, earlier version of calculator.

You heard about the use of ABAQUS this morning for at least trying to see if this hot leg failure is meaningful, is reasonable and that was talked by Dr. Iyengar. And that's basically the activity that was done for V&V. I think that's --

CHAIR REMPE: So predicted 50 percentile so you were right with the deterministic --

DR. AZARM: No, we weren't.

MALE PARTICIPANT: I think that was your papers right on the microphone.

DR. AZARM: Oh, sorry. I think, believe when he did his analysis, 50 percent was saying that the hot leg is going to fail at time T0 and he was basically coming up that hot leg was going to fail at least couple of minutes later. Am I correct or the reverse?

CHAIR REMPE: Okay. So MELCOR predicted a couple minutes later than the calculator or what are you -- I'm looking at the last bullet here, the 50 percentile, so the predictions.

DR. AZARM: Yes, that is related to what Dr. Iyengar did.

CHAIR REMPE: So it was the ABAQUS in this.

DR. AZARM: ABAQUS. ABAQUS basically tried to use the same MELCOR. They did their own heat transfer. They did their own distress calculation accounting for the delta T, et cetera, and they tried to deterministically, because they cannot calculate any uncertainty, calculate a time of hot leg failure.

And they find out at least their number is, like, couple of hundred second low of 50 percentile. So he was basically saying for Westinghouse the C-SGTR probability is, what we are calculating is bounding because he's predicting that the hot leg is going to fail area. That was the last bullet.

CHAIR REMPE: Okay. I didn't understand both bullets were connected. Okay, let's take a break, right, and come back at 2:45. Thank you.

(Whereupon, the above-entitled matter went off the record at 2:28 p.m. and resumed at 2:47 p.m.)

CHAIR REMPE: We're going to go back on the record, folks. And just before you start at about 3:30 Corradini has got to go catch a plane, and so I may interrupt during your presentation just to ask him if he has any final comments if that's okay, and I apologize, but we're juggling with a lot of different people and their schedules.

MEMBER POWERS: Do you know that your conference is still on?

MEMBER BALLINGER: Yes, I was curious, I thought it hung up.

MEMBER POWERS: Yes, I think it hung up.

CHAIR REMPE: It did, and he, we're aware of it and he's trying to reconnect and that's why I was trying to see if he wanted to do it now.

MEMBER POWERS: How unfortunate.

CHAIR REMPE: It's a good thing he can't talk to you right now, Dana. Please start.

DR. AZARM: I think Kevin wants to say a summary?

DR. COYNE: Yes, just a quick -- Kevin Coyne, Research. Just a quick clarification, well I guess it's not even a clarification, but just to re-summarize this key point on how the calculator handles tubes and hot/cold/hottest.

DR. AZARM: All right.

DR. COYNE: So we get thermal-hydraulic input from Mike Salay.

DR. AZARM: He's been most wonderful.

DR. COYNE: He has three categories, average cold, average hot, and hottest. The way they broke that up was informed by the detailed work they did for the Westinghouse Plant earlier and then judgment from how the CE CFD work went, but they did not redo that work for CE, they just believe that's not, and I'll say it kind in the negative, not an unreasonable assumption to make for CE based on their experience with Westinghouse.

But that breakdown between average cold, average hot, and hottest wasn't redone for CE. It's informed from our work with Westinghouse. The calculator doesn't actually, you know, count tubes, what it does is when you generated flaw distribution from the data that we talked about earlier it will allocate those flaws into one of those bins based on a user set input into the calculator.

So the user could choose to put all the flaws in the hottest, all the flaws in the average hot, all the flaws in the cold, or some distribution of doing that.

For the work that Ali is going to talk about in the next section how that allocation went was 1 percent in the hottest, 9 percent in the average hot, and 90 percent in the cold, again, informed by some of the earlier work done for the Westinghouse and some of the CFD work that was done.

That's a user input that could be changed to do sensitivity studies if need be.

MALE PARTICIPANT: Thank you.

DR. COYNE: And I don't know if we said outride the significance of the six square centimeters is an equivalent to diameter, so which seem to be a reasonable presumption for when you have C-SGTR bypass.

CHAIR REMPE: Thank you.

DR. AZARM: All right. So now I'm going to try to explain what we have done as a part of simplified probabilistic risk analysis for estimating LERF due to C-SGTR.

And I'm going to discuss both for severe accidents and the so-called DBA accidents. However, the main focus of this study was on the severe accidents.

MEMBER BLEY: Excuse me. Let me ask a simple question, I don't remember. What part of this work was borrowed from previous PRAs and what part is new for this work?

DR. AZARM: What --

MEMBER BLEY: Could you use the existing, previous PRA and how much of it, or what parts have you changed?

DR. AZARM: Oh, okay. We will get to that.

MEMBER BLEY: Oh, okay, if it's coming that's fine.

DR. AZARM: We basically for the identifying sequences of accidents we used IPEEE for Zion and Calvert Cliffs.

MEMBER BLEY: Okay.

DR. AZARM: So we went through them and we identify sequences that you are interested and then we try to work in the LERF into it.

MEMBER BLEY: Okay.

DR. AZARM: So that was for the severe accident part. For the other one, the DBA one we used the stylized analysis, and I talked about it.

MEMBER STETKAR: You didn't look at how the existing PRAs that you looked at might not have identified all of the conditions that could lead to this, did you?

DR. AZARM: No.

MEMBER STETKAR: Let me get to the point that I'm going to make.

DR. AZARM: Okay.

MEMBER STETKAR: I have a 2-train plant, two steam generators. Something happens that causes a plant to trip and my atmospheric relief valves open on both of those steam generators, as they would, there's no reason to believe that one would open versus two.

DR. AZARM: Okay.

MEMBER STETKAR: And one of them sticks open. What do the operators do? Here's a question for you, what do the operators do? You know the answer to this right away, right?

The operators are instructed to immediately isolate all feedwater flow to a faulted steam generator. They will do this, I have a faulted steam generator.

So they will immediately isolate all feedwater flow to that steam generator leaving that steam generator open to the atmosphere with no water in it, leaving one steam generator left to cool the core, and there's a whole bunch of ways I can get the core to melt after that.

Now how does the PRA that you used look at those scenarios? The answer, it doesn't. It looks at it perhaps from the perspective of a spuriously open safety relief valve on the secondary side as an initiating event, which is one of your so-called design basis initiating events.

But it doesn't look at a risk assessment that says how confused are the operators going to be now if they're starting to get into trouble where they followed the procedures, they've had an event, a safety injection that occurs because it had an overcooling transient, and they have a whole lot of other stuff going on.

You presumed that the operators behave as if this is an initiating event that starts by the valve opening by itself or that somehow you can model this as a station blackout with failure of the turbine-driven aux feedwater pump, which it certainly isn't.

The question is is in terms of drawing conclusions about the risk implication of these induced tube rupture events, consequential tube rupture events, with large early release fraction, fraction contribution to large early release frequency, I maintain that the risk assessments that you've used are incomplete.

Now how important they are I don't know because I have never done a PRA. I've been advising people to put these overcooling transients into the model because they can cause problems that increase the core damage frequency and perhaps increase the large early release frequency because of consequential tube rupture over the last four or five years, but I haven't seen anybody who's actually done it.

Now I haven't done it in any of the studies that I've ever done.

DR. AZARM: It's a very good question.

MEMBER STETKAR: So I don't know and that's the whole point I'm getting to in terms of trying to draw conclusions about how important might this be based on the snapshot of particular scenarios that you've taken from preexisting PRAs.

DR. AZARM: Okay. Let me see if I -- Okay.

MEMBER STETKAR: And that's --

DR. AZARM: -- can shed some light to them.

MEMBER STETKAR: -- because that doesn't, none of what I said changes anything to do about some of the stuff that I missed, either the thermal-hydraulics or the materials properties.

It does change some of the timing I think in terms of the relatively likelihoods of hot leg rupture versus larger releases.

DR. AZARM: Okay. Let me see if I can shed some light, and I'm sure we have not covered all the bases.

We actually did look at a spurious actuation of a steam generator relief valve as an initiator. We did understand that a steam generator is considered faulted.

MEMBER STETKAR: Okay.

DR. AZARM: And we did understand that the plan is going to behave that it got a main steam break and the operator is going to go and try to close the high pressure injection because there is no reason to inject SI because of the overcooling.

Anyway, we were, and then we said what if I get a tube rupture because of this large delta P I have caused on the steam generator to fault.

MEMBER STETKAR: Yes.

DR. AZARM: Now I have something that acts like a main steam line break, at the same time has a steam generator tube rupture in it, and the operator is already confused because early on he had to stop injection, now he has to turn it on.

We did that but not on the severe accident --

MEMBER STETKAR: Not on the severe. See that's my concern, yes.

DR. AZARM: On the design basis accident.

MEMBER STETKAR: Yes. I'm transitioning from, I don't like this severe accident design basis, I'm just trying to paint a scenario for you.

DR. AZARM: Yes. Even though the DBA, so-called DBA accident was not our focus we did try to look at a series of them.

MEMBER STETKAR: I saw that, but most of what you focused your results and conclusions are on that severe accident which focuses, in my understanding, if not exclusively, primarily on the station blackout with two different findings of the turbine-drive aux feedwater pump failure.

DR. AZARM: Okay. I will try to -- And you are right.

MEMBER STETKAR: And then adding SAMG space though, but in SAMG space and SAMG timing operator actions to perhaps depressurize to get a low pressure feed into the steam generator or something like that.

But, again, it's a timing issue. It's not the same scenario context that I painted originally.

DR. AZARM: Yes. No, I fully agree that when NRC came to us the main focus was that core damage, onset of core damage has already occurred --

MEMBER STETKAR: Yes.

DR. AZARM: -- and you have finished Level 1 and you just want to see the conditions of the --

(Simultaneous speaking)

MEMBER STETKAR: But see the thing that I took issue with in the NUREG, the draft NUREG, there are certain statements that says well, if you have a Level 1 PRA and a Level 2 PRA it's easy to identify these intermediate states.

And my point is that not from any PRA I've ever seen yet, because none of them that I've seen challenged this. They all look at do I have enough steam release yes or no, and if I have enough steam release few of them ask do I have too much and what happens if I have too much, in terms of it, well I call it plain vanilla transient response.

DR. COYNE: And just so I understand, because I think we've talked about this with similar scenarios on another project.

MEMBER STETKAR: We have, on another project that's true.

DR. COYNE: So the case that we would have is a core damage event with a faulted but isolated steam generator.

MEMBER STETKAR: Faulted but isolated steam generator, right.

DR. COYNE: And one or more steam generators with presumably aux feed availability so the aux feed system is available however there might be other reasons if you get into a core damage scenario such as RCPC.

MEMBER STETKAR: Aux feed could later go away to the remaining steam generators. I mean I don't know how we get to core damage on this.

DR. COYNE: Right.

MEMBER STETKAR: I'm not trying to --

DR. COYNE: So I think a short answer would be the way they count or identify the scenario is if you lose aux feedwater that would be counted as something that would lead to a dry steam generator condition.

I think the more difficult one to count are the cases where you don't lose aux feedwater. Say you have this case where you may have one dry steam generator --

MEMBER STETKAR: That certainly would be more difficult.

DR. COYNE: Right. And that is just -- I'd agree that would be difficult to find in the PRA where you actually have done a thermal-hydraulic analysis to see how that --

MEMBER STETKAR: What I don't know, and I don't know whether it was covered when I was out so excuse me if it was.

What I don't know is, I know that you've assumed that a steam generator without feed is depressurized, is that statement throughout, but it says that the assumed whatever it was sized break was enough to depressurize you to get a delta P across the tubes but not enough to I think depressurize the primary side, or something like that.

In other words, I don't know what, the thing I'm trying to understand is if I go to core damage, however I got there, with a stuck open dry, a dry steam generator with an open atmospheric relief valve, or safety valve, atmospheric relief valve, is the timing of hot leg rupture under that case different from the conditions that were used for your station blackout core damage progression because of the presumed pressure in the secondary side of the plant, given the assumed whatever it is, half square centimeter or whatever it was, break is not nice, but leakage?

DR. AZARM: Mike had a set of runs which we didn't like very much, but what he did, he assumed for the secondary side SRVs stuck open at the beginning of transients.

MEMBER STETKAR: Okay.

DR. AZARM: And then he did that run, the situation got worse.

MEMBER STETKAR: I've read those words, yes.

DR. AZARM: And when recalculated the conditions of C-SGTR probability, instead of the normal

0.22 we got 0.99. So --

MEMBER STETKAR: Well, and it would go to a large early, a larger release also because the whole core is a fail. It's a --

DR. AZARM: Right, yes. Yes, there is a run done like that and, you know, I heard this morning or this afternoon they quoted CES.22, but that's for various stylized --

MEMBER STETKAR: Yes, that's --

DR. AZARM: When we did this specific scenario with SRV open we get 0.99 for CE plant. Now we didn't have a Cleveland run --

MEMBER STETKAR: For Westinghouse.

DR. AZARM: -- as we did for Westinghouse.

MEMBER STETKAR: Couldn't we presume it would be higher than the 10 to the minus 2 --

DR. AZARM: Going to be higher, yes.

MEMBER STETKAR: You don't know how much higher, yes. Okay. This, by the way, the reason I said 2-train plant, it's a much bigger issue on a 2-train plant.

Your conditional core damage probability given, especially when you start throwing in possible operator responses is a lot higher. But a 4-loop plant like, you know, the design plant, some more forgiving.

DR. AZARM: Yes.

MEMBER STETKAR: You know, you're not going to get all of the valves stuck open on all the steam generators for example.

DR. AZARM: Anyway, this slide is very simple. It basically says there was an NRC/NRR user request. In response to that they have put together a draft NUREG together and it involves several different offices of NRC Research to pull this together and integrate it.

What was the objective of this simplified PRA analysis? Just basically they wanted a quantitative assessment, simplify for the use of NRR, that can address the C-SGTR during a severe accident after the onset of core damage and during a DBA event before the onset of core damage.

We wanted to have some simpler way to calculate the change in LERF and core damage probability due to containment bypass and they want it to be piloted on Westinghouse and CE, and later on a guide to be written that has not yet been written.

And there are a couple of terms that needs to be defined, C-SGTR I put in blue means a leakage that are greater than the threshold value of some leak area before a large vent path in RCS is established.

This vent path could be either due to the component failure at hot leg or surge line or could be intentional depressurization. Of course, LERF is a C-SGTR that happened within the first ten hours.

MEMBER STETKAR: But LERF, and Kevin help me out, LERF in this study is defined as something before you have confidence that 95 percent of the population can be evacuated, right? Is that the definition that you're using?

DR. AZARM: That's the definition.

MEMBER STETKAR: Okay.

DR. AZARM: We are very much driven by evacuation.

MEMBER STETKAR: Yes.

DR. AZARM: Okay. This basically gives you an example, again, my notes doesn't read, of the differences between what the distinction between the C-SGTR after and before core damage.

Occurrence of C-SGTR after core damage is usually due to creep rupture high temperature. It can contribute to LERF if RCS remains intact. Of course, it cannot contribute to core damage because we assume core damage has already occurred.

On the contrary the C-SGTR before core damage could occur due to pressure-induced, there is no reason for us to believe very high temperatures can be experienced.

It can lead to core damage and LERF so it can have delta increase in core damage on LERF exactly like some of the scenarios that Dr. Stetkar has mentioned.

An example of C-SGTR after the onset of core damage could be a fast or a short station blackout. This results in secondary side boil off establishment of counter current flow, hot leg, and onset of core damage, additional heat up and occurrence of C-SGTR when RCS is intact.

This is what you heard a lot this morning, so there's nothing really new in this slide. This is a pictorial that shows the timeline of various events and classification of them, or classification of releases into LERF.

It also highlights in some cases the C-SGTR may occur prior to hot leg failure, but is not of sufficient size to depressurize the primary so a hot leg could fail shortly after, and I think that's Case 3.

The selection of critical C-SGTR area should ensure that sufficient depressurization would fall opposed to a failure. So you started, I think a couple of things, one, the point about evacuation was mentioned earlier, and, two, a consideration that why we are choosing six centimeters squared.

What is example of C-SGTR before core damage? I tried to give one example, blow down of one steam generator, rapid and complete blow down, delta pressure across tube increases almost by a factor of two, affected steam generator is dried out, operator fails to isolate the affected steam generator, primary remains high pressure and high pressure system continues to inject.

One or more tubes may rupture if they have deep flaws. Again, the condition as you will see, the calculator tells us don't worry about the small flaws, you need to have a really deep flaw for this to happen.

Operator might have turned off or stopped HPI. Now in the accident sequence not that he has a small LOCA to a steam generator tube rupture, he has to reestablish HPI and if he doesn't do that, for example, as an example if operator fails to do that then now you have a LOCA going outside to an open SRV, core uncovers, you get a fast core damage, and you are going to contribute to both CDF and LERF.

So, yes, this type of scenario you can find in the PRA and we kind of tried to bound it when we did the DBA or before core damage scenario.

CHAIR REMPE: Okay. So I think this is the time, if you don't mind before you go to the next slide, Mike, if you're out there and --

MEMBER CORRADINI: I'm out here. What's up?

CHAIR REMPE: Do you want to give like -- Because I know you're going to be leaving soon to catch your plane, did you want to break in and provide some comments that we normally go around and get from people who attend?

MEMBER CORRADINI: Well I don't want to interrupt. I can stay on for a while longer and maybe there's a more natural breakpoint in about 20 minutes.

CHAIR REMPE: Why don't you just do it now?

(Simultaneous speaking)

MEMBER CORRADINI: Why don't we all just let John go after him again?

CHAIR REMPE: Do you have any insights? Actually, I would assume with respect to the thermal-hydraulics and the materials we discussed earlier today that you wanted to convey --

MEMBER CORRADINI: Well I think, I guess my general comment is I know the Staff has done a lot of work over many years trying to advance this.

I think what at least the current NUREG and, the draft NUREG, excuse me, as it sits is that it still needs some additional -- I'm sorry, I'm getting a lot of background noise, is that me?

(Simultaneous speaking)

MEMBER CORRADINI: Is that me, I apologize, I'm on a handset.

CHAIR REMPE: I don't see anybody moving any papers here.

MEMBER CORRADINI: Okay, all right. But I guess my only point was the draft NUREG still needs to be further improved in terms of clarity and that's probably my overall comment.

There was a number of questions we had relative to the thermal-hydraulics and I think as we asked the questions Staff did a very good job of trying to explain where things were coming from, it just was hard to pull that out from the NUREG itself.

The second general comment, in terms of understanding, if this is going to be the last report in terms of the generic C-SGTR work and the next application would be the Level 3 PRA for Vogtle, it seems to me that there's got to be a story to be told at the beginning of this to see how it can be utilized and that's kind of missing. So I'll stop there.

I think it's appropriate to make those changes prior to issuing it for draft for outside comments, but I don't want to push the issue, I just think staff would probably want to have a cleaner version before they go out to have additional comments from industry and stakeholders. That's it.

CHAIR REMPE: Okay, thank you. And, again, I apologize for interrupting, but I appreciate your tolerating it.

DR. AZARM: No problem.

CHAIR REMPE: Go ahead.

DR. AZARM: Okay. Now what is the steps in this simplified PRA analysis are now just focusing on the severe accident part. Later on I'm going to come back and do this with the so-called DBA part.

We have kind of a 7-step process. First we select accident sequences then we try to determine the TH characteristics of those sequences, or at least bound them, consider a single flaw and get some idea of that what are the size of flaw that can fail in the steam generator tube, where should be our focus, and you will see why we are doing that, because if you try to run the calculator with many samples of flaws it's going to take a long, long time so you better have some idea that where is the region that that is done in a contributor.

Depending on what you get from the screening analysis of flaw you generate samples of flaw that probabilistically can support your analysis, you perform case runs and post processing and then you estimate your containment bypass probability and then you look at your timing and calculate your LERF.

Now I'm going to go through each of these steps in an overview fashion. Okay, focusing on C-SGTR for severe accident, and the first is that we need to identify the core damage sequences that are identified with high primary pressure, dry secondary side.

As you saw, at least for the severe accident, they are considering most of term to be low secondary pressure. If Level 2 PRA are available, and we had a case that they were available in one of our pilots, we can basically look at those two questions because these are part of PDF identification for Level 2 analyses and get the frequency of the sequences of interest.

If Level 2 PRA is not available then we have to go through Level 1 PRA, examine the dominant sequences for both internal and external event and try to identify that core damage sequence is that the primary was high pressure and a steam generator was dry, more so if the core damage after feed and bleed comes though that category, most of the station blackout scenarios come under category.

You have to make sure to look at external event as internal event and also you have to think, because remember if you talk about SBO these are multi-unit issue, so if you have multiple unit on one side and you have seismic event, most probably both are going to experience SBO, most probably, or the good probability both are going to go through the cycle scenarios and both are going to get a steam generator tube rupture, so somehow you have to keep track of multi-unit effect.

I'm not going to go through detail, I'm basically showing you we went through the two prior, the Zion and Calvert Cliffs and we looked at internal events, seismic, fire, flood, and I think in the case of Calvert Cliffs they had high wind, tracked a group of the sequences we found under short term SBO and long term SBO.

It doesn't mean they were exactly short term, it's more long term, but to group them on those two categories, and basically what we got, we typically got 2 times ten to the minus 6, this is from IPEEE's and their updates.

For short term SBO and about 2 times 10 to the minus 5 for long term SBO, and it's consistent for both of them. Now it was a little bit surprising to us because kind of Calvert Cliffs was addressing some of the common initiator issues in multi-unit core damage and when you look at that they had fires, seismic that dominated short-term SBO and long-term SBO affecting both units.

So, again, we captured that, I am showing that, but really at the end we looked at one unit or more, so we didn't look at the 2-unit risk. So that led to step one in PRA to find a frequency and a class of sequences that you want to start doing your calculation with and that comes from existing PRAS, at least for the severe accident side. For the DBA side, no, it's a different story.

Now the next thing is that the TH characteristics for these sequences and, you know, we can say we want many other sequences, but what was available basically is a base-case for short-term SBO and for Westinghouse or Zion based on a RELAP/SCDAP.

I tried to basically, SBO with early failure of TDAFW. The base case assumes 21 gpm RCP leakage. Also, the base-case assumes 0.5 square inch hole in the secondary side, which is sufficient to depressurize you before any, a steam generator tube rupture occurs and no recovery of AC power.

Long-term SBO is the same thing. If the turbine-driven AFW fails after battery failure, they have assumed that battery depletion is at four hours and we know that sometimes they can drag it to six hours or eight hours by load shedding and other stuff.

And as SCDAP ruled out there was a bunch of sensitivity analyses that was performed, they did sensitivity analysis based on RCP seal leakage rate. They went all the way to 450 gpm per pump.

That was important because in one case that they put 450 gpm RCP leakage it cleared the seal. So according to the SCDAP/RELAP analyses the seal clearing should happen somewhere between 350 gpm leakage of RCP 450.

It doesn't happen in any other condition. They put a little bit larger hole in the secondary side, including open, the steam generator SRV and ARV, they take on that some of the threshold of leakage area and they did rapid early depressurization to show that the hot leg did not fail.

You have seen this result this morning. Again, this graph shows information we used for the PRA and calculator software.

We used hot leg temperatures, shown in red. Average hot tube temperature, shown in green. Primary P, secondary P, and actually we should have also the hottest fuel, but I don't think I'm showing it here.

CHAIR REMPE: The blue line, the hottest tube, it's on there.

MALE PARTICIPANT: The blue.

DR. AZARM: Oh, it's the blue, okay. All right, I am showing that. The important thing you know, and we talk about that several times, after 1500 minutes, or whatever, 1600 minutes, 160 minutes, what am I talking about?

Really nothing is happening. It's basically your steam generator going dry 2-1/2 hours, 2 hours the steam generator goes dry, et cetera. And then you'll start using water until you get to core uncovery and once the onset of core damage happens you will see in a few minutes you will start getting zircaloy oxidation and everything shoots up.

So if you look at here, somewhere between 180 to 214 minutes everything should be over. So, again, this is, and usually nothing happens frankly after you are going to the temperatures above 800 degrees C.

So actually the situation is worse than

that. It's a much shorter time, usually within ten minutes everything is over. The other thing that you have to notice, just notice in this region the difference between the temperature of hot leg and the temperature of the hottest fuel or average hot tube.

There is a big, about 400 degrees C difference between them. So it says hot leg is really, really hot.

CHAIR REMPE: That gets more cleanup there, that's why --

(Simultaneous speaking)

CHAIR REMPE: Actually, that brings up another question, and maybe Mike needs to answer it, but I guess there is a MELCOR stainless steel oxidation model, and is it turned on so you'd even see it?

I'm guessing it doesn't consider any blocking because of hydrogen in the system, but has this been considered at all?

DR. SALAY: To my knowledge it wasn't considered, the stainless steel oxidation. I don't think the model is activated, but I'm not sure. I'd have to check for that.

CHAIR REMPE: Okay. Again, this might be another example of something that should be clarified what the document did or didn't do and maybe it can be ruled out because you can say yes, there's hydrogen in the hot leg that didn't go to the top of the vessel or whatever, but it would be nice to know if that's been looked at.

DR. SALAY: Right.

CHAIR REMPE: Thanks.

DR. AZARM: So that was the SCDAP/RELAP and this is the MELCOR one. No, I'm sorry. I'm sorry, this is just I draw it different. This is the long term station blackout. It's still a SCDAP/RELAP.

Here we will see the activities or the interesting part happens after 800 minutes, which is like 12 hours because you had turbine-driven AFW running for four hours and another two hours for the steam generator to dry out.

And most of the time we have done quite a bit cool down by that time, so that's why it delayed so much. But you will see, notice the same behavior if you look at your hottest fuel and average hot tube compared to hot leg, 800 degrees it starts shooting up and goes up very fast.

Next I am going to talk about MELCOR and Calvert Cliffs and just see the difference. They basically run a two base-case of short-term SBO and long term SBO.

They had I believe two sensitivity analyses, one was what I said earlier, they assume all this always is stuck open at times zero, that was one sensitivity analysis that basically gives us almost probability of one of C-SGTR.

And they did some analysis with the RCS component, fail or not fail, a suppression trip rupture, et cetera.

MEMBER STETKAR: Are you going to address, I don't know when or if, something I brought up this morning about uncertainties in these various temperatures and times, and I don't -- Are you going to address that?

DR. AZARM: We did address as a part of calculator the uncertainties almost for everything except the TH analysis.

MEMBER BALLINGER: But what you said was you identified the uncertainties. You didn't say "I addressed them."

MEMBER STETKAR: If you go back to -- Let me see if I can ask it in the context of the picture, if you go back to three or four slides. Yes, that's good.

This leads me to believe that the hot leg always heats up faster and more dramatically than the hottest steam generator tube, therefore, reinforcing the notion that the hot leg is going to fail first, but maybe not by much in time.

So the question is, what uncertainties are there in that timing such that if I have two distributions in time and I look at the inner section of those distributions where the intersection measures the likelihood that the tube heats up faster than the hot leg, have you looked at that?

DR. AZARM: Let me see if I can --

MEMBER STETKAR: If I could put it into words that way.

DR. AZARM: -- give you that picture. This is showing the survival probability. It's kind of like the two PDF but it's showing it in terms of cumulative.

When you look at Westinghouse there might be something to section in the tails that you would integrate over the time and then, you know, there are some other stuff.

So it's very low. It's not -- Even with the uncertainties we have included that does not include the TH uncertainty. With the uncertainties we have included we don't see that in Westinghouse.

Look at comparatively for even good case of CE Plant, you will see that overlap, and that's what in post processing you use for calculating the C-SGTR, but we'll get to that.

DR. SALAY: Yes. I also want to mention, although I wasn't involved in the steam generator action plan they looked at -- This is Mike Salay, the NRC -- that they tried to look at many different things in TH that may affect what fails first.

And from my understanding based on the flow patterns that you get the timing of when you start to heat up may vary a lot, but they seem to get a pretty consistent failure of the hot leg first unless you had two, because it's the same temperature as when you start rising the temperature they both rise at about the same time and it really depends on how the flows go around inside.

MEMBER STETKAR: Okay.

DR. AZARM: I think you'll see that in this picture, this is, just keep what you saw in Westinghouse in mind and now look at the CE results for MELCOR.

I tried to kind of keep the same colors, but look at your hot leg and look at your hottest tube.

MEMBER STETKAR: Yes.

DR. AZARM: They're basically tracking each other.

MEMBER STETKAR: No, and I'm not arguing Westinghouse versus Calvert versus CE. I'm asking about within the context, for example, it's clear for the CE that any uncertainty would certainly affect the conclusions.

What I'm asking about is what's the extent of the uncertainty in the Westinghouse where one example you showed seemed to indicate that there might not be, that you'd have to have very large uncertainties to get much of an intersection.

On the other hand, if I go back to that slide, you know, Page 106, on my handout here, four or five before that, it doesn't to me look -- The one before that, even this one is the same.

I'm looking at hundreds of minutes here and on that scale, you know, I'm within ten minutes or less, you know, on those slopes. They seem pretty close to me.

CHAIR REMPE: And this was done with SCDAP, which has certain melt progression models, and if, again, so you have certain assumptions on how the core heats up in blocks or whatever, but if you went to the CE Plant that was done with MELCOR, which has assumptions, now they've been benchmarked sort of against each other but I don't know.

Are there going to be some differences in the heat up because it's a MELCOR analysis versus a SCDAP core heat up analysis? And so if you've done the Westinghouse with MELCOR would you get the same method uncertainties?

MEMBER STETKAR: The only reason I bring this up is to me there's a lot of analogies in the way we do seismic stuff.

If you just look at point estimate fragility, mean fragility and mean hazard you might conclude that there is low likelihood of something failing.

On the other hand, if you looked at the uncertainties the whole analysis is out in the uncertainties if you don't treat the uncertainties correctly or reasonably.

DR. SALAY: I'll admit you can't treat them correctly.

MEMBER STETKAR: Reasonably you're going to have optimistic results.

MEMBER POWERS: I fully --

MEMBER STETKAR: So that's the only I'm asking about here is how carefully did you think about the uncertainties in these different times in the models to support those times.

MEMBER POWERS: I wonder -- I fully support your concern about the uncertainties, but I would like, you know, if what you see in this block, the hot leg, the surge line, and the tubes coming up to temperature fairly shortly, but that said, the temperature is not in the condition it is when it gets up to a temperature where a creep rupture becomes very broad, and in a case the hot leg would cross that temperature during the rise, whereas you don't cross these temperatures in the other locations to substantially in the clean parts of the legs.

And so the sharp rise might fool you that it's really not the 10-minute gap where everything's coming up, it's where the hot leg comes up versus the 60 minutes later when the --

I mean the way the creep rupture works in these things is slow, slow, slow, bing, it just goes and so don't let that fool you. Nevertheless, your point is correct.

But I think that when you go through the details of analysis they quickly come to the conclusion that it's how much mixing we get in the lower plenum that really controls all of these temperatures in the tubes whereas the hot leg is really controlled by what goes on in the core.

MEMBER STETKAR: Okay.

DR. AZARM: And just one point of correcting the opinion, I am not going to say that the uncertainty evaluation done is 100 percent complete, but when I see the Westinghouse shift is so large I think even with the incompleteness of our uncertainties we have I certainly do not change my position in the Westinghouse --

MEMBER STETKAR: And I'm not advocating trying to, you know, do a perfect uncertainty analysis here. I'm trying to understand how broad those uncertainties might be and where there might be areas of overlap that haven't been considered.

And the uncertainties might be broad, but as long as there's enough margin I'm not worried about a minuscule part of overlap. That's all, so I'm trying to sort of probe to see how much of that thought had been put into the analyses or how much you thought about it outside of the context of what's written in the report anywhere.

MEMBER POWERS: Well I think I do share with Dr. Rempe the concern and within the computer codes themselves there are probably capable of the performing. Assumptions have been made that could affect the calculations, and certainly the entropy, the gas coming into the piping system.

They are assumptions that are made and developed into codes so operating the codes that were made were not, nobody ever wrote it down exactly what they were thinking.

With time they are carried on by tradition and things like that --

MEMBER STETKAR: Well and they may have been made --

MEMBER POWERS: You don't know what they are.

MEMBER STETKAR: You don't know where they are and they may have been made for other purposes.

MEMBER POWERS: Oh, yes, I would agree.

MEMBER STETKAR: Theoretically a conservative treatment of some other issues.

MEMBER POWERS: But what happens is the codes, things get developed and sequences don't and the issue of induced steam generator was not on anybody's mind when the code was written.

MEMBER STETKAR: Right, yes.

MEMBER POWERS: And some of these things, you know, a comprehensive uncertainty analysis to identify is a formidable job.

CHAIR REMPE: It is, but if there is a

difference that is just because of what MELCOR would predict versus SCDAP and we're saying oh, the CE Plant's worst, is it because, is there something that's different in the way those codes are predicting the scenario that would show --

MALE PARTICIPANT: No, I think --

MEMBER POWERS: I mean I think we trace it, like I say, back to the mixing in the lower plenums would get you into trouble.

CHAIR REMPE: Well then the CFD analysis would justify that, but I'm just wondering about time because I don't have --

MEMBER POWERS: No, I don't think, I mean you just don't see how you alter that conclusion by changing the accident analysis very much because it's really an entity of dilution that's affecting --

CHAIR REMPE: But if it were done -- Okay, but if the BWR with MAAP and MELCOR we know that it does delay in how the melt progression occurs.

MEMBER POWERS: That one would shift everybody, okay?

CHAIR REMPE: Yes.

MEMBER POWERS: To shift CE versus Westinghouse you got to have something very imaginative to even see, but it's hard to see how you get into that trouble if you ask me. I mean that's --

MEMBER STETKAR: In the code world.

MEMBER POWERS: Yes.

CHAIR REMPE: Okay.

DR. AZARM: Also there's a good news. I always like to look at the good news --

MEMBER STETKAR: Why?

DR. AZARM: The good thing is that these are coming from a code. So the hot leg temperature is going to drive your hottest tube temperature, so all of these curves are correlated. We know they are.

So even if the curves are shift, they're relative two times is still, you know, there are, you are right, the big margin between the two --

MEMBER STETKAR: That hurt. Yes, it's still, you know, the shift does this.

DR. AZARM: Yes.

MEMBER STETKAR: What I'm worried about is how far apart these distributions are.

DR. AZARM: Yes. But, you know, the margin I think is the main thought process we have regarding Westinghouse.

MEMBER STETKAR: Okay.

DR. AZARM: Okay. I am now in the step three of PRA that we talked about. We basically tried to see if I have different sizes of flaw in my steam generator I can take it one flaw at a time, what is the probability of C-SGTR, what is the probability of flaw tube fails before hot leg.

When I look at Westinghouse and I look at one flaw at a time and it looks like if I use the average hot temperature I couldn't see anything under 50 percent.

Actually the first time I start calculating numbers, that 70, 75 percent deep flaw, and similarly in the hottest tube almost nothing interesting happens unless you have flaws that is 70 percent or more.

So it's obvious in Westinghouse the probability that you have one or two flaw bigger than 75 percent is going to drive the C-SGTR up. When we look at the CE Plant basically if you look at the hottest tube almost anything's going to --

(Laughter)

DR. AZARM: So, yes, you are hoping that you don't have a flaw where the temperature has gone this --

MEMBER POWERS: Probably didn't even do you good in the hottest tube if it's pristine.

(Simultaneous speaking)

DR. AZARM: Yes. Well even with pristine might fail, but at least with the pristine one I don't know what is leak area and I can justify the leak area might be small, but I can have a long shallow area and, you know, and the average hot tube, you know, it's you have to have greater than 50 percent in order to get some number.

So, again, just for the information if I have 20 hottest tube the chance that one of them is flawed, that is the way we, we are not accounting for pull moving around and affects more tubes, but if I am talking about 20 hottest tube and I have 4000, 5000 tubes and I have a hundred flaws the chance of one of them shows up in hot test is going to be a small.

So that's why we don't get one like what the TH got, but it basically tells you I need to focus on different sizes of flaws when I do the Westinghouse and when I do for CE Plant, even if I take the average flaw bins I should get more or less a good calculation of the mean and perhaps I do a few sensitivity analysis to get some idea for the uncertainties with this one.

I think that's what the next slide is saying. So for the representative of Westinghouse Plants we used limited samples of large flaws, estimated frequencies that there could be one, two, or more large flaws within a cycle.

For the representative CE Plants we basically had an expected average number of flaws within each flaw bins and we used that as a sample. This is shown in the next slide.

The next slide basically is telling you for the Westinghouse first of all, so what is the probability that if I'm on Cycle 15 and going to Cycle 16 what is the probability that I have a flaw between 16 to 17 percent depth, it's 10 times 10 to the minus 3, and if I have X number of flaws the probability that I see one is 0.4.

And, you know, the same thing for other bins. So it's lower probability that I see a single flaw. For the CE Plant I actually create a bin of 10 percent, 20 percent, 30 percent, so on in depth and the length and I can calculate from my flaw distributions how many flaws do I expect to see in the CE steam generators.

And, you know, yes, there is probability that I have a big flaws but I didn't try to put 0.1, 0.01 there because I know that the smaller flaws are going to dominate it anyway.

And this is a graph that I showed earlier. So when you do that and you run the Westinghouse and

CE Plant for one of the cases you basically see totally different behavior of the probability plot of RCS survival in a C-SGTR.

For Westinghouse versus CE, in Westinghouse you have this larger margin that's separating them saying that yes, most probably the RCS is going to fail, the hot leg going to fail, where in CE plants you say hey, I have to do calculations.

The next slide basically, I'm not going to even go through it, when we say you have bunch of time steps and each time step you got accumulative probability, now in each time step you have to calculate incremental probability of containment bypass and you have to compute them and integrate them.

So all I'm saying is that in all the routine that does this post-processing integration, et cetera. So after we do all of those steps what do we get?

Let's first look at the CE Plant is 692 and then in parentheses I have put the sensitivity case when might assume the SRVs and the steam generators are stuck open from beginning.

So for the base case I get C-SGTR of containment for bypass probability of 0.22. For the case that SRV is open I get 0.99, I get one. This is just because of we have capability to calculate.

For Westinghouse I get about 1.3 times 10 to the minus 2 considering all these large flaws happening with certain probability if it is made out of Inconel 600. If it's made out of Inconel 690 I get a little bit lower.

We did not do the sensitivity case with no --

MEMBER STETKAR: With the stuck open --

DR. AZARM: Yes. I do believe we have some runs but there was reason that we couldn't use the data.

(Simultaneous speaking)

MEMBER STETKAR: That would certainly be -- You know, from my perspective I'd be interested in that one to see how it affects the --

DR. AZARM: Yes. I also want to mention for Westinghouse we also included the loop circular and probability, because as I said earlier the Westinghouse, when the RCP seal leakage hit 450 it cleared the seal.

So we assume for RCP seal leakage between 350 and 450. At 450 probability one, at 350 probability 0.1, and we included that. So if I add RCP seal leakage in this scenario --

MEMBER STETKAR: And the tubes fail --

DR. AZARM: Of course. When the seal clears --

MEMBER STETKAR: So, for example, in the context of another project that shall remain unnamed that Kevin is interested in where they explicitly track those different seal leakage rates they would have much different consequential steam generator tube rupture probabilities, right?

DR. AZARM: Correct, you should.

MEMBER STETKAR: But for the larger end of the seal.

DR. AZARM: But basically we assume when the loops are clear that we basically assume the hot leg is circulating and not that much mixing or cooling, so basically tubes saw what hot leg's seen.

MEMBER STETKAR: Yes.

DR. AZARM: And they fail, yes.

MEMBER BLEY: Let me take you back to your last slide, you don't have to go back to it. Your bullet talks about you can do the calculations using a Fortran routine or an Excel worksheet. What did you do?

DR. AZARM: Okay.

MEMBER BLEY: Let me tell you where I am headed. I don't, you know, if I write a Fortran routine

I know exactly where, usually, where the calculations are going.

If I use Excel, Excel has some things built into it that I might not fully understand and the order it does things might generate funniness. I've seen that somewhere. I haven't played with that enough to know.

If you used Excel how did you make sure it was doing all the calculations right for you?

DR. AZARM: Let me tell you the story behind it because it's much more involved than what we saw. The Westinghouse was relatively very easy. There was one run.

We didn't differentiate between loop with pressurizer and loop without pressurizer. You put all your flaws in one steam generator and you have the hot legs and basically what we did in Excel, since we have one run, one result, we did even the integration numerically. We didn't use anything of the imbedded Excel function, so it was easy to perform.

MEMBER BLEY: Okay.

DR. AZARM: When we got to CE Plant times to mark, it gave us different TH for Loop A than Loop B, so instead of we do two runs now we do four runs.

We have to integrate Loop A and Loop B,

first come up with those results then come up with it over time, it was a nightmare to do it with Excel. I have a bunch of routine, Fortran routine for convolutions and we basically modified that and used it for Fortran.

MEMBER BLEY: You stretched out, okay. That makes me more comfortable because I know when you get those complicated situations --

(Simultaneous speaking)

DR. AZARM: Yes, it was complicated. I couldn't do it with Excel anymore.

MEMBER BLEY: Okay.

DR. AZARM: Okay. Now we are going to get to Level 2. This is the most simplified analysis we did and we will tell you why.

MEMBER BANERJEE: Just one question for clarification.

DR. AZARM: Yes?

MEMBER BANERJEE: How sensitive are the results to loop seal clearing, the what you assume for that?

DR. AZARM: Basically the one case that we run we basically said if loop seal clears there's no constant current flow and the temperature that hot leg sees is more or less the same temperature the tube sees.

So we did a sensitivity analysis and the tube failed when it sees the same high temperature as hot leg.

MEMBER BANERJEE: Yes, so that also given, is going to fail, yes.

DR. AZARM: If loop seal happens, the good thing about the Westinghouse is that, or Westinghouse analysis it says loop seal will not happen unless you have a very large RCP leakage, which is by itself is a probability of Level 1.

So that's why instead of we get 0.1 we got 0.013 when we included loop seal. It was a small contribution. Again, because of to get loop seal clear based on RELAP/SCDAP analysis you have to have a very large RCP seal leakage on all four pumps.

MEMBER BANERJEE: Yes, I can see that. Okay.

DR. AZARM: Okay. When we started doing Level 2 analysis, actually at the beginning we had a very good intention. We did understand if I do a complete Level 2 analysis I can differentiate between large and the small releases and I can account for all the actions in SAMG and hopefully we can get timings.

And, again, this is four years ago when we started this, we were very idealistic, so we can do a good Level 2 analysis. We started working on developing the event trees.

The first thing that happens is that these are post-core damage sequence. So as soon as I am post core damage and I am saying what is the probability of PORV opens, it's not qualified for that environment.

What is the probability of SRV to operate? Is SRV going to be jammed or is it going to be open? I don't know. What are the HRAs associated with SAMG? This is now core damage for us, C-SGTR.

And it always easy to say I'm going to depressurize my secondary and fill it up with fire water, but am I going to do it under this condition with the high radiation on the secondary side?

So there is a slew of problems that we couldn't handle, not in this small scope project. So it was NRC and our agreement to go to a very simplified conversion analysis.

MEMBER BLEY: You know, those things you talked about, bounding might not be so bounding given some of those issues.

DR. AZARM: Yes. Our bounding you will see it is --

(Simultaneous speaking)

DR. AZARM: I'm going to go through it.

You basically give credit to nothing.

MEMBER BLEY: Okay.

DR. AZARM: This is just to show you that we did go through SAMG, we understand SAMG, we tried to model them, we understand , or it was we are guiding to do and, you know, describing in the secondary by filling up, but we didn't model them.

When we look at SAMG a couple of issues are important in this slide. So we did look at SAMG, we tried to understand it, but we decided not to model them, but we saw two things that might be important.

One is that sometimes if they recover AC after core damage they try to bump RCPs and now they are going to clear the seal and they are going to put us in the problem that we had before, failing the tubes.

However, it does require that your AC to be recovered and we are not modeling that. If your AC is recovered than you can fill up your steam generator, you can do many things that, you know.

But, yes, bumping RCP may not, may clear your seal, but you have so many ways to help you to, so. The other thing was injection of cold water into a dry steam generator as a part of SAMG.

Again, we didn't verify it just to be aware that they have this guidance document. They are aware of the issue and supposedly, you know, they do a slow injection, whatever, so they have addressed it. We did not.

Okay. Now I'm going to get to this simplified LERF analysis. Basically it's a six factor formula. We know the frequency or the probability that we enter the accident condition.

We know from the calculator and other stuff I described how to calculate the probability of C-SGTR. So those two we know how to do. Well actually it's a five factor formula.

There's three other factors that we don't know about. One is the conditional probability that RCS is not depressurized. The other one is failure probability of SAMGs, and the last one is evacuation.

We basically do not give credit to neither SAMGs or depressurization post-core damage, so we are very bounding. So if you enter this accident scenario, severe accident scenario, and you have got the C-SGTR, the only thing else we care is evacuation and that's why we call it bounding. Nothing is credited.

And evacuation I believe we used a timeline that comes from SOARCA, I think it's ten hours or something like that for 95 percent to be evacuated. So if the short-term SBO becomes a LERF or the long-term

SBO, most of them, not become a LERF.

And this basically shows you those factors, you know, that the C-SGTR due to the single tube break. If loop seal clears we assume C-SGTR is conditional one, and, you know, the other factors is one. The same thing for the CE Plant.

This slide basically shows the summary results that we have got for changing in LERF. It is basically showing you CE, this half of Calvert Cliffs is higher than Westinghouse.

When you look at all hazard model for internal and external event we get for Calvert Cliffs 5.7E to the minus 7. Remember Calvert Cliffs has two turbine-driven AFW.

So really for other CE Plant might be a factor of ten higher, okay. So that's why we are saying CE Plants is most susceptible. Now look at the all-hazard model for Westinghouse, we get about 2 times ten to the minus 8 and Zion had only one turbine drive-AFW and they were kind of generic.

So we think that CE Plant is one to two orders of the magnitudes after LERF than Westinghouse Plants.

MEMBER BLEY: And that's kind of compounded because it's both in the fractional approach to LERF as well as the case feeding it.

DR. AZARM: Yes. For Calvert Cliffs.

MEMBER BLEY: Yes.

DR. AZARM: Now we are going back, so I'm done with severe accidents. I have fewer slides on the pressure-induced just to tell you a short activity we did and what results we got.

Pressure induced is basically we look at all the scenarios that we think the delta P across the two is at least twice what is in nominal operation. We develop scenario frequency from existing PRAs in some modifications we do.

We basically used a very simplified bounding pressure-temperature curves, okay. For example, I think for ATWS, because we didn't have it for ATWS, we used a pressure of 3200 and nominal temperature, et cetera.

We had some results for the main steam line breaks or we idealized that. So we kind of generated the TH we want but in a very established manner. This doesn't come for MELCOR or RELAP.

We again tried to look at the leak area that at least is one or more tube, but frankly what we look at is six centimeters squared. I shouldn't say one or more tube.

We modified the existing PRA accident progression to reflect that everything is more complex when you have this DBA accident and you have a consequential steam generator tube rupture on top of it.

It's bad enough to have ATWS, if you get the C-SGTR you are gone, okay.

MEMBER BLEY: Let me ask you something because I'm a little, I'm remembering some work done a long time ago looking at ATWS and to get to pressures twice normal, operating pressure, there's probably other things that are going to blow out that will limit the pressure long before you get to that point, reactor coolant pump seals, a whole variety of things, so I'm not sure what this tells us.

DR. AZARM: Okay. An ATWS was a specific and, again --

MEMBER BLEY: Well I --

DR. AZARM: Yes.

MEMBER BLEY: No, no, I'm not --

(Simultaneous speaking)

MEMBER BLEY: Okay. During looking at ATWS's when pressure was going up people looked to see what would break first.

DR. AZARM: Yes.

MEMBER BLEY: And it sure wasn't the loops, but it was things that would keep you from continuing to pressurize up.

So this kind of sets a bound that says if you get there, I don't know whether tubes would break or not, probably not at twice nominal pressure, but I just don't know what the point of this is because I didn't think you can get there.

DR. AZARM: Let's perhaps let's look at this.

MEMBER BLEY: Okay.

DR. AZARM: And going back to your question about ATWS, first of all it's not a big contributor.

MEMBER BLEY: I didn't raise a question about ATWS. I said the idea of --

(Simultaneous speaking)

MEMBER BLEY: Okay. I just was referencing --

(Simultaneous speaking)

DR. AZARM: And just we go back to your comment. So we looked at ATWS and we did the nominal that if the ATWS hits the 3000 psi ASME limit we are going to rupture, but we don't know that, are we going to break?

So we had to style it as we look at a steam line break inside containment and outside containment. Now that gives you almost twice because the secondary side depressurized, primary stays at HPI, okay.

We look at established opening of a steam generator relief valve. I know we looked at some PRAs, we even looked at it due to fire and non-fire causes.

We looked at some of the plant have a very high pressure pump for feed and bleed --

MEMBER BLEY: I'm getting confused again. You know, when you say you depressurize the secondary side, well, that then gives you 2000 instead of 1000 pounds differential, but the tubes aren't designed for 1000 pounds, they're designed for full system pressure and they're hydro'd to that line, in fact something above that.

DR. AZARM: Yes. The tubes, if they are pristine, can take easily 5000 psi. We are putting a probability of flaw of 75 percent in this stuff.

MEMBER BLEY: Okay.

DR. AZARM: So we have still our calculator calculates flaws that are big enough that can fail during --

MEMBER BLEY: At these pressures.

DR. AZARM: At these pressures. And that

is the probability that I see a flaw of that big size. So these numbers are coming, not only taking the existing PRA, modifying it, changing HRAs, but also on top of that multiplying it with the probability that I have such a large flaw that it's going to rupture.

Again, it's done just to get a feeling about how, about how good these things are, and the very conservative bounding calculation which we explained in the document, and I don't know what the details right now, we got about 2 times ten to the minus 7 and we think it's below that.

And we did the same thing for CE and for CE was a little bit higher and I think indicative to what Dr. Stetkar was saying because they have twisting generators, we got 4 times 10 to the minus 7, but we have addressed, we tried to address that issue and we have documented each of these accident sequences.

MEMBER STETKAR: Now your conditional probabilities of core damage given the initiating event and a consequential tube rupture runs through that little simplified event tree that's got a whole bunch of operator actions in it that you say things, oh, this is kind of a 10 to the minus 3 and this is kind of a 10 to the minus 2, and without really doing a real human reliability analysis.

I will point you to an event that occurred at a real plant where they had an uncontrolled cool down, not due to a stuck open valve, but due to heater drains that blew through because of the power failure, and the operators never figured out what was going on.

The only thing is a guy bumped a DC circuit breaker that happened to interrupt power and the interruption of power closed the MSIVs and stopped the cool down miraculously.

The operators never figured out what was going on. Now they didn't have a tube rupture to contend with in this thing, they just never figured out what was causing the cool down in the beginning.

DR. AZARM: Oh, yes. I --

MEMBER STETKAR: So the simple models about saying well the operators can equalize pressure across the ruptured steam generator the same as if this is a steam generator tube rupture initiating event kind of ignore --

MEMBER BLEY: But that was partially because they thought that was normal.

MEMBER STETKAR: Yes. Well, yes. That's a different part of the story, that's right. So my point is that a lot of these numbers without the context of a fully integrated PRA treatment are just that, they're numbers, and that's okay as long as you're not trying to draw global conclusions about how important or how unimportant this issue is to overall public health and safety.

I honestly don't know how important it is and I'm willing to say that I don't know how important it is because I haven't seen anybody really investigate it, which is okay, but I know this isn't a thorough investigation.

DR. AZARM: I agree it's not.

MEMBER STETKAR: So that's another concern is about what message comes across in this NUREG in terms of is this issue, not necessarily from the research of more thermal-hydraulic analysis or more material science because my sense, knowing nothing about those topics, is that it's pretty mature in those areas. Again, knowing nothing about those topics.

But in terms of drawing overall conclusions about, you know, are we ready to check off the box that we look at everything and can conclude that it's not important, that is not as clear to me.

DR. AZARM: I think I do agree. I think --

MEMBER STETKAR: And essentially, you know, what I'm challenging in this particular slide is that center column there that says the conditional probability of core damage given the initiating event frequency, I have reasonable confidence in all because I have no other reason to doubt it by your 4 times 10 to the minus 3 conditional tube rupture probability.

I'm just not sure that I have any confidence at all about 3.2E to the minus 2 conditional core damage probability given those conditions, because that relies, my recollection, heavily on a whole bunch of operator actions.

DR. AZARM: If I may, and I hear you, I do agree we should not make a conclusion, but perhaps I should do a better job.

The first column just gives you the frequency of initiating events. The second column, which is added because of this work, it says given what we know, and, yes, there are, better analysis can be done, this is the probability that you get C-SGTR.

MEMBER STETKAR: Right.

DR. AZARM: So what is the probability of conditional core damage given IE and C-SGTR.

MEMBER STETKAR: Right. There's a model in there for that now.

DR. AZARM: And that's the one that you are questioning and that's the one --

MEMBER STETKAR: Exactly.

DR. AZARM: I do -- Again, we tried to be bounding but I can't define every details of it. Yes, we shouldn't have made the conclusion. And the left column, of course, it's all one or zero.

MEMBER STETKAR: Well I mean in this case you've assigned it to LERF, so I, you know, I can't believe that it can be worse than that.

DR. AZARM: Yes. No, I agree with you. I do agree. The first conclusion talks about we do believe at least an order of magnitude is higher for CE than it is for Westinghouse when we talk about the severe accident caused the C-SGTR.

We say PRAs indicates delta LERF, delta CDF. For the C-SGTR for this pressure-induced accident, the one we just said, we think it's less than 5E to the minus 7 with some grain of salt.

We don't say for this not important. To me, when you have a criteria of 1E to the minus 6, 5e to the minus 7, doesn't say it's insignificant, it doesn't say it is dominating either, but, yes, I agree with your point.

I think the next bullet it comes from the summary or abstract. Basically it's emphasizing that, you know, there are ways, you know, like adding additional turbine-driven AFW or diesel generator pump can help you.

It talks to concisely and clearly says that some work was done in loop seal clearing for Westinghouse Plant as I said, nothing is done for CE. So the Staff we tried to control, avoid large and deep steam generator tube.

At least now we know large and deep means 70 percent or more when we talk about Westinghouse. It is not 40 percent, so we have enough margin there that I think that's an important thing we are trying to say.

We talked about the inlet plenum and surge leg plenum, hot leg geometry, et cetera. The design features are an important contributor. Depressurization of reactor, intention of depressurization could help you here and also in other places. For the DBA accidents we feel, again, that the deep flaws is the one that is contributing.

One area that we just weren't for sure and we haven't talked about it, where does the six centimeters square coming from? Why do we assume this critical area of six centimeters squared?

There's a summary of reason for it. When we start a big drop we wanted it to be sufficient to pressurize the secondary side such that it forced the opening of the secondary relief valve.

So we wanted to have a leakage area that even the 1/2-inch hole it still can open up your secondary. We wanted to depressurize primary. We wanted it to provide sufficient release path that it can be considered from the source as a LERF. If you have a very small hole it may not be.

And we didn't it want it to be too large. If I got 22 failure I wanted if I still had the counter current flow, so we didn't want it to be that large. So we put all these things together.

I think initially we did some back of envelope calculations and we came up with the six centimeters squared and we also looked at the input from both RELAP, SCDAP and MELCOR.

Now I think the last item talks about increasing the battery life to facilitate longer operation of TDAFW and support SAMG operation and probability of equipment survivability post onset of core damage. I think that's a million dollar question.

If we want to do Level 2 I still don't know how we can do it if we don't know how a PORV, SRV, and others are going to operate.

MEMBER STETKAR: Now before we -- I'll stay on the horse that I've been riding for the last hour and a half.

But when I first started to think about this stuff, we thought about it 25 years ago and people concluded ah, induce your consequential tube ruptures, no big deal, and so we kind of dropped it, probably wrongly.

But I started thinking about it again saying well, gee, are our PRA models actually capturing all of the scenarios for which this phenomenon may present a vulnerability, I quickly came to the conclusion that no, we're not.

I've done some work at a plant several years ago that had a concern about pressurized thermal shot for their reactor vessel, not a U.S. plant, it will be unnamed, so we for that particular plant because of that concern looked pretty carefully at overcooling scenarios, but from that perspective.

So we started looking at stuck open secondary side valves, you know, how fast could you cool down, all that sort of stuff, but from that perspective and at the same time recognized that oh, gee, it's true, if the operators, if you have a faulted steam generator they're going to isolate that steam generator so that indeed there was -- The good news is that the overcooling scenarios for that particular plant we could justify it didn't have much of a chance of really harming the reactor vessel but indeed there were visible contributors to core damage because they put the operators in situation that was more confusing to them and, in fact, removed one of their steam generators as a cooling mechanism, but then never thought about it in the sense of getting the high-dry-low condition.

DR. AZARM: Right.

MEMBER STETKAR: And I have not seen PRAs really look at that. So I'm curious about why an insight from this whole study is that PRAs may not actually be looking for these scenarios.

They're looking for the -- You found them in places like your, you know, main steam line break or your stuck open secondary relief valve initiating event, you found them in station blackout with whatever timing you want to give on failure of your auxiliary feedwater flow.

But you didn't say are the PRAs actually identifying these scenarios. So I'm curious about why that's not a PRA insight or should it be?

MEMBER BLEY: Yes, see, to me --

(Simultaneous speaking)

MEMBER BLEY: I've kind of sat back and thought maybe people were waiting to see where the researcher was going before they started addressing that.

MEMBER STETKAR: Yes. Well, but I mean some of the feedback from this says well it's not a big deal contributor, it's certainly not a big deal contributor to core damage, and furthermore it's not even a big deal contributor to LERF, so what incentive do I have to go out and look for it.

But it's a self-fulfilling issue that if you've only looked at what you could see and people haven't looked for the other things you don't know how important those other things might be.

And, again, I don't have a sense, I honestly don't. If I had, you know, any example believe me I would've been screaming about it that look, you know, this study had been done and it's a 50 percent contribution to something or other, but I don't have that.

But I do know that the results that you're looking at are not complete in that sense. And that it's not a simple, it's not just a simple binning of things because the actual accident progression becomes, the event scenario, I won't even, I need to stay away from this design basis and core damage thing, the scenario progression becomes much more complicated because the operators now have to deal, if they follow the procedures and isolate the faulted steam generator that's good from their perspective, but it reduces their ability to cope perhaps with other things that are going on and their performance might not necessarily be all that well in an integrated sense. So I'm just curious. It's a comment.

DR. AZARM: I think your comment is well taken. I haven't seen also, or I haven't seen neither any PRA systematically look for all possible scenarios and the complications involved.

And what we did, you know, just trying to pick up the low hanging --

MEMBER STETKAR: I can't -- As I said I did, this was several years ago because of that one particular issue that we went and looked at, in particular looking for overcooling scenarios that we had not looked for before because typically we'd use models like these that said do we have enough steam relief --

DR. AZARM: Right.

MEMBER STETKAR: -- and if no, it's a bad day, as long as you have enough, you don't ask anymore about do I have too much and what happens if I do have too much.

After that one plant then I put it in models that I'd work on for other plants. Typically it wasn't a big --

OPERATOR: Please pardon the interruption. Your conference contains less than three participants at this time. If you would like to continue press star 1 now or the conference will be terminated.

MEMBER STETKAR: The interruption is pardoned. None of the plants that I worked on was at a, yes, I have to be careful. I'll just stop the discussion there.

DR. AZARM: Regarding your --

MEMBER STETKAR: It's we've identified some interesting scenarios by looking at it. Things that make life pretty complicated for the operators and can indeed have impacts on core damage.

I have never looked at consequential tube rupture. I've never built a model so I have no insight, you know.

OPERATOR: Please pardon the interruption. Your conference contains less than three participants at this time. If you would like to continue press star 1 now or the conference will be terminated.

MEMBER STETKAR: This is an indication that the line is not there.

CHAIR REMPE: Anything else?

MEMBER STETKAR: No.

CHAIR REMPE: Okay.

DR. AZARM: Thank you so much.

DR. COYNE: Can I get a quick clarification on it just to make sure that we understand this feedback?

DR. AZARM: Sure.

DR. COYNE: So my impression on how we identify and bin these scenarios is that if you had a loss of aux feedwater from whatever the cause you would count that as a dry scenario which our assumption would lead you to also put a low secondary pressure scenario then you could systematically look for higher pressure scenarios in the RCS.

So those scenarios seem more likely to be counted with the current PRAs we have.

OPERATOR: Please pardon the interruption Your conference contains less than three participants at this time. If you would --

MEMBER STETKAR: Yes, I'm assuming that it was stopped midstream that it's been intercepted. Let me just --

DR. COYNE: The one thing that causes me pause is the scenarios you brought up where I've had a faulted steam generator then I then have the operators isolate it and then get myself into a core damage scenario where I may not have the cut set that tells me I completely lost aux feedwater.

MEMBER STETKAR: Right.

DR. COYNE: So therefore I may have aux feedwater available to other steam generators but I have this steam generator potentially in a vulnerable state.

Is this an example of the majority of the scenarios you're thinking of?

MEMBER STETKAR: Kevin, I don't know.

DR. COYNE: Okay.

MEMBER STETKAR: I don't, you know, I don't know is the key because I have not thought about the consequential tube rupture.

I have to say I've never built a model that's got consequential tube rupture in it, so I'm not right now able to say well here are the class of scenarios that you wouldn't otherwise identify that are important for this.

But I know that the typical PRA models do not quantify the frequency of scenarios that actively isolate feedwater to a faulted steam generator such that that steam generator is sitting there open, big hole open to the outside world and dry waiting for whatever other combination of things finally get you into trouble.

I just, you know, that's all I can say. I haven't done, you know, I haven't done the study.

MEMBER SKILLMAN: Ali, you made a comment about the origin of the one square inch or the six centimeters squared. In the late '60s and early '70s all of the PWR designers, Babcock, Combustion, and Westinghouse, were designing for a 1-inch break hands off makeup system coverage.

And so the CDCS on the Westinghouse and the Combustion systems and the makeup and purification on the B&W Plant could ride through a 1-inch break and if the 1-inch break was in the steam generator that'll be this half, then the only thing you would see changing is your makeup tank level changing because your pressurizer level would hold because of your pressurizer level primary system control systems.

And so in time you begin to see a reduction in inventory. So that was a purposeful design and that became kind the knee in the curve for what's a real small break and what's a break that you could communicate to the NRC, we've had a leak but we can handle it.

It's handleable within the normally operating equipment.

DR. AZARM: Normal --

MEMBER SKILLMAN: But that was a number that all of us were branded with back in the late '60s and early '70s and we designed all the equipment for that.

As time went on it became clear that actually it wasn't a one square inch it was a one inch Schedule 160, which is about a 5/8 of an inch, just a little bit smaller.

But that's I believe where that number came from. It was a nominal 1-inch break.

DR. AZARM: Now I can tell you honestly how we calculated that number. It wasn't based on charging and perhaps we didn't document it. We basically tried to look at, again I am getting myself in trouble.

So what is created, we created a 1-inch hole, we tried to assume a choke flow through that assuming no pressure in the secondary and so it dried out, and see how fast it can depressurize that primary because now you have nothing, because uncovered you have all the steam, et cetera.

So that was one and I think we had the

criteria for those or something. And the other thing we looked at, we did a bunch of calculations. We thought, again I think Sanjoy is going to jump over me, but what we did we did some fraud analogy.

Basically saying if I have a counter current flow, like the one I have in tunnels, the fire, how much leakage I should have before that counter current flow breaks, and then we calculated that.

I think we got equivalent to three, which was 18 centimeters. And then we did some other sensitivity analysis and finally we said okay, six centimeters squared looks like conservative.

Also we looked RELAP and SCDAP and if you look at that it shows that if you have one tube breaks and you have half an inch hole you are not going to even pressurize.

But if you have two tubes break you are going to pressurize the secondary, open up the secondary SRV and you are going depressurize the primary.

So then with all of these factors that came together and we tried to pick up a conservative lower bound, which was the six centimeters squared, which coincidentally is the 150 gpm of charging from CVCS that is designed. But that's the way it came up.

MEMBER STETKAR: Interesting. Thank you.

DR. AZARM: You're welcome.

MEMBER STETKAR: Thank you.

CHAIR REMPE: Okay. Well we're running a little bit behind schedule, but why don't we switch to the last presentation, which is fairly short and, even though this isn't bad.

DR. AZARM: Thank you very much.

CHAIR REMPE: Thank you very much for your presentation.

MR. ZOULIS: I think I'll bring you back to schedule.

CHAIR REMPE: I don't know, you've only got one slide. I don't think --

MEMBER STETKAR: No, no, Antonios is the master.

(Laughter)

MR. ZOULIS: Now there is prioritization -- Oh, sorry, so wrong presentation. Yes, we're talking about the severe accident induced steam generator tube rupture and we appreciate all the hard work that Research has done in the last past five years researching this phenomenon.

And, of course, NRR endorses the

completion of the draft NUREG and we're looking forward for the review and public comment period, assembling all of this information that we've created in the last five years.

Now we've planned to develop the RASP, Risk Assessment Standardization Project, handbook guidance, we've already, Kevin didn't know this earlier, but we have revised the current user need that we're going to send to Research recently to incorporate an update to that guidance.

We also may look at updating 0609 Appendix J, which is the steam generator significance determination process attachment --

MEMBER BLEY: Antonios, but the RASP, and that began as like a, just a collection of tools, right?

MR. ZOULIS: Right. That's right.

MEMBER BLEY: But now you are going to formalize that?

MR. ZOULIS: Well we have the RASP, I mean Kevin can speak to that better than I, but we have the RASP handbook, it's available publicly, and in there we provide ways to make sure that the senior reactor analysts are using standard methods of when they do their risk assessments.

Again, it's guidance, it's not a

procedure, it's not a requirement. It was more developed to kind of try to assist the senior reactor analysts when they are performing -- Any of the accident sequence progression analysts --

MEMBER BLEY: Is it a NUREG or how is it --

MR. ZOULIS: Kevin? I'm sorry.

MEMBER BLEY: I thought it was just a place on your website that had the tools sitting on it.

DR. COYNE: It's in ADAMS, but it's not a NUREG.

MEMBER BLEY: It is in ADAMS, okay.

DR. COYNE: It actually came about from a commitment to the, well it had started and then we had an OIG audit about how we ensure that our risk assessments match the as-built as-operator plant.

One of the commitments we made was that we were going to update the RASP handbook to include some quality assurance aspects, and that's what's used by both the ASP analysts and research and the SDP analysts and NRR and the regions.

And over time it's grown, it's a 4-volume set now. It has volumes on quality assurance, volumes on external hazards, shutdown, and internal events.

MEMBER BLEY: Okay.

MR. ZOULIS: So those efforts we're going to pursue immediately, right now I guess, and the next thing that we plan to do or considering are issuing an information notice with the issuance of the NUREG once it's available and final, which we've already communicated with the generic communications branch and that's an acceptable process, and perhaps evaluating the issuing under the generic issue programs, whether it meets further action and we, of course, have that process in Research to evaluate those issues.

That's all I really have today. Thanks again for the opportunity to present to you today. I really appreciate all the work that Kevin has done. I think Raj hit the nail on the head when he mentioned that Kevin has been a big supporter of this and kind of the engine that's been driving this for the last five years over the bumps that we've had.

CHAIR REMPE: So I have a question, because we don't want you to get us ahead of schedule.

(Simultaneous speaking)

CHAIR REMPE: We've listened to what John was saying about a self-fulfilling prophecy with respect to the risk assessment results and if you would put this under the generic issues program, unless there's some change in the risk assessment or the results from LERF and the CDF changes, would it really go anywhere?

MR. ZOULIS: I can't speak for the generic issues program, but I was kind of thinking about John's comment before, or Dr. Stetkar's comment, and just from my personal experience if you have already a faulted steam generator and now you've somehow lost the ability to feed to the other steam generators and you have now a consequential steam generator later on, my gut says to tell me that it's a very low probability sequence, but I mean that's just my, that's just my gut --

MEMBER STETKAR: I don't know. I'm not try to be coy here, I honestly don't know and one of the things that we try to do in risk assessment is what can happen, what's the frequency and what are the consequences of that.

I've identified something that can happen. I don't know what the combination of the frequency and consequences taking it all the way out to releases are, I just don't know.

MR. ZOULIS: Yes, right. Well I'll tell you, John, when we did a lot of the modeling back when I was the licensee we would test whether, so if you had a consequential steam generator fault, I'm sorry, a steam generator fault, you would then test whether you still have offsite power, whether you still have your emergency -- If you did it would take you then to a station blackout tree.

You would then see whether you get offsite power recovery or not and then you would continue to progress down the line. And, again, just my, kind of my gut --

MEMBER STETKAR: But, again, that's --

MR. ZOULIS: I'll come back.

(Simultaneous speaking)

MEMBER STETKAR: The SPAR models are very heavily influenced by this notion that the world revolves around station blackout and failures of turbine-driven aux feedwater pumps.

There are things in the world that can make life really complicated, failures of DC power, partial failures of DC power --

MR. ZOULIS: In particular --

MEMBER STETKAR: An error in some place can get pretty darn interesting.

MR. ZOULIS: Yes.

MEMBER STETKAR: And I'm not saying that necessarily all plants have the same vulnerability. I personally tend to worry a lot more about two plants, only because they only have two steam generators and if you're giving up one you have a lot less available to you. But, again, I don't know.

MR. ZOULIS: We could look and I think we have the ability to look into it. I mean we can probably --

MEMBER STETKAR: Having the ability to look into it requires some fairly clever changes to event models, that's all I have to say is it's not, you can't just look at preexisting cut sets, you can't look at, you know, it's --

MR. ZOULIS: Oh, no, no, No --

(Simultaneous speaking)

MEMBER STETKAR: You actually need to walk yourself into these things systematically.

MR. ZOULIS: I'm sure Selim would love to do that analysis for us.

(Simultaneous speaking)

MEMBER SKILLMAN: Building on John's comment, loss of half of vital DC or half of vital AC, a portion of the battery, or part of the instrument air system can really push the operating team to their limit because of the permutations and combinations of squirrelly behavior.

I mean all kinds of crazy things happen and none of it is diagnoseable right on the front end, and so I'm with John in just maintaining a challenging curious attitude on this.

MR. ZOULIS: I think we've done that by continuing this project and if there's any other areas that we need further evaluation we'll take it back for consideration.

MEMBER STETKAR: I mean one thing that this project has least educated me on is that the conditional probability for a consequential tube rupture is not inconsequential.

I mean we're not talking about numbers that are 10 to the minus fifths largely, we're talking about, you know, somewhere between a 1 percent conditional probability up to perhaps 100 percent conditional probability depending on the plant design and the nature of the scenario and so forth, which is enough to get my interest in terms of thinking about the effects on what are already very, very low frequency scenarios.

You know, it's not too difficult to get factors of two when you're talking about very, very low frequencies, if not more.

CHAIR REMPE: Okay. Anymore questions?

MEMBER BALLINGER: This is only my second time around this because I come late to this, but I just look at this and say to myself there isn't a single here that's not uncertain in a lot, in a big way.

And so am I -- Is that wrong? Just having a computing of a probability without getting a pretty good handle on what the uncertainty is on that probability just seems to me like in this case just not a good thing to do.

It's just extracting more information from the calculations that we do than we're justified in doing.

MR. ZOULIS: The only thing I could say is that we're risk informed, so if it's uncertain then we'll have either defense in depth guidelines, or we have FLEX, or other ways that perhaps the licensee could assure that the steam generator maintain water inventory.

And so we understand that there is some sort of vulnerability and I think that's half the battle and the rest we need to decide how we're going to move forward.

I mean knowing the exact number I think is not the most important thing.

MEMBER BALLINGER: Yes, I just worry that a calculation that gets done, somehow gets enshrined, and is a number which has a very large uncertainty which at this point doesn't make any difference.

But somewhere down the line who knows what happens and all of a sudden that number does make a difference for some other unforeseen set of circumstances that we have to deal with and then not having quantified the uncertainty becomes a big albatross around your neck. That's all.

DR. COYNE: If I could add to Antonios's comment, I think the comments about needing to identify areas of uncertainty in particular areas that we believe introduce large uncertainties is very important in the report so we really appreciate that feedback.

You know, a lot of the discussions on the timing and these high temperature effects, you know, I had the same issue when I first encountered this phenomenon six years ago of, you know, wow, these things are, you know, three minutes seem to be a big uncertainty from expert judgment from Argonne National Lab, that's what they put on these timings, three minutes.

It turns out that probably isn't so far off given how fast things are happening. So we will probably never be able to tell exactly that it's at

14,210 seconds, but the relative behavior and the shifting, you know, that part I'll probably --

(Simultaneous speaking)

MEMBER STETKAR: Yes, a picture like this. This might be, I have small hands so I can't get the uncertainties large enough, but if we have confidence that it works this way --

DR. COYNE: Right.

MEMBER STETKAR: -- and there's good reason to believe that despite the fact the uncertainties in both the, and here I'm talking about the relative times of hot leg and the tube rupture, I think that's, to me that's compelling information that indeed, yes, I have confidence in the physics.

DR. COYNE: Right.

MEMBER STETKAR: On the other hand if you hadn't thought about the uncertainties and indeed there is a substantial overlap such that even though you might show right now 20 minutes, which is sort of what that shows, the uncertainty is plus or minus an hour, you'd have a problem.

DR. COYNE: So where we are when we started this work is we had the detailed work for Westinghouse. I forget the exact words we used at that time, but it was not a large contributor to LERF but it was not an insignificant contributor, so it was something that was worth paying attention to.

CE we expected behaved, the CE style designs I should say, behaved worse than the Westinghouse style designs. We had not done the calculations to show that.

So that was the big contribution from this work is being able to show that for CE you had that overlap that you were demonstrating for Westinghouse much less so.

I acknowledge the uncertainties hadn't been quantified honestly, they probably won't be quantified because of resource limitations that we have and would it appreciably change how NRR would use the final regulatory product.

But I do agree that the report needs to identify those areas of uncertainty and if it can be easily or readily quantified we should pursue that, but a lot of these issues that came up today would be multi-year research projects in and of themselves and probably beyond our practical ability to do that quantification.

MEMBER BALLINGER: Can you identify the areas of uncertainty and then make some kind of statement about what your judgments are as to what the effect of that uncertainty might be?

DR. COYNE: Well that would be up to the individual technical aides for the various aspects of the project and I, you know --

MEMBER BALLINGER: Okay.

MEMBER STETKAR: I think I heard Kevin say that they try to at least identify a little bit more clearly the areas of uncertainty, unless I'm putting words in our mouth.

DR. COYNE: We're trying to and I, you know, we got a lot of feedback today, which is good, and there is some areas that we probably need to clarify in the report to further make it clear of what's feeding into the uncertainty on the evaluation.

CHAIR REMPE: Antonios mentioned something that hasn't really been brought up much today or if at all of FLEX and how it impacts the results currently documented in the report, which doesn't really discuss FLEX very much, and so that's another issue to --

DR. COYNE: Out of my scope, but honestly --

(Simultaneous speaking)

DR. COYNE: -- when we stared this work in 2010, and I think I might have mentioned this offline to several people, one of the potential outcomes was the identification of having alternate means for plants with CE style steam generators to be able to mitigate this better, and by mitigating having alternate means of adding water to the steam generators.

Since we started this work we have the FLEX program that has come about, which probably addresses, we haven't looked at it as part of the project, but it probably addresses a lot of the need that we initially thought would probably arise from the initial work when we started in 2010.

But we didn't evaluate that as part of this project though it would be a beneficial outcome from the FLEX program in general.

MEMBER STETKAR: I mean in a sense, although it's not called FLEX, there is at least some discussion about SAMGs in this, and I'm still not quite sure about where the distinct black line is between FLEX and SAMGs because there seems to be some overlap in those areas, so in some sense it does address some of those issues about depressurizing and alternate ways of feeding steam generators, which now might be called be FLEX, but in the context of this report might have been called SAMG.

CHAIR REMPE: Okay. So if there are any

more comments or questions before we go around the table we usually open up the line for public comments and while we're dealing with that are there any folks in the audience here today and you feel obliged to want to make a comment?

MR. ZOULIS: Thank you very much. Thank you.

CHAIR REMPE: Okay, are we open?

MR. BROWN: Yes.

CHAIR REMPE: Is anyone out there, because we don't have any way of checking you're out there unless you'll speak up and just say the public's out here and I have, you know, let us know you're there.

So based on the no response I am assuming that we have no public --

MR. HOFFMAN: There is one member.

CHAIR REMPE: Oh, good, good.

MEMBER STETKAR: Thank you.

CHAIR REMPE: Thank you. Do you have any comments today?

MR. HOFFMAN: Yes, I had a couple of comments.

CHAIR REMPE: Please identify yourself and then go ahead and give your comments.

MR. HOFFMAN: Yes. My name is Ace Hoffman and I live near the San Onofre Nuclear Generating, former nuclear generating station, so I've been very interested in the steam generator issue for a number of years now.

There were a lot of times when it was very hard to hear what was going on and when there were less than three listeners the machine wanted to cut us off rather repeatedly, so I wasn't able to pay as close attention as I would have liked.

But I did hear a lot of condemnations of the way PRAs are used and I couldn't agree more with that, and also the 95 percent evacuation figure, I wonder how carefully that's been decided whether or not it's accurate and whether or not it wouldn't be a good idea to go to say oh, 98 percent or 99 percent because those percentages can each be tens of thousands of people, and so it could make a huge difference.

And I heard talk about the operators making mistakes in not understanding what the real situation is with the steam generators as one of them loses pressures and various issues like that are just not going to come out in a PRA.

So I think that based on everything I heard today that the NRC needs to rethink the use of PRAs. That's about the only comments that I could make because, again, it was rather hard to hear a lot of times during the day.

But thank you very much for a very interesting discussion. Thank everybody.

CHAIR REMPE: Thank you for your comments. Are there any other individuals out there on the public line who'd like to make a comment?

So with that I'll ask you to close the public line and this is the time when we usually go around the table and ask for comments. As you start making your comments, because the current plan right now is to have a full committee meeting to discuss this document.

The document will not be changed, right, Kevin, before the full committee meeting and so it is what it is.

DR. COYNE: Correct.

CHAIR REMPE: And then they will, actually the document is already apparently available to the public if the delve into it, the NRC system, and so at that point then they'll make some changes and issue it officially for public comment at the end of this year, or however they can with the schedule.

DR. COYNE: Correct. Following NRC's feedback, whatever feedback we receive following the

June meeting we'll reflect those comments as best we can in the document and do some other cleanup and editorial work and then with the goal of hopefully issuing it formally as a draft NUREG for public comment by December.

CHAIR REMPE: Okay. So as you go around the table and you're making your comments, maybe highlight your key points you'd like to have mentioned at the end of this meeting to help Kevin a bit more.

MEMBER BANERJEE: So, Joy, can you give us a timeline again? I'm a little confused, this will be discussed in the full committee meeting in June?

CHAIR REMPE: Yes.

MEMBER BANERJEE: We'll write a letter in June?

MALE PARTICIPANT: Yes.

CHAIR REMPE: You know, that's always the full committee's decision, but that is the potential hapicure and so if you have some thoughts now that would be interesting for me to hear, things that, I mean they'll probably have an hour or two of full committee meetings, things you think they should highlight at the meeting, those are the issues to bring forward at this time.

MEMBER BANERJEE: We've written a lot of

letters, right, on this, and what would be the particular issues that we would, just the new material we are seeing or is it about the CE Plant, I'm just trying to grapple with --

CHAIR REMPE: Okay, so the letters that have been written were on the older steam generator program, not since the user need was issued and their work.

MEMBER BANERJEE: Okay.

CHAIR REMPE: And so, no, there's not any letters from ACRS since this addressing of --

MEMBER BANERJEE: In 2009 or something, right?

CHAIR REMPE: So it's this document, what's going to occur in the future, those kind of issues. Do you want to start or do you want a minute more to think about what you want to say?

MEMBER BANERJEE: Well I can start, but, you know, I need to be clear in my mind what the objectives are in what we are trying to accomplish with this letter.

In any case I can give you my views. I think that the Staff did a great job, at least for the part of it that I am more familiar with, which is the thermal-hydraulic part.

They presented their results very well and I think I got a clear picture. What I don't know, and I think this is really also what Ronald brought up was the effect of uncertainties, in this case primarily related to the thermal-hydraulic calculations.

As we were discussing earlier one of the issues which is very important is how much mixing occurs, you know, and that's sort of key to determining what will happen and as your hot legs get larger, really for the CE design, what you have is the possibility of a core region which is not very well mixed so it sort of gets into water in the plenum which is also shallower.

So that we've got a situation which is somewhat different from what was happening to the Westinghouse plants and where we were more sort of willing and prepared to consider that because there was quite a distance and the tube was sort of waving around and the way the, you know, the technical details which I don't want to get into, but which give us let's say more assurance physically that mixing would occur.

So we need to understand really what the uncertainties in these calculations are in terms of the mixing. So at the bottom line that's the key parameter that we have to look at.

CHAIR REMPE: So with respect to the draft NUREG do you believe you have enough information or do you believe you need the underlying documents to better --

MEMBER BANERJEE: I'd have to go through the NUREG in sufficient detail to find out whether these types of thermal-hydraulic uncertainties have been taken into account or not.

CHAIR REMPE: Okay.

MEMBER BANERJEE: And maybe the Staff can address this, how they would address this issue.

CHAIR REMPE: Okay. Dick?

MEMBER SKILLMAN: I thank the Staff for a very thorough presentation. It's obvious there has been a tremendous amount of work. I have no comments. Thank you.

CHAIR REMPE: John?

MEMBER STETKAR: I've been admonished that I can't just say I have no additional comments what I said, so I have to say yet one more time --

CHAIR REMPE: That's the idea here.

(Laughter)

MEMBER STETKAR: But not all of it. The two things, short and sweet, are one, in that PRA insight is the thing that I had mentioned about are the current constructs of PRA models really looking for all of the scenarios which might have a vulnerability for these consequential tube ruptures.

In other words, are they complete in that sense, are the scenario definitions complete. So that's one thing in terms of PRA insights, if you will.

The other one is because of that use of extreme caution when you try to, if you try to derive any comprehensive PRA insights about how important this phenomenon might be either to core damage frequency, which is probably not all that much, but more importantly to large early release frequency that the analyses that had been done are the analyses that were done but they aren't necessarily a comprehensive look at the picture, so be careful in terms of selling the overall conclusion that way.

CHAIR REMPE: Okay. Ron?

MEMBER BALLINGER: Well I mentioned the uncertainty issue. I just, it's hard for me to think of being able to assign a value for LERF in a PRA sense without factoring in the uncertainty part.

It just, I don't know. It just makes, and I know you've guys have done an awful lot of work, obviously, very good work, and hampered by the fact that you have no data, especially on Alloy 600 and 690, you know, projecting where forward when you have zero data effectively is just a tough, tough, tough, job to do.

CHAIR REMPE: And, Dennis?

MEMBER BLEY: Yes. You know, I went back and I reviewed my notes from the original meeting we had on these issues and boy a lot has been done and a lot has gelled in the intervening years. It's impressive.

I found the meeting really helpful. There has been much more thorough thought on many of the sticky issues that affect the phenomenology and the risk assessment than one finds in the draft report and I hope they can eventually get that rectified in the document itself, especially with regard to uncertainties in the thermal-hydraulic analyses and the use of them in the PRA.

I do agree with Mr. Stetkar that the uncertainty in the scenario development and especially how the operators might interact with that is still a little bit on the weak side, but I think they made many of their cases pretty well.

The pressure-induced steam generator tube rupture piece of the NUREG and the presentation are certainly much less convincing than the other work.

CHAIR REMPE: Okay. I think that's it,

right? So the --

MEMBER BLEY: That's all from me.

CHAIR REMPE: Okay. So I also wanted to congratulate the Staff on a lot of work and I realize that it's been difficult with other priorities coming into the NRC and I think they've done a good job.

In addition to documentation of the uncertainties I wish there would be also more documentation on the limitations of the analyses, that's always an issue no matter how much money you have, you have to stop somewhere and I don't think it sometimes has been brought out as clearly as it should be in the document.

I would really like to have, or at least to ACRS if possible before the full committee meeting and as soon possible the calculation reports or background for the MELCOR as well as the CFD analysis for the CE Plant.

I think that there are some differences in the way SCDAP does a core progression versus MELCOR and I think seeing more details might be educational because if we're going to write a letter I think it may be helpful so that we have a better understanding of what was done.

Sanjoy mentioned the CFD analyses and the fact that we don't have the underlying report, there just may be some information. If you think about the flaw analysis and what was done about the different plants and understanding how the assumptions were brought forward in the model it helps us to be more educated, so I would like to request that strongly.

I also would like to know a little bit more about what was going on about stainless steel oxidation, if the temperatures when you think about that they can occur at low temperatures at 600 degrees C.

I mean Dana has mentioned well if there's some hydrogen there it can delay that, but I mean that could really make a difference if you start having some issues with the iron oxide forming and becoming molten or if it's falling off it would change the situation, and so it would be good to know if there is some hydrogen there and, again, maybe the analysis reports will give us that information.

And so with that if there's nothing else from the Staff I'd like to --

(Simultaneous speaking)

CHAIR REMPE: Oh, Dr. Powers, you're back in time to make some comments.

MALE PARTICIPANT: Finally closing --

MEMBER POWERS: Oh, on this --

(Simultaneous speaking)

MEMBER POWERS: On this particular subject or am I free to talk about anything?

(Laughter)

CHAIR REMPE: If you don't mind let's keep it to the subject.

MEMBER BALLINGER: You're not aloud to not make comments.

MEMBER BANERJEE: Why? It would be more amusing if he could --

CHAIR REMPE: We'll do that after I bang the gavel, okay, if any of us would like to stay.

(Simultaneous speaking)

MEMBER POWERS: Yes. We're working on a tough little issue here. The things we saw were very enlightening, very useful. It is clear that the uncertainties we have and how they affect the results here are pretty serious in this.

We need serious consideration in this analysis because yes, we see a horse race between failure at the vessel nozzle versus failure at other locations and it makes a difference in that.

But that's a fairly formidable job to take into account a full epistemic so I tend to look on this work here as a snapshot of what we think we know now and maybe identifying some of the crucial uncertainties, and I think we've captured some of those areas of uncertainty.

I think one of the things that we need to think about is in ACRS when we talk on the research report is where should we be looking for some experimental support to help this along so we're not just totally relying on the computer grids to do this.

And I guess my final comment is I really appreciated learning about this gamma distribution, that was something I had not appreciated how useful a tool that was for multiple mechanisms affecting things and I learned something here.

CHAIR REMPE: So with that, let's close the meeting.

(Whereupon, the above-entitled matter went off the record at 5:03 p.m.)

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Severe Accident-Induced Steam Generator Tube Rupture (SGTR)

Opening Remarks

Dr. Raj Iyengar, RES/DE

Consequential Steam Generator Tube Rupture (C-SGTR) Subcommittee Briefing
April 7, 2015

Purpose and Background

- NRR User Need Request “Developing Analytical Bases and Guidance for Future Risk Assessments of Consequential Steam Generator Tube Rupture (C-SGTR) Events” issued December 2009
  - Requested development of improved analytical bases and guidance for probabilistic risk assessments of C SGTR events
- Subsequent to an April 2011 ACRS sub-committee briefing, NRR Management requested RES to restructure project to focus on near-term deliverables and to allow for an incremental approach
  - RES issued a document (Jan. 2012) identifying “hold-points” to resolve near-term deliverables before proceeding with the full scope
- Informal meetings with lead ACRS member for C-SGTR issues (Dr. Rempe) held January 2012, January 2013, and April 2013
- ACRS full-committee meeting in May 2013
- Subsequently, staff prepared a draft NUREG (transmitted to ACRS staff on Feb 19, 2015)

Severe Accident-Induced Steam Generator Tube Rupture (SGTR)

Thermal Hydraulic Analyses

Dr. Michael Salay, RES/DSA
Dr. Christopher Boyd, RES/DSA

Consequential Steam Generator Tube Rupture (C-SGTR) Subcommittee Briefing

April 7, 2015

Topics

- CSGTR Scenario Description
- TH analyses
- Method (CFD & System Code)
- Experimental Basis
- Differences Between CE and Westinghouse Plants
- Combustion Engineering MELCOR analyses

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.

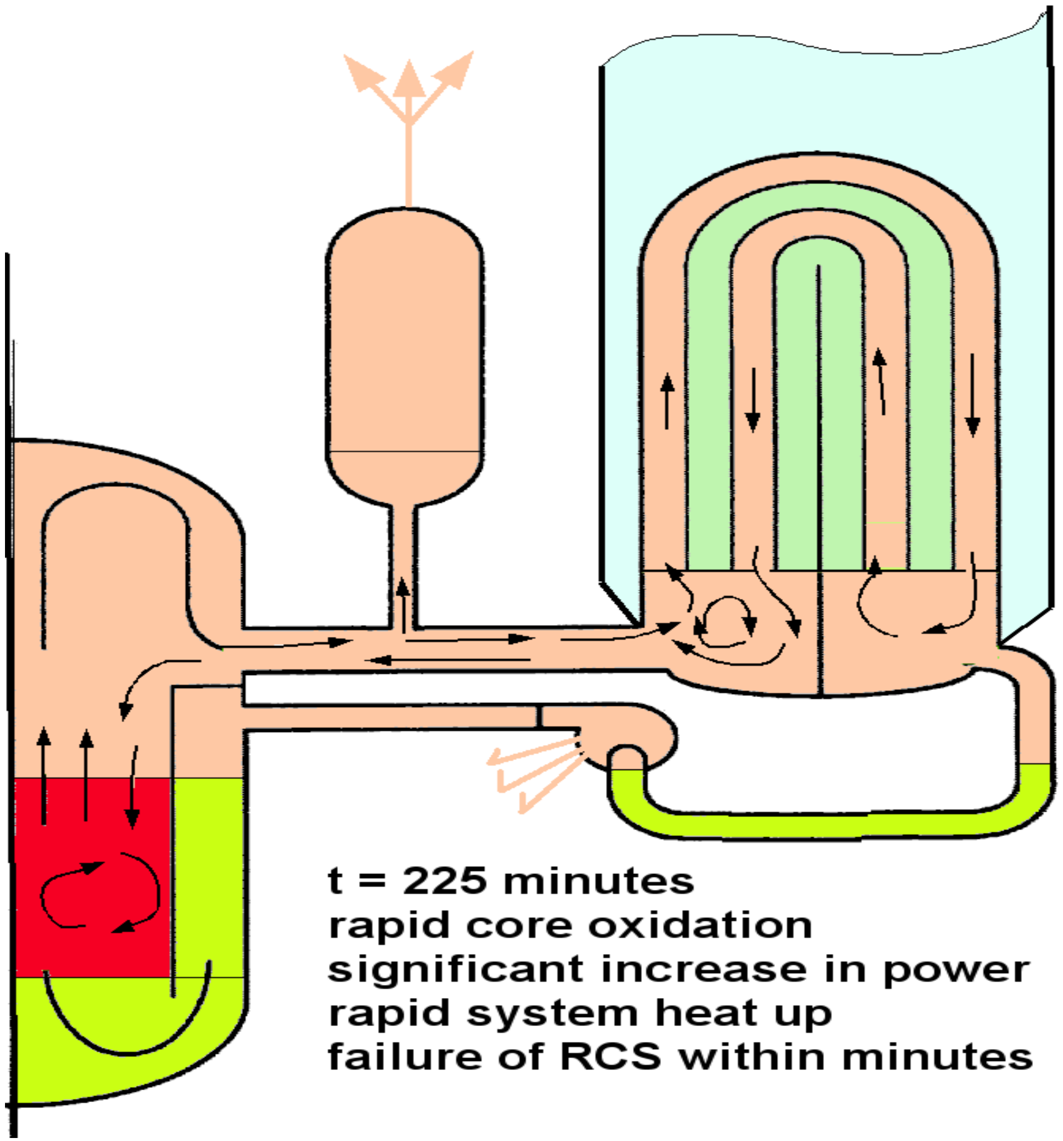
- If SG tubes fail first, then a flow path is created that bypasses the containment
- Failures of other RCS components (hot leg or surge line), RCS blow down into the containment
- Determining SG tube failure is important in consequence analysis

A Fast Scenario

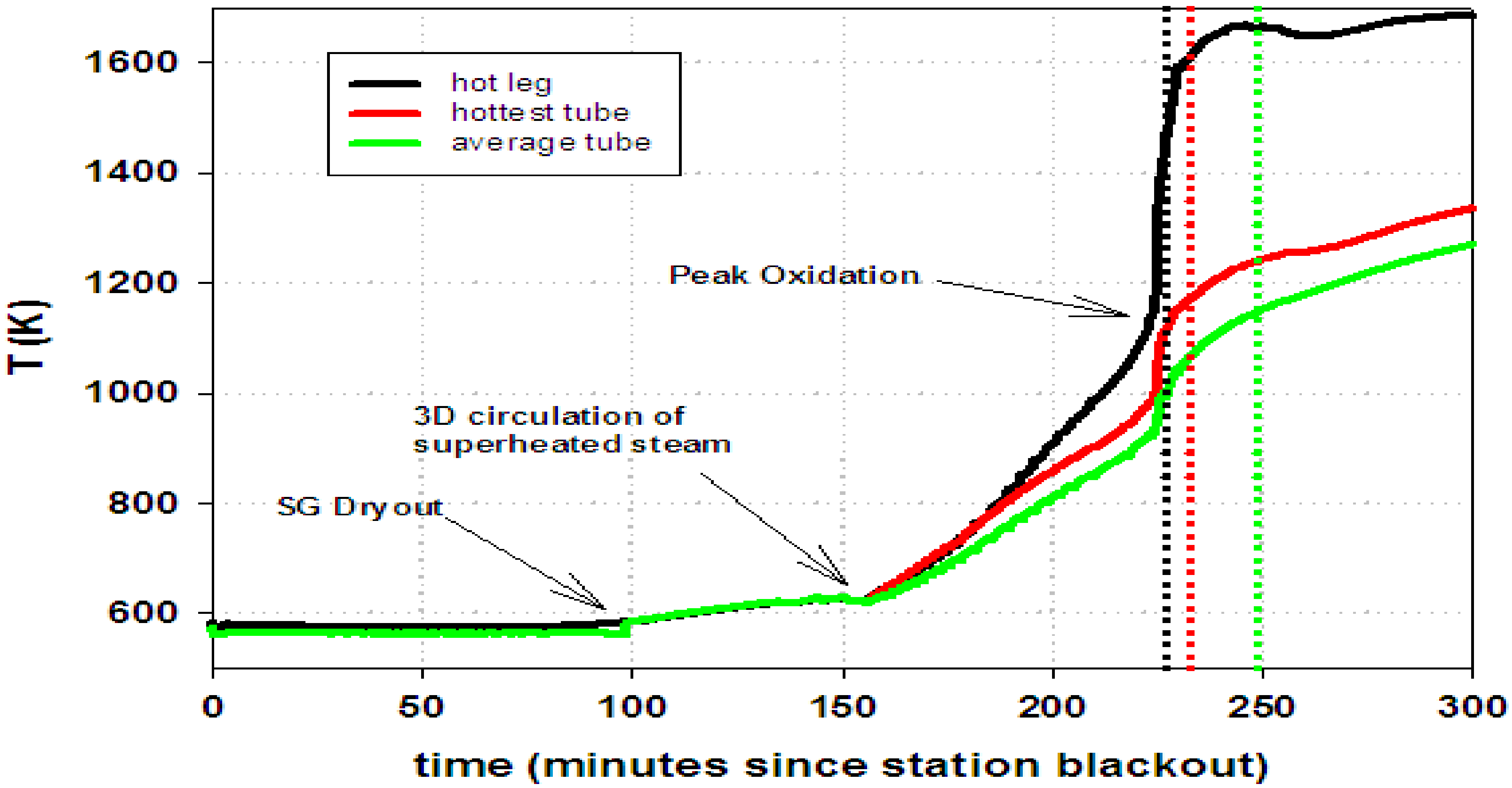
RCS failure within 4 hours

- 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.

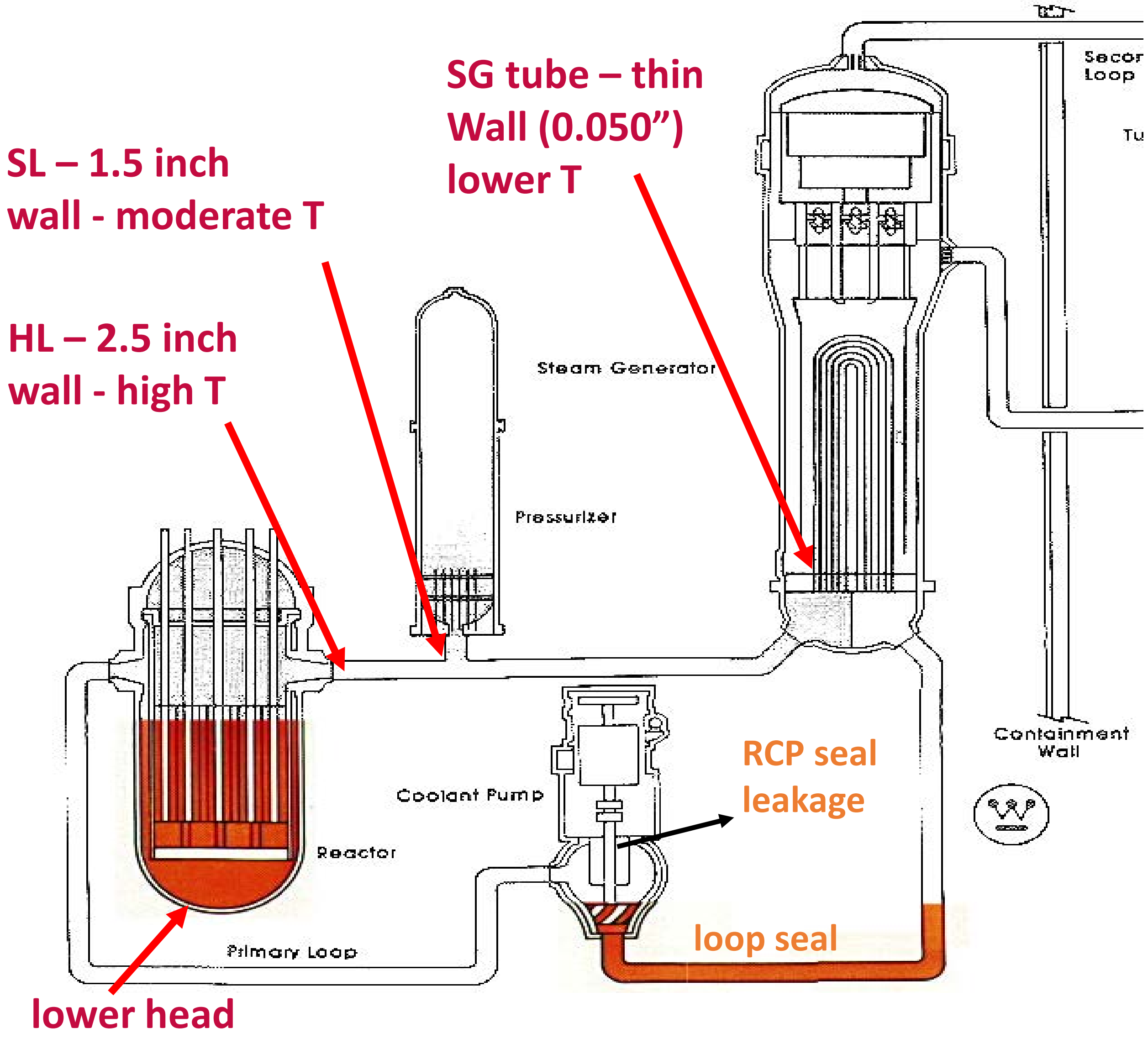
- More likely scenarios involve some auxiliary feedwater or operator actions that significantly delay the failure time.



RCS Structure Temperatures –Fast Scenario



RCS Points of Interest



Considerations

~ T=1475 K, start of rapid Zirc Oxidation

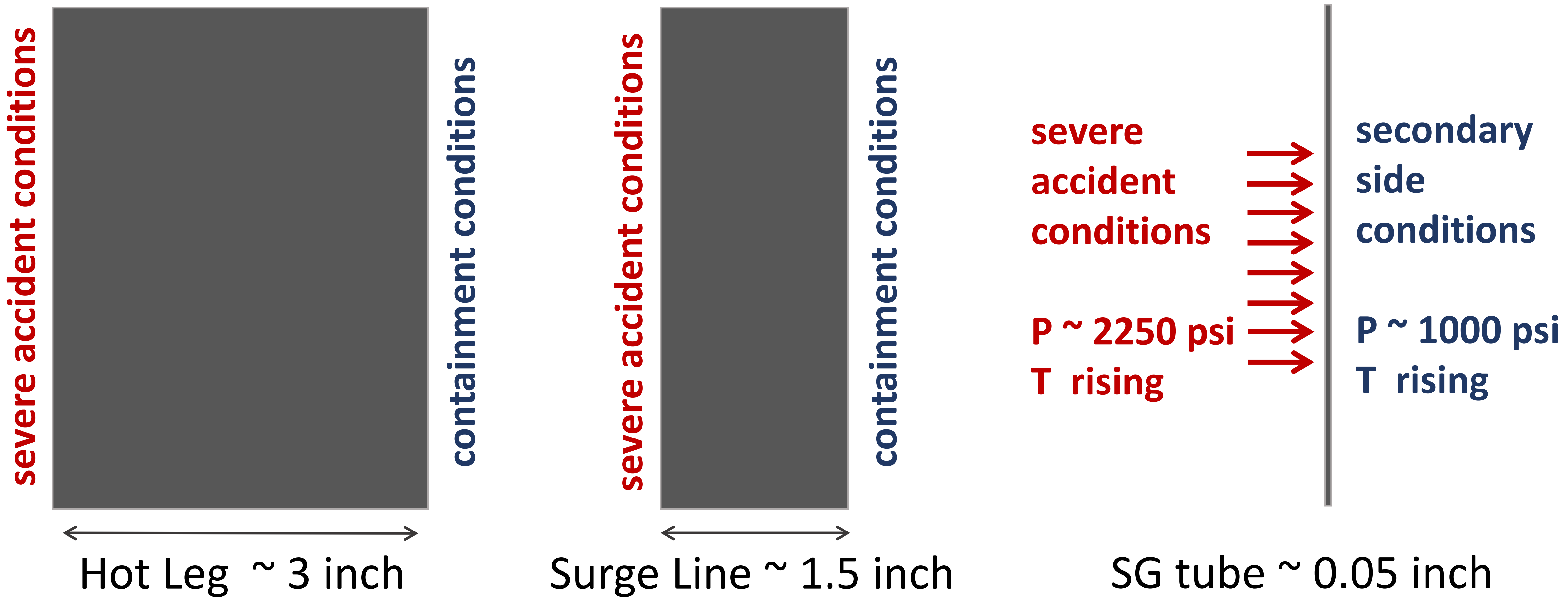
~ T=1725 K, Melting Stainless and Inconel

~ 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.**

RCS Boundaries



High-Dry-Low

Primary Side

High Pressure

* no significant leakage to reduce pressure

Secondary Side

Dry

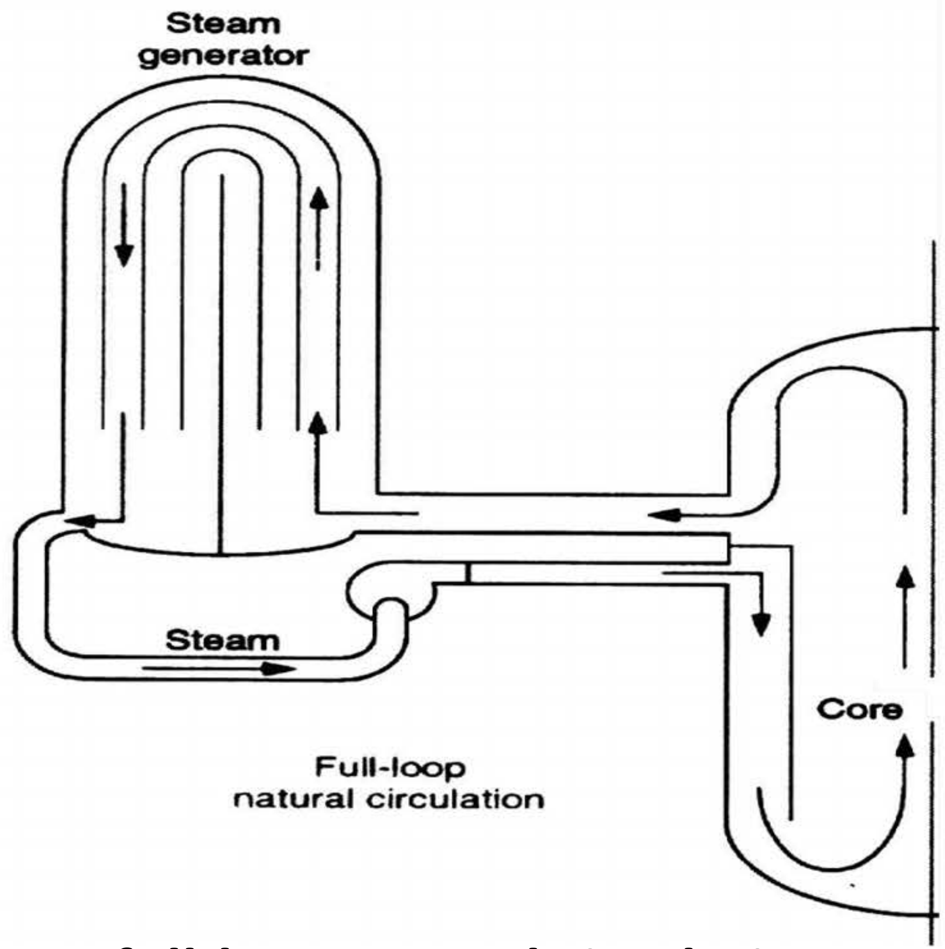
* Loss of water allows tubes to heat up

Low Pressure

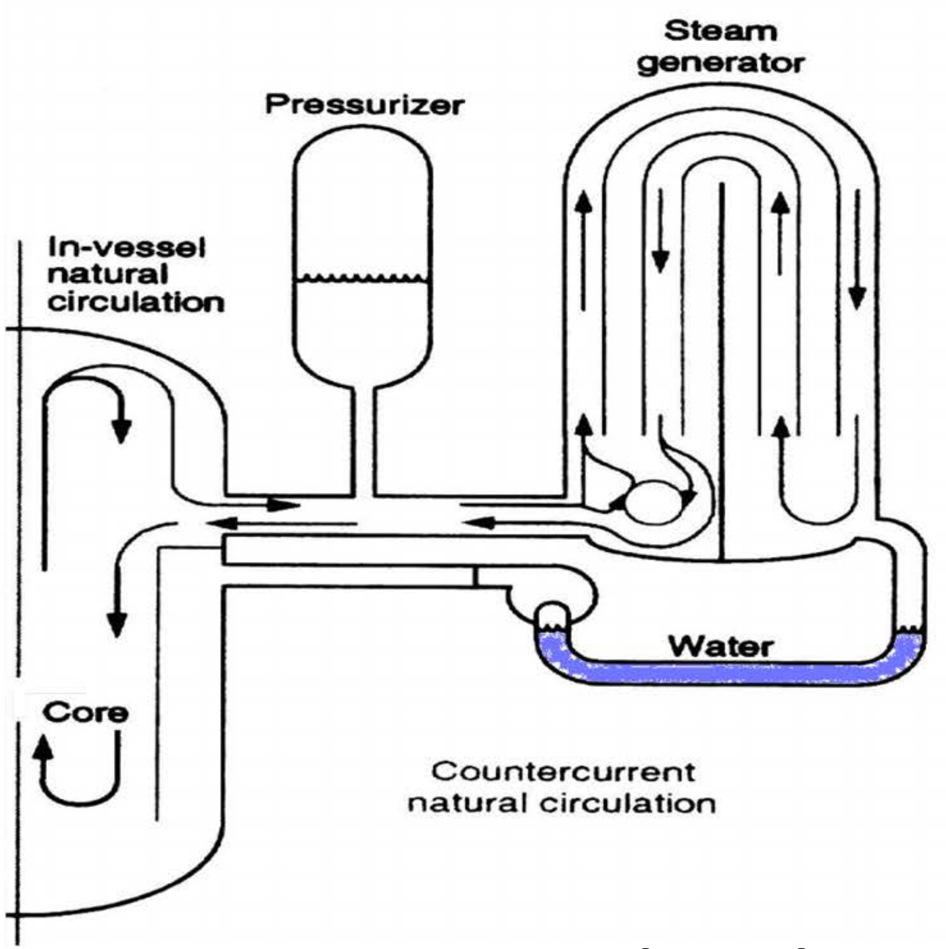
* Secondary side leakage increases pressure difference (i.e. mechanical load on wall)

SG tube wall

Two Flow Patterns - PWRs with U-Tube SGs



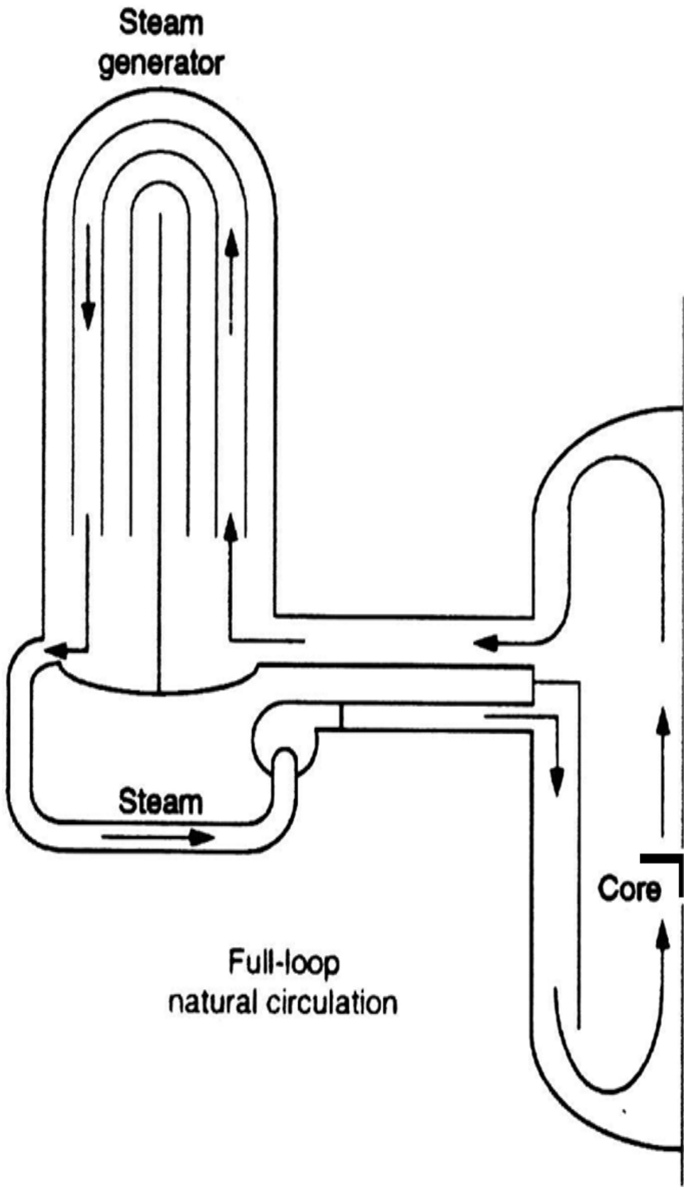
full-loop natural circulation



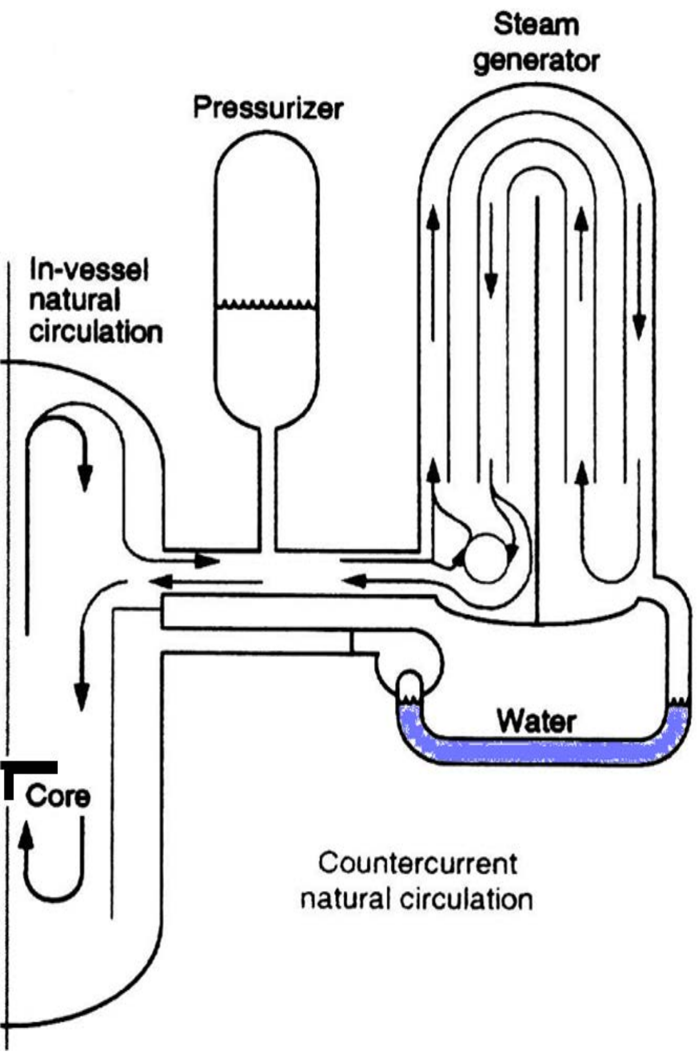
Counter-current natural circulation

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

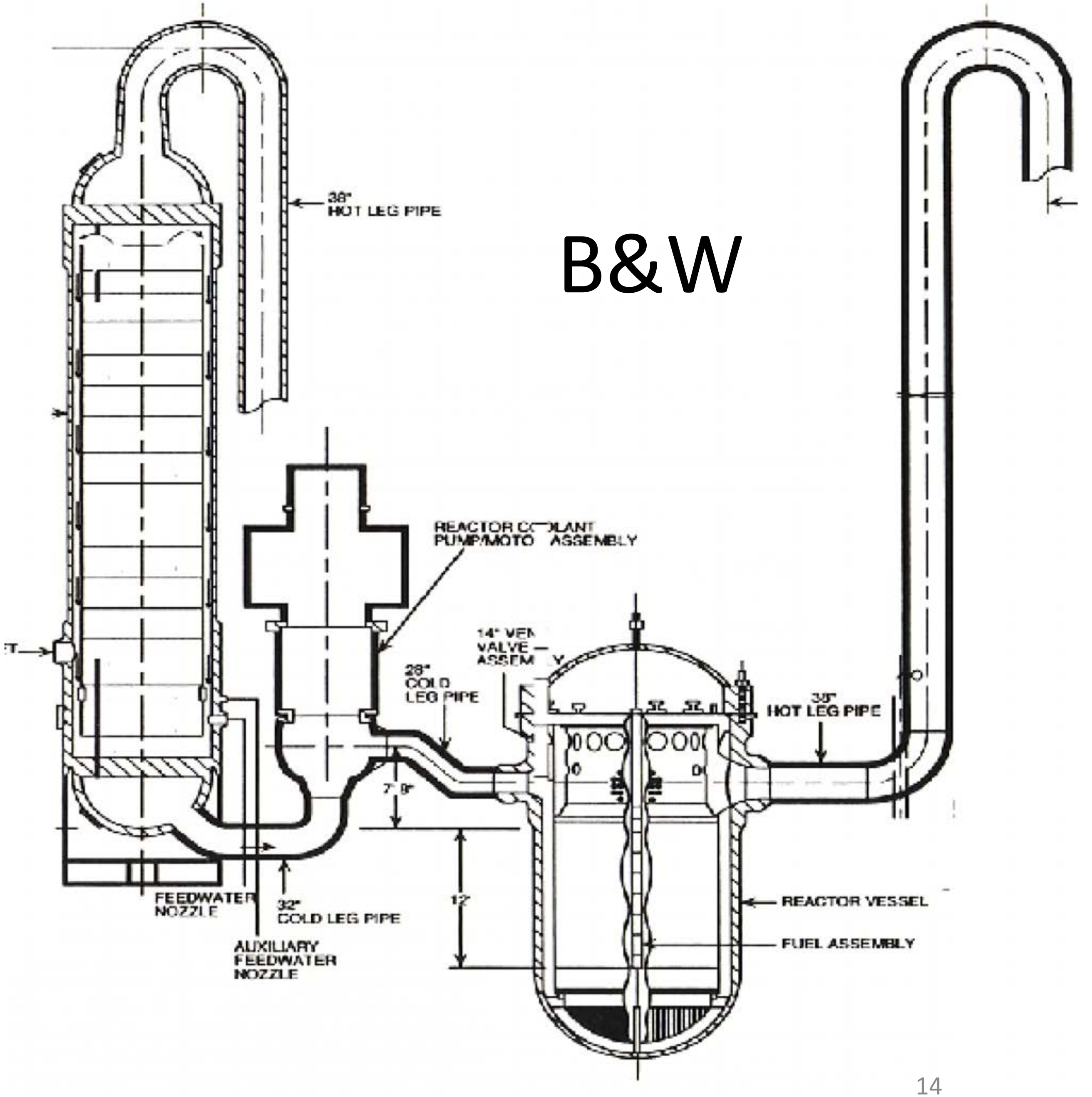


- With the pump loop seal filled with water, a counter-current flow field is established.
  - This flow pattern mixes the hot gases with cooler flows returning from the SG. The thermal challenge to the tubes is reduced but not eliminated.
- System code models require external information to ensure consistency:
  - 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.

B&W



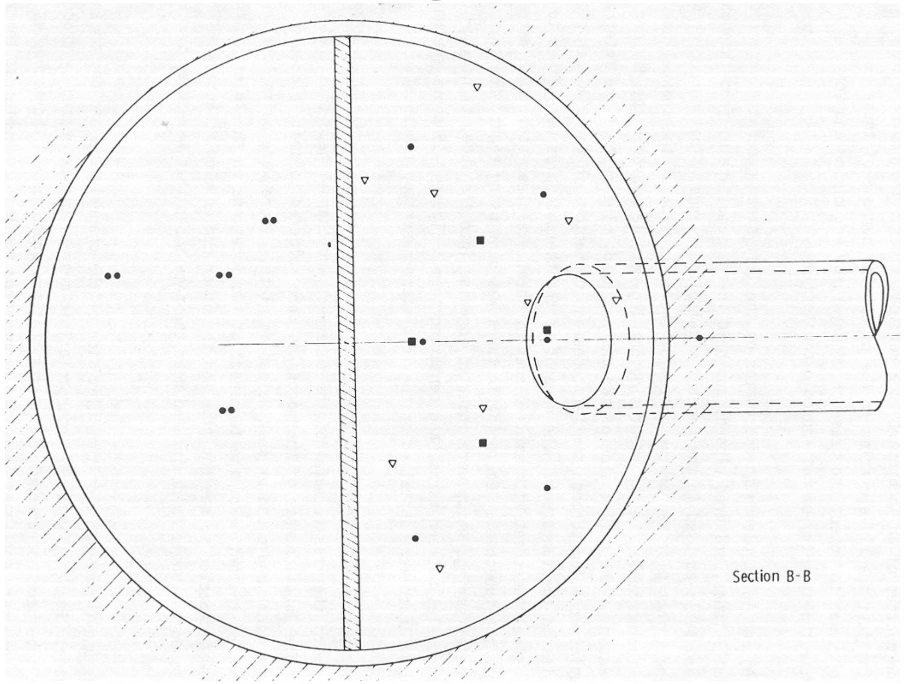
TH Analyses

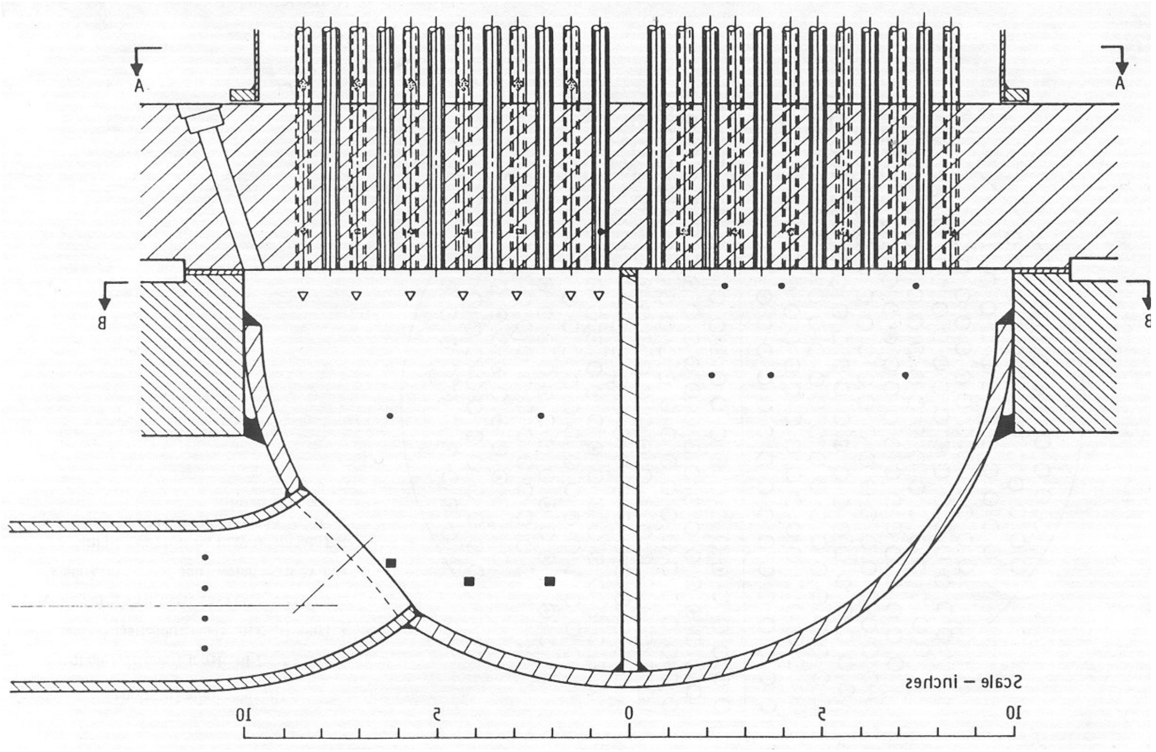
- Both Westinghouse and Combustion Engineering TH analyses used
  - Input to tube-failure calculator and Finite Element Models
- Westinghouse TH analyses performed for the Steam Generator Action Plan (SGAP)
  - Documented in NUREG/CR-6995
  - TH analyses for Combustion Engineering (CE) plants did not receive the same level of attention
- TH analyses conducted with CE under C-SGTR project

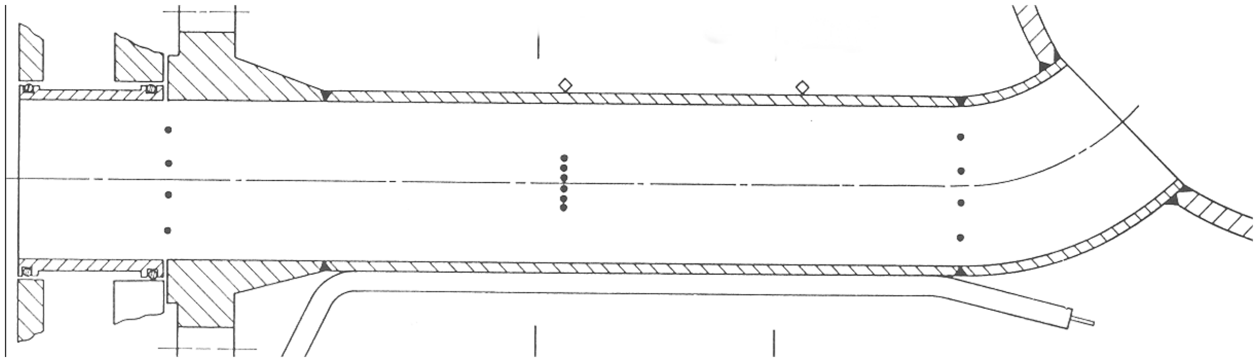
Method

- Computational Fluid Dynamics (CFD) and System Code
  - CFD predicts spatial flow and temperature distribution
  - System code predicts transient behavior
    - Uses CFD results for counter-current flow in hot leg
    - Results can be combined with those of CFD to obtain a transient spatial temperature distribution
  - CFD approach validated using Westinghouse 1/7th scale tests

Westinghouse 1/7th Scale







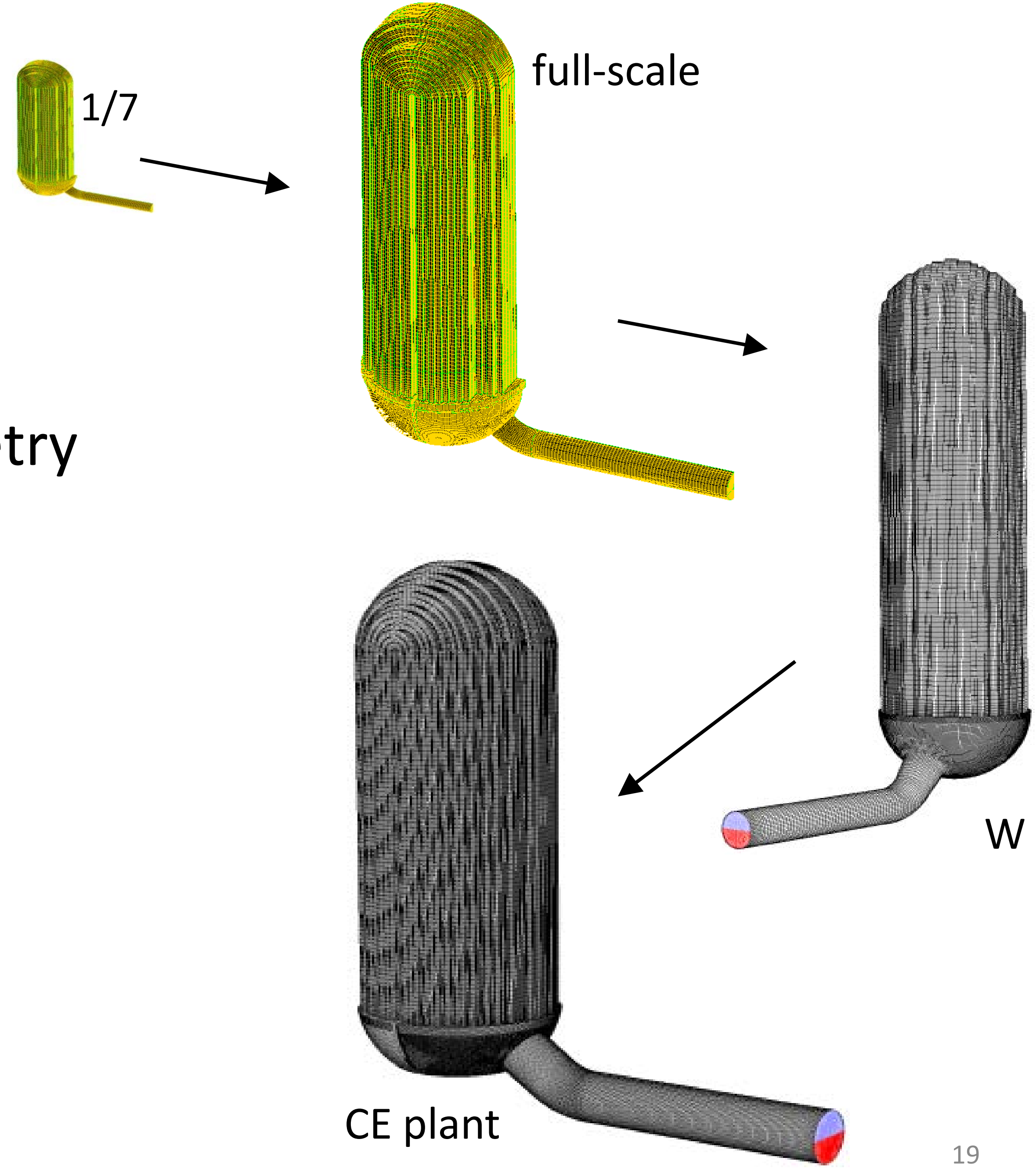
* 4 inch hot leg
* 216 SG tubes
* Temperature measurements
* Facility included vessel, electrically heated core, and two steam generators.

Westinghouse 1/7th scale tests

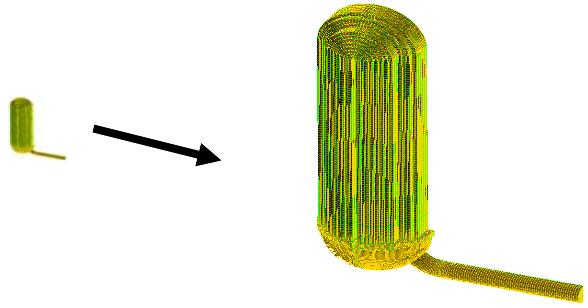
- Demonstrated the counter-current flow path
- Not focused on tube integrity but provide valuable insights
- Many scaling studies demonstrate applicability to full-scale
- Results helped inform modifications made to system codes (SCDAP/RELAP5 or MELCOR) used to study the station blackout scenarios.
- 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



Scale-Up

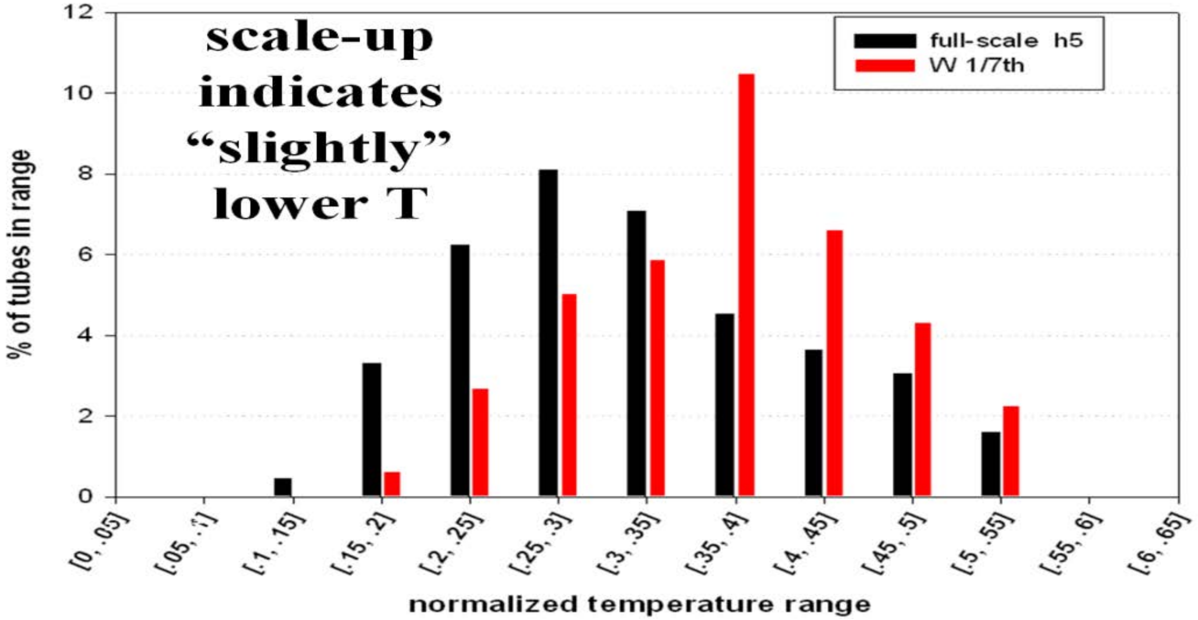


1/7th scale / SF_6 >> full scale severe accident conditions

Solutions are similar when heat transfer rates are scaled.

Heat transfer rates under severe accident conditions are different, however, and the geometry is not similar.

Impact of these distortions needs further analysis

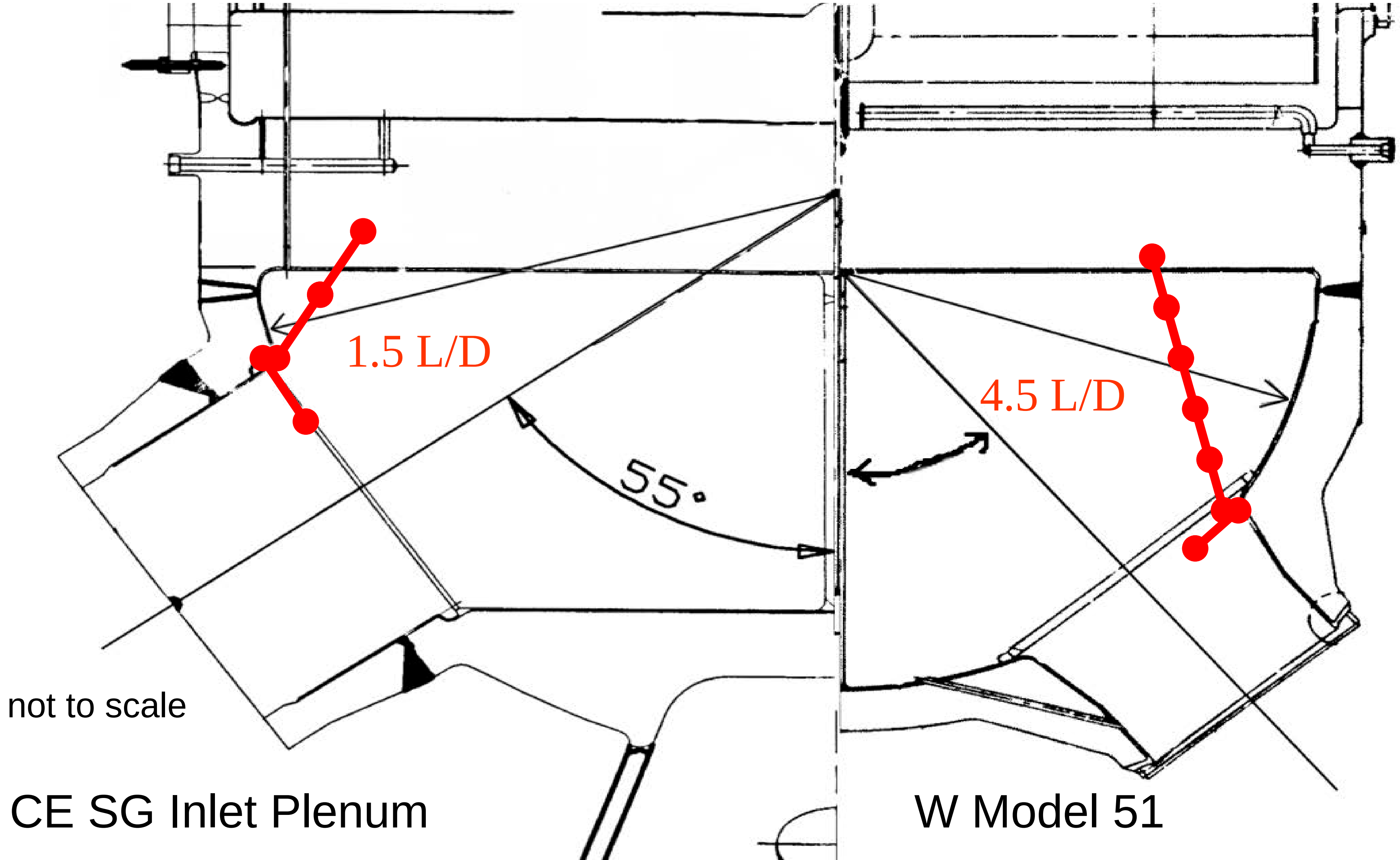


	full-scale h5	1/7th scale case
Tube heat loss (kW)	5047	3.7
m (hot leg flow – kg/s)	4.2	.059
Th (hot leg hot temp)	1404 K	428 K
Tht tubes (average tube hot temp)	1046 K	373 K
mt (tube flow - kg/s)	9.1	0.12
m_t / m_L (recirc. Ratio)	**2.16**	**2.06**
% of Hot Tubes	**38**	**38**
f (mixing fraction)	**.92**	**.81**

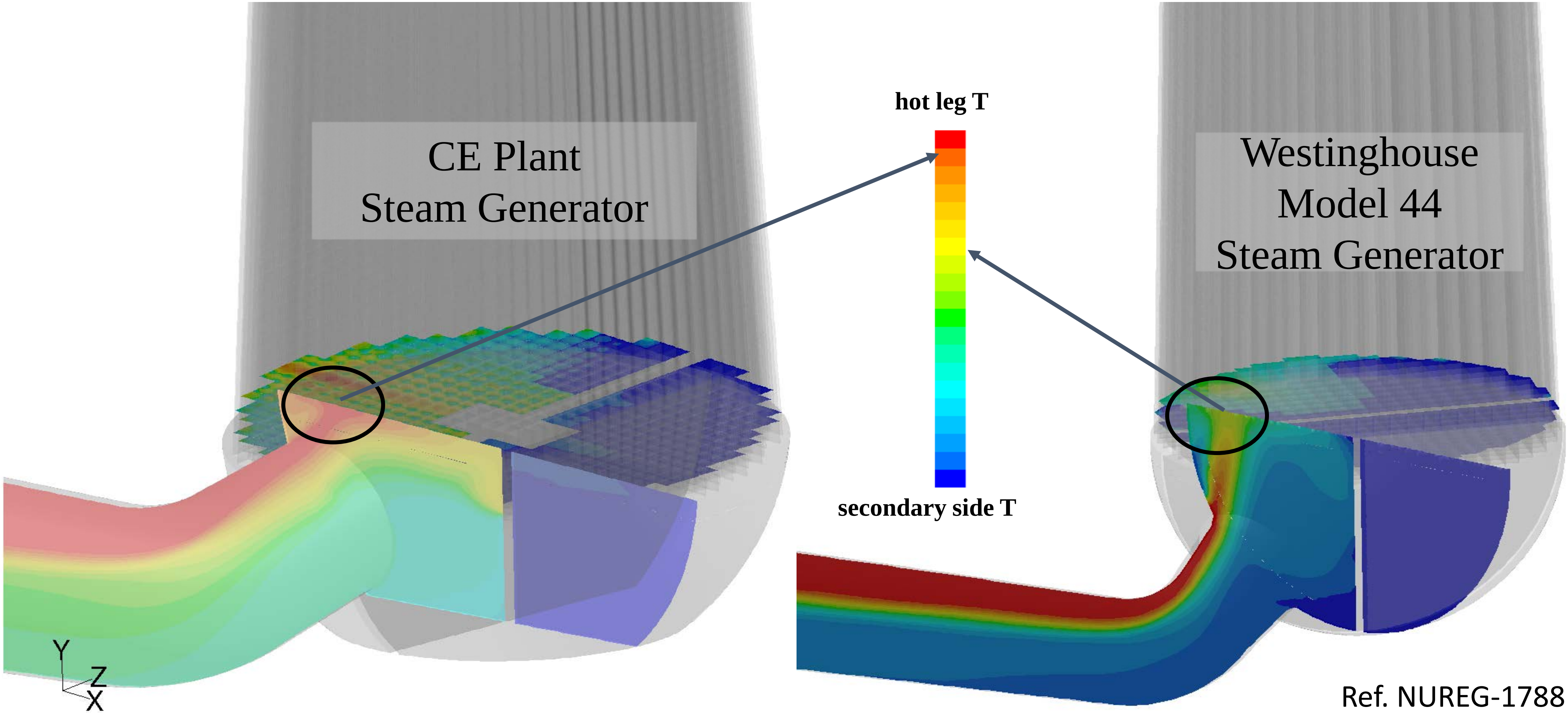
CE SGTR Behavior Differs from Westinghouse Plants

- 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)



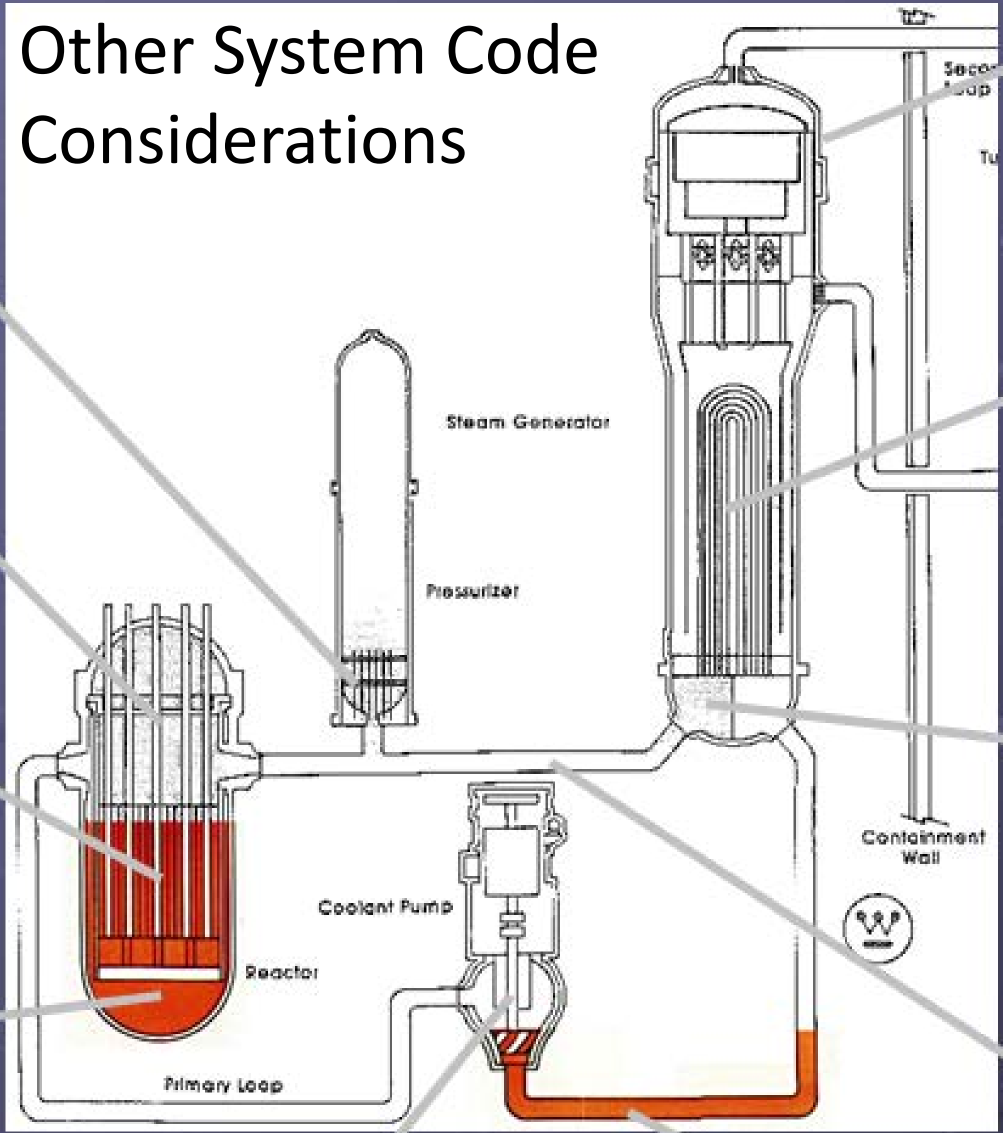
CFD Predictions - Westinghouse and CE (hottest tube region circled)



Ref. NUREG-1788

(temperature contours on vertical centerline plane of hot leg)

Other System Code Considerations



pressurizer draining
surge line orientation

natural circulation
core bypass flow

oxidation rate
core blockage
nodalization
natural circulation

nodalization
downcomer clearing

pump seal leakage
suction height

loop seal clearing

shell heat loss
SG depressurization

tube heat transfer
secondary flows
mass flow
tube fraction
leakage
plugging
vertical node count

inlet Plenum mixing
recirculation ratio
plume T distribution

HL Flow rate
entrainment
radiation modeling
entrance effects

MELCOR CE calculations

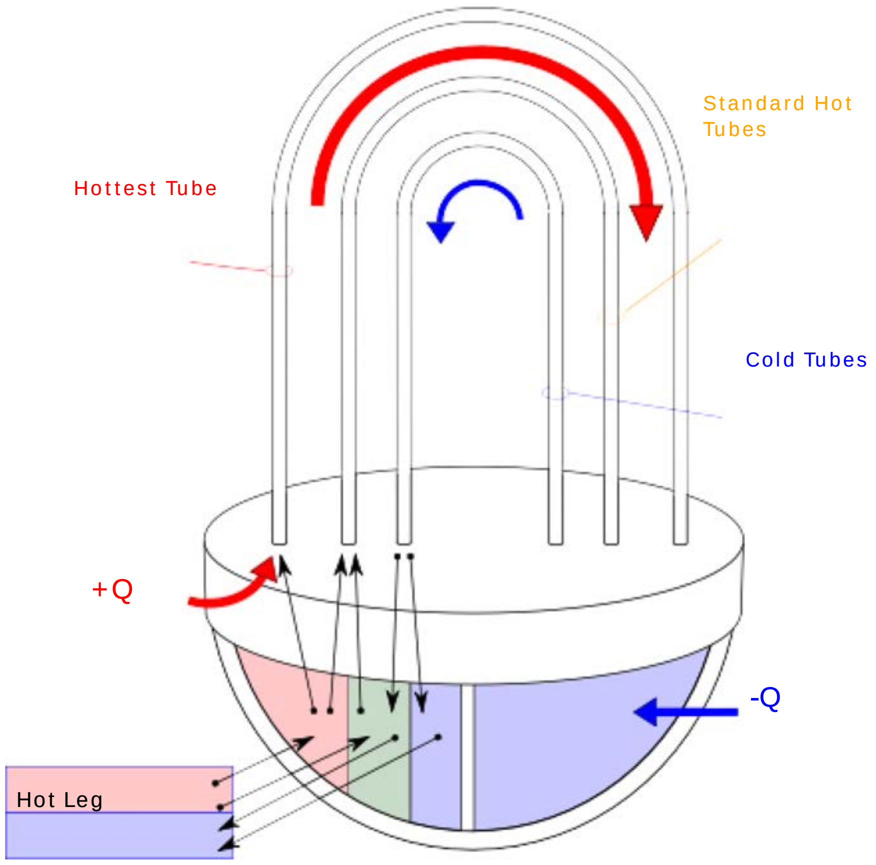
- SNL generated Combustion Engineering deck
  - based on previous RELAP and MELCOR decks
- Areas of focus

- Addition of Hottest Tube
- Estimation of tube temperature profile
- Secondary relief valve opening criteria
  - ➢ Valve sticks open after n cycles
  - ➢ Valve sticks open upon high T cycling
  - ➢ Valve sticks open as far as it has opened
  - ➢ Operator opens valve to depressurize secondary

- Results

Addition of Hottest Tube

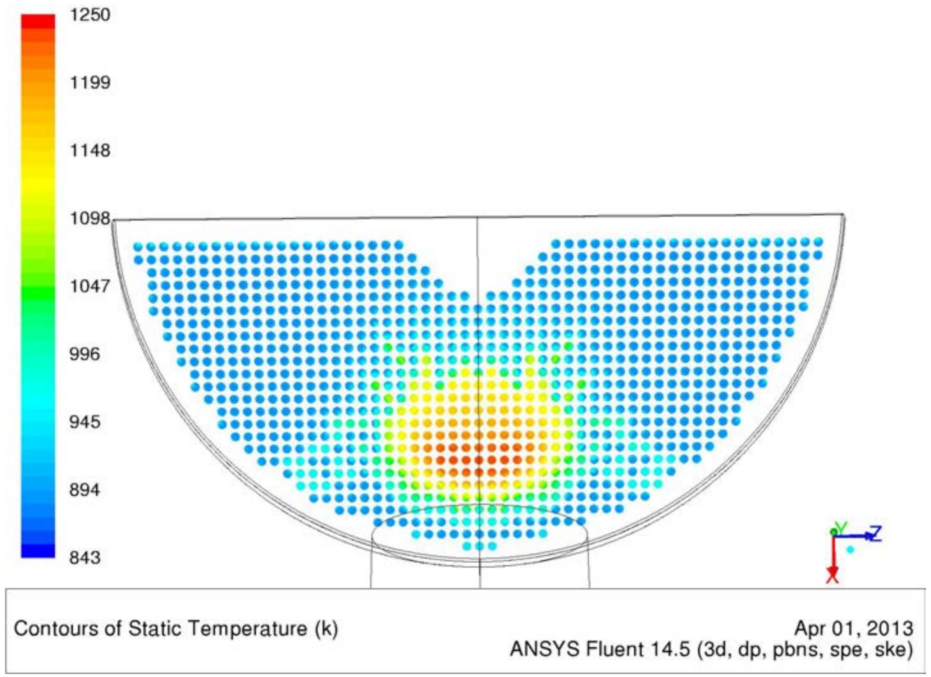
- Hotter tubes fail earlier
  - ➢ Need to capture peak T
- Calc based on CFD results
  - ➢ Provides “anchor” to estimated temperature field
- A few methods were tried
  - ➢ Postprocessing method
    - o Reasonable results
  - ➢ Side-calculation method
    - o CF link to main calc
    - o similar to SGAP RELAP calculations
    - o issues with CF copy
  - ➢ Heat addition method
    - o similar to more recent calculations
    - o Used this method



Heat addition

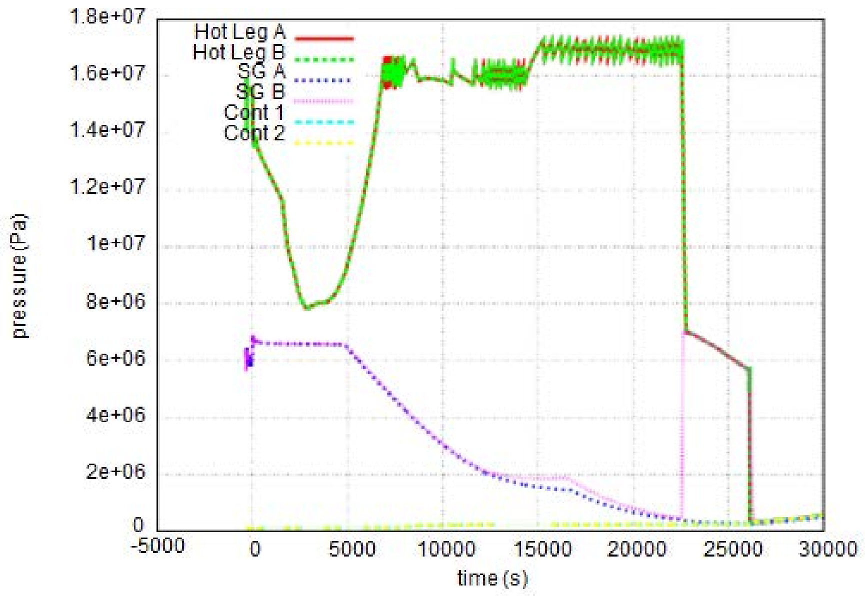
Approximation of Tube Temperature Field

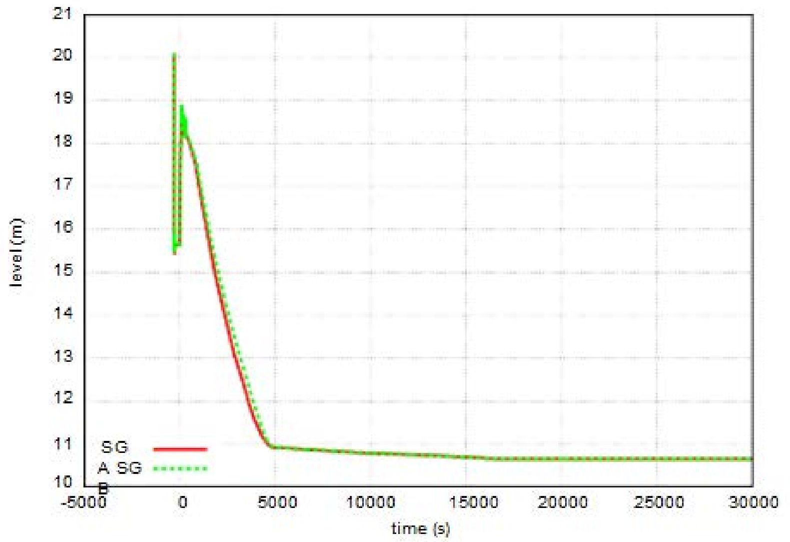
- MELCOR calculates T(t) for 2 sets of hot tubes (hot and avg)
  - ➢ Apply CFD-calculated profile to get time-dependent T field for tube failure calculations
- CFD SG temperature field approximated using parabolic shape
  - ➢ plume assumed circular, symmetric, and to reach top of SG tubes
  - ➢ radius at bottom set to match the number of in-plume tubes calculated by CFD
- Can map flaws to T(t) field
  - ➢ Lower T on average than current methods
  - ➢ Still bounds CFD results
- SG Inlet Plume T Profile



Results

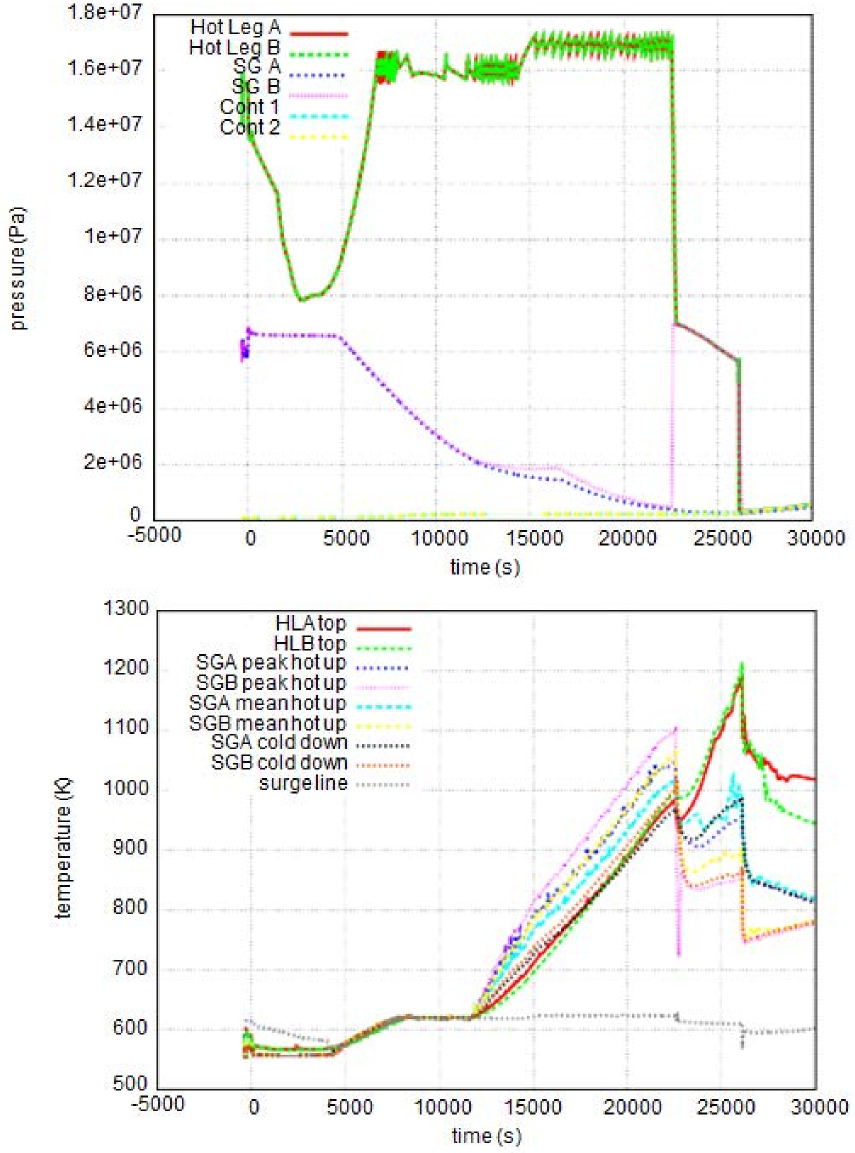
- Most results for STSBO
  - ➢ Early TDAFW failure
  - ➢ Small SG secondary leak
- Features
  - ➢ Primary P ↓ when SG wet
  - ➢ Secondary P high until SG dry
  - ➢ Primary P increases as SG dries out
  - ➢ Primary P limited by SRV setpoint
  - ➢ Structure T ↑ when Zr oxidizes
  - ➢ P Primary and SG equilibrate upon tube rupture
  - ➢ P remains high enough to rupture HL





Results

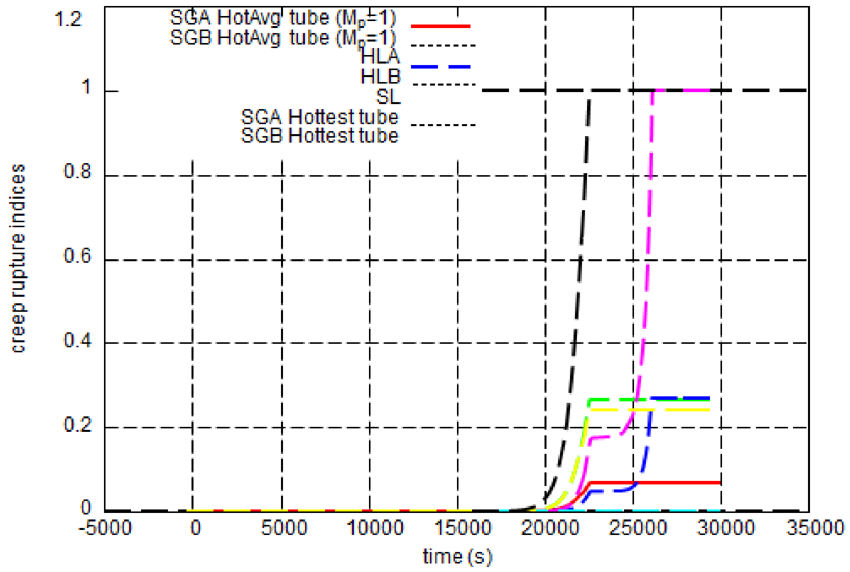
- Most results for STSBO
  - ➢ Early TDAFW failure
  - ➢ Small SG secondary leak
- Features
  - ➢ Primary P ↓ when SG wet
  - ➢ Secondary P high until SG dry
  - ➢ Primary P increases as SG dries out
  - ➢ Primary P limited by SRV setpoint
  - ➢ Structure T ↑ when Zr oxidizes
  - ➢ P Primary and SG equilibrate upon tube rupture
  - ➢ P remains high enough to rupture HL

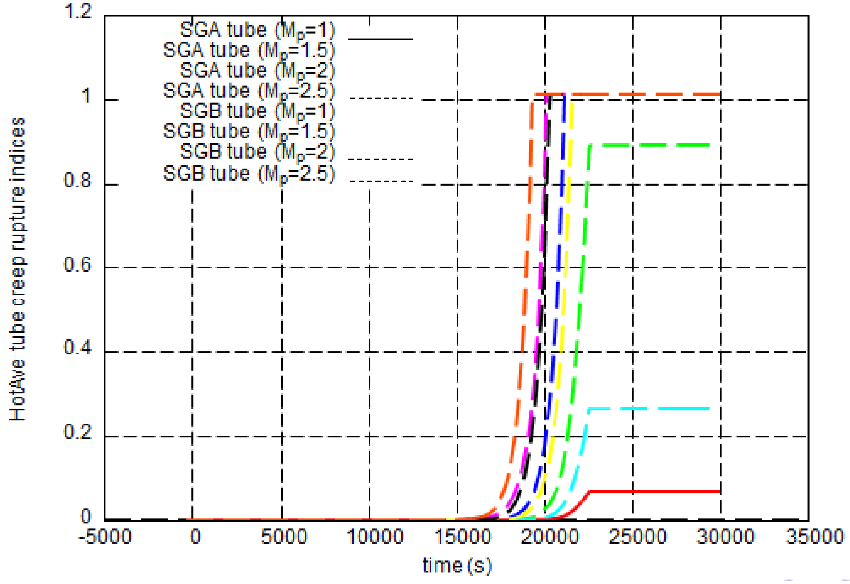


Rupture Behavior

Rupture also calculated with MELCOR as screening check

- Rupture behavior for STSBO case
  - Loop B hottest tubes fail first
  - Loop B Hot legs fail some time after
  - Other components did not reach failure conditions
  - Only unflawed average-T and hottest-T plume tubes considered
- Also considered tubes with different stress multipliers
  - Represents different flaws
  - Can be used to characterize rupture behavior after the fact

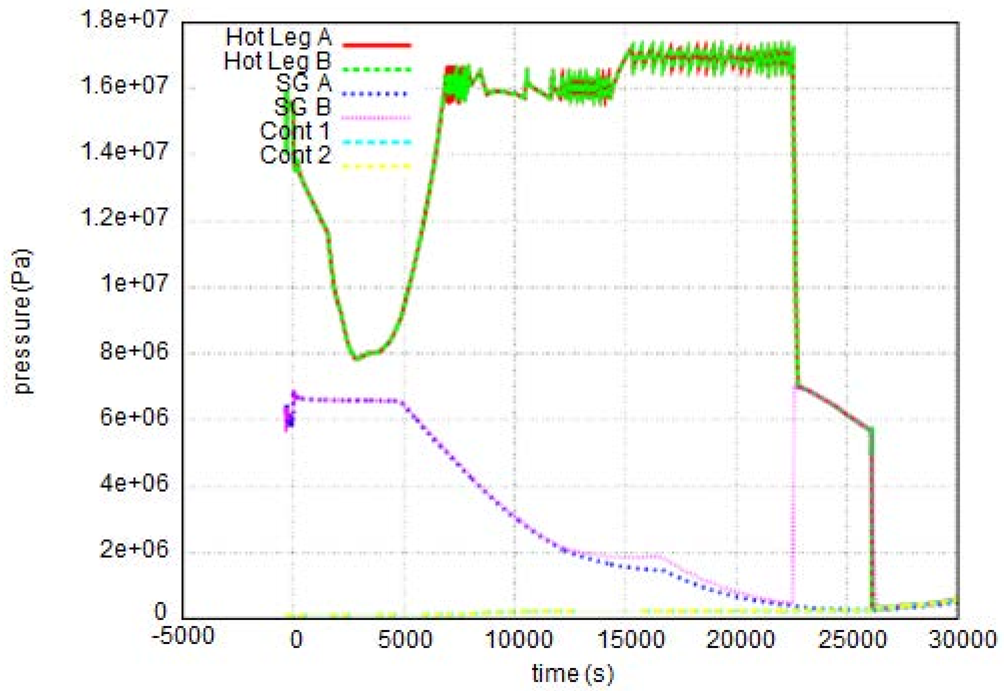


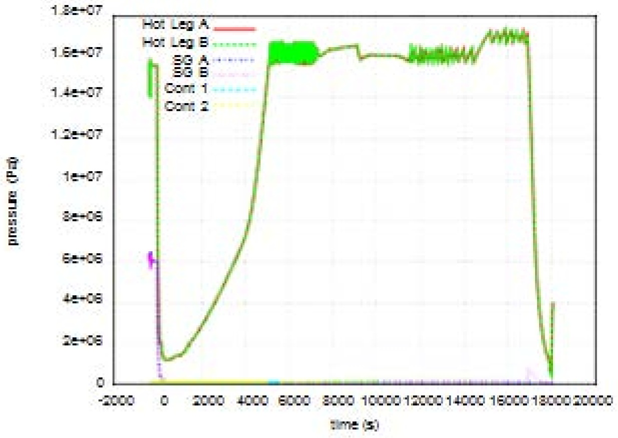


Some sequence variations

A few of the variations that were run

- Compare operator of opening secondary relief valves
  - Depressurization to enable low-P SG injection
  - Much faster and stronger depressurization of primary system
  - Earlier SG tube rupture
- Compare long term station blackout
  - TDAFW assumed to operate for 4 hours
  - Similar, but 4 hour shifted, behavior





Some sequence variations

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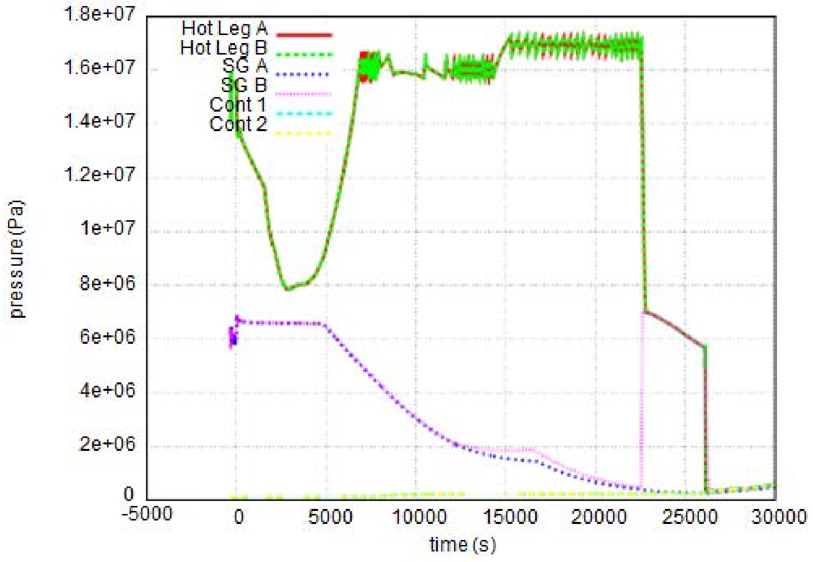
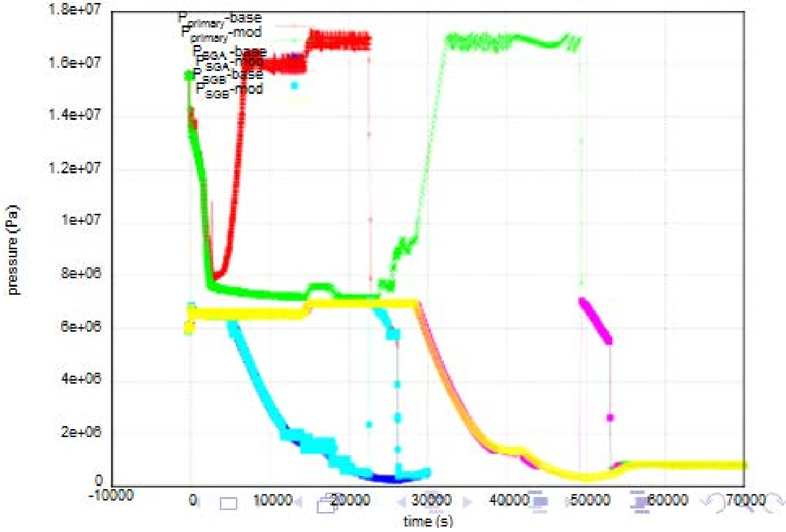




Summary/Conclusions for MELCOR CE Analyses

- CE CSGTR behavior differs from Westinghouse behavior
  - ➢ This is caused by less mixing of hot gases before reaching SG tubes
    - o Smaller hot leg length-to-diameter ratio
    - o Shallower SG inlet plena, especially for some replacement SGs
  - ➢ Because of this CE SGs are thermally stressed relative to HL in comparison to W plan
    - o Greater likelihood that tubes will fail earlier relative to Westinghouse plants
- RCS components failed first in most simulations

Severe Accident-Induced Steam Generator Tube Rupture (SGTR)

RCS Modeling and Failure Prediction

Dr. Raj Iyengar, RES/DE

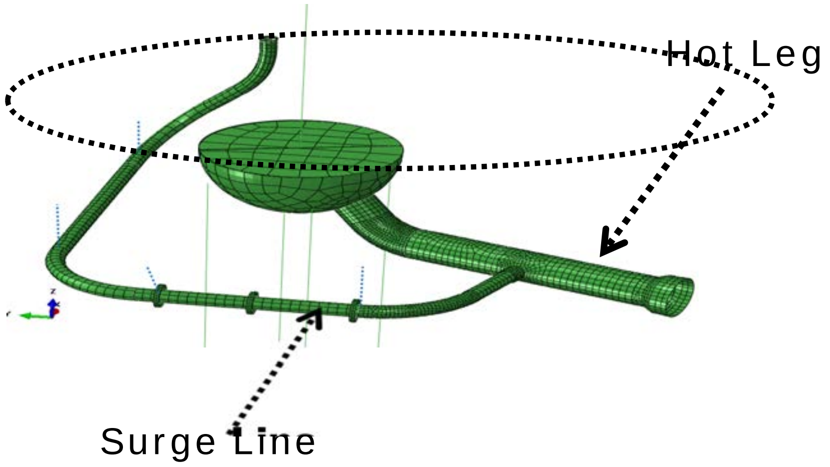
Consequential Steam Generator Tube Rupture (C-SGTR) Subcommittee Briefing

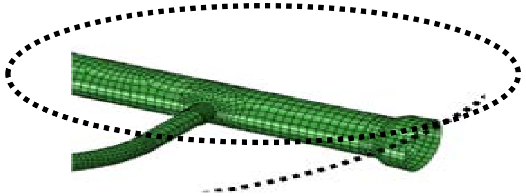
April 7, 2015

Hot Leg-Surge Line 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 Model

- Total strain = elastic + plastic + creep
- Creep Law – time and rate-dependent
- Plasticity Law – rate-independent
    - piecewise-linear stress-stain input from experimental data

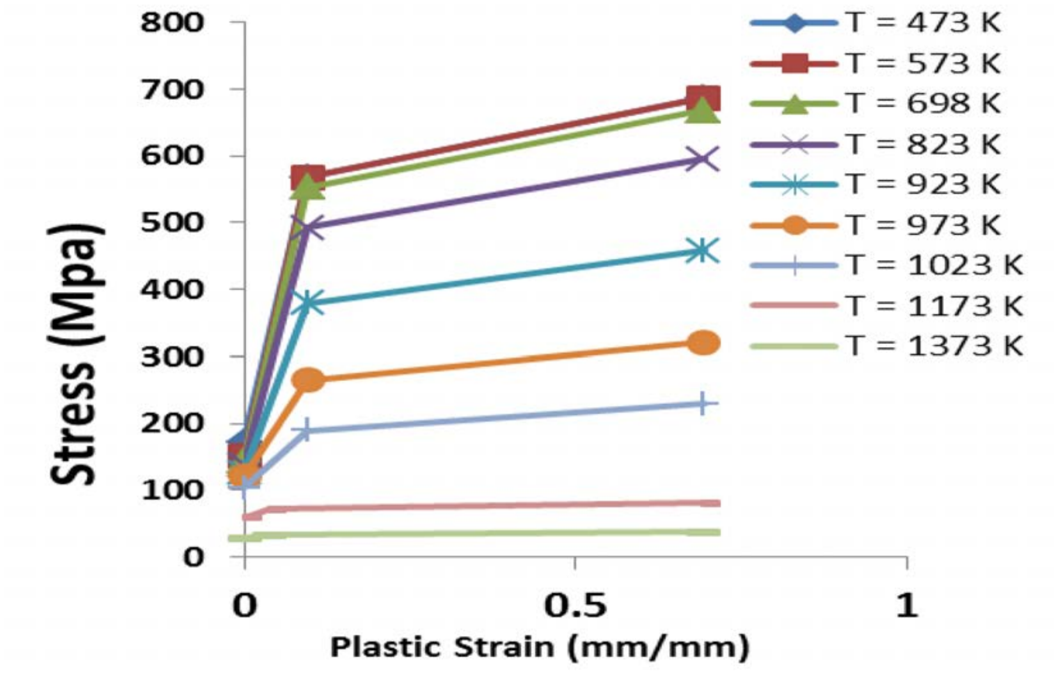
Analysis Procedure

- HL/SL structural temperatures for initial conditions (steady-state condition)
- Spatial-temporal distribution of gas temperatures (RELAP) - Use time-dependent heat transfer coefficient
  - Assume upper and lower temperature split
- Heat transfer coefficient spatially adjusted 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
- Apply gravity load and temporal distribution of internal pressure
- High temperature (up to 1373K) material properties (316 SS)
- Run a thermal-mechanical simulation

Material Model – SS316 SS (HL)

United States Nuclear Regulatory Commission
Protecting People and the Environment

Combined rate-independent plasticity and power-law creep models

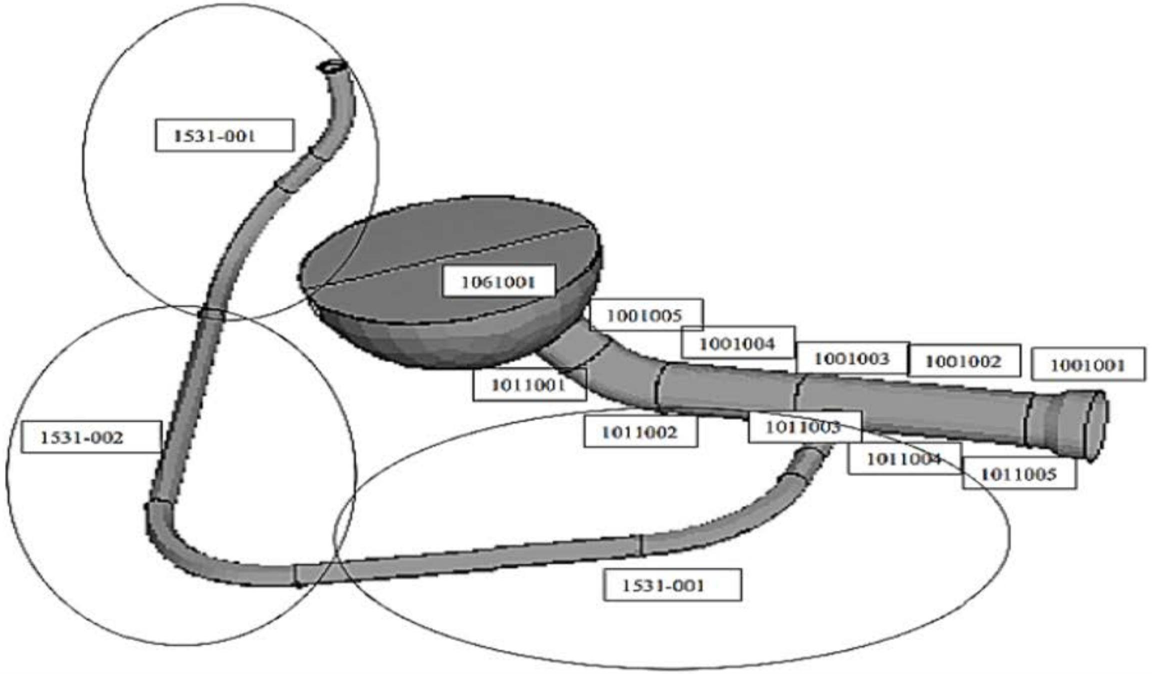


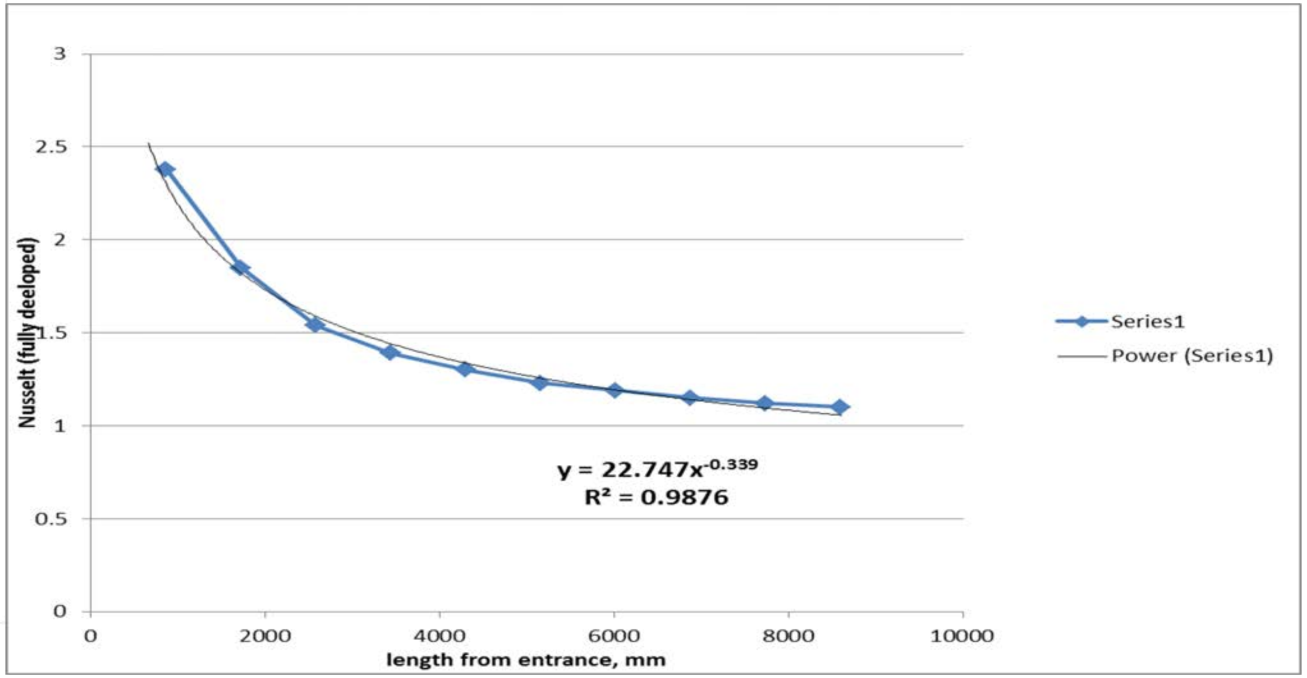
Temperature dependent stress strain curves for 316 stainless steel

$$\dot{\bar{\varepsilon}}^{cr} = A\tilde{q}^{n}t^{m}$$

A	m	T (K)
1.23E-43	4	294.1
1.62E-36	9.78	748
5.68E-36	9.97	773
1.15E-32	9.06	798
1.47E-29	8.2	823
1.02E-28	8.2	848
6.31E-28	8.2	873
4.49E-27	8.18	898
2.88E-26	8.16	923
5.68E-24	7.42	948
8.56E-22	6.72	973
3.34E-20	6.25	998
1.30E-18	5.77	1023
1.56E-17	5.89	1100
9.39E-17	5.85	1150
2.21E-15	5.85	1250
8.91E-15	5.85	1300
1.08E-15	5.85	1400

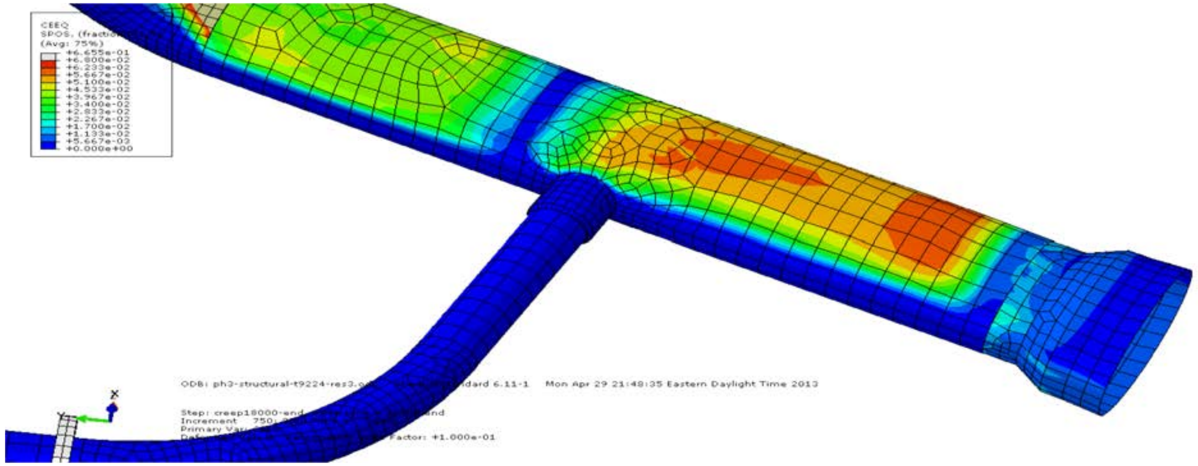
Thermal heat flux model





Spatial adjustment for heat transfer coefficient (based on information in in NUREG-1922).

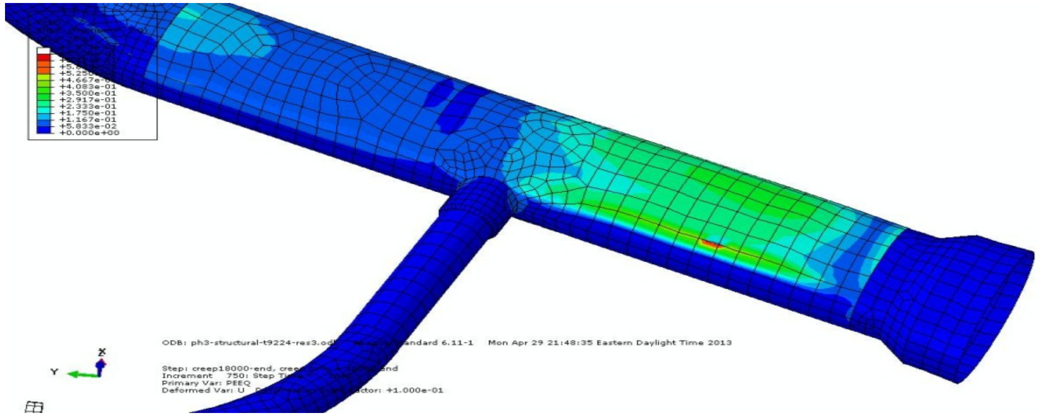
Creep and Plastic Strains



Accumulated Creep Strain

t = 12302 seconds

Accumulated Plastic Strain



Damage Prediction

Damage at any material point determined using

Larsen-Miller Parameter (LMP)

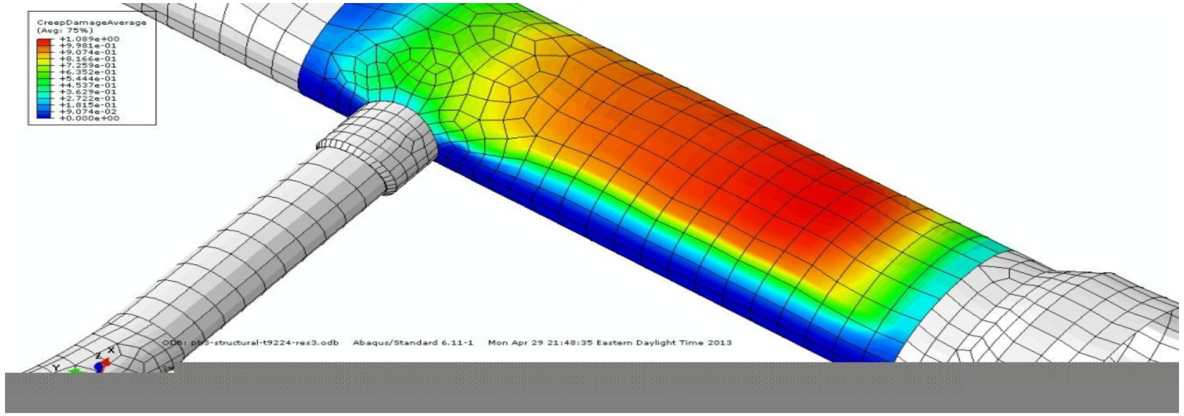
$$LMP = A*\text{Log10}(\sigma) + B$$

σ - effective stress; T – temperature

Time to rupture

$$t_r = 10(LMP/T - C)$$

A, B, and C - constants



Failure time - 12302 seconds

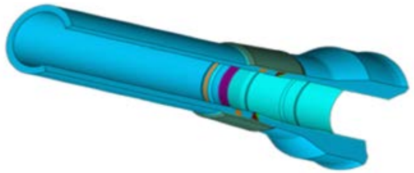
Damage is averaged through thickness to determine failure time.

Failure Behavior of RCS Components

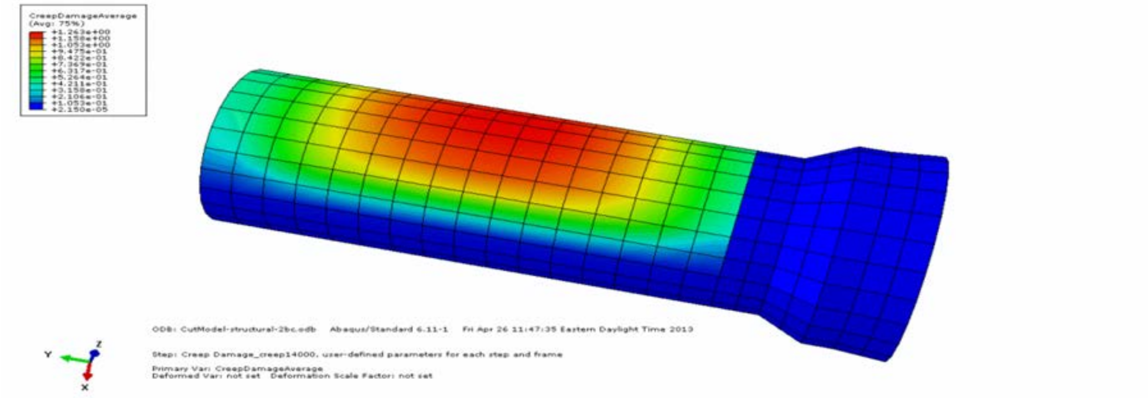
- System-level model simulations
  - computationally intensive
  - poses issues with convergence
  - Not well-suited for understanding sensitivities to input parameters
- Failure location in the hot-leg region predicted by the system model
- A sub-model of hot leg and reactor pressure vessel nozzle used 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

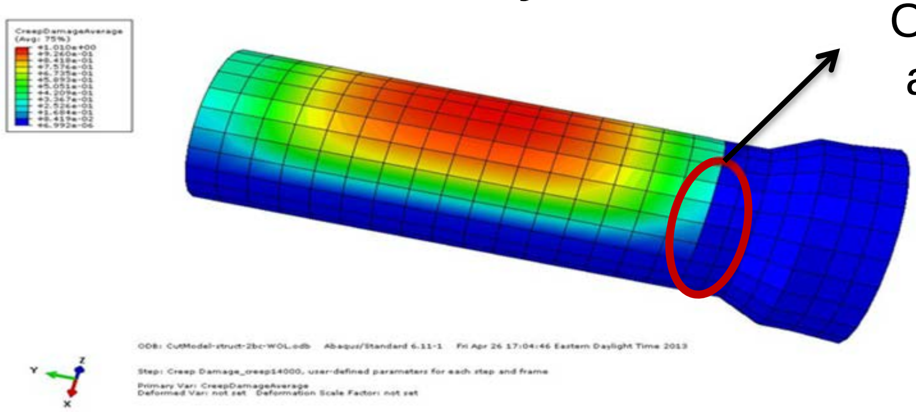


No Weld Overlay



Weld Overlay

Overlay added



t_r = 12428 secs

t_r = 12500 secs

Failure time increases by 72 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	12302
Hot leg only	Creep and Plasticity; Spatially adjustment of HTC	No	12428
	Creep and Plasticity; Spatially adjustment of HTC	Yes	12500
	Creep only; Spatial adjustment of HTC	No	12140
	Creep and Plasticity; HTC not adjusted spatially	No	12560

Using the simplified procedure in C-SGTR Calculator, predicted failure times for hot leg:

1) *12800 s (5th percentile), 13000 s (50th), 13100 s (95th) (considering one hot-leg model)*
2) *12700 s (5th), 12900 s (50th), 13000 s (95th) (considering a mode of four hot leg and a surge line)*

Failure Behavior of RCS Components

Summary

- Hot-leg model yields similar failure location and time compared with the system model (Westinghouse).
- 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.
- Most importantly, the complex analyses predict lower failure times compared with the simplified analysis used in the C-SGTR calculator.

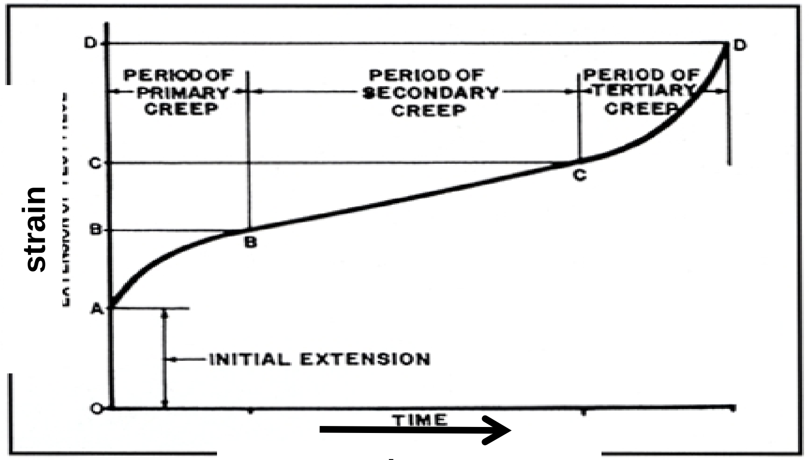
BACKGROUND SLIDES

Background
Pre-C-SGTR

- NUREG-1570 – Issued March 1998
- Steam Generator Action Plan – Initiated November 16, 2000 (Revised May 11, 2001)
- RIL 09-03 Issued – August 21, 2009 – Closes SGAP Item 3.5
- SGAP Closed – December 3, 2009
- User Need NRR 2010-005 Issued December 16, 2009
  a) T-H Analyses of CE Plants and Evaluation of Incore Instrument Tube Failures
  b) Updated SG Flaw Distributions and Enhanced RCS Structural Analyses
  c) Development of Guidance and Tools for Future Risk Assessments
  d) Preparation of a document compiling and summarizing the key research.

High Temperature Behavior of RCS and SGT Materials

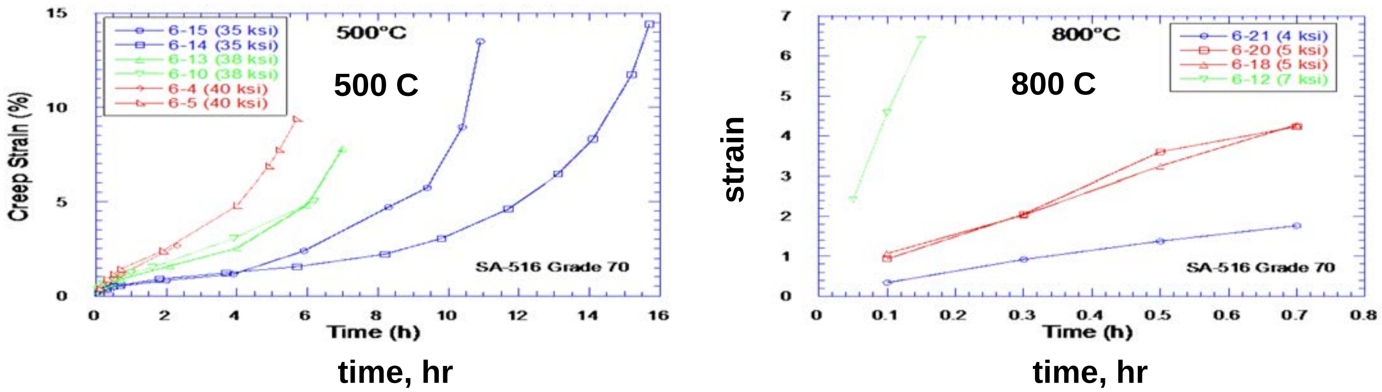
High Temperature Deformation
Metals



http://www.nationalboard.org

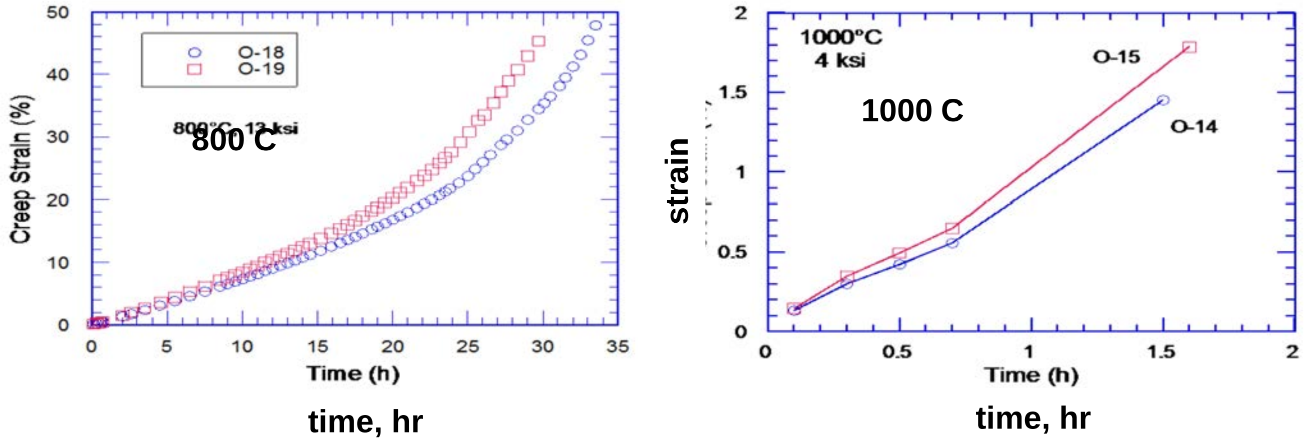
- Primary (transient) creep
  - Increase in creep resistance
  - Low temperature
- Secondary creep
  - constant creep rate
  - power law behavior
- Tertiary creep
  - rapid increase in creep rate
  - damage accumulation

Carbon Steel G70 **ANL Tests**



Increase in stress or temperature accelerates creep deformation and damage

SS316

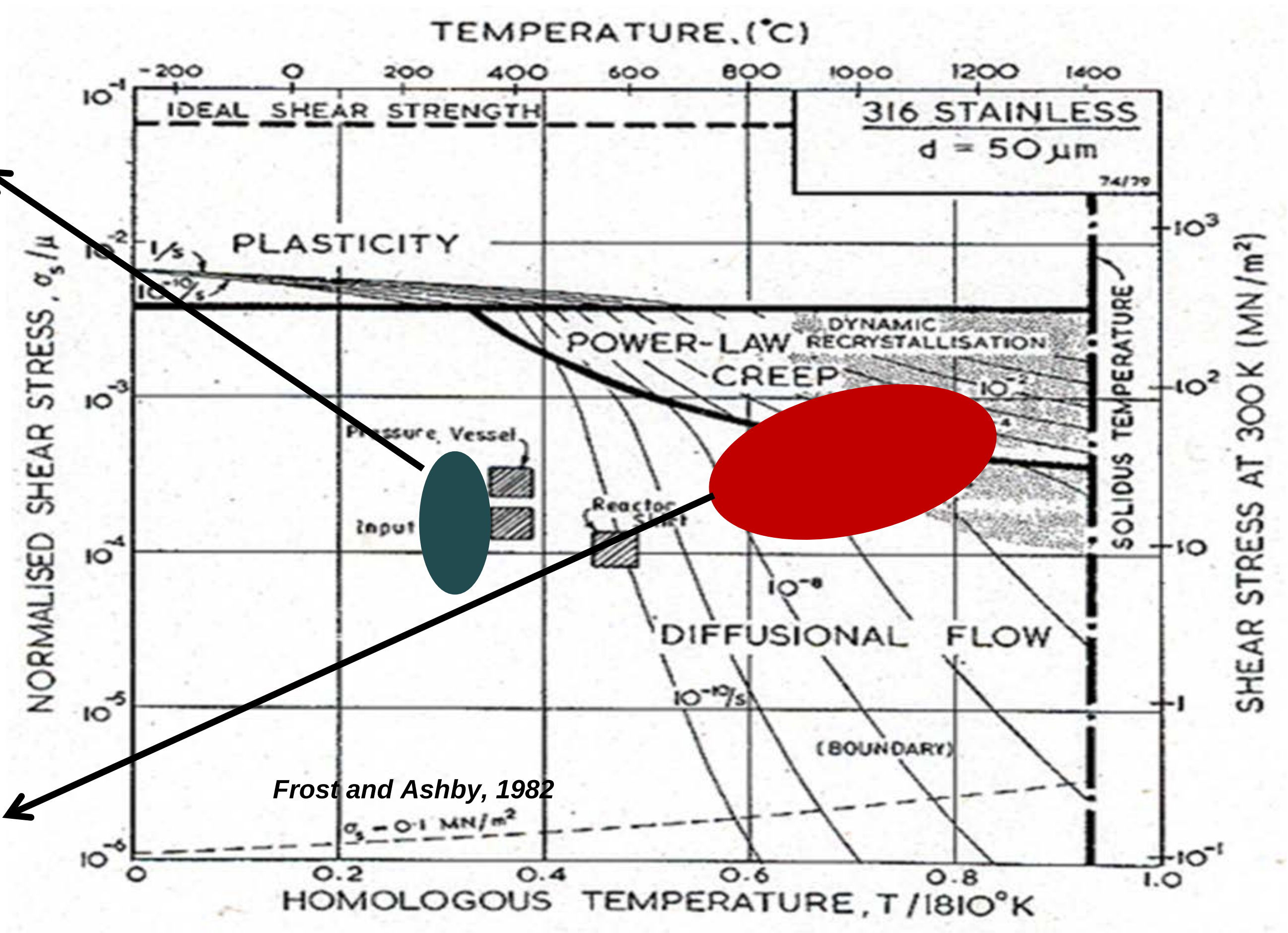


Deformation Map
316 Stainless Steel

temperature

Normal Operating Conditions

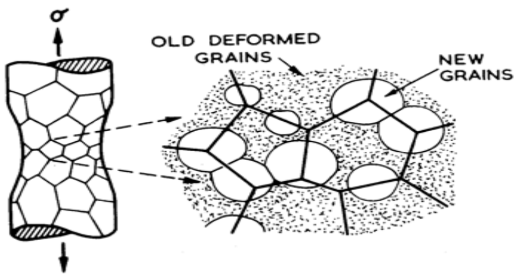
shear stress



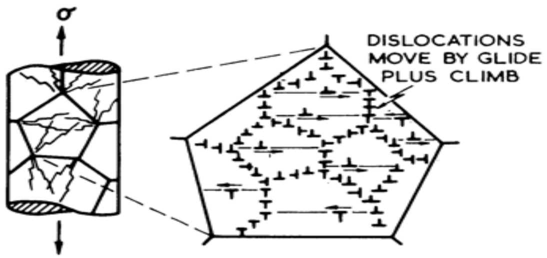
Frost and Ashby, 1982

Severe Accident Conditions

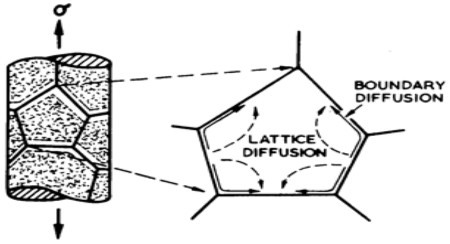
Dynamic Recrystallization



Deformation by Dislocation Glide and Climb



Deformation by Diffusion (Lattice and Grain Boundary)



- Transient effects influence extent and appearance of various regions
- Diffusion effects in stainless/alloy steels at high temperatures not well understood

High Temperature Failure

Ductility-driven Damage
(dislocation motion)

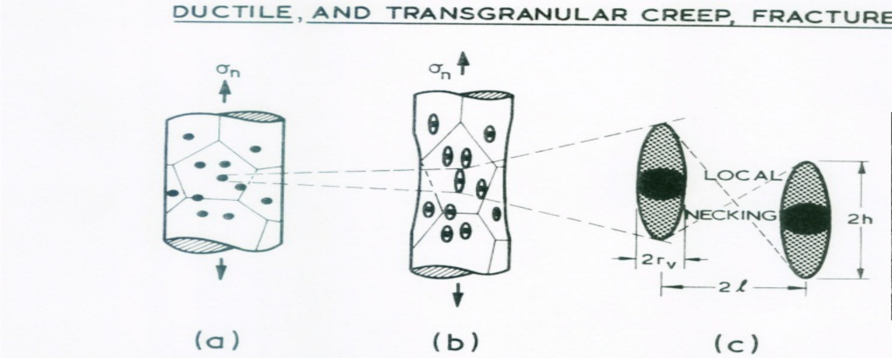
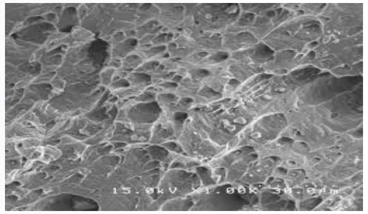


Fig. 3. (a) Ductile fracture and transgranular creep fracture requires either that holes pre-exist or that they nucleate at inclusions which concentrate stress. (b) The holes elongate as the specimen is extended. (c) They link, causing fracture, when their length, $2h$, is about equal to their separation $(2l - 2r_v)$.



Diffusion-driven Damage
(grain boundary sliding)

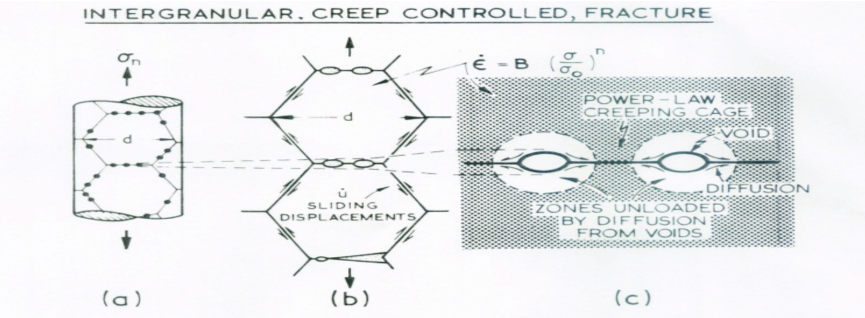
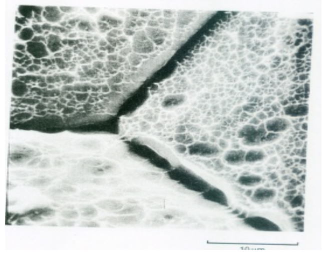


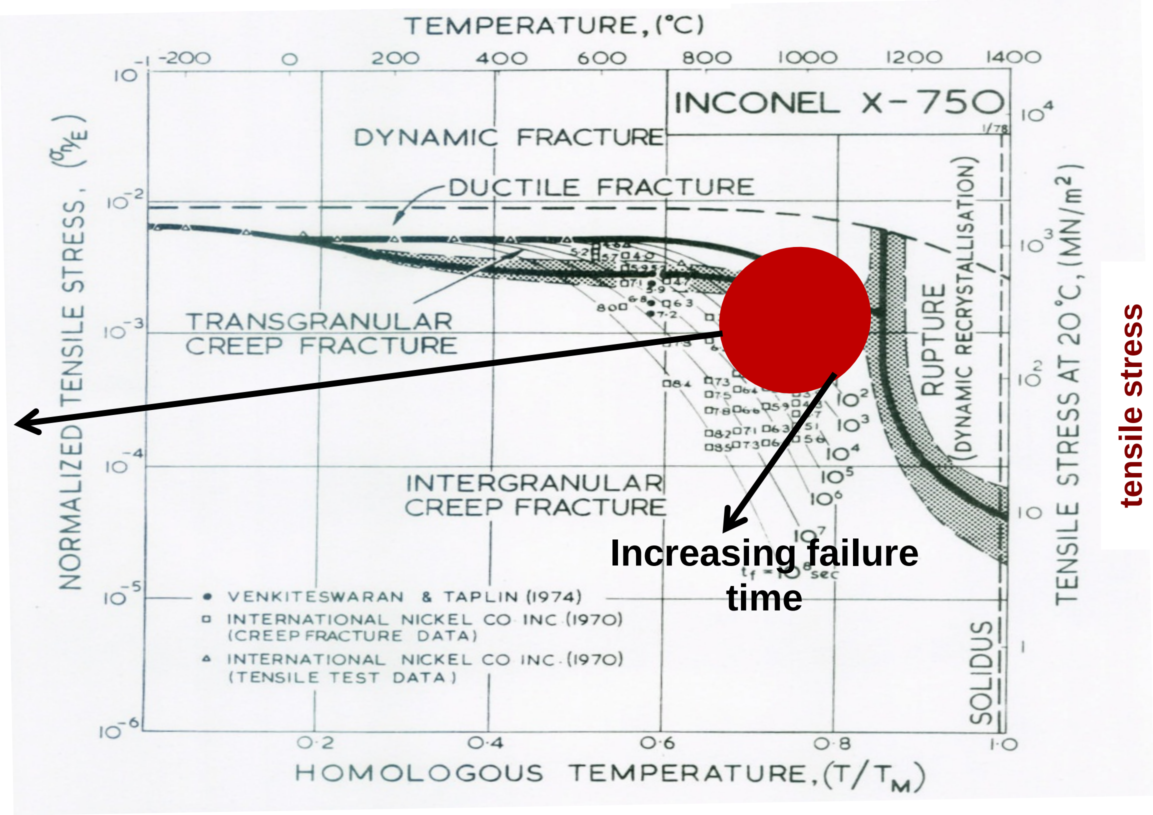
Fig. 4. (a) Grain boundary sliding stimulates the nucleation of grain and (b) boundary voids. (c) The voids grow by diffusion, but the diffusion fields of neighbouring voids do not, in general, overlap, so that each void is contained within a cage of power-law creeping material.



Ashby, Gandhi, and Taplin, 1979

Severe accident conditions

temperature

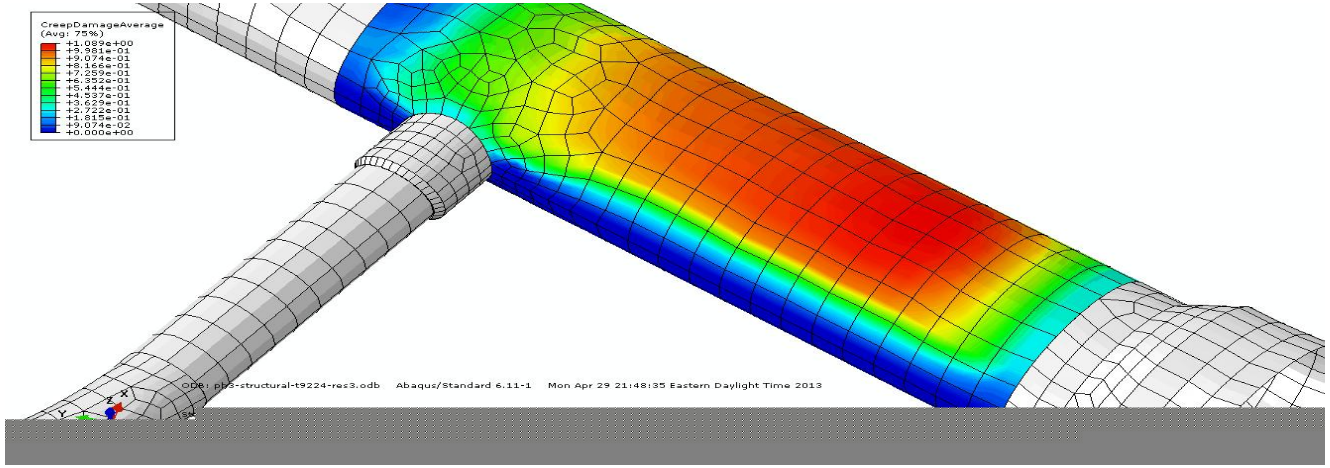


tensile stress

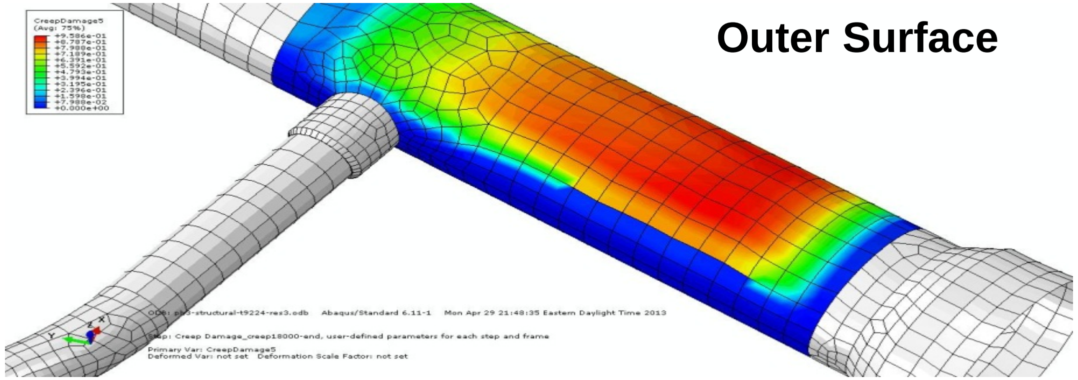
Increasing failure time

Transient effects and multi-axial stress state influence extent and appearance of various regions

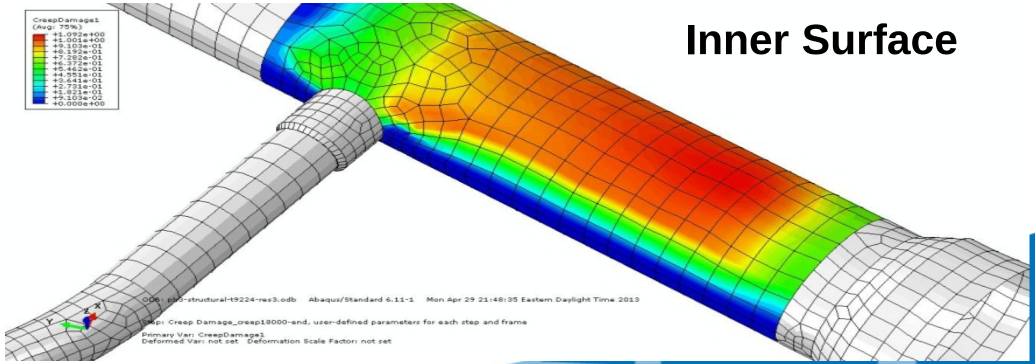
Damage
at t= 9222 + 3080 seconds



Through Thickness Damage



Outer Surface



Inner Surface

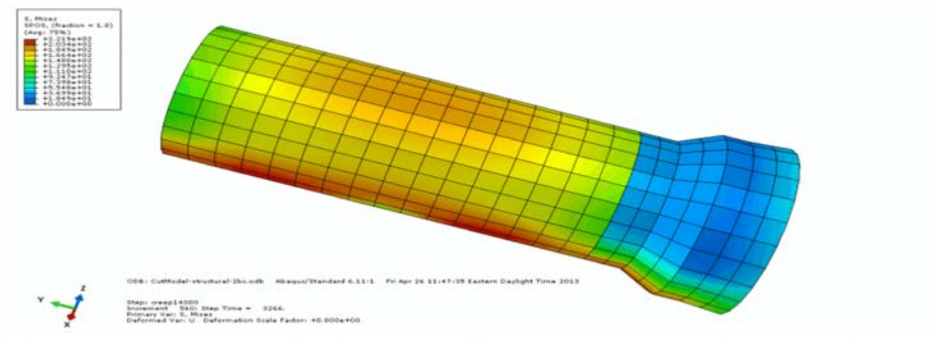
Effective Stress Distribution

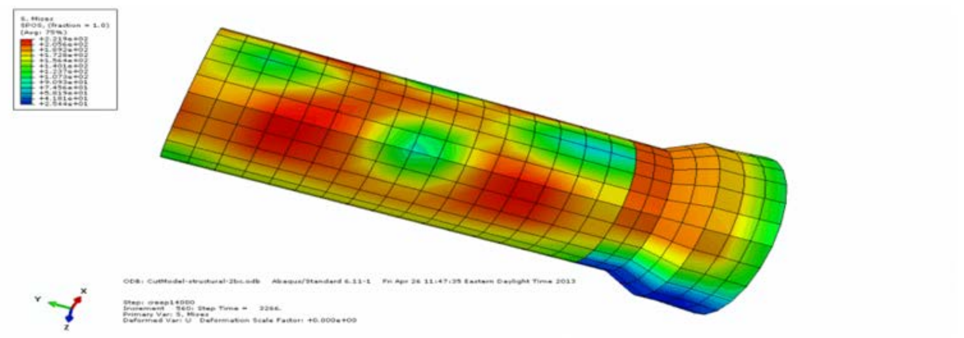
No Weld Overlay (t = 9222 + 3206 secs)

Hot side

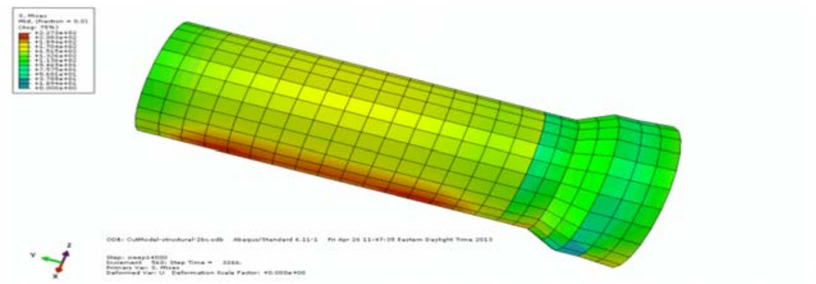
Colder side

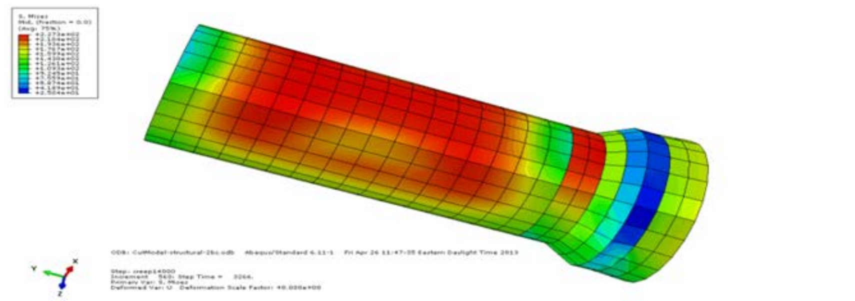
As creep progresses
stresses relax
redistribution occurs



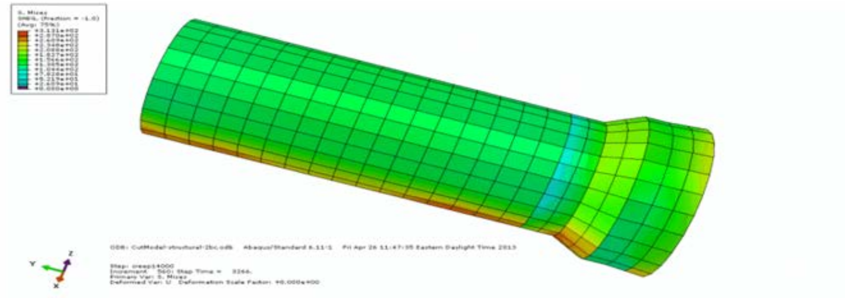


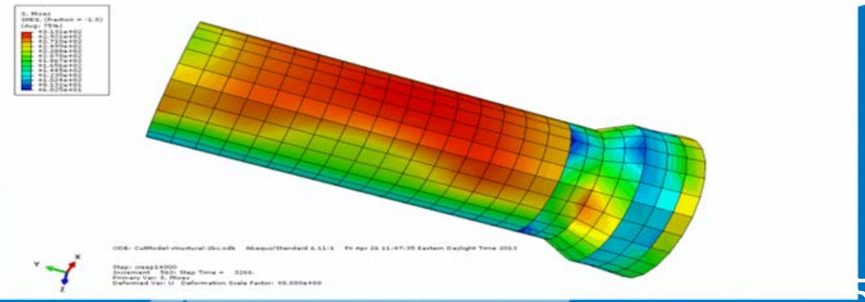
Outer Surface





Mid-thickness





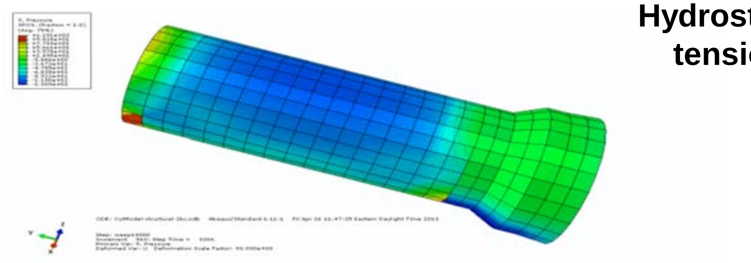
Inner Surface

Hydrostatic Stress Distribution

No Weld Overlay (t = 9222 + 3206 secs)

Hot side

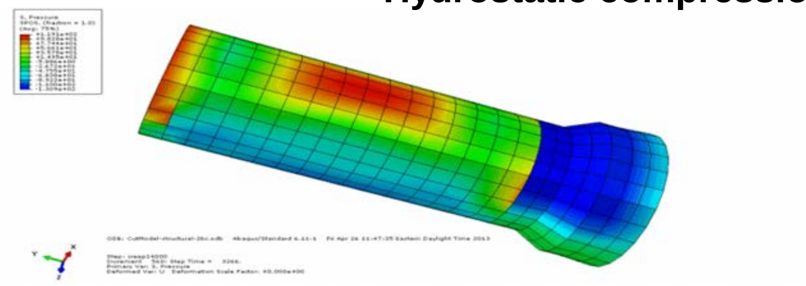
Hydrostatic tension



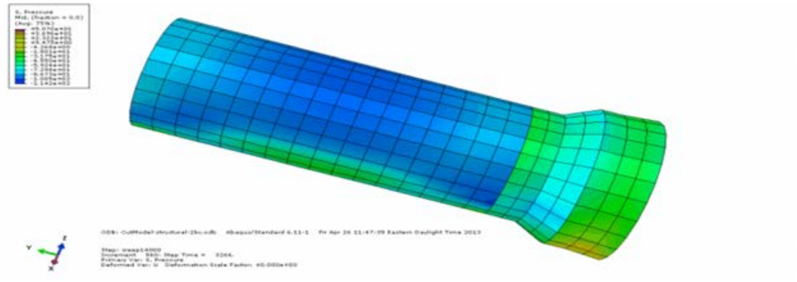
As creep progresses stresses relax redistribution occurs

Colder side

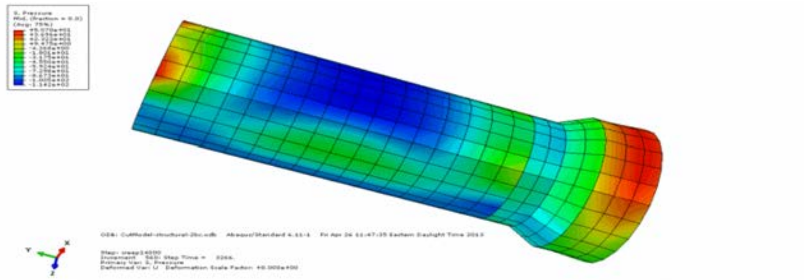
Hydrostatic compression

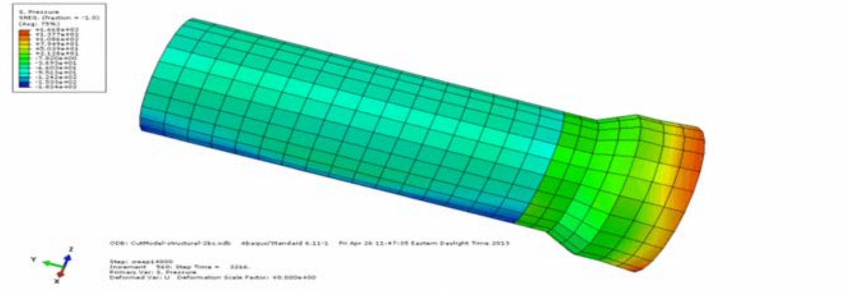


Outer Surface

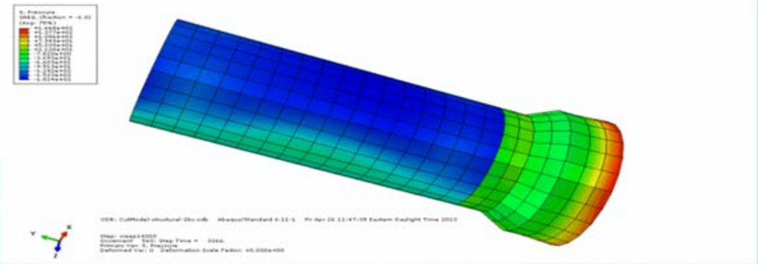


Mid-thickness





Inner Surface



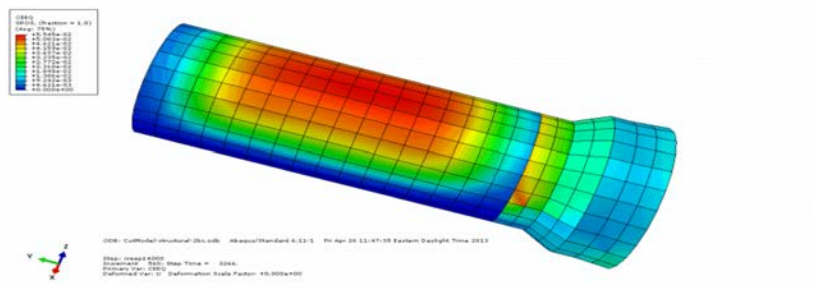
Hydrostatic Stress Distribution

No Weld Overlay (t = 9222 + 3206 secs)

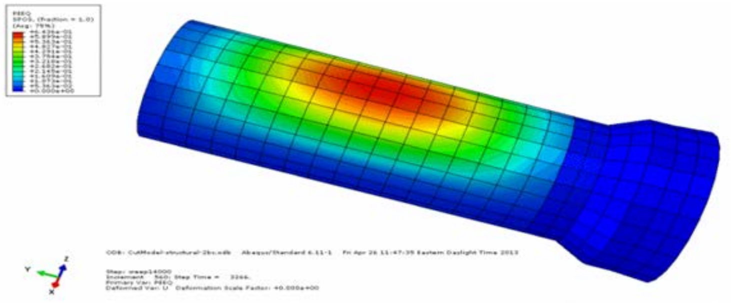
Accumulated Creep Strain

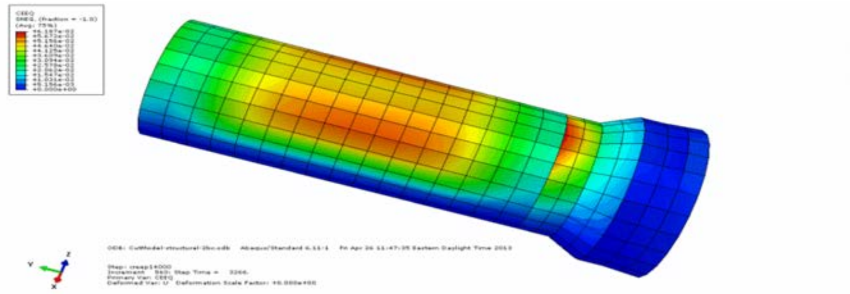
Accumulated Plastic Strain

Higher creep & plastic strains near the outer surface on the hot side of pipe

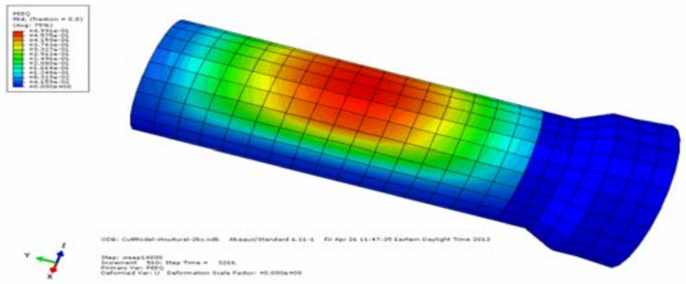


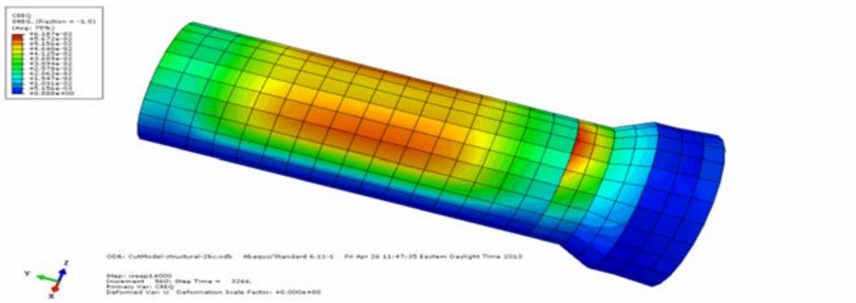
Outer Surface



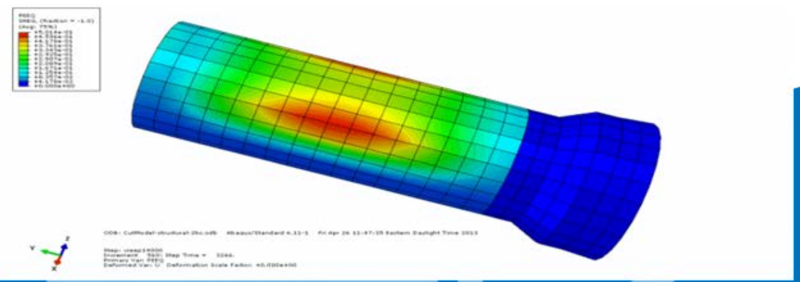


Mid-thickness





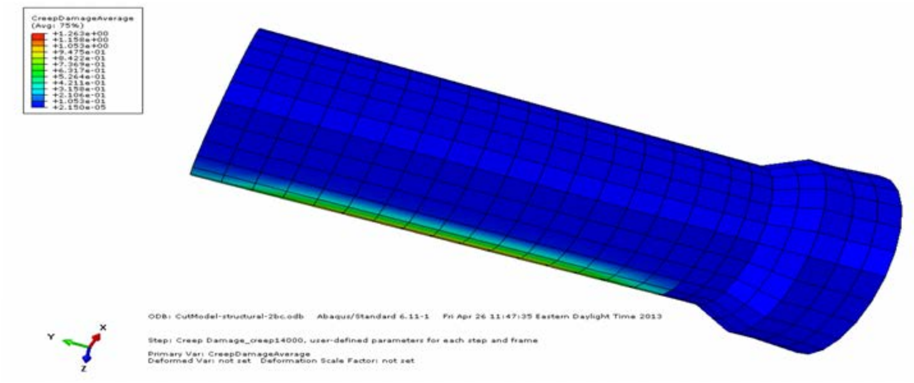
Inner Surface



Damage Accumulation

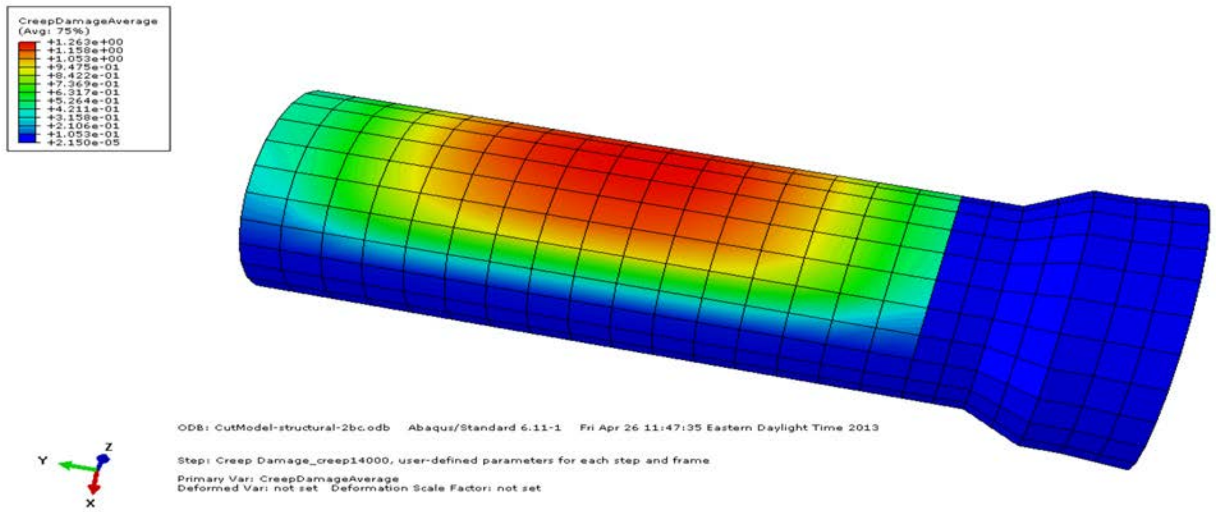
No Weld Overlay (t = 9222 + 3206 secs)

Cold Side



Red - Through Thickness Damage > 1
Blue – Little or No Damage

Hot Side

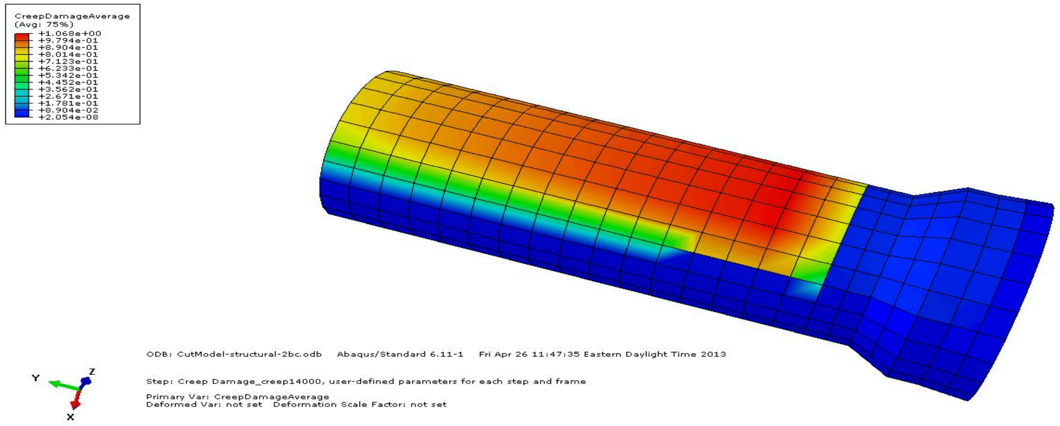


Damage Accumulation

No Weld Overlay

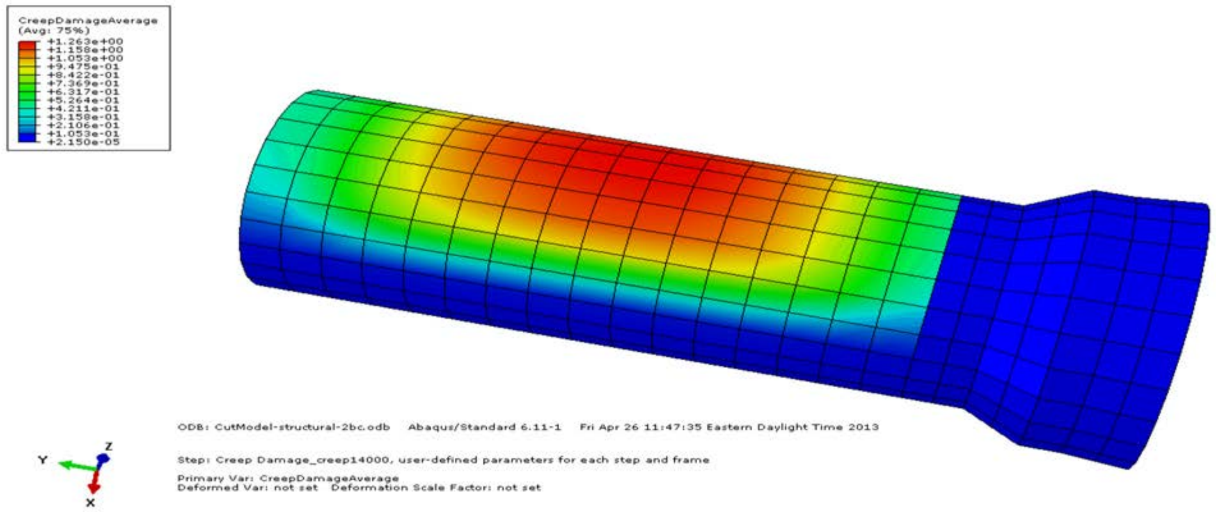
Red - Through Thickness Damage > 1
Blue – Little or No Damage

EPRI Failure Criterion



T= 4010 secs

ANL Failure Criterion

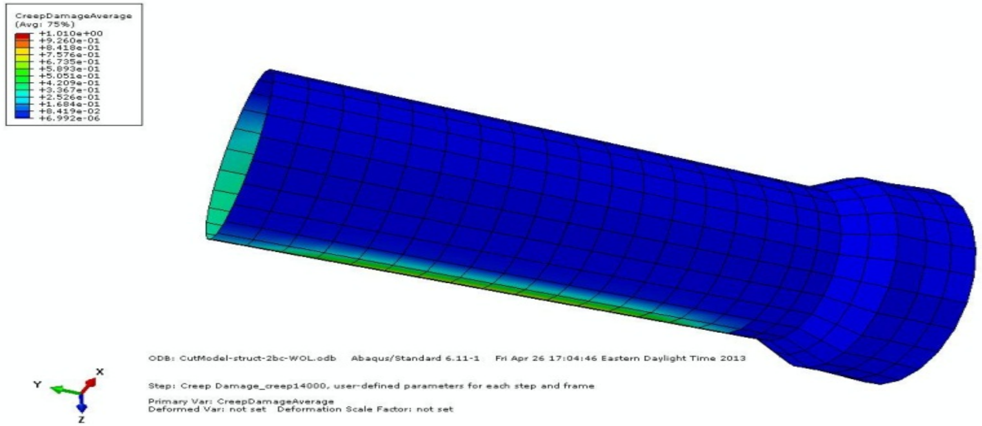


T= 3206 secs

Damage Accumulation

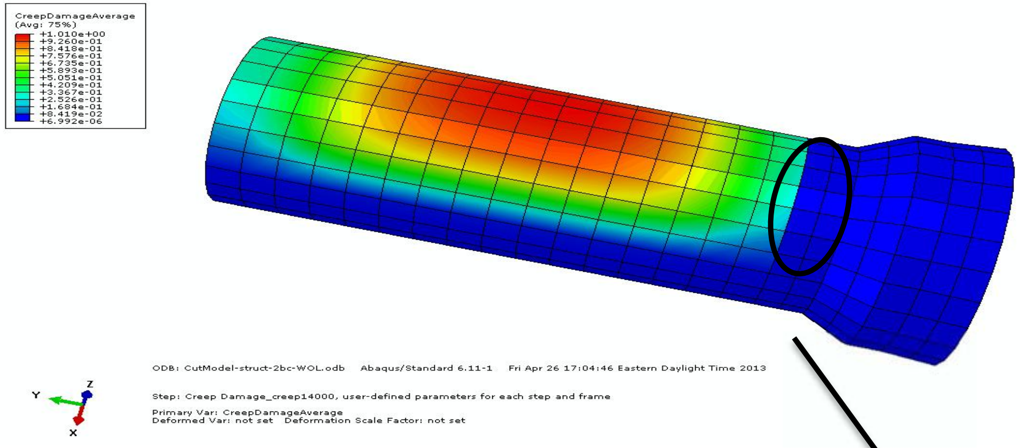
Weld Overlay (t = 9222 + 3278 secs)

Cold Side



Red - Through Thickness Damage > 1
Blue – Little or No Damage

Hot Side



Overlay

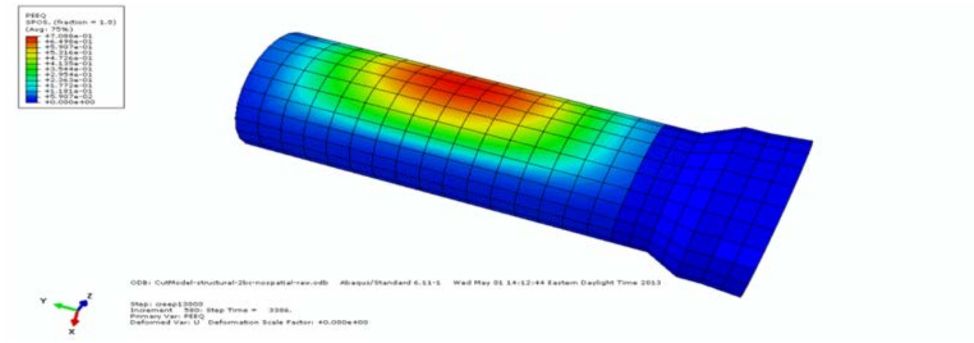
Failure time increases by 72 seconds with weld overlay.

Failure location does not change.

Damage Accumulation

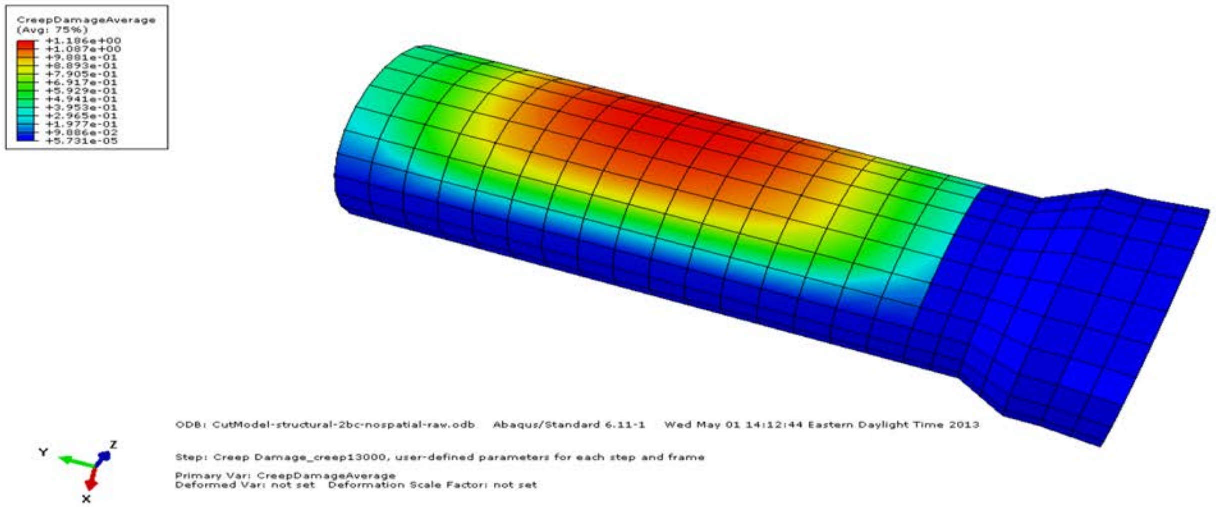
No Weld Overlay (t = 9222 + 3386 secs)

Accumulated Creep Strain

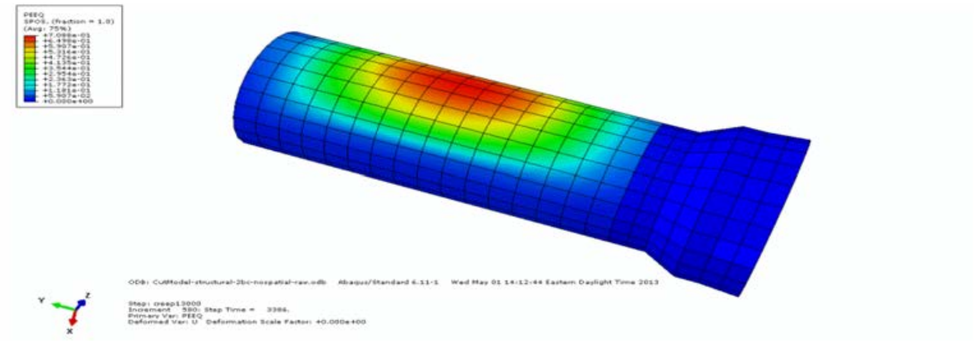


No spatial adjustment of heat-transfer coefficient

Red - Through Thickness Damage > 1
Blue – Little or No Damage



Accumulated Plastic Strain



Failure time increases by 180 seconds relative to spatially adjusted HT coefficients

Damage Accumulation

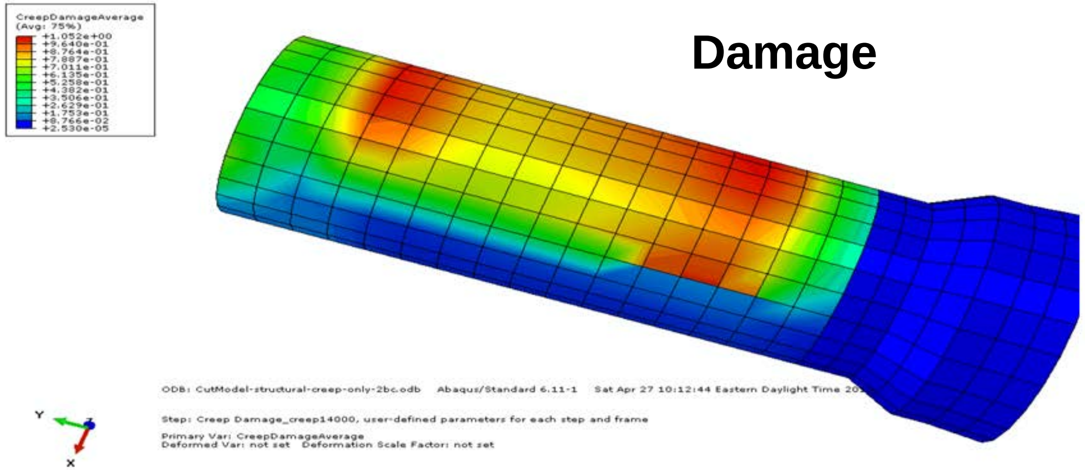
No Weld Overlay

Creep Only
(t = 9222 + 2918 secs)

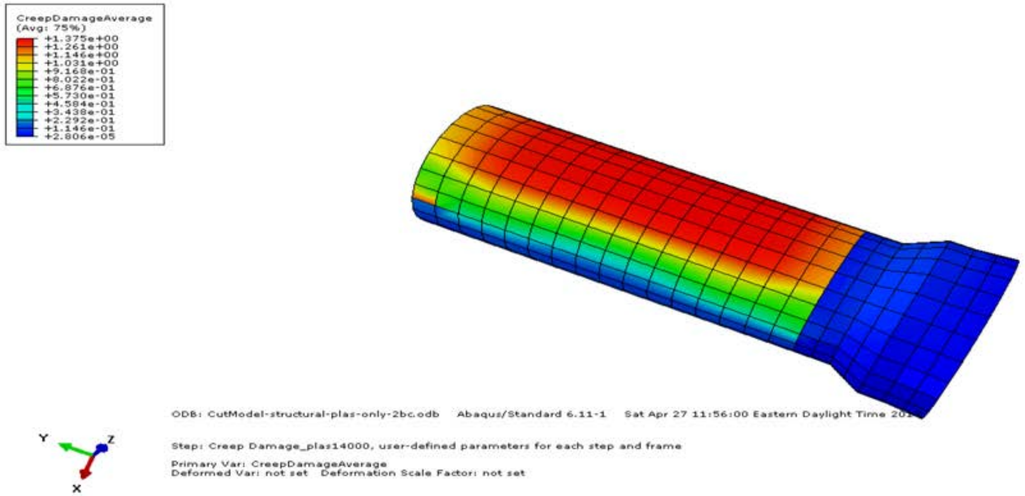
Red - Through Thickness Damage > 1
Blue – Little or No Damage

Plasticity Only
(t = 9222 + 3802 secs)

Damage



Damage



Plastic Strain

Shorter failure time when creep is the only dominant inelastic mode of deformation.
Primarily caused by the stress redistribution due to creep –
aided by the difference in temperature transients on the hot and cold side

Severe Accident-Induced Steam Generator Tube Rupture (SGTR)

SG tube Flaw Distribution Characterization

Dr. Ali Azarm, Innovative Engineering & Safety Solutions (IESS)
Mica Baquera, RES/DE

Consequential Steam Generator Tube Rupture (C-SGTR) Subcommittee Briefing

April 7, 2015

SG tube flaws

- Objective
  - To update the previous study on flaw statistics and provide current statistics sufficient to generate flaw samples for C-SGTR analysis (input to the C-SGTR calculator)
- Background
  - The previous work on estimating SG tube flaw distributions was for 600 MA tube materials (NUREG/CR-6521 Gorman Report) and for cracks only using data that existed pre 1995.
  - These (U-tube) SGs are replaced with those having new SG tube materials (Thermally Treated Alloy 600 and 690).
  - Use of the information from previous studies could not be justified

SG tube flaws (Cont'd)

- Flaw data for Thermally Treated Inconel 600 and 690 (600TT and 690TT) were collected from selected in-service inspection reports available to the NRC
- Flaw data was manually extracted and compiled into a data base for further analyses
- The data were binned against operating time (*measured in Equivalent Full Power Years-EFPY*) and flaw types

SG ISI reports Used as Input

Plant Type	Inconel 600	Inconel 690
Westinghouse	Surry U1 & U2 (26 Cycles)*	DC Cook U1 & U2 (6 Cycles) McGuire U1 (2Cycles)
C-E	—	Millstone U2 (3 Cycles) St. Lucie U1 (3 Cycle) Calvert Cliffs U1 & U2 (7 cycles)

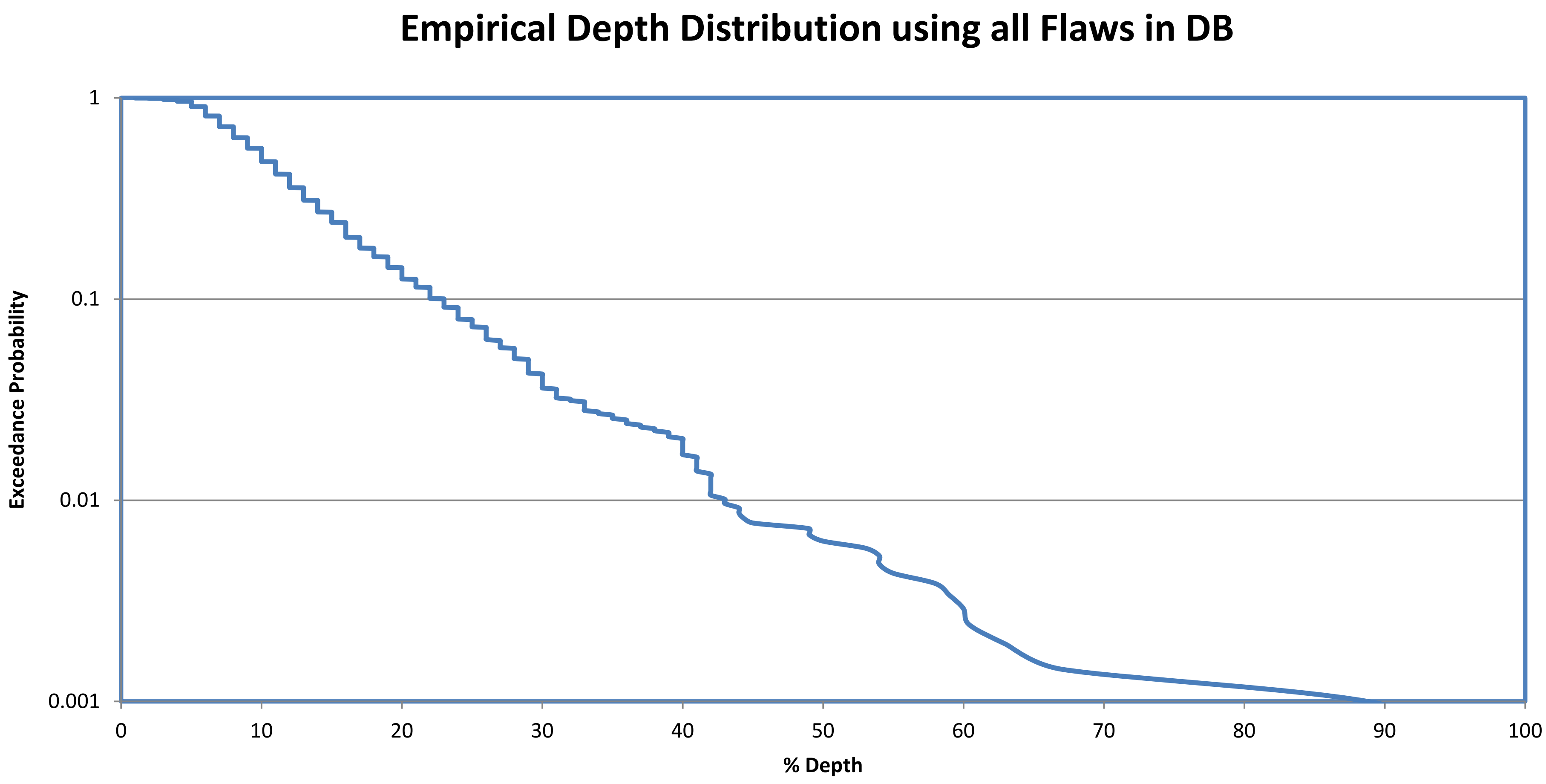
* Surry unit 2: 12 RFOs, 13 ISI reports (cycles), 268.5 EFPM (or ~22 EFPY)

Flaw Binning for Statistical Evaluation

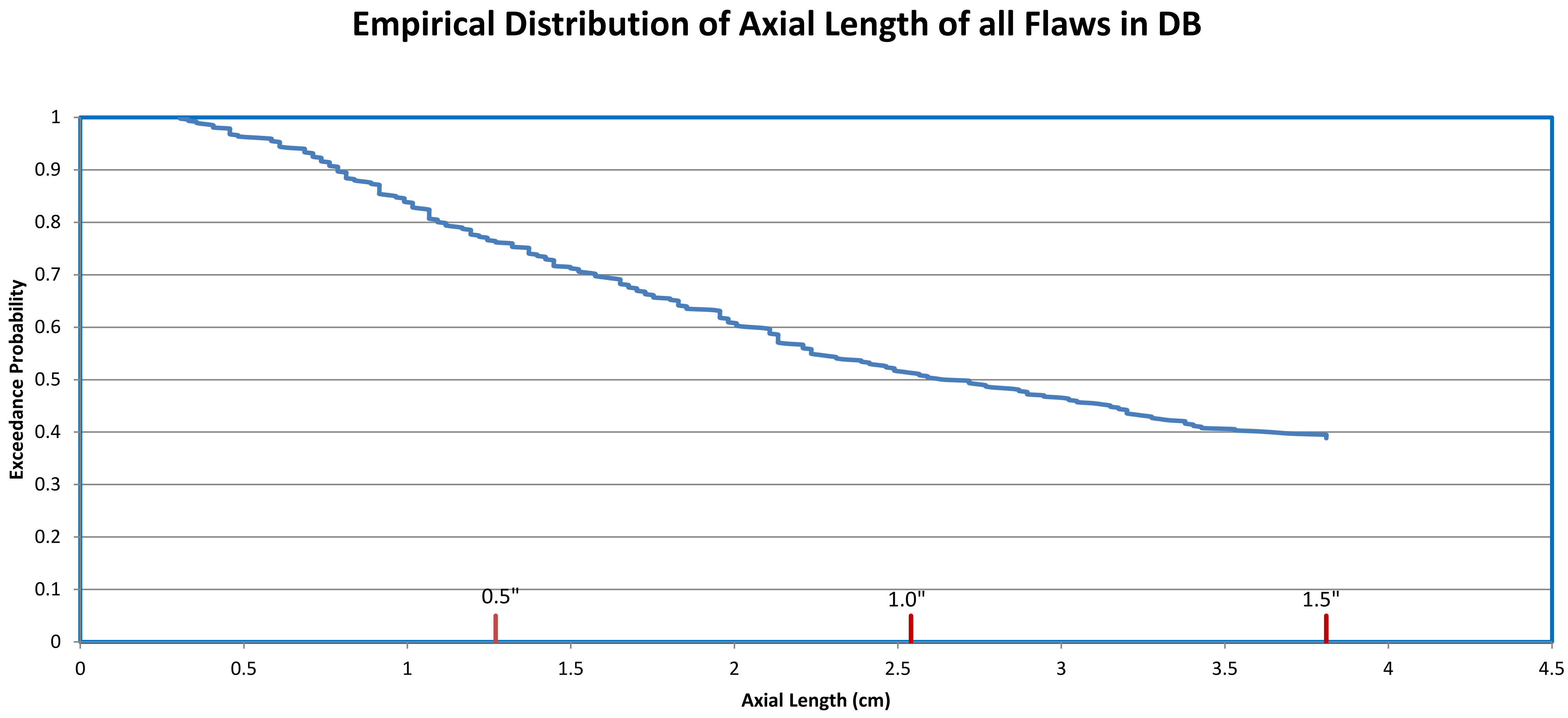
- Flaw types and names in ISI reports are different for different types of SGs and their design
- For the purpose of statistical analysis and use of calculator software, the flaws were grouped as either volumetric/wear or cracks.

Flaw Group for Statistical Analysis	Flaw Name in Database
Volumetric Flaw or Wear	Anti Vibratory Bar (AVB) wear Foreign Object Wear Free-span Wear Pit Volumetric Wear Tube Support Plate (TSP) Wear Loose Part Wear Fan Bar Wear Manufacturing Burnish Mark that changed Lattice Support Wear
Crack	Single Axial Indication (SAI) Single Circumferential Indication (SAI)

Graphical Presentation of Aggregate Flaw Data



Graphical Presentation of Aggregate Flaw Data



Flaw model (rates)

- Flaw Generation Rate per tube as a function of SG service life [measured in EFPY] for
  - Volumetric/Wear Flaw 600TT
  - Volumetric/Wear Flaw 690TT
  - Axial Cracks 600TT
  - Circumferential Cracks 600TT
  - *No Crack data was found for 690TT*

Flaw Model (sizes)

- Flaw length Distribution
  - Wear and Cracks
  - One distribution for both 600TT and 690TT
- Circumferential Flaw Length (arcs) Distribution
  - Cracks for 600TT only
- Flaw depth Distribution
  - Wear and Cracks
  - One distribution for both 600TT and 690TT

Model Parameters

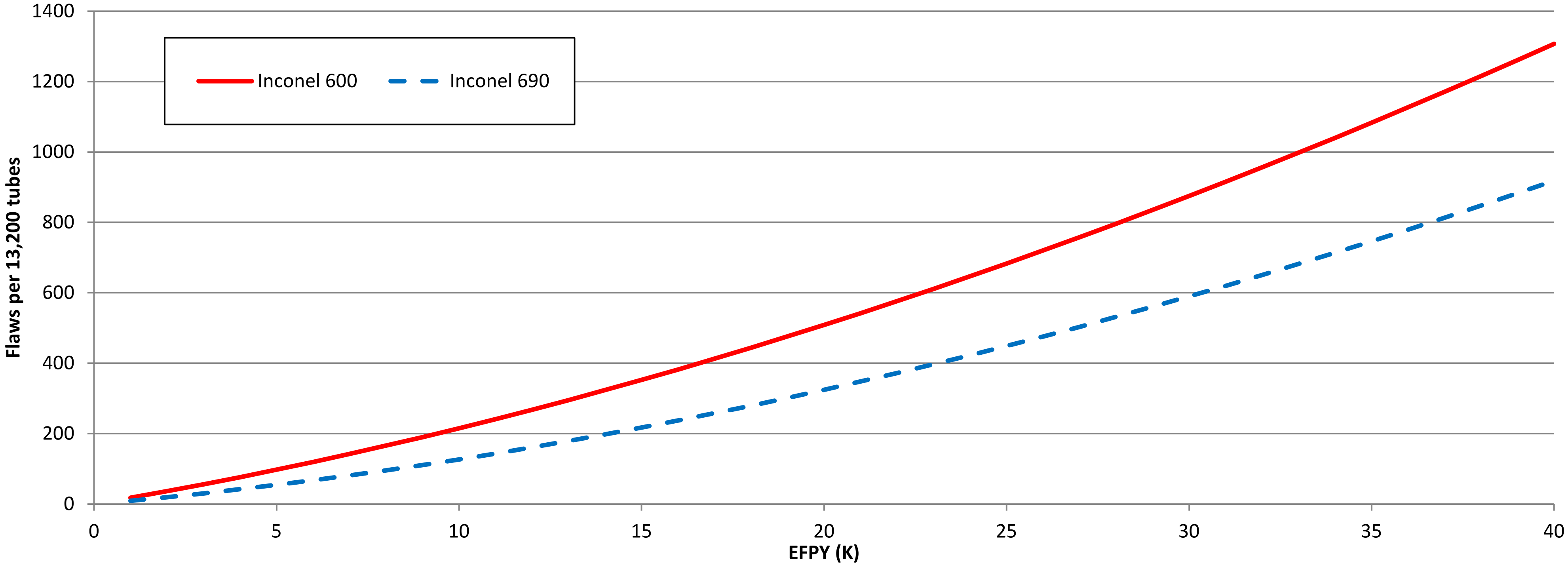
- A flaw model was developed by
  - Linearly increasing rate of volumetric flaws generation as a function of time (i.e. EFPY)
  - Linearly increasing rate of crack flaws generation as a function of EFPY
  - Gamma Distribution of flaw length
  - Gamma Distribution of flaw depth
- Statistical Estimation Approach
  - Regression using Excel routine for estimating the linearly increasing rates
  - Matching the first two moments for estimating the parameters of Gamma distributions

Summary of Statistical Results

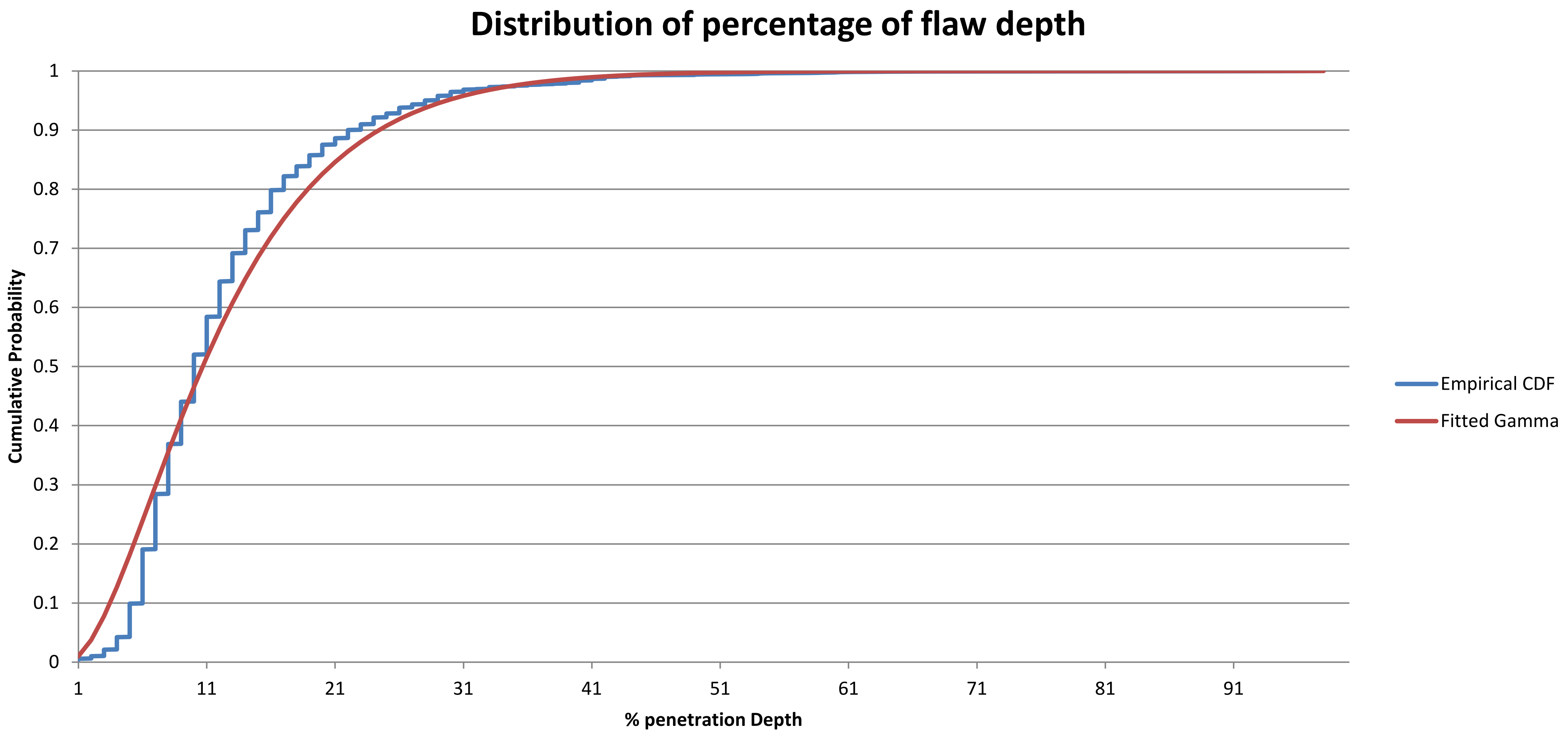
(prior to Adjustment in PRA Analysis)

Flaw Characteristics	Thermally Treated Inconel 600	Thermally Treated Inconel 690
Volumetric/Wear Flaw Rates	$h(k) = 6.4166 \cdot 10^{-5} K + 1.3236 \cdot 10^{-3}$ $\mu = 6.4166 \cdot 10^{-5}$, $\Omega = 1.3236 \cdot 10^{-3}$	$h(k) = 5.5826 \cdot 10^{-5} K + 6.8627 \cdot 10^{-4}$ $\mu = 5.5826 \cdot 10^{-5}$, $\Omega = 6.8627 \cdot 10^{-4}$
Axial Crack Flaw Rates	$K<15$, $h(k)$ = Negligible $\mu = 0.0$, $\Omega = 0.0$ $K>15$, $h(k) = 2.0 \cdot 10^{-4}$ $\mu = 0.0$, $\Omega = 2.0 \cdot 10^{-4}$	$h(k)$ = Negligible $\mu = 0.0$, $\Omega = 0.0$
Circumferential Crack Flaw Rates	$K<15$, $h(k)$ = Negligible $\mu = 0.0$, $\Omega = 0.0$ $K>15$, $h(k) = 1.0 \cdot 10^{-3}$ $\mu = 0.0$, $\Omega = 1.0 \cdot 10^{-3}$	$h(k)$ = Negligible $\mu = 0.0$, $\Omega = 0.0$
Axial Flaw length: Axial Cracks, Wear Marks, or Volumetric Flaws	Gamma(a =2.33318781, β= 2.0847)	
Circumferential Crack Angle	0.58·Gamma(a=28.6565, β=0.4187) + (1-0.58)·Gamma(a=9.5638, β=0.0670)	
Flaw Depth: Cracks, Wear, Volumetric Flaws	Gamma(a =2.0658, β=16.3274)	

Average Number of Flaws (both wear and Crack if applicable) as a Function of EFPY for 4 SGs (3300 tubes/SG)



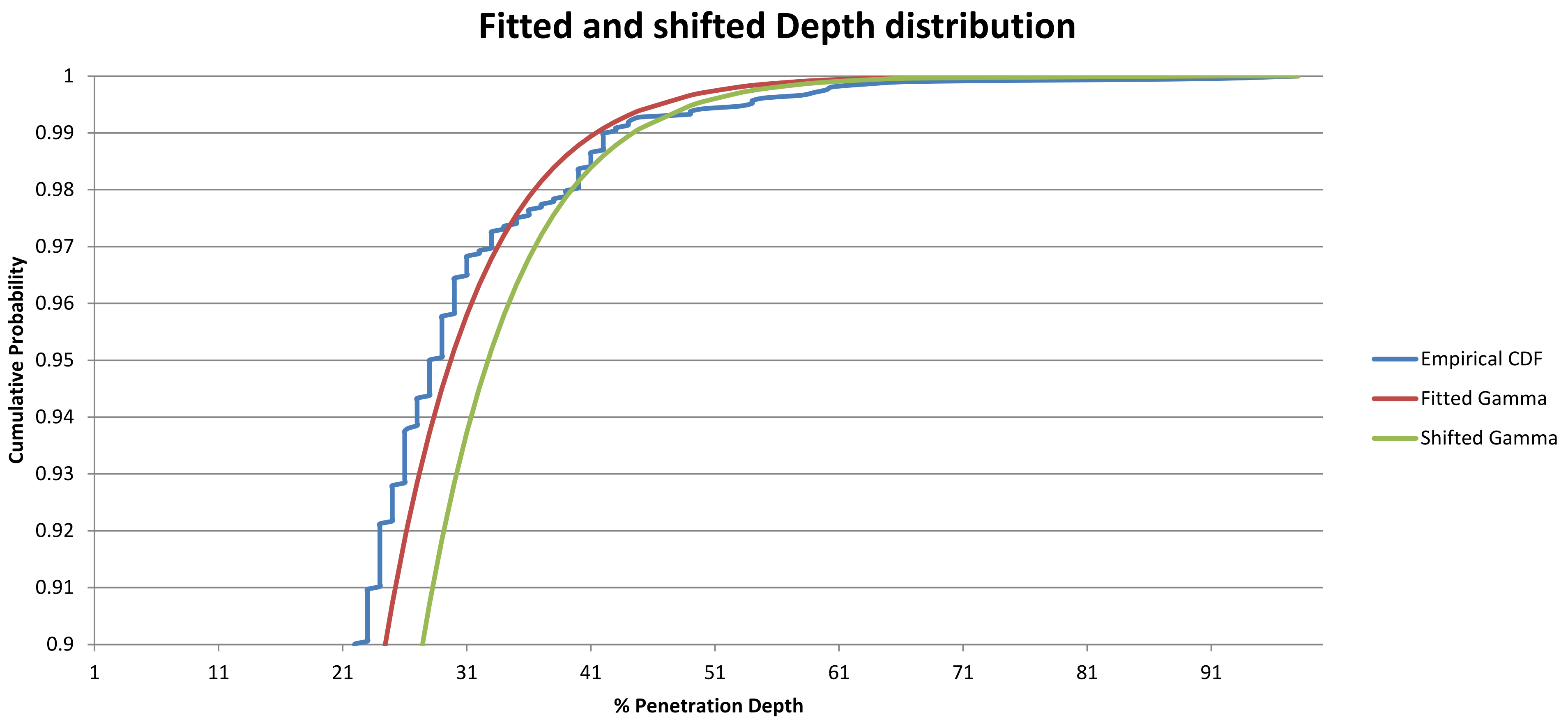
Flaw Depth Distribution



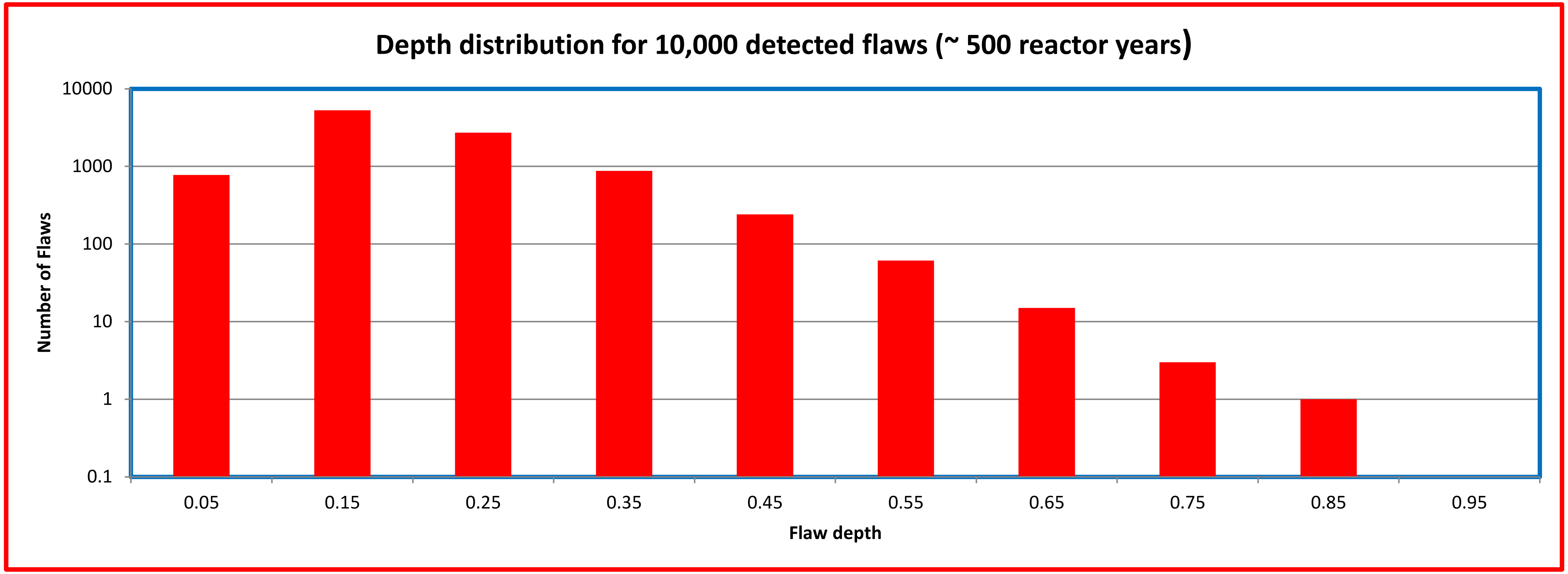
Adjusted (shifted) Flaw Distributions used in PRA

- During PRA analysis, adjustments were made to the original estimated distributions of flaw depth and length. For example, for Westinghouse plant, the large and deep flaws were the major contributor. A better fits at the tail of distributions of length and depth therefore were examined.
- There were also a large number of unreliable small depth/length measurements; i.e. depth less than 10%, which skewed the size distributions of depths and lengths towards the lower values.
- To improve the distribution fit for large flaws which are more important to C-SGTR and to compensate for the perceived distortion of flaw size distributions towards the shallower and smaller flaws, the previous distribution were shifted by a small amount of depth and length (adding a scale variable to Gamma distribution).
- This adjustment also provided much closer estimates of the number of tubes that are plugged in each cycle (better estimate of the number of large/deep flaws at the tails).

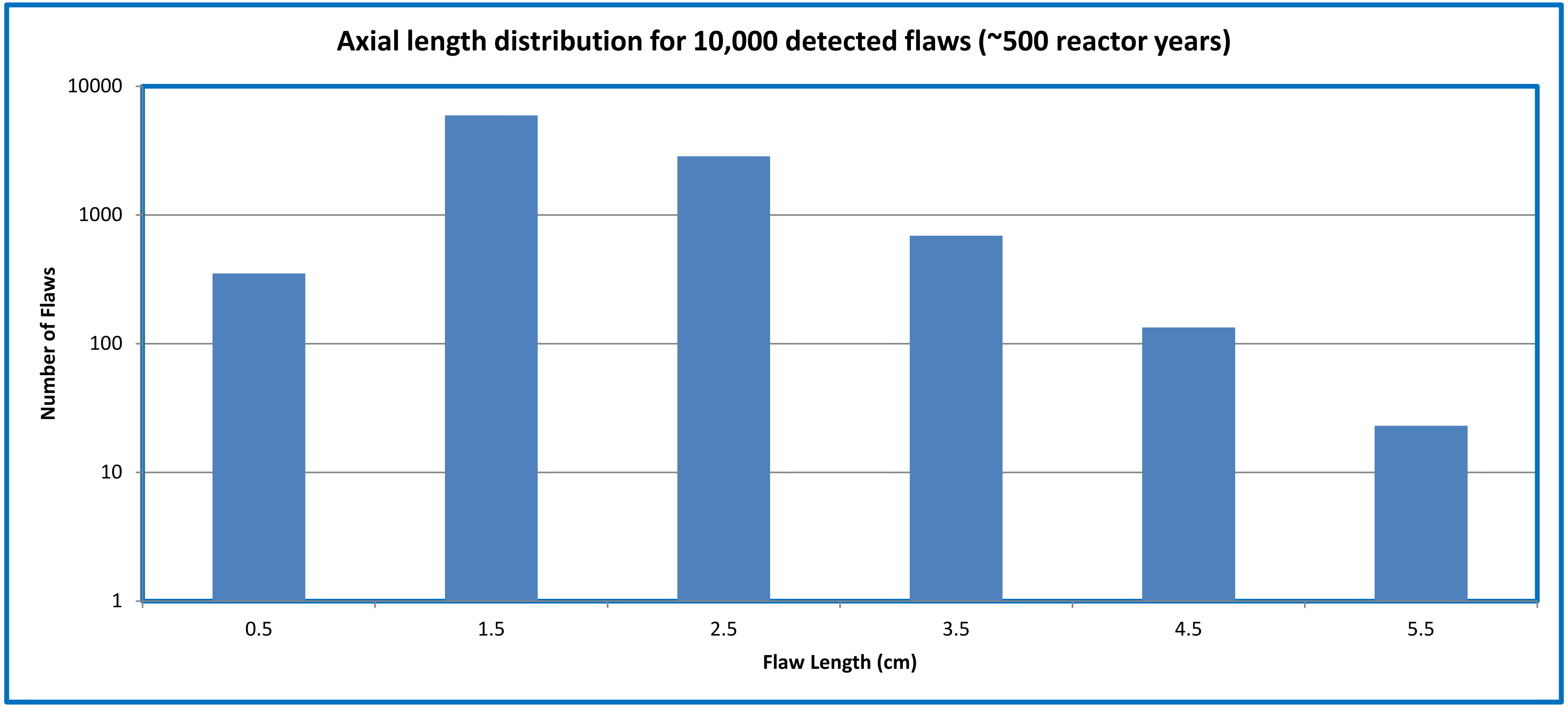
Flaw Depth Distribution (Shifted Gamma)



600TT Flaws for 10 SGs – Distribution by Depth



600TT Flaws for 10 SGs for illustration purposes – Distribution by Length



General Findings

- Sufficient statistical results were developed to generate flaw samples for the C-SGTR calculator software.
- New material 600TT/690TT flaw rate generation is about an order of magnitude less than what was reported for MA 600.
- The majority of flaws observed are volumetric rather than cracks
- The flaw length and depth distribution is somewhat smaller than MA 600.

Severe Accident-Induced Steam Generator Tube Rupture (SGTR)

C-SGTR Calculator

Dr. Ali Azarm, IESS
Dr. Selim Sancaktar, RES/DRA

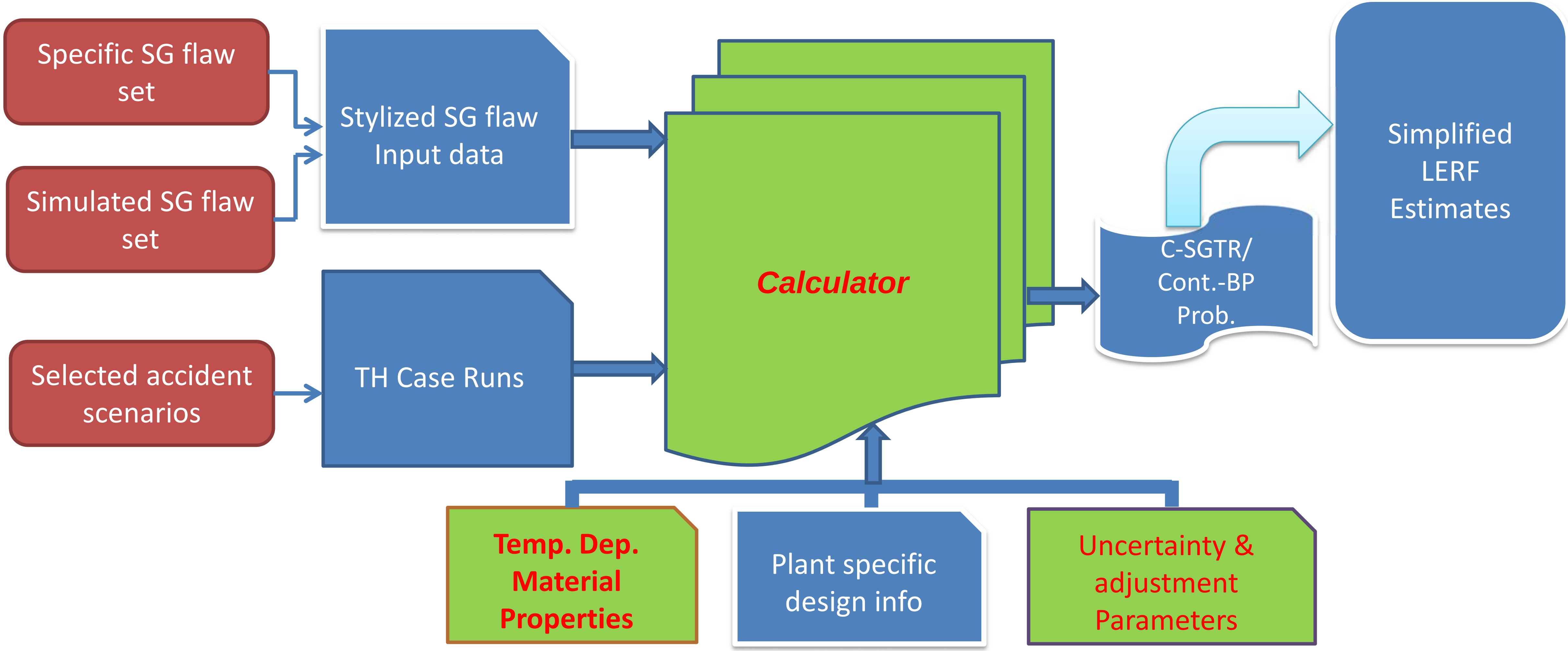
Consequential Steam Generator Tube Rupture (C-SGTR) Subcommittee Briefing

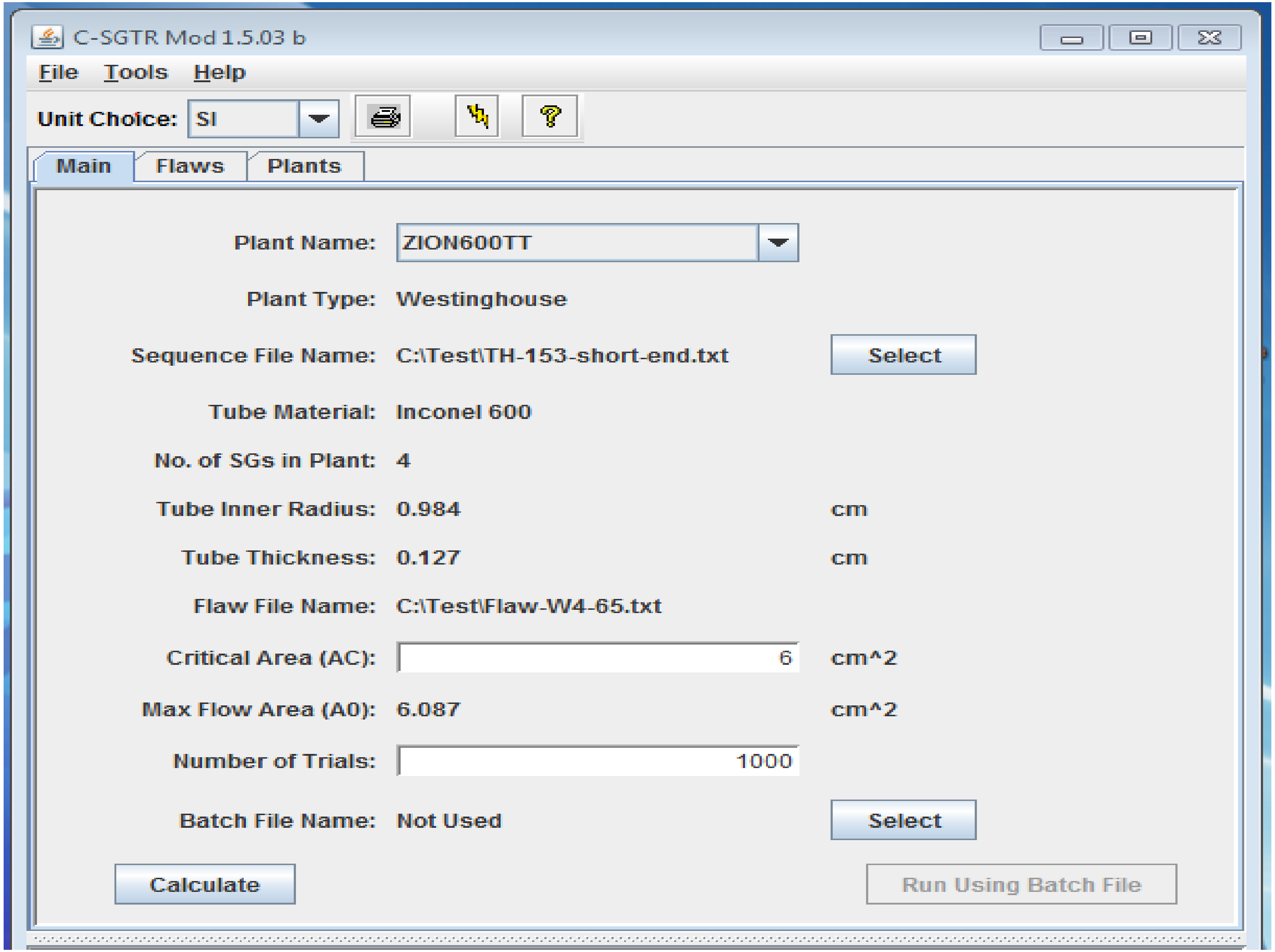
April 7, 2015

C-SGTR Calculator

- A software referred to as the C-SGTR Calculator has been developed to support the recent C-SGTR PRA analyses.
- The calculator is used to estimate failure times and leak sizes of SG tubes with different types of flaws.
- The software also has built in models for failure of HL and surge line due to creep rupture failure mechanism to estimate failure times and probabilities of HL and surge line.
- Input material data for thermally-treated Inconel 600 and 690 (TT600 and TT690) is used for SG tubes modeled.

Role of the Calculator in the C-SGTR PRA Analysis




C-SGTR Mod 1.5.03 b
File Tools Help
Unit Choice: SI
Main
Flaws
Plants
Plant Name: ZION600TT
Plant Type: Westinghouse
Sequence File Name: C:\Test\TH-153-short-end.txt
Select
Tube Material: Inconel 600
No. of SGs in Plant: 4
Tube Inner Radius: 0.984 cm
Tube Thickness: 0.127 cm
Flaw File Name: C:\Test\Flaw-W4-65.txt
Critical Area (AC): 6 cm^2
Max Flow Area (A0): 6.087 cm^2
Number of Trials: 1000
Batch File Name: Not Used
Select
Calculate
Run Using Batch File

Calculator: Models

I. Pressure Induced Failure Model for SG tubes and Estimated Leak Area

II. Creep Rupture Failure Model for SG tube and Estimated Leak Area

III. Creep Rupture Failure Model for Hot Leg

IV. Creep Rupture Failure Model for Surge-line

Calculator: An Overview

- Models
  - Failure of a flawed tube [Mostly NRC models]
    - Wear (volumetric) and Cracks for SG tubes
  - Creep rupture failure for hotleg and surge line [EPRI model]
    - No flaw postulated
- Material Properties as a function of Temperature for:
  - Inconel 600 TT/690 TT [Open literature]
- Larson Miller Creep Rupture Parameters for:
  - Inconel 600TT/690 TT for SG tubes [NRC & Open Literature]
  - Hot leg and surge line [EPRI]

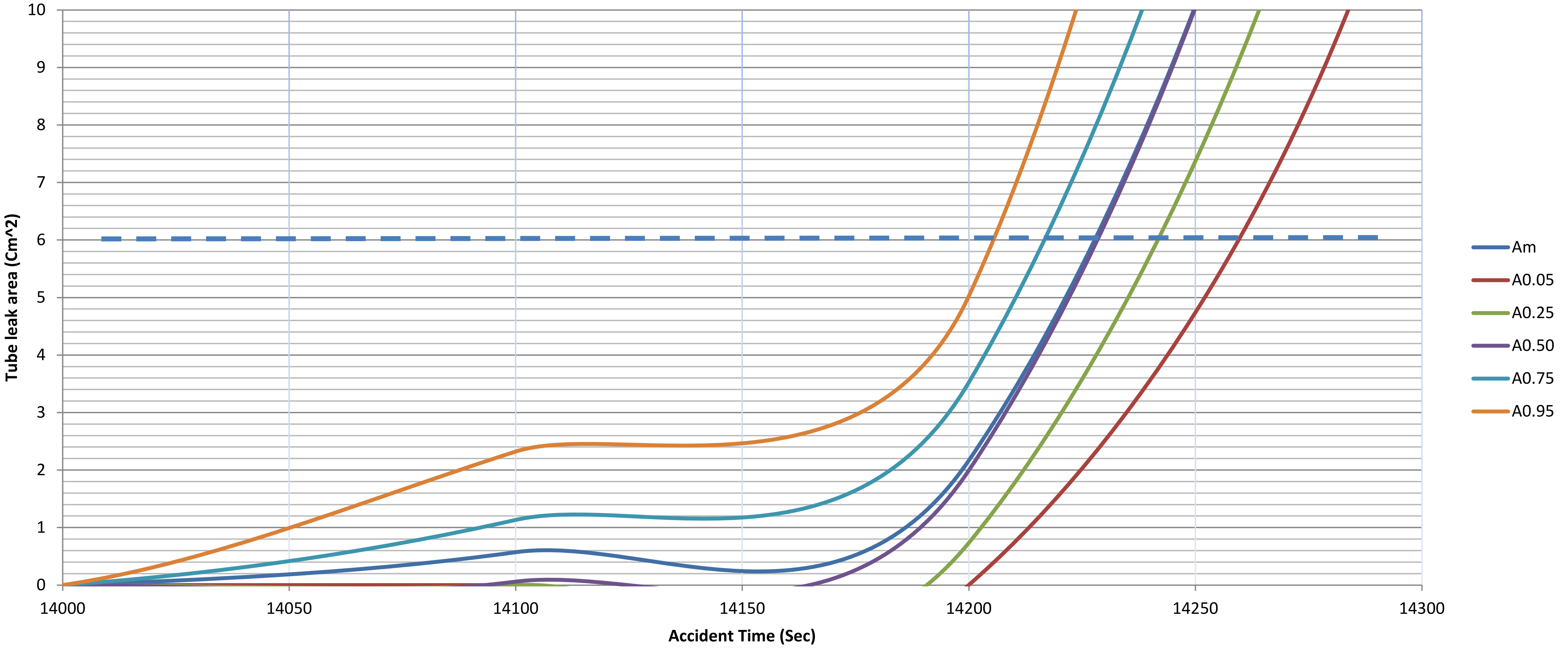
Calculator: Sequence Input

- Thermal Analysis of the accident condition, such as from MELCOR informed by CFD :
  - The fraction of hot tubes and their average temperature
  - The fraction of cold tubes and their average temperature
  - The fraction of hottest tubes and their average temperature
  - Primary/secondary pressure
  - Hot leg temperature
  - Surge-line Temperature

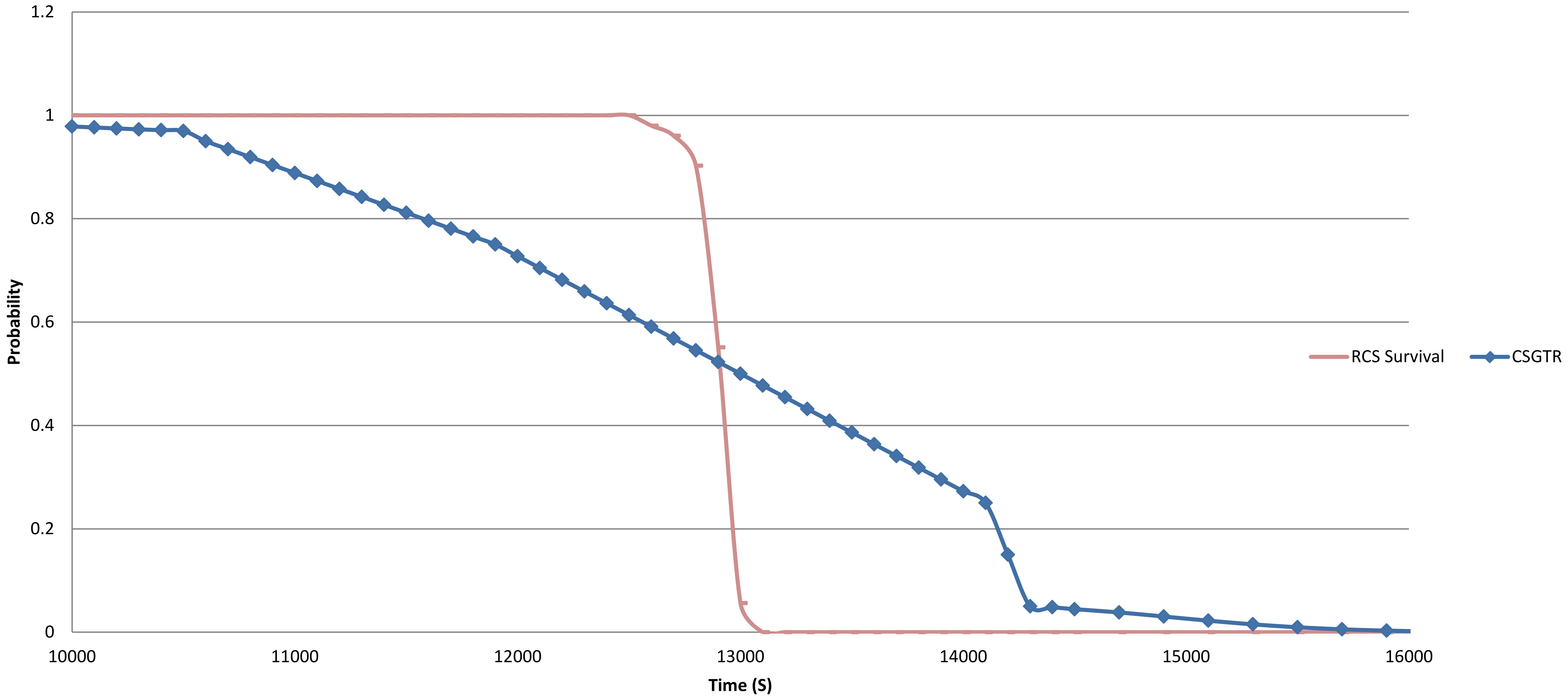
Calculator: An Overview

- Calculator Output
  - C-SGTR leak area probability distribution [5%, 25%, 50%, 75%, 95%] as a function of accident time
  - Probability of failure of one hot leg or the surge line as a function of time
- Probabilities are estimated when uncertainties and variations are accounted for

Graphical Presentation of Output: Total SG Tube Leak Area



Graphical Presentation of Output Information: Survival Probabilities



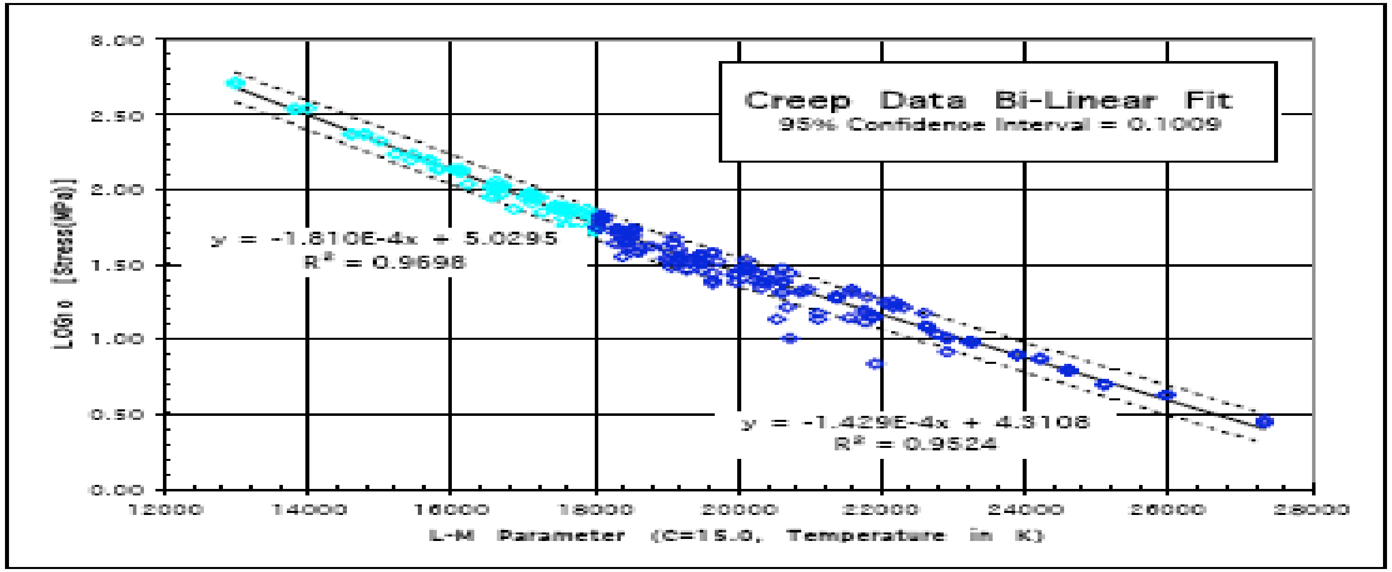
Calculator: Overview of Uncertainties

- Modeling uncertainties
  - – Parameter uncertainties for creep rupture model for SG tubes, hot leg, and surge line
  - – Aggregate (lump) uncertainty for leak rate and failure pressure predictions for both pressure induced and creep rupture failure of SG Tubes
- Measurement Uncertainties
  - – Uncertainties in material properties
  - – Measurement uncertainties of Flaw size (depth and length)

Calculator: Overview of Uncertainties

- Numerical Errors in estimation
  - Not handled Formally
  - All runs were performed using 1000 MC samples
- Probabilistic Variations
  - Variations represented by flaw samples [handled outside the code]
  - Variations of TH analyses results for class of accidents [handled outside the code]

Example: Uncertainties for Larson Miller Parameter

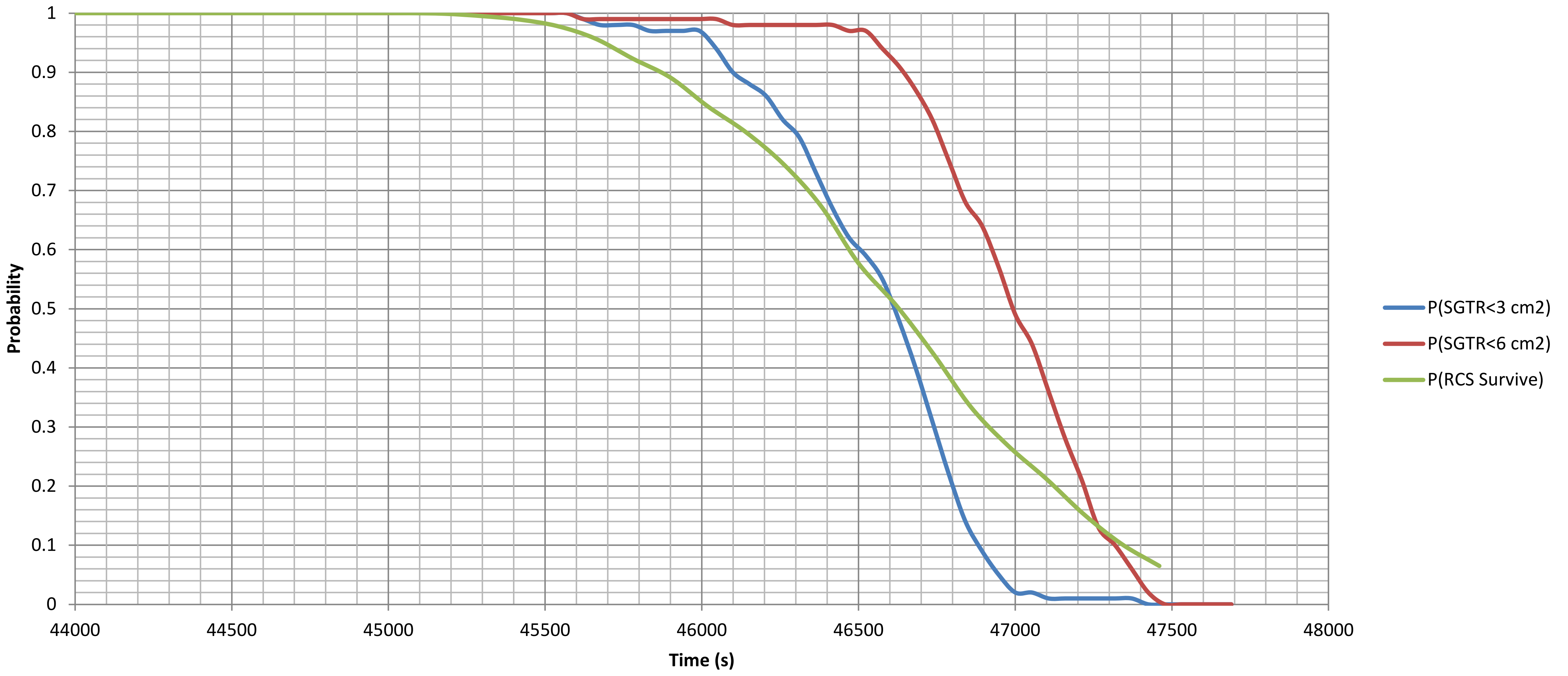


$$P_{LM} = ((27.78 \pm 0.44) - 2.4\ln(m_p\sigma)) \times 10^3 \qquad m_p\sigma \geq 5.7\,ksi = 39.3Mpa$$

$$t_R = 10^{\frac{P_{LM}}{T} - 15} \qquad \int_0^{t_f} \frac{dt}{t_R} = 1$$

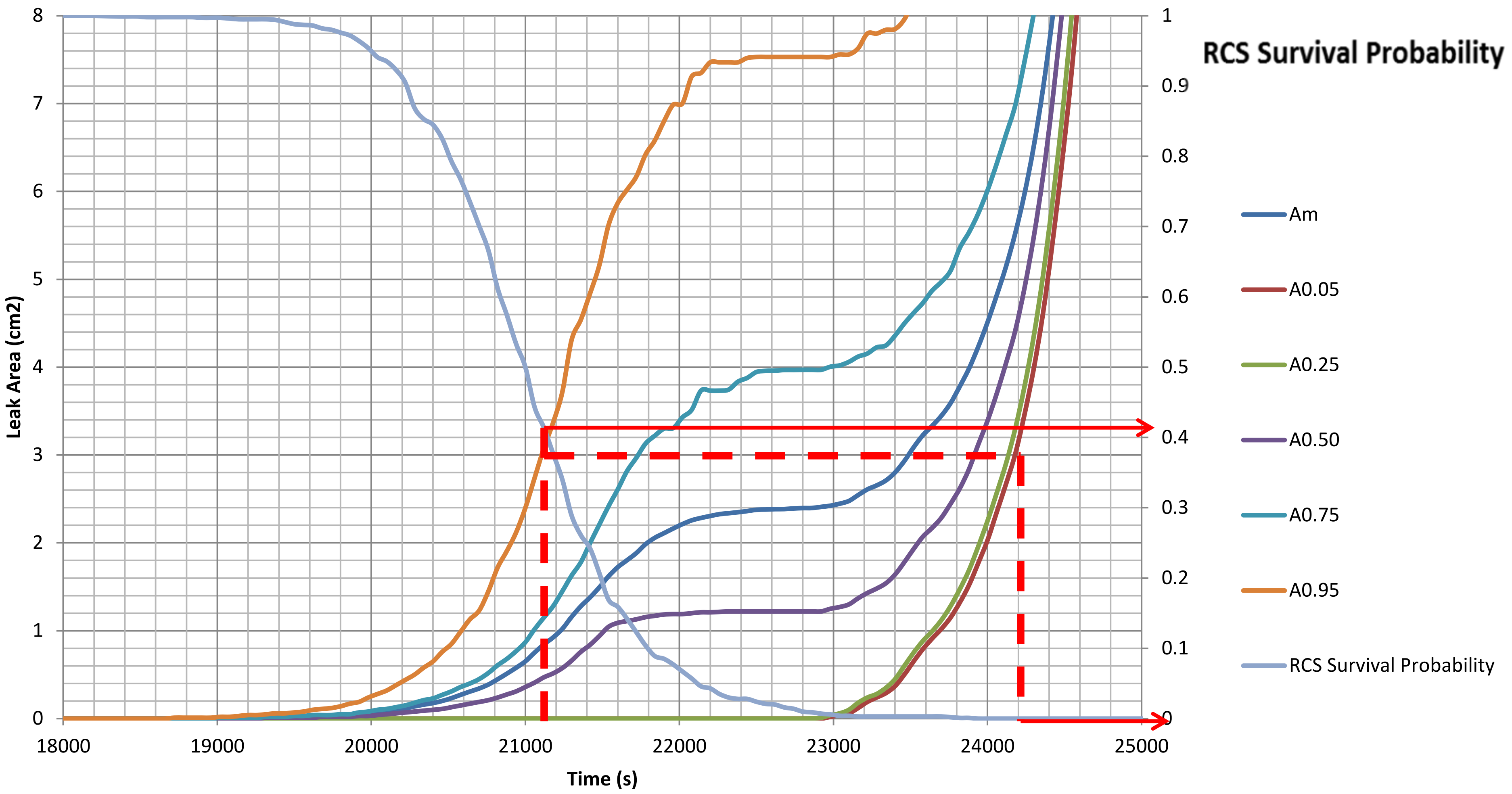
<u>For a T value of 1000 degree Kelvin (~727 C), the overall error interval (two sided 5% and 95%) for time to failure; t_f, can vary by a factor of 10.</u>

Illustrative Graphical Examples



The RCS survival Probability and the probability of SGTR with a Leak Area less than 3 and 6 cm^2 for ltsbo-a-b-scf

Illustrative Graphical Examples



The RCS survival Probability and percentiles of SGTR Leak Areas for stsbo-a-hottest tubes

Calculator V&V

- Limited results from individual equations were tested against published results from testing
- Case runs for each module was checked against the results obtained from similar MCAD routine
- Initial Version of software and Probabilistic Fracture Mechanics Models for SG tubes were reviewed by Argonne National Laboratory (ANL).
- Models for hotleg and surge line model examined for reasonableness by RES/DE using finite–element program ABAQUS®
- Predicted 50 percentiles of the predictions were compared to deterministic results for limited simple case runs

Severe Accident-Induced
Steam Generator Tube Rupture (SGTR)

Probabilistic Risk Analysis of C-SGTR

Dr. Ali Azarm, IESS
Dr. Selim Sancaktar, RES/DRA

Consequential Steam Generator Tube Rupture (C-SGTR) Subcommittee Briefing

April 7, 2015

Background

- In response to a NRC/NRR user need request, RES has performed a C-SGTR Analysis for two selected plants; a Westinghouse (W) and a Combustion Engineering (CE).
- A draft NUREG report titled

  Consequential SGTR Analysis for Westinghouse and Combustion Engineering Plants with Thermally-Treated Alloy 600 and 690 Steam Generator Tubes
- Is completed.
- The draft report contains work done by three disciplines in RES:
  - thermal-hydraulic and computational fluid dynamics analysis (RES/DSA),
  - Fracture Mechanic analyses (RES/DE), and
  - probabilistic risk assessment (RES/DRA)
- This presentation discusses PRA aspects of the work.

Objective

- To develop a methodology for a quantitative assessment of the risk associated with C-SGTR during a severe accident after the onset of core damage, and during a DBA event before the onset of core damage.
- Estimation of the LERF* as a result of containment bypass due to C-SGTR and its application to two PWRs: a Westinghouse and a Combustion Engineering design consistent with NRC/NRR user request.
- Estimation of the probability of C-SGTR* using calculator software.

* C-SGTR is defined by a leakage area greater than a threshold value occurring before a large vent path in RCS is established; either due to failure of RCS components or intentional depressurization.

C-SGTR before and after the onset of core damage

- After Core Damage
  - A C-SGTR ensuing from a severe accident (CD already postulated) is mainly due to creep rupture (temperature-induced C-SGTR).
  - Such a C-SGTR may increase LERF, but does not affect plant CDF.
  - The race between SG tube failure and failure of another RCS component (hot leg, surge line, PORVs/SRVs, etc.) determines whether the majority of fission products may be retained in the containment or not.
- C-SGTR Before Core Damage
  - A C-SGTR before core damage may be caused by accidents that create a delta pressure across the SG tube walls significantly larger than its nominal value.
    - A C-SGTR Before core damage is mainly caused by pressure induced failures of the SG tubes.
  - Such a C-SGTR will complicates the accident progression and will increase both CDF and LERF.

Example of a C-SGTR after the onset of core damage a fast (short) SBO

- Initial Phase
  - loss of offsite power, failure of diesels, and failure of auxiliary feedwater systems
  - natural circulation is established
  - primary inventory lost through reactor coolant pump seals.
  - Secondary side boils off, secondary side dry
  - Additional primary inventory lost through safety valve cycling and RCP seals
- loop natural circulation stops as primary inventory falls in SG tubes.
  - Counter current flows of superheated steam begins as inventory falls below hot leg; Core and primary system heat up.
- Core uncovers
  - core oxidizes and produces significant power, system heat up accelerates
  - This can induced failure of RCS piping or Steam Generator tubes (A race in time)

Selected Timelines for C-SGTR after Core Damage

SGTR Occurs ↓↓

Case 1

←Reactor Trip	←Evacuation Starts	←Core Damage	←Evacuation Ends

C-SGTR occurs too late to contribute significantly to LERF

SGTR Occurs ↓↓

Case 2

←Reactor Trip	←Evacuation Starts	←Core Damage	←Evacuation Ends

↑ Hot Leg Fails

C-SGTR occurs after hot leg failure thus fission product release is limited

SGTR Occurs ↓↓

Case 3

←Reactor Trip	←Evacuation Starts	←Core Damage	←Evacuation Ends

↑ Hot Leg Fails

C-SGTR occurs before hot leg failure; fission product release not curbed until hot leg failure

Example of C-SGTR before core damage – Blow down of one SG

- Rapid and complete blow down
  - Delta Pressure across SG tube increases from approximately 1200 psi (nominal) to about 2250 psi
  - Affected Steam generator dries out
  - Operator fails to isolate the affected SG
  - Primary remains high pressure, High pressure system inject
- One or more SG tubes may rupture if they have a deep flaw(70% or more flaw depth)
- Operator fail to re-establish HPI (If HPI was terminated early on in response to SG blow down)
  - Core Uncovery and onset of core damage due to loss of primary inventory through ruptured SG tubes and failure to reestablish HPI
- This example scenario could contribute to both CDF and LERF

Steps for a simple PRA for LERF estimation
Severe Accidents (CD already postulated)

1. Select accident sequences and estimate their frequencies from the existing PRAs.
2. Determine T&H characteristics of the sequences.
3. Consider a single flaw of various depth and determine the probabilities of SGTR before RCS failure as a function of flaw depth
4. Depending on the results from step 3 generate appropriate flaw samples for the needed calculator runs and calculate the associated flaw probabilities
5. Perform case runs with the calculator software
6. Estimate containment bypass probability
7. Estimate the associated LERF contribution

1. Select Accident Sequences and Their Frequencies

- Identify High and Dry Sequences
  - Level 2 analysis is available: High primary and dry secondary are Level 2 binning questions, so this task is straight forward for plants with existing Level 2 analysis
  - Level 1 analysis is available: search the sequences to identify those when at the onset of core damage the primary is at high pressure and the secondary dry [e.g. SBO scenarios, feed and bleed scenarios].
- Consider CDF contributors, internal and external events
  - Consider the potential for multi-unit impact; e.g. extended SBO due to seismic impacting multiple Units

1. Select Accident Sequences and Their Frequencies (Cont'd - two Pilot Plants)

Zion (W-Plant) Sequences	Short Term SBO CDF	Long Term SBO CDF
Internal events including internal floods	5.23E-7	5.2E-6
Seismic	~5.6E-7	5.6E-6
Fire	~9.5E-7	9.5E-6
Total	2.03E-6	2.03E-5

Calvert Cliffs (CE-Plant) Sequences	Short Term SBO	Long term SBO
Internal events	2.5E-08	1.7E-07
Seismic	6.4E-08	2.0E-7
Fire	2.4E-06	2.4E-05
Flood	ε	1.6E-06
High wind	4.7E-08	4.3E-06
Total	2.6E-06	3.1E-05

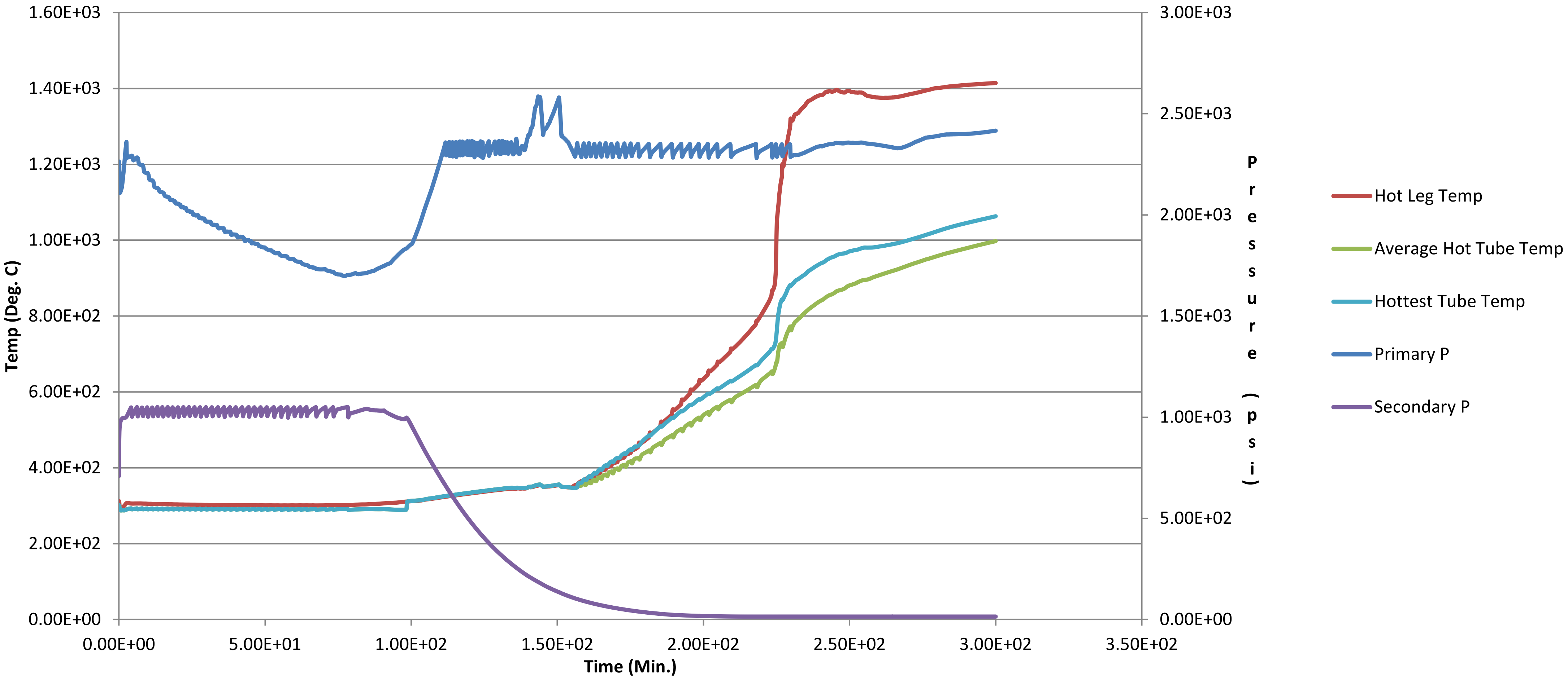
1. Select Accident Sequences and Their Frequencies (Potential Sequences for Dual Unit: Calvert Cliffs)

Initiating Event	Short Term SBO		Long term SBO		At least one Unit CDF	
	Single Unit *	Dual Unit	Single Unit*	Dual Unit	Short Term SBO	Long term SBO
Internal events	1.9E-08	5.5E-09	4.5E-08	1.2E-07	2.5E-08 (~13%)	1.7E-07 (~87%)
Seismic	5.0E-08	1.4E-08	ε +	2.0E-07	6.4E-08 (24%)	2.0E-7 (~76%)
Fire	ε	2.4E-06	2.2E-05	2.2E-06	2.4E-06 (~9%)	2.4E-05 (~91%)
Flood	ε	ε	1.6E-06	ε	ε	1.6E-06 (~100%)
High wind	ε	4.7E-08	ε	4.3E-06	4.7E-08 (~1%)	4.3E-06 (~99%)
Total	6.9E-08	2.5E-06	2.4E-05	6.8E-06	2.6E-06 (~8%)	3.1E-05 (~92%)

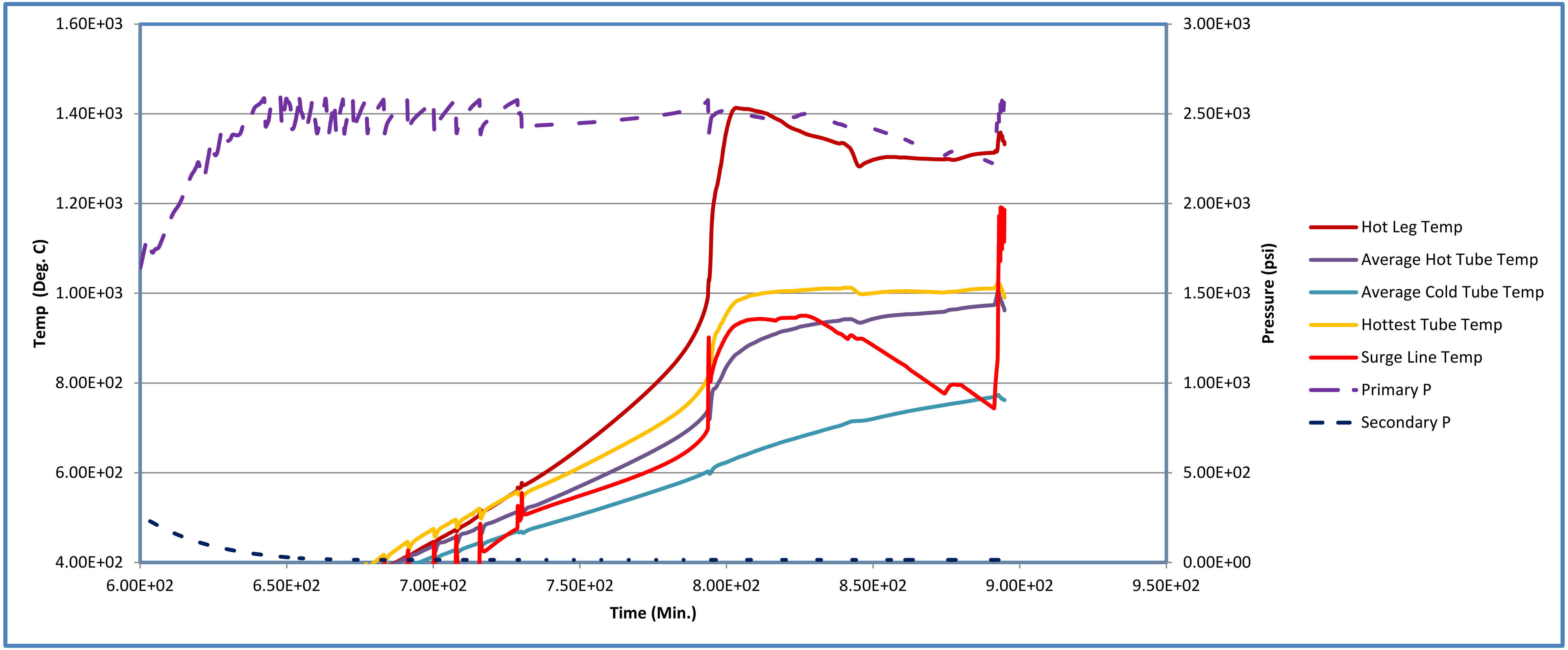
2. Determine T&H characteristics of the sequences (RELAP/SCADAP for Zion)

- Base-Case for Short term SBO
  - SBO with early failure of TDAFW
  - 21 gpm RCP leakage per pump
  - 0.5 square inch hole in secondary side of SG
  - No recovery of AC
- Base-case for Long term SBO
  - SBO with Failure of TDAFW after Battery depletion
  - 21 gpm RCP leakage per pump
  - 0.5 square inch hole in secondary side of SG
  - No recovery of AC
- Several sensitivity Case Studies
  - Rapid and early depressurization
  - Different RCP leakage rates
  - Larger hole in secondary including open SG SRV/ARVs
  - Sensitivity of the threshold of C-SGTR leakage areas

TH Results for Westinghouse SBO Sequence with Failure of TDAFW Pump at T=0



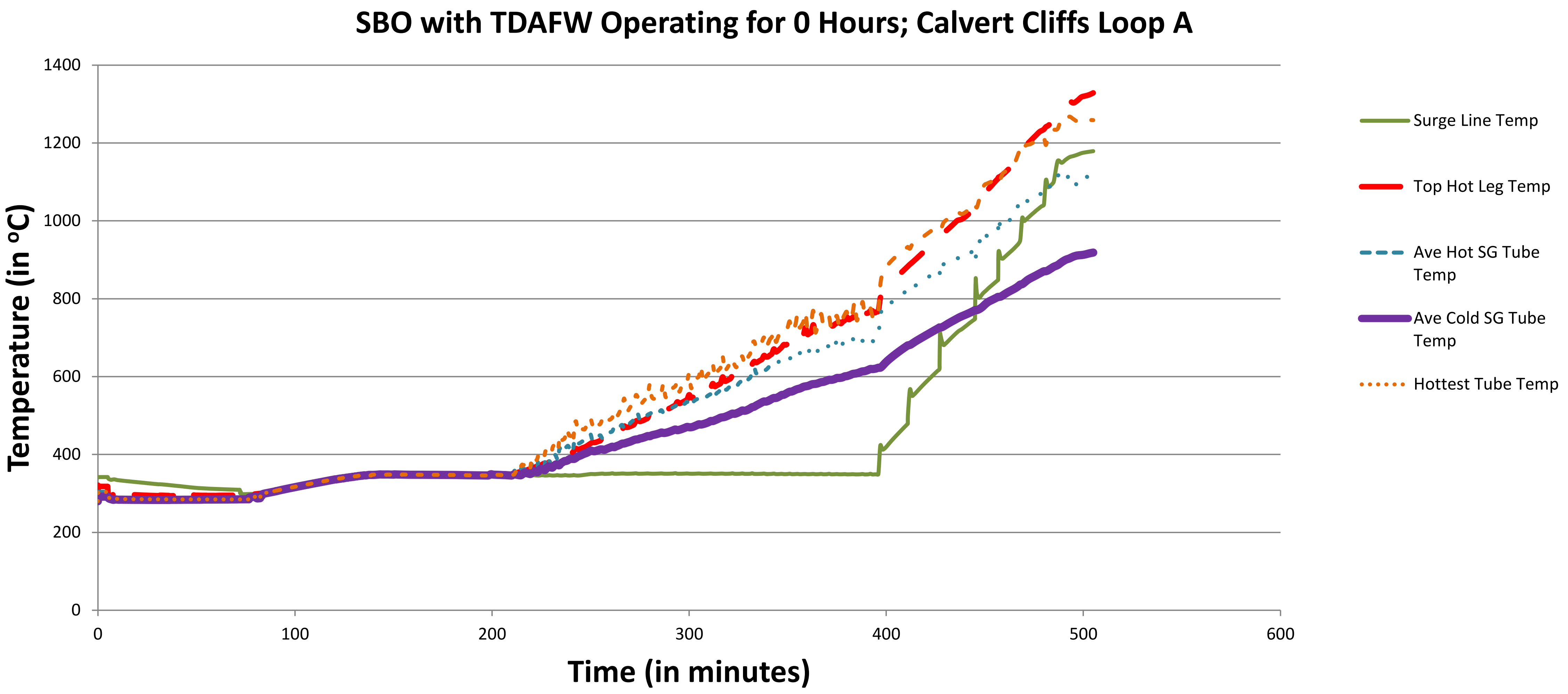
TH Results for Westinghouse SBO Sequence with Failure of TDAFW Pump at T=4 Hours



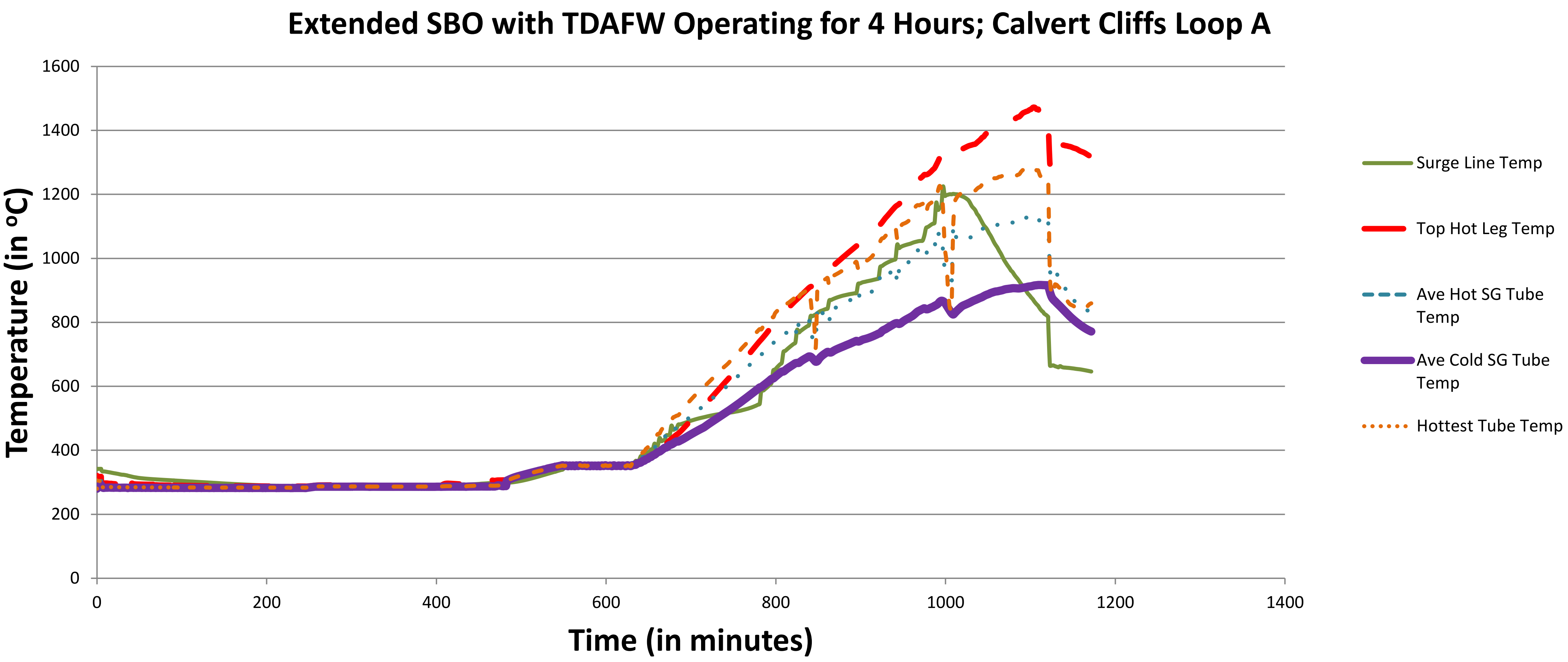
2. Determine T&H characteristics of the sequences (MELCOR for Calvert Cliffs)

- Base-Case for Short term SBO
  - SBO with early failure of TDAFW
  - 21 gpm RCP leakage per pump
  - 0.5 square inch hole in secondary side of SG
  - No recovery of AC
  - Differentiated between primary Loop with and without pressurizer
- Base-case for Long term SBO
  - SBO with Failure of TDAFW after Battery depletion
  - 21 gpm RCP leakage per pump
  - 0.5 square inch hole in secondary side of SG
  - No recovery of AC
  - Differentiated between primary Loop with and without pressurizer
- Sensitivity Case Studies
  - Larger hole in secondary - failure of SG SRV to reclose after the first demand
  - With or without failure of RCS component s

TH Results for CE SBO Sequence with Failure of TDAFW Pump at T=0 Hours



TH Results for CE SBO Sequence with Failure of TDAFW Pump at T=4 Hours



3. Flaw depth screening using C-SGTR calculator for base case short (fast) SBO scenarios

Flaw Depth	West. Representative Plant		CE Representative Plant	
	Ave. Hot Tube	Hottest Tube	Ave. Hot Tube	Hottest Tube
<0.05 (Pristine)	~ 0	~ 0	~0	~100%
0.25	~ 0	~ 0	**~5%**	~100%
0.50	~ 0	~ 0	~80%	~100%
0.75	50%	~100%	100%	~100%
0.9	100%	~100%	100%	~100%

- Only Large Flaws will contribute to C-SGTR probability for the representative Westinghouse plant.
- Both large flaw and medium size flaws are contributing to C-SGTR for the representative CE plants. Small flaws can contribute if located at hottest tubes.
- There is a small fractions of tubes that are exposed to hottest tube temperature and the flaws Will be assigned randomly to them. Failure of pristine tube exposed to hottest temperature are not expected to create large leakage area.

4. Flaw samples for calculator case runs

I. For cases where a few large/deep flaws (much larger than SG tube plugging criteria) are the major contributor to C-SGTR; Perform the analysis with a small sample of large flaws.
   - In such cases we expect low C-SGTR probability, since the probability for one or more very large flaws within a cycle is expected to be small.

II. For cases where flaws of all sizes contribute to C-SGTR probability; Perform analysis using average set of flaws in addition to sensitivity analysis using several sets of flaws for addressing the uncertainty due to flaw variation .
   - In such cases we expect high probability of C-SGTR and some uncertainties due to variations in flaw samples.

4. Flaw Samples for Calculator Case runs (cont'd)

- For the Representative Westinghouse Plants
  - Used a limited sample of large flaws
  - Estimated the frequency that there could be one, two, or more large flaws within a cycle
- For the Representative CE Plant
  - Develop average or expected sample bins to estimate the mean probability of C-SGTR
  - When needed, developed limited sample flaws for sensitivity analysis and addressing the uncertainties

4. Flaw Samples for Calculator Case runs (cont'd)

For Rep. West. Plant: Large Flaws only

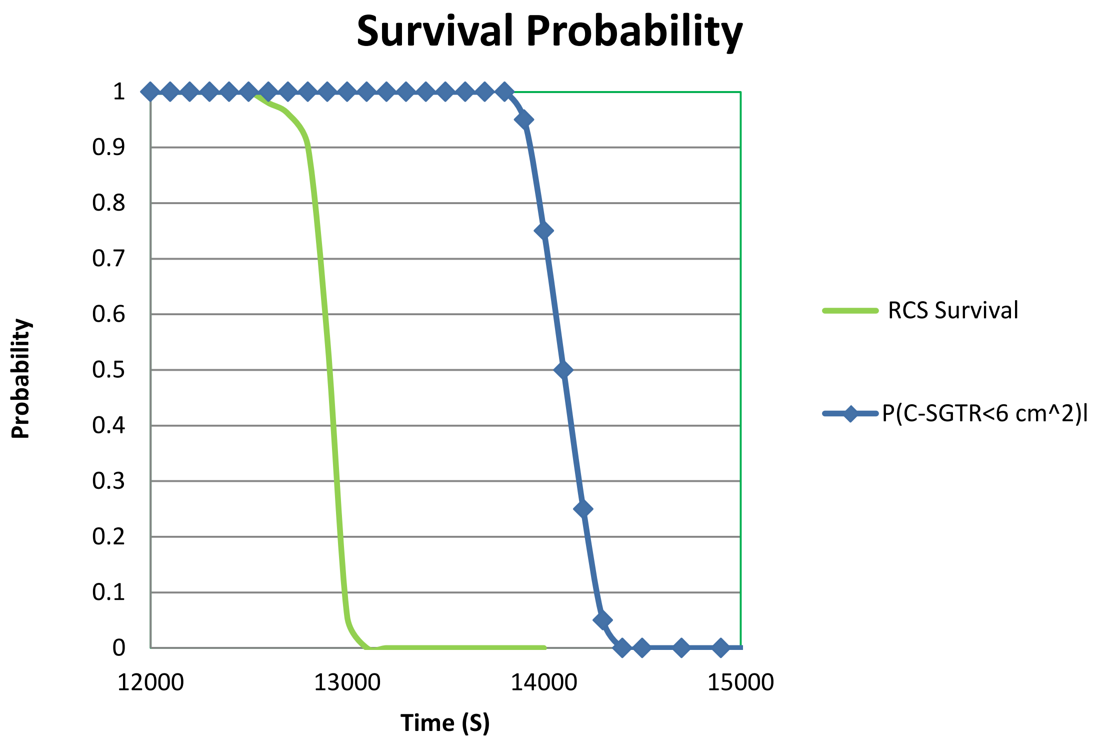
Depth Bin	Probability of Flaw Belonging to Depth Bin	Expected Number of Flaws in Depth Bin
0.6 – 0.7	1.46E-03	0.0453
0.7 – 0.8	3.39E-04	0.0105
0.8 – 0.9	7.70E-05	0.00239
0.9 – 1.0	small	small
Total	1.86E-3	0.06

For CE Plant :Expected Number of Flaws for both SGs

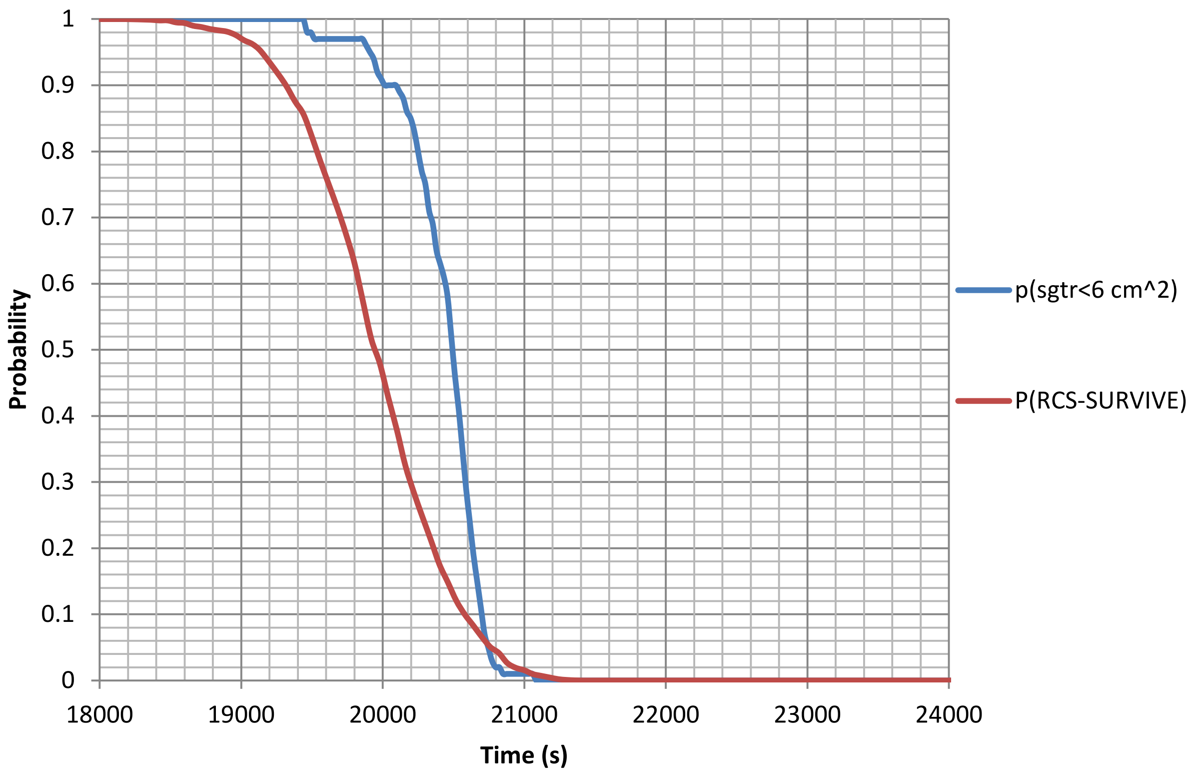
Depth / Length	1 cm	2 cm	3 cm	4 cm	5 cm	6 cm	Total
0.1	1	13	7	2	0	0	22
0.2	6	88	43	11	2	1	151
0.3	3	45	22	5	1	0	76
0.4	1	1	1	0	0	0	3
0.5	0	1	0	0	0	0	1
0.6	0	0	0	0	0	0	0
0.7	0	0	0	0	0	0	0
0.8	0	0	0	0	0	0	0
0.9	0	0	0	0	0	0	0
Total	10	148	73	18	3	1	253

5. Case Runs with Calculator Software (short/fast SBO)

West. Rep. Plant: Very Small C-SGTR Probability



CE. Rep. Plant: relatively high C-SGTR Probability



6. Estimate containment bypass probability

- Containment bypass Mathematical Equation for Each Set of Flaw

$$Prob(CSGTR) = \int Prob\ (RCS\ survive\ at\ t) * Prob(CSGTR\ occurs\ between\ t\ and\ (t + dt)) * dt$$

- The integration is summed over all flaw samples accounting for their probabilities
- The calculations generally performed using a Fortran routine or using MS-Excel worksheet.

6. Estimate containment bypass probability (Cont'd)

CSGTR Leak Area	Probability Of Containment Bypass			
	SBO with Early Failure of TDAFWs		SBO with Failures of TDAFWs after Battery Depletion	
	CE-690 (with SG SRV open)	WEST. 600 (690)	CE	WEST. 600 (690)
Greater than one tube (~6 cm^2)	0.22 (0.99)	1.31E-02 (8.90E-3)	0.31 (~1)	2.6E-02 (1.8E-2)
Greater than two tubes (~ 12 cm^2)	0.06 (NC*)	8.23E-5 (3.85E-5)	0.10 (NC)	1.6E-4 (7.8E-5)
Greater than three tubes	(NC)	~0	(NC)	~0

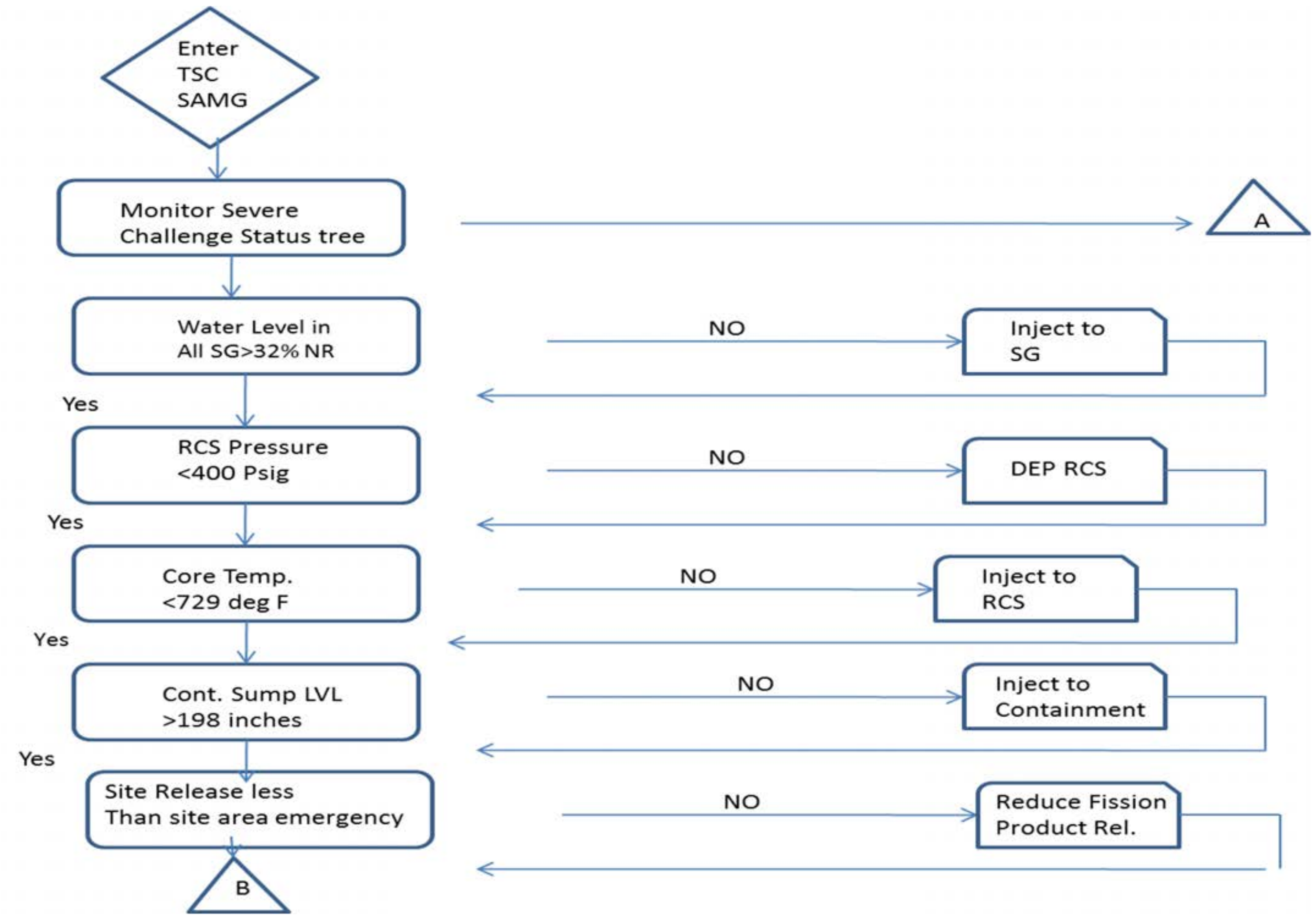
*NC: Not Calculated

7. Simplified LERF Model

- Detail Level 2 Models
  - Advantages
    - Differentiate between Large and small releases
    - Account for SAMG activities
    - Provide detail timing of releases
  - Challenges
    - Equipment operations after onset of core damage (PORV, SG SRVs, ARVs, etc.)
    - HRA associated with SAMG actions
    - Adequate differentiation in magnitude of source terms and release timing
- Bounding LERF analysis

7. Simplified LERF Model

Typical SAMG Actions for C-SGTR



7. Simplified LERF Model

SAMG actions related to C-SGTR

- Objectives of SAMG
  - To arrest the core melt within the vessel by depressurization and injection
  - To reduce radioactive release magnitude (scrubbing) by depressurizing and injecting to/refilling the SGs
- Actions depends on
  - AC recovery
  - DC availability
- Additional issues related to C-SGTR
  - Bump RCPs
  - Injection of cold water into a dry SG

7. Simplified LERF Model

- Simplified bounding LERF Evaluation using factor analysis
  - Frequency of severe accident sequences with potential for C-SGTR (f_{AC}),
  - Containment Bypass Probability due to C-SGTR (P_{CSGTR}),
  - Conditional Probability that the subsequent failure of RCS including the stuck open relief valves do not occur (P_{NDEP})
  - Failure Probability of all SAMG actions (P_{SAMG})
  - Probability that early effective evacuation is not successful (P_{EVAC})

Conditional LERF Probabilities for an SBO with Early and Late Failures of TDAFW for W plant

Factors	Applicability	LERF Factors
P_{CSGTR}	Due to one or more tube breaks in a SBO CD Sequence	1.3E-02
	- Due to single tube break only	1.3E-02
	- Due to multiple tube breaks	8.2E-05
	- Loop seal cleared	1.0
P_{NDEP}	Multiple tubes with loop seal cleared	1.0
	Break of one SG tube and w/o loop seal cleared	1.0
P_{SAMG}	Multiple tubes with loop seal cleared	1.0
	Break of one SG tube and w/o loop seal cleared	1.0
P_{EVAC}	In a SBO, CD Sequence with early failure of TDAFW	1.0
	In a SBO, CD Sequence with late failure of TDAFW	0

Conditional LERF Probabilities for SBO with Early and Late Failures of TDAFW for CE plant

Factors	Applicability	LERF Factors (early, late)
P_{CSGTR}	Sequences with no stick open primary or secondary relief valves	(0.22, 0.31)
	Sequence with loop seal clearing	(1.0,1.0)
P_{NDEP}	Sequence without loop seal clearing	(1.0, 1.0)
	Sequence with loop seal clearing	(1.0, 1.0)
P_{SAMG}	Sequence without loop seal clearing	(1.0, 1.0)
	Sequence with loop seal clearing	(1.0, 1.0)
P_{EVAC}	For all sequences	(1.0, 0.0)

Summary Results

SG TYPE	Tube Material	#of SGs	EFPY	Hazard Category	SBO CDF (per RY) (short & Long)	Cont.-Bypass Frequency per year	LERF Fraction (%)	LERF (per RY)
CE	690	2	15	All[1]	3.3E-05	1.01E-05[2]	5.6%[2]	5.7E-07[2]
CE	690	2	15	Internal	1.9E-07	5.65E-08	9.5%	5.4E-09
W	600	4	15	All	2.03E-05	8.8E-07[3]	3.7%	3.2E-08
W	600	4	15	Internal	5.23E-06	2.3E-07	3.6%	8.4E-09
W	690	4	15	All	2.03E-05	6.3E-07	3.5%	2.2E-08
W	690	4	15	Internal	5.23E-06	1.6E-07	3.5%	5.8E-9

1. All refers to contribution of CDF from internal events, internal flood, fire, and seismic PRA.
2. CDF for short and long SBO from all hazards models are ~2.6E-6/RY and 3.1E-5/RY respectively. The total containment bypass probability is estimated by [(2.6E-6*.22+3.1E-5*.31) = [5.72E-7+9.61E-5] =1.02E-5.The LERF contribution is from the short sbo. It is estimated at 5.7E-7 or about 5.6%.
3. As discussed in Section 7.1.5, the probability of C-SGTR is about 1.3E-2 for short SBO with Inconel 600 materials and 8.9E-3 for Inconel 690. The probability of C-SGTR caused by a cleared loop seal due to RCP seal failures was also estimated at 2.5E-03. The overall probability of C-SGTR is estimated to be about 1.6E-2 and 1.1E-2 for short SBO and for Inconel 600 and 690, and 2.85E-2 and 2.0E-2 for long SBO and for Inconel 600 and 690.

Pressure Induced C-SGTR Scenarios

- Identify Scenarios where the delta P across the tubes is two or more times its nominal value during normal At-Power operation
- Develop scenario Frequencies from existing PRAs
- Identify simplified bounding pressure-temperature curves for the accident
- Estimate the C-SGTR probability equivalent to one or more tubes
- Modify the existing accident progression tree and related HRA to account for C-SGTR
- Re-quantify LERF and CDF

Summary of Pressure Induced C-SGTR for Westinghouse Plant

Pressure Induced Accidents	**f(IE) per year**	**P(CSGTR|IE)**	**P(CD|IE,CSGTR)**	**P(LERF|IE,CSGTR,CD)**	**Δ-CDF per year**	**Δ-LERF per year**
ATWS-Electrical	1.5E-5	0.01	1.6E-4	1	<1.0E-9	<1.0E-9
ATWS-Failure of rods	1.2E-6	0.01	1	1	1.2E-08	1.2E-08
SLBIC	1.0E-3	2.50E-03	3.2E-02	0	8.0E-08	0
Spurious opening of SG relief valves	3.0E-3	2.50E-03	3.2E-02	1	2.4E-07	2.4E-07
SLBOC	4.0E-5	2.50E-03	3.2E-02	1	3.2E-9	3.2E-9
High Pressure Feed and Bleed Scenarios	2.0E-5	0.01	2.5E-2	1	5.0E-9	5.0E-9
All IES – Total Contribution					3.4E-7	2.6E-7

Summary of Pressure Induced C-SGTR for CE Plant

Pressure Induced Accidents	f(IE) per year	P(CSGTR|IE)	P(CD|IE,CSGTR)	P(LERF|IE,CSGTR,CD)	Δ-CDF per year	Δ-LERF per year
ATWS-Electrical	1.5E-5	8.0E-3	1.6E-4	1	<1.0E-09	<1.0E-09
ATWS-Failure of rods	1.2E-6	8.0E-3	1	1	9.6E-09	9.6E-09
SLBIC	1.0E-3	4.0E-3	3.2E-02	0	1.3E-07	0
Spurious opening of SG relief valves	3.0E-3	4.0E-3	3.2E-02	1	3.8E-07	3.8E-07
SLBOC	4.0E-5	4.0E-3	3.2E-02	1	5.1E-09	5.1E-09
High Pressure Feed and Bleed Scenarios	2.0E-5	8.0E-3	2.5E-2	1	4.0E-09	4.0E-09
All IES – Total Contribution					5.3E-7	4.0E-7

PRA Summary

- PRA results indicate that the conditional containment bypass probability (given high-dry-low conditions) is approximately an order of magnitude greater for the CE plant analyzed by this study compared to the analyzed W plant
- PRA results indicates that Δ-LERF and Δ-CDF due to C-SGTR during DBA accidents prior to onset of core damage is expected to be <5.0E-7 per year based on somewhat bounding analysis.
- Plant features that reduce the likelihood of severe accident high-dry-low conditions (e.g., diversity in AFW system) can reduce containment bypass frequency
- The loop seal clearing for the pilot Westinghouse plant was considered for very large RCP seal failures and was found not be significant. No analysis currently available for CE plants.

PRA Insights

- The conditional probability of C-SGTR during severe accidents is reduced by:
  - Avoidance of large and deep SG tube flaws (particularly for Westinghouse-type SG designs)
  - Plant and SG design features (e.g., SG inlet plenum and hot leg geometry) that increase the temperature difference between hot leg and hot (or hottest) SG tubes during severe accidents (therefore increasing the probability of hot leg failure prior to SG tube failures).
  - Depressurization of reactor coolant system
- The conditional probability of C-SGTR during DBA accidents can be reduced by:
  - Avoidance of large and deep SG tube flaws

Other PRA Insights

- The critical size of C-SGTR Leak area may depend on the details of the accident sequences; an area of 6 cm^2 which approximately equals to guillotine break of one tube in a W-plant is assumed. The following issues were considered for estimating the critical area:
  - Sufficient to pressurize the secondary side forcing the opening of the secondary relief valves and subsequent failures to reclose
  - Sufficient to provide sufficient release path before RCS failure or vessel breach to be considered LERF (source term/release fraction)
  - Not too large to disturb the counter current flow
- Importance of SAMG Operation
  - Increasing the battery life to facilitate longer operation of TDAFW and SAMG operations
  - Probability of equipment survivability post on set of core damage – during severe accidents (PORVs, SG SRVs and ARVs)

Severe Accident-Induced Steam Generator Tube Rupture (SGTR)

Regulatory Implications

Antonios Zoulis, NRR/DRA

Consequential Steam Generator Tube Rupture (C-SGTR) Subcommittee Briefing

April 7, 2015

NRR Perspective

NRR:

- Endorses the completion of the draft NUREG for review and public comment summarizing the information and research developed over the past 5 years.

- Plans to develop Risk Assessment Standardization Project (RASP) Handbook guidance and update Inspection Manual Chapter (IMC) 0609 appendices to support risk assessments (SDP) for the Reactor Oversight Program

- Considers evaluating findings using generic issues processes – potential actions:
  - Issue information notice on issuance of final NUREG
  - Evaluate issue under the generic issues program