# Official Transcript of Proceedings

# NUCLEAR REGULATORY COMMISSION

| | |
|---|---|
| Title: | Advisory Committee on Reactor Safeguards<br>604th Meeting |
| Docket Number: | n/a |
| Location: | Rockville, Maryland |
| Date: | Friday, May 10, 2013 |

Work Order No.: NRC-4208 Pages 1-75

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

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

+ + + + +

604TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

FRIDAY

MAY 10, 2013

+ + + + +

ROCKVILLE, MARYLAND

+ + + + +

The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 8:30 a.m., J. Sam Armijo, Chairman, presiding.

COMMITTEE MEMBERS:

J. SAM ARMIJO, Chairman

JOHN W. STETKAR, Vice Chairman

HAROLD B. RAY, Member-at-Large

CHARLES H. BROWN, JR. Member

MICHAEL L. CORRADINI, Member

DANA A. POWERS, Member

JOY REMPE, Member

MICHAEL T. RYAN, Member

COMMITTEE MEMBERS: (Continued)

STEPHEN P. SCHULTZ, Member

WILLIAM J. SHACK, Member

GORDON R. SKILLMAN, Member

NRC STAFF PRESENT:

CHRISTOPHER BROWN, Designated Federal Official

ALI AZARM

KEVIN COYNE

MIRELA GAVRILAS

RAJ IYENGAR

KEN KARWOSKI

MICHAEL SALAY

SELIM SANCAKTAR

ANTONIO ZOULIS

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 ARMIJO: Good morning. The meeting will now come to order. This is the second day of the 604th meeting of the Advisory Committee on Reactor Safeguards.

During today's meeting the Committee will consider the following: First, consequential steam generator tube rupture; second, future activities and reports of the Planning and Procedures Subcommittee; third, reconciliation of ACRS comments and recommendations; and fourth, preparation of ACRS Reports.

The meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act. Mr. Christopher Brown is the designated federal official for the initial portion of the meeting.

We have received no written comments or requests to make oral statements from members of the public regarding today's sessions.

There will be a phone bridge line. To preclude interruption of the meeting, the phone will be placed on a listen-in mode during the presentations and Committee discussion.

A transcript of portions of the meeting is being kept and it is requested that the speakers use one of the microphones, identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

Our first topic is consequential steam generator tube rupture and Dr. Joy Rempe will lead us through the briefing. Joy?

MEMBER REMPE: Thank you, Mr. Chairman. Colleagues, you're well aware of the fact that the staff and industry have contributed considerable resources over the last two decades to better understand the safety implications and risk associated with consequential steam generator tube rupture events. Significant previous activities include an assessment of temperature-induced tube rupture of the reactor coolant system in the NUREG-1150 study, a representative analysis of the potential for induced containment bypass by an ad hoc NRC staff working group in NUREG-1570, and recent thermal-hydraulic analysis and risk assessments as part of the steam generator action plan.

More recently severe accident analysis performed as part of the state of the art Quality

Reactor Consequence Analysis Project provided additional insights into the likelihood and impact of subsequent failure of the reactor hot leg shortly following a consequential steam generator tube rupture event.

The last time we met on this topic was back in January 2011. Our Subcommittee for Materials, Metallurgy and Reactor Fuels heard about the research program proposed to address user need NRR-2010-005, which is entitled, "Support in Developing Analytical Bases and Guidance for Future Risk Assessments of C-SGTR Events." At that time our Subcommittee had several comments related to the research program being developed to assist risk-informed decision making related to the C-SGTR.

Today the staff is here to report on how they modified their proposed research program to address our comments and to discuss the progress that they've made for completing this research program. Note that this is just an information briefing on their interim progress at this time, but because it's been over two years since we've heard about this effort and because the staff will soon be completing this effort, I believe it's important for us to hear about their progress and provide our comments at this time.

And at this point I'd like to turn the meeting over to the staff, and I believe Dr. Raj Iyengar will start their presentation.

DR. IYENGAR: Thank you, Dr. Rempe. Good morning. I'm Raj Iyengar. I'm from the Division of Engineering, Office of Research. I'm very pleased to present some of our work and receive your comments on this Consequential Steam Generator Tube Rupture Program.

Just a little bit -- a minute on the background of this User Need. Dr. Rempe already captured the important objectives of this User Need. We had engaged with ACRS in 2010 Subcommittee and we received a number of comments, very valuable comments from the members. And subsequent to that meeting we met with NRR management and at their request to restructure the project so that we would take into account all the questions and comments we received from the ACRS.

Accordingly, research then devised a revised document which re-scoped and the effort so that we don't complete the full effort as outlined in the User Need, but to reduce the scope and determine the important key points or key salient things and so we developed a hold-points document which we shared with the members through Dr. Rempe. And subsequent to that we have been having some informal meetings with Dr. Rempe on our progress and updated up the recent activities that we accomplished in this project.

Just to capture this document, you already have received the hold-points document. In this document we have re-scoped the project to determine the relative effect -- relative importance of Westinghouse versus Combustion Engineering plans. and based on our understanding then we would either submit the results through NRR. NRR would reevaluate and see whether this project needs to go into the full scope of the User Need or there may be some interim steps we could take. So that was whole purpose of this exercise. And it's largely based on the ACRS feedback as well.

MEMBER REMPE: Could you show us where you are exactly in the progress right now?

DR. IYENGAR: Oh, in the progress we have evaluated the Westinghouse area here on the right-hand side and we have done some sensitivity studies of the RCS components to determine how it reflects the calculator calculations. And then the

CE side we have done a MELCOR calculation to determine thermal-hydraulics, And Mike Salay will talk a lot about that. And I think we have made considerable progress there as well. And we are also in the process of completing the RCS component analysis for the CE plant as well. And the calculator is all ready to go. I think the calculator, we will hear about the calculator today.

MEMBER REMPE: And one of the things; and maybe you'll get to it later, that I was interested in, you know, they're getting ready to do the level 3 analysis. And so some of its insights and tools will be interfaced hopefully with the level 3 analysis that they're going to do for the both of them.

DR. SALAY: We are interacting with the people at least, you know, from hydraulics about development of that modeling.

MEMBER REMPE: Okay.

DR. IYENGAR: Yes, thank you. Now I'll turn over to NRR. Antonio Zoulis will talk about the NRR perspectives. And following that we will have some technical discussions from Mike Salay on thermal-hydraulics and flaw distributions from Mirela Gavrilas. And then I'll talk a little bit about the RCS components. And then we end up with the calculator discussions.

MR. ZOULIS: Good morning. My name is Antonio Zoulis. I'm with the Division of Risk Assessment in NRR and I'm going to talk about our path forward after we've summarized all the findings. And we're developing a report to make sure that the research that was conducted over the past years, couple years is documented. And the basis for the findings and the assumptions will be part of the appendix file of the NUREG that we're going to be issuing.

And one of the options that we're thinking about going forward is taking that information and developing RES guidance and for the use of future significance determination or ASP analysis risk assessments. And then also possible actions for the generic issues, the implications of the Combustion Engineering steam generators or the shallow plenum steam generators is whether we'd issue a generic communication and if it's necessary to maybe further revise the Severe Accident Management Guidelines to emphasize the importance of making sure that the steam generators are covered to prevent the rupture either using a B.5.b-type equipment or FLEX, which is being recommended now as part of the Fukushima actions, the Fukushima Daiichi.

So those are the kind of things that we're going to be thinking about going forward. We're actually meeting today to discuss it with Research and we're not making any formal commitments, but that's what we're thinking about right now.

MEMBER REMPE: Do you have an estimated schedule when you think they will be completed? I know it's coming down sooner, but what's the date now?

MR. ZOULIS: That I think we need to still discuss long term.

So if there aren't any other questions, then I'll give it over to Mike.

DR. SALAY: In an earlier set of these graphs you provided you had -- for this same chart you had one additional point. It said, "Ensure agency has better understanding of G-SGTR for steam generators with shallow inlet plenums." Is that just a limited number of steam generator designs that you're concerned about?

MR. ZOULIS: Selim, are we talking about

the shallow plenum steam generators? Are those limited?

DR. SANCAKTAR: I think at this point it is, but really this is not an issue of one type of reactor versus another. It's not like a Westinghouse versus CE, but more of a --

MEMBER ZOULIS: Certain design feature.

DR. SANCAKTAR: Designs. And earlier I heard what I would call rumors that even some newer reactors might have different geometries of steam generators that might affect the results in an undesirable way, a consequential steam generator tube rupture. But we haven't really sat down and evaluated it and counted it.

CHAIR ARMIJO: Okay. So the tools developed by this project will be applicable to all --

DR. SANCAKTAR: Exactly.

CHAIR ARMIJO: -- kinds of steam generators, not just a particular set.

DR. SANCAKTAR: That's the intent.

CHAIR ARMIJO: Okay. Thank you.

MEMBER POWERS: Sam, our EPR seems to have what I would call a shallow greater than --

CHAIR ARMIJO: Okay.

DR. SALAY: Okay. Good morning. I'm Mike Salay and I'm going to talk a little bit about the thermal-hydraulics work that we've been doing.

We're focusing on a specific scenario of consequential steam generator tube ruptures. That's the severe accident induced steam generator tube rupture. It's the so-called high-dry-low scenario characterized by high primary pressure, dry steam secondary side and low secondary side pressure. The scenario which this results from is a station blackout, loss of off-site power, loss of generators. And within that we're looking at one, the long-term station blackout, which is the one that fits in the chart where the turbine-driven auxiliary feedwater works, is assumed to work until the assumption of battery failure at four hours. However, we also run the short-term station blackout simultaneously because it runs in half the time and you can look at that scenario and get results quicker and get a turnaround quicker.

Secondary depressurization can occur really by -- there are two methods that stand out that -- which can occur. You can either have the failure of a secondary for a MSSV, but what's characteristically done is that there's assumption of leakage on the secondary side to the containment, a small leakage to ensure that even when the valves do not -- are not assumed to fail open that the pressure goes down.

Also of interest is -- from people -- that people have asked that what happens if you don't have this leakage? Will your tubes still fail? So there on the right you see two severe accident natural circulation flows, one on the left for cleared loop seal and one on the right for an intact loop seal. The flows are different.

With the cleared loop seal gas is heated by a reactor, go up, use the full hot leg area, go through the steam generator where they're cooled, come down through the cold leg, down the downcomer and back to the core to be heated up again. This is a relatively efficient heat transfer to the steam generator tubes and is typically considered to result in failure.

The one we're focusing on here is the situation where you have counter-current circulation with the intact loop seal. In this situation the flow to the steam generator is more restricted. Hot gas is coming from the core. Have to go up and flow against colder gases. Returning it flows up, down the hot leg, up through some tubes where it's cooled, back through other tubes and then down the cold leg. And which --

MEMBER RAY: You said down the cold leg. Did you mean to say --

DR. SALAY: Sorry. Down the hot leg. Excuse me. And so the big thing we're trying to determine -- well, which is characteristically been looked at is what fails first. Do you have a tube fail first or does an RCS component fail first? If an RCS component fails first, you depressurize the system in the containment. The stress on the tubes goes away and you will not have containment bypass from that event.

Scenario is a little different if a tube fails first. And as in previous analyses with Westinghouse, if you have a -- where un-flawed tubes do not fail, you need a flaw to fail. Then you have one tube failing at the flawed tube failing. If you have one tube, it -- the power of this slowly -- doesn't depressurize the primary very rapidly, so your system stays at high pressure and it expected that your hot leg will fail shortly thereafter. However, if you have un-flawed tubes failing, then multiple tubes reach their failure set point at the same point, at the same time and then you can have a large -- multiple tubes fail and a lot faster pressurization. It's expected that this would result in large releases to the environment.

Combustion Engineering plants with replacement steam generators and other steam generators with shallow plenums are considered especially susceptible to the steam generator tube rupture with -- in case of the CE of a low high leg length to diameter ratio. So there's not a lot of opportunity for mixing a lot of hot leg. So your hot gases pretty much make it all the way through hot. And also there's a shallow steam generator inlet plenum and so there's not a lot of mixing there. So as a result the steam generator tubes see nearly the same fluid gas temperature as the hot leg and they're much thinner. They respond much faster. So they end up becoming much hotter than -- the structures are -- the tubes become hotter than the hot legs.

MEMBER SHACK: Is that why you identified the replacement steam generators, is because they typically have thinner tubes, or is the plenum changed?

DR. SALAY: It's the plenum change. And

so there's less mixing that occurs there. And so you just get a hotter temperature up into the tubes.

MEMBER SHACK: But the original CE design had a --

DR. SALAY: Had a -- was a --

MEMBER SHACK: -- deeper plenum?

DR. SALAY: -- deeper plenum, yes.

MEMBER SHACK: All right.

MEMBER STETKAR: Just quick, because I know you have a lot of slides to cover here and you're going to focus mostly on materials things. But because you do cast this in a risk-informed framework, the scenarios that you talked about focus only on equipment failures that get you the dry-low conditions on the secondary side of the steam generators. And I know we discussed this in the past; have you looked at actual emergency operating procedures? Because in many plants operators are instructed to depressurize the secondary side of the steam generators actively to try to get low pressure; for example, condensate or other sources of feedwater in there. So therefore, it may be very likely in these scenarios where you don't have secondary side heat that indeed you have low pressure on the secondary side open, permanently open.

DR. SALAY: Intentionally.

MEMBER STETKAR: Intentionally?

DR. SALAY: Yes. No, actually we --

MEMBER STETKAR: So in terms of vulnerability of plants -- and in fact there's one new plant design where automatic signals do that on every safety injection.

DR. SALAY: No, we didn't specifically look at the OPs, but we do look at the scenario where you do have this open to containment area and sort of represents -- can represent in some sense that --

MEMBER STETKAR: Okay. It's not open to the containment. It's open to the outside world.

DR. SALAY: Oh, it's open to the -- oh.

MEMBER STETKAR: Yes.

DR. SALAY: Intentionally?

MEMBER STETKAR: Yes. I'm an operator and I'm instructed to open my main steam relief valves to depressurize the steam generator. Suddenly those sitting around the plant see a large cloud of steam coming out into the atmosphere and second pressure goes down.

DR. SALAY: Oh. Oh, no. Okay. Yes, so

not leave open. Just -- yes, just open it.

MEMBER STETKAR: Open and leave open.

DR. SALAY: Oh.

MEMBER STETKAR: To get the pressure down.

DR. SALAY: And to keep it --

MEMBER STETKAR: That's considered to be a good thing to do in operator space.

DR. SALAY: So --

MEMBER STETKAR: It increases -- in FLEX space for example it allows me to get low pressure feed into my steam generators. I can't do that if the steam generators are at 1,000 pounds.

DR. SALAY: Yes. No, we didn't look at that specifically, but we do have different scenarios.

MEMBER STETKAR: It doesn't affect anything else if you're going to talk about materials. It does -- it make affect --

DR. SALAY: Yes, the scenario.

MEMBER STETKAR: -- vulnerabilities depending on how plants organize their accident response.

DR. SALAY: One of the things that we do we do look at different stick-open models, and so there are cases in which for MSSV stick-open and -- so which --

MEMBER STETKAR: Okay.

DR. SALAY: -- effectively have the same behavior and --

MEMBER STETKAR: They certainly have the same behavior. I'm just talking about the frequency of vulnerability.

MR. ZOULIS: In the scenario though you mentioned, the -- if this is a station blackout, you wouldn't be able to use condensate. The pumps wouldn't be available. So I think the probability of those sequences may be a lot lower than --

MEMBER STETKAR: After the fact you --

MR. ZOULIS: -- station blackouts.

MEMBER STETKAR: After the fact the PRA knows that I can't do that. During the event eyes on operator don't necessarily know that.

MR. ZOULIS: Well, you wouldn't have power. You won't be able to us --

MEMBER STETKAR: If you've got a firewater pump.

MR. ZOULIS: Right. Well, yes, but then --

MEMBER STETKAR: Most plants do.

MR. ZOULIS: You want them to take that action though, because you want them to cover the --

MEMBER STETKAR: Yes.

MR. ZOULIS: But I think what you're looking at is the sequences may be a lower probability of -- then if you have a station blackout, you have nothing else available and your FW turbine -- your driven pump fails. So once you add additional equipment and it's already proceduralized, your probability of the sequences I think goes down and --

MEMBER STETKAR: By the time the operator knows that the feedwater pump, the firewater pump has failed, he's already opened the valves.

MR. ZOULIS: But I think that would be --

MEMBER STETKAR: My point is the PRAs tend not to look at those sequences because they tend not to look for this high-dry-low condition. They don't look at the frequency of a sequence where things have failed but the secondary relief valves are open because the operators would have opened them. They just don't look at that because the PRA only looks for fail, fail, fail, fail, fail.

MR. ZOULIS: We have success with the successful --

MEMBER STETKAR: Okay. Anyway, I don't want to take up too much time. I just wanted to probe because we'd had this discussion a couple of a years ago and I wanted to see whether you had sort of probed, you know, the frequency in terms of vulnerability to these conditions.

DR. IYENGAR: Okay. Kevin has looked at that.

MR. COYNE: Kevin Coyne from the staff. One thing that may mitigate the concern somewhat is that for the purposes of the PRA if we have high and dry; we essentially assume low, that we impose this leak on the secondary side of the steam generator. So high and dry is going to lead to low for the purposes of the PRA.

MEMBER STETKAR: And I recall you saying that before, so that's --

MR. COYNE: Right. Now Mike has done several other sensitivity runs to see what the impact of retaining secondary side pressure is, but for the thermal-hydraulic ones that were used for the PRA analysis we assume high and dry at leads to a low.

MEMBER STETKAR: Leads to a low? Okay. Thanks.

MEMBER SKILLMAN: I'd like to follow up because I'm in agreement with John Stetkar. On loss of off-site power you lose your reactor coolant pumps and you trip. So you're decay heat begins to drop rapidly. You still have -- excuse me, you still have decay, but you stop the fission process. So you've got six or eight percent power, but it's decreasing very, very rapidly. And I think before this event you'd find, if you go through the procedures, operators will open the atmospheric dump valves and condenser dump valves. And they'll ride that as long as they can to get the secondary pressure down so as to make the sink that's available for the heat. So I think that there are some procedure issues here that are worth exploring because the outcome might give more time before you go to the loop seal in the primary. So I think there are some procedure elements of this that are worth exploring.

DR. SALAY: Okay. And, yes, here I have an animation of what MELCOR predicts and gives a little indication of how PFCI can switch over at any time.

CHAIR ARMIJO: Well, I looked at this video last night and I thought it was nice, but I didn't know what to look for. So if you could give me --

DR. SALAY: Yes. No, no. I'm going to talk through it, so --

(Laughter.)

CHAIR ARMIJO: A little narration would be helpful.

DR. SALAY: No, I will narrate.

CHAIR ARMIJO: It didn't tell me look here, now look there.

DR. SALAY: There are a few things I've added.

CHAIR ARMIJO: You get a little reward if you see the cat pop up quickly over in the left steam generator.

DR. SALAY: So I mean there are selected things from the output. I mean things I'm looking at, you have your system. You're looking at water levels, whether it's the void fraction of the water, whether you have bubbles, whether it's frothy or not. Also it would show fission product distribution through it. And I added pressures last night here for the secondary side B and the primary in the containment and atmosphere and didn't have time to put a scale on, but it's essentially -- that's one atmosphere out and, you know, you can sort of see relative.

Up top you have these cylinders. They represent the relief valves for the loop A, which has the pressurizer for these PORVs, for the SGPORVs. The large red cylinders are the SGPORVs and the small ones are the SRVs and the MS -- the different MSSV banks.

So I'll start seeing -- what you're going to see is the water level in the secondary drop. The SG secondary PORVs are going to open somewhat. They open a lot initially, but it's kind of an accelerated animation so you don't really see that. It happens kind of quickly. And your primary pressure is going to drop. Your pressurized level is going to drop. And so here it goes. Your level is dropping. Your pressure is dropping. As the secondary side dries out, you're losing your heat sink. Your system is going to heat up again. It's going to fill up the pressurizer and then the primary PORV is going to start cycling.

When the water level reaches the top, the PORV is going to stay open and you're going to start creating bubbles in the core which then start voiding out the system at the upper vessel, the tubes, the upper head, the hot and cold legs and then down through the core and the system starts heating up. And you can see the pressure. At four hours the battery is going to fail, so you're going to switch over from the PORV to the SRV and that's going -- that happened right there.

And then you're going to see a few things in rapid succession around six hours into the accident. There's -- you're going to see fission product releases indicated by a yellow cloud here. And then the secondary -- the hottest tube on the secondary side will fail shortly thereafter. It's indicated by first starting to go clear and then turning black when -- as it approaches failing, turning black when it fails.

So there's the fission product release. It's heating up. And there a tube fails. You have some fission products enter the secondary side and one of the MSSVs open the system. That cooled the system, but it's still heating up. Then the hot legs just failed and your accumulators kick in and flow in some water. And I'll turn it -- so, and I'll tap back. I'll tap you.

MEMBER POWERS: Mike, what was the key to getting the tube to fail first rather than the nozzle leaking?

DR. SALAY: Well, right now what we're getting, one versus the other is -- there are a few things that affected it that we're looking at -- is the velocity in the hot legs and tubes. I mean only a small change makes it go either way. And this is, you know, un-flawed.

One of the things that -- well, I'll go over it a little later, but is that the MSSVs don't open very much. They're not stressed. So it didn't seem like they would be sticking. So I'll just go over our task for the update MELCOR and fluid models for our representative plant, evaluate thermal-hydraulic behavior and some releases for some significant accidents. The ones that we're looking at are the station blackout and there was also a part of it that was to perform technical assessment of incore instrument tube failures on natural circulation for CE and Westinghouse plants, which was in the fore.

So what we essentially provide is to the other divisions is the thermal-hydraulic transient behavior which they use as boundary conditions for their calculations. And we also provide the initial check and then check of when tubes failure -- when tubes and when the hot legs fail, or when other components fail.

So as Dr. Rempe mentioned, we're -- this is in progress, so we're still updating it with recently updated CFD predictions. And we've been looking at the differences between what we're getting now and what was obtained in previous work to make sure we didn't miss anything. We're looking at the hot legs and steam generator flows and its impacts, as mentioned earlier that -- the tube failure time and which one fails first. And the previous RELAP analyses had different pressurizer drainage behavior, and I'm not sure why at the moment. And the other divisions also performed analyses on Westinghouse for T-H, thermal-hydraulic behavior. That was obtained from previous work from the NUREG-6995.

So, next the component failure time. It's very close. And one of the findings that we had relates to the assumption of this secondary leakage. The previous analyses considered once you have multiple tubes failure prior to hot leg that you'd get containment bypass and the operations actions, notwithstanding this, may not necessarily be the case based on the results we're getting.

If you have the leaky valves that were leaky steam generator, that was just sufficient to pressurize the SG behaviors. Steam generator behavior is different than we expected. And yes, the leak advances the tube rupture timing relative to your hot leg, but it also resulted when the tubes finally do fail, that you don't really stress the MSSVs. And the pressure still stayed high if you don't assume MSSV failure, a stick-open, and your hot leg still ruptures.

And so what we're getting again of a star caveat saying that whether -- which one fails first depends highly on parameters that we're looking into right now and comparing against other previous analyses. And it makes a difference. But what we're getting is currently that un-flawed tubes do fail before the hot legs for the high-dry-low, but the hot leg fails first for the high-dry-high. We're looking at different stress multipliers which represent different flaw sizes to simultaneously, even though they don't affect the T-H behavior, just to get an indicator of what size or what stress multiplier will result, translate to failure.

So if the relief valves, secondary valves were not opened intentionally, even if multiple tubes fail first, unless an SGPORV fails early, the predictions we were getting for volatile releases were low, significantly low or orders of magnitude than what we would call a large release.

And on the right you have a little diagram of pressure for long-term station blackout. It was only the right region that was really interested in, but building the whole thing. It's -- long-term station blackout your turbine-driven non-auxiliary feedwater works until the battery fails at 14,400 seconds or 4 hours. So your primary pressure goes down. Your secondary is at the PORV set point. Then after the battery fails, it goes to an SB set point. It boils off. And then as it starts to pressurize, the tube fails pretty far in the accident. Well, not tube. This is multiple tubes. This is a 20 tubes failure because it's un-flawed. And but depressurizing improves rates relatively quickly. The MSSV does open, but not -- I mean its again, not stressed. And I mean it's -- and yet the pressure stays high enough that the hot leg ruptures near the end of the transient.

And the last thing I want to mention is that MELCOR has not been predicting loop seal failures, although it's not something that we've been looking at exploring. And that's --

CHAIR ARMIJO: Michael, I had a question. What constitutes a flawed tube? How flawed does a tube have to be before it falls into this category of flawed?

DR. SALAY: Well, what -- again we provide initial estimates and the other divisions are doing the more analysis failure. But what we have is typically the way it's been handled before, and the way we're doing it now is that you have -- just apply a higher stress, and some flaw relates to higher stress in some manner. And they've looked at it indifferent ways so it's -- I didn't get -- I used similar stress multipliers to what was done before and just -- and let the other divisions really deal with the details. Yes, so two is -- a stress multiplier of two was considered a possible flaw from what I've heard.

CHAIR ARMIJO: So you define in terms of stress --

DR. SALAY: In terms of --

CHAIR ARMIJO: -- rather than some

geometrical defect?

DR. SALAY: Rather than a geometric flaw. So some geometric flaw would result in a stress multiplier, yes.

CHAIR ARMIJO: Okay. Somebody else has got numbers on that?

MR. ZOULIS: Yes, somebody else figures that out.

CHAIR ARMIJO: Okay.

MEMBER REMPE: Michael, it sounds like you are getting some unexpected results and you're still investigating them. Can you talk a little bit about what you're doing to try and reduce some of the uncertainties in --

DR. SALAY: Well, one of the things is primarily when I have time to work on it, it's looking at trying to resolve the differences between the different codes and making sure that what -- the results we're getting are matching the CFD and that -- because the CFD provides the detailed flow information that -- and you take the output of the CFD and then that's what's used to set the flows within the counter-current flows and the hot leg and mixing ratios in the inlet plenum and the flows in the tubes. And so wanted to make sure that matches up. And so just trying to see why -- trying to explain why we're getting differences from what was done before.

MEMBER SCHULTZ: Michael, when you say the results are very sensitive to parameters, is this different with different approach of calculations that you're using, different methods, or is it across the methods you're -- are you using different methods to try to --

DR. SALAY: Well, this is --

MEMBER SCHULTZ: -- figure out what the sensitivity is?

DR. SALAY: Well, this is -- you have a prediction of again for CFD on the flows to the hot leg. What if it's 10 percent off or something? That's enough to cause a difference between one failing first than another and --

MEMBER SCHULTZ: Yes. And then you get different overall results?

DR. SALAY: Yes, and it's --

MEMBER SCHULTZ: It's almost, you know, fail or no fail of the entire -- the important consequential results. Is that -- that's what I'm hearing?

DR. SALAY: Yes.

MEMBER SCHULTZ: And so what would you say is your time frame to try to figure out the sensitivity and -- it sounds like you're at a very complicated place and I'm not sure in terms of your analysis what you feel is your --

DR. SALAY: What I think is --

MEMBER SCHULTZ: -- opportunity for success here --

DR. SALAY: Well --

MEMBER SCHULTZ: -- to figure out what they're sensitivities are and determine which direction to go next.

DR. SALAY: What I think is that we will probably end up saying -- once we verify that that stuff matches, put some -- how -- we'll look at how much it varies and probably give uncertainty on which way it can go. And if there's nothing -- and it's sort of a competition for resources, but it seems like it's on the order of a month or two, but --

MEMBER SCHULTZ: Okay. Thank you.

MEMBER STETKAR: I understand you're competing for resources and things, but -- I don't understand anything about materials, but I know that we are right now in the throes of thinking about a more integrated response to long-term station blackout in the post-Fukushima era. And if some of these conclusions are real, it seems to me that we need to think carefully about how we instruct the operators to respond to a blackout situation, in particular with respect to active depressurization of the steam generators. Because if they win and get water in, that's a good thing, but if they try and don't get water in, this seems to be saying that it's not a good day at the electric factory, and it might have implications on how we try to manager those events. So do you have any comments in that?

DR. SALAY: No, it seems reasonable.

MEMBER STETKAR: Okay.

DR. SALAY: Yes.

MEMBER CORRADINI: Can I ask John's question differently? Is water always good? What you're basically saying there may be times when adding water everywhere might have to be thought about. Because the natural response would be -- is to put water into the secondary side of the generator.

CHAIR ARMIJO: Yes, and if you succeed, that's the natural response.

DR. SALAY: That's the expected

response.

MEMBER CORRADINI: That's the expected response, but if you're sitting here with a loop seal lock, you could exacerbate it.

DR. SALAY: If you don't get water.

MEMBER STETKAR: If you don't -- yes, if you don't get water in.

DR. SALAY: Right.

MEMBER STETKAR: I always thought getting water in is good, but operators' procedures tend to think in success space. You're doing something with the presumption that you will get a source of water. FLEX will work, for example. You will get the hose hooked up in enough time.

DR. SALAY: I guess you could say only open it if you have water available and the pump --

MEMBER STETKAR: Exactly. I mean that's part of my point. If indeed for some -- and that's counter-intuitive to the way most people write emergency procedures.

MEMBER REMPE: I have a question, too, about your comparisons with the CFD Code, and maybe this is something that we should have a longer day and discuss all of these things with the Subcommittee in more detail, but the CFD results, how well are they known? I mean is there data and, you know --

DR. SALAY: Well, yes, there's actually the Westinghouse 1/7th scale test that -- for Westinghouse products. I mean there actually is an experimental program being initiated in Switzerland that will look at both Westinghouse and CE geometry that intends to look at both Westinghouse and CE --

MEMBER REMPE: Oh, that's what I'm not aware of. So that would be good to give us a little more information about that in the future. Thank you.

DR. IYENGAR: Thank you very much. We are going to make a slight brief detour into the high-temperature behavior of some of these materials, just to put things in perspective for one reason, but the main reason is because this comes at the request of Dr. Powers to you. He wanted some brief about it primarily to say goodbye to his friend Dr. Bill Shack for all his contributions in this and other areas of materials. So we obliged.

MEMBER SHACK: So this is a goodbye talk?

(Laughter.)

MEMBER POWERS: No, just so that people

can remember this is important.

DR. IYENGAR: On, it's too ambitious to try to capture a vast field in couple of slides, but I just want to put things in perspective here. When the temperatures increase, the metals we deal with; the stainless steels or steels or even alloy 600, they exhibit different kind of complex behavior because of the internal dynamics that the crystals inside go through and the atoms go through. So as a result you would see when these metals are stressed in addition to the plasticity we normally think of, instantaneously occurring deformation, you also have a time-dependent behavior, whereas the increase of time the strain increases. And there are different regions associated with this and these regions are dependent on the type of mechanisms that are involved within the crystals themselves.

I don't want to get into all of those, but I would mainly focus on the point that as you increase the temperature some of these regions vanish and then you would probably have -- for example, higher the temperature you have less of the secondary stage and more of the tertiary stage, but you have damage accumulation. For one reason we do not have, not surprisingly, not much of data that was available for the regions that we are interested in, the severe accident region because that's -- here you don't design components for those. So we didn't have --

MEMBER STETKAR: Not of these materials.

(Laughter.)

DR. IYENGAR: Right. And more recently ANL through Dr. Majumba had done some tests which pushes beyond the envelope, you know, up to 1,000 degrees or so for some of these areas. And as you can see these curves are displayed here for stainless steel and carbon steel. And in all these cases you have what is called more of the tertiary stage which actually implies that some kind of damage is happening.

A clever man, for his Ph.D., Mike Ashby, in the '70s came up with I think a fantastic way of putting everything in perspective, what he called is a deformation map. He put -- you know, well, you can see, look at the map and see how the materials will behave at various temperatures. So he plotted the temperature, normalized temperature and versus, you know, shear stress for any given material. Here's an example of that just for illustrative purposes of stainless steel. So normally -- I just want to show you that we operate these steels in our RCS, for example, in this region.

So in this region we don't have much of the dynamics going on, not much action, but as the temperature increases, you start having many different mechanisms within the crystals going on. As you can see, some of them are shown here where you had these individual grains. You know, you start having dynamic recovery of dislocations and then these grains themselves start, you know, re-crystallizing and that will lead to quite a bit of deformation, time-dependent, because re-crystallization process is time-dependent. And then you would also have because of the diffusion process that you have something called climb of dislocations and cross-slip, which you don't normally see in these regions.

And there's another one which is quite dominant is diffusion of atoms and point defects and interstitial or substitute atoms from the grain boundaries, and that actually causes what's known as grain boundary sliding. It's contributes to a lot of deformations. It's a very fascinating field. That's one of the reasons why they design for high-temperature turbine applications. They design to meet the -- based on -- make a lot of single crystals where you don't have any grains at all. So there's no way you could have this kind of diffusion effects. So this block is mainly based on steady set conditions. And this changes quite a bit on transient conditions, which is what we're dealing with, so this is just a process for illustration.

And I want to tell you that the diffusion processes are not very well understood. This was a statement Mike Ashby made in 1982, that it's very poorly understood and it's -- I believe after doing enough work I also found that it's still true today. So if you really want to go into the severe accident range and materials, you probably need to have a better understanding of some of these mechanisms.

So this is deformation only, but there's also an accompanying process that happens during these high temperatures which lead to damage of materials. This is -- you have two different types of damage. There's ductility-driven damage. It is mainly caused by dislocation motions in these materials. This also happens at lower temperatures when you increase the stress, but it's more profound at higher temperatures because of certain additional mechanisms that are available for the dislocations to move. That's mainly because they want to reduce the activation energy and the internal energy.

Then you also have a diffusion-driven mechanism of failure that -- as I mentioned diffusion of atoms to the grain boundaries start nucleating voids and the voids grow and you have grand boundary failure. And here's an example of a nicely voided grain boundary, as you show these voids in the grain boundary that fails.

And Ashby, along with students, also developed a fracture map for many of these metals. Here's an example to illustrate that of alloy 750, which is same family as alloy 690 and 600. You can see that the regions of interest for us. In these you have lots of different mechanisms of these happening. So you could actually have damage happening in parallel with deformation which is not captured in many of our models that we use. And these effects are actually more profound when you have transient and multi-accident situations. This is for a simple case and in our case RCS components you see a multi-axial stress state which actually axialates all these processes.

And that's about all I want to say. I

think we would like to move on to some of our more technical work, and Mirela would go next.

MS. GAVRILAS: Yes, I'm Mirela Gavrilas. I'm branch chief for Corrosion and Metallurgy in the Office of Research. Charlie Harris is on annual leave. He is the one who prepared this work. Luckily for me I have Emmett Murphy and Ken Karowski in the audience. So they can field any questions that go beyond what I'm covering.

So the original risk calculator that will be discussed alter today used flaw distributions from steam generators with 600 mill-annealed. There are only four plants out there that are still using 600 mill-annealed tubes in their steam generators, and three of these plants are scheduled to have replacements within the next couple of years, so clearly the flaw distributions were obsolete and needed to be brought up to speed. And that's what my branch was tasked with. Specifically we needed to revise the probability distributions of flaws to better represent the new materials that are used in steam generators, these new materials being thermally-treated alloy 600 and also thermally-treated alloy 690.

The need for this revision is because

the flaw distributions in 600 mill-annealed differed in substance from the flaws in 600 thermally-treated and 690. Specifically 600 mill-annealed is more susceptible to cracks and just an order of magnitude all steam generators had thousands of tubes plugged because of indication of cracks, while the replacement steam generators made of these newer materials have anywhere from none to about 50 or so per steam generator. The other significant difference between the old and new materials is that 690 is actually much more susceptible to wear than its predecessors.

So again to give you an idea of what we're talking about, 600 thermally-treated plants have hundreds of indications of wear while the St. Lucie and San Onofre 690 steam generators have thousands of indications of wear. On the other hand, there are both 600 thermally-treated and 690 that have hardly any indications of wear.

So regarding the actual behavior of the tubes during severe accidents we did not do additional work because while there's a shift towards wear type of flaws, we felt that the flaws were -- the failure of the wear flaws are adequately captured. So from a pressure burst perspective the wear flaw is modeled similar to the crack and there is data that was collected by Argonne several years ago that compared a rectangular flaw with an elliptical flaw and showed that assuming an elliptical-type flow to be rectangular is adequate, if somewhat conservative. So that's the basis for that. And in terms of creep where flaws are simply simulated as erosion of the thinning of the wall over a significant area. So that's pretty well understood.

MEMBER RAY: The alloy 690 tubes, aren't they thicker walled than the 600?

MS. GAVRILAS: Ken Karwoski said no.

MEMBER RAY: No?

(Laughter.)

MS. GAVRILAS: He shook his head and said no.

MEMBER RAY: I thought that was the reasons why the San Onofre had -- part of the reasons for their design went different is that the alloy 690 wasn't as strong as the 600.

CHAIR ARMIJO: It's the heat transfer coefficient.

MEMBER RAY: It was the heat transfer coefficient? So that made them thicker?

CHAIR ARMIJO: It made the walls thinner.

MEMBER RAY: Thinner? Okay. So now we're going -- that's the wrong direction. And so are the stress rupture properties/creep properties for the 690 the same as 600 in a non-flawed condition?

MS. GAVRILAS: Ken, do you mind answering this?

MR. KARWOSKI: This is Ken Karwoski from NRR. We currently have a research project with Argonne National Labs where we compile the creep rupture properties of alloy 690. That task just recently started and we've seen some preliminary results, but I'm not prepared to talk about those today. But we are compiling that information.

MEMBER RAY: Okay. But the negative thing with this stuff, we're starting off with thinner-walled tubes. So whatever flaw you have --

MEMBER STETKAR: Well, or more heat transfer surface. That's one way to deal with it.

MEMBER RAY: Unless it -- okay. So it could --

MEMBER STETKAR: It could have more tubes.

MEMBER RAY: You'd have more tubes and you'd -- each reading depends on the temperature of the tube and the stress on the tube which would ultimately stress your tube.

CHAIR ARMIJO: Yes, I guess a basic question is would -- everything else being equal at one of these events, is an alloy 690 steam tube more vulnerable than the old alloy 600?

MS. GAVRILAS: I think the assumption was that the old alloy was more vulnerable. Ken, can you build up on that?

MR. KARWOSKI: Yes, I guess first with respect to the tube wall thickness, in general most plants would have the same tube wall thickness in going from alloy 6 -- in -- during their replacement projects some did change it. In the case of San Onofre they had what I'll call this thicker-walled alloy 600 than most plants. And so they went to the more conventional size tubing in their replacement steam generators.

With respect to the heat transfer coefficient, yes, you accurately pointed out several plants have done different things to accommodate the lower heat transfer coefficient associated with 690. Some plants have increased the number of tubes.

Some have increased the length of the tubes. Some have reduced the thickness of the tube wall material. So it all depends on what a specific plant chose to do or did not.

CHAIR ARMIJO: Thank you.

MR. KARWOSKI: And with respect to the creep properties of alloy 690, like I said, we're studying -- we're compiling the existing data that's out there to see if there is a good mix between them.

CHAIR ARMIJO: Okay.

MEMBER SHACK: There was another question I had though. You made a comment that there was more wear in the 690. Now is that a difference in the actual wear properties of the material, or that's really the thinner -- I mean the smaller-diameters tubes, more and more vibration, more fretting? Is it a design or a true material difference?

MR. KARWOSKI: I think all we were trying to point out there is there's some replacement steam generators with a lot more wear than the original steam generators. Certainly there are replacement steam generators with no wear. So we won't try to say that there's a higher degree of susceptibility of 690. We're just pointing out the observation that some plants with 690 tubes have a lot more wear in the replacement steam generators than they had in the original steam generators.

MEMBER SHACK: For whatever reason.

MS. GAVRILAS: To get back to what Charlie actually did for this, he looked at CE 690 plants, three CE 690 plants, three Westinghouse 600 thermally-treated plants, and three Westinghouse 690 plants. And he looked at ISI data that the Agency received to come up with the new probability distribution of flaws characteristic of these plants. His intention initially was to come up with a correlation that described the flaws, the number of flaws as they evolved as a function of operating history, but what he found in looking at the data was, one, the data scatter was huge; and two was there were some ISI reports that weren't entirely reliable.

So what he did instead is come up with a description of a -- typical is what he calls the -- sort of typical CE 690, typical Westinghouse 600 and Westinghouse 690. And then he assigned to the flaws an error band that was intended to capture the variability that we would see in each of these steam generator types. And that error can be propagated into the final risk number through the calculator.

So that's the work that was done in the context of the consequential steam generator tube rupture by our branch. And if you have additional questions, I'll try to field it, or Ken and Emmett will do so.

DR. IYENGAR: Thank you very much. We have just about a half hour left and I think --

CHAIR ARMIJO: I have a quick question. For your analysis in your model, is there a specific location in the steam generator where you would expect the tubes to rupture, at the hottest location or something like that? And then in your flaw analysis do you specifically look for flaws in those regions of the steam generators? Is that a part of your model?

DR. SALAY: I'll answer that, the first part. Yes, there's a specific region. The hottest part is -- there's -- right coming in on the hot side. The center of the hot plume are the hottest tubes right as you come up over the tube sheet, yes. So right in there is the hottest part.

CHAIR ARMIJO: So then when you do your ISI is that where you pick the flaws to get an idea of --

MS. GAVRILAS: The ISI picks the flaws, but it still allows the tube, but when it comes to describing the population of flaw, we describe them by type. So axial, circumferential, wear and location.

CHAIR ARMIJO: Okay.

MS. GAVRILAS: So, yes, it goes by size and location.

CHAIR ARMIJO: So when you run this model, when you're finally running this model, you pick the flaw distribution in the region of interest where it's likely to be the hottest and to see --

MS. GAVRILAS: That I'm not sure about. The data we have, but I'm not sure if we do that.

DR. SALAY: I think what's been done before is that they just assume that the flaw occurs at the hottest part. What we're doing with the CFD and the MELCOR analysis, we can provide a temperature distribution that -- and so that we're trying to evolve the capability to be able to link up the flaws with the location. I think it may be done to some extent already and I now Selim or Ali -- you know, it's done in the calculator and --

DR. IYENGAR: Okay. I think Ali will go

through that in his presentation.

This is task is mainly to provide an independent check on the failure time of RCS for both Westinghouse and CE plant.  So we were interested in developing more rigorous finite-element models and, based on that, examine the effects of the thermal-hydraulics on failure locations, as well as try to put in a weld overlay on, you know, the weld to see if that has any influence on the failure time and the location.

We developed a finite-element model, a full-system model for the Westinghouse plant as shown here.  It's based on three-dimensional shell elements.  We also developed a smaller sub-model here of the hot legs, because that's where the action was.  And it was better to do a lot of sensitivity studies because our larger model had some -- had a long run time and converging issues.

The material behavior we observed was a total strain occurring any point in the material is based on elastic, and the rate-independent plastic behavior indicates instantaneous plastic strain that I pointed out, plus a creep strain which is time-dependent as well as rate-dependent.

The analysis procedure is outlined here.

We initially ran the model for a steady-state condition and then we took the time-dependent gas temperatures from RELAP as a boundary condition. This is for the Westinghouse. For the CE we take the MELCOR thermal-hydraulic transients. And we also use a time-dependent heat transfer coefficient which is actually a fully developed solution. So with that, these heat transfer coefficients were adjusted to ensure that we capture the transient effects well.

MEMBER SHACK: You felt you were comfortable with the shell elements even right up to the nozzle?

DR. IYENGAR: Yes, we actually are running another independent check with 3-D elements. And some of these calculations were done -- similar calculations for different transients were done at ANL and they had done some extensive analysis to show that the shell in this case was quite adequate. You know, for example, used five section points for the thickness versus seven section points. And I think we ran some 3-D calculations. We see similar effects. And whether failure occurs is in such a short time I think it's pretty -- we're pretty comfortable with that.

And as I mentioned, the heat transfer coefficient was adjusted based on the NUREG-1922. We also modeled the heat loss to the ambience through the insulation -- I mean even the insulation due to convection and radiation. We though the constants from that -- from Mike Salay from his earlier experience with RELAP and MELCOR. And combining all these together, we ran a thermal-mechanical simulation for the short-term station blackout.

Here I'm showing you the creep and plastic strains at 12,300 seconds in the full model. As you can see that, yes, this is the area here, the hot leg. And it's not surprising because this area here that was also three times thicker than the stainless steel hot leg here.

We wanted to plot for the damage. The way we defined damage was using a Larsen-Miller Parameter which we got the constant from the most recent tests from ANL. So I think we have a little bit more confidence in that because these were run at temperatures up to 1,000 C, which in the earlier cases we did not have the open literature. And from that we estimated time to failure.

And for the sake of plotting here we

plot the damage average to the thickness to determine the failure time. And you can see I plotted here for the average -- I mean the failure in the hot leg region at this time. And as I mentioned earlier, the system level model simulations are computation intensive and all components convergence because of rapid transience that happens around 12,000-15,000. They're not well-suited for understanding sensitivities. And since failure location is in the hot leg region, we tried a sub-model, and the sub-model kind of predicted similar results as the large-scale model.

Here is an example of that. They show without a weld overlay and with a weld overlay. The weld overlay is not very visible here because of the shell model, but this cartoon here shows you that the weld overlay is applied over a small weld region.

CHAIR ARMIJO: Why would anyone expect a major difference?

DR. IYENGAR: Excuse me?

CHAIR ARMIJO: The hottest part is far away from the weld overlay.

DR. IYENGAR: Yes, I don't expect a major difference, but --

CHAIR ARMIJO: It seems like --

DR. IYENGAR: Well, but we wanted to, you know, have a check, I think a check on that as well. That was the reason why we ran those and it was of interest.

CHAIR ARMIJO: Now is this pipe -- these are all insulated pipes, right? And if you really wanted that region, the hot region to fail first, why wouldn't you just heavily insulate that part so it would get hotter faster and give you some advantage?

DR. IYENGAR: Oh, well again, this is for one scenario, right? I mean if you have to do it for different scenarios, you might have different --

CHAIR ARMIJO: Yes, but would it make any difference, I guess is a question in a sensitivity study? You know, right now you have certain insulation properties --

DR. IYENGAR: Yes.

CHAIR ARMIJO: -- throughout the system --

DR. IYENGAR: Right.

CHAIR ARMIJO: -- and if you said, well, I'm going to really insulate this part because if I want anything to fail, I want that thing to fail rather than the tubes.

DR. SALAY: I think these may be stresses.

DR. IYENGAR: This is a damage --

DR. SALAY: Damage. So it's a damage index.

CHAIR ARMIJO: It's a damage index, but ultimately it's stress and temperature, right?

DR. IYENGAR: Stress and temperature, because that's what it's dependent on.

CHAIR ARMIJO: Okay. Anyway, just --

DR. IYENGAR: Anyway, the failure location and the failure time didn't change very much. And we ran additional cases to show that we are all in the neighborhood of 12,000 to 12,500 range. The system model we ran, I already showed you the one with the weld overlay and without the weld overlay.

One of the things is that I -- oh, this is actually no special resolution here. Creep only. When I turned off the plasticity and had only creep time-dependent behavior -- this is of interest mainly because there was an earlier thought that if you didn't have creep, if you had only plasticity, that the failure would happen at a much longer time. That actually is not quite true based on our simulations only because there is a significant redistribution that occurs during creep. And in addition to that, because of the context of current circulations the redistribution effects are much more significant because you have a constant pressure applied over time and that has to be balanced. And because of that, you have -- when you have only creep you actually have much shorter times.

And all times compared quite well with the median failure time estimated by the CE calculator which you're going to hear about in a minute.

MEMBER SHACK: What's your damage model for combined creep and plasticity?

DR. IYENGAR: Well, all the damage is the same based on the Larsen-Miller Parameter, which is stress and temperature. The stress changes because of the redistribution effects when we have creep. If you don't have that in the combined --

MEMBER SHACK: So you just integrate the Larsen-Miller Parameter over the --

DR. IYENGAR: Yes, it's actually postprocessing plant information. As you go --

MEMBER SHACK: Okay. So the --

DR. IYENGAR: -- you do not have a damage --

MEMBER SHACK: -- plasticity really enters in because it changes the stress?

DR. IYENGAR: It changes the stress and it changes the -- it gives you it instantly, yes. It relaxes the stress.

MEMBER SHACK: Relaxes the stress, yes.

DR. IYENGAR: But I think the more significant thing as you'll see later in my report is that the redistribution effect is so significant because of counter-current circulation. You have hot on the top and cold on the bottom and it's really -- that I think changes the scenario significantly.

And in summary, we've done our paces with Westinghouse and then we are almost completing our CE analysis based on the thermal-hydraulic input from MELCOR and our draft report is in progress.

If you don't have any questions, we can move on to the risk assessment topic.

DR. SANCAKTAR: We would like to give you a summary of the work done by the Risk

Assessment Group. We have contracted an organization to work on the risk portion of the tasks. And the main principal analyst is Ali Azarm, who is sitting over here, and he will be presenting the work done.

We have worked with the information available from different disciplines to produce three draft reports which will be distributed within the NRC cognizant and interested offices for comments and review. I'll let Ali to proceed and explain.

DR. AZARM: Thank you, Selim. Okay. The work we did is to support NRC to come up with a risk tool. This risk tool was supposed to submit the probability of consequential steam generator tube rupture for two types of accidents; design basis accident, pressure-induced failures and severe accidents, what you've heard the most today about.

We basically were supposed to come up, and we are still working on it; it's a work in progress, with the probability of containment bypass and defining the fraction of that containment bypass that we believe is going to happen early enough and is large enough release that constitute a LERF.

So that was the objective we had. I'm

going to go very fast through this stuff. Stop me at any time. Basically this is a pictorial of what is the approach we are doing. We basically start with a specific set of flaws, or we can simulate flaws. And I will tell you shortly after how we do that.

So we can have a set of flaws that we feel -- we select our accident sequence and we us MELCOR or RELAP or T-H, which is basically pressure and --

MEMBER STETKAR: How do you select your accident sequence? Somebody else gives you that?

DR. AZARM: We basically initially went through and identified all the accident sequences from level 1 PRA we believe is candidate.

MEMBER STETKAR: And whose level 1 PRAs?

DR. AZARM: Yes, so --

MEMBER STETKAR: From -- no, from whose level 1 PRAs?

DR. AZARM: Mainly we focus on the SPAR.

MEMBER STETKAR: On the SPAR? Okay. Thank you.

DR. AZARM: You know, then they decided.

MEMBER STETKAR: Okay. Thanks.

DR. AZARM: The T-H that we need is

basically pressure, temperature for hot leg, surge line and the hottest tube, cold tube, average hot tube, etcetera. That is fed to a software we call calculator. And the calculator basically comes up with a probability of steam generator tube rupture, probability as a function of time, of hot leg failure, and surge line failure. And again as soon as you say probability, these are deterministic codes. That means we have to be very, very careful to account for all the sources of uncertainties and all the probabilities.

For calculators equipped from bunch of libraries, of default values, of certainties of parameters and the models. We have libraries of materials properties alternate as stress as a function of time for 600, 690. Some of them could rupture. Equations. Some of them come from NRC work. Some of them were not available. Come from public document. And, you know, is right now placed hold until you get better models.

So once those probability comes out of this software, what we do we'll start to combine them to come up with a containment bypass probability and then from that and the timing of accident sequence evacuation, the discussion that we went to regarding release path to the open steam safeties, those are basically accounted for to calculate LERF.

MEMBER STETKAR: Probabilities that they might have failed or stuck open not considering the fact that the operators might be very likely to have opened that path because the SPAR models do not account for that?

DR. AZARM: Actually we are not there, but we were thinking -- let me tell you one thing and -- at least what PRA assumes right now is that if you have one tube failure or a an equivalent area of one tube failure, even with the SRV open, your primary pressure, because it's restricted with the area of that one tube, is not going to drop below -- no, 1,200-1,300 PSI. So it's still used with the hot leg failure. The problem comes multi-tube failures or you have a SRV open. Then we basically assume -- and you will see that -- again this is very preliminary and we are going to work very closely with this. So we did account for the possibility of SRVs either fail open or open. We also look at SAMG and --

MEMBER STETKAR: Okay.

DR. AZARM: -- what you mention regarding to depressurize the secondary and all that to put firewater in. And then you can do the firewater. However, those event trees have not yet been developed --

MEMBER STETKAR: Okay. Thanks.

DR. AZARM: -- even though it's discussed.

MEMBER STETKAR: Thank you.

DR. AZARM: Okay?

MEMBER SCHULTZ: Ali, from what you just indicated, looking at Michael, you haven't benchmarked or compared or gotten input from the thermal-hydraulics work that we heard earlier? You have your own models?

DR. AZARM: No, we did get --

MEMBER SCHULTZ: No? You have collaborated?

DR. AZARM: Everything we are doing is informed by thermal-hydraulic models and the work that Mike has done. Informed doesn't mean we exactly copied it. You know, so our assumption is informed, but as, you know, we discussed, on multiple tube right now because they didn't assume any SRV stuck open or fails open, they don't get released. But we do account for it.

MEMBER SCHULTZ: Yes. Okay. Thank you.

DR. AZARM: That's for everything we use.

MEMBER SCHULTZ: Yes.

DR. SALAY: Yes, we periodically meet and coordinate that.

MEMBER CORRADINI: So can I say that back to you just so I get it? So what you're really saying is you have then these more detailed calculations with MELCOR informed by the CFD that gives you a range of results. Knowing those results you put those results in as an input, but there's a bit of back and forth as these are improved, whatever you put them in?

DR. SALAY: Correct.

MEMBER CORRADINI: Got it.

CHAIR ARMIJO: And before you leave that, in the calculator the inputs for the material properties and plant-specific design information, you're also addressing at the same time the piping, right, the hot leg piping, whether its properties, its dimensions and the race between failure of the tubes and failure of the hot leg?

DR. AZARM: Correct.

CHAIR ARMIJO: Okay.

MEMBER SCHULTZ: Ali, I understand uncertainty. I don't know where you're getting the uncertainty information and I don't quite understand the parameters.

DR. AZARM: Okay. One by one.

MEMBER SCHULTZ: Yes. Thank you.

DR. AZARM: For example, when we look at the sources, two, three different sources for 690, stress as a function of temperature there are some variation, data we observe, we account for that parameter.

MEMBER SCHULTZ: Okay.

DR. AZARM: When you look at some of the models we have used for hot leg we have not used yet, you know, just used the work you heard as a consistency. The models we use it comes from EPRI document and EPRI document puts the modeling uncertainty in what they propose is an empirical equation. Whenever we had empirical result or empirical fit for the leak area or size of leak area, when you look at the actual data when it was available, there was large uncertainty associated with that. So that is built in, but before the uncertainty.

MEMBER SCHULTZ: Thank you.

DR. AZARM: You're welcome. I'm going to go through very fast. The first one basically says we use any model that's available at this time. So for failure fracture mechanic models we use NRC models, we use EPRI models, etcetera. Flaws --

MEMBER SHACK: What do these EPRI models for the hot leg and surge line look like? I mean are they --

DR. AZARM: It's basically --

MEMBER SHACK: It's a pressurized tube and they just --

DR. IYENGAR: It's just a Larsen-Miller Parameter applied, you know, to calculate the stress based on the pressure, the whole stress.

MEMBER SHACK: Right. Okay.

DR. IYENGAR: And you have stress, you have the temperature. It's the PLM parameter.

DR. AZARM: Exactly. It's a PLM. It's a Larsen-Miller creep rupture equation and they have given us the empirical numbers there to calculate the stress based on a pressure temperature. And then we put it there. And then the good thing is that it has an uncertainty when it gives you output. It gives you output first minus some variance. So the only interpretation we have done we have assumed that variance coming from a normal distribution and the sample.

MEMBER SHACK: Now from your work you assume that that failure area is far enough away from the nozzle, that that's a good approximation for the stress and the tube?

DR. IYENGAR: Yes, actually what makes it even sooner is -- and from our work we're finding that the stress-based predistribution and the multi-action effects are more profound. So that actually decreases the failure time as in comparison with what you see in the EPRI report.

MEMBER SHACK: Okay.

DR. IYENGAR: So actually it's we're in a better position in this circumstance.

DR. AZARM: Okay. You've heard the presentation on the steam generator flaw. We received that data, that limited data. For our job we needed to do some statistical estimation. So what we have right now, we have a flaw generation rate as a function of a steam generator service life, what is called EFPY, effective full-power year of operation. So if you tell me that steam generator is 10 years old, 10 EFP, then I can basically tell you we expect to have 50 plus/minus so and so.

We also have distribution, gamma distribution fit for the length and depth of the flaws. How good they are, hopefully we have additional data to check that. This is based on things that they're doing. And that's what we use for simulation.

PRA consideration. The first area, critical steam generator leak area, we're just discussing the difference between one tube and multiple tube. We do look at SAMG actions for the PWRs. Really for this issue there's not that many because if you want to get water inside the vessel, you have to do make up to RWP. You need the power back. So that's almost impossible. The only other thing you can do is depressurize the steam generator. So we have gone to SAMGs and we have discussed that. Again we haven't modeled these.

We have looked at the -- for these -- most of these SBO scenarios when is going to be the declaration of general emergency and if we can have effective evacuation? So Mike talked about the LTSBO or SBOs that turbine-driven AFW is initially available. Usually by the time you get released is 12-15 hours later and we think the evacuation is effective so LERF is much less than that.

We have been doing some analysis for two cases study. Okay. Again additional for the calculator it can look at flaw tube as well as pristine tube. It can -- at each run you could have two different group of temperature, average cool tube and average hot tube. So if you want to do average cool, average hot and hottest tube, right now we do two runs, two separate runs.

The calculator has a document that in detail documentation, some of the early version was reviewed by ANL and NRC staff. We have -- since then have added the EPRI models, documented the EPRI models on hot leg and surge line.

For right now again for a very simplified model we use a five-factor equation. We basically look at some level 1 PRA and coming up with a frequency of entering to level 2. We calculate the consequential steam generator tube rupture probability from calculators. Then we look at the possibility of primary stay pressurized for hot leg failure or a stuck-open relief valve in primary in order to come up with the probability that primary doesn't get depressurized. And we multiply those three numbers by containment bypass and then we go from containment bypass to LERF. We account for SAMG and EVAC, evacuation.

What we believe is important parameters is; you heard them, mixing in the steam generator. This is this whole shallow steam generator inlet. Mixing in hot leg. Okay. And this is the ratio of hot leg. We think pressure drop, that constituted a flaw. Heat transfer and I was, you know, surprised that somebody asked the condition of insulation because that really tells you how hot hot leg gets. So those are basically the thermal-hydraulic issues.

We are very concern about the performance of primary and secondary relief valve. Pre and post up to the core damage. Remember, these are all after core damage and endures harsh environment. The chance of these things failing or sticking open or jamming closed is going to be very high.

Duration of DC availability is going to play important role. Effectiveness and success with SAMG activity. And we do not yet know the effect of FLEX and EDMG on the equipment. We have to for sure add to this list.

MEMBER SCHULTZ: And the condition of the tubes does not show here, or did I miss it?

DR. AZARM: No, we --

MEMBER SCHULTZ: The flaws in the tubes or the condition of the generator.

DR. AZARM: We missed that. We should have put that.

MEMBER SCHULTZ: So it should show there? It should show there?

DR. AZARM: Yes.

MEMBER SCHULTZ: Okay. Thank you.

DR. AZARM: We did two cases study. I don't call it representative. There's nothing representative. There was one Westinghouse, one CE. Just remember the Westinghouse and CE are almost in opposite spectrum which come from first four items I mentioned the last slide. And also when you look at the CE plant you looked at, it's kind of odd. It has two turbine-driven AFW trains, so it's not really susceptible as much to LTSBO. However --

MEMBER RAY: What is the CE plant? Do you know which one it is?

DR. AZARM: Yes.

MEMBER RAY: Because that's not true of all CE plants, for sure.

DR. AZARM: No, not at all.

MEMBER RAY: Okay.

DR. AZARM: Not at all. The other thing --

MEMBER SHACK: He didn't say it was representative. He just said it was a case study.

MEMBER RAY: Okay. So two trains as opposed to one train makes a big difference.

DR. AZARM: Big difference, yes. Yes, we noticed. The other thing it will --

MEMBER RAY: There are plants that have just one train. That's why I asked.

DR. AZARM: The other thing about this CE plant, when you look at the external event, even back in IPEs, there are two unit size. They have -- large fraction of SBOs aren't affecting both sides. So this issue is going to be affecting both units.

Just for the comparison, since these two plants are opposite, we get almost containment bypass probability very low for the Westinghouse plant and almost one for the CE plant. And that's because of thermal-hydraulic and everything else we looked at. However, remember, we go to LERF, they are different, but as -- not as really as you see here because they have two turbine-driven and large fraction of long-term steel is not LERF.

Okay. So that's basically, I believe.

Yes, basically we are saying that we feel this is very design-specific, condition-specific. We think we are better understanding that this the influencing factor in the integrated PRA domain and we have to assume that there will be the FLEX system and try to integrate that.

Okay. Thank you.

MEMBER SHACK: You mentioned that you had regular failures of not just creep. Can you do an operational assessment with the tool, too?

DR. AZARM: We actually, both Selim and I, have looked at, you know, like different scenarios, like even normal operation and, you know, we can tell you if you have 90-percent deep crack or 90- percent deep wear, you are going to leak or the tube is going to fail even under psi 30. So, yes, we can do the same thing. Yes.

MEMBER REMPE: Are there any other questions?

(No audible response.)

MEMBER REMPE: My feeling that I'd like to have it on the record and have your feelings documented are that we should have another Subcommittee meeting that's even a day, or at least a half-day before this comes back to the full

Committee. And in fact this would have been a Subcommittee meeting if the schedule had allowed it. If you concur, then perhaps we ought to think of something within six months or something, or whenever it seems appropriate depending on your progress.

DR. IYENGAR: Yes, can we get back --

MEMBER REMPE: Absolutely.

DR. IYENGAR: -- and discuss this and back to you?

MEMBER REMPE: Yes, I just think that it would be good --

DR. IYENGAR: Sure.

MEMBER REMPE: -- to go into a lot more details, although I think it's good to have everybody's -- to hear a brief overview today.

If there are no other comments or questions, I'll turn it back over to you.

MEMBER STETKAR: Thank you very much. Appreciate it. I was a bit concerned about time, but you appreciate the staff.

And with that, we will take a break and reconvene for P&P at 10:20.

(Whereupon, the above-entitled proceeding was adjourned at 10:02 a.m.)

# Consequential Steam Generator Tube Rupture (C-SGTR)

**Full Committee Briefing**

**Advisory Committee On Reactor Safeguards**

**May 10, 2013**

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

United States Nuclear Regulatory Commission
*Protecting People and the Environment*

# Staff Opening Remarks

Raj Iyengar, RES/DE/CIB

# Purpose and Background

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

- 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 identifying “hold-points” to resolve near-term deliverables before proceeding with the full scope project in January 2012
- Informal meetings with lead ACRS member for C-SGTR issues (Dr. Rempe) held January 2012,  January 2013, and April 2013.

# Revised RES Plan

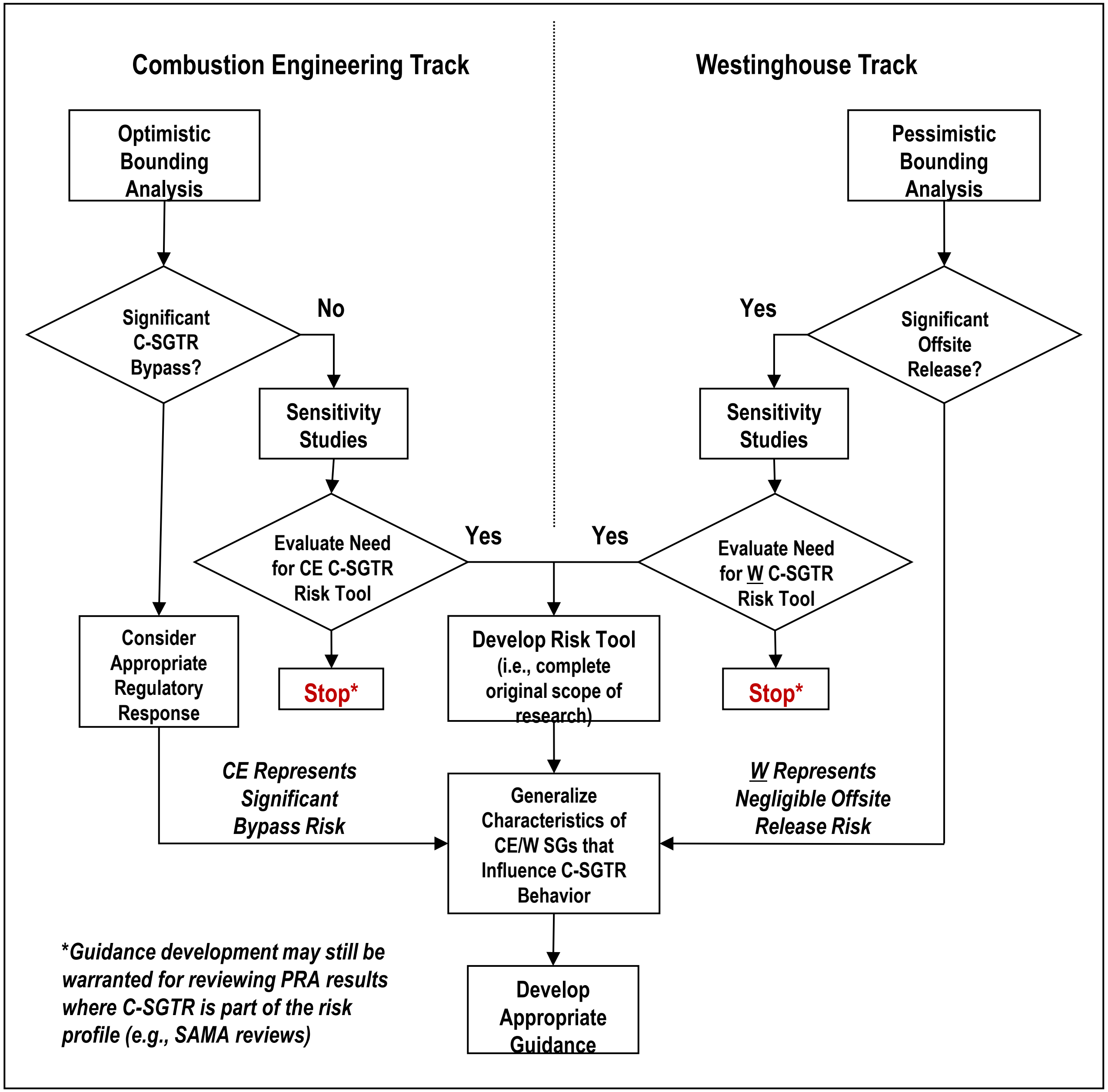



Illustration of the revised research plan of January 2012

# Outline

- NRR Perspective
- Thermal-Hydraulics
- SG Flaw Distribution
- High Temperature Material Behavior
- Behavior of RCS Components
- Probabilistic Risk Assessment Considerations

# Origin of User Need, User Need Details & Regulatory Implications

Antonios Zoulis, NRR

# C-SGTR Path Forward

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

- Develop a summary report compiling key insights and state-of-knowledge

**Options Under Discussion**

- 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

- Evaluate findings using generic issues processes – Potential actions:
  - issue generic communication
  - revise SAMGs to emphasize importance of using additional equipment (FLEX/B.5.b) to extend battery life and/or inject water into steam generators using diesel driven pumps.

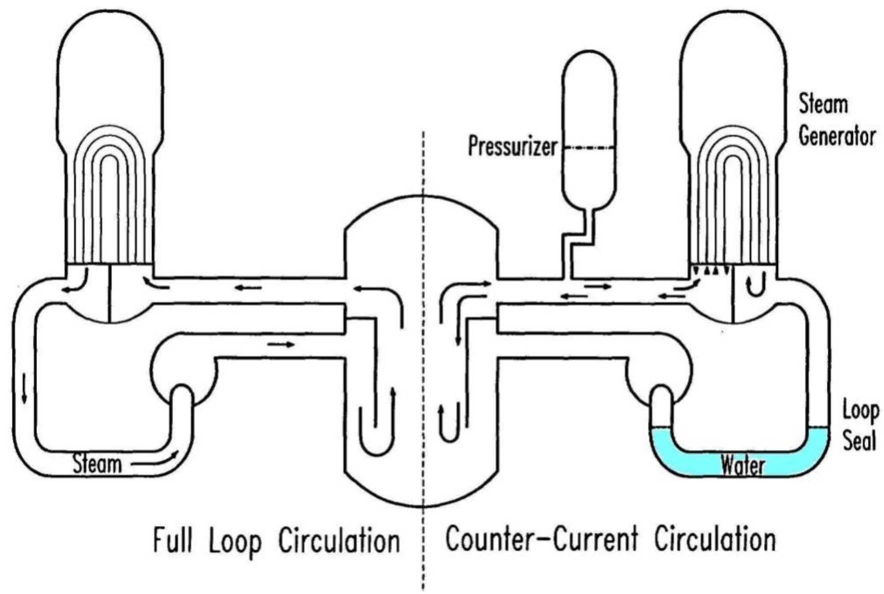
U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

# Thermal-Hydraulics

Michael Salay, RES/DSA

# Scenario

- "High-Dry-Low"
  - High primary pressure, Dry SG secondary, and Low SG secondary pressure
    - SBO with loss of TDAFW either immediately (Short Term SBO) or after battery depletion (Long Term LTSBO)
    - SG secondary Depressurization assumed to occur by some secondary leakage
  - High-Dry-High – high secondary pressure if no secondary leakage
- CE plants with replacement SGs susceptible to CSGTR – low amount of gas mixing in HL/SG plenum under Natural Circulation
  - Low hot leg (HL) length to diameter ratio
  - Shallow SG inlet plenum
  - Gas T into SG tubes nearly that of gas T into HL
- Releases depend on what components fail first – SG tubes or something else
- It has been expected that if unflawed tubes fail before hot legs that system depressurization will be sufficient to prevent HL failure
  - Multiple tube failure
  - Expectation of large releases
- Different behavior if loop seal cleared

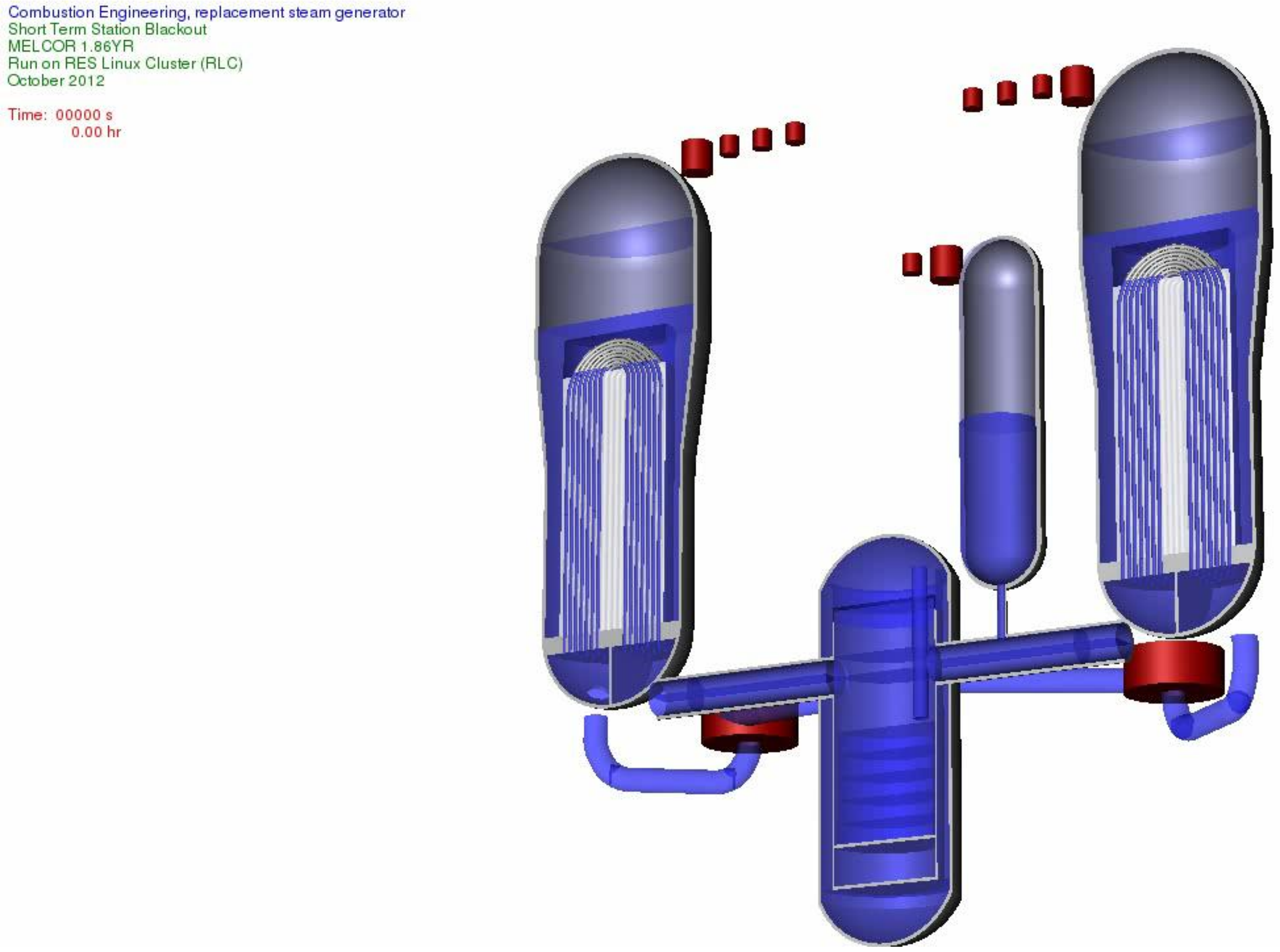



Severe Accident Natural Circulation Flows

# STSBO:SBO; SG leak;

**no tube flaws; nominal seal leak;**
**no MSSV stick open model;**
**fail TDAFW@t=0; fail battery at 4h**

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

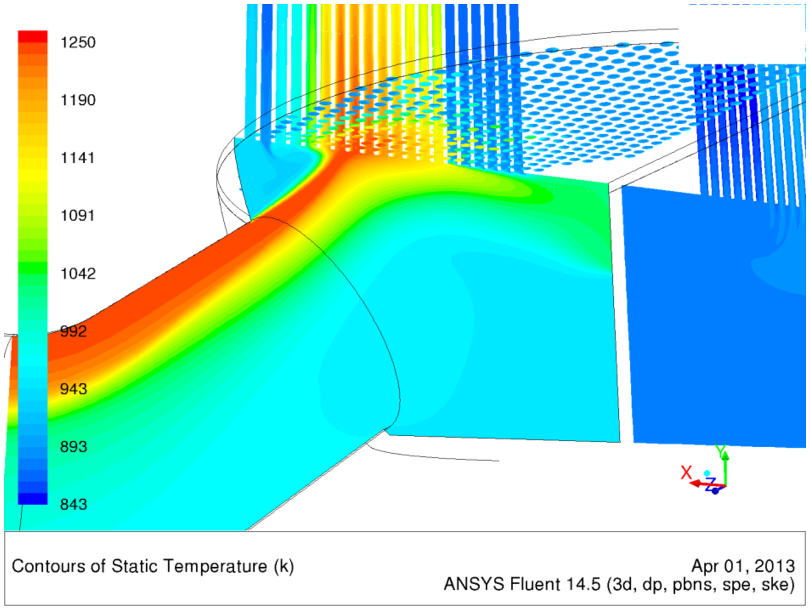


# Thermal-Hydraulics Work

- Tasks
  - Update CFD and MELCOR models for a representative CE plant
  - Evaluate T-H behavior for selected risk significant accidents
  - Perform technical assessment of incore instrument tube failures on natural circulation for CE and Westinghouse plants

# T-H Modeling Status

- Recently improved CFD predictions
- Comparison of MELCOR, CFD, and RELAP for CE
  - HL and SG flows
    - Impact on tube failure time and sequence of failures
  - Pressurizer drainage behavior
  - SG Secondary leakage
- MELCOR predictions updated
- Westinghouse T-H from previous work (NUREG/CR-6995)




**Why both FLUENT and MELCOR?**

- **MELCOR provides system behavior for full transient**
- **FLUENT provides detailed local 3D TH behavior**
- **Both tools needed to fully understand phenomena**

**→ CFD results used for MELCOR input and SG spatial temperature profile**

# **Major CE T-H Findings** (1/2)

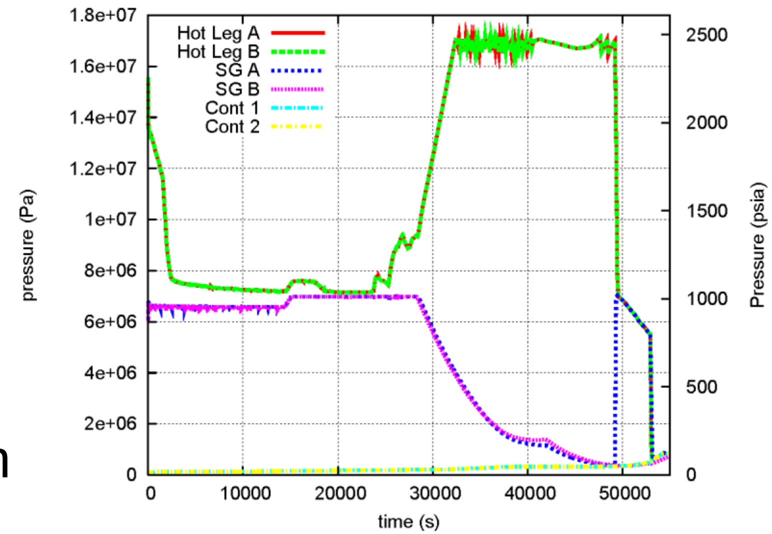
U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

- Relative failure timings still being evaluated*
  - Component failure timings are very close
- Previous analyses considered multiple tube failure prior to HL to result in containment bypass
  - This may not necessarily be the case based on MELCOR results
  - Impact of leaky (sufficient to depressurize) SGs different than previously expected
    - Depressurization advances SG tube rupture timing
    - Low secondary P resulted in primary/pressure equilibration at level near MSSV opening pressure – MSSVs barely open if at all
    - Equilibrium pressure high enough to rupture hot legs
    - Resulted in many more runs than planned to evaluate releases

*** Results are very sensitive to parameters currently under evaluation**

# Major CE T-H Findings (2/2)

- Current Results summary*
  - Unflawed tubes fail before HLs for high-dry-low
  - HL fails before unflawed tubes for high-dry-high (no SG leak scenario)
    - Determine failure time as t_failure(sm), sm(flaw size)
  - Even if multiple tubes fail first, unless an SGPORV fails early for leaky SGs secondary sides, volatile releases low
  - MELCOR does not predict loop seal clearing
    - Phenomena not fully explored




**LTSBO Pressures**

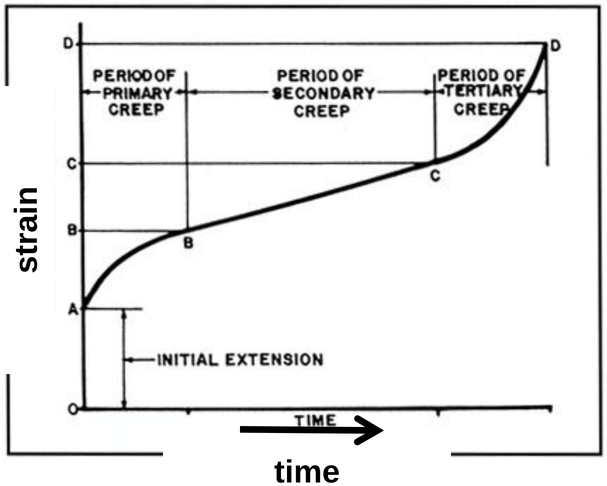
*** Results are very sensitive to parameters currently under evaluation**

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

# High Temperature Behavior of RCS and SGT Materials

Raj Iyengar, RES/DE/CIB

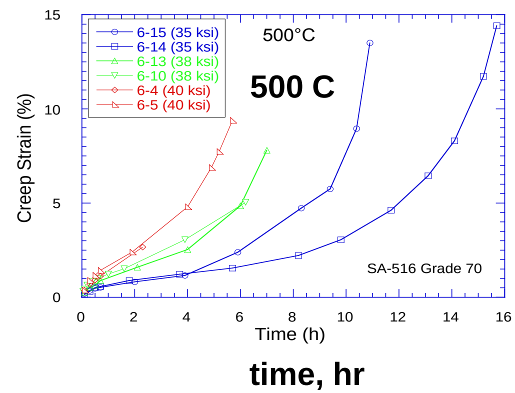
# High Temperature Deformation
## Metals

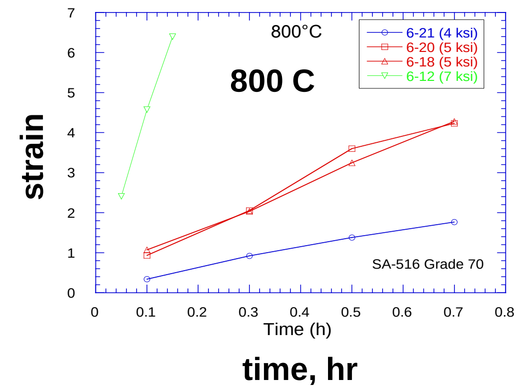



*http://www.nationalboard.org*

- 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

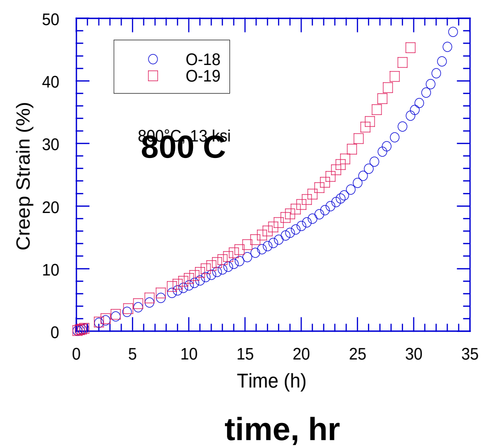
**Carbon Steel G70** **ANL Tests**

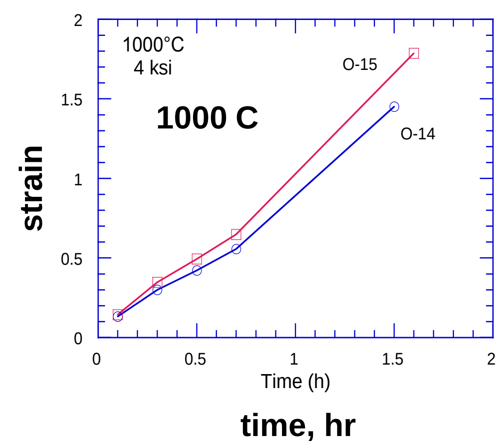



**Increase in stress or temperature accelerates creep deformation and damage**

**SS316**



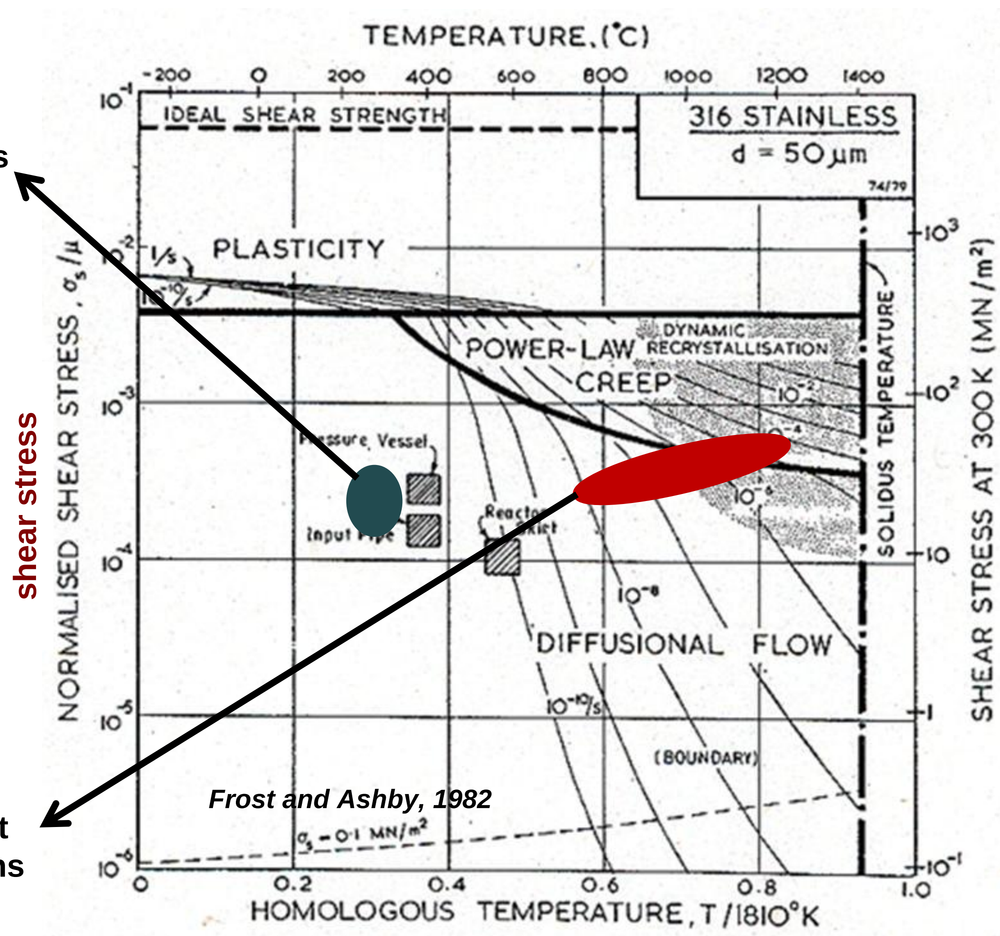



# Deformation Map
## 316 Stainless Steel

temperature

Normal Operating Conditions

shear stress

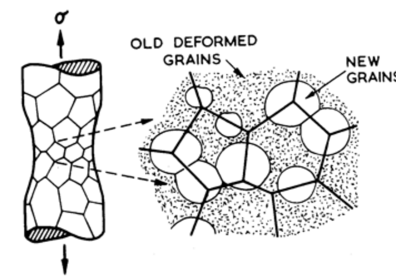



*Frost and Ashby, 1982*

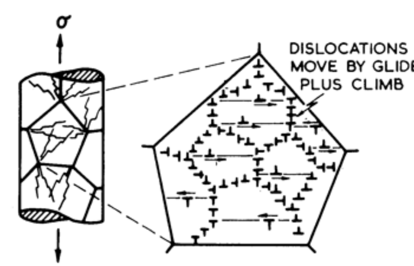
Severe Accident Conditions

- Transient effects influence extent and appearance of various regions
- Diffusion effects in stainless/alloy steels at high temperatures not well understood

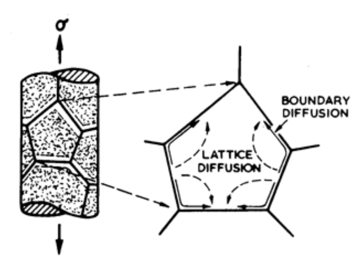
### Dynamic Recrystallization




### Deformation by Dislocation Glide and Climb




### Deformation by Diffusion (Lattice and Grain Boundary)



# High Temperature Failure

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

**Ductility-driven Damage**
**(dislocation motion)**

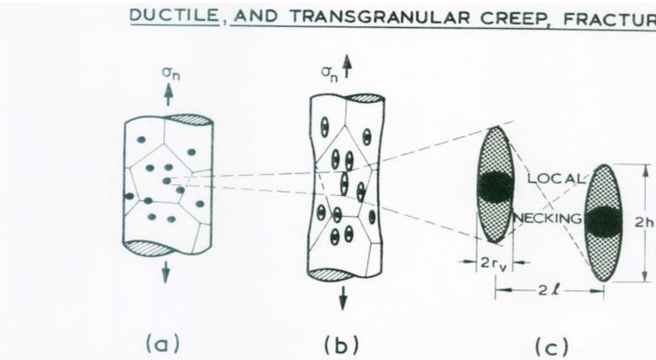
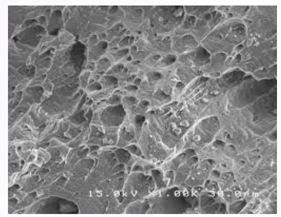



Fig. 3. (a) Ductile fracture and transgranular creep fracture requires either that holes pre-exist or that they nucleate at inclusions which concentrate stress. (b) The holes elongate as the specimen is extended. (c) They link, causing fracture, when their length, 2h, is about equal to their separation ($2l - 2r_v$).



**Diffusion-driven Damage**
**(grain boundary sliding)**

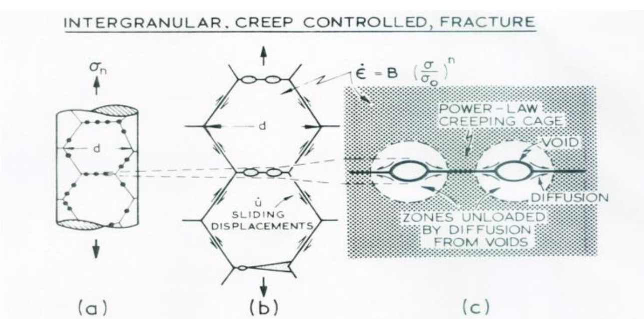
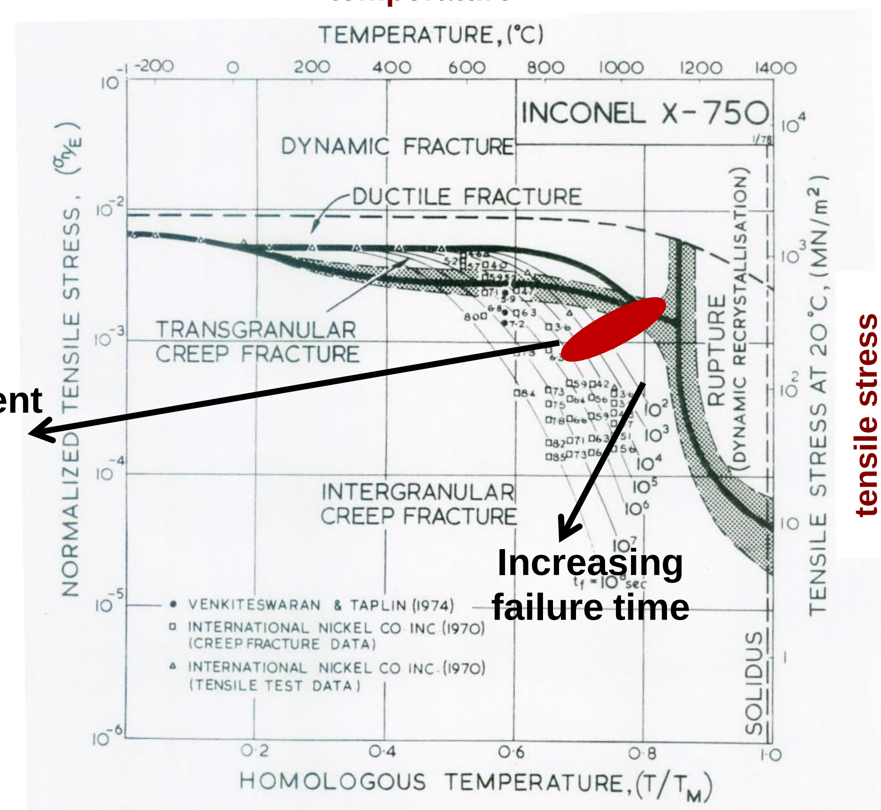



Fig. 4. (a) Grain boundary sliding stimulates the nucleation of grain and (b) boundary voids. (c) The voids grow by diffusion, but the diffusion fields of neighbouring voids do not, in general, overlap, so that each void is contained within a cage of power-law creeping material.

*Ashby, Gandhi, and Taplin, 1979*

**temperature**

**Severe accident conditions**




**tensile stress**

**Transient effects and multi-axial stress state influence extent and appearance of various regions**

# Flaw Distribution in SGs

Charles Harris, RES/DE/CMB

# Condition of SG Tubes

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

- Represent current fleet
  - Describe flaws in CE and W
    - Number, size
    - Type, location
    - Total leak area
  - New Materials
    - Alloy 600TT, alloy 690

# Condition of SG Tubes

- Update NUREG on flaw distributions
  - NUREG/CR-6521 (1998)
    - Original statistics still valid
    - 1998 - applied to Alloy 600MA

    - Adjust for new materials
    - Incorporate newer ISI data
      - number, size, type, location

# SG Tubes - Plant Data

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

| Combustion Engineering Plant Designs | | | |
|---|---|---|---|
| Plant | Current Model | Material | Replace Date |
| Calvert Cliffs 1 | BWC - 7811 | 690TT | Jun-02 |
| Millstone 2 | BWC | 690TT | Jan-93 |
| St. Lucie 1 | BWC | 690TT | Jan-98 |

| Westinghouse - Alloy 600TT SG Tubes | | | |
|---|---|---|---|
| Plant | Current Model | Material | Replace Date |
| Byron 2 | D5 | 600TT | Jun-05 |
| Seabrook 1 | F | 600TT | Jun-05 |
| Surry 2 | W/51 F | 600TT | Sep-80 |
| Vogtle 1 | F | 600TT | Jun-05 |

| Westinghouse - Alloy 690TT SG Tubes | | | |
|---|---|---|---|
| Plant | Current Model | Material | Replace Date |
| Donald C. Cook 2 | W/54F | 690TT | Mar-89 |
| McGuire 1 | BWC | 690TT | May-97 |
| Prairie Island 1 | Fr 56/19 | 690TT | Nov-04 |
| Sequoyah 1 | ABB/Doosan | 690TT | Jun-03 |

# Failure Behavior of RCS Components

Raj Iyengar, RES/DE/CIB

# Failure Behavior of RCS Components

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

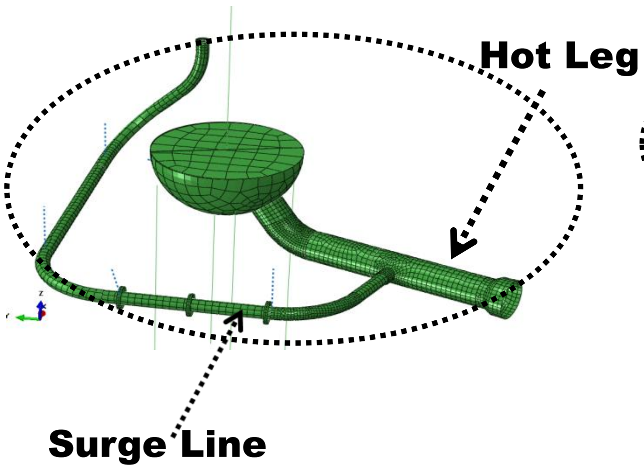
- Identify, characterize, and model relevant RCS nozzles to assess their potential for failure during a severe accident for both Westinghouse and CE plants
- Develop finite-element models, addressing variables such as nozzle geometries/configurations, boundary conditions, loading conditions, primary water stress corrosion cracking mitigations (overlays)
- Evaluate adequacy of simplified C-SGTR Calculator failure time estimates

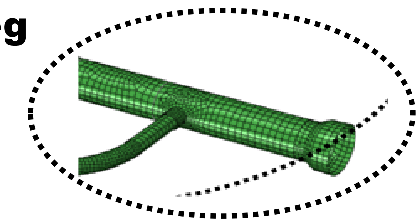
# Model Aspects

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

## 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

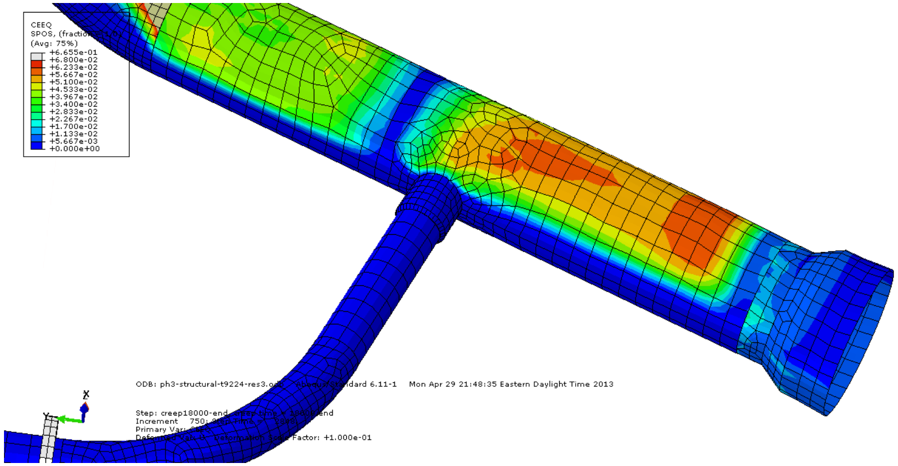
- Total strain = elastic + plastic + creep
- Creep Law – time and rate-dependent
- Plasticity Law – rate-independent
  - piecewise-linear stress-stain input from experimental data

# Analysis Procedure

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

- HL/SL structural temperatures for initial conditions (steady-state condition)
- Time-dependent gas temperatures from system code (RELAP) as a boundary condition
  - Use time-dependent heat transfer coefficient
  - Assume upper and lower temperature split
- Adjust the heat transfer coefficient spatially in the hot-leg region (based on the developing curve provided in NUREG-1922)
- Model heat loss to the ambience due to convection and radiation
- Run a thermal- mechanical simulation for short-term SBO

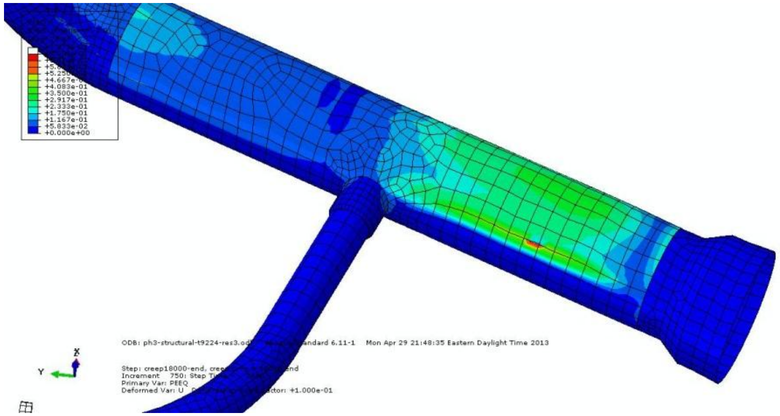
# Creep and Plastic Strains




**Accumulated Creep Strain**

**t = 12302 seconds**

**Accumulated Plastic Strain**



# Damage Prediction

**Damage at any material point determined using**

Larsen-Miller Parameter (LMP)

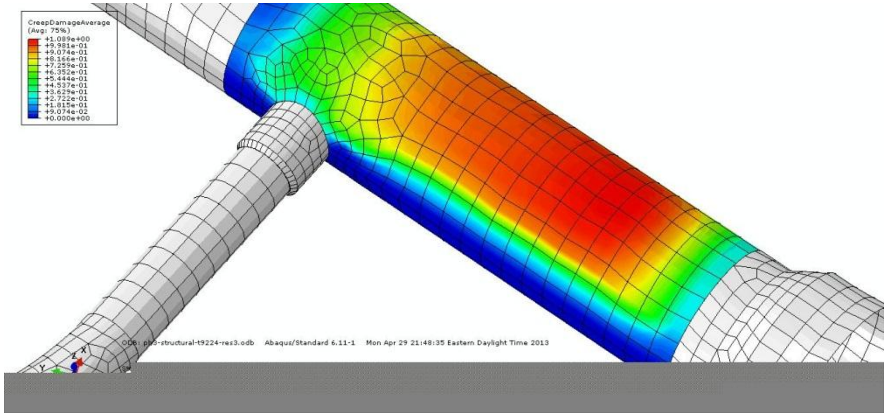
$$LMP = A*Log10(\sigma) + B$$

$\sigma$ - effective stress; T – temperature

Time to rupture

$$t_r = 10(PLM/T - C)$$

A, B, and C - constants




**Failure time - 12302 seconds**

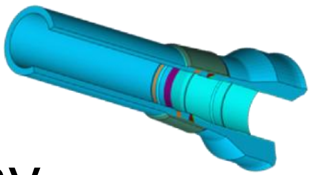
**Damage is averaged through thickness to determine failure time.**

# Failure Behavior of RCS Components

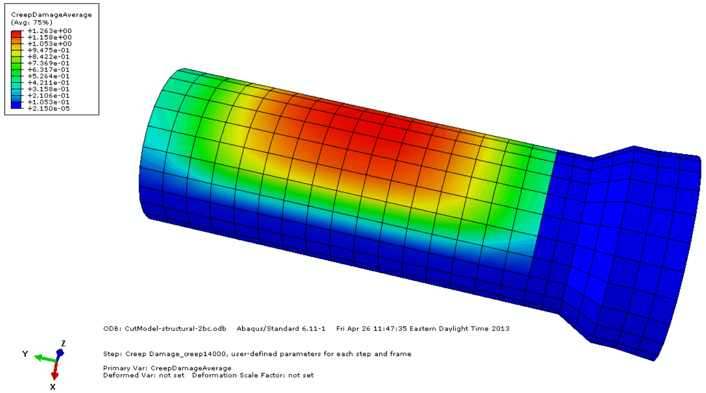
- System-level model simulations
  - computationally intensive
  - poses issues with convergence
  - Not well-suited for understanding sensitivities to input parameters
- Failure location in the hot-leg region predicted by the system model
- A sub-model of hot leg and reactor pressure vessel nozzle used 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

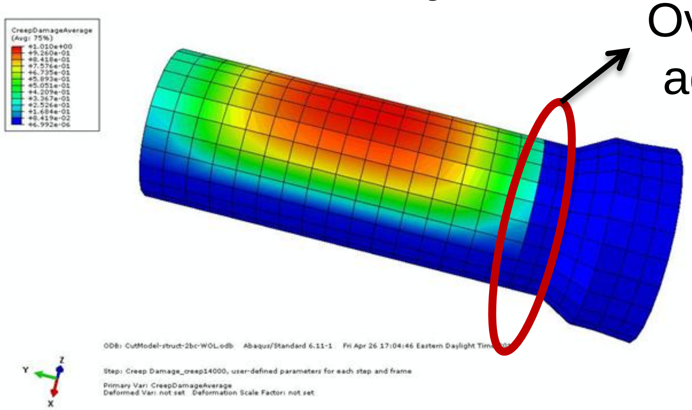


## No Weld Overlay



$t_r$ = 12428 secs

## Weld Overlay



Overlay added

$t_r$ = 12500 secs

**Failure time increases by 72 seconds with weld overlay**

**Failure location does not change**

# Failure Behavior of Hot Leg

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

## 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 |

**Hot leg failure time - 12600 seconds**
**( median failure time estimated by CSGTR Calculator)**

# Failure Behavior of RCS Components

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

## 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.

## Additional work

- Conduct analysis for CE pipe (T-H input from MELCOR)
- Draft report in progress

United States Nuclear Regulatory Commission
*Protecting People and the Environment*

# Probabilistic Risk Assessment Considerations

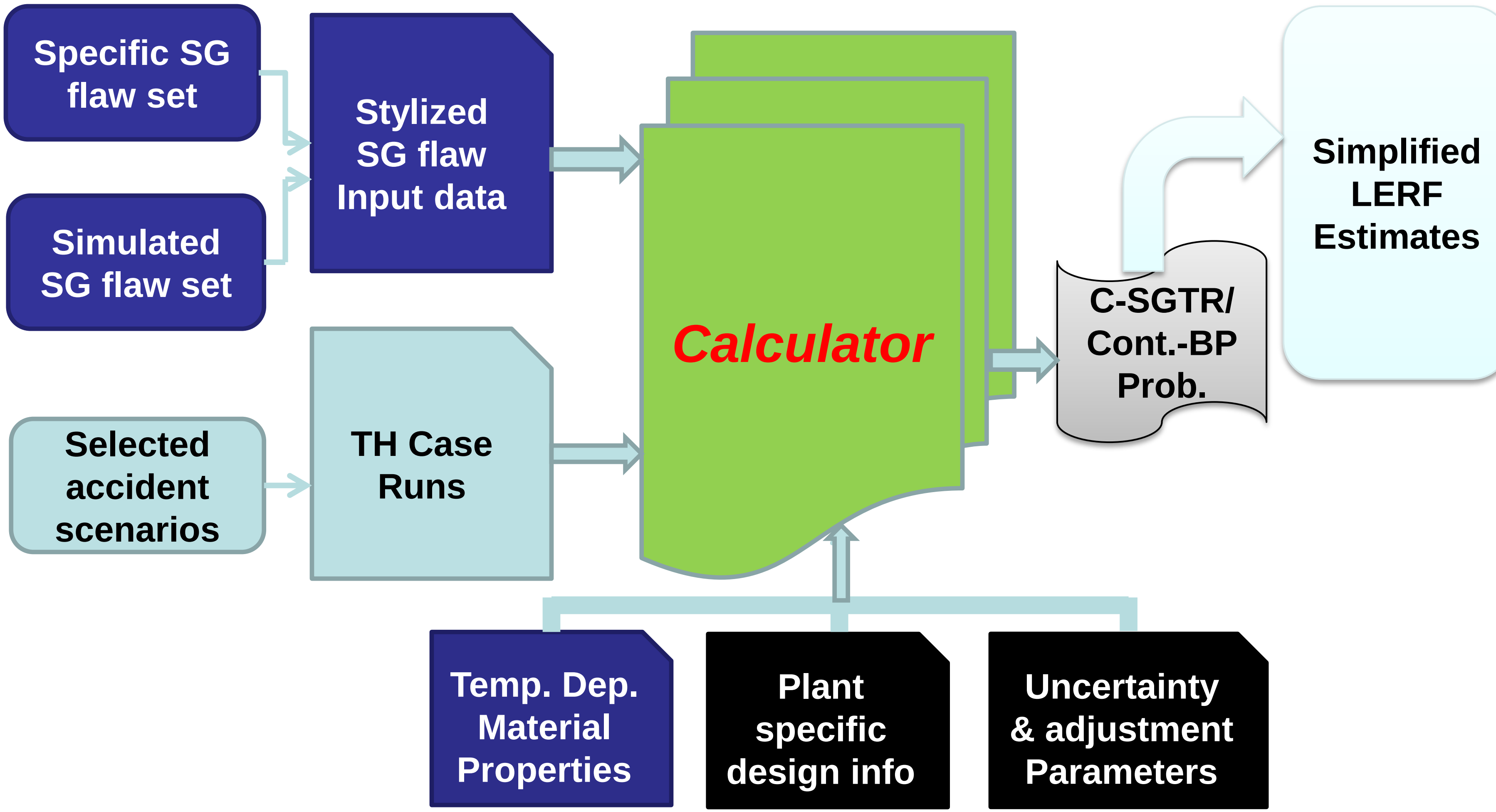

Selim Sancaktar - NRC/DRAB/PRAB

M.A. Azarm – IESS

# Objectives

- Develop a risk tool to
  - Estimate the probability of C-SGTR for
    - Design Basis
    - Severe accidents
  - Estimate Containment Bypass probability (Cont.-BP)
  - Estimate the fraction of Cont.-BP that is considered to be LERF
- Demonstrate the risk tool

# Overview of the Approach



# Main Activities

- Fracture Mechanics Models (FMM)
  - Available NRC's FMM for SG tubes
  - Documented FMM for hot leg and surge line
  - Compilation of material properties from multiple sources for TT-600 and 690
  - Both parameter and model uncertainties were treated
- SG Flaw Statistics
  - Limited SG flaw data was used
  - Flaw generation rate developed as a function of EFPY
  - Distributions for flaw length and depth
- PRA Considerations
  - Critical CSGTR leak areas important to PRA evaluation were determined
  - SAMG actions were identified
  - Triggering and declaration of General Emergency during accident were determined
  - Issues related to component performance post core damage were identified
- Performed two case studies

# Calculator: An Overview

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

- Flawed or pristine tubes
- Tubes with two different temperatures were modeled
- Library for material properties, default values and uncertainty parameters were prepared
- Technical Basis Document and a User Manual were peer reviewed by ANL and NRC staff
- FMM for Hot Leg and Surge line were added later using documented EPRI models

# Simplified Level 2 Model

- Five-Factor formula for LERF
  - Frequency of severe accident sequences with potential for C-SGTR ( $F_{AC}$)
  - C-SGTR probability ($P_{CSGTR}$)
  - Conditional Probability that the subsequent failure of RCS including the stuck open relive valves do not occur ($P_{NDEP}$)
    - Probability of Cont. BP is the product of

      $F_{AC} * P_{CSGTR} * P_{NDEP}$
  - Failure Probability of all SAMG actions ($P_{SAMG}$)
  - Probability that early effective evacuation is not successful ($P_{EVAC}$)

# Important Features Affecting C-SGTR Probability

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

1. Mixing in SG (deep or shallow SG inlet plenum)
2. Mixing in hot leg (physical characteristics such as length and diameter of hot leg )
3. Pressure drop in hot leg and SG tubes (i.e. an integral effect)
4. Heat transfer and heat losses from the hot leg walls (e.g. heat up and condition of the insulation on the hot leg)
5. Performance of primary and secondary relief valves post onset of core damage
6. Duration of DC availability including load shedding capabilities
7. Effectiveness and successful SAMG activities
8. Success of Flex and EDMG

# Case Studies

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

- Two case studies were performed for a Westinghouse (W) and a CE plant (CE) Plant
- Specific features appeared to be very important
  - The W plant and the CE plant are on the opposite side of spectrum for items 1 through 4 in the previous slide
  - The two plants were treated similarly for items 5 through 8, however the following differences were noted:
    - CE Plant has two TDAFW trains.
    - CE plant is susceptible to SBO affecting both units due to external events

# Preliminary Results
## from the case study

| CSGTR Probability as a Function of Critical Leak Area for the W Plant | | |
|---|---|---|
| **Critical Leak Area** | Probability that CSGTR Exceeds the Critical Leak Area | |
| | SBO with Early Failure of TDAFWs (Inconel 600) | SBO with Early Failure of TDAFWs (Inconel 690) |
| **Equivalent to one or more tubes** | 1.3E-02 | 8.9E-03 |
| **Equivalent to two or more tubes** | 8.2E-5 | 3.9E-5 |

| CSGTR Probability as a Function of Critical Leak Area for CE Plant | | |
|---|---|---|
| **Critical Leak Area** | Probability that CSGTR Exceeds the Critical Leak Area | |
| | Extended SBO with Early Failure of TDAFWs | Extended SBO with Failures of TDAFWs after Battery Depletion |
| **Equivalent to one or more tubes** | 0.95 | 0.98 |
| **Equivalent to two or more tubes** | 0.87 | 0.84 |
| **Equivalent to more than three tubes** | 0.75 | 0.64 |

# Concluding Remarks

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

- The contribution of C-SGTR to LERF can vary significantly based on plant design features.
- Research is now at a stage to better understand what are the most influencing parameters for C-SGTR LERF and how to control them.
- The contribution is expected to be reduced as a result of additional actions that are underway for EDMG and Flex programs.