Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards
Materials, Metallurgy and Reactor Fuels
Steam Generator Action Plan

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Thursday, September 24, 2009

Work Order No.: NRC-3101 Pages 1-306

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

<u>DISCLAIMER</u>

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

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

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

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

+ + + + +

ADVISORY COMMITTEE ON REACTOR SAFEGUARD

(ACRS)

SUBCOMMITTEE ON MATERIALS, METALLURGY AND

REACTOR FUELS

+ + + + +

THURSDAY, SEPTEMBER 24, 2009

+ + + + +

ROCKVILLE, MARYLAND

The Subcommittee convened in the Commissioners' Hearing Room at the Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, at 8:30 a.m., Dr. Dana A. Powers, Chairman, presiding.

SUBCOMMITTEE MEMBERS PRESENT:

DANA A. POWERS, Chair

J. SAM ARMIJO

SANJOY BANERJEE

DENNIS C. BLEY

OTTO L. MAYNARD

WILLIAM J. SHACK

JOHN D. SIEBER

JOHN W. STETKAR

NRC STAFF PRESENT:

CHRISTOPHER BROWN, Designated Federal Official

TIM McGINTY

DAVID BEAULIEU

CHRISTOPHER BOYD

JEFF HIXON

TIM LUPOLD

KEN KARWOSKI

ROBERT PALLA

ED FULLER

ALSO PRESENT:

DON FLETCHER

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

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

(8:31 a.m.)

OPENING REMARKS AND OBJECTIVES

CHAIR POWERS: Again, the meeting will come to order.

This a meeting of the Subcommittee on Materials, Metallurgy and Reactor Fuels Subcommittee.

I am Dana Powers, chairman of the Subcommittee for the Steam Generator Action Plan.

ACRS members in attendance include William Shack, Sanjoy Banerjee, John Stetkar, Dennis Bley, Otto Maynard, Sam Armijo and Jack Sieber. Christopher Brown of the ACRS staff is the designated federal official for this meeting.

I wanted to begin with a little bit of an introduction on this subject. We are here to talk about the steam generator action plan. The steam generator is of course a part of the primary pressure boundary. There is a rumor that there are designs of reactors out there that don't actually have steam generators, but we will not concern ourselves with them today.

The ACRS itself has had a long involvement with this action plan. It probably culminated in the publication, what, eight years ago, ten years ago, in a report on the voltage-based repair criteria. And in that report we made a variety of recommendations to the staff on what action they should take as part of their action plan on the steam generator.

I'd remind people of course that ruptures to the steam generator are indeed design basis accidents. We worry about them because it is possible for them to progress to become bypass accidents that result in severe core damage if they are not arrested. But they are in fact a design basis accident, and they do occur.

At the time we made our recommendations to the staff, in addition to their action plan, the Commission made an explicit request that we keep them informed on progress made in the action plan with respect to the steam generators. While substantial amount of water has passed through the turbines, over the years, and the ACRS membership has evolved. And we've not been coming back as regularly perhaps as we should to the steam generator action plan, and progress has been made by the staff.

And they are here to review that progress and to make recommendations on what they would like to do with respect to close out of some action plan items, and to transfer other items into ordinary research programs.

I have asked that they spend some time in their presentations provided background, since there are some members that may not be thoroughly familiar with everything that has gone before in connection with the action plans.

So the meeting today is to review the staff's activities and technical basis, related to closure of remaining items in the steam generator action plan. We will hear presentations from representatives of NRR and RES. The subcommittee of course will gather information, analyze relevant information and facts, and formulate proposed positions and actions as appropriate for the deliberation of the full committee.

Essentially what we want to do is to formulate a fairly succinct status report for the Commission itself. And I will comment that I believe there is nobody on the current commission that was present when we wrote our report on the voltage-based repair criteria, which in a sense summarized many of the technical issues that have arisen concerning steam generators.

When we formulated our report we are going to have to devote some portion of the task to background for the Commission themselves. So to the extent the staff can help us formulate some of those words, we will appreciate it greatly.

A transcript of the meeting is being kept, and will be made available as stated in the Federal Register Notice. Rules for participation in today's meeting were announced as part of this notice of this meeting that was previously published in the Federal Register on August 20th, 2009.

We have not received any requests from members of the public wishing to make oral statements.

We request participants in this meeting to use microphones located throughout the meeting when addressing the subcommittee. Participants should first identify themselves, and then speak with sufficient clarity and volume so they can be readily heard.

I will note that Dr. Shack has an organizational conflict of interest, since he was directly involved in some activities in the steam generating action plan. He is however at liberty to provide clarification and technical data as he sees fit. So we will call upon you to provide clarification and technical facts as we see fit. It will of course be unusual to hear facts coming from Dr. Shack, but -

First I'll ask, are there any members who would like to make opening statements? I will comment that a lot of the activities that the staff is undertaking have to do with the interface between what is ordinarily a metallurgical issue of steam generator corrosion and risk assessment. And I am particularly anxious to get insights on this from Mr. Stetkar and Dr. Bley. And I will comment that this is an aspect of the committee that has not been investigated very thoroughly in the past, relatively new to us - new to me, anyway.

If there are no other opening statements, we will now proceed with the meeting. And I call on Tim McGinty for the introduction.

And Tim again I will comment that background and helping us formulate positions that we can subsequently communicate to the Commission will be very helpful.

OPENING REMARKS

MR. McGINTY: That is clearly in our common interest.

Good morning, Mr. Chairman, and members of the subcommittee. My name is Timothy McGinty.

I'm the director of the division of policy and rulemaking in the Office of NRR. My division is responsible for the project management of the steam generator action plan that is before you today.

I'd like to thank the subcommittee for taking the time to review the staff's work, particularly in light of the large amount of supporting documentation that has been provided to you.

To my right is David Beaulieu. He is the NRC project manager for the steam generator action plan who will be providing the opening staff presentation. In the audience we have various NRC and contractor staff who contributed to the action plan closeout effort, including the scheduled presenters, Bob Palla, Chris Boyd, Gene Carpenter, Jeff Hixon and Selim Sancaktar.

And other knowledgeable staff members such as Ken Karwsoki and Emmett Murphy, some of whom have been involved with the steam generator issues sine the early 1990s.

Today we will begin with a staff presentation that provides a background and overview of the steam generator action plan, and the desired outcome of this meeting and future plans regarding steam generator research activities. Staff has completed its work on steam generator action items, and has provided the closeout documentation to the ACRS.

As Mr. Beaulieu will explain further, the desired outcome of the ACRS review is a letter from ACRS that finds acceptable the staff's closeout of each steam generator action plan item, that ACRS has not previously reviewed and closed. Staff presentations over the next two days cover each of these items. Following the ACRS review the staff would like to be able to close the steam generator action plan.

Future work activities associated with this topic will be coordinated using other agency tools, such as the user need process and planning, budgeting, performance management process.

With that, I turn the presentation over to the project manager.

CHAIR POWERS: Let me ask you one question. You have a lot of documentation on this project, but it's individual piece meal as far as I can tell. Are there any plans to write what I would call a comprehensive status report on all the work collected together in something that I would say is suitable for archival publication?

MR. McGINTY: At this point in time we hadn't anticipated that or envisioned that.

CHAIR POWERS: I wonder why not?

MR. McGINTY: Why not?

CHAIR POWERS: I presume there is a fair amount of public interest in this. We have on the books NREG-1740 which in a sense is a kind of status report in its time. It's probably not titled adequately for someone to refer to as a status report, but it certainly lists the issues. And it seems to me you have done so much in so many diverse areas, some of it fairly arcane admittedly, that a document - of a summary nature - that says okay here is what the status is on this. And I would see it more as knowledge preservation --

MR. McGINTY: Right.

CHAIR POWERS: -- and a guide to this rather dense forest of topic-specific documentation that you have available that would be particularly useful as you bring new people into the agency in a variety of roles who have to confront this action plan. And I'm particularly interested in this because I see the potential for a great deal of apathy to develop in the area of steam generator tubes. Because they quite frankly are performing very well right now. I believe it's something like nine years since we had the last rupture, and we have material that in fact is better but still not immune to the cracking and rupture problem.

And so I can foresee and time when people will become unfamiliar with the kinds of challenges that can arise with steam generators, and having a resource guide that says, here is what we know about these machines, and how we got that information, strikes me as something that would be very useful to the agency. It might even be useful to the staff simply to provide a summary.

MR. BEAULIEU: Research has created a short summary document. We will make sure that if we haven't given you a copy, we'll make sure that you received it. But it does integrate all the research activities that were done to some extent and references other documents. In addition the user need created by NRR provides a summary of what was done and where we'd like to go further on that. And that will be covered too.

CHAIR POWERS: I think that is a useful start. I think you need to think seriously about something that you can publish in the archival literature.

MR. McGINTY: Understood.

CHAIR POWERS: That would be readily available to the public, readily available to the metallurgical community. I hesitate to pick the forum, but I would say something like progress in nuclear energy or something like that, something fairly visible, that would - I'm not supposed to prejudge these things. But I think you've done a good job.

MR. McGINTY: There is clear merit to that, and so I'd like to take that under consideration.

CHAIR POWERS: I would propose that members give this some thought, and that maybe it's something that should figure in our draft position that we develop for the committee's deliberation. They do this, because I mean it's just too - there is a forest of stuff. And some of it is dotting i's and cross t's nature, some of it is very significant, some of it is innovative. Some of it is actually profound. I think of particularly the stuff that you have done on probability of detection and things of that nature.

And I think it's too easy in the present business for the agency not to brag on them itself, once in awhile in the literature, and say we really do significant things here. And too easy to forget that there is a community out there that has an interest in community power and would like to know what the status is.

Let's go, please continue.

MR. McGINTY: Fair enough.

SGAP BACKGROUND AND OVERVIEW

MR. BEAULIEU: Just to start out, I want to make a reminder that the copies of the slides that you all have, the color as well as the black and white slides, cover all of today's presentations as well as tomorrow's presentations. So you won't get a new package for tomorrow, so you want to bring the packages tomorrow with you if you plan to attend.

CHAIR POWERS: You ask way too much of us.

MR. BEAULIEU: If you'd like me to bring another copy, I'll be happy to.

But let's see, thank you, Mr. McGinty, for your introduction. Good morning, members. I am Dave Beaulieu, project manager for the steam generator action plan which is the subject of this meeting today and tomorrow.

I'd like to thank the subcommittee for taking the time to review the staff's work.

I will begin by providing a background and overview of the steam generator action plan, the staff's completion of work on all items, desired outcome of the ACRS review, and future steam generator research activities.

In terms of the steam generator action plan history. During the NUREG 1150 studies from 1985 to 1990, the issue of consequential steam generator tube rupture was first identified.

CHAIR POWERS: I have to generate here. Bypass accidents and the risks associated with them was first identified in WASH-1400. That was 10 years before the NUREG-1150.

MR. BEAULIEU: I stand corrected. By consequential steam generated tube ruptures, we mean that the steam generated tube rupture itself is not the initiating event. Consequential tube ruptures refer to those steam-generated tube ruptures that may be caused as a result of another initiating event which could be such as a very large steam main break that leads to high differential pressure across the steam generator tubes, or severe accident induced consequential steam generated tube ruptures.

The current concern was that the high temperature gases created during core damage sequences could cause steam generated tubes to be the first component in the reactor coolant boundary to fail, resulting in potential containment bypass and the release of large amounts of radioactive material outside of containment.

NUREG-1150 quantified the frequency of this occurrence in the low 10^-6 for reactor year range on the basis of expert elicitation.

In the early `90s --

CHAIR POWERS: Well, that frequency of occurrence somewhat begs the point made by 1150, which was, though the frequency is 10^-6, it is the risk-dominant accident. Because of the high consequence.

MR. BEAULIEU: Right, and it also was really based on expert elicitation, their best judgment is a way of saying it. We've progressed a lot further since then.

CHAIR POWERS: It's still based on our best judgment.

MR. BEAULIEU: Yes, that's true. That's true.

In the early 1990s the industry made

several requests for the relaxation of requirements with respect to steam generator tube integrity. And this involved the voltage based repair criteria.

A differing professional opinion was filed in the early `90s, it began with the concerns with these relaxation requests. At the time the staff reviewed those relaxation requests and identified that granting them might substantially increase the conditional probability of containment bypass during core damage accidents.

CHAIR POWERS: I think I would characterize the situation a little bit differently. Or maybe I'd augment the characterization a little bit. Are you saying that we'd have an evolution from a set of regulations written primarily for a wastage mechanism to one where cracking, stress corrosion cracking, was dominant, and that the regulations were ill suited for addressing that issue. The regulatory requirements spoke of 40 percent for wall thinning and things of that nature. So how do you translate that into when the mode is cracking rather than wastage.

And so we had an evolution in mechanism. My colleague, Dr. Shack, tells me - I think he knows on this subject very well - that when an alloy is not susceptible to wastage corrosion it probably is susceptible to cracking erosion. There seems to be no escape, these metals.

MEMBER SHACK: No impermium.

CHAIR POWERS: Silicon carbide. And that the translation of those regulatory requirements to this other mechanism posed a challenge, both for the industry and for the staff. So then everything else you say is absolutely true.

MR. BEAULIEU: Thank you.

Approximately the same time in the early 1990s NRR with the assistance of research began a study of the effects of severe accident conditions on steam generator tube integrity as background information for a proposed new rulemaking on steam generator tube integrity. Results from this study, published as NUREG-1570, indicated that the risk is controlled by the current tube integrity requirements to a value that is low enough that no new rulemaking was needed. The NUREG was never intended to specifically address the DPO. It covered some issues, but the DPO remained open following the publishing of that NUREG.

In 2000, following the rupture of the steam generator tube at Indian Point Unit 2, additional focus on the resolution of several long standing issues, which includes the DPO. The executive director of operation referred the DPO to ACRS for resolution. After extensive public meetings and review of the issues raised in the DPO, the ACRS published NUREG-1740 to present its conclusions to present is conclusions and recommendations.

In particular ACRS concluded that the methodology being used to quantify the risk of containment bypass, due to high temperature challenges to steam generator tubes, was not technically defensible.

Technical staff in NRR and research jointly reviewed the full text of NUREG-1740 to extract the list of issues that required additional work. Those tasks were incorporated into a new section, which is Section 3 of the steam generator action plan, is the last remaining section open, which identified individual staff members with lead and support responsibilities for each task and schedules for completing each task.

Most but not all of those steam generator action plan tasks are directly related to the work to define the risk associated with severe accident induced tube ruptures, leading to containment bypass.

Section 3 also includes work such as the work performed under items 3.1 and 3.11 that involved design basis events, which address the potential for damage progression of multiple steam generator tubes due to steam generator depressurization, such as what would occur during a steam line break or other type of secondary side design basis accident.

The staff's work to address a steam generator action plan items involving design basis events is complete, and the ACRS has reviewed and endorsed the closure of these items. The technical basis behind this was primarily that it was based on the overall conclusion of this work which is that the dynamic loads from such design basis events are low, and do not affect the structural integrity of the tubes or lead to additional leakage of ruptures beyond what would be determined using differential pressure loads alone.

As of today, where do we stand today is that the staff has completed its work to close all steam generator action plan items. The closeout documentation has been provided to ACRS. The purpose of this two-day ACRS subcommittee meeting is for ACRS review of all steam generator action plan items that the ACRS has not previously reviewed and closed.

The desired outcome of this review of this review is that following this ACRS review which includes this two-day subcommittee meeting as well as an ACRS full committee meeting scheduled for October 8th, the desired outcome is that ACRS will issue a letter which finds acceptable the staff's closeout of each steam generator action plan item that ACRS has not previously reviewed and closed, These are items 3.1.k, 3.4, 3.10, 3.11 and 3.12 that are the subject of our presentations for the next two days.

What can you expect to hear from us during the next two days? Essentially all of the items are directly related to work to define the risk associated with severe accident-induced steam generator tube ruptures leading to containment bypass. The work involved the following technical areas of research: thermal-hydraulics; steam generator tube material failures; reactor coolant system material failures; component behavior studies; and probabilistic risk assessment.

How these integrate together, that is an iterative process. The thermal hydraulic work takes the PRA sequence being evaluated and determines the fluid temperatures and pressures as a function of time. Conditions are then used as inputs into the reactor coolant system, material failure and component behavior models.

Finally all of the thermal hydraulic information and material failure information are logically combined into a PRA model to determine the risk associated with a consequential steam generator tube rupture issue.

Future activities. Does this mean: that all steam generator questions have been answered? I don't think we'll ever reach that point. While the closeout documents for each steam generator action plan items provide a solid basis for closing the items, an NRR need to research is in concurrence that requests specific research products to facilitate the development and review of future risk assessments involving consequential steam generator tube rupture events. Products will build upon analyses, methods, tools, and expertise developed as part of the steam generator action plan. However, this research work that is needed to address the NRC user need no longer requires the level of coordination and agency focus that is required to implement the action plan process.

Consequently, staff would like to close the steam generator action plan, and like I said this does not preclude future consequential steam generator tube rupture research activities. In fact we would simply like to have these future work activities associated with this topic be coordinated with other agency tools: user need; planning and budgeting; and address those as a whole of NRR work based on risk significance and prioritized accordingly with our other processes.

CHAIR POWERS: I think that is a singularly important point that you make there. The action plan is one vehicle that the agency has for addressing the specific issues that arise, and it has its place. And those action plans by design are of finite duration. The technical issues may well continue, and the research program provides a venue for addressing those technical issues, and they should be used appropriately.

And there is nothing being proposed here that precludes further investigation of any one of the disciplines that you have listed down here. But on a need basis, and in competition with all the other demands on finite resources. I think that is just a very singularly important thing. We are not saying, okay, no one ever has to look at these issues ever again, because I guarantee you that there will be lots of details that come up some of which may be quite important. It's just that the action plan may not be the appropriate vehicle for doing that.

Then I will augment my opening comments to say, Professor Banerjee, the thermal hydraulic aspects that we are going to hear about next, I think, are very important to us here and I think there has been, to my mind, some innovative work, and I'll be very interested in your tutored perspective in this regard.

This introduction I think is fairly important. Do members have any questions that they want to pose on the strategy? I personally have no troubles with the strategy, I would propose that that also figure as one of the points we make in our report to the full committee.

MR. McGINTY: Once again, thank you.

CHAIR POWERS: Yes, that was a very useful introduction. I would however like to get from you a succinct list of the titles for all the sections of the action plan - one, two, four - well, it's three. And just help me with the letter.

MEMBER BANERJEE: Something which would help me is were there industry actions taken in response to the program, and are they documented somewhere, summarized?

MR. BEAULIEU: There will be a slide presentation on this, PRA standard has been revised to include - to require licensees to consider consequential tube ruptures in their PRA analysis.

MEMBER BANERJEE: This was the PRA analysis, but with regard to inspection, there are quite a number of actions, I take it, right?

MR. BEAULIEU: Yes, in terms of inspections and repair, that was primarily addressed through tech spec changes that had been previously discussed here. And that particular aspect of it has been reviewed and closed.

MEMBER BANERJEE: It doesn't have to be addressed now, but if there is some summary document of the way we can take a look at what has really happened in response to this action plan, not just on paper, but in actual actions, that would be useful.

MR. BEAULIEU: I will get that for you.

MEMBER BANERJEE: Any time. Before the full committee.

MR. BEAULIEU: The tech specs was the big piece, and the requirement to address consequential tube ruptures in the PRA in place now. And I will provide you further information.

That said, Mr. Boyd is up next.

SGAP ITEMS 3.4.a-d

MR. BOYD: Hello, my name is Christopher Boyd. I'm here together with Don Fletcher from Information Systems Laboratories. And we are going to present the thermal hydraulic work that has been done in support of the steam generation action plan. These are items 3.4.a through g.

Don has been involved with the system code modeling with SCDAP/RELAP, and he is from the Office of Research, and we have done the computation fluid dynamics that supports and extends the experimental database to support the system codes.

I'm going to give a quick introduction, and give some background in the thermal hydraulics, then Don is going to talk about some highlights from the SCDAP/RELAP work, and then I'm going to come back and talk about some highlights from the computational fluid dynamics that was done.

So we are going to talk about starting out an action plan, Section 3.4. Section 3.4's general mission is to develop a better understanding of the reactor coolant system conditions, and the component behavior. And then we are going to focus now on 3.4.a-g, which just focuses on the thermal hydraulic behavior.

So our goal in general, and we'll look at the specific tasks, but in simple terms our goal is to, one, predict the overall system behavior during these severe accidents, and we need to get the plant behavior properly modeled.

And then we are going to go in and look at specific locations in the reactor coolant system that are potential failure locations, and look in detail at the thermal and mechanical loads. And we do some screening calculations for failure, but we mainly are then going to pass our boundary conditions off for materials, a further detailed material study of those components.

We need to understand that when we look at the thermal hydraulics, we are focused on the worst conditions. We don't want to run conditions where nothing fails. So we're in a situation where hot leg failure is a wonderful thing. I mean it's kind of crazy. So I'm trying to orient. These are bad scenarios, and we stay in that bad area. So I don't want to give the wrong impression. These are not the normal operations, of course.

What we need for a tube to fail we call high dry low, and I want you to understand that so we know where we are looking. We need to have high primary side pressure, so if there is a leak in the reactor coolant system and it depressurizes, we do not predict tube failure. So we are going to look of course in the high pressure scenarios, and then we are going to look at a various spectrum of leaks from different things and see if it depressurizes enough. And our goal would be to create boundaries. If you have a leak of a certain size, you won't maintain the high pressure.

We also need a dry steam generator. If the tubes have water on the secondary side we are not going to have a severe accident induced tube failure. So we are going to look of course - our first runs are going to be with no auxiliary feeds. Everything fails right away; no backup systems work. They are dry. But we will also look at various combinations of four hours of aux feed, eight hours of aux feed, and kind of look at the boundaries of where things will fail.

In addition to being high and dry, we also need to depressurize the secondary side of the steam generator. If we maintain the pressure boundary on the secondary side, that essentially cuts the stress in half on the tube, and we do not predict these induced tube failures.

So we look at a stuck-open valve on the secondary side, or possibly leaks of various sizes on the secondary side, but that will depressurize, and get us into this high dry low.

So we start out dead center in this high-dry-low, and we assume a bunch of things to make us get into the high-dry-low condition. And then we look at ways out of it, and we try and create maps where we are in that area, and where we are not.

Let's look at a typical scenario, so we have an idea what goes on. So this is our fast scenario, our scenario where we assume everything fails. At time zero we have a loss of offsite power. The diesel generators fail to start. At some point the auxiliary feedwater system is going to kick on. We assume that fails to start.

Now the primary inventory will have a small LOCA through the reactor coolant pump seals. The typical assumption is 21 gmp. as a minimum. And will also at this point going to be transferring heat to the steam generator and it is going to boil off. So you will see the secondary system boiling off. At about 100 minutes we will dry off the steam generator. At this point the primary system will expand a little bit. We will get some relief valve cycling, and the reactor coolant pump seal LOCA will continue, and we'll be basically losing inventory now on the primary side.

At some point that inventory will drop to the point where we will lose full loop circulation. We will continue with a slow heat up of the primary system. We will be sitting at the valve set points and releasing out through the pressurizer relief valves.

The inventory continues to drop. At some point the inventory falls below the hot leg. And at this point we set up a three-dimensional natural circulation flow pattern that carries superheated steam from the core out into the loops. And basically now our energy sink now is the metal mass out in the loops. This flow pattern has been experimentally established. There was a set of one-seventh scale tests that were done. So at about 150 minutes in this fast transient is when we drop below the hot leg, we get superheated steam from the core, and we get this three-D natural circulation flow pattern.

Again we are still losing inventory from the primary side.

MEMBER BANERJEE: Chris, let me ask you a question here. If you depressurize the secondary side rapidly, and it becomes a heat sink. And you still have some liquid that is formed by condensation in the steam generator, you can on the riser side of the steam generator pull up a lot of liquid if you have flooding conditions at the elbow. That would depress core levels much earlier. Because now if the loop seal hasn't cleared, with the delta h, which is the standard reflux condensation scenario. So you get core uncovery quite early, not as late as you are showing here. What happens in that case?

MR. BOYD: I'm going to ask Don if we have run that. We have run maybe 100 or 200 different scenarios. Did we --

MR. FLETCHER: Your concern again, I guess I'm not quite sure what you are saying?

MEMBER BANERJEE: What I'm saying, while you still have water, and you haven't cleared the loop seal, and you have depressurized the secondary site very rapidly.

MR. FLETCHER: On purpose?

MEMBER BANERJEE: No, it's happened due to whatever events you've got. So you still have water you are boiling off the secondary site. So it's becoming a big heat sink early in the transient.

MR. FLETCHER: Yes.

MEMBER BANERJEE: Early in the transient. Now you are starting to have all that water in the steam generator which has no way to go on the riser side. This is a very common scenario. You see it in semi-scale.

MR. FLETCHER: But if you have depressurized the steam generator, and you are removing heat to the steam generator, the steam will escape in the secondary --

MEMBER BANERJEE: No, I'm saying on the primary side.

MR. FLETCHER: You are talking about condensing inside the tubes?

MEMBER BANERJEE: It's usually the refluxing mode, right?

MR. FLETCHER: Well, inside the primary system the tubes are still filled with water.

MEMBER BANERJEE: Yes, so now that head, I can show you in a diagram, that head balances the head and uncovers the core. And the reason this happens is that you get flooding at the elbow that leads to the steam generator. This is a well known phenomenon.

MR. FLETCHER: You are talking about a small break LOCA accident --

MEMBER BANERJEE: Like a small break LOCA.

MR. FLETCHER: -- where you have hold up of the liquid on the upside of the tubes.

MEMBER BANERJEE: Your steam generator is being rapidly depressurized, and dropping in temperature makes this much worse. And a small break LOCA doesn't.

MR. FLETCHER: You tend not to get into this situation, because the steam generators are still removing heat, and that keeps natural circulation going. We don't lose enough primary inventory to get into that situation.

MEMBER BANERJEE: That's a sort of presumption that you don't lose primary inventory.

MR. FLETCHER: Well, I guess the question is, how do we lose primary inventory? If there is a LOCA clearly you're losing it. Here the only break we have in the system is the pump shaft seals. Which --

MEMBER BANERJEE: That's the only postulated break.

MEMBER SHACK: I think in his case you

got to remember a LOCA is a good thing.

MR. FLETCHER: That's correct.

MEMBER SHACK: If he busts that thing, he - he wants to lose pressure.

MR. FLETCHER: If you lose primary pressure, you lose --

MEMBER BANERJEE: Well, what happens is, you hold up the pressure, because you tend to - you know, you pop open your PRV, it goes down, it goes up again, and you lose inventory, right? But I just want to understand the sequence of events. Why don't you uncover the core in the early stages if you rapidly depressurize and boil off on the secondary side?

MR. FLETCHER: Because if you remove heat to the secondary then the primary circulation keeps going. And you can't get into the liquid hold up situation that you are talking about.

MEMBER SHACK: We're liquid solid at that point on the primary.

MEMBER BANERJEE: You're liquid solid, that's true. But if you've lost some inventory it breaks the natural circulation.

MR. FLETCHER: The only LOCAs we have considered are pump shaft seal leaks. The base case is at 21 GPM per pump, which is a small rate. But we have also evaluated much higher shaft pump seal rates up to 480 GPM, which is considered the top end of the pump shaft seal leaks. We haven't looked at combined LOCAs such as a late break or something like that that would --

MEMBER BANERJEE: You don't lose enough inventory to uncover the top of the U-tube?

MR. FLETCHER: Eventually we do, but we get there by pressurizing the primary system and expelling the water from the pressurizer safety release valves.

MEMBER BANERJEE: The reason is ask this is that many reactors are considering rapidly depressurizing the secondary side to try to depressurize the primary side, and all of them have this problem which occurs, which is that you hold up liquid, then on the riser side, you lower the core level so you uncover the core at relatively high power.

MR. FLETCHER: We have looked at pre-core damage operator intervention, which is just what you stated. But it assumes that you have turbine-driven auxiliary feedwater available to keep the secondary wet.

MEMBER BANERJEE: Right.

MR. FLETCHER: We have analyzed that, and we found that --

MEMBER BANERJEE: You are going to have it for four hours or something you said.

MR. FLETCHER: We've analyzed starting this steam generator what I call a feed and bleed, where you on purpose depressurize the secondary and inject auxiliary feedwater. We have looked at that being implemented 30 minutes into the event, which is still while you have quite a bit of water in the secondary system, and the primary system is full at that time. And our analysis shows that that strategy for operator intervention is successful.

MEMBER BANERJEE: The problem with it is that most people are using the wrong flooding correlation at the elbow, so that what happens is that you show that you get into natural circulation after looking at what you used, but in fact, you are flooding. So there is no refluxing period. So that is because they don't put the elbow effect in the flooding correlation.

And your velocities are a factor of two to five higher typically - five to two or whatever - at an elbow. So you can be completely wrong on this one. Because the most recent experiments on the Dresden - that's not the station but the lab - have indicated that using flooding correlations from vertical tubes gives you very, very much higher velocities than at that elbow. So I would like to understand exactly where you would uncover the core at the early stages.

MR. FLETCHER: We have not seen any situation that goes to reflux cooling in the analysis we have done.

MEMBER STETKAR: Don, can I ask you a question? You mentioned something that is kind of important to me, you said the only cases you've ever run you've always assumed you have a turbine-driven auxiliary feedwater pump. Think about Sanjoy's scenario. Suppose I have only motor-driven auxiliary feedwater pumps, and the station blackout, and I quickly depressurized the secondary side of the steam generators. How does that affect your analysis?

MR. FLETCHER: Okay, we have not analyzed that.

MEMBER STETKAR: Okay, but you thought about the problem an awful lot. How would you suggest that the results would change?

MR. FLETCHER: When did we stick up a

valve on the second --

MR. BOYD: Is it on the third lift did we stick open a valve?

MR. FLETCHER: Are you talking about the old analysis? It was actually on the first lift.

MR. BOYD: Okay, so we have done that scenario when we have depressurized the secondary side quickly and with no aux feed.

MR. FLETCHER: That is correct. With no aux feed. He's saying you do have aux feed.

MR. BOYD: No, no, no feed. I'm saying no initial feed to the steam generators. Steam generators are full initially.

MR. FLETCHER: Yes.

MR. BOYD: Lose all offsite power and then open up the secondary relief valves relief valves quickly. Because that cools down the primary side drastically and it delays the transient. We don't see these types of behaviors you are talking about because we haven't combined that with somehow getting into a situation where we got a big LOCA and the water level is down at the same time.

MR. FLETCHER: We've not analyzed a primary LOCA in addition to everything else that is being assumed in this.

MEMBER BANERJEE: You haven't varied the leak rate on the primary side. It's not a big LOCA, but enough to uncover the top of the U-tubes.

MR. FLETCHER: We have evaluated RCP shaft seal leak rates up to as high as 480 GPM.

MR. BOYD: As high as about 1600 or 1700 GPM, right.

MR. FLETCHER: Which is a fairly large break. The old analysis, when we started this five years ago, the original assumption was, we stuck open a steam generator PORV valve early, which would be the situation you are describing with no aux feed available whatsoever. And that leads to the same type of event that we are going to show you here.

MEMBER STETKAR: Earlier though than the timing you have here?

MR. FLETCHER: About the same.

MEMBER STETKAR: Oh, is it? Okay. That's what I was looking for.

MR. FLETCHER: The assumption of the stuck-open secondary relief valve really doesn't affect the outcome of the analysis we have looked at here, which are clearly assuming a small leak rate from the secondary.

MEMBER BANERJEE: Okay, you can go

through it. The real issue here is whether you do get earlier core uncovery then, your number here which is what, 2:25 minutes or something.

MEMBER SIEBER: Yes, it seems to me that you have to be reasonable in how many breaks and how many failures you assume. Because the probability of all these things occurring at once in large measure becomes pretty remote. I think pump seal failure is - it can happen under the loss of AC power and failure of diesels, have relief valve, inadvertent relief valve, but multiple failures after multiple failures I think get us out of range. Though I think where you are headed it's in the right direction.

MEMBER STETKAR: Just for a little perspective, there is at least one plant design that automatically blows down the secondary side. Operators hands off very quickly under these types of conditions. And that is the particular design that I think Sanjoy and I were both asking this question.

MR. BOYD: And I guess we're saying we have run that, we just haven't - and with substantial LOCAs, but not working the LOCAs to get it into the situation that you are concerned about. We can probably get into it.

MEMBER BANERJEE: Yes, let's carry on.

CHAIR POWERS: Let me ask one question. Based on your picture, you are looking at accidents that all involve the loop seal intact. And my recollection during our discussions prior to publication of NUREG-1740 that question of intact boot seals arose a lot. Have you done analyses with loop seals either gone or intermittent?

MR. BOYD: Yes, we do - we have looked at the loop seals gone, and we looked at the conditions that would lead to the loop seals clearing. We have looked at different pump suction heights to see the impact of that; different nodalizations on the vertical legs; and then done hand calculations to verify the pressure differences across the loop seals. So that has been studied.

And we do predict the loop seals to clear under some conditions.

MR. FLETCHER: And we will talk about what those conditions are.

CHAIR POWERS: Good. I just recall that as an issue that occupied some few minutes at least in our discussion.

MR. FLETCHER: The loop seal issue is key here. If the loop seal remains filled with water you end up with one type of behavior in the loop. If the loop seal clears, then the steam can flow all the way around the loop, the tubes are hit with the hot steam, and tube rupture is highly likely in that situation.

MEMBER BANERJEE: Do you have four or three loops depending on the PWR you are looking at?

MR. FLETCHER: Yes, the standard plant we are looking at is a four-loop Westinghouse Plant, and we have modeled each of the four loops individually.

MEMBER BANERJEE: Each of the loop seals will clear at different times. Do you take that into account?

MR. FLETCHER: They have the capability in the model to clear at different times if that is the case, yes.

MR. BOYD: In addition to clearing the loop seal, the loop seal could clear intermittently. You also have to clear the lower downcomer before you get that full loop circulation. So what we find is it's not something the loop seal can just kind of clear like that. We really need to boil it off and clear it in that way.

MR. FLETCHER: Okay, so back to this slide, the goal here was to give you an idea that we've got a secondary side boil off, and then we've got a LOCA and a primary side boiloff through the relief valves. And if at some point the core uncovers, we have this three-dimensional natural circulation flow patterns it sets up, assuming the loop seals remain intact, which is the typical behavior. And then inventory continues --

CHAIR POWERS: Let me ask you a question.

MR. FLETCHER: Yes.

CHAIR POWERS: You said that is the typical behavior. I will surely admit that that is the typical behavior in a calculation. Is it in fact typical behavior we would expect in plants?

MR. FLETCHER: Yes. Well, from our hand calculations of the pressure differences and the work we have done to try to see if that makes sense, we find that it is only under certain conditions. You have to preheat that water, and then you have to have a very large LOCA. And we have a LOCA of about 17 or 1800 GPMs, and then we can start flashing and boil off that water and clear the loop seals in time before something else fails.

So - so we've tried to map it out. Clearly there is uncertainty with what we have done.

CHAIR POWERS: And that I believe. I believe that's what the ACRS was asking was to map it out and say what conditions.

MR. FLETCHER: That's right, and we have tried to do that. And he's going to present some maps in his presentation. He's created a good deal of maps. He's going to present two as examples of that.

CHAIR POWERS: Very good.

MR. BOYD: Okay, so we've got the water level continuing to go down. We've got this 3-D natural circulation pattern going on. We've got the system slowly heating up. The next step is that we start to get fuel failures, and we start to get a rapid oxidation of the core. At some point the core reaches a peak oxidation and the power from that reaction will be five to 10 times the decay power at that time. So what you will see in the temperatures coming off the core, there will be an elbow, and it will go up at a much higher rate. And at this point we have jumped off the cliff, and something bad is going to happen, and we are looking for failures.

And it's at that point where we predict, shortly afterwards, where we are going to predict either a hot leg, a circ line, or a stressed tube to fail.

Now I wanted to just give a few notes on the way we do the calculation. So as to give some background. This is a simple diagram of the RCS system. I want to make the point that we use SCDAP/RELAP 5 as a screening tool. So when we started about five years ago, our PRA group came to us, and they had a list of about I think it was 1,400 scenarios is where we started. And we tried to do calculations at various points to eliminate large groups of these. But we started with a very large group of scenarios. We're running SCDAP/RELAP, and we're monitoring failure points at the hot leg, the surge line, and the steam generator tubes. And we think we have a fairly conservative screening approach, and we're trying to find scenarios where it's potentially - or where at steam generator two could potentially fail prior to one of these other components.

So we run the calculation, generally with no actual failures. When the hot leg is predicted to fail, we don't actually open up a break in most cases. We continue to run, so that we can predict subsequent failure times.

For the high-dry-low scenarios in the

four-loop Westinghouse plant that we are looking at, we predict the hot leg to fail first.

MEMBER BANERJEE: Is this due to thermal stresses or what?

MR. BOYD: It's a creep rupture calculation that we do.

MEMBER BANERJEE: So is the hot steam running back - I mean running up, and the cold steam running counterclockwise?

MR. BOYD: That's right. And the top of the hot leg would be subjected to the heat transfer from that hot steam, and these temperatures, you know, we are melting the core at this point, we have temperatures coming off the core that are extremely hot, beyond the metal masses. The only thing that is saving the hot leg at this point is it's 2-1/2 inches thick, and it takes a little bit of time to get that heat into the hot leg.

So the points I wanted to make here is that we predict the hot leg to fail first. And then what we do is, we look at the steam generator tubes, and we apply stress multipliers to them, so we will double the stress on the tube, we will triple the stress on the tube, we will quadruple the stress on the tube. And the idea is, we want to find out what additional stress is needed to fail a tube prior to the hot leg. Now we will pass that off to the materials people to tell us what that means from a flaw perspective. As an example, half an inch flaw three-quarters of the way through the tube would fail in testing it about the same time as a pristine tube with the stress doubled, a stress multiplier of two.

Our idea, though, is to screen the calculation. And we screen up to - we consider that if we triple the stress and it fails, then we capture that as a potential tube failure. We don't believe that tubes with flaws in them that would equate to a stress multiplier are in the plans but that's our screening criteria.

MEMBER BANERJEE: About how long after the core uncovers does this happen? Core uncovery, what, dropped it to 25 minutes?

MR. BOYD: Don has got some exact numbers here, or closer numbers here.

MR. FLETCHER: In our base case which we will present later the hottest tube with no flaws undegraded strength fails about six minutes after the hot leg, about 360 seconds.

CHAIR POWERS: You have not unreasonably for this kind of presentation presented very exact things. You said the hot leg fails first. And one of the primary issues that the ACRS was concerned about was the uncertainty in those kinds of numbers. What kind of uncertainty would you attach to those numbers, and how do you come about getting that uncertainty?

MR. BOYD: We looked - we did a simple set of sensitivity studies. We did not do an elaborate uncertainty analysis. And what we would look at, first we ran - I would say we ran about 100 runs just to see what impacts these failure times. We had a PERT meeting and tried to identify the key phenomenon, and then we tried to identify ranges that they could vary.

Some of the key here is to identify parameters that affect the hot leg and not the tube. So some issues like the core heating up faster, well, all that heat to get to the tube goes to the hot leg, and we don't see a difference in the relative timing failures, which is really what we are after.

So we identified a list of things that can impact the tubes and not the hot leg and vice versa, and then we varied these parameters to try to get some idea of what our uncertainty would be.

CHAIR POWERS: I'm not being critical, I'm being curious here. What was the inhibition to launching a rigorous uncertainty analysis, where you defined distributions for each one of these parameters and sampled from them, the sorts of things that were done in connection with say the pressurized thermal shock?

MR. BOYD: That's right. We looked at that. And we - I may not be speaking for management I guess from the NRC, but I believe one of the issues was the cost and the amount of effort involved with that to do that properly. So the approach we took is that we were going to make a decent estimate of our uncertainty and then feed it into the system. And then we were going to watch that and see how big an impact our uncertainty had on the final answer.

And if our uncertainty was critical, then we would then have to go back and do the refined uncertainty analysis. But there are of course in a multidimensional problem like this there is a whole host of uncertainties.

In the end our calculation boils down to a number between zero and one, at the end of an event tree.

CHAIR POWERS: If you take your

confidence levels high enough all answers are zero to one, I'll admit that.

Just as a heads up I would expect this uncertainty issue to come up in the committee discussion, the full committee discussion. I just can't imagine Professor Apostolakis not raising the issue. So you may want to think about explicitly speaking to the uncertainty issue.

MR. BOYD: But to summarize our approach was to make an approximation of the uncertainty so that we could feed it into the PRA system. Now of course with our thermal hydraulic conditions that went to, let's say, the hot-leg failure, we feed that into a three-dimensional finite element model that the materials guys can operate. And then from there they also can predict an uncertainty on the failure times.

So uncertainty is being calculated along the way, and then when it is integrated into the PRA model, there is again a final assessment of what we believe the uncertainty would be.

MEMBER SHACK: Of course I need to make my usual statement that it all depends on the flaw distributions in the steam generator tubes.

CHAIR POWERS: No, nothing matters, it's

just the flaw distribution. Yes, that is a boundary condition on all our discussions. A well known boundary condition.

MEMBER BANERJEE: But still it's sort of a tradeoff between flaws and temperatures and pressures, right?

MEMBER SHACK: Well, just to go back to Chris' point, what we found when we did the finite element analysis, we varied things like the materials properties. And from our point of view his uncertainties were driving the picture. In other words we could change our uncertainties and all it did is sort of change the absolute times; the relative scenario wouldn't change. He made a change in his thermal hydraulic model, and all of a sudden the failure point shifted from the surge line to the hot leg. And that was sort of the driving condition.

But we sort of found that with a sort of fixed thermal hydraulic condition, our parameters didn't change the relative time. We can change absolute numbers, whether it's 13,000 seconds or 17,000 seconds, but the relative numbers, but we felt we were being driven by the thermal hydraulic uncertainties.

MR. BOYD: I would argue with you there

that what we were doing --

MEMBER BANERJEE: If you try to reduce the uncertainties.

MR. BOYD: That's right, all the uncertainty came from the material - no. What he's talking about though, at the time we were doing that, we were removing large biases. For instance the surge line, when you connect it to the sides of the hot leg in SCDAP/RELAP it draws all of the fluid out of the upper hot flow, even though it's connected equally to both pipes. And that's because one pipe in a sense is far upstream from the other pipe.

We corrected that based on our three-dimensional analysis, and that shifted the surge line.

MEMBER SHACK: Yes, I didn't mean to address that, but what I was saying is that with a fixed thermal hydraulic input, it seemed that everything we did to the material parameters wouldn't change the relative --

MR. BOYD: And I guess what I'm saying is that after we went through over years and looked for biases and removed them, the information we gave you settled down quite a bit, and from your perspective it probably looked a lot crazier than it really is.

MEMBER SHACK: Well, it certainly drove the expense up.

MR. BOYD: All right, so I wanted to make a few points here. In our screening calculations we are looking at three locations, the thick hot leg, the moderate surge line, and the thin tube. Typically we don't open up a hole when we have a failure. We continue to run so we can predict the subsequent failures. We add stress multipliers. I wanted you to get the concept that we increase the stress on the pristine tubes such that we can find out what it takes to fail a tube prior to the hot leg. We have a screening criteria three.

And some other points to make. If we open up a failure of the hot leg, which we did in some calculations, and we assumed an 11-inch hole when the hot leg broke, that depressurizes the system very rapidly. Within maybe 30 seconds we've reduced the load on the tubes 10, 15 seconds, we've reduced the stress on the tubes. We're not going to fail tubes once the hot leg fails. We are going to put these fission products into the containment.

We also looked at it from the other way around. We assumed enough stress on the tubes such that it would fail prior to the hot leg, and then we opened up what would be the equivalent of two tube areas, when the stress multiplier hits two. So we open it up a minute or two before the hot leg failure time, and we just let the system start to depressurize.

Now that's a relative slow depressurization; the hot leg still fails. So then we went to a stress multiplier of three, so we could open it up - a weaker tube, we open it up I don't know five, six, seven minutes before the hot leg fails. That still was not enough of a depressurization,. The hot leg still fails.

At that point what we're saying is that when we predict a tube failure in the typical case we are still going to fail the hot leg, and we are still going to push all the fission products into the containment. The SOARCA guys have run this type of calculation. They see the same behavior, and they see that it significantly reduces the release of material. Most of the material gets hung up in the containment somewhere. You will still get a slow leak out through the tube, but it is not driven by 2,000 psi any more. It's driven by containment pressure.

MEMBER BANERJEE: Is this true if you clear a couple of loop seals then you have a natural circulation path through the steam generator, a couple of steam generators, or one maybe. So imagine you have cleared a loop, so now you have a path to circulate, right?

MR. BOYD: When we clear a loop seal, what we get is - we don't get that three-dimensional counter-current flow pattern. We get a full direct circulation. And now we are going to put flow from the core through the hot leg through the inner plenum right into the tube sheet. At that point we don't benefit from the mixing that significantly reduces the temperature, and we are going to fail the tubes earlier than the hot leg.

MEMBER BANERJEE: So that's the scenario that - why I was saying that if you had multiple loops it's possible that you will clear one or two loop seals and have a direct circulation, because it will bypass around in the downcomer.

MR. FLETCHER: Yes, the model will handle that. What Chris is saying is that the tube failure comes very quickly in the situation.

MEMBER BANERJEE: Yeah, but in that case the hot leg would fail maybe later, and the tubes would fail --

MR. BOYD: That case, we have not run the case. I mean in that case, we could fail a number of tubes and depressurize the system. That case is actually easier to calculate.

MEMBER BANERJEE: No, I know, but to me it seems a higher risk of bypassing the containment in some sense.

MR. BOYD: Well, risk-wise, it's a much lower probability that that would occur, based on the way we have mapped things out, defined loop seal clearing.

MEMBER BANERJEE: Is it because loop seals in most cases are difficult to clear? Is that why --

MR. BOYD: We're finding you need a very large LOCA to clear the loop seals.

MEMBER BANERJEE: Okay, well, that we need to look into. It's really the question I asked you whether you had models.

MR. FLETCHER: I will show you some results on that.

MEMBER SIEBER: Is that the only condition that would clear the loop seals, a very large LOCA? And the second question to that is, if you have a very large LOCA that does clear the loop seal, the steam generator is still intact, is that not correct?

MR. FLETCHER: But the tradeoff is that the primary system pressure - it's the difference between the primary system pressure and the secondary system pressure that is stressing the tubes.

MEMBER SIEBER: Right.

MR. FLETCHER: And if you have a very large LOCA, the primary system pressure comes down so far that the tubes are safe.

MEMBER SIEBER: Right.

MR. BOYD: So they are in the range of LOCAs that are large enough that we have looked at that can clear the loop seals and still have the pressure. And it becomes a risk assessment at that point.

MEMBER SIEBER: Yeah, and what range are those? Those are the ones that would generate steam generator tubes without rupturing the RCS, right?

MR. FLETCHER: That's correct. Shall we show him the map?

CHAIR POWERS: If you're going to get to it.

MR. FLETCHER: It is the example I'm

going to show.

MEMBER SIEBER: Okay, if you are going to get to it, that's fine, I'll wait.

MEMBER BLEY: There is something less than satisfying, uncomfortable, about turning on a second failure to protect you from the first, and giving credit to ourselves for that. And I guess the thing you worry about is, is that a figment of the particularly severe condition you are looking at? Could there be another accident that might get the tubes and not be essentially guaranteed to open up the hot leg? Because I suspect when we get into the details of risk, we are counting an awful lot on that second hole to not let things get outside of containment.

MR. BOYD: I don't know how to answer that other than we've looked at a large number of scenarios and have not come across that type of scenario that challenges the tubes with an induced failure, a thermally induced failure, without thermally challenging the hot leg and surge line.

CHAIR POWERS: Maybe we should make clear that what these gentlemen are looked at are the induced failures.

MR. BOYD: We are looking at severe

accident induced or thermally induced tube ruptures at this point.

CHAIR POWERS: Not looking at severe accidents initiated by tube generator rupture?

MR. BOYD: That's right.

MEMBER BLEY: Okay, thanks.

MR. BOYD: So the point of this slide was to let you know we are doing a screening calculation, and just a note that we have failed the hot leg to demonstrate - or failed a hole in the hot leg to demonstrate that that will protect the tubes if we depressurize. We've also failed tubes. We failed the equivalent of two tube areas and eight tube areas to demonstrate that even under those conditions we still fail the hot leg.

And again we are talking about the case where the loop seals are filled at this point.

I'll also note that we believe that our screening criteria has some slight conservatisms in it with our SCDAP/RELAP 5 model. These would come from the fact that when we fail a hot leg, we are looking at an unflawed hot leg, and we look at a stainless steel hot leg. I believe the failures are predicted at the carbon safe end of the actual nozzle, and this will fail a little bit earlier than what we predict.

We also use an average heat transfer coefficient. I call it a limitation of the SCDAP/RELAP model, but we only can use one correlation for the hot leg heat structure. So we use the average heat transfer coefficient. Now this hot leg has a length to diameter ratio of about seven, so we are in the entrance effects region.

And the heat transfer coefficient will be higher at the nozzle region. And when we pass this information along to the materials guy, we do give them the entrance effects. So they apply the hotter heat transfer, and they apply - they look at the carbon steel and the weld, and they predict the hot leg to fail earlier than we do. But for our screening purposes I just wanted to point out.

And when we look at a tube failure, we are assuming the flawed tube is just above the tube sheet in the hottest region of the plume; again, this would be a conservative assumption.

Okay, so here we'll move on to some easier slides, I hope. The thermal hydraulic predictions are integrated into the overall project. And Dave talked about this. This is an integrated project, and it's an iterative project, so you can think of this going around the loop many, many times before we get to an answer.

MEMBER ARMIJO: I'm sorry, I'm going to just pull you back with one question. When you failed the hot leg, is that strictly a pressure stress rupture failure, or are there other loads taken into account?

MR. BOYD: We do a simple pressure load, and that's it. We do a simple creep rupture, Larson-Miller creep rupture calculation in the hot leg when we fail it. Now when Argonne looks at the hot leg I believe they took some other stresses into consideration.

MEMBER SHACK: Yes, it's still a creep failure. We just have a much more detailed model than they do in the SCDAP/RELAP.

MR. BOYD: We fail a one-dimensional infinitely long stainless steel pipe with the pressure and thermal loads on it.

CHAIR POWERS: Are Larson-Miller type approaches appropriate for these kinds of heat fluxes and temperature?

MR. BOYD: This was a question that was asked awhile ago. I don't personally have the information on that, but I guess I was assured that we were doing something that was reasonable when we started this. And again this would go along the lines of the materials guys looking over our shoulder when we do these types of calculations.

MEMBER ARMIJO: I am just wondering, you do a thermal analysis in that nozzle and hot leg area to show that you know all the stresses, where they are. I would imagine especially if you have some sort of flow, natural circulation.

MR. BOYD: The finite element model at Argonne does that.

MEMBER SHACK: They give us the very detailed heat transfer conditions. But we do the full thermal analysis of that.

MEMBER ARMIJO: They know if it's pressure or whatever.

MEMBER SHACK: All the stress. I mean it's not a simple PR over T kind of calculation.

MEMBER ARMIJO: Yeah. That's my question.

MR. BOYD: In SCDAP/RELAP I think we are talking about the simple PR over T simplistic calculation in our screening calculations. And we have compared our screening calculations directly to the 3-D model. They fail - and I forget the number -

I believe it's two or three minutes earlier than we do.

And when we talk about two or three minutes, or five minutes, that is substantial in this particular transient. Even when we go out to 18 hours, the action starts when the core oxidizes, and the core power goes up to five or 10 times the decay heat power, so it condenses the heat up into something more like 20 minutes. So I mean a 10-minute margin would be huge, because you know the temperature differences over 10 minutes are very large. So even two to three minutes can be a substantial difference when we are talking about these heat up rates.

MEMBER STETKAR: Can I - this will help me tomorrow I think a little bit. I have to publicly admit my utter lack of knowledge about anything related to materials.

CHAIR POWERS: It's a truism of everyone.

MEMBER STETKAR: Yes, but most people won't admit it. You mentioned that when the hot leg opens you, you opened up an 11-inch hole, and that's a good thing for eventual releases.

In the real world when one of these

things fail do you actually open up an 11-inch hole, or do you open up a small split and relieve all the stresses and actually have a relatively small hole? I'm asking that of the materials people.

MEMBER SHACK: We do a very good calculation up until the time of failure. After that it gets more difficult. I don't think for that particular case we actually try - I can't remember modeling which we - it's going to - because a small hole won't relieve the pressure, it will rip to some larger hole.

MR. BEAULIEU: A 30-inch pipe is going to have enormous loads at this point.

MEMBER SHACK: It's - I'm fairly confident, an 11-inch hole doesn't sound unreasonable. Whether we can predict that or not, I think that is an engineering judgment that says, you've got this material so hot, and until it depressurizes that thing is just going to keep opening up.

MEMBER SIEBER: Yes, but everything is changing while you are doing that, so it is very hard to analyze.

MEMBER BLEY: And you are leaking through the holes, too.

MEMBER MAYNARD: The 30 inches was an input, that wasn't a calculated --

MR. BEAULIEU: Oh, of course not. Not from our one-dimensional code. Eleven inches happened to be about the size of the surge line, so by doing it that way we also looked at the potential for the surge line dislodging itself.

MEMBER SHACK: We've done these calculations much more detailed for a steam generator tube where we try to follow the crack as it opens up. It opens up from a crack to basically a round hole. And I think something very similar would happen, we would start with a small tear in the hot leg. But we didn't try to calculate how that would grow. But I think on an engineering judgment basis it would grow pretty rapidly and to a fairly substantial --

CHAIR POWERS: Is engineering judgment another word for wild-ass guess?

MEMBER SHACK: That was yesterday.

MEMBER SIEBER: You are talking on the order of seconds, right, for the full failure to develop?

MEMBER SHACK: No, we're talking minutes.

MEMBER SIEBER: Minutes?

MEMBER SHACK: It's down in Chris' one to two minutes, probably. But it's still rapid I think compared to the difference we see between the hot leg failure and the tube failure.

MEMBER SIEBER: I've witnessed a number of coal-fired failures and they're fast.

MR. BOYD: I am going to try and page down through that slide so I can avoid any more questions.

CHAIR POWERS: A strategy many have tried before the ACRS. They have universally failed I think.

MR. BOYD: Okay, so we've got an integrated project of thermal hydraulics as one small part of it. The thermal hydraulic issues, I pulled out a few issues from NUREG-1740. Some of the concern from the ACRS is that we have this one-dimensional code, and we are trying to predict, obviously, an important part of the prediction is this three-dimensional natural circulation phase, because that is where the failures occur, and that is where we are pulling heat transfer rates and things from, so we need to get those mass flows and heat transfer rates properly.

In the past it's typically been done

through scaling from a set of one-seventh scale experiments. There is always some criticism of that. There is concern that the mixing may be overestimated. Concern that we didn't have leakage in the testing. And concern over our sensitivity studies.

We presented this information to the ACRS in 2004, and we presented our CFD models, and a fairly decent assessment with SCDAP/RELAP of the entire picture.

The staff had two concerns they wrote to us in a letter. They requested one that we take our CFD models which were focused on inlet plenum mixing, and extend them such that we could also predict the hot leg flow. I think Graham Wallis used the term, you guys don't know what the hot leg flow is, and have no idea. We argued with them, but in the end we went ahead and modeled the hot leg flow.

They also requested us to look at the reactor coolant pump seal - I'm sorry, I said coolant pump seal, the loop seal clearing issue, and to kind of - which is the same question asked a few minutes ago.

MEMBER SHACK: You covered everything with the language.

MR. BOYD: Right. That looks like some cut and paste and bad editing.

MEMBER BANERJEE: Well, if I look back, I was just reading some old stuff, really the request was beyond that, it was to develop a much larger 3-D simulation, and staff argued that the codes you are using, it would need about 10^9 meshpoints, and therefore it was impractical to do.

I recall reading that now. It was because I guess you couldn't paralyze your codes or something.

MR. BOYD: No, the question was, we were making the argument that in order to model the vessel circulation we would need to model the vessel. And modeling the vessel is extremely difficult because of all the structures.

MEMBER BANERJEE: Well, at least the top part of the vessel.

MR. BOYD: Well, we were arguing that we needed to model the bottom part of the vessel also. Now in the end the ACRS letter said no you don't, you just model the top part. So in the end we modeled the top part of the vessel with the simplified core region that did not require that many cells. So we created a vessel, a much simplified vessel, such that we could just add heat to the core region, and predict the countercurrent natural circulation flows.

So we basically ended up doing --

MEMBER BANERJEE: Yeah, I think the steam part is what's important. I mean the liquid is boiling off more or less. It's only that portion, what's happening to the full distribution, things like that.

MR. BOYD: And in the end that's what we ended up doing.

MEMBER BANERJEE: Okay, we'll see what you did, so let's go on.

MR. BOYD: So the action plan tasks A through G, basically these tasks are focused on SCDAP/RELAP 5 and the CFD work. We want to perform plant sequence variations. We have done a whole variety of these. We want to reevaluate the system code assumptions, and we have looked at a whole series of assumptions, everything from pressurized draining, hot leg radiation, core nodalization, downcomer nodalization. We've really scoured the deck and looked for things we could consider changing, and consider the effect of changing them. And we did update the model as necessary.

We needed to estimate the two temperature variations from the one-seventh scale data. We did that, but then we demonstrated that we could even expand on that with the CFD predictions. We want to perform a more rigorous uncertainty analysis on the system level predictions, and this is where we estimated the uncertainty through sensitivity studies.

Back to CFD we requested to benchmark our tools with the available data, and that was done, and that's NUREG-1781. In Section f we are estimating the uncertainty due to core melt progression, and this was part of our sensitivity study that went into the uncertainty estimation. We did a variety of changes to the oxidation rate and circulations that would affect the core melt progression.

And then the final one is to perform additional experiments. And in this case we looked at the results we had from the computational fluids, and we looked at the experimental results we had, and we felt that the experiments at this point were not necessary, or would not be worth the expense to get us further from where we already were.

So the talks today will tell us about NUREG-6995 which summarizes all the system level work, and that's Don's presentation where we'll get the highlights of that. That covers the system level plans, and then I'm going to talk about the CFD analysis in 3.4c, e and g. And this is just a quick run through the highlights from NUREG-1781 and 88 NUREG-1922, which is in draft form right now.

MEMBER SIEBER: I have a quick question. You assume that on loss of all power that the loop seal fails, and that the leak rate is, what, 440 gpm?

MR. BOYD: We are talking about the reactor coolant pump seal leakage.

MEMBER SIEBER: Right.

MR. BOYD: This is a tough issue to know what that leakage rate is from our perspective. So we assume a whole spectrum of leakage rates, and leaking at different times.

MEMBER SIEBER: It depends on the model of the seal and the pump vendor.

MR. BOYD: That's right.

MEMBER SIEBER: How much it leaks. And 440 is the outside if my memory - the high limit.

MR. BOYD: That's right. That's like all the stuff, all the seals are just somehow disappear. We found that to be a very difficult issue, a thorny issue to sort out. So the way we addressed it is, we just mapped out the leak rates that are problematic or not problematic, or mapped out the consequences of various leak rates, and assigning a probability to what leak - to whether you are going to have 180 gallon per minute leak or 440 gallon per leak, we did not do that.

MEMBER SIEBER: You get roughly the same plant response but a different timing, I presume. Depending on the leak rate.

MR. BOYD: No, the leak rate, if it's 21 gallons per minute, which is the default leak rate. That's like everything is normal. We do not depressurize the primary side, and we keep the stress on the tubes, and we potentially fail them. Now if we leak at 180 gallons per minute, which is another standard leak rate based on various change in the seals, and that's actually more probable than the 21 gallon per minute leak rate at least from the PRA numbers I've seen, that leak rate will depressurize the primary system and take the load off the tubes and preclude tube failure.

MEMBER SIEBER: Have you searched for the seal leak rate where you can distinguish between what fails, like the hot leg or the steam generator tubes, and if so what would that leak rate be?

MR. BOYD: We did not do incremental

leak - we did a whole spectrum, 21, 60, 90, 120, 180. We did kind of a spectrum, so we have an idea. I believe whenever we have a large enough leak rate to depressurize a system, we don't fail the hot leg either, do we? Don't we end up eventually with a lower head failure in that case.

MR. FLETCHER: I believe that is correct, yes.

MR. BOYD: Typically if you are going to fail the hot leg you are sending heat out into the loop, so the tubes end up seeing that heat also. But when you depressurize a system that natural circulation goes way down, and we end up slumping the core and eventually failing the lower head in that case.

MEMBER SIEBER: Okay.

MEMBER STETKAR: Chris, are you going to talk more about this later? Because this is a really important topic for me in particular from a PRA side. I don't care about seal failures. It's the size of the break where you transition. Will you talk about that?

MR. BOYD: We will talk about that. This was supposed to be the 10-minute introduction.

CHAIR POWERS: And it has fully lived up

to expectations for 10-minute introductory presentations.

MR. BOYD: This is a warm up.

Okay, so this is where I will transition over to Don, who is going to talk about the SCDAP/RELAP 5 model.

CHAIR POWERS: I think - in looking at your slides, I'll go ahead and take a 15-minute break here, and we'll come back at 20 minutes after.

(Whereupon, the above-entitled matter went off the record at 10:05 a.m. and resumed at 10:24 a.m.)

CHAIR POWERS: Let's come back into session.

Before we resume our technical discussions, our reporter would like to advise us on the use of the microphones, because they are so different than the ones we are used to.

(Off the record comments.)

CHAIR POWERS: Thank you very much. Don, I guess you are up.

MR. FLETCHER: Don Fletcher from ISL Idaho Falls, and the teleslide also acknowledges five other analysts from ISLL that have been involved in the program.

The purpose of the SCDAP/RELAP 5 thermal hydraulic analysis is to determine the sets of plant configurations, conditions, and accident event sequence scenarios that can lead to containment bypass through induced steam generator tube failure.

The risk associated with the accidents is affected by the order in which the reactor coolant system component structural failures occur. In particularly if a hot leg pressurizer surge line with the reactor vessel lower head fails, these failures lead to depressurization of the RCS into the containment, and the depressurization of the RCS precludes subsequent steam generator tube failures and containment bypass.

On the other hand if steam generator tubes fail first this leads to a discharge from the RCS into the steam generator secondary system, and may lead to containment bypass by relief through the safety relief valves of the steam generators or through other leakage paths in the steam generators.

The depressurization that the RCS sees in that situation is not sufficient to preclude subsequent failures of the hot leg.

MEMBER BLEY: Doesn't that depend on - I guess it doesn't depend on how many tubes rupture, because you still have the safety valves, is that the trick?

MR. FLETCHER: You're talking about the release?

MEMBER BLEY: Well, and why don't you depressurize.

MR. FLETCHER: Well, if you have a huge number of failures of tubes, of course you could depressurize the primary system down. But the limiting pressure would then be the safety relief valves on the secondary side, the open exit point pressure.

MEMBER STETKAR: Let me follow up on that a little bit. I thought that we were concerned with high-dry-low conditions, so by definition secondary side pressure is low. So if I already have a preexisting open steam relief path like an open surge relief valve, I'm not relying on the secondary side safety valves to hold secondary side pressure.

MR. FLETCHER: That's correct. In order to get the low secondary side pressure you need to have a stuck open relief or a significant leakage path of some kind.

MEMBER STETKAR: Initially?

MR. FLETCHER: Initially, yes.

MEMBER STETKAR: That's right. So I

don't have that secondary side safety valves holding up pressure for me?

MR. FLETCHER: That is correct.

MEMBER BLEY: And I guess if you are getting - this probably goes back to things Bill was talking about - if you are getting the tubes rupturing first, due to this thermal effect, it wouldn't seem to me it would be reasonable to expect only one or two. It seems you would expect larger numbers, because larger numbers seeing the same thing. So what happens if you get four or five, did you look at that? At what point do you get away from having the secondary failures.

MR. BOYD: We did look at multiple tubes, and we failed the tube when the stress multiplier was two. Now I would have to let the tube integrity guys speak to this better than I, but it's my understanding that to have a tube that is that flawed in the generator would be rate.

Now to have --

MEMBER BLEY: Well, we're looking for rare. I mean that's what this is all about.

MR. BOYD: It's got to be in the generator. It's got to be in the hottest part of the plume. And then it's got to have neighbors with the same flow density. So we could conjure up situations where all the flow tubes are in the hottest area.

CHAIR POWERS: None of those are independent probabilities. If you have a highly flawed tube, very likely its neighbors will be flawed.

MEMBER BLEY: Whatever mechanism caused one may have caused the others. Are we treating these as independent when they are not independent?

MEMBER SHACK: There's various meanings of what you mean by very likely here. Yes, it's certainly more likely that you will have one than --

MR. BOYD: Well, we failed up to eight tubes. And we still failed the hot leg. We said eight tubes are in the hottest region, and have a stress multiplier of two, and they will open up on the side with a flow area of one tube. And we failed them at that point. And then we also -- we still got the hot leg failure. And we failed them with our screening numbers which are a little conservative. In reality the hot leg would have failed even sooner if it were passed on to the materials.

MEMBER STETKAR: And when you failed those tubes they were open to let's say atmospheric pressure?

MR. BOYD: They were open to the atmospheric pressure, that's right.

MEMBER BLEY: You said one thing which confused me. You failed eight tubes with an equivalent area of one full tube?

MR. BOYD: Each tube had - we failed - we put a hole in there with eight tube flow areas. And that was not sufficient. We also looked at two tube flow areas first, and then we went to eight, and then we went to a stress multiplier of three which would fail it even earlier.

MEMBER BLEY: What happened with that?

MR. BOYD: In that case we also failed the hot leg. There was a delay I think of maybe 30 seconds in the hot leg failure, so there was some small impact of a depressurization. But that may or may not even be realistic. These calculations can vary based on the timing, core cycling, and things like that.

MR. FLETCHER: Slide 3 provides an overview of the SCDAP/RELAP 5 thermal hydraulic evaluations, and the containment bypass, that has been run since 1998. We've already discussed NUREG-1570 and NUREG-1740 analysis, and the issues of loop seal clearing, reactor coolant pump shaft seal leakage, potential limitations of the system code, considerations of steam generator tube leakage.

In 2000 - 2002 we made a significant revision of the SCDAP/RELAP 5 model to address the previous concerns. Important in those were revisions to replicate the natural circulation behavior. In the Westinghouse one-seventh scale experiment since CFD calculations, the expansion of the model to include tube stress multiplies for failures of average and hottest tubes.

In 2003 and 2004 we performed a significant number of sensitivity evaluations looking at variations in the mixing parameters, shaft seal leakage, steam generator tube leakage, core bypass issues, core damage progression, and other event sequence assumptions.

The February 2004 ACRS meeting was talked about already. The concerns there were the steam generator power fraction approach that we were using at the time for hot leg circulation. We've improved the model in that respect. The issue of lube seal clearing came up again, and also issues regarding the reactor vessel internal circulations. And it was an expression at that time for some better information on the flows of energy during the event, because it was quite confusing to the ACRS at that point.

In 2004 and 2005 we performed a number of analyses to support the PRA, looking at various options on the reactor coolant pump shield, seal set, shaft seal leakage rates. The operation of turbine driven auxiliary feedwater, various assumptions on the station battery depletion times, steam generator secondary steam leakage rates, and some of the operator interventions that might be possible.

In 2005 we had an NRC and consultant peer review and PIRT evaluation. This was in response to the ACRS recommendation that we do so. Some of the things that were talked about at that time were core axial mobilization of whether it should be expanded; the need for finer axial nodalization of the region of the steam generator tubes and the tube sheet; discussion of the need for a better way to come up with the hot leg circulation rate. We went to a hot leg CD rate.

Hot leg - it was determined by a CD that was calculated based on some experiments. We did some hand calculation, evaluations of loop seal clearing behavior, and then the PIRT was used to identify the uncertainty study, independent and dependent variables.

In 2006 we performed the energy flow and uncertainty evaluations. In 2007 we had a public peer review meeting where EPRI sat in and made some comments based on their map analyses, and in particular, they commented on the SCDAP/RELAP 5 steam-to-wall hot leg radiation model. They believed that we were under predicting the heat transfer in that respect. They also made some comments regarding the assumptions we were using for the hottest tube inlet temperatures, and that they were too conservative; and also that we were not considering the creep rupture failure in the hot leg nozzle carbon steel safe end, and that was being too conservative.

MEMBER ARMIJO: Were you treating the carbon steel the same as the stainless steel?

MR. FLETCHER: In SCDAP/RELAP 5 the model is too simple to do so. We're modeling the hot leg strictly with stainless, and the calculations were performed on that basis.

The - an analysis of that safe end would have to be done with a more complex model.

MEMBER ARMIJO: Did EPRI do that when they claimed that the carbon steel would be the weak link?

MR. FLETCHER: That was their claim, yes, sir.

MEMBER ARMIJO: Did they show you anything persuasive?

MR. BOYD: I believe the NRC has confirmed that also. W e have done an ABAQUS 3D finite element model, Argonne I believe modeled that in detail.

MEMBER ARMIJO: So is that basically where you would expect right now if this thing happened?

MR. BOYD: That is the hottest part of the hot leg is over at the nozzle end.

MEMBER ARMIJO: The porous creep rupture properties.

MR. BOYD: Right, that is where we expect failure.

MEMBER SHACK: But it is big and thick, so it's not intuitively obvious that --

MEMBER ARMIJO: It is big and thick. It's got good thermal conductivity, and --

MEMBER SHACK: But it's in the analysis, so you look at it. It wasn't in the SCDAP/RELAP analysis.

MEMBER ARMIJO: I'm not familiar with

the nozzle design, but where it comes down to be welded to the stainless steel I think it's not quite so thick.

MEMBER SHACK: Right, but there are big shoulders on it. It is something you have to do with a finite elements analysis.

MEMBER ARMIJO: Thank you.

MR. FLETCHER: The current report draft NUREG-6995 summarizes all of the previous analysis. And it presents a final base case analysis using an upgraded hot leg steam-to-wall radiation model. It also performs some screening analysis, where we are categorizing the events, the severe accident events, into groups that lead to containment bypass, might lead to containment bypass, or don't lead to containment bypass.

For those of you that are not familiar with SCDAP/RELAP 5, it's a combination of the RELAP 5 thermal hydraulic system, fluid flow and heat transfer models, and SCDAP course core severe accident models. RELAP 5 solves conservation of mass momentum and energy, to fluid formulation, nonequilibrium, non-homogeneous model with noncondensable gas that is trapped with the steam.

The SCDAP model severe accident core

behavior such as fuel rod heatup, oxidation, ballooning rupture, and so forth. The SCDAP/RELAP 5 is capable of predicting buoyancy driven flows in one-dimensional systems such as you might have in steam generator tubes that have long runs of vertical flows upward and downward.

What it can't do is handle the multidimensional effects, some of which are important for this application. This includes the mixing of the steam generator inlet plenum; the countercurrent flow in the hot leg where hot steam is flowing towards the steam generator, and cool steam if slowing back toward the reactor vessel.

And it can handle the mixing in steam generator tube bundles themselves.

The diagram that Chris showed earlier is repeated here.

MEMBER BANERJEE: How many parallel paths can you practically handle?

MR. FLETCHER: Practically?

MEMBER BANERJEE: I mean if you were dividing the tubes in the steam generator up in groups, how many?

MR. FLETCHER: The number, you could do individual tubes if you wanted to, I believe. From a practical viewpoint maybe something like 10 - 20 would be the most that I have ever seen used. But we have not done that. Mainly because the assumptions of which tubes are in which groups are imposed by the CFD analyses. And the CFD analyses define how many tubes are flowing upward, and how many tubes are flowing downward, in the steam generator. So we have modeled two sets of tubes, one that is the hot average tube where the steam is carried toward the outlet plenum of the steam generator. Another average cold average to that returns that steam to the inlet plenum with the steam generator.

And then we are modeling the hottest tube --

MEMBER BANERJEE: Do you have to iterate with the CFD analysis there?

MR. FLETCHER: There have been several iterations with the CFD analysis. And that frankly is the reason why our base case results have changed over the past five or six years; the models have improved. And I must say the iteration with CFD has been very well - the SCDAP/RELAP is often applied as you are well aware in situations where we are stretching beyond where it should.

So I think the CFD iteration with us, the tuning -- that is a bad word to use - the adjustment of the RELAP model to match the CFD behavior is a good way to go.

MEMBER BANERJEE: Let me understand. May you will explain this. If so you can do it later. But clearly you are providing some sort of boundary conditions to the CFD calculations as well. The interplay between these two calculations in the iteration is what you are understanding a bit more about.

MR. FLETCHER: The iteration has been done -- this is not a continuous thing. We are not doing this online if you will.

MEMBER BANERJEE: No, no, I realize. You understand something about the CFD and you feed it back.

MR. BOYD: It turns out that it's not as sensitive as it might be because of the way we use a discharge coefficient to predict the hot leg flow. So from Don's work I can get some upper plenum conditions in the vessel, and some steam generator secondary site conditions. I can go off then with only those two facts and then the geometry, the energy coming in, and I can predict now hot leg flows, and I can do sensitivity studies to see how sensitive it is.

And we are able using a densometric Froude number correlation to come up with a discharge coefficient that based on the density difference between the steam generator in the plenum and the upper plenum of the vessel, we can get the hot leg flows.

So I, and I'm making sure that I use conditions that are very relevant based on his calculations, now I feed that off to him, and now he's got a model that can predict an appropriate hot leg flow in his SCDAP/RELAP 5 model.

Now the next thing we are concerned about is bundle flows. Again, I can do a calculation where I again do this, and I can predict bundle flows. We call it the recirculation ratio, the ratio of the bundle flows to the hot leg flow.

And again then I can pass that off to Don. It's not so sensitive that we would - the way we are doing it with physically based correlations, for instance, with a hot leg flow, it's not so sensitive that we go back and forth. I found that discharge coefficient to be pretty constant over a wide range of conditions.

MEMBER BANERJEE: Let's say the flow in

the bundle is dependent on how things mix in the various inlet and outlet.

MR. BOYD: That's right.

MEMBER BANERJEE: So let's say the flow is going up in the plenum and I'm going to resolve that. That would take some fluid which is coming back from the upper plenum into the inlet plenum.

MR. BOYD: That's right.

MEMBER BANERJEE: So you tell him roughly the ratio of tubes going forward and tubes going backwards, and he puts that into his calculations.

MR. BOYD: That's right.

MEMBER BANERJEE: But it doesn't affect the boundary conditions that you use too much; that's what he's saying.

MR. BOYD: No it does not. The plant is forgiving. When we found that it is not super critical the fraction of tubes that are in upflow, that varies a little bit. And if we ran something with 35 percent of the tubes in hot flow, or 45 percent of the tubes, we would end up with the same general result in the end.

And all of the flow returning, after it goes through the hundreds and hundreds of L over D of steam generator tube length, it comes back at the secondary side temperature, because this is pretty significant heat exchanger we are going through. So in that case that boundary condition isn't very sensitive, and doesn't change.

So we found that the iterations back and forth are not that sensitive, and you do not have to bounce back and forth very many times before you are convinced that you've got the SCDAP/RELAP model predicting the mass flows and the temperatures around that loop that are consistent with the CFD predictions.

MEMBER BANERJEE: Yeah, we always go back to the most critical aspect of it, which is what you have shown. One loop seems clear and one not.

MR. BOYD: Well, this is a schematic. This is not trying to represent an actual situation.

(Simultaneous speakers.)

MEMBER BANERJEE: That's of course what separates the fact that you may massively fail the steam generators.

MR. BOYD: That's correct. I was trying to show on the right hand side the typical behavior, and by typical, I mean loop seals generally do not clear. You have to work at it to clear the loop seals, and I will show you the cases where we did that.

MEMBER BANERJEE: But that is done by your SCDAP/RELAP calculations basically.

MR. BOYD: Yes.

MEMBER BANERJEE: RELAP, let's forget the SCDAP, it's really RELAP.

MR. BOYD: The model looks at the loop seals, and if the loop seals are predicted to clear, because of the thermal hydraulic effects, the temperatures, the flow rates, pressures, the delta P's across the loop seal and so forth.

MEMBER BANERJEE: Then you got problems.

MR. BOYD: We go to the model on the left if that is the situation. We remove the recirculation path on the right hand side for that loop, and we do it on a loop by loop basis. And we retain the model on the left hand side in that situation.

MEMBER BANERJEE: In that case of course all the hot steam goes up, the back flow is very small.

MR. BOYD: There is no back flow, right. There is no mechanism for the back flow.

MEMBER BANERJEE: There is no

mechanism, and then you have the potential to massively fill.

MR. BOYD: Massively fill tubes, yes.

MEMBER ARMIJO: In that case, in this set up here, which gets the hot leg hotter faster? Is it the full loop circulation, or the counter-current natural circulation? From the standpoint of failing the hot leg which is the more - which is more likely, or are they both the same?

MR. BOYD: I don't think we have the exact answer for that. But we could imagine that when you've got full loop circulation, you are going to have larger flows down the hot leg, and we are going to get the hot leg hotter faster in that case. The problem is the tubes get hotter much faster; not just faster, but much faster.

MEMBER ARMIJO: Okay, so they are both accelerating.

MR. BOYD: Yeah, you can't get heat to the tubes without passing the hot leg.

MEMBER BANERJEE: The left hand scenario is what we --

MEMBER ARMIJO: More risky from the standpoint of bypassing the containment; is that the way it works out?

MR. FLETCHER: That is correct. That is correct. The nodalization that we are using for the situation where you have the circulating flows and the hot legs and the steam generator tubes as shown on this slide, the key points are that the hot leg has been split into two, an upper half and a lower half. And the steam generator tubes have been split into two, the tubes flowing forward towards the outlet plenum, and then the tubes flowing in reverse towards the inlet plenum.

MEMBER BANERJEE: Sorry, I just missed - the steam generator you work it out from your CFD calculations, what fraction will flow backwards?

MR. FLETCHER: The adjustments in this are made in the region of the steam generator, in the plenum - on this diagram it shows up in volumes 105, 106 and 107. The flow from the upper hot leg comes into a combination of 105 and 106, and the flow returning from the outlet plenum of the steam generator with the cooler steam flows into 106 and 107.

So the inlet plenum is modeled with 105, 106 and 107, where 105 is a hot inlet; 106 is a - I'm sorry, 107 is a code plenum, and then 106 is a mixing plenum. And the adjustments made in the models are in the flow coefficients in that region.

MEMBER BANERJEE: The thing that puzzles me there is really the 106.

MR. FLETCHER: The 106 is where the action occurs, that's for sure, and the adjustments made to the model are made to match the CFD calculations as far as the mixing process and the flow that is actually donated from 106 back into the tubes.

MEMBER BANERJEE: Yeah, but in a sense 106 is flowing in two directions of the same fluid.

MR. FLETCHER: Well, first of all it's all steam, and you're right, it has flows entering and exiting, and mixing - so 106 is appropriately named the mixing plenum; that's exactly what it is.

MEMBER BANERJEE: Right so how do you proportion one-dimensional model to a single fluid of momentum and mass. I mean this must be a strictly empirical mode that you take this, pull it out --

MR. FLETCHER: We are forcing the mixing to match the CFD calculations.

MEMBER BANERJEE: But it's arbitrary - it's purely empirical. You are just saying, this is going in, this is going out, this is based on CFD?

MR. FLETCHER: It's not arbitrary nor is

it empirical. It's adjusted to match the CFD and the Westinghouse --

MEMBER BANERJEE: Well, it's empirical in the sense that your code does nothing there.

MR. FLETCHER: That's correct. We are not trying to model this on the first principles basis with the two temperature steam paths.

MEMBER BANERJEE: It's just a note where you are prescribing what is going on, prescribing what is coming out.

MR. FLETCHER: No, we aren't prescribing what is going in.

MR. BOYD: I can take a crack at this. If you just remove 105 and 107 from the picture. Okay, now we have the flow coming from the hot leg, it mixes in the plenum, and goes up to the bundle. And what we find is, that would be 100 mixing. What we find is that the temperatures entering the bundle are slightly hotter than that, so they have a little bypass out around that big mixing volume, maybe 5 percent, 10 per cent of the flow bypasses around through 105, mixes back in, and now we match the temperatures going into the bundle.

So 105 gives us the ability to bypass a little bit of the hot flow around the mixing, and then we're applying a mixing fraction. If the mixing fraction is point nine, 10 percent of the flow is going to bypass the mixing. We mix it back together, with this approach we can get the hot leg mass flow, and we can get the tube bundle mass flows to have the appropriate temperature and mass flows from what we would see in experiments in computational fluids.

So this is a simplistic way in one-dimensional code to establish the appropriate mixing and mass flows.

MEMBER BANERJEE: So that node really tries to conserve mass and momentum along with other things, basically have an inlet flow at a certain temperature coming from the left-hand side.

MR. BOYD: It has two inlet flows.

MEMBER BANERJEE: And then it will have an inlet flow coming from the top. So it's just like a cross junction if you like.

MR. BOYD: Yes.

MEMBER BANERJEE: Then those flows somewhere are prescribed, because the inlet flow coming in, you're saying 10 percent of it will bypass through 105. Is that what you are really doing?

MR. FLETCHER: That's roughly what's going on. Now the momentum is not appropriate here, because we have a one-dimensional code, but we have a highly three-dimensional flow pattern in that inlet plenum. So the one-dimensional code has loss coefficients. If you don't do anything to it you guess, you will get flows up into the bundle but they won't quite be right. So what we do is, we say no. We want the bundle flow to be a certain ratio of the hot leg flow, or a certain mass flow. And we adjust those coefficients. And what we are really doing then is we are adjusting coefficients that will account - that ultimately are going to account for all the turbulent mixing and resistances that are occurring in the inner plenum.

MEMBER BANERJEE: The way you do it is you adjust the flow resistances until you more or less get the amount that you would like based on the CFD. That's what you are really doing?

MR. FLETCHER: That is correct.

MEMBER BANERJEE: The loss coefficients so that 10 percent of the flow goes to 105.

MR. FLETCHER: That's correct.

MEMBER BANERJEE: That's why I said it was frankly empirical procedure.

MR. BOYD: And also adjusting it so that the right amount of flow ends up entering the bundle, and the code again would do this on its own, but it wouldn't quite match what we've seen from experiments in the 3-D calculations, so there are adjustments made in those loss coefficients to account for some of the physics that's not in the code.

MEMBER BANERJEE: And that is also what you are doing at the hot leg, right? You fit it into two fluids, two paths.

MR. BOYD: That's correct.

MEMBER BANERJEE: With some interchange between the paths?

MR. FLETCHER: In SCDAP/RELAP 5 we don't interchange between the paths.

MEMBER BANERJEE: They are just completely without mixing?

MR. FLETCHER: From the viewpoint of RELAP 5 they are without mixing. We do have thermal radiation wall to wall from the upper half of the hot leg into the lower half of the hot leg, but other than that there is no mixing - no interaction between the upper and lower sections.

MR. BOYD: We do predict some mixing in our 3-D calculations between the upper and lower flows. Let's say we postpone that mixing until we get to these three boxes in the inner plenum, and we account for it there. So we do that mixing, we just do it all down there. So all of our mixing takes place down there.

MEMBER BANERJEE: Now why is 582 pointing to the left?

MR. FLETCHER: Okay, 581 and 582 are nodes in the upper plenum of the steam generators.

MEMBER BANERJEE: No, I mean --

MR. BOYD: That's a type arrow on Don's arrow.

MEMBER BANERJEE: Oh, okay.

MR. BOYD: It's an arrow error.

MEMBER BANERJEE: Oh, the arrow is in the wrong direction. I was just puzzled.

MR. FLETCHER: And I can see why now.

MEMBER BANERJEE: Okay, then it's all clear.

MR. FLETCHER: I'm glad we've succeeded.

MEMBER BANERJEE: At least I know what you are doing. I don't know if I agree with it, but I know what you are doing.

MR. FLETCHER: The next slide shows the SCDAP/RELAP 5 calculated base case event sequence, the timing of it. And I want to highlight that this assumes that plant systems fail immediately. This isn't considered a most likely accident scenario; it's just a convenient accident scenario for us to use from which to evaluate other operations, or other variations.

And again we are not modeling the hot leg surge line and two break flow paths directly in the model. This allows us to decide when the hot leg fails, and then subsequently look at what might happen to steam generator tubes and so forth.

At time zero the accident sequence assumes we have a loss of all AC power. We get reactor and turbine trips; loss of all feedwater; reactor coolant pump trips. We assume reactor coolant shaft seal leakage begins at a 21 gpm per pump rate. Not shown on the slide we are also assuming that there is a leak in each of the steam generators, with a flow area of a half a square inch.

At 5905 seconds, steam generator one becomes dry. During this period we're circulating through the coolant loops, we're passing the core heat to the steam generators, we are boiling off the secondary inventory.

MEMBER BANERJEE: What percentage for a small break is this? Like 1 percent or 2 percent or smaller?

MR. FLETCHER: It's tiny. I don't have the number handy here.

MEMBER STETKAR: Don, are you going to - is this an appropriate time, or should we wait to explore the issue of the sensitivity to that leakage rate?

MR. FLETCHER: I'm going to show it directly. It's coming.

At 9226 seconds the steam at the core exit begins to superheat. The hot leg of the steam generator circulation patterns begin that we've discussed. The system heats up, and at 10747 seconds we get an onset of fuel rod oxidation. That process continues to a peak, a peak oxidation that occurs at 13566 seconds. And the key failure is the hot leg one, and this is the hot leg in the pressurizer loop, fails at 13625 seconds. This assumes no degradation in the hot leg material themselves.

Approximately six months later at 13985 we have the hottest tube in the pressurizer loop steam generator with nondegraded tube strength fail. Shortly thereafter the pressurizer surge line fails, and then a key point here is that we do have some spare accident behavior. There is no coolant source provided for the core, so the temperatures are going to continue to heat on up. So we end up with molten fuel pools forming near the center of the hottest channel starting at 14241 seconds.

The point being that the severe accident behavior, the core damage progression behavior, comes after the time when the hot leg fails, and after the time when the tube ruptures, if they do occur, would happen.

MEMBER BANERJEE: I'm sure that Dana has asked this question already, but the difference between these numbers is quite small.

MR. FLETCHER: It's quite small.

MEMBER BANERJEE: So I mean within the uncertainties, you could almost say that everything fits together.

MR. FLETCHER: I've got a slide coming up that hopefully will help this.

MEMBER BANERJEE: But it's not that everything fits together.

MR. FLETCHER: It's that the heat up rate is so fast that everything is compressed.

MR. BOYD: If you subject let's say a very strong material and a very weak material to an extreme heat source, and the strong material is going to fail in one minute, the weak material will always fail before that. Now it's a very compressed time, but we could argue that the weak material fails first.

What we have - these times are close together, we mentioned it earlier. But these times, you have to look at the temperature flux, and the energy coming out of the vessel. The peak oxidation occurs, and all of a sudden we increase the power because of this exothermic reaction by 10 times over the decay heat power.

The temperatures are going up, then all of a sudden they go up very fast. And a few minutes, the entire rising temperature takes place on the order of 20 - 25 minutes, so for instance 10 minutes would be half of that total heat up, and it would be a very substantial difference in temperature.

So although some of these scenarios last for 18 hours when we have auxiliary feed, the action all still occurs in 20 - 25 minutes.

MEMBER BANERJEE: So you've saying that physical reasons for the sequence of events --

MR. BOYD: I'm arguing that six minutes is --

MEMBER BANERJEE: Is a lot of difference

in temperature.

MR. BOYD: -- a big deal when we're talking about the type of heat up rates that we are talking about.

MEMBER BANERJEE: All right.

MR. BOYD: A bigger deal than what it looks like when you see something that could last for a day.

MEMBER BANERJEE: I think what would be useful would be to put the temperatures down there.

MR. FLETCHER: It's coming. It's coming. The next few slides portray the behavior that we see in the primary and secondary system. The base case event leads to high RCS pressure, dry steam generators are low pressures. The slide on the left at the top shows the RCS pressure. It falls initially as we cool to the steam generators while they are wet; then the steam generators dry out, and as a result we've lost the heat sink. The primary system fluid heats up, pressurizes the primary, takes the primary pressure up to the POVR opening set point; that's at the 16 mpa point there.

MEMBER BANERJEE: My original issue was with that pressurization in the first 5000 seconds as to whether you could get core uncovery there or not.

MR. FLETCHER: You don't see that here.

MEMBER BANERJEE: Not in your calculation.

MR. FLETCHER: Not in these calculations. We have not evaluated the LOCA type situation that you discuss.

You can also see on here that were a few spikes up to open - the PORVs are insufficient to limit the pressure increase, and we have to go up and open the SRVs for a few cycles; that's what the spike above that is.

The key point is that we need to remember that we don't open the hot leg failure. We don't open the break. And therefore you don't see a depressurization in the primary system pressure as a result of that.

The slide on the bottom shows the steam generator pressures. There's four of them on here. We are modeling point five square inch leak flow pass from each of the four steam generators. And so the secondary pressure goes up to the secondary PORV relief valve pressure set point. That's what the flat point there out through the period when the steam generator remains wet.

And then once the steam generator dries

out, there is no longer any water to vaporize the steam. The leak becomes very important at that point, and the pressure falls on down.

The key thing to point out here is that for the half-inch square break, by the time you get out to where the hot leg break occurs, which is 13600, you completely depressurize the steam generators.

MEMBER STETKAR: Don, let me make sure I understand, something I asked earlier. If you open the steam generator PORV fully, early on, within the first couple of minutes, I think I understood you to say earlier that that really doesn't significantly affect the timing of the subsequent event progression. Is that right?

MR. FLETCHER: That is correct. We've looked at those cases early in this project. We used this point 25 square inch assumption quite a bit in the last six years or so. But what we saw on those early runs was that the steam generator that had the stuck open PORV on it, it depressurizes very rapidly, starting right from the beginning. If you dry out very quickly - I think the dry out time was like 1900 seconds for the one PORV, so it's roughly one-third of the time that you see the drying taking at the half-square inch.

And for that case you clearly ended up down at low pressures by the time you get to the hot leg failure time, and the results are pretty much the same, the only distinction being that if you got a leak in all four steam generators then you've got a potential to break tubes in all four steam generators, where in the old case we only had the stuck open PORV and one steam generator, and that was the one that had the failures. The others did not fail at all.

MEMBER STETKAR: Thanks.

MR. FLETCHER: The next slide shows the pressurizer level on the stop, and the core hydrogen generation rate on the bottom. The pressurizer level is falling at the beginning, as we are cooling the RCS by steam generator heat removal. Then we lose the heat sink. The RCS fluid begins to heat up, swells the heat up into the pressurizer. Actually fills the pressurizer with water for a brief period of time. And that period when the pressurizer is filled is when the PORVs on the pressurizer weren't sufficient to limit the pressure excursion, and we went up and opened the safety relief valves for a few cycles.

Subsequently we expel water and two phase out the pressurizer PORVs, and the remaining water in the pressurizer drains down into the hot leg. And you can see that by about 11000 seconds the pressurizer is completely empty.

So at this point at about 11000 seconds the entire RCS system above the elevation of the hot leg is completely filled with steam, and we are still losing mass at that point. The core hydrogen generation rate that results from the oxidation of the fuel rods is show on the bottom slide. And you can see that the oxidation rate starts slowly, and as the temperatures increase the rate continues to accelerate, and we end up with a peak that is roughly about 10 times the fission product decay heat.

As another rule of thumb, it's about 9 percent of normal operating power at that point. So we end up with a significant spike with power being added in the core from a combination of decay heat and oxidation. And the reason this is important is shown on the next slide. It's the rapid RCS steam temperature excursion that leads to the structural failures. The slide shows that metal temperatures in the surge line, the upper portion of the hot leg, the average steam generator tube, the hottest steam generator tube. You can see that prior to about 13500 seconds the temperatures are creeping up as the core power keeps adding - the core decay heat keeps adding power in the core, and the system temperatures are rising relatively slowly.

Then we get the oxidation peak, and it's that rapid increase in temperatures that you see there at 13500 seconds as a result of the oxidation peak that leads to all the failures we see.

The vertical lines, the color critical lines, show the failure times. The red line is the hot leg, and you can see where the hot leg failure is there. And following that is the average undegraded tube failure time. And the pressurizer surge line, and then average tube failure time.

The takeaway from this is that the creep rupture failure is affected by the material, by the structure, that the degradation that you assume in its strength, its thickness, the differential pressure across it and the local steam temperatures; and that hot steam flows from the reactor vessel outward into the RCS, so you first see the hot steam effect in the hot leg, because that's immediately closest to the reactor vessel.

The effect on the surge line is limited, because there is mixing involved, with the temperature of the steam that goes into the surge line, and also because you are opening the pressure as a relief valve on an occasional basis, not continuously.

And then in the tubes the effect is buffered by the time delay for the flow of the hot steam to go through the hot leg, and once it gets into the steam generator inlet plenum, it has to mix with the cooler steam coming back from the outlet plenum, and that mixing provides a benefit that keeps the temperatures of the steam going into the steam generator tubes lower.

MEMBER BANERJEE: But this must be fairly sensitive to the heat transfer coefficient you are using at the - between the tubes and the rest of the world, to the secondary side.

MR. FLETCHER: On the outside of the tubes. It is somewhat sensitive to it. And we performed a number of sensitivity analyses on that.

MR. BOYD: Sanjoy, I'll make a comment. It does turn out that that heat transfer coefficient plays a role when we do our CFD sensitivity studies. But in the end the net heat transfer out of the tubes generally we are finding goes into heating up the metal mass of the tubes themselves, so really it becomes a low CPDTD time.

MEMBER BANERJEE: Just the thermal inertia of the system.

MR. BOYD: The thermal inertia is very important, and that's well modeled. And it's a one-dimensional flow. So the inert wall heat transfer coefficient would be fairly - as well modeled as you can model heat transfer.

And then the secondary side is steam, and it's at low pressure. So it really doesn't have the heat capacity let's say to compete with the massive amount of steel in the bundle to carry heat away. So what we find is, it's not as - the heat transfer to the tubes is important, but we find that we can kind of narrowly bound it somewhat because of the fact that it is governed by that --

MEMBER BANERJEE: And there is very little radiation from the hot tubes to the cooler tubes and things like that.

MR. BOYD: The radiation, we model radiation from the steam to the tube wall. As far as radiation between the tubes, on the secondary side, we are not. The tube temperatures are just approaching about 1,000 Kelvin where radiation starts to get interesting whenever the ruptures occur. One might argue that with 3,000 tubes the view factor from tube to tube would be to tubes that are very similar temperatures.

MEMBER BANERJEE: Depends on which tubes are coming back, and which tubes are going forward.

MR. BOYD: But the hottest tubes that we are doing for our screening calculations are at the core of the upflow, and they would be surrounded by similarly - but I agree. there is some effect.

MEMBER BANERJEE: It's fine, I mean to be a bit conservative is good. But you want to fail them as quickly as you can, right?

MR. BOYD: That's right. If we did allow them to radiate some heat away that would slow it down a little bit.

MEMBER ARMIJO: This graph is a little bit hard to read. What fails first, the surge?

MR. FLETCHER: The hot leg fails first. That is the red curve.

MEMBER ARMIJO: Yeah, I can see that.

MR. FLETCHER: And you can see the vertical red line on that, I hope.

MEMBER ARMIJO: Yeah, my question is, is it the - the colors are hard to read. Does the surge line fail before the hot - the average tube?

MR. FLETCHER: The sequence of events is given back on slide seven. And so the hot leg fails in the hottest tube. And the average tube is like 10 minutes later.

The one on the right side is the average hot tube.

MEMBER ARMIJO: Got it, I understand, thank you.

MR. FLETCHER: The next slide shows in tabular form what you saw --

MEMBER BANERJEE: In your model, how are you - you don't have the hottest tube, right?

MR. FLETCHER: Our model, we're handling the hottest tube as a single tube.

MEMBER BANERJEE: But the way you set it up in the nodalization diagram --

MR. FLETCHER: It doesn't show up in the diagram. The hottest tube doesn't show up there. We are doing that as a side calculation.

MEMBER BANERJEE: How is that done?

MR. FLETCHER: We are doing that with a heat structure that is connected to the secondary side on its outlet, on the outside of the tube. We are feeding the hottest temperature into that tube at --

MEMBER BANERJEE: Hottest means unmixed, 105.

MR. FLETCHER: No, it's not unmixed. It's slightly mixed. This will become a lot clearer whenever we look at the CFD predictions. But we predict what the hottest temperature will be from the CFD predictions, and we feed that temperature into that tube. It's not the 105 temperature. That's the hot leg temperature. But it's also not the mix temperature either.

MEMBER BANERJEE: What would happen if you vent 105? I mean I can see from the safety calculations you always get some mixing. But 105 is really unmixed.

MR. BOYD: 105 is the unmixed temperature. Yes, that would really challenge the tubes. And that's the case of the full loop circulation where you have the hot leg temperatures going right into the inner plenum.

MEMBER BANERJEE: And that would - well, it's just like a full loop circulation. So that is assuming that there is some mechanism where some part of the stuff coming from the top of the hot leg doesn't mix. No matter what the CFD calculations show.

MR. FLETCHER: If we do it that way, it makes that assumption; that is correct.

MEMBER BANERJEE: Then what would happen to that failure? Would it be before the hot leg or after the hot leg? Or have you done that?

MR. FLETCHER: We have not done that.

MR. BOYD: That would be before the hot leg.

MEMBER BANERJEE: It would be before the hot leg.

MR. BOYD: We could be pretty confident without doing it that that would be before the hot leg.

MEMBER BANERJEE: It would be like the left hand scenario.

MR. BOYD: That's right.

MR. FLETCHER: The next slide, slide #11, shows in tabular form what was shown in the previous plot. Hot leg one, the one that is connected to the pressurizer, fails first at 13625 seconds, and that's shown in red because that is the first failure.

The other hot legs fail slightly later because they don't see the effect of the flow going out the pressurizer relief valve pulling extra steam into that hot leg the way it is in hot leg one.

MEMBER BANERJEE: Excuse me, let me just go back to my train of thought. Your hottest tube right now is based on basically the CFD calculations that you've done at full scale.

MR. BOYD: That's correct.

MEMBER BANERJEE: For different - I guess you did one for the Westinghouse system and one for BNW or something.

MR. BOYD: We looked at that for some different inlet plenum geometries, right, the GE plants and the Westinghouse.

MEMBER BANERJEE: But is that roughly the effect of the different inlet geometries is not all that significant?

MR. BOYD: It can be. There are some inlet plenums that are very flat, and not - and maybe only 1-1/2 pipe diameters, hot leg diameters, deep. So that becomes an inlet plenum that doesn't have a lot of mixing region.

MEMBER BANERJEE: So the one that you are showing here is based on what type of inlet plenum?

MR. BOYD: We are looking at the

Westinghouse type inlet plenums for the - very similar to - I think we used a Model 51 generator. The inlet plenum is very similar to a Model 44 generator. We based this on the Zion plant.

And we looked at a variety of Westinghouse steam generators, and the inlet plenums have a similar design.

MEMBER BANERJEE: So for a flatter inlet plenum, would you have a different curve for the hottest tube?

MR. BOYD: Yes.

MEMBER BANERJEE: It would fail faster?

MR. BOYD: That is correct.

MEMBER BANERJEE: Fail before the hot leg, any of them?

MR. BOYD: We don't have results today for the CE plant. We are not presenting that. But there can be a much more significant challenge if you don't have that in the plenum mixing.

MEMBER BANERJEE: So the current results are for Westinghouse?

MR. BOYD: That's correct.

MR. FLETCHER: The reason the hot leg failure is shown in red is so that if you go to the bottom part of the diagram there, where we are showing tubes with various stress multipliers on them, when they fail, compared to the hot leg failure time. So the red numbers on the bottom bracket for the average tube and the hottest tube, the time of hot leg failure.

And what it shows is that for the average tube in generator one, and by this I mean the tube that is carrying the average temperature steam from the inlet plenum upward into the tubes, it takes a stress multiplier of 2.74 to fail the average tube coincident with the hot leg. And in the case of the hottest leg it takes a stress multiplier of 1.68 to fail coincident with the hot leg.

MEMBER BANERJEE: I'm not familiar, not being a stress analyst, with the stress multiplier.

MR. BOYD: And you are asking the thermal hydraulic guys this question?

MEMBER BANERJEE: Well, you guys have introduced this. I didn't.

MR. BOYD: From our perspective it's very simple. I mean if we have a stress multiplier of two, we double the pressure, or double the stress, double the pressure on the tube.

MEMBER BANERJEE: What is the physical significance?

MR. BOYD: The physical significance is that there are flaws, potential flaws, in the tube. And you can take a certain flaw, and you can relate it to an equivalent stress multiplier.

MEMBER BLEY: Earlier you related a two-inch multiplier to a flaw of certain dimensions. Would you repeat that again?

MR. BOYD: I called Saurin Majumbdar, so I'll blame this on Saurin out at Argonne. But he sent me a graph. If you have for instance a half an inch flaw three-quarters of the way through the thicknesses, he told me the ligament stresses were double what the stresses were on a pristine tube without a flaw. And he said that would fail in a similar way to you taking a brand new tube and doubling the stress on it.

So we just work with a simple multiple of stress. We'll let the experts figure out what's in the generator, and what stress multiplier that would correspond to.

Now, my understanding is, most tubes with their little tiny surface flaws, they have stress multiplier equivalents of like 1.02, or something like that, in the generator. So when we look at stress multipliers of two we're really pushing things. And our screening criteria of a stress multiplier of three, this become a tube that is significantly flawed.

MEMBER BLEY: Ready to fail on normal operations.

MEMBER MAYNARD: Isn't a simple way to look at it is, if you are using a stress multiplier of two, you're really saying you are going to fail that tube at half the stress that a good tube --

MR. BOYD: That is another way to look at it. Or you could say the tube wall thickness is half of what the tube wall thickness would normally be.

MEMBER BANERJEE: What is the stress multiplier at which it would fail under normal operations. Is it two or three?

MEMBER ARMIJO: It depends on the stress.

MEMBER BANERJEE: There is some amount of flaw or stress multiplier where the tube would just fail if you were operating it normally, right? What multiplier would that be? Would it be three? Or two?

MEMBER SHACK: More like four, probably.

MR. BOYD: It's kind of like asking what

the safety factor is on the tubes.

MEMBER SHACK: That would be equivalent to a multiplier of about four.

CHAIR POWERS: So what is - in the inspection of the tubes - is to look for a voltage signal corresponding something on the order of 40 percent through wall. Now there is some probability of course that you could have a flaw that is deeper than that that you would just not detect because of pigments or geometry or something.

MEMBER BANERJEE: Right, I remember your report on this.

CHAIR POWERS: And so that's why we worry about it. That's why they don't take one as the stress multiplier, because there are going to be flawed tubes in there and a variety of other things.

MEMBER BANERJEE: Well, if you looked at say at your 40 percent in the matter of detection, or whatever --

CHAIR POWERS: No, they can detect down to about 10 percent. Where detection capabilities fall apart is in the incipient formation of the cracking, because there is nothing really to detect there.

MEMBER BANERJEE: What would be a

realistic stress multiplier to sort of assume is --

CHAIR POWERS: I think they are right in the range. I don't think there is anything objectionable to taking a screening criteria of three. I really don't have any troubles with that.

MEMBER ARMIJO: I'll tell you, I think 3.5 is a pretty crummy tube.

MEMBER SHACK: Under the voltage requirement these cracks are either - they could be shallow, longer. Most of them probably are very short, and they would have fairly small stress multipliers as Chris said. If you have a stress multiplier of two, that is a big crack to be left in the steam generator.

CHAIR POWERS: I mean there is no question.

MEMBER SHACK: Some probability of it, yes.

MEMBER BLEY: All of this helps us get a feeling for the uncertainty. And when you talk about the average tubes, and you've shown us numbers on that, he said that is the tube with the average temperature of the inlet plenum, right? To help me think a little bit more about uncertainty, that one doesn't help me much, but to have some idea of how many tubes see the higher temperature. The hottest tube isn't one tube. There are several tubes.

MR. BOYD: Yes, we try and predict that, and we will show that in a little while if you want to hold off on that question.

MEMBER BLEY: Okay, I'll be happy to wait.

CHAIR POWERS: That is a very crucial aspect of the FD calculations. And similarly it was a crucial aspect of looking at source trend issues associated with these things.

MR. BOYD: We're going to be looking at the PRA later. But the PRA is not the thing. The thing is all these things we are talking about right now, and the uncertainties that end up over there.

CHAIR POWERS: The PRA people will readily admit, a strong phenomenological analysis is the first step.

MEMBER BLEY: I don't know what you are doing. It's just a way to write a report and put all that information in one place.

CHAIR POWERS: That is not an affliction that affects some fields however.

MEMBER BANERJEE: Yes, there is an uncertainty associated with of course the distributions, how well they are predicted, but also with the fluctuation field, because what happens in the mixing vat things is that you have large eddies, and the large eddies can be very persistent. So you could get a bunch of hot fluid going up for 30 seconds or 40 seconds or something, depending on the eddy turnover time. So you have to also stop to factor those things in. We'll discuss this in some detail when we go into the CFD.

MEMBER BLEY: That will be very interesting.

CHAIR POWERS: But Sanjoy, suppose you had a large --

MEMBER BANERJEE: It may not matter.

CHAIR POWERS: Remember, you've got a very thick tube sheet, and a very reasonably thick wall here, and a reasonably low CP for steam.

MEMBER BANERJEE: It may not matter, because the thermal inertia of the system. But all I'm saying is, if you take this into account and drive it up and down, you may find there is no effect, or there may be an effect. It's a comment.

MEMBER ARMIJO: What fraction of the tubes are in the category of hottest tubes? I would imagine that there would be many, many, many that are within just a few degrees difference.

MR. FLETCHER: Again this is - I want to concentrate on the word, average. The hot average tubes are 41 percent of the total; 59 percent would be then cold returning tubes.

MEMBER ARMIJO: So maybe 40 percent of your hottest average tubes that have a stress multiplier of one, that is, perfect tubes, fail in 1395, and your hot leg fails maybe two or three hundred seconds earlier.

MR. FLETCHER: That's correct.

MEMBER ARMIJO: If everything works the way you say. That is not much difference.

MR. BOYD: You said 40 percent of our tubes. You are looking at the hottest tube. That's on the order of maybe half a percent of the tubes are at that temperature. We have a live spread. And we bend the tubes and put them in a histogram, and we are talking less than a percent of the tubes. It's still a large number of tubes. One percent would be about 30 tubes.

MEMBER BANERJEE: I think a lot of these questions get answered with the CFD.

MEMBER STETKAR: Are we going to see that histogram later? You said you bend the tubes, and you have a histogram?

MR. BOYD: I do show that.

MEMBER STETKAR: Later? Okay.

MR. BOYD: It ends at 11:30, my talk.

CHAIR POWERS: I want to disabuse you of that notion.

MEMBER BANERJEE: You are just lucky that Graham Wallis isn't here. Otherwise it'd go on for three days.

CHAIR POWERS: My only constraints are that the presentations on the 25th will begin no earlier than midnight tonight.

MEMBER BANERJEE: Okay. I think it's clear.

MR. FLETCHER: Okay, good.

The next slide shows the parameters that we selected to vary for the purpose of categorizing event outcomes into containment bypass, no containment bypass, or having a potential for containment bypass. The words at the bottom say that if we have a non-degraded hottest tube that is predicted to fail prior to the hot leg, we consider that a definite containment bypass outcome. If we have a hottest tube failure margin of a 3.0 stress multiplier or higher, we consider that a negative outcome for containment bypass. And in the gray areas between one and three stress multipliers.

So we are screening event sequences based on a 3.0 stress multiplier hottest tube consideration. And at the top we list the behavior that we looked at varying, to try to map out which event sequences might lead to containment bypass, which ones do not and so forth.

We looked at RCP shaft seal leakage. Again the base case assumption is 21 gpm per pump. We looked at increases at 13 minutes into the event, and at the time when the pump fluid reaches saturation, which is somewhat short of two hours into the event. I can't speak directly to the single failures that lead to that, but there are seals that can fail at about 13 minutes, and at about the time of saturation, and that's the reason we selected that criteria.

We looked at variations in turbine driven auxiliary feedwater operations. The base case assumes there is none. However a more likely case might be that turbine driven aux feed runs for awhile and then maybe fails at some later point.

We looked at variations in the steam generator secondary system leakage flow area. The base case uses a half a square inch for each steam generator. We looked at mitigative operator action, including a pre-core damage and a post-core damage strategy, and I will discuss those in some detail. And we also looked at the effects of actually opening the steam generator tube rupture and hot leg flow paths.

Again, we are not typically modeling those paths, and the sensitivity evaluation indicates that if we open a tube rupture flow path early in the event for example, a high stress multiplier tube might fail before the hot leg, it doesn't significantly effect the results that follow. The hot leg still fails, and the outcome is pretty much the same.

I think on the --

MEMBER BANERJEE: But when you say the tube rupture, is that just a single tube or several tubes? At what effect does it start affecting --

MR. FLETCHER: Well, we looked at rupturing two tubes and eight tubes, and that was -- neither of those was significant enough to really affect the outcome.

CHAIR POWERS: Something that has not been reviewed in detail here is the staff has expended some substantial effort looking at the potential for progression in tube rupture. In particular, does the jetting of the gas cause failure of adjacent tubes with and without particulate in the flow.

MEMBER BANERJEE: This was the DPO?

CHAIR POWERS: Well, it was one of the issues that was addressed very early in the action plan. So you can't just assume one. But the mechanism from getting from one to many is another issue.

MR. BOYD: So when you think about failing eight tubes, we're thinking of maybe 30 of the tubes being in that hottest bin, so about a third of those tubes, almost a third, we are assuming, have a stress multiplier of two equivalent flaw in them and they buckle, which would - I don't know what the probability of that is, but it's a fairly drastic assumption.

MEMBER BANERJEE: When do you assume that rupture to occur, early in the transient?

MR. FLETCHER: We took the time when the 2.0 multiplier tube fails in the calculation, and we open the tube rupture at that time.

MEMBER BANERJEE: Okay.

MR. FLETCHER: And then we observe the subsequent hot leg failure --

MEMBER BANERJEE: That would be, if I looked at that little table, that would be typically 13500 seconds or so?

MR. FLETCHER: 13360.

MEMBER BANERJEE: Yeah, and the hot leg failing at 13625?

MR. FLETCHER: 13625. And what we found is that if we actually opened the path, the tube path, at 13350 --

MEMBER BANERJEE: Nothing happens.

MR. FLETCHER: The hot leg failure moved up by five seconds is all.

MEMBER BANERJEE: Okay, that's more or less to be expected.

MR. FLETCHER: That's correct.

MEMBER ARMIJO: But the difference in time is, are you saying, compressed? You have a potential for 30 perfect tubes failing six minutes before the first hot leg failure at this time. It's not much time. You took 1 percent of the hottest tubes, so that is 30 perfect tubes. And they fail at 13985, and that's a stress multiplier of one. And then six minutes later at 13625 - no, earlier - so that means that's a really short time between the - so that hot leg really has to fail early.

CHAIR POWERS: No, I don't think - I think the salient point, at least the take home point I wrote down here is, if the tube fails first, the hot leg fails shortly thereafter, and that relieves promptly the driving force for bypass release. That is - that I see as the significant contribution here, because I don't think that has entered into the thinking about induced bypass accidents. The thinking had always been that once you got into the bypass you were on the road to doom. But quite frankly six minutes of release is not very much release. Half an hour of release is not very much release.

MEMBER STETKAR: Let's see if I can understand, though. They said that up to eight didn't affect the hot leg failure. But if I understand Sam's question correctly, would 30 ruptured tubes affect the likelihood of hot leg rupture, a hot leg failure?

MEMBER ARMIJO: No, they happen after.

MR. BOYD: We did not try all the combinations.

MEMBER ARMIJO: I was not just noticing

that you fail a lot of tubes.

MR. BOYD: Clearly we could push it to the limit where they will have to bundle and save the - I guess you call it saving the hot leg in that case. And we did not --

CHAIR POWERS: It takes a peculiar change of mind to see hot leg failure as being a good thing.

MEMBER BLEY: We see that in some of the uses of the word, conservative.

CHAIR POWERS: Yes, I understand.

MR. BOYD: We thought given that with the assumption of the stress multiplier and the way we did it.

MEMBER STETKAR: Yes, I see what you did, thanks.

MR. FLETCHER: The next couple of slides show examples of containment bypass outcome maps. The graph NUREG has many of these maps in it. The two I picked out were for no operator intervention, and variations in the steam generator secondary steam leakage rate, or leakage flow area I should say. And then the RCP shaft seal leakage rate that increases at 13 minutes.

What the slide shows is that if you have a steam generator leakage flow area of .1 square inches or smaller, you don't get sufficient depressurization of the steam generator secondary side to lead to tube failure, and so you lose the low steam generator pressure condition from high-dry-low and you don't have containment bypass. That's the bottom half of the vugraph.

MEMBER BLEY: Can you do that again? I didn't quite follow that.

MEMBER STETKAR: Let me see if I can get something first before you get into this. You said, assume no operator intervention. What exact operator intervention are we talking about here that we are assuming doesn't happen?

MR. FLETCHER: The operator intervention that we have looked at, there is a pre-core damage and a post-core damage, and I'm going to talk about that in a minute. But the pre-core damage is, slow down the steam generators, use turbine driven auxiliary feedwater at 30 minutes into the event.

MEMBER STETKAR: Okay.

MR. FLETCHER: The post-core damage says, the operator sees 1,200 degrees Fahrenheit at the core exit thermocouples and decide they've got to do something. And they either take action at that time, or they convene a committee and 12 minutes later take action.

MEMBER STETKAR: But this is all action on the secondary side. It's to depressurize the secondary side and feed?

MR. FLETCHER: The pre-core damage is secondary side.

MEMBER STETKAR: Got it.

MR. FLETCHER: The post-core damage is opening pressurizer PORVs to depressurize the primary. Completely different.

MEMBER STETKAR: Okay, thanks. Just wanted to make sure. Thank you.

MR. FLETCHER: Okay, the other question? I'm sorry.

MEMBER BLEY: If you would just walk through that. Point one square inches, that is much less than even one steam generator tube rupture, much less?

MR. FLETCHER: Tube flow area is roughly half a square inch or something like that. About three-quarter inch to about a half inch, half square inch of flow area roughly.

So if the leakage flow area from the secondary side of each steam generator is less than

.1, then we don't get sufficient depressurization to fail tubes. And so we've pretty much screened out everything below .1 square inches on the bottom as leading to containment bypass.

However if you have a flow area that's larger than that for a steam generator leakage path, now what becomes important is the assumption that you make for RPC shaft seal leakage, the reason being that the larger the shaft seal leak, the lower the primary pressure gets. And what the results show is that if the leakage rate is greater than 180 gpm per pump, and that higher leakage rate starts at 13 minutes --

MEMBER BLEY: I always get confused when we use gallons per minute. So this is at system pressure?

MR. FLETCHER: This is at - let me back up clarify what we assumed.

MEMBER STETKAR: And that is 180 gpm per pump.

MR. FLETCHER: Per pump.

MEMBER STETKAR: Meaning a total of 720.

MR. FLETCHER: That is correct.

MEMBER STETKAR: Total of 720 gpm. Pretty reasonable sized break.

MEMBER BLEY: In more than one tube.

MEMBER STETKAR: No, no, I'm thinking primary side break, in terms of what we define typically as a small LOCA, or a medium LOCA or a very small LOCA. This is well into the small LOCA size break range, well into it. Possibly even close to a medium LOCA.

MR. FLETCHER: You are correct, the gpm equivalent that we are citing here is at the initial point, initial operation. You open a hole that gives you the 180 gpm from each pump at that time, and we use that - we maintain that area, that flow area throughout the transient calculation.

MEMBER BANERJEE: What do you mean by increases at 13 minutes?

MR. FLETCHER: The way the pumps can fail, the way I understand it is, there are several seals in the pump, and the failure rates have a - a set of failures occurs at about 13 minutes into the event is what they found. If you lose pump power, the pumps stop operating, and you still have primary system pressure, you get this 21 gpm per pump is what we call normal leakage without any failures whatsoever. But at 13 minutes into the event, if you maintain that condition for 13 minutes, then a seal will fail at that point.

MEMBER ARMIJO: But is this from temperature increases or just pressure or what?

MR. FLETCHER: You are well beyond our capabilities at this point.

MR. BOYD: The pump seal LOCA issue is a complex thing that we didn't study. What we did is, we found some interactions between the agency and the industry, and we found some event trees that listed the different pump seal LOCA failure modes, and the gpm equivalents, and we used that, and we made sure we ran through all that spectrum of leakage rates.

MR. FLETCHER: What they told us is that there are failures that are clustered at 13 minutes. Other failures that are clustered at the time after the system heats up at about two hours into the event.

MR. BOYD: And we are not ready to defend what we found, but that's what we used.

MEMBER STETKAR: I am assuming you took the expert elicitation results from the 1150 analyses as the input for this? Or something --

MR. BOYD: We found several things, and we found something that was as late as about 2002, which was the last thing we used, and it was refined a little bit from some of the earlier work, not much.

CHAIR POWERS: There has been quite a lot of work on that since 1150. Not only has there been work, there's been refinements of the seals.

MR. BOYD: If I can make a thermal hydraulic guy's comment, we really aren't that interested in the specific uncertainties that are on those event trees. And the timing, the point of saturation and the leakage rates, 180 gpm, 60 gpm, that didn't change a whole lot. We ran all of them, and we will - when the dust settles someone is going to have to verify those fractions on that event tree, and we did not do that.

MEMBER STETKAR: Sure, the key to your analysis, though, is anything less than 180 gpm per pump is - says you are susceptible to a containment bypass. The pressure is high enough to be high.

MR. BOYD: And we ran all the plausible leakage rates that we could find.

MR. FLETCHER: The slide after that shows the same situation, steam generator, secondary steam leakage, but with the shaft seal leakage increasing at the time the RCP fluid reaches saturation, which is somewhat less than two hours. The results are the same except for the right hand side of this, which shows that above 400 gpm per pump, that's 1,600 gpm total leakage from the RCPs, that we ended up in a containment bypass situation. And the reason for this is that the loop seal water heats up between 13 minutes and two hours, as a result of interaction with the hot steam. The steam gets keeping hotter and hotter during this period, and you warm that water up. It doesn't actually clear the loop seal except in a situation where you have a high enough depressurization rate that is greater than 400 gpm you depressurize, and you flash the loop seal water out because it's been warmed.

So there is a distinction between the 13-minute curve and the two hour curve or the saturation curve when you get to very high RCP leak rates. And the distinction has to do with flashing out loop seals as a result of the depressurization when the water is hotter.

The next several word slides summarize the findings of the analysis. First where the operators are assumed to take no action, for even sequences that assume very small leakage rates, less than .1 square inch per steam generator, they generally don't result in containment bypass, because the in generator pressure does not get low enough to have that occur.

Where even sequences where the RPC shaft seal leakage rate is below 180 gpm, you get the potential for containment bypass, a stress multiplier between one and three.

Where the shaft seal leakage rates are above 180 gpm, generally don't have containment bypass except for the situation I just mentioned where the seal failure occurs late and the water is hot.

Slide #16, for even sequences where turbine driven operates and continues to operate, there is no dry out of the steam generator, secondary side, and there is really no event - there is really no severe accident in this case. And so those did not result in containment bypass.

And then for sequences where a turbine driven system operates initially and then subsequently fails we end up with steam generator tube failure margins that are very comparable to what you have if you don't have turbine driven aux feed operating at all. It does buy you some time, it does buy the operator some time, though, as a result of the turbine-driven aux feed system operating for awhile.

And then when you get to the point where the system stops operating, then you are at a lower decay heat so the event proceeds a little slower.

The next two slides talk about operator intervention --

MEMBER BLEY: On that last one you had up there, that must assume a fair amount of time before they recover that turbine driven pump, is that right? If they get it back in five minutes or something like that --

MR. FLETCHER: No problem.

MEMBER BLEY: Which is a kind of failure that often happens with those pumps.

MR. FLETCHER: That's correct. If you have aux feed you don't have a problem, as long as you can keep it running.

MEMBER BLEY: As long as you get it within that, what, 100 minutes or something like that?

MR. FLETCHER: Right.

MEMBER SIEBER: Well, but the issue - it doesn't happen that fast, because usually the failure trips the pump, and someone has to manually relatch it.

MEMBER BLEY: It doesn't take 100

minutes to do that.

MEMBER SIEBER: No, but you have to know that it's happened.

MEMBER BLEY: That first assumption.

CHAIR POWERS: It all depends on why the turbine driven feed pump failed.

MEMBER BLEY: Absolutely.

MR. FLETCHER: For the pre-core damage strategy, this is essentially a steam generator feed and bleed cooling at 30 minutes into the event, the assumption being that the aux feed system is available, and the operator has opened the steam generator PORVs. The assumptions we made is they depressurized the steam generators to about 270 psi, and then closed the PORVs, and then modulated them to keep the pressures down to around 270 psi.

The results show that this strategy is effective in the short term for preventing containment bypass. At a minimum it buys some time. It delays the onset of RCS heat up. In the long run the steam generator PORVs are going to fail closed when the station batteries are depleted, and there's nothing the operators can do about keeping the secondary depressurized at that time.

But continued success therefore requires that the aux feed system continue to remain available, and some means of getting it into the steam generator is effective.

Finally, the post-core damage strategy, the operators are assumed to depressurize the RCS by opening either one or two pressurizer PORVs, when the core exit temperature reaches 1,200 Fahrenheit, or 12 minutes after that point. What we found is, opening one PORV limits the RCS cooling rate that you see. You don't get quite as much accumulator injection. The core fails early; so does the hot leg, and the containment bypass is avoided for either time, either the 1,200 degree Fahrenheit time or 12 minutes afterwards.

If you open two pressurizer PORVs it prevents the early core failure, and also prevents the early hot leg and steam generator tube structure failures, but then eventually you have to deal with a slower heat up of the RCS, as the lower decay heat heats the RCS, so you lose fluid at that point.

What we found is the tube failure margins are significantly improved in those situations over where the operator takes no action at all, and containment bypass is avoided from either of the action times that we looked at.

MEMBER BLEY: As you say there is a 12 - minute time interval for doing that?

MR. FLETCHER: There is a 12-minute time interval.

MEMBER BLEY: What happens if they do it earlier?

MR. FLETCHER: Earlier is better, earlier than the 1,200 Fahrenheit. I don't know that we have looked at that in detail. The presumption is, that makes things a lot better if they do that. I can't think of a reason why it wouldn't, but again, we haven't analyzed it in detail.

In summary, the previous ACRS review comments on the SCDAP/RELAP 5 analysis have been considered in the current analysis. This includes a need for improved thermal radiation modeling, which was a comment that the ACRS made a long time ago, and EPRI reiterated in a more recent review.

The improved method for determining the hot leg circulation rate was incorporated into the model. We have done some evaluations of loop seal clearing, sensitivity to reactor vessel internal circulation rates, which was something the ACRS requested. We did an analysis of the system energy flows, and performed some independent peer reviews of the methods and the results. And the steam generator action plan items addressed by the draft NUREG/CR listed at the bottom there. The plant sequent variations, reevaluation what we are doing with SCDAP/RELAP 5 and the wisdom of the way we are doing it. The need for more rigorous uncertainty analysis, and admittedly the analysis is a sensitivity calculation based one, which is not a thorough, but one could argue it's more rigorous than what's been done in the past. And estimate the uncertainty due to the core melt progression.

That concludes my talk. Further questions?

CHAIR POWERS: Do people have other questions to ask?

The central issue that the committee of course is addressing is, have we done enough to - I mean there are always going to be technical issues, and there are going to be technical issues associated with accident analyses forever, I'm sure. The question is, have we done enough to satisfy the needs for this steam generator integrity action plan. And I reiterate that question not because I expect an answer now but just to make clear what we are asking.

You have summarized us before the

subtasks. I think it will become a little clearer when we discuss the CFD portion of it. But I would propose doing the discussion of the CFD proposed analysis after lunch. Why don't we plan on reassembling at 1:00 o'clock, and we will CFD, k-epsilon and k-omegas and all kinds of interesting - who knows what all turbulence models.

(Whereupon, the above-entitled matter went off the record at 11:55 a.m. and resumed at 1:09 p.m.)

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

(1:09 p.m.)

CHAIR POWERS: We will come back into session. And I think we move to the wonderful world of computational fluid dynamics. This is an area of exceptionally low controversy. So I think this will go quite quickly, I'm sure.

MR. BOYD: Thank you.

6. SGAP ITEMS 3.4.E-G

COMPUTATIONAL FLUID DYNAMICS

MR. BOYD: So, again, I am Chris Boyd from the Office of Research. We are going to talk about the computational fluid dynamics that was done to look at the experiments and help to extend those into various other full-scale conditions so we could use those to refine our SCDAP/RELAP5 model.

The CFD builds upon the experimental results, and it provides additional insights. Even from the experiments themselves, after modeling them with CFD because of the limited instrumentation, we were able to fill in a lot of gaps and answer a lot of questions on what was going on behind some of those measurements.

We use this to adjust flow loss coefficients in our 1D model to ensure that the flows are consistent with the experimental observations and our 3D CFD predictions of the natural circulation flows.

Here is the system code regions that are of interest. The hot leg flow itself is one of the key natural circulation flows. And we use our results to help ensure that flow is right: the surge line flows, not necessarily the flows. That is really a repressurization. But the mixing of that flow in the hot leg, what temperature enters the surge line, is something we adjust.

The tube bundle flows and the amount of tubes and the temperature that enters the tube bundle are another feature that we would adjust with the CFD for the experiments.

So at this point, what we have done in adjusting these flow coefficients is we have this leap in a natural circulation flow pattern where the mass flows and the heat transfer are appropriate based on our best knowledge from 3D simulations.

Now, that gets us to system code response. One key feature that is very important, then, is on top of that, we predict the hottest tube. That hottest tube is very important, as we have discussed. So that is another aspect where, instead of having a single temperature going into the tubes, the computational fluids give us a range of temperatures. So that hottest tube calculation becomes very important.

Thermal hydraulic issues. These are the same issues that I had listed earlier about the test scaling issues on concern about mixing being overestimated and a lack of leakage and the sensitivity studies.

In addition to those initial concerns that ended up in the action plan, after we finished the work in the action plan and presented it to the ACRS, they had an additional concern. They wanted us to use our three-dimensional tools to model also the hot leg flows themselves.

And so those are the concerns that we have going forward. The action plan tasks deal with benchmarking our tools and then extending them to full-scale conditions. And we have NUREG-1781 and NUREG-1788 to document the basic action plan tasks.

This is basically the material that was presented to the ACRS in 2004. And, again, the ACRS suggestion at that time was to extend our model. We did that. And we have a NUREG that is drafted and ready to be published, 1922. And that is going to document the work we did to assess that.

We also did a lot of other work in that NUREG. And we took the opportunity to reassess all of our assumptions and limitations from the previous work and how we put the data into the system code. And I think we made maybe an order of magnitude more improvements than just predicting the hot leg flow when we did that final NUREG.

So we'll start out. This is kind of a quick run through the work that was done: one of the tasks at benchmark at one-seventh scale. I don't show the hot leg flows, but one of the things we did was we demonstrated that we could predict the hot leg flow patterns very well, the distribution of the temperatures, the profiles.

So we believe we had the hot leg flow well-documented. Then there was a question of the inlet plenum mixing, which was our focus at this point.

We had some temperatures at the end of the hot leg. And we had some temperatures in the tubes. And we were able to demonstrate that we could after getting the hot leg flow properly modeled then predict the mixing in the inlet plenum and get the temperatures into the tubes properly predicted.

In the red box there, we show that we were able to predict the temperatures to within a degree. And, surprisingly, we were actually able to predict the two bundle mass flows right on, which was very good for our benchmarking work.

I'll show a quick animation up in the

right corner. This was a symmetric design. And you can see the flows coming in. This is a flow path colored by temperature. You will see the red flow paths at the top of the hot leg, and they quickly turn to yellow and lower before they impact the tube sheet.

MEMBER ARMIJO: Roughly what kind of temperature difference do we see red to yellow?

MR. BOYD: Roughly we're seeing --

MEMBER BANERJEE: Just a minute. Microphone. Say it again.

MR. BOYD: In this particular test, we are seeing temperatures in the hot leg of about 155 degrees. This is a scaled facility with sulphur hexafluoride, about 155 degrees. And the temperature going into the tubes is on the order of 100 degrees. So it's about a 60-degree temperature drop through the inlet plenum.

Okay. So the next step, we were asked in the action plan to evaluate the impact of scale. So our goal would be to see, you know, how can we scale these test data up full scale.

We realized when we started out on this path that the geometry of an actual prototypical Westinghouse steam generator was a little bit different than the geometry of the facility. And we realize we're going to get different answers based on scale and based on geometry.

So we took an intermediate step. And we scaled up our model just by multiplying the geometry by seven. And then we replaced the sulphur hexafluoride with steam at the appropriate severe accident conditions and basically ran full-scale conditions in a geometry that is consistent with the facility.

What we found is that generally the results are about the same. We saw the same general mixing, maybe slightly more mixing, in the full-scale facility. But the flow pattern in the tubes, the mass flows were all consistent.

MEMBER BANERJEE: Now, Chris, the previous slide you showed us, even though it's an animation, it's, I assume, a RANS calculation you did.

MR. BOYD: That is correct.

MEMBER BANERJEE: Basically it's just a steady flow pattern, then?

MR. BOYD: In that symmetric facility, we show a steady flow pattern. That is right.

MEMBER BANERJEE: Well, you are not doing any transient or if you did, that is not the transient calculation.

MR. BOYD: In this particular case, we did run that as a transient. And it looked like that, too. In the full-scale facility, what we found is that there is unsteadiness in the rising plume. And we are going to talk about that.

MEMBER BANERJEE: Okay. So it's like unsteady RANS still. It's not an LES?

MR. BOYD: That is correct.

MEMBER BANERJEE: Yes. Okay. It's not a large eddy simulation. A RANS is a Reynolds-averaged numerical simulation. So basically it's averaging over all length scales and time scales. And the turbulence model which is used, then, is usually very geometry-sensitive because the larger structure of turbulence are determined by the geometry. So a large eddy simulation only models the very fine scales of turbulence; therefore, has a better chance to be geometry-independent. That's always the down side of a RANS calculation.

MR. BOYD: Well, we went back to this slide. I think I missed something very important. Let's go down to the bottom right corner of the slide. You see a histogram. What we see here, first we have to understand what the histogram is.

On the bottom axis, the x-axis, we have created a normalized temperature range. If we take the hot leg temperature and call that a hot temperature, the return flow from the cold steam generator tubes and call that a cold temperature, we can normalize the temperature. All temperatures now will be between zero and one in the inlet plenum.

What we see in this particular test is that on a normalized scale, we have got temperatures of about .5 to .55 in that bin. We have made 20 bins 5 percent apart. And that is the way we have stuck with that kind of a category.

You will see on the left scale it's the percent of tubes in the bundle. So if this were something that we were doing, we wanted to know how many tubes were how hot, in this particular case, we would say that about 2.1 percent of the tubes fall into that category. And that's how we could get the temperature.

Then with that normalized temperature, then we can apply that back to the full-scale conditions. And we have shown that this histogram is pretty fairly stable over a wide range of conditions so that we can look at it in that way.

And, of course, we repeated this actual scale so we would get full-scale conditions.

MEMBER ARMIJO: The differences in those histograms on the temperature scale, is that about ten percent from one box to the other?

MR. BOYD: Well, they should be .05 apart. I made 20 bins on a zero to one scale, so about 5 percent.

MEMBER ARMIJO: Five percent?

MR. BOYD: Right. So that histogram becomes very important. Now, the test had about 25 percent of the tubes instrumented. I re-created the histogram from the test data. And I got a very similar spread but with only every other row instrumented. And in the rows that were instrumented, every other tube was instrumented. It makes it tough to get that.

But I did see the same central tendency. I just didn't see quite the peaks out at the edges. But I felt very good that what we saw was very similar.

And the mass average temperature entering the tube bundle we matched right on as well as the mass flow.

MEMBER BANERJEE: So it must depend on

the flow rate into the tube. So the mass flow rate, how was that sort of fixed, the mass flux? Was that on the basis of a system code calculation? How did Westinghouse select .06 as the mass flux?

MR. BOYD: Where do we see .06?

MEMBER BANERJEE: It's in your --

MR. BOYD: Oh, no, no. Wait. Let's go back. From the facility, the one-seventh scale facility, it had a reactor vessel with electrically heated rods. So that is indirectly measured, but it is a measured value of .06 kilograms per second.

In the CFD prediction, we were interested in inlet plenum mixing. So we took the hot leg back to the vessel, and we used that as a boundary condition. And we put in .06 kilograms per second.

That was Graham Wallis' concern is that you're running a CFD model and you're inserting the hot leg mass flow, you don't really know what the hot leg mass flow is.

MEMBER BANERJEE: Right. But this must depend on various factors, right, like how much steam is being generated and what the reactions are going on.

MR. BOYD: That's right. If you change the conditions in the steam generator, you will change the conditions in the inlet plenum of the steam generator. You will change the --

MEMBER BANERJEE: What about the core?

MR. BOYD: And the same with the core. These are all dependent. What I'm going to show you is that we have come up with a way to predict the hot leg flow based on the conditions in the upper plenum and the steam generator. And it holds constant over a wide range of conditions. So we have solved this problem of how to get the hot leg flow.

MEMBER BANERJEE: The source of the flow is the core, right?

MR. BOYD: The source of the flow is the buoyancy-driven flow, like a chimney effect, up into those tubes. You've got the rising hot plume. And it drives --

MEMBER BANERJEE: Right, but --

MR. BOYD: -- it out of the core.

MEMBER BANERJEE: So there is a reverse flow coming back, and there is a flow going, right?

MR. BOYD: That's correct.

MEMBER BANERJEE: The net flow going is going to be if there is a loop seal, if there is a difference between the reverse flow and the forward flow, then it has to be generated in the core, right, by a mass balance?

MR. BOYD: If the system is slowly depressurizing, there will be a difference. And also if you are on the loop with the pressurizer and the pressurizer is pressurizing, that draws a little mass flow off. So in some cases, you can have more forward mass flow than you have returning. And the source of that extra mass would come from the core.

MEMBER BANERJEE: Right. So now what is the state in the core? You have water in the core still at this point?

MR. BOYD: Well, I mean, it's various states. Its natural circulation flow pattern sets up as soon as the water level gets down below the hot leg and we start getting super heated steam off the core. And it continues all the way down to the point of failure, where the collapsed water level is below the fuel, active fuel.

MEMBER BANERJEE: So let's take a scenario now that your loop seals have not cleared because if the loop seals have cleared, then you have a completely different flow pattern.

MR. BOYD: That's correct.

MEMBER BANERJEE: Right. So loop seals have not cleared, and the core is boiling off. Of course, if the core is completely exposed, then you're getting hydrogen and all sorts of things happening.

But let's say the core is boiling off. You're getting some steam flow. You're getting a little steam cooling. You haven't got a big hydrogen generation rate yet. Okay.

Still, there is a positive flow. And as the core level decreases and goes down, until you start to generate a lot of hydrogen, I am assuming that the steam cools things until the core is pretty well uncovered. Your mass flow is going to drop because the level is going to drop, right?

So eventually until you generate hydrogen, your mass flow is going to get down to almost zero when the core gets uncovered, right?

MR. BOYD: That's not what we see. The mass flow, you mean the mass generation coming off the core?

MEMBER BANERJEE: Steam generation.

MR. BOYD: The steam generation? That, of course, would go down. I don't have those numbers.

MEMBER BANERJEE: So the only flow you're getting is due to, if you like, if you put a block at the top of the core or, say, at the bottom of the core and say there's no more steam generation because the core is uncovered and I'm not getting a lot of hydrogen yet, just starting to get hydrogen, you can get a situation where the net mass flow is close to zero.

So when you talk about this mass flux from Westinghouse experiments, what is that? The mass flux is the net mass flux or is it the mass flux? Because it makes a huge difference.

MR. BOYD: That's right. Let me --

MEMBER BANERJEE: When that mass flux goes to zero, you're going to get very little mixing.

MR. BOYD: Yes. And I did put "mass flux." That's a bad word there. It's mass flow. And it's the net mass flow from the vessel --

MEMBER BANERJEE: In kilograms per second?

MR. BOYD: In kilograms per second from the vessel into the hot leg.

MEMBER BANERJEE: Okay. So it's mass flow?

MR. BOYD: Mass flow. That is --

MEMBER BANERJEE: Net mass flow.

MR. BOYD: And I guess it's the total

mass flow going into the hot leg, entering the hot leg.

MEMBER BANERJEE: Right. Now --

MR. BOYD: Now, that same amount comes back in this experiment.

MEMBER BANERJEE: It is exactly the same?

MR. BOYD: Yes. These experiments were set up. There was --

MEMBER BANERJEE: Flow natural --

MR. BOYD: There was no steaming. They were running sulphur hexafluoride and just heating it. So if .06 went over, .06 came back. They weren't leaking sulphur hexafluoride out into the --

MEMBER BANERJEE: All right. So it was just a closed natural circulation loop?

MR. BOYD: That's right. This was to show that we could cool the core in this environment. And it also helped study the tubes.

MEMBER BANERJEE: The core was completely uncovered. That's turned around this --

MR. BOYD: In this particular experiment, what they did is they had a floor on the core, which would have maybe been your low water level. And then we just had sulphur hexafluoride in that facility, heated it up, and had a heat sink over at the generator. And then this pattern sets up.

MEMBER BANERJEE: When you say one-seventh, is it one-seventh reduced height scale as well or one-seventh --

MR. BOYD: Yes. It's reduced height, and I think it's geometrically scaled specifically in all directions. So it's one-seventh down. Now, of course, you didn't take 3,000 tubes and make 3,000 coffee stirrers. They then combined tubes into larger tubes and ended up with a few hundred total tubes.

MEMBER BANERJEE: So you have a one-seventh height core and a one-seventh height --

MR. BOYD: That's right.

MEMBER BANERJEE: -- and a one-seventh height plenum as well?

MR. BOYD: That's right.

MEMBER BANERJEE: So the geometric scaling in all directions?

MR. BOYD: That's the theory, right.

MEMBER BANERJEE: Okay. I see. So it's Reynolds numbers, and everything is very, very different from full-scale?

MR. BOYD: They're not very, very different. That is the point of the sulphur hexafluoride.

MEMBER BANERJEE: Scales are not different? Oh, because you used more dense--

MEMBER BANERJEE: Yes. The sulphur hexafluoride ends up this stuff is almost 100 kilograms per meter cubed when it's pressurized. I mean, it's a very, very dense liquid. So, surprisingly, the Reynolds numbers and the Grashof numbers, a lot of these numbers were within an order of magnitude --

MEMBER BANERJEE: Of full-scale?

MR. BOYD: -- of full-scale conditions.

MEMBER BANERJEE: So the length scale differences are compensated for by the density being higher?

MR. BOYD: That's right. And, of course, you don't match everything, but that's why sulphur hexafluoride was chosen, though, to get close.

MEMBER BANERJEE: There is a report on the scaling of this, right?

MR. BOYD: There have been a lot of studies on that scaling. That's right. And we could dig up a reference for you on that.

MEMBER BANERJEE: So first somebody satisfied themselves, I assume, including Graham

Wallis, that it was properly scaled? That's the first thing that you would worry --

MR. BOYD: Those arguments that are made that it's properly scaled, that's right.

(Laughter.)

CHAIR POWERS: I think it is safe to say that it's properly scaled for some purposes.

MEMBER BANERJEE: So it has been examined. So, you know --

MR. BOYD: That is right. It definitely has been examined several times.

MEMBER BANERJEE: The scaling? Then the appropriateness of, I guess, the next thing would be the turbulence model. Somebody has actually looked at that other than you, Chris, in terms of your using a Reynolds stress model and the mixing?

MR. BOYD: That's right. That's NUREG-1781. And we ran a whole series of turbulence models. Quite frankly, for getting these mass flows nailed down and the general mass flows and temperatures nailed down, we didn't find it to be very sensitive.

As you know, with CFD, if you're looking at a specific profile in a corner somewhere, the turbulence models can make a huge difference. The flow might even be going in a different direction. But if you're looking at some integrated number, like a mass flow and a mass average temperature, which is what we're trying to get to tune SCDAP/RELAP5, then it's far less sensitive. And that's what we found.

MEMBER BANERJEE: But my concern is more with whether you can get regions where you've got very hot fluid which is not mixed. You know, how --

MR. BOYD: Given the way this plume takes off and comes into the inlet plenum, the concern would be, can some of the flow coming out of the hot leg somehow not mix --

MEMBER BANERJEE: Yes.

MR. BOYD: -- and make it to the inlet plenum? At least in the Westinghouse plant, that doesn't seem likely.

MEMBER BANERJEE: Because, for whatever reason, that plenum is well-mixed, relatively well?

MR. BOYD: It's not well-mixed, but there is a lot of activity going on in there. And it's fairly well-mixed.

MEMBER BANERJEE: So you don't get a sort of a Coanda effect, where part of the plume hugs the wall, the hot plume, and just goes up the wall to the --

MR. BOYD: No. This comes out. It actually --

MEMBER BANERJEE: Enough of a jet?

MR. BOYD: -- accelerates up the nozzle. The hot flow accelerates down the hot leg. And it starts to decrease and form a jet, really accelerates up the nozzle. It's a very strong plume. And it comes out of that nozzle at an angle and then goes up. And it is not slow enough to see that Coanda effect.

MEMBER BANERJEE: Right. Okay. But the concern, of course, here is in some way how we will be able to predict this mixing, which will be very dependent on the turbulence model.

MR. BOYD: I'm telling you we didn't see that at one-seventh scale, this great dependence on it. You would think, but we did not see that.

MEMBER BANERJEE: Okay.

MR. BOYD: And I show this example here. We demonstrate that, at least at the one-seventh scale, we have really nailed --

MEMBER BANERJEE: What is the diameter of the pipe?

MR. BOYD: At one-seventh scale, we had a four-inch pipe.

MEMBER BANERJEE: The real situation --

MR. BOYD: About 29 inches, I believe. Is it 30 or 29?

MEMBER BANERJEE: They used a schedule 40 in a 4-inch pipe. It may not be exactly scaled. They used what is available.

CHAIR POWERS: You get specialized pipe.

MR. BOYD: Right.

CHAIR POWERS: It might be a little expensive.

MR. BOYD: That's right.

CHAIR POWERS: What strikes me remarkable on your next slide is that the mixing fraction is ten percent different.

MR. BOYD: That's right.

CHAIR POWERS: That strikes me as a lot of difference. And it's a difference in the direction of being less threatening to the tubes.

MR. BOYD: I'll tell you what. It turns out that the mixing fraction for the way we do our calculations isn't critical. One, the mixing fraction determines the average tube temperature.

Two, the temperatures that go into the tube sheet are very close to that middle volume 106. So we're taking eight percent of the hot leg flow in that case, and we're bypassing.

There is also the effect of we get entrainment of two times going in. So when you look at the total flow that we bypass, it's a very small fraction of the tube bundle flow.

And it turns out that the mixing fraction is not a really big impacter on the average temperature going on. And then it also is much more important what our hottest tube temperatures would be because that's where we're looking at the failure.

MEMBER BANERJEE: That's where the action is.

MR. BOYD: That's where the action is. So this mixing fraction that we have focused on for the last ten years I can personally consider to be a little bit of a red herring. It's not as important as you might think.

MEMBER BANERJEE: It's the tail of the distribution that matters here.

MR. BOYD: More important. That's right.

MEMBER BANERJEE: Let me ask you another question about this, the recirculation. I mean how much is going down the hot leg and coming back up.

That is very dependent on the sort of stuff, resistances and so on, in the whole core region, right, with the fuel and what is there in the upper plenum?

MR. BOYD: You're saying that the hot leg flow is dependent on the core resistances?

MEMBER BANERJEE: Well, it is because it is going and then it's rising through the core, right, and back? There is a resistance there, not --

MR. BOYD: That sounds a lot like the letter I wrote back to the ACRS.

MEMBER BANERJEE: Oh, did you?

MR. BOYD: They argued and said, "That is not the case."

MEMBER BANERJEE: Whoops. But I thought they asked that you did a more detailed core sort of model or something?

MR. BOYD: No. Well, I think the core of --

CHAIR POWERS: The intent of the ACRS was wrong. We wanted to motivate you to --

(Laughter.)

MR. BOYD: I'll tell you what I found, Sanjoy, is that what we're saying is that the flow in the hot leg is dependent on the upper plenum conditions in the vessel and the inlet plenum conditions in the steam generator.

We have been able to vary those and show that the densometric Froude number correlation gets the same coefficient. There have been experiments on that, too, in different geometries that show over a wide range of conditions that the mass flow is proportional to a densometric Froude number with a coefficient that should be determined --

MEMBER BANERJEE: Between these two?

MR. BOYD: Between those two. Now, where the core flows will come into play is they will impact the temperature in the upper plenum, that will then impact the flows in the hot leg.

All of that is somewhat modeled in SCDAP. In other words, if the core acts differently, we will get hotter conditions in the upper plenum. And that will impact our flow in the hot leg because that is a physically based correlation based on a density difference between those two.

MEMBER BANERJEE: Let me just understand the physics. What is happening is that core steam, say, or SF6 is coming down and it runs down into the core, where it heats up, and it rises in a plume through the core maybe or --

MR. BOYD: That's right. That is going to come back up out of the core.

MEMBER BANERJEE: And it comes into the upper plenum and runs along the top of the hot leg back to the steam generator more or less.

MR. BOYD: There are obviously some circulations going on in the vessel.

MEMBER BANERJEE: Right.

MR. BOYD: But we are going to assume that a big chunk of the upper plenum is probably fairly well mixed. And it's going to be similar to the temperature coming out of the core. And this --

MEMBER BANERJEE: Is that really true? There is no temperature stratification in the upper plenum?

MR. BOYD: There would be a little, but the point is there are flows there. So that is going to keep that from just forming some sort of a stratification layer because there are some flows going on in the vessel, some circulation.

MEMBER BANERJEE: So let's say that there is enough just for the purposes of continuing the discussion, there is enough mixing so that the upper plenum is more or less well mixed --

MR. BOYD: That's right.

MEMBER BANERJEE: -- at some temperature, some average temperature, --

MR. BOYD: Okay.

MEMBER BANERJEE: -- and these is steam leaving this, which is heating -- as hot as it can get. There is no stratification, though, that is leaving the upper plenum as the mixed mean temperature or is it leaving at the hottest temperature that you could get out of the outlet of the core?

MR. BOYD: We find very little stratification in our 3D calculations. So we're getting it leaving at a fairly close to the mix mean. In SCDAP, there is a series of volumes in there. They also show a flow pattern. And we're pulling it out of the --

MEMBER BANERJEE: So what is heating it up is basically you have to get rid of the decay heat?

MR. BOYD: That's right.

MEMBER BANERJEE: That is what is heating it up.

MR. BOYD: And the fact that you are dumping cold flow from the generator that has been cooled by this huge heat sink and you're dumping it back into the vessel, that helps stir things up and keeps the natural circulation flows agitated in the vessel also.

MEMBER BANERJEE: The heat sink in this case is basically the structures?

MR. BOYD: That's right. And primarily the surface area of the generator makes it the big heat sink.

MEMBER BANERJEE: That's because that is the largest structure.

MR. BOYD: I mean, the hot leg has got a seven length diameter ratio. And the surface area is not that large compared to the generator.

MEMBER BANERJEE: I am just trying to picture this physically. If there is no stratification --

MR. BOYD: I think when you see some of the pictures that I show later that kind of shows some of the CFD model, it may be easier to picture that.

MEMBER BANERJEE: All right. Let's hold it, then. Keep going.

MR. BOYD: The point here on this slide was to show that the tests were fairly well-scaled if you kept the geometry the same. And, surprisingly, we got very similar results.

However, heat transfer had to be adjusted somewhat to be consistent with the facility. So we had an issue with heat transfer being important and, in addition, geometry of the inlet plenum.

So then we took our full scale --

MEMBER BANERJEE: When you say "heat transfer," this is heat transfer to the metal mass?

MR. BOYD: To the metal mass. That's correct.

MEMBER BANERJEE: Is that different amounts of metal mass or when you say --

MR. BOYD: At one-seventh scale, those tests were -- the metal mass would not quite be the same depending on how you look at it as a 3,000-tube generator. We had 216 tubes in the --

MEMBER BANERJEE: Metal pass per unit volume was different.

MR. BOYD: And a lot of times many of the tests were done in the facility at a steady state condition where they were driving heat all the way through, as opposed to heating it up. They would run them at a steady condition to get these.

So what we show here is that when we compared the facility to a prototypical generator, we had some geometric differences. And what we see is that the nozzle is a little bit closer to the tube sheet. So there is going to be a little less mixing distance.

And, in addition, the nozzle is a little bit contoured. And it expands out. It makes it a little easier for the hot flow to enter the inlet plenum and get up to the tube sheet.

So if we go back to our histogram on the right, this is running severe accident conditions through our scale-up model, which is the facility geometry at full-scale size with steam compared to running what we consider a prototypical generator geometry with the same conditions.

And what you see is there is a larger tail out to the right. We have got the red bars go out to the right a few more distances. And that represents the reduced mixing in this inlet plenum.

MEMBER STETKAR: Chris, on this particular display here, can you explain? There are error bars there called the standard deviation value, which I start to interpret as some measure of our uncertainties about this process. How are they evaluated?

MR. BOYD: What we find is that in the model 44 geometry and the model 51 and all the Westinghouse geometries, the hot leg comes in at an angle, as you can see from the bottom left. And what we get is some unsteady flows in the inlet plenum.

So we run these as a transient simulation. And we collect data over long periods of time and average it. And these are just simply the standard deviation on that average.

MEMBER BANERJEE: The pattern is not steady, even in our answer. You have to run it on steady, and it fluctuates.

MR. BOYD: I will show you an animation in a little while, and you'll see what we're talking about.

MEMBER BLEY: Do you consider those fluctuations representative of the real physical situation or some attribute of the modeling?

MR. BOYD: Well, when you look at a Reynolds-averaged Navier-Stokes code, there is some issue because of the basis for the model itself. So this would have to be validated.

There are some cases where these types of codes can predict the unsteady behavior very well, but this has to be studied.

MEMBER BLEY: We're not sure.

MR. BOYD: And we did not have any data to do that, to tell you that those fluctuations are real.

MEMBER BLEY: Okay.

MEMBER ARMIJO: Could you explain what you mean by "percent of hot tubes" in your previous slide?

MR. BOYD: Percent of tubes in the range. Let's look at the -- well, if we --

MEMBER ARMIJO: That's a big number compared to what we were talking about before, you know, percent.

MR. BOYD: What that means is percent of tubes, I believe, would be -- if it was two percent, then two percent of the tubes in the bundle would be at that temperature.

MEMBER ARMIJO: I'm trying to understand that 38. What is --

MEMBER BLEY: Maybe back up.

MR. BOYD: The 38? Am I on the wrong slide?

MEMBER ARMIJO: Yes, on page 7. Yes.

MR. BOYD: Okay. "Percent of hot tubes."

MEMBER ARMIJO: Yes.

MR. BOYD: What we predict with the computational fluids is when this hot plume rises, it chooses a fraction of the total bundle and has forward flow in those tubes. It can't choose the entire bundle because there has got to be some return flow.

In this particular case, 38 percent of the tubes in the bundle carried the hot flow forward. And in both cases, it was 38 percent.

MEMBER ARMIJO: Okay. Okay.

MEMBER BANERJEE: I guess if you looked at the previous slide, that --

MEMBER ARMIJO: Yes. I got it now.

MEMBER BANERJEE: I guess, continuing Chris' comments there, the reason you might get an unsteady flow pattern is you see the actual steam generator plenum has aximetric entry, as opposed to a symmetric entry, where you might get a nice --

MR. BOYD: That's right. It is a non-symmetric inlet plenum, and the flows are buoyant, which can be a little unsteady. And the return flows fluctuate a little bit. So all of that drives the plume around.

There was a French study a few years ago where they did a large eddy simulation on this. They saw similar plume behavior --

MEMBER BANERJEE: Oh, really?

MR. BOYD: -- and hot leg behavior from

what I saw.

MEMBER BANERJEE: What code did they use, their own?

MR. BOYD: I believe it was TRIO.

MEMBER BANERJEE: TRIO? Okay. That's a very good code, actually.

MR. BOYD: They were doing something that was running on the order of six months or something.

MEMBER BANERJEE: Right. But who did this one?

MR. BOYD: I don't remember. It's been a while. There was some behavior I observed where the plume actually seems to back up for a second. It gets choked by some return flow. And they had seen the same behavior. And I thought that was interesting.

MEMBER BANERJEE: Well, if you could give me a reference to that?

MR. BOYD: I can try to --

MEMBER BANERJEE: Do you have a copy of the paper?

MR. BOYD: I can try to dig that up.

MEMBER BANERJEE: Yes. That would be very helpful.

MR. BOYD: Okay. Let's move on. So what

we did, we showed that we scaled up and we got similar results. But then we showed that when we look at actual geometry, we get a slightly different result for a few reasons. There are two things going on in there.

One, we get a little less mixing. And we're tracking the hottest region. So it looks like it's getting hotter. But at the same time, the plume is unsteady. And it doesn't stay at one region. And that mitigates some of the hotter temperatures. And we're going to talk about that as we go forward.

Now let's look. After we did the Westinghouse, we move on to the CE plant. And what we found is that the geometry of the inlet plenum here was important. You have a 42-inch hot leg now, 8,000 tubes in the generator. And the hot leg is on a relative basis closer to the tube sheet.

So we probably didn't need computational fluids to estimate that we might get less mixing. But we went ahead and did it anyway. And what we see is that there are some tubes, the hottest tubes now, if we plotted that histogram, the bins go out to the .9 kind of range on that scale.

This color scale that I have here is sort of your zero-to-one histogram scale. This isn't perfect. I just drew some arrows. But that gives you an idea of what we're looking at. That is explained in more detail in NUREG-1788.

Okay. So here is where we are. We have modeled and answered the questions in the action plan. We benchmarked our code, convinced ourselves that it does a reasonable job. We looked at various inlet plenum geometries. We also considered tube leakage, which I didn't show you. I looked at single leaks, distributed leaks, and things like that.

We determined that given that we had all of this information, probably doing more experiments was cost-prohibitive unless we really felt it was necessary if we needed more refinement than what we have.

And that's where we ended up. And we presented that to the ACRS.

MEMBER BANERJEE: Can you just back up to the previous slide, Chris?

MR. BOYD: Sure.

MEMBER BANERJEE: You see that on the Westinghouse visualization there. Is that corner the upper corner where the hot leg joins the plenum? Is that a fairly sharp corner or is it a rounded corner?

MR. BOYD: It's sharp in the CFD model.

MEMBER BANERJEE: What about in real life?

MR. BOYD: And I don't know in real life. The drawings I have --

MEMBER BANERJEE: That is going to make a huge difference.

MR. BOYD: That could impact the way it jumps off of the plenum, but I think when you see the way the plume moves around, it may not make as much difference as you think.

CHAIR POWERS: Let me ask you a question about the moving around business. Now, one of the speculations in our initial discussions of this work was if we had a leak in the tube that might tend to stay, the plume starts going up a particular set of tubes.

MR. BOYD: I've got some really nice animations on the leaks if you give us about five more slides.

CHAIR POWERS: I'm not the controlling factor so far.

MR. BOYD: Okay.

CHAIR POWERS: Go ahead.

MR. BOYD: Well, if you save that until we get to that point, --

CHAIR POWERS: I will.

MR. BOYD: -- you will see exactly what you want.

MEMBER BANERJEE: Also in the model, was that a sharp corner?

MR. BOYD: It was a sharp corner in the model.

MEMBER BANERJEE: I mean in the one-seventh model.

MR. BOYD: The drawings I had did not list any curvature on there. So I don't know. I had things that were glorified PowerPoint slides to build from. I didn't have machine drawings.

MEMBER BANERJEE: Yes. So you did the best you could, which is you --

MR. BOYD: I used the drawings and dimensions in the report, the test report, but it did not give me machine shop drawings.

MEMBER BANERJEE: Your point about it being unstable, though, and moving around could change it because, even if it was a more rounded corner, because it moves around, you might not get the corner effect. But usually if it's rounded, you get the corner effect, little stick.

That makes a huge difference to what

happens to the plume in real life. You see that wall jet will just go right around a corner. I have seen this. And then you get vortices, but it will stick.

Anyway, carry on. So clarify that.

MR. BOYD: Okay. We have talked about the CFD modeling effort, and we presented that to the ACRS. And here is a note from the ACRS meeting. Now, Dana requested us to help him write his next report. So I left the first sentence in here in case --

CHAIR POWERS: Beautiful.

(Laughter.)

CHAIR POWERS: Just copy the first one.

MEMBER BANERJEE: This is just copied from the last letter, right?

MR. BOYD: It is copied from the last letter, you know, nothing shameful here.

MEMBER BANERJEE: This letter?

CHAIR POWERS: I think we'll maybe change "excellent" to "marginally adequate" or something.

MR. BOYD: But the point is what is in blue. And that says, "The ACRS has requested we perform a similar study and let's predict the hot leg flow." So let's go through that.

So we updated our CFD model. We had time to look back on the two NUREGs we published. We had time to review the assumptions and limitations.

We also met with the industry or EPRI in a public meeting and discussed and presented our results and took comments. We also had expanded computer resources, which always helps. And, in addition to looking at our model again, we also took another look at how we implement these results in SCDAP/RELAP.

So all of that was done in response. And we have NUREG-1922. This spells that out. And let's take a quick look at some of the highlights.

Here is our updated CFD model. What we have got is a quarter of a vessel inside the core barrel with regions and loss coefficients to represent the fuel region, the upper core support plate, the upper plenum, and various regions going up. We had some drawings and dimensions of the vessel. And now we have the hot leg connected to this.

We also put a surge line on this loop, which we didn't have before. We put a side and a top-mounted, which we could turn on and off. We also added hydrogen to the mix, the ability to track hydrogen and have that impact on the density.

And we didn't get into the details, but the prior tubes were square. The goal was just to create a boundary condition. And it did a very nice job, the boundary condition, on the inlet plenum. And it got the flow resistances going in.

This tube bundle actually uses tubes that each tube represents a three-by-three array of tubes. So we matched the flow area. We're going to match the transit time of the flow through the bundle. And then we used some adjustments in the code that we could make sure we had the proper total pressure drop along the tubes and the right heat transfer. So we created a tube bundle that is far more realistic. We also used a lot more nodes in the inlet plenum.

MEMBER BANERJEE: How did you do the adjustment?

MR. BOYD: What we did is we took a three-by-three array of tubes. And we put various mass flows through it at various temperatures. And we looked at the heat transfer drop-off, the temperature drop-off rate, and the friction rate. We went up and around the bundle.

And then we created one tube with the same flow area. And we did that. And, of course, the first run we didn't get the right pressure drop and we didn't get enough heat transfer because we don't have enough area.

So we jacked up. We called it a porous media so we could add a source term to it to adjust the pressure drop. And we matched the pressure drop right.

You know, the two plots are right on top of each other all the way around the bundle. And then we jacked up the effective thermal conductivity so we could drive the heat out a little faster towards the wall. I mean, that's the problem. The tube's got too big of a cross-section. By doing that, we were able to match the temperature drop. We sat there and played with that. And we did it over a range of conditions.

This is stuff you can't do in an experiment when you have less tubes, but in the CFD world, since it's all just numerical, we were able to create conditions. We were able to create a tube that has the same behavior as the three-by-three array of tubes that --

MEMBER BANERJEE: In some way you jacked up the losses in the heat transfer?

MR. BOYD: We jacked up the losses in the heat transfer. So that is our model.

Here is just an animation. This is a cross-section, a vertical cross-section, of the hot leg. It shows the surge line mounted on the side in this particular case. And the tube bundle is taken away so it doesn't block our view. And you will see the tube sheet face and the temperatures at the tubes. And you can see this. I guess we really can't see the contrast here too well. These screens don't do it justice.

But the hot tubes are moving around in space or across the tube sheet. That plume is moving.

MEMBER BANERJEE: Getting a form of flooding at the inlet.

MR. BOYD: That's what it looks like. That's right.

MEMBER BANERJEE: Is that true?

MR. BOYD: And that's what the French had predicted, too, in that return flow from the plenum. That's right.

MEMBER BANERJEE: You do see that even in the vapo-liquid flow.

MR. BOYD: But I'll tell you --

MEMBER BANERJEE: That's why we worry about reflux condensation.

MR. BOYD: Some of that can fool you, though. Some of it is the flooding issue where the plume is backed up and some of it is the plume has moved off the center line. So there are two effects going on there. But you are right. That is exactly what some of it is.

Anyway, let's move on. So here are now the key points. We go back to that diagram of the SCDAP/RELAP nodalization. And we think of what we have to adjust in SCDAP/RELAP. One of them was the surge line temperature.

With a side-mounted surge line, if you don't make an adjustment in SCDAP/RELAP, it only draws from the top of the hot leg. And that is because the top of the hot leg and the bottom of the hot leg are disconnected. And to get from the top to the bottom, you probably have to travel about 40 meters to get back there. So there is a pressure difference.

Anyway, we found from the CFD that it oscillates, but it averages out to about 50/50, which is lucky. So we were able, then, to force SCDAP to take equally from the upper and lower side. And that significantly reduces the temperature of the surge line, as you might expect.

The next thing, the hottest tube location, now we've got individual tubes. And we can track each one individually. So we now have the mass flow and temperature at each of these tubes.

What we found, the lessons learned, the hottest tube location varies with time and that the tube upflow pattern varies with time. What we see, you see this crazy up flow pattern.

It locks into that for a short while. And then it's slowly adjusted. Then, all of a sudden, it will shift a little bit. And then it can shift around.

What we found, though, is it can even change from 30 percent of the tubes in up flow to 40 percent of the tubes in up flow, 45, but the hottest core region doesn't change. And the hottest tube doesn't change. So that makes our lives a little easier when we're trying to predict the hot --

MEMBER BANERJEE: This is buried in that?

MR. BOYD: It's at the edges where things adjust, but in the core region above the hot leg, where the plume is strong, things aren't going to reverse flow.

Let's see what I see here. So we look at the tubes individually I guess is the key result here. What this means, what we have done now, is on the left side of this graph, I look at my old work from a few years ago. And what we used to do is we would track the hottest region and how many tubes were that hot.

I show the data on the left. The concern was, well, the hot tube moves around. It's not fair to always assume the same tube is always that hot.

So if I take the same data now with this new model and I look at it a different way and see individual tubes, what is their mass average flow temperature, I get this histogram, which has all of the edges muted off and all the peaks. So all the peak, when the hot plume goes by, that is cut off. And now I get normalized temperatures in the range of .4 to .45.

Now, what this graph is showing us -- and I can't read the numbers on mine either, but it looks like I'm saying about two percent of the tubes are in the range from .4 to .45 on my normalized scale.

Now, in our screening calculation, when we did the hottest tube, we used .5 for the hottest tube. And we found that we ran a series of sensitivity studies and we found that .5 sort of bounded what we did. And we used that.

But if we were going to do a detailed analysis, we would come to this histogram. And we could find out how many tubes are how hot.

MEMBER BANERJEE: You said in your previous slide the hottest region always stayed -- even though the edges were shifting around, the hottest region stayed fairly stable.

MR. BOYD: Well, let's go back to that. Now, that is a misunderstanding. I said it wrong.

MEMBER BANERJEE: Okay.

MR. BOYD: The average hottest region. So in this case, I show in the center of those 21 tubes, in that little box in the middle, the hottest tube. But if we took snapshots of this every second, that hottest tube moves all the way outside of that. And sometimes it's maybe outside of that box completely. This is the average over maybe 120 snapshots in time. This is what it averages out to.

MEMBER ARMIJO: What is that time scale? Is it like --

MR. BOYD: It is fairly fast. I have some plots of it. I don't think I have one with me. But what we have are the oscillations are fairly quick. It does not lock onto a hottest tube and stay there for 20 seconds.

MEMBER ARMIJO: Yes.

MR. BOYD: This is something that is moving around.

MEMBER ARMIJO: So it keeps your peak temperature down on any tubes.

MR. BOYD: And that is the effect here of what I am saying. We have improved our estimation of the hottest tube by looking at individual tubes. And that is what I am trying to point out on this slide. We still use .5 as a screening criteria. We believe we are below that. This is the actual data. Now, for a different steam generator, this would look different.

We'll go on to the next. Now, the next one, this is actually the subject of the ACRS concerns with how do we predict the hot leg flow. And up in the top right, you will see the correlation that we used.

There are some reports out where a scaling analysis is done and they come up with this Froude number correlation, which should be the governing parameter for mass flow between two volumes of different temperatures, different densities.

It has been demonstrated that that coefficient, the discharge coefficient, will be constant over a wide range of conditions. We found the same thing. We found a value of .12. And it was steady over a wide range of conditions.

So now by doing this, we are able to take the CFD calculation and determine what the hot temperature coming in on one side of the hot leg is, the cold temperature coming in on the other side. We relate those to densities, plug them in, and we can get our discharge coefficient. And then we'll do a series of sensitivities to see how stable it is.

And now in our SCDAP/RELAP model, we can monitor those temperatures and make sure that the flow rate is consistent. And that is how we predict the hot leg flow.

This becomes a physically based correlation. If something happens in the core, the temperatures get hotter. The density is going to change. It's going to change the mass flow.

I am going to buzz on. The next one is an updated mixing model. Prior to this, all of the mixing models were based on where the thermocouple was in the one-seventh scale tests. So there was a temperature measurement. And from there to the tubes, there was a mixing calculation done.

We realize there was some entrainment in the hot leg, some mixing. It's not the majority of the mixing but some mixing. So we moved the reference temperature for the mixing all the way back to the vessel. It also corresponds to the reference temperature we're using for our hot leg discharge coefficient. So this makes things a little more consistent. It's also a lot easier to describe.

We re-derived the mixing model using that temperature. We had to re-derive it because now we have this surge line in the mix. And if a surge line is present, it draws mass out of the system.

We basically now have come up with a new formulation, a slightly formulation for the mixing. So we get a little more mixing, and we get a little bit more recirculation ratio calculated. And this is implemented in our SCDAP/RELAP model.

MEMBER ARMIJO: Now, before you leave that, what is the magnitude of the temperature difference between T-hot and T-cold right at that nozzle?

MR. BOYD: Well, of course, that changes during the transient. Oh, at the nozzle itself?

MEMBER ARMIJO: Yes, right in that area.

MR. BOYD: At around a time to failure, that T-hot is on the order of about 13 to 14 hundred kelvin. And the return flow is on the order of about 900 kelvin.

MEMBER ARMIJO: That should put a lot of stress analysis in that area. Is that part of your model if somebody does --

MR. BOYD: That is not part of our model, but whenever the finite element model -- our upper hot leg is not connected to our lower hot leg from a thermal point of view. So that is not in our model, but that would be included in the finite element model that is done after --

MEMBER ARMIJO: From a standpoint of failure time, either by creep rupture or pressure loads or bending loads and all of that stuff, has that all been treated in the analysis?

MR. BOYD: Again, our model just uses a simple Larson-Miller creep rupture calculation. And I think we can wait for the materials guys, who will talk about what they did with their ABAQUS models, what their assumptions were and what they -

MEMBER ARMIJO: Will that be tomorrow or something or are we going to cover that?

CHAIR POWERS: Coming up.

MEMBER ARMIJO: Great. Great. Thank you.

MEMBER BANERJEE: The point I would make, it is absolutely right because there has been a huge number of experiments done to look at this problem of countercurrent flow, just to look at the thermal striping and thermal stress problems. Whether it's important here I don't know, but in many cases, it is.

MR. BOYD: If we look at the next slide, we talk about leaking a little bit. So we have a leak of about 1.5 kilograms per second, was the first leak that we created.

Now, just to give you an idea, at these conditions, if we broke open one tube and had a side entry, side hole on it that was about one tube area, we would get a leak rate of about six kilograms per second. So this puts that in some perspective for you, how big a leak we're talking about.

What we found is that the natural circulation flow pattern continues and that the tubes around the leaker did not shift in that histogram to any noticeable difference. They stayed within a five percent band. So a leak of this size is not a big deal, coffee straw in the big river.

Now we doubled the leak: Three kilograms per second. The flow pattern still persists. We still get returned flow. A hot plume does have some attraction to the leak occasionally, and you can see that in the animation.

On average, the tubes around the leaker have a normalized temperature of .5, a little bit higher, maybe .05 to .1 higher, than what they did without the leak.

We doubled that again: six kilograms per second. Now, this is equivalent to a tube popping basically at these conditions. What we find here is that the natural circulation flows are almost gone.

The return flow actually chokes off every now and then. The hot plume occasionally locks onto the leak. You still get that flooding, it looks like, with flows coming back because it is a little unsteady. And the normalized temperature went up to .55. So we are getting hotter. When I say, "normalized temperature," I am talking about the tubes around the leaker, not at the leaker.

Now we will double that. This will be a double guillotine break. What we have done here is we have completely broken off the natural circulation flows. We still get strong natural circulation flows in the generator. So the inlet plenum still has a lot going on. There are still a bunch of return flows coming back.

And, as you can see, there is a stream of hot -- I moved the leaker off to our side a little bits we could see it better in this view. You will see that we're -- there is some decent attraction to the leaker.

But it doesn't stay on it all the time. And the tubes around it have a normalized temperature of .8. Now, that normalized temperature drifts up to one occasionally and then drifts down a little but, but it averages --

MEMBER BANERJEE: Now, there is no counter-current flow in this loop?

MR. BOYD: Not in this loop with this kind of a leak --

MEMBER BANERJEE: Other loops giving --

MR. BOYD: The other loops would continue in there.

MEMBER BANERJEE: That is what is moderating the temperatures.

CHAIR POWERS: But I think the concern at the time the issue was raised, we would be very sensitive. This shows a very smooth transition, almost intuitive. But the concern originally was that it would be very sensitive to the existence of leakage in the tube. And so what you're showing is not huge.

MR. BOYD: The concern, I thought, yes, was even a small leak could really mess up the mixing. And what we demonstrated in our previous work that was published before that the one and a half-kilogram size leak really wasn't changing the overall picture of this story.

But here we ran it out to the point where it does change it.

CHAIR POWERS: Yes.

MR. BOYD: And if we go any more than this, the flows down the hot leg normally are only about five kilograms per second during this part of the accident. And now we've got a leak that's 12. So we've got a fairly substantial leaker going on.

Okay. So the results of our updated modeling, we have got a physically based discharge coefficient for the hot leg flow. We have updated our mixing model. It is more consistent. It considers the hot leg entrainment, considers the surge line flows. We have got the surge line mixing considered. And we believe that we are modeling that far better than we were.

The tube bundle flows have been studied

in more detail. We have changed the way we come up with our hottest tube. We believe it is more realistic because of the motion of the tubes.

We have histograms of these temperature distributions. And we have looked at the impact of tube leakage in a better way than we had before to quantify what leaks are needed to break down the natural circulation flows.

We also completed various other sensitivity studies on hydrogen and things like that that are in the report. And NUREG-1922 documents that updated work.

So, in summary, we have got action plans 3.4.c, e, and g. And those are addressed by our previous NUREGs. There were some concerns on the way we modeled it. We have refined the model. And we've got NUREG-1922, which spells out the details of that refined model with some of these corrections.

One interesting thing I found was that with this new tube bundle model, we went from porous square tubes to this new tube bundle model. If we looked at the hottest tube and the distributions in the same way we used to look at them, we got a very similar spread.

So it is an indication that what we were doing before wasn't bad at all. This is --

MEMBER BANERJEE: The geometry should not matter that much, right?

MR. BOYD: What we were doing before, though, I was concerned about the oscillations. We saw those before. But the residence time of the flow through the tube bundle was different because we didn't match flow area.

Now we match flow area. So the time if there are different slugs going in, the time to get back will be similar. And that will help make the oscillations in time more physically based.

MEMBER BANERJEE: Let me ask you about these oscillations because the validation of the code, which is a RANS code, was for a symmetric inlet, where this thing was not cycling around.

Now, you are going to apply this in a transient calculation. You have done it in a transient calculation, but I guess you can't get a steady state with the aximetric.

What sort of validation do you have for that situation?

MR. BOYD: We would have to look at plume validations, which we have not done. We --

MEMBER BANERJEE: It can be any

experiment where the plume is --

MR. BOYD: If you were going to validate that, yes, we would need to get out into the plume literature and see if we could validate against something.

These have been done with the CFD tools. We did not do it at these conditions in this geometry for this problem.

MEMBER BANERJEE: The French did their aximetric inlet the same sort of --

MR. BOYD: You know, I don't remember. They did a real plant. So probably not. They shift the hot leg off to the side to leave room for the manway, it looks like to me in these generators. So I doubt that they did something aximetric of symmetric either.

The CE generators are large enough that they can come in the center and still have room for the manway with an 8,000-tube generator, but these 3,000-tube generators, they seem to shift them off to the side.

MEMBER BANERJEE: Yes. It would be very interesting because I know TRIO. And TRIO, of course, is a top-class code. I'm not saying that Fluent is not, but then there is -- and they did an

LES, right?

MR. BOYD: The work that I had seen was an LES model, right.

MEMBER BANERJEE: That would be useful if they saw a similar phenomenon.

MR. BOYD: The very small report that I saw I'll have to admit John Mahaffey helped me out with that. He was at a meeting, and he brought it back to me.

The key I saw, what I saw that I liked, was that they also saw that flooding, as you mentioned.

MEMBER BANERJEE: Yes.

MR. BOYD: And it made it a little more easier to understand.

MEMBER BANERJEE: If you know who did it, I mean, there are only a few people in France who use TRIO or N3S. We can always get the details.

MR. BOYD: Okay. That's all I have.

CHAIR POWERS: My own impression is our November 17th, 2004 was indeed perspicacious. So we may crib from that a little bit.

Are there any questions the members would like to have?

(No response.)

CHAIR POWERS: Thank you very much. I think we can move on now to a less technically arduous topic.

7. SGAP ITEMS 3.4.H-I

MATERIALS - POTENTIAL RCS FAILURE LOCATIONS

MR. CARPENTER: Good afternoon or should I say good morning based on the schedule?

CHAIR POWERS: By my schedule. And we know we're getting close to lunchtime. So you may want to move right along.

MR. CARPENTER: We will work this right along. I am Gene Carpenter. And I am with the Office of Research. Jeff Hixon is also with me from the Office of Research.

CHAIR POWERS: You are going to have to get a little closer to the action here, Gene.

MR. CARPENTER: How is that? I did consider bringing some raw meat here and throwing it based on what we --

CHAIR POWERS: You are the raw meat, sir.

(Laughter.)

MR. CARPENTER: I understand that. Thank you very much.

We are going to be talking about the steam generator action plan item 3.4h, the potential

RCS failure locations. Basically what the steam generator action plan had asked of us was to investigate the concern that during a postulated severe accident, core effluents may bypass the containment if failures are experienced of the steam generator tubes. However, obviously if some other RCS components fail before the tubes, then containment bypass may be averted, as you heard previously today with Chris, et al.

So what we did in research was that we performed a scoping review to determine potential failure locations, modes, and times to failure for these non-steam generator tube components during the postulated event.

For 3.4h, we conducted a three-phase scoping study. Phase I reviewed the methods and models for predicting failure modes and times to failure. It identified additional information needed for the study, and it also scoped out the components that might be considered weak links.

In Phase II, we took this information and developed the three-dimensional computer models of these selected components for the representative Westinghouse four-loop plant. And with that, we also utilized detailed mechanical and structural drawings and included analysis of operating histories of these components.

Finally, in Phase III, we utilized the Reactor Leak and Power Safety Excursion, RELAP, code and the CFD calculations and with use of the expanded high-temperature materials database to calculate the failure sequence of the selected components.

So in Phase I, we went and put together a workshop in November of 2001 to discuss the behavior of these components and the bolted connections during the postulated severe accidents. The participants included valve and gasket manufacturers, industry, EPRI, and some of the people from Argonne National Laboratory.

The workshop concluded that it would be possible to analytically predict behavior during severe accidents of certain components, certainly not all of the components but some of them.

Following the workshop, we then went and took a look at some of these components and bolted connections to model to predict the failure times. And we initiated an effort to develop improved models and also included variables not addressed in previous analysis.

For Phase I, the components that were

selected for the analysis were the hot leg and surge line, including the nozzles and supports; the steam generator primary side manway; top-dead-center RTD scoop that penetrates the hot leg, including the welds; the socket weld connection of the instrument line to the RTD flange; and the PORVs, plug-to-cage impact.

We also did a review of the operating histories of the relief valves, bolted and flanged connections, and spiral-wound gaskets.

MEMBER BLEY: I don't remember. How big are those RTD connections?

MR. CARPENTER: If I remember correctly, approximately one-inch. I think that's about the size and diameter.

MEMBER BLEY: Okay.

MR. CARPENTER: The analysis was based on the Zion Nuclear Station and simply because we had the access to the drawings and had a fairly decent PRA.

MEMBER STETKAR: Did you look at reactor head vent lines or aren't they at all interesting?

MR. CARPENTER: I apologize. I don't remember if we did look at those. I will find that out and try and get back to you.

MEMBER STETKAR: Thanks.

MEMBER SHACK: I don't remember doing that at Argonne. Whether somebody else did --

MR. CARPENTER: I don't know. We will find out.

Again, as Chris had mentioned earlier today, we looked at this for the hydraulic sequence under SPO. Results from the RELAP5, thermal hydraulics analysis for surface heat flux or flow is used as input for the thermal conduction and stress-strain analysis, failure times due to tensile and creep rupture calculated with data from literature when available, and extrapolated when data was only available at lower than severe accident temperatures.

CHAIR POWERS: Let me ask you --

MR. CARPENTER: Yes?

CHAIR POWERS: -- why did you select Zion?

MR. CARPENTER: Zion was we had drawings that were available. We were able to get some fairly good mechanical drawings. We also had a PRA that was available for that.

CHAIR POWERS: That is the problem. We don't really care.

MR. CARPENTER: Use it as a representative plant.

CHAIR POWERS: How representative is it?

MR. CARPENTER: It could be very representative. I don't remember how many of that type of plants were made, but we basically use it as a representative plant. I can find out.

CHAIR POWERS: I assume, I mean, it's very representative.

MR. CARPENTER: Yes.

CHAIR POWERS: Okay. Just curious. Yes?

MR. CARPENTER: Going on, failure times due to tensile and creep rupture were then calculated with the data from the literature. Sensitivity analyses were conducted to determine the variability of predicted failure times due to variations in surface heat, thermal conductivity, creep rate, and yield strength.

CHAIR POWERS: There is this line that you have there that says, "Extrapolated when data were only available at lower than severe accident temperatures." Isn't that the problem with applying these Larson-Miller-type formalisms in the RELAP code?

MR. CARPENTER: I'm sorry? I didn't hear

you.

CHAIR POWERS: Isn't that the problem with applying Larson-Miller type failure analyses, like they do in the RELAP code.

MR. HIXON: Good afternoon. Well, I believe this is Phase II. So in Phase III, you did get high-temperature data to address that.

MEMBER SHACK: We essentially ended up with experimental data on all of the materials that we needed at the time. When we did the initial analyses, we didn't have data for some of the materials because most of these materials are not designed to operate at this temperature and nobody ever bothers to get data.

MR. CARPENTER: Which takes us to the last bullet here that the available temperature material properties data were collected from the literature. And then over the temperature range was not available was identified and, as went into that for Phase III.

This graph is basically the same thing that you have seen earlier today, where, as we did the calculations, we saw for the initial calculations of the RTD and failure shortly after the Corps given instrument line going on to the RFD flange socket weld, surge line to hot leg nozzle weld, a hot leg near the PRD nozzle and then certainly hot leg does sometime after the steam generator tube.

In Phase III, we did improvements to the thermal hydraulics modeling. This would have been refinements made to the surge line to the hot leg connections of the RELAP model. Thermal hydraulic data calculated using RELAP5 was improved to account for entrance effects and flow reversals during the PORV cycling.

And, as Jeff mentioned, high-temperature materials database was expanded by conducting high-temperature tensile and creep tests on stainless steel and carbon steel weldments. Enhancements changed calculated failure sequence, and that resulted in the hot leg failing first. And it also suggested that the reactor coolant pump seals could fail prior to the steam generator tubes failing.

We then held another expert workshop held to evaluate the findings. And among the other findings from the workshop was an agreement that seal failure could occur sooner than previously estimated and could possibly avert or mitigate containment bypass.

CHAIR POWERS: Agreeing with whom?

MR. CARPENTER: Pardon?

CHAIR POWERS: Who is agreeing? It says an "agreement." And I'm not sure who is agreeing with whom.

MR. CARPENTER: During the expert workshop?

CHAIR POWERS: Yes.

MR. CARPENTER: As we did before, we had people there who were experts in flanges, who came from EPRI, from the industry, and other laboratory people.

CHAIR POWERS: Okay. So the next time somebody from the industry comes in and I ask him about pump seals, he will say, "Oh, yes, they fail."

MR. CARPENTER: Under certain conditions, it is possible.

(Laughter.)

MEMBER BLEY: You would like them to.

MEMBER SHACK: Again, these temperatures are well-outside the design range for these seals.

CHAIR POWERS: It's not very hard for me to get there either.

MEMBER SHACK: Well, let's hope it's hard for you to get there.

MR. CARPENTER: So, finally, for this,

the conclusions were that the improved models for determining time-to-failure of non-steam generator tube components under severe accidents were developed; times-to-failure between the non-steam generator tubes with the exception of the reactor coolant pump seals were relatively close to each other; and it was determined that seals could fail prior to the steam generator tubes, which could avert or mitigate containment bypass. And NRR and Research are looking at follow-on research.

CHAIR POWERS: The issue of failure, I mean, thermal hydraulic guys look at failure in a more subtle fashion than this. They have big failures and little failures,

You just declared failure. I mean, that's all you did was say, "Okay. Fail." So it could be a 22-gallon-per-minute failure or a 480-gallon-per-minute failure, and it's all just one failure to you.

MR. CARPENTER: Once it no longer contained pressure, yes.

CHAIR POWERS: So they are still stuck figuring out what the flow rate is. I mean, they just have to do that arbitrarily and find a sensitivity over the potential range. You offered them no insights on the volumetrics of the failure?

MR. CARPENTER: I don't think it was no insights, but we did find some information. And that was provided to the thermal hydraulics people, yes.

MEMBER SHACK: Well, I mean, if we blew out the RTD, we knew that was a relatively small leak. The failure of the hot leg, no, we don't have a good way of calculating the failure size of a creep burst. But, as I said, the judgment is that if this thing failures fails by creep, it's going to --

CHAIR POWERS: Do we have that capability?

MEMBER SHACK: It's always nice to have that, but it's a difficult problem, especially if you want to do an experimental verification. You know, this is a gas-type leak. You know, it's not going to relieve the pressure rapidly. It's going to continue to depressurize.

My judgment would be that this thing would be a fairly dramatic --

MEMBER ARMIJO: Since that is so important, that hot leg failure, how detailed was the analysis done on that? Where was the failure expected to occur in the carbon steel, the carbon steel this transition material where it's welded to the --

MEMBER SHACK: I talked to Saurin. And, again, we have to go back and refresh our memory as to which thermal hydraulics model we were using at the time we calculated which failure. But I think with the refined thermal hydraulics model, the failure occurred in the hot leg but in the stainless steel adjacent to the nozzle but in the stainless steel.

Part of this is, again, as I said, the general size of the nozzle, even though the material is weaker. It is also sort of a question of the heat transfer into the thing. So it is a combination of things getting hot.

MEMBER ARMIJO: The reason I am concerned or confused is that you've got these big temperature gradients from the bottom around the sides where it is really cold, maybe as much as 400 degrees Centigrade from the very top. So you have got a very complicated stress.

Of course, you have got the pressure stress, and that is probably the dominant theme, but I don't know.

MEMBER SHACK: Well, just think. This stuff is creeping pretty heavily at this point. We are getting all sorts of deformations due to these temperatures.

We're not building much stress. The stress is probably Pr/t because everything else is --

MEMBER ARMIJO: That is what I am trying to get at. Is it --

MEMBER SHACK: By "creep," you know, it doesn't take a whole lot of creep to relieve a thermal stress.

MEMBER ARMIJO: Right, right. Well, even at the colder part of that nozzle --

MEMBER SHACK: The colder part is still relatively cold, yes.

MEMBER ARMIJO: It's strong.

MEMBER SHACK: Yes.

MEMBER ARMIJO: Yes. That's very strong. So all the deformation would be concentrated --

MEMBER SHACK: Up at the top.

MEMBER ARMIJO: -- up at the top. So I am just wondering --

MEMBER SHACK: I think it's blowing out because of the Pr/t. It's not blowing out because of the temperature deformations. Saurin has the deformation pictures. I actually don't remember what they quite look like.

MEMBER ARMIJO: Is there any kind of a topical report or just kind of focused on the failure of the hot leg, you know, what went into it and --

MR. HIXON: Yes. I have a report, "Behavior of PWR RCS Components Other Than Steam Generator Tubes," November 2008. I think it was --

MEMBER ARMIJO: Well, as long as I know it is there, I will go take a look at it.

MR. HIXON: Right.

MR. CARPENTER: And it does have some of the pictures that Dr. Shack was discussing.

MEMBER ARMIJO: Okay. All right.

MEMBER STETKAR: I am a little curious about the conclusion that the RCP seal failure could avert containment bypass scenarios. I guess I understand the discussions.

If everything is driven by the hot leg, I am not at all concerned about that. But you seem to have raised that as yet another way that these induced tube ruptures are not an issue.

How much did you actually look at the available flow paths through those seals due to thermal failures of the elastomer materials because you remember the original analyses of those seals were looking at LOCA conditions and bigger flows are bad for LOCAs.

Here bigger flows are good things again. And seals when they fail tend to cock. They tend to get pretty tortuous paths. So I was curious how much of your work looked at that in terms of, in particular, of averting what we are trying to avert, which is the eventual bypass.

MR. HIXON: At the workshop there were seal vendors, correct? And pretty much we relied on their expert opinions for a variety of --

MEMBER SHACK: No. It was really the --

MEMBER STETKAR: But they have always thought about the bad thing --

MEMBER SHACK: The person, right, is a big hole.

MEMBER BLEY: So being conservative from that point of view.

MEMBER STETKAR: It is conservative to say that everything goes away and it remains in its original geometry, which maximizes the flow areas.

MEMBER SHACK: People were aware of that problem that what is conservative for one answer is not conservative for the answer we are interested in at the moment.

But we talked about it. I mean, this was only a conceptual idea. The thought was there was going to be some follow-on work, but that never happened.

MEMBER STETKAR: The only reason I raised it is it is given some prominence here in terms of the --

MEMBER SHACK: I think all you can really say is it's a potential mechanism and we haven't really investigated it. Now, how much credit you want to give it at this point is certainly a different --

MEMBER SIEBER: Well, I think there's another factor. Over the years, people have been working to improve these seal packages. And you can replace them fairly easily during a refueling outage. And people have done that.

MEMBER SHACK: Yes. There are never --

MEMBER SIEBER: And so 400 GPM is a cold number, I think, for many plants.

MEMBER SHACK: This is a very different condition. You know, the elastomers are gone.

MEMBER SIEBER: Yes. It depends on what the --

MEMBER SHACK: All those numbers are suspect that people use for these leakages, which is why the thermal hydraulics people compute a range.

MEMBER BLEY: We've been focusing on temperatures in the hot leg, and I haven't seen anything on what temperatures in the cold leg are as long as you maintained the loop seal. That's the temperature --

MEMBER SHACK: But certainly it's much cooler than this, but it's still --

MEMBER BLEY: The elastomers are hot, but what about other stuff?

CHAIR POWERS: You have to come to a microphone, identify yourself, speak with sufficient clarity and volume and --

MR. LUPOLD: I am Tim Lupold. I am Branch Chief for the Corrosion Metallurgy Branch. I happen to have a little bit of plant experience under my belt. And I know that these seals require cooling. And in the scenarios provided under this event, they are going to lose that cooling immediately. And those seals are going to heat up.

Typically the seals you are talking about these days are cartridge seals that are used on the RCPs. They usually have three stages. And they break the pressure down for leakage.

And those seals without the cooling are

going to overheat. They're going to fail. They are going to start to leak. Now, the exact magnitude of leakage, no one can really say that and point to it and say, "This is what it is" without actually doing some testing on them. Okay?

We haven't done that testing on them, but without that cooling going into these seals, they are going to fail fairly quickly and start to leak.

MEMBER BLEY: Actually, some time ago you had ACEL do substantial testing. Now, the seals may have changed since that time, but there were a number of experiments.

MR. LUPOLD: But, as John said, they were from the point of view of maximizing this leakage.

MEMBER SHACK: And those were the people we had at the workshop. And there was some thought that they would sort of do additional testing, but that never --

MR. LUPOLD: And a lot of the testing that has been done in the seals has been relatively around operating parameters. And these are considerably different than operating parameters. So seal experience is going to be a lot different than under operating conditions. That's all I just really want to point out.

MEMBER ARMIJO: Are these all elastomer materials or what is a design of the seal?

MR. LUPOLD: I would have to go back and check that. It's been a long time since I looked at the seals. I really couldn't tell you off the top of my head. I hate to give you wrong information.

We could do some research on that, dig some drawings out from the vendors and --

MEMBER ARMIJO: Well, I don't know anything about these kind of pumps, but people are always improving things. And if somebody has some in with a better, less temperature-sensitive seal, would we know that?

MEMBER BLEY: They are very different as you go from vendor to vendor. The ones he's talking about are the Westinghouse standard packages. Some of the others are quite different.

MR. LUPOLD: The ones I am most familiar with are the ones produced by Sulzer-Bingham.

MEMBER BLEY: But in the Westinghouse pumps?

MR. LUPOLD: Actually, the ones that I am used to are on the CE RCPs.

MEMBER BLEY: Okay. That is different.

MR. CARPENTER: Now for my afternoon

presentation.

CHAIR POWERS: Yes?

9. SGAP ITEMS 3.10

USING LABORATORY DATA FOR PREDICTING FIELD EXPERIENCE (CRACK INITIATION, CRACK GROWTH RATES

MR. CARPENTER: Staff closure of steam generator action plan item 3.10. The staff basically went back and took a look at the 3.10 language. What we determined was that this was not based on any specific ACRS-recommended action in 1740. And then based on that and also on the fact that the staff monitors plant operating experience through inspection processes and reviews of results of steam generator tube inspections, we determined that it was not appropriate for us to continue with that, that basically we thought 3.10 was mission creep and that we should really be focusing in on what we need to for the steam generator action plan.

The final item is that as we continue to look at this, we see that there are future actions that need to be accomplished. Based on operating experience, we will go back and reprioritize that depending on what we need to be doing.

Any questions?

CHAIR POWERS: Well, I think that we

hoped that there would be continuing activity to guide any laboratory studies on stress corrosion cracking based on observations in the field --

MR. CARPENTER: Of course.

CHAIR POWERS: -- and that there would be some systematic attempt to see if we were seeing anything weird happening in the field that merited laboratory study. I mean, is there a mechanism to look at the results of tube inspections to see if things are changing or getting weird or --

MR. CARPENTER: Well, we are continuing to do quite a bit of work on steam generators. So we have the various programs ongoing, including the steam generator TIP program, Tube Integrity Program. And that is a multinational program. So we are not looking just at U.S. activities. We are looking globally.

We have various other activities ongoing right now with steam generators. We have a fairly active operating experience, a program that goes out and looks at this.

So it's not like we're basically saying, "We're done. We're never going to look at this again." We are continuing to look forward as to what we need to be doing in steam generators.

And as we go forward and we see new items coming down based on operating experience, based on other research that's going on, either domestically or internationally, we will focus our research as appropriate.

MEMBER ARMIJO: So you are saying that recommendation is being addressed through these --

MR. CARPENTER: Through these other programs, right, but they weren't necessarily specific to the steam generator action plan.

CHAIR POWERS: Fair enough. I think the item was simply to do something and we didn't get -- just out of curiosity, how translatable is the experience from other countries to this country?

MR. CARPENTER: As far as what is going on?

CHAIR POWERS: Their steam generator tube integrity. I mean, other countries do different things on water chemistry, for instance.

MR. CARPENTER: Of course.

CHAIR POWERS: And some plants -- what is it, alloy-800. I mean, it's just not pertinent. And so I was just wondering. Is there translation ability? I mean, do the experiences of other countries have any use to us at all?

MR. CARPENTER: I would say yes. And the basis for that is the program that I just mentioned, the steam generator TIP program, it's a five-year program. We are about to start four. We are about to start number four.

So we have already got 15 years of international experience on this steam generator TIP program. And we consider it valuable enough that we are going for another five years worth of work.

CHAIR POWERS: Well, you told me you are doing it. I'm asking, did you get anything out of it? What are you getting out of it that is useful?

MR. KARWOSKI: This is Ken Karwoski from NRR. Maybe I can address that. Your specific question is, is operating experience from foreign countries applicable to the U.S.? And the answer is yes.

We continue to monitor foreign operating experience along with our own to make sure that there aren't any safety issues. We are following various issues. Not all of the experience from other countries is applicable to the United States, but we evaluate it if we have any questions on its applicability, engage the industry in order to make that determination.

MEMBER SHACK: Just as an example, you know, the CANDU people have different materials in their steam generator, but we all get together as part of the TIP program to talk about water chemistry because the concentration --

CHAIR POWERS: This is all because you want to take a vacation in some salubrious locale like -- these are metallurgists. These are metallurgists. They're very, very strange people.

MR. CARPENTER: It also turns out that their most common problem is probably fretting wear-type problems, which is now becoming -- you know, now that we have essentially eliminated stress corrosion cracking, we find these things still wear and still fret and still fatigue. So we have a lot in common, even if we have different materials. With the Koreans, of course, we have the same materials.

International experience certainly in many ways is applicable, even if --

CHAIR POWERS: Boy, I am having a hard time finding out why. I mean, fretting, okay, granted, you can find out from the foreigners you don't want a lot of foreign materials in your steam generator. I actually knew that beforehand.

MR. CARPENTER: Techniques for detecting

them and analyzing them.

CHAIR POWERS: Ah. Now, that is where we are getting someplace. That is the kind of information. That sounds good. Now, I'll bet the Koreans have nifty ways or nifty ideas.

MR. CARPENTER: The Canadians probably expend the most effort looking at that problem.

CHAIR POWERS: They don't have anything else to do in Chalk River. I'm going to hear about this.

MR. CARPENTER: I've been to Chalk River. I agree.

(Laughter.)

CHAIR POWERS: Okay. Well, that is probably for pursuit on a different venue and particular pursuit in the research report. I was just curious on that subject. It seems plausible to me.

Are there any other questions in the blacksmithing area here?

(No response.)

CHAIR POWERS: Okay. I think we are scheduled to take another 15-minute break here, aren't we? So why don't we break until 3:00 o'clock.

(Whereupon, the foregoing matter went off the record at 2:43 p.m. and went back on the record at 3:01 p.m.)

CHAIR POWERS: We have the esteemed Bob Palla to talk to us, who has been an unfailing source of information and detailed insights in this particular area for well over a decade. How about that for an introduction?

MR. PALLA: Is that enough?

CHAIR POWERS: Thirty or 40 after the --

MR. PALLA: I had better stop here.

10. SGAP ITEMS 3.1.K

PROBABILITY SG TUBE FAILURES BY

SG DEPRESSURIZATION EVENTS

MR. PALLA: My name is Bob Palla. I'm with the Probabilistic Risk Assessment Branch in Division of Risk Assessment, NRR. I am going to be talking to you about four tasks in the action plan that I inherited from a senior staff member, who worked them for several years. They were basically deferred in around 2005 and not actively completed as originally intended.

So, as I am going to describe, as part of the steam generator action plan closeout, we took a closer look at what the original intent of these items was, looked at progress made in other related items under the action plan, and also looked at what we thought were reasonable expectations for future work.

Between the work that was completed in other areas and the user need that I am going to discuss at the end of my presentation, we think that the items that I will be discussing here, the intent has been made, and that these can be closed.

I did not number the slides. My apology. So please don't shuffle them up.

CHAIR POWERS: Leave us just totally lost. Our rules are very explicit on this matter.

MR. PALLA: So I could skip to the last one. The first task I am going to be discussing is --

CHAIR POWERS: Well, what this means, Bob, is that you can't possibly use more than ten slides because of my limited ability to count.

MR. PALLA: The first task is numbered 3.1k. The task called for using information developed in tasks 3.1a through 3.1j to evaluate the conditional probabilities of multiple tube failures for appropriate scenarios in risk assessments for steam generator tube alternate repair criteria.

In the way of background, tasks 3.1a

through j addressed a number of physical processes the DPO writer asserted could cause steam generator tubes to open up and leak, specifically additional tube leakage or ruptures from the growth of existing cracks resulting from dynamic loads.

All of these processes were applicable to steam generator tubes in the free span except for concerns involving tube support plate movement during large blowdown loads, such as main steam line breaks. Tasks 3.1a through 3.1j were completed in the 2002 to 2004 time period.

The conclusion of this work was that the dynamic loads from the steam line break are low and do not affect the structural integrity of the tubes or lead to additional leakage or ruptures beyond what would be determined using differential pressure loads alone.

The completion of each task was documented in a separate memorandum. And I have listed here on this first slide the nature of the work that was carried out under each task.

Now, for each one of these within ADAMS is a separate closeout memo that makes references to all of the supporting documentation. I wasn't planning to reiterate it here, especially because if you look at the next slide, the ACRS concluded that the analyses of steam line break had been completed and that this action plan item 3.1 is closed.

Now, the only caveat there is that there wasn't any real discussion of 3.1k. So this was bringing up the rear. Just we wanted to bring this item up here and just make clear what was going on with regard to this.

The objective of 3.1k was calculate the leakage from existing steam generator flaws under differential pressure loads alone for a design basis steam line break.

The plan was to express this in the form of a probability distribution for total steam generator leak rate from the population of flawed tubes.

The planned approach was to develop steam generator leakage probability distribution based on the research-developed steam generator flaw information for flaws in the free span.

We in conjunction would use formulas for predicting the occurrence of bursts and leaks in the associated leak areas. And then, finally, we were planning to do RELAP5 calculations, providing realistic flow rates through the leak areas associated with the various bursts and leaks.

This information was to be used to support the resolution of GSI-163, involving multiple steam generator tube leakage, in DBAs, the steam generator item 3.11 that the Committee heard about several months ago.

The results could also be used in a risk assessment to determine the effects of steam generator the degradation in the risk from steam generator blowdown events.

Work on this task was deferred to staff reassignment on other activities. And the need to complete this work was revisited as part of this action plan closeout, taking into account the results from preceding tasks and the results -- progress made towards resolving GSI-1630.

Now, in looking back at the 3.1 subtasks -- each one of those related issues, regarding vibration, displacements, jets, impingement on adjacent tubes. They were systematically closed out. And at the end, as indicated, their conclusion was that they would not prorogate into multiple tube ruptures. This was addressed and agreed upon with the ACRS.

So, in essence, the concern didn't

materialize. In addition, as described in the NRR technical report on resolution of GSI-163, the industry has adopted a performance-based technical specification.

And the requirements that are in place at all U.S. reactors as part of that initiative in I-9706 would provide reasonable assurance that the potential for one or more ruptures or the equivalent leakage for multiple tues under normal conditions or DBAs would be well within what we assumed in previous risk studies and that the leakage from one or multiple tubes under DBAs would be limited to very small amounts consistent with the applicable regulations for off-site and control room dose.

So, in essence, the performance-based tech specs and the way that those have been implemented provide added assurance that the likelihoods would be small.

And, finally, the need for the calculation was diminished by the fact that most plants have installed replacement generators with more corrosion-resistant materials. This has resulted in a lower number of flawed tubes being left in services and also fewer proposals to increase the amount of leakage would be allowed under DBA events.

CHAIR POWERS: Many plants have chosen different materials, but not all plants have chosen --

MR. PALLA: Not all plants. There is a constantly shrinking set of remaining plants that have the 600-alloy.

CHAIR POWERS: Do you know what fraction of the plants now has the 600-alloy?

MR. PALLA: If you want to get it right --

CHAIR POWERS: Somebody does.

MR. KARWOSKI: This is Ken Karwoski from NRR.

There are 69 PWRs in the U.S. Of those, 42 have 690 material. Seventeen have thermally treated 600, which is a little more corrosion-resistant than the mill-annealed 600. There are three of those that we plan to replace this fall and basically one a year from then on out until they are all replaced. So 10 out of 69 have the --

CHAIR POWERS: So we really are getting down to the point that most will have 690.

MR. KARWOSKI: That's correct.

CHAIR POWERS: And so apathy can set in.

MR. KARWOSKI: Well, the plants with

thermally treated 600 have now started to exhibit cracking, but I share your concern with respect to the potential for apathy with more corrosion-resistant materials.

MR. PALLA: It didn't completely go away, but the target is getting smaller.

CHAIR POWERS: Sure.

MEMBER SIEBER: Yes. And the other thing is that the old 600 tube steam generators are the later models of that brand, where the chemistry was much better controlled. And so their susceptibility to degradation is lower than the earlier steam generators were.

So I don't know. I don't think you can quantify that. On the other hand, you can look at tube-plugging rates for those remaining steam generators. And they're quite a bit smaller than what had been experienced in the past.

MR. PALLA: So our conclusion is that this work wasn't really needed as we had originally planned.

CHAIR POWERS: I think the original thought was when we looked at the various tasks that were being done to get propagation from one tube to the next in the event of failure or mechanisms to get multiple failures, with multiple failures, the concern would probably not be one or two tubes but more in the eight category. Certainly 12, we knew we were in desperate straits at 12.

We were not seeing Korean-spirit mechanisms that prompt led to those multiple failures. I think we were interested in seeing well, you know, you don't see it in these particular tests and investigations.

We are looking for a feel on the probabilities that you get, understanding that they were likely to be small. And so what you are saying is we're not going to get that.

MR. PALLA: You're not going to get that. You're going to get assurance that the leak rate would be small based on what we know, but these other mechanisms have been --

CHAIR POWERS: We can live with this. Okay.

MR. PALLA: Okay.

CHAIR POWERS: I understand what the status is.

SGAP ITEMS 3.4.J-K

SG TUBE LEAKAGE RATES; PRA FOR EVALUATING.

SG TUBE INTEGRITY REQUIREMENTS

MR. PALLA: Okay. The next task I will be discussing is task 3.4j. This task called for putting the information developed in task 3.4i into a probability distribution for the rate of tube leakage during severe accident sequences based on the measured and regulated parameters for alternate repair criteria applied to flaws in restricted places.

As background, 3.4i provided analytical predictions of flaw opening areas and leak rates from axial and circumferential cracks under the tube support plate during steam line breaks and severe accidents. This work was performed by Saurin Majumdar at Argonne.

3.4i was closed by issuance of a technical letter report in May of 2004. That described analyses for predicting leak rates of degraded tubes in restricted areas under DBA and severe accident conditions.

The leak rate models presented in the Argonne report provide upper bound leak rates assuming no crevice deposits are present. The report also describes Argonne's evaluation of test results that show that crevice deposits could reduce the leak rates by as much as a factor of 1,000 compared to leak rates with no deposits.

Although the objective of 3.4j was to put this information into a probability distribution for the rate of tube leakage during severe accident sequences, this work was effectively completed as part of action plan item 3.5, which you will be hearing about tomorrow.

Under task 3.5, the Office of Research, Sandia National Labs, and SAIC with developing a methodology to integrating the results of PRA with results from supporting thermal hydraulic and materials engineering analyses.

The results of this research effort will be discussed tomorrow, as I said, but I just wanted to summarize the nature of their conclusions here.

As described in a report issued February 2008 -- it's a contractor report on task 3.5 -- SAIC developed an Excel spreadsheet to compute the probability of tube failure during an accident using steam generator flaw distribution in the pressure temperature history for an accident.

Uncertainty distributions for key model inputs were developed using Excel add-in called Crystal Ball. Flaw distributions were provided for six defect types, including circumferential and axial outer diameter stress corrosion cracking at tube support plates.

The circumferential cracks considered in the analysis were located at either the top of the tube sheet or at the tube support plate and were expected to be surrounded by a buildup of sludge.

The Argonne models were used to calculate the growth of each crack during the transient. Now, example calculations were performed. And in these calculations, the maximum crack opening displacement for circumferential flaws was set to one millimeter based on consideration of the test at Argonne, which showed that sludge deposits would significantly restrict the flow through the flaw. Nevertheless, the model would provide you a mechanism for looking at alternative assumptions.

The conclusion based on this is that the effort performed under 3.5 has achieved the intent of ask 3.4j and that we can close the 3.4j task.

Next task is task 3.4k. This task calls for integrating information provided by tasks 3.4a through 3.4j as well as task 3.5 to address ACRS criticisms of risk assessments for alternate repair criteria that go beyond the scope and criteria of generic letter 95-05 as well as dealing with other steam generator tube integrity and licensing issues.

As background, action plan item 3.4 addresses ACRS comments on previous risk assessments by developing a better understanding of the reactor cooling system conditions in corresponding component behavior in severe accident sequences in which the RCS remains pressurized.

In the previous presentations, you have heard about the various subitems under 3.4, the thermal hydraulic work discussed by Chris Boyd and the structural analyses described by Gene Carpenter as well as the assessment of leak rates for degraded tubes in restricted areas. That's the 3.4 task that we are basically saying is not really needed. Well, it's covered by the other item.

The objective of task 3.4k was to integrate information provided by the above tasks as well as task 3.5 to address the ACRS comments.

Now, I have divided the task into, really, two broad areas. One, I'll refer to it as a specific concern. And then on the next slide, I am talking about a broader concern.

The specific concern on this item was specific, actually, to South Texas project, steam generators that had stainless steel drilled hole tube support plates. And this was the only steam generator of this type in the U.S. This generator has been replaced since then.

Now, because the tube support plates were stainless, instead of carbon, they did not corrode and the tubes were not clamped in place or dented. However, the crevices did accumulate deposits, which caused steam generator tube cracking. And in a depressurization event, the tube support plates could move and expose these flaws, the cracks.

To limit the displacement of the tube support plates, several tubes were expanded at various tube support plate elevations. And this expansion essentially locked the tube support plates in place and dealt with the concern about the movement of the plate.

When the flaw is located adjacent to tube support plate and the flaw burst pressure is exceeded, the tube will not burst because of the physical restriction of the tube support plate, but the flaw can open up, resulting in increased leakage.

The staff had calculated an estimated leak rate for this constrained opening flaw to be about five gpm per burst flaw in that region.

MEMBER BLEY: Mixing a few things up in

my head. The generators in which they expanded the tubes, they did that for all of the tubes?

MR. PALLA: They expanded certain tubes. I don't believe it was all of them.

MEMBER BLEY: Ones where they thought they had flawed, where they found --

MR. PALLA: No, no. I think it was to restrict the motion of the plates.

MEMBER BLEY: Okay. Just to lock the plates to --

CHAIR POWERS: This tube support lift problem was --

MEMBER BLEY: Okay. And are these the generators that you said had been replaced?

MR. PALLA: Yes.

MEMBER BLEY: Okay. So they're no longer there?

MR. PALLA: No. So this was another reason for not pushing too hard on this.

MEMBER BLEY: Fair enough.

MR. PALLA: But we did estimate the leak rate. We realize we could have built this into a PRA model somewhere, but we didn't take that step because of the fact these generators aren't there anymore. We don't expect that there would be any more of them.

So that was the specific concern.

CHAIR POWERS: The only thing that would cause you to think farther that it's not obvious to me that the newer generators self-lock. And then you go back to having this support plate wafting around during a blowdown and things like that.

MR. PALLA: It would be an issue in a different design.

CHAIR POWERS: Some of them were going to have drilled hole support plates.

MR. PALLA: Okay. The broader concern involved other steam generator tube integrity and licensing issues related to flaws in the free-span of the tubes and I think in general the ability to perform severe accident calculations in a technically defensible manner. I think that may be kind of a recurring theme with a number of the comments, just the defensibility of these analyses.

Now, here is where I shift the shell in a pea game, task 3.5. Well, it took a long time. I was trying to separate this issue out from task 3.5 for a long time. These essentially would seem to be doing the same thing. So my claim is that task 3.5 was specifically intended to address that concern.

If you look at task 3.5a through d, they involve development of an integrated framework for assessing the risk of high-temperature, high-pressure accident scenarios of interest, including the treatment of uncertainty in operator actions and example applications of the methodology.

Tasks 3.5e through 3.5g involve extension of the methodology, to include treatment of combustion engineering plants, external events, events at low power, and secondary depressurization events, details to be revealed tomorrow unless you read ahead and reviewed some of the voluminous documentation there.

Based on the results of this example calculation performed under task 3.5, research concluded that the contribution of consequential steam generator tube rupture events to the overall containment bypass frequency is lower than or at the same order of magnitude as containment bypass frequency due to other internal events. So we're kind of right at about the level of bypass from interfacing system LOCAs.

MEMBER STETKAR: That means without considering it, you might be a factor of two too low, right?

MR. PALLA: Yes. And, in fact, I think

the more we look at some of the more traditional containment bypass and early failure modes, things that contribute to LERF, you look at DCH. The closer you look, the smaller it gets to the point that you say it's not really much of a likelihood of failure, steam explosion, large hydrogen burn.

One by one these things, you know, you can reject. They will not significantly contribute. So what you might have, actually --

MEMBER STETKAR: This might be more than --

MR. PALLA: -- inconsequential steam generator tube rupture could actually be the dominant contributor. Now, I'm going to mention something later that --

MEMBER STETKAR: The largest don't --

MR. PALLA: -- something you all heard about already, the SOARCA insights. If you actually have a subsequent RCS piping failure, you are going to have a little blip of a release. And it's going to depressurize the -- you are not going to have a driving force. You are going to reduce the magnitude, the quantity of materials release to the point that you are not going to have a large release. It might be early, but it won't be large. So it won't be LERF to looking at that.

MEMBER BLEY: Bob, give me a head start before tomorrow. Depending on where I look, one thing is not completely clear. There were seven subtasks under 3.5, I think.

MR. PALLA: Okay.

MEMBER BLEY: In some places --

MR. PALLA: Well, Selim is going to walk you through that tomorrow.

MEMBER BLEY: Are they all completed or are some of those still ongoing?

MR. PALLA: I think they're all completed, yes.

MEMBER BLEY: All completed? That's what wasn't completely clear to me.

MR. PALLA: Yes.

MEMBER BLEY: So the reports we have cover all of the --

MR. PALLA: That will be the claim, yes.

MEMBER BLEY: Okay. That's where I wanted to know where to start.

MR. PALLA: Okay.

MEMBER SHACK: Are they planning additional work?

MR. PALLA: Yes. Well, because I think

that the intent of the plan has been met, but I think honestly we would still like to know more about some of these things.

And I am going to discuss. In the very last slide is some discussion on that.

CHAIR POWERS: I mean, no research program ever gets finally resolved. I mean, it is kind of a definition of a research program.

MR. PALLA: We have some stimulus money here to --

CHAIR POWERS: That is really not the issue that we are addressing. The question is, have we done enough that we can move it into the regular research program to address those issues? And have we gotten adequate understanding for the purposes of the action plan?

MR. PALLA: Okay. And so, in light of the conclusions that research was producing about the magnitude of the consequential tube rupture, its contribution to overall risk, they recommended that plant PRAs continue to evaluate consequential steam generator tube ruptures on a plant-specific basis in accordance with the existing PRA standard.

The methods and results developed through the RES activities provide valuable insights into the risk significance of consequential ruptures as well as a foundation from which the risk implications of future tube integrity and licensing issues might be assessed. So it is really more of a starting point than an ending point from the point of view of future assessments.

Although additional research related to consequential ruptures is planned, the work completed has achieved the intent of action plan item 3.4k.

Now, tomorrow on 3.5, I guess you can deliberate on whether the full intent of 3.5 has been met, but we consider that what we have learned from the 3.5 work plus what we intend to do further as part of an action steam generator user need would put us in a much better position for any future risk assessments that might be needed.

MEMBER BLEY: Let me sneak in one more of those questions just to help me prepare for tomorrow if I prepare anymore. We had a report, "The Risk Assessment of Consequential Steam Generator Tube Ruptures" from back in March, which kind of integrated the work that had been done up to that point. But a lot more has been done since then.

I don't think you have put together a report that integrates the things that have been done since that point. And the question is, am I wrong? Is there something that --

MR. PALLA: Which report? What is that report?

MEMBER BLEY: It's actually called "The Risk Assessment," '09, March '09.

MR. BEAULIEU: Yes. That is just a few months ago. There is nothing else that has been done since then, right?

MEMBER BLEY: There it said only the first three tasks under 3.5 had been completed. The others would be completed later. That is why I am asking, if there is something that has been written to integrate the ones that were completed later.

MR. BEAULIEU: Okay. That will be --

MEMBER BLEY: I will wait until tomorrow to get to the details.

MR. BEAULIEU: That will be covered by 3.12, which will be covered in a few minutes.

MEMBER BLEY: Okay.

MR. PALLA: There is a contractor report. And then there is a more recent staff report. I think you must be talking -- that's a Research staff report?

MEMBER BLEY: Yes.

MR. BEAULIEU: It doesn't have a number on it. It says it's March.

MEMBER BLEY: It says "March 16th."

MR. BEAULIEU: March 2009.

MR. PALLA: Okay.

MR. BEAULIEU: RES.

MEMBER ARMIJO: Before you go, the previous slide, you said based on -- you decided not to issue this proposed reg guide and, instead, you endorsed NEI 97-06.

MR. PALLA: I think you are one step ahead.

MEMBER ARMIJO: Oh, am I ahead?

MR. PALLA: I should catch up to you.

MEMBER ARMIJO: I'm usually behind. Sorry. I'll wait.

MR. PALLA: Well, if we're ready, I'll just go on to that

MEMBER ARMIJO: Okay. The question is, how do you actually do that, endorse industry document? What is the process that you use that effectively makes it an NRC position?

MR. PALLA: Okay. Well, Ken can clarify it, but if it's like we did on severe accident management, essentially NEI has a process by which they bind the industry. If they get an 80 percent approval on a particular initiative, then they commit to it as an industry.

We did that on severe accident management. I suspect it was similar on this particular initiative.

MR. KARWOSKI: This is Ken Karwoski again. In the case of NEI 97-06, as Bob indicated, the industry decided to voluntarily adopt it with a greater than 80 percent load, I think, of --

MEMBER ARMIJO: So, from that standpoint, you were happy that it would be done?

MR. KARWOSKI: Well, one more piece. What is in NEI 97-06 is basically a performance criterion that we have adopted in all the technical specifications at the plants. So all of the technical specifications at the plants have been changed to basically reflect the performance-based approached that NEI 97-06 basically proposes. So we've essentially adopted that in the technical specifications.

Now, with that said, NEI 97-06 has guidelines, which we have not endorsed, but that is usually what plants implement in order to ensure these performance criteria that are in the technical specifications are met. And those approaches have been successful.

MEMBER ARMIJO: Okay. Thank you.

MR. PALLA: Okay. So this particular task called for reviewing the risk insights developed under task 3.5 and assessing the need for completing what was called DG-1073. It was labeled "Plant-Specific Risk-Informed Decision-Making for Induced Steam Generator Tube Rupture." Now, when you go back and you try to find that document, it doesn't exist, but the plan was to develop it.

Now, the plan essentially came about -- as I believe Dave may have mentioned, there was originally a plan to do a rulemaking on steam generators. Then there was I guess a decision to transition, instead of a rule, go with a compliance-based generic letter. And as part of that generic letter, the plan was to have a pair of reg guides. And one of these was to be this risk-informed decision-making reg guide.

Now, given that the decision was made to, instead, adopt the industry initiative under 97-06, work on DG-1073 was never completed. I'll be you are surprised about that.

CHAIR POWERS: Appalled.

MR. PALLA: But there was no driver. The industry initiative didn't require that guidance document in order to implement it.

12. SGAP ITEM 3.12

NRR USER NEED TO RES FOR FUTURE WORK

MR. PALLA: So we re-looked at this task 3.12. Consistent with that, we further assessed the need for regulatory guidance on induced ruptures given the risk insights under steam generator action plan 3.5.

Based on that assessment, we conclude that additional guidance and tools are indeed needed to support future assessments of the steam generator tube ruptures. And the rationale is summarized on the next slide.

First, task 3.5, as well as the numerous other studies, performed by NRC and industry over the last decade have not generically dispositioned the issue of induced rupture or substantially reduce the inherent uncertainties in the analysis of these events.

The final report, as I mentioned, determined that the contribution from these events to the overall containment bypass frequency could be at the same order of magnitude as that from other causes.

So it's not like we have been able to drive that thing down into the grass and be able to walk away and declare victory. It's very much an important contributor to bypass.

CHAIR POWERS: You mentioned the inherent uncertainties in these processes, which are not small to my mind. Earlier we discussed at some length the system's thermal hydraulics and the more detailed CFD mixing-type calculations.

Those particular studies seem singularly ripe for what I would call a rigorous uncertainty analysis, but the investigators resisted doing that and relied, instead, on a reasonable range of sensitivity studies but, nevertheless, sensitivity studies which inherently suffer from being typically one at a time variations or small set variations.

At the same time, there was a discussion of whether the uncertainties were dominated by thermal hydraulics or dominated by flaw distributions and the like, another issue that would seem to me readily resolved by a rigorous uncertainty analysis, which apparently has now been done.

So now I hear you're saying, "Gee, we would like to understand these uncertainties," and you have got investigators that seem to be unenthusiastic about doing rigorous uncertainty. And are you looking for rigorous uncertainty analysis? And can you get that?

MR. PALLA: Well, I don't know how much rigor has to go into the analysis of uncertainties of each of these pieces. I mean, there are enough uncertainties to go around, whether it is the flaw distributions, the thermal hydraulics, the PRA itself.

Where this comes together -- and you will hear about it tomorrow -- is this probabilistic code that basically takes as input the thermal hydraulic, the pressure temperature histories, which could be fed in as the point estimate plots or one could ascribe uncertainty bounds to that, however that might best be done.

That gets fed into the probabilistic code as well as failure time estimates for the surge line and for the hot leg, which could be point estimates or, better yet, point estimates with some distribution, some kind of range in --

CHAIR POWERS: These are not an enormously difficult thing to do nowadays.

MR. PALLA: But where it comes together

and where it might be more tractable is in the context of the probabilistic code, where you could ascribe uncertainties to the various pieces.

And the flaw information is also going into that code. You turn the crank, and it is convoluting all of these different distributions. It is sampling. We will be discussing that.

But that is perhaps one -- I think you would still want to do it for the structural analyses. You'll want to have some not just the point estimate but some kind of a measure of uncertainty as well as the TH in --

MEMBER STETKAR: Do we have any description or documentation of this wonderful code?

MR. PALLA: The probabilistic code or the --

MEMBER STETKAR: Yes.

MR. PALLA: Yes. There is some form of documentation. I don't know if it's been given to the Committee yet or not.

MEMBER BLEY: I think it is one of the ML documents that are in the --

MEMBER STETKAR: Is it? Okay. I missed it.

MEMBER BLEY: Yes. There's one in there

that deals with something about --

CHAIR POWERS: Somehow labeling all of the documents on a .pdf with ML numbers is not the most useful thing I've ever seen.

MR. PALLA: There is a brief description of it in the contractor report of 2008, I believe the date is.

Yes. The answer is yes. We think that more should be done in the way of uncertainty and expect that it will be done still.

CHAIR POWERS: You have to admit that when I have engaged in what I would call rigorous parametric uncertainty analysis, I have found I have been unfailingly wrong in my intuition on how things couple together.

I mean, I have an exact batting average of the zero on that. The things that I think will cause things to go up, invariably cause them to go down, and vice versa, because of very strange couplings that arise among multiple varying phenomena, my experience with this has nothing to do with steam generators. But I often find not only the coupling among phenomena but the ability to rank, which are the most important uncertainties, which are less than important certainties, to do unfailing interests.

MEMBER BANERJEE: This code you are talking about, let me just try to understand. With regard to thermal hydraulics, the primary uncertainty in my mind is whether the hot leg will fail at all.

I mean, if it does, even if it fails within a few minutes before or after the steam generator pops, the tube pops, then you have got an alternate path into containment. If it doesn't fail, then it is a very different scenario. And I think the uncertainty is really -- I don't know how you would quantify that without actually doing thermal hydraulics calculations more in the best estimate sense with certainty, as we do for things like peak clad temperature.

The methodology is very clearly laid out. It's called a CSAU methodology, even if it is not applied in its full glory to this problem. Nonetheless, we have to determine whether the hot leg would fail at all or not and what sort of assumptions have gone in. And if we can find conditions which are sort of reasonable, they won't fail. It seems it's on or off in some ways. It's a big change.

So is this code able to handle this sort of thing?

MR. PALLA: Well, the way that this code would operate, essentially it's looking at the distribution of the predicted failure times for the tubes versus the other --

MEMBER BANERJEE: No failure of the other at all, right?

MR. PALLA: Like let's say you did a stand-alone finite element analysis of the hot leg. You would feed it the TH results from your latest systems-level calculation. You will feed that same result. Well, you'll feed it the results for the hot leg.

Now, you will peel out of the same run the results for the tubes in various parts, portions of the steam generators. You will have a pressure temperature history.

And you will basically look at the probability or the timing of failure, really, is calculated for the structures, calculated for the tubes. And basically you are looking at the distributions. And you are trying to figure out what is the likelihood that this goes first versus the other component goes first.

MEMBER BANERJEE: The way I see it, these guys have run various scenarios, right? Without actually saying, is it likely or not, let's say one scenario is a scenario where a loop seal clears. Clearly if a loop seal clears, you are going to just fail a lot of tubes and the hot leg is going to stay intact and there is going to be a bypass of containment.

Now, they have done this. Now, we don't know what is in there with this clearing or not. How are you going to assign a probability to this?

MR. PALLA: Well, I think you would run a sequence with clearing. You would run a sequence without clearing. The results are applicable if those conditions are applicable.

MEMBER BANERJEE: It seems that the whole thing seems to depend on whether or how you are going to assign a probability to --

MR. PALLA: Yes. Well, I mean, you are going to have a spectrum of sequences, each with its own pressure temperature loading that is going to be seen at the various places within the RCS. Each of those sequences has a likelihood.

I guess the challenge is to try to establish what is a reasonable set of calculations to represent the bulk of the sequences that are of concern.

You don't want to have to analyze thousands of sequences, but maybe if you can analyze the sequences that provide reasonable representation of the spectrum, maybe there are a dozen sequences that -- I mean, we are still talking basically things that go to core damage at high RCS pressure, high primary-side pressure, depressurized secondary side.

Now, there are some complications. There might be some of these sequences that have stuck-open relief valves that you could have partially depressurized if you had a RCP seal LOCA. So there are some variations on that.

In NUREG-1570, for example, was the staff's first cut at developing a risk perspective on this. A relatively limited number of sequences was run there to try to represent the range of conditions that the RCS might --

MEMBER BANERJEE: I am going to sort of leave this to my PRA colleagues, but they can understand my --

MR. PALLA: You will have a better chance tomorrow to bore in on it, I think.

CHAIR POWERS: Please continue.

MR. PALLA: Okay. The second bullet there is that although the work to date may be sufficient to resolve the technical concerns related to task 3.5, in our view certain limitations of the work would restrict its usefulness in supporting future risk assessments.

For example, the risk analyses did not account for updated flaw distributions or the results from the most recent thermal hydraulic analyses. The results that are in the most recent probabilistic report basically go back to flaw distributions as they existed probably, really, around 1990. The thermal hydraulics is probably early 2000.

So it doesn't have the benefit of everything we have learned on CFD and more precise -- well, maybe it's never precise but better thermal hydraulics.

I believe that the separate structural analyses would predict somewhat earlier hot leg failure times. I don't believe that those earlier times got plugged into the probabilistic calculations either.

So we have got some basic tools in various portions of the organization. We have developed new information. But we haven't really plugged it in and turned the crank and tried to see, well, what is the picture today?

Because if you have replaced a lot of these generators and if you have got a performance-based tech spec that would basically give you more assurance that you are removing from service some of these more seriously flawed tubes, it doesn't make a whole lot of sense to use a flaw distribution that was developed before you even had that kind of control of those tubes.

The third item is the idea that the PRA standard identifies the need to address induced steam generator tube rupture as a supporting requirement for a quality PRA.

It refers to NUREG-1570 as I guess a reference document that could be used to help structure the event tree, quantify the probability, conditional failure probability.

But it's dated. And I think the picture could be quite different if you accounted for the current flaw distributions; if it's replacement generators, totally different picture perhaps, updated thermal hydraulics, the whole nine yards.

It seems more appropriate to basically use some of the new information to bring it all together and to kind of take another snapshot of what we think the situation is.

I wanted to just mention two other things. The industry has developed some methodologies to be used. The staff had developed 1570, but there is a Westinghouse topical report on simplified level 2 event trees that has attempted to do a better job of addressing this induced tube rupture.

There is an EPRI methodology that is out there. And based on anecdotal information, licensees are incorporating these consequential tube rupture models into the PRA. And the intent of the model developers was that these models would meet PRA capability category 2.

So if a utility person is aware that they're implementing this model, they think it meets category 2, they are peer reviewers on someone else's model, they see they are doing it the same way, it must be good. It all meets category 2.

It would pass the peer review test. But, yet, we have not really looked at those numbers. Of course, we don't have much to compare them against, but what I am going to explain on the user need is one of the items we would ask for is kind of a cross-comparison.

Number one, we would turn the crank based on our best estimates, our thermal hydraulic analyses and flaw distributions, et cetera, but then look across at these other methods to see are we in the same range, what are reasonable. Maybe it's --

MEMBER SHACK: It does seem somewhat dated, too.

MR. PALLA: They are.

MEMBER SHACK: I mean, I think the EPRI report is 2002.

MR. PALLA: I'm sure we would prefer our numbers over those numbers if they were different. Now, sometimes you study these things and you go around all the way, 360, and you're back at where you started. You are on a much better technical basis. But you ended up not much different than you were before.

So, anyway, yes, peer review.

CHAIR POWERS: I mean, I think I am very sympathetic with your view that we really need to pull all of this together in a fashion that it can be used, both in probabilistic and non-probabilistic applications. And you're right. Until that is done, the peer review is liable to be quixotic.

Well, enough said. Let's continue on.

MR. PALLA: Okay. Well, based on these

observations on that aforementioned slide there, the NRC staff concludes that additional guidance and tools are still needed to support future risk assessments.

The guidance would address acceptable approaches for the modeling and quantification of consequential tube ruptures in future NRC and/or licensee risk models. The guidance would also support NRC staff assessments of the risk implications of new licensee-proposed alternate repair criteria, if any. And development of the guidance is part of the user need letter that we currently have in concurrence. It's at the office level right now.

CHAIR POWERS: Is there any realistic expectation the licensees are going to try to propose new alternate repair criteria?

MR. KARWOSKI: Yes.

CHAIR POWERS: Okay. Short answer. That's all I needed to know. Thank you. Admirably succinct.

MR. PALLA: Okay. Now, ready for this one. With regard to task 3.12, our conclusion that additional guidance and tools are still needed and the decision to proceed with this development effort has achieved the intent of action plan item 3.12 and, therefore, provides a sufficient basis for closing 3.12.

3.12 basically says, do we or don't we need it? The answer is yes, we need it. Here is what we think it --

CHAIR POWERS: Right.

MR. PALLA: And I have thrown this in just for general information. This is a user need that has been coordinated with our colleagues in Research. It's across three different divisions in Research and two different divisions in NRR.

I have identified four major areas that are part of that user need. I don't know if you want to go into them or hold this off for some future discussion.

I believe that the document was provided to the Committee. You have an ML number for where the draft is. Again, it's at the office level, directors, for signature, more details to be worked out as it goes.

This thing is extremely complicated. I think the challenge will be to keep it manageable and to keep it from growing.

CHAIR POWERS: Yes. There are two

observations here. One is that yes, this is one of those marvelous issues that involves the confluence of both thermal hydraulics materials and PRA.

The second observation is the agency is pretty good at managing those kind of multidisciplinary activities.

MR. PALLA: That is a challenge, though.

CHAIR POWERS: It is.

MR. PALLA: You are cutting across offices. And you are cutting across divisions. But yes, to succeed, you are going to have to communicate very well across organizational boundaries.

So, really, what I have identified there is additional thermal hydraulics analysis. The work would focus, really, on the CE plant, although there is some thinking about whether replacement generators from Westinghouse plants, the lower plenum configuration, is different than the original ones and could be somewhat more like CE in terms of the proximity, the orientation of the hot leg and how it enters the lower plenum. So that is something to be discussed and to pin down some more, but at this point, we know that we're very light on CE thermal hydraulic analysis.

We want to look at this concept of the

observation, really, from the SOARCA work that you have heard about that, even if you fail multiple tubes, the hot leg failure or some other piping failure shortly thereafter could relieve the pressure and the driving force and render a bypass, actually a non-LERF. So it may or may not occur for CE. So if it does seem to be the case there as well, we would want to see how sensitive that result is to major input assumptions.

Another area that is part of that first major bullet is in-core instrument tube failures. There is some interest in these analyses that stems from recent NRC-sponsored analyses as well as some industry-sponsored work.

Fauske and Associates had looked closer at TMI. They believe that the failure of the in-core instrument tube may be not as significant in TMI as it might be in Westinghouse plants that have larger diameter tubes, but this could disrupt the natural circulation flows.

So, as a result, it could affect the timing and maybe the likelihood of the consequential tube rupture. So we flagged that for further analysis. Some work is already ongoing there, but we wanted to indicate support for that.

The second major area, updated flaw distributions and RCS structural analyses, the idea here was the work that you'll hear tomorrow was predicated on quite old distributions. It's our thinking that the performance-based tech spec would result in reduced number of severely degraded flaws that are in service in a generator.

CHAIR POWERS: Let me ask a little bit about that because flaw distributions; whereas, they are the fundamental uncertainty in the universe, I understand this. But all uncertainties pale in comparison.

MR. PALLA: We have had some initial discussions.

CHAIR POWERS: Is it feasible to get flaw distributions that are of sufficiently reliability to make a difference.

MR. PALLA: Somehow we need to characterize what is out there. And it doesn't make much sense to use a distribution that is outdated. I don't know the best way to replace it.

Chris Boyd's presentation focused on the multiplier concept. I don't know if there is a way to basically take flaw information, map it into multipliers. There needs to be some kind of massaging done. But it is undoubtedly one of the key areas of uncertainty, not only the types of flaws and where they -- well, the types of flaws and where they occur is the key thing because they have to realistically occur coincident with the hottest tube.

So you're interested, really, in flaws that are prone to occurring in those particular regions. And things that are far removed from that zone of influence may be of less significance.

I think when you talk in flaw distributions, inherently there are uncertainties, well, not maybe so much uncertainty as physically where does this mechanism manifest itself.

But given that it does, I mean, do we know enough to preferentially say that these will be on the periphery versus the center or if they're somewhere above the tube sheet, where would they be?

So I think there would be a challenge to do that, but something that I think is worthy of pursuit there just because we think that that performance-based tech spec, had we done the kind of work that was done by Gorman in that original effort to develop that distribution, if you operated off of the generators that are in service today, it might be much different.

CHAIR POWERS: Is there any effort -- I mean, we have steam generators being replaced right and left. Do we have people looking at the extracted generators to understand these flaw distributions in greater detail?

MR. PALLA: I thought that we did, but --

MEMBER SHACK: No. You can never get the right steam generator, you know. I mean, in some ways, the problem may be simpler for this particular problem.

I mean, if you do an inspection, it would be possible to perhaps bound the kind of flaw distribution that would be reasonable to expect, rather than -- I think it is possible to bound these flaw distributions.

I think it is probably impossible to get an accurate flaw distribution. But on a plant-by-plant basis, as they do their inspections, you know, you have an idea of their inspection capability?

MR. PALLA: You can get actual results.

MEMBER SHACK: Yes, how typical that is is --

MR. PALLA: Well, it would be a plant-specific thing. You know, if the next outage they came up with a different distribution, it would be back to the drawing boards, but if you were sort of conservative about picking a flaw distribution that bounded that, something like you are going to do with PTS, I suspect.

MR. FULLER: This is Ed Fuller from NRO. I would like to comment based on my experience in the last job I had, when I was still at EPRI.

The industry does that all the time. In terms of their whole performance-based approach, they have developed guidelines to take inspection findings; apply probably of detection to these various flaws of various kinds; and then, in turn, put that into ways to calculate potential for bursting of the various tubes given the degree of degradation. They got something called the flaw handbook.

And one of the things that we also did when I was helping out EPRI, developing the steam generator tube integrity risk assessment reports, was to take some of those and apply the multipliers to some of the flaw distributions that were actually developed in some of the plants.

One case in point is ANL2 came in to try to get an extension of their running their very last cycle, and they had this kind of information. They provided it to the NRC.

The point I am trying to make is what Bob was saying a little while ago about this probabilistic code has an analogous approach that has already been developed by the industry, perhaps somewhat dated as well.

But, in principle, you can do this. It has been done, and it can be done better.

MR. PALLA: Okay?

CHAIR POWERS: Okay.

MR. PALLA: I guess the last item under the second bullet there is a finite element analysis of RCS components. I think the focus here would be on the hot leg, possibly the surge line, probably not reactor coolant pump seals.

The idea would be use the latest thermal hydraulic analyses in concert with finite element analysis, develop point estimates or distributions ideally to feed into the probabilistic code.

The third item deals with guidance and tools for future risk assessments. This would include developing generalized event trees, guidance on the treatment of critical operator actions and steam generator operating strategies, guidance on the use of probabilistic computer codes or screening techniques to quantify the probability of consequential tube rupture and consideration of subsequent RCS piping failures.

MEMBER STETKAR: Bob, I know we are going to talk about the PRA stuff tomorrow, but why do you feel there is a need for guidance on development of generalized event trees?

MR. PALLA: Well, I think that --

MEMBER STETKAR: Each plant seems to be somewhat different.

MR. PALLA: I was thinking more from a staff assessment point of view.

MEMBER STETKAR: Okay. So if you feed into the SPAR models, you mean?

MR. PALLA: Yes. If you look at 1570, I think it is pretty cumbersome.

MEMBER STETKAR: Yes.

MR. PALLA: If you look at NUREG/CR-6595, simplified containment event tree, it's overly simplified. I think that maybe the level of effort that's in maybe the topical report or the EPRI document is more appropriate.

CHAIR POWERS: Your point on simplified tools, maybe it's not simplified but less cumbersome.

Is that what you're looking for?

MR. PALLA: Or user-friendly and perhaps enable somebody that hasn't been working this for the last ten years to be able to actually pick it up and use it.

CHAIR POWERS: Okay. So it's not really simplified. It is --

MR. PALLA: I think you would have to simplify it in order for them to make it useable. But yes, I think user-friendly. And I think the more you know about something, at the end of the day, when you have developed all of this detail, you could usually fold it down. You're smart enough to figure out what is important. And that's what you put into the model.

CHAIR POWERS: It's surprising how difficult that is.

MR. PALLA: Well, you have to have --

CHAIR POWERS: To put your arms around it, the ability to simplify it down is gone.

MR. PALLA: The work would also include the use of the probabilistic computer code together with updated flaw distributions, updated thermal hydraulic results, and insights from the finite element analysis to derive updated conditional probability failures.

The thinking is for a set of sequences that might provide a reasonable range of the kind of conditions in the RCS that you would need to evaluate these conditions for.

I guess it depends on how detailed the event tree is, but the values that we have now in 1570, for example, could be quite different if you used the latest information.

So this would be an attempt to develop updated numbers and then to compare them to the information in these other documents to get a reality check. Where there are big differences, we will pursue that further to try to see if there is a right answer.

Do you have a question?

MEMBER BLEY: No. I am good.

MR. PALLA: Okay. This work would also include developing a draft reg guide. Hopefully it will end up being a final reg guide on the risk-informed decision-making related to consequential ruptures, as identified in 3.12. And there may be corresponding changes to the inspection manual chapter, things that are used for significance determination process.

MEMBER BLEY: Yes, I do. Earlier I indicated that we might have had the report that described that probabilistic computer code for putting all of this together.

We have some reports on, have some progress reports on, the development of the methodology. I don't know if that is what you are talking about or if there is a specific report that really lays out in detail how this code works. If that is true, we would like to see it.

MR. PALLA: The reports that I am thinking about -- and, Selim, maybe you can help him out when we finish up here. There is a contractor report that I thought was February 2008.

MEMBER BLEY: Okay. That's the one. Okay.

MEMBER STETKAR: I think Chris went to go find it.

MEMBER BLEY: Okay. We have an earlier version of that, not the 2008 version.

MEMBER STETKAR: If that's the one you're talking about, that's --

MR. PALLA: I think it's basically the 2005 report with a bunch of editing in it.

MEMBER STETKAR: Okay. We have the 2005.

MR. PALLA: The results are not different, but the TH calcs are the same. Everything, the calculations, are the same.

The final item, I guess it kind of echoes what Dr. Powers had mentioned about the desire to have some kind of a document that tries to encapsulate the key items of research that have been done.

Now, we may not be thinking of it to the same level of detail that Dr. Powers did, but it was our thought that there has been a lot of work.

It's documented in a lot of different places. If there was a document that could provide kind of the unified discussion of this at a reasonable level of detail with -- you know, you've got to have some fancy CFD figures in there and things like that.

CHAIR POWERS: I think the presentations have certainly refined my thinking in that. And I no longer think it can be done in a single document. I think there have to be two: one of a summary nature and one providing a summary of the details.

MR. PALLA: Appendix.

CHAIR POWERS: Yes, an appendix.

MR. PALLA: Like NUREG-1150 and the

supporting documents.

CHAIR POWERS: Well, I would hope that it would not be that way. I would hope that it would still remain a tractable document but a reference document, nevertheless. And so it is something for us to discuss further. I just don't think it can be done in one document.

MR. PALLA: Okay. So that is our thought. That is the content of the user need.

CHAIR POWERS: Well, the existence of this tentative user need, of course, substantiates the ability to close the action plan and move the research into the regular research program. So, I mean, it's not orthogonal to this.

MR. PALLA: No.

CHAIR POWERS: It's supporting of this.

MR. PALLA: It's a direct follow-on.

CHAIR POWERS: Yes.

MR. PALLA: It's a continuum, really.

CHAIR POWERS: Are there any questions for Mr. Palla, who is an unfailing source of good ideas here?

(No response.)

CHAIR POWERS: I think that gives us a lot to think about here.

I would say that we have one last presentation on 3.11 and GSI-163. Notice that it is abbreviated in the extreme.

13. SGAP ITEM 3.11 (GSI-163)

REQUEST ACRS TO DOCUMENT SGAP 3.11 CLOSED

BASED ON ACRS CLOSEOUT OF GSI-163

MR. BEAULIEU: Yes. This one, there is a slide for it, but you don't really need it.

(Laughter.)

MR. BEAULIEU: It is truly administrative.

MEMBER MAYNARD: How can we go back to the previous slide if we only have one slide?

(Laughter.)

MR. BEAULIEU: Item 3.11 is GSI-163. So ACRS has reviewed and closed 163, but our presentation never mentioned that this is also 3.11. Therefore, ACRS letter never closed 3.11. And so we would just like ACRS to document that.

CHAIR POWERS: I think we can handle that one. That's one that we can cover.

MR. BEAULIEU: So that's it. I wish they were all that simple.

I would like to also reemphasize Bob raised a bunch of points about all these unanswered questions. Well, that illustrates the point that all of these products really build upon what we have already learned, the new tasks.

And it shows that we are not pulling any punches in terms of we still think we need to do. And we will continue to track and do that work outside the plan. That should give you a level of confidence that that is a true statement.

That's it.

CHAIR POWERS: Any other questions?

(No response.)

14. COMMITTEE DISCUSSION

CHAIR POWERS: Okay. Now we turn to the issue of what we have to do. We are going to hear more tomorrow, but I wanted to cover this material now and get feedback and comments on the letter.

One has the choice in thinking about this of writing an extremely exhaustive letter or writing a much more summary letter. It may come as a surprise to you that I lean heavily to the much more summary letter kind of idea.

Nevertheless, I think we can't escape a certain amount of specificity in this. We have heard a lot from the staff about resolving the thermally induced steam generator tube rupture, particularly from the thermal hydraulics view. Sanjoy, your views? Does that look like they've got what they need for this?

MEMBER BANERJEE: I think they have certainly made a lot of progress. And they have tried to respond to what the ACRS had asked for. And it looks like very nice work. And we have been discussing with Chris.

Let me ask that. Some of the sensitivity studies they have done have been useful. Whether that really quantifies or whether we can quantify the uncertainties which could lead to rather major differences in consequences, I am not yet clear about. I really need to look at the details. The devil here is in the details.

CHAIR POWERS: I come away with the impression that the answer is going to be no, that they cannot because of the multidisciplinary nature of the problem.

It really requires a more integrated approach than was conceived in the original task, that, even if you did the thermal hydraulic uncertainties in exhaustive detail, you would only be partway through the problem. So I kind of lean toward Mr. Palla's presentation that, yes, there needs to be a more integrated approach to this thing.

I am a little surprised because it is such a juicy problem and susceptible to a more rigorous uncertainty analysis that I am surprised people avoided it because I would have -- I mean, I didn't even do that kind of stuff, but I would have jumped on this and said, "Yeah, I can do this. And I will Monte Carlo this thing to death," you know, because it is very susceptible to that sort of thing.

MEMBER BANERJEE: Yes. It is exactly what you're saying, but, as our colleague George likes to point out to me, thermal hydraulists look at the world in very deterministic terms.

CHAIR POWERS: They do. They do.

MEMBER BANERJEE: And they're frightened of actually assigning probabilities to various events or working with the people who can and working through it because clearly what is important here is to bring the PRA people and the stress analysts all together and do this problem, --

CHAIR POWERS: Well, I think the --

MEMBER BANERJEE: -- which has enormous sort of implications --

CHAIR POWERS: Oh, it does.

MEMBER BANERJEE: -- on SOARCA and things like that as well. So how you go about doing it I don't know. So I'm just looking at it purely from a thermal hydraulics viewpoint at the moment and saying I don't know what the likelihood of loop seal clearing is.

I don't have a clear picture of the uncertainties in the CFD calculations. Remember that there is only one sort of one-seventh scale experiment that to sort of anchor it. And we all know the major uncertainties in CFD, which is usually called color fluid dynamics, --

CHAIR POWERS: Yes.

MEMBER BANERJEE: -- computational fluid dynamics. So when you sort of compound all of this and you try to draw an uncertainty band, you might find that these uncertainty bands are very large indeed, I mean, if you do it fairly carefully.

Whether that is true or not I simply don't know yet. The amount of information we have, without looking at it in detail, I cannot give an answer as to whether it's sufficient to get an idea of it or it needs quite a bit more work on that.

Even if you worked with the PRA people and they assigned certain uncertainties or whatever various events and we work with the stress analysts, we may still have to do thermal hydraulics calculations to bound the uncertainties better or we may not. I think it would be helpful, though.

What I intend to do is to go over these reports before we have to write a letter or ask Professor Wallis, if he would, to take a look at it as well since he started the problem, you know.

CHAIR POWERS: My intention, at least what I have drafted for a letter, is to specifically address the issue of thermally induced steam generator tube ruptures, be relatively complimentary on what has been done to say that in the end, they have really not done the uncertainty analysis that was looked for, but what they have revealed is that that job is much bigger than what was conceived of at the inception and say in that regard that it is reasonable to defer that and reference the forthcoming user need as the more appropriate vehicle for trying to carry out that bigger job.

MEMBER BANERJEE: I have no problem with that. I think that they have made a lot of progress and the counter-current flow and the mixing all go in the right direction.

CHAIR POWERS: Yes. I think they have. I mean, I am very complimentary for the way they have interfaced CFD with their systems-level analysis. And I come away thinking CFD may be more useful than I usually give it credit for, but yes, I thought they could derive some unusual insight.

MEMBER SHACK: Well, it is not often you get to work with just a gas.

CHAIR POWERS: Yes.

MEMBER BANERJEE: Only a gas.

MEMBER SHACK: I can do calculations with gases.

CHAIR POWERS: Well, I spent a lot of time working with just aerosols. And the aerosol is a minuscule perturbation. Somehow the insights gained on CFD on aerosol physics is so minimal that they are generally abusive. I mean, I think we derived a lot out of this.

MEMBER BANERJEE: And they have done what we asked them to do.

CHAIR POWERS: Yes. They have indeed. And they have not really done -- we specifically called for rigorous uncertainty analysis. Now, I think our determination on the word "rigor" was left to the reader to decide.

I don't think they've done that, but I think they have shown enough that had they done it, they wouldn't have accomplished very much and that it is simply a bigger issue than --

MEMBER BLEY: But there is a part of that where they could have accomplished a lot, I think. And that is not having done the quantitative detailed uncertainty analysis but cataloguing and comparing and evaluating the range of uncertainties and putting them all in one place and what's aleatory and what is epistemic and trying to get a sense of how they all line up.

What you need to do before you can do any of the quantitative work would pull it together quite a bit. We've got lots of pieces out there and no easy-to-grasp catalogue. What you think is important is what you remember most recently, rather than the whole set.

MEMBER BANERJEE: Yes. I think there's --

MEMBER BLEY: That's the place I think they could have.

MEMBER BANERJEE: I think they can still do that.

MEMBER BLEY: Oh, yes, absolutely.

MEMBER BANERJEE: They have a lot of the pieces there.

MEMBER BLEY: Absolutely.

CHAIR POWERS: I make no secret that I think that, at least in the draft, pulling it all together is going to be commendation.

MEMBER STETKAR: Well, it would be useful to capture that information while it's alive.

CHAIR POWERS: Well, I say it would be even worse. One of the problems is that this entire effort has spanned well over a decade. And some people working over here remember some pieces from an allied discipline. Others remember more recent pieces from an allied discipline. And it gets confused.

MEMBER BLEY: Many are beginning to retire. So it would be nice to get it before.

CHAIR POWERS: That is right. I mean, at least in my draft paragraph, what I call attention to is it's not just for the technical status and the public interest in this but also from the knowledge preservation point of view that pulling it all together in some tractable document I think is going to be crucial as the agency brings on young people to replace the old fogeys that are going away and whatnot.

Okay. Well, we heard a very summary

presentation on the materials aspects of this problem. And that, of course, has been a central issue here.

Comments on the materials aspects of the problem?

MEMBER ARMIJO: I would like to make a comment. First of all, I think the work is really impressive. I am not sure whether I can really believe yet whether the hot leg fails six minutes before a bunch of perfect tubes fail.

And I don't know enough about this to know how important it is, whether it fails six minutes before or six minutes after, but there is a report on that subject that I failed to read. And I am going to read that before the full Committee meeting and see if I understand what went into the determination of when a material, that hot leg material, fails.

I think we reviewed the closure or at least the completion of the pressurized thermal shock program, which I thought was really a good piece of work. And this has the similar elements, complexity and uncertainty.

And I think maybe they had a bigger budget or maybe they had an easier problem to solve.

But I think they dealt with uncertainties in all of these areas, thermal hydraulics materials, flaw distributions, all of that.

I think they handled it very well, and maybe that's one of the things that will be done in the future research. I believe you have done everything you have been asked to do in the action plan.

So from the standpoint of closing things, I think it should be closed. That's all I have.

CHAIR POWERS: Any other comments in that area?

(No response.)

CHAIR POWERS: When Bob spoke to a variety of the tasks that are probabilistic in nature, we're going to hear tomorrow about 3.5. And Bob's essential contention throughout most of this was either it wasn't worth doing or it's all done in 3.5. So I guess we have to wait until --

(Laughter.)

CHAIR POWERS: We have to wait until 3.5.

Yes, sir?

MEMBER SIEBER: Yes. I guess something that I will dream about tonight or have a nightmare over is with respect to consequential steam generator tube ruptures.

I agree that what it seems to me to be is a horse race as to what fails first. And if you are unlucky and have the steam generator tubes fail first, you have lost defense-in-depth.

I would like to feel a little bit more comfortable that something else will fail. And a thermocouple with its scoop or some instrument line won't do it for me.

CHAIR POWERS: Well, I got the impression from their thermal hydraulics analyses that, even should a steam generator fail first, that the hot leg is imminent and that when a hot leg fails, it --

MEMBER BANERJEE: If you fail a lot of --

MEMBER STETKAR: It goes back to what Sam said. He used the technical term "a buncha," --

(Laughter.)

MEMBER STETKAR: -- as in "a buncha tubes." Buncha is what the buncha does.

MEMBER SIEBER: The point of my concern is, is there something that should be done in emergency planning space, mitigating beyond the design basis accident?

When you get into the horse race, you know, one in a million chance or whatever it is, you get to the horse race and you see on the chart, you're five minutes away from something failing. Should you do something?

And the alternatives are not good. One of them is allowed to happen and have a containment bypass. Then the core melts. And you don't have enough time to get out of town or the other thing is do something that causes the steam generator tubes not to rupture that dooms the core. And that's a bad choice.

So if I felt more comfortable in the timing, maybe more than six minutes or something like that, then maybe I wouldn't be thinking about those alternatives.

And so I need to get into perspective --

MEMBER STETKAR: Yes. You raise a good point.

MEMBER SIEBER: How close to the edge are you?

MEMBER STETKAR: I think what we'll hear tomorrow a little bit is that a lot of the things that are in place to try to protect the core have the secondary benefit of mitigating the consequential steam generator tube rupture, too.

So a lot of the things that the operators are now being trained to do --

MEMBER SIEBER: I can hardly wait for tomorrow.

MEMBER STETKAR: -- hope that they're not counterproductive; in other words, you aren't going to face the situation where you have to give up the core to save the tubes --

MEMBER SIEBER: I think the key day is tomorrow.

MEMBER STETKAR: -- in terms of anticipatory actions that you might be able to do.

CHAIR POWERS: But Jack raises a good point that when we switch our perspective to one of accident management, that the focus on bounding analyses that get this horse race very close together, it does not serve us well when we think about that --

MEMBER SIEBER: That's right.

CHAIR POWERS: -- and that we need to think about the more realistic separation in time between these --

MEMBER SIEBER: That is exactly right.

MEMBER SHACK: But I think, as Chris points out, it is never going to separate. It's a vain wish. Physics just essentially forces this thing into a race that there's nothing you can do about.

MEMBER ARMIJO: Once you get that hot --

MEMBER SIEBER: Once you get the core oxidation or clad oxidation, you've got a whole energy source that overwhelms everything else and is driving against every weak point in the system. The question is, which one is the weakest or do you make one?

CHAIR POWERS: As a famous member of the ACRS once said, decay heat doesn't melt cores. The zirconium clad oxidation melts cores.

Okay. Are there any other comments we should factor in into thinking about this letter? Otto, please?

MEMBER MAYNARD: I've got a couple of just general comments in consideration for the letter. You know, one is that overall I think the effort has been very good.

It has helped, although there are other reasons and motivations too. This effort overall has helped improve steam generator performance, better material, better inspections, you know, a lot of things that -- this effort has contributed to those positive things. There are other motivations, too, but I think it is very good that this effort did that.

The other, a little bit along the lines of what you were just talking about, one thing I haven't heard really talked about in this whole discussion is what has been learned about better or different ways to manage severe accident.

We can argue all day long about what happens first or whatever, but, in reality, have we learned something that might be of use in factoring in for handling severe accidents and doing things like that? That is something that --

CHAIR POWERS: Well, I think you will hear more in that vein tomorrow and whatnot, but your general point that the overall quality of this work is very good, I think that will be central in the letter.

I mean, I think we have looked at many of the points in the action plan in the past and have been pleased with the sincerity of the effort. And we asked a specific question here that we haven't got a final answer on yet, but I think we can certainly be complimentary on all of the points that we have heard today.

And where they have deferred items I

haven't seen a -- it hasn't caused any heartburn for me. The points were much more shotgunny in nature. And as they get smarter, some of them yes. There's just no point in doing it because it is being done elsewhere or doesn't need to be done.

I think we will be relatively complimentary on each of the tasks. I can't imagine we won't.

MEMBER BLEY: I agree. I would like to follow up what Otto said, the one area where we did hear an action suggested. And that is the one of opening the PORVs, kind of after the core melt to the open path and drop the pressure if the key would be the 1,200 degrees exit temperature and the 10 minutes.

Right now that kind of stuff is off in the SAMGs and the Tech Support Center. And you can't really hope, I don't think you can really hope, that a ten-minute time window is going to be something you will control out of the Tech Support Center.

So, again, the idea of how do we go forward with trying to institutionalize some of the lessons is something pretty important or could be.

CHAIR POWERS: Okay, gentlemen. Any other comments that we would like to make?

MR. BROWN: Just one thing.

CHAIR POWERS: Mr. Brown?

MR. BROWN: Dennis wanted me to send you all a 2008 Sandia letter report. So it's in your e-mail. It relates to the subject tomorrow.

MEMBER BLEY: It's the report they were talking about.

MEMBER BANERJEE: Which one is that?

MEMBER BLEY: The one that describes this probabilistic computer code that lets them do the risk work and pull all of these pieces together into one place.

CHAIR POWERS: Are there any other comments to make? I guess we all look forward to tomorrow's presentation. I think we appreciate all of the discussions today.

The presentations I thought were all at exactly the level that we needed them for for our developing a draft position and appreciate very much what you have done for us here.

MR. BEAULIEU: You mentioned about the summary document. It was one of the closure documents in that mountain of information that you got. It's called a RIL. It's a research document. So that was the attempt at pulling all of the research effort.

MEMBER BANERJEE: Is that in the CD?

MR. BEAULIEU: That is in the CD provided to you.

MEMBER BLEY: If someone could point us to where the specific document --

MR. BROWN: The way the CD is, we labeled them by item numbers followed by the ML number. So next time we'll put the subject.

MEMBER BLEY: On this one he is talking about, if you will give us the numbers?

CHAIR POWERS: Maybe you can just print a disk of yours, then. It will save me a lot.

MEMBER BLEY: I can e-mail the ones with --

CHAIR POWERS: Having read research information letters in the past, it may be not exactly the document I am looking for, but we'll see.

With that, I suppose that we can recess for the day and to study up our probabilistic risk assessment jargon and be prepared for tomorrow.

(Whereupon, the matter was recessed at 4:33 p.m., to reconvene on Friday, September 25, 2009.)

ACRS
Meeting of the Subcommittee on the SGAP
September 24, 2009
Rockville, MD

SGAP Items 3.4.a-g
Introduction
Thermal-Hydraulic Studies

Christopher Boyd RES/DSA

Background

- SGAP Section 3.4
  – **develop a better understanding of RCS conditions and component behavior under severe accident conditions**

- Section 3.4.a-g
  – **thermal-hydraulic** behavior of the system during severe accidents
    - focused on conditions that challenge the integrity of the steam generator tubes

High-Dry-Low

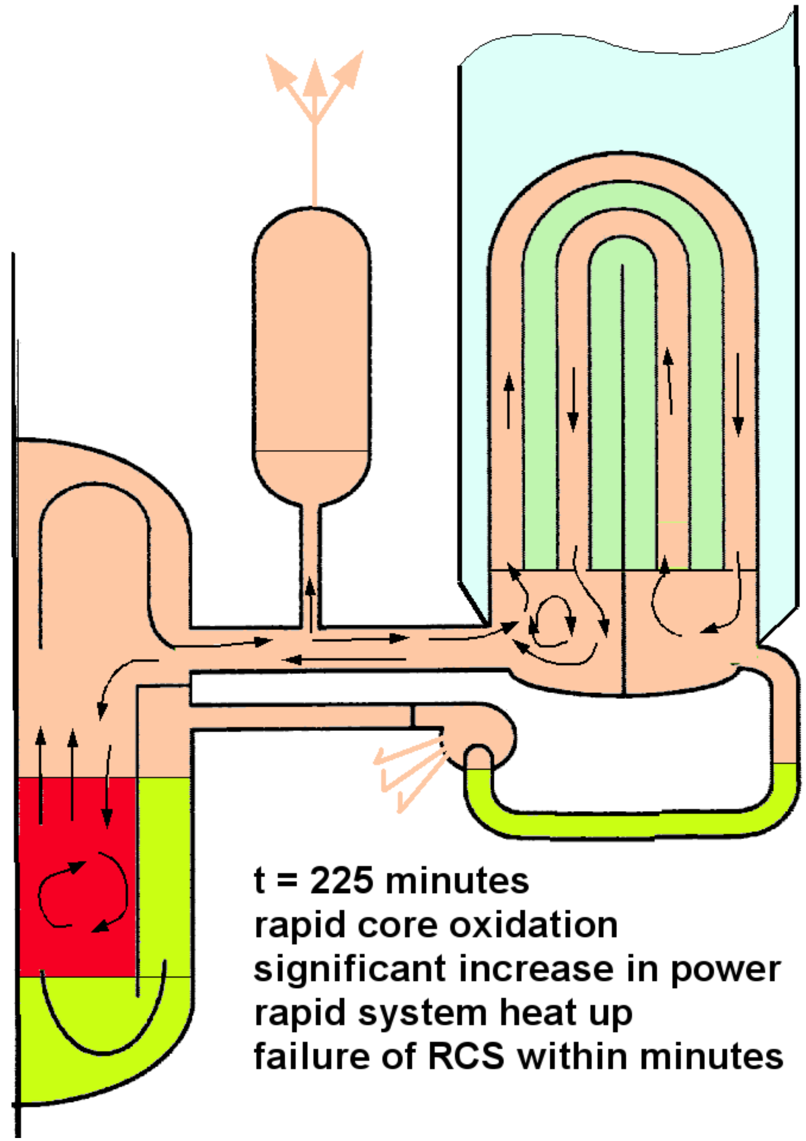
- The primary challenge to the tubes is when the plant is in a “high-dry-low” condition
  - <u>High</u> primary side pressure
    - RCS must remain intact with no significant leaks
  - <u>Dry</u> steam generator secondary side
    - all auxiliary feedwater systems fail
  - <u>Low</u> pressure on the secondary side
    - leakage or valve failure must occur to depressurize the secondary side.

A Fast Scenario

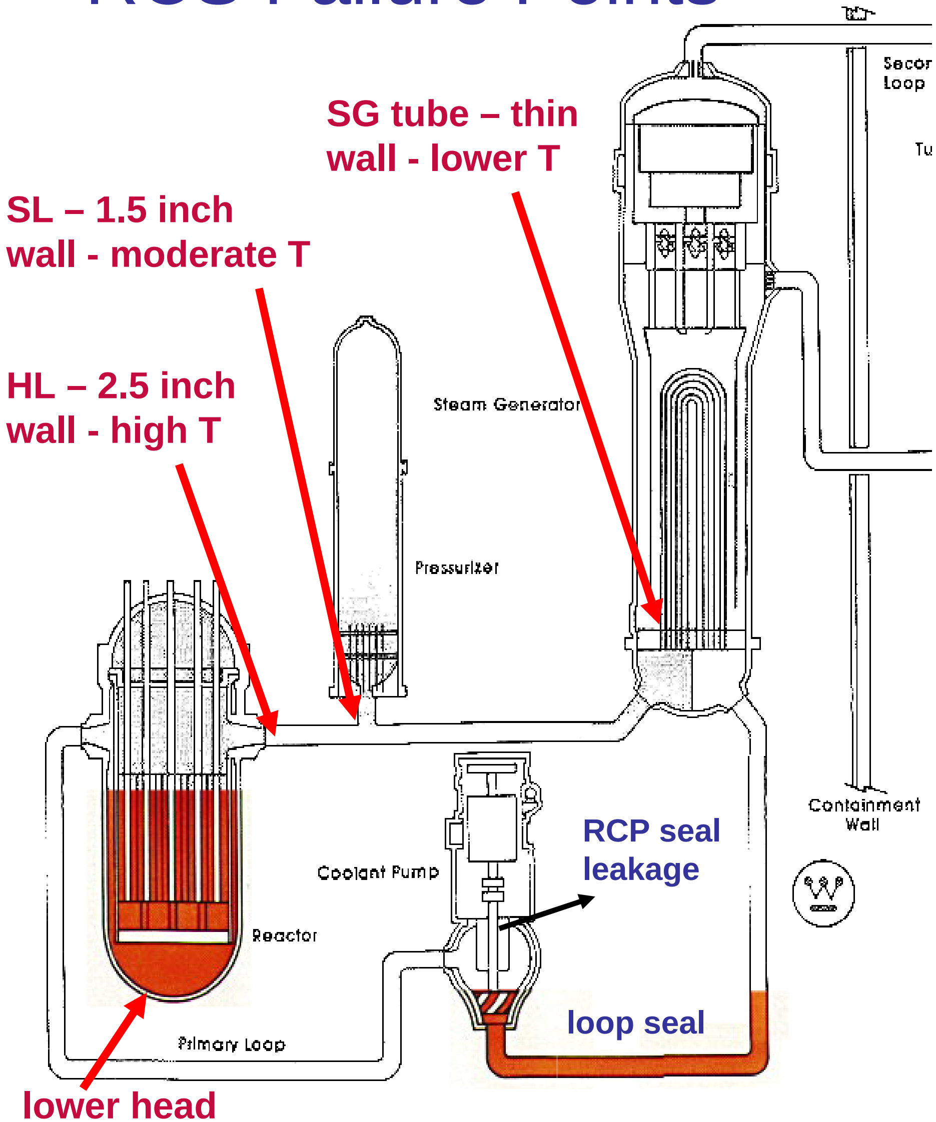
RCS failure within 4 hours

- loss of offsite power, failure of diesel generators to start, and failure of all auxiliary feedwater systems
- primary inventory lost through reactor coolant pump seal LOCA and secondary inventory is boiled off
- secondary system dries out and safety relief valves start cycling, primary inventory lost through valve cycling and pump seal LOCA
- loop natural circulation stops as primary inventory falls, inventory falls below hot leg
- natural circulation of superheated steam begins, core and system heat up.
- Core uncovers, core oxidizes and produces significant power, system heat up accelerates and induced failure is predicted for RCS components.

- More likely scenarios involve some auxiliary feedwater or operator actions that significantly delay the failure time.



RCS Failure Points



Technical Highlights

-SR5 used as a screening tool for potential induced SG tube failures.

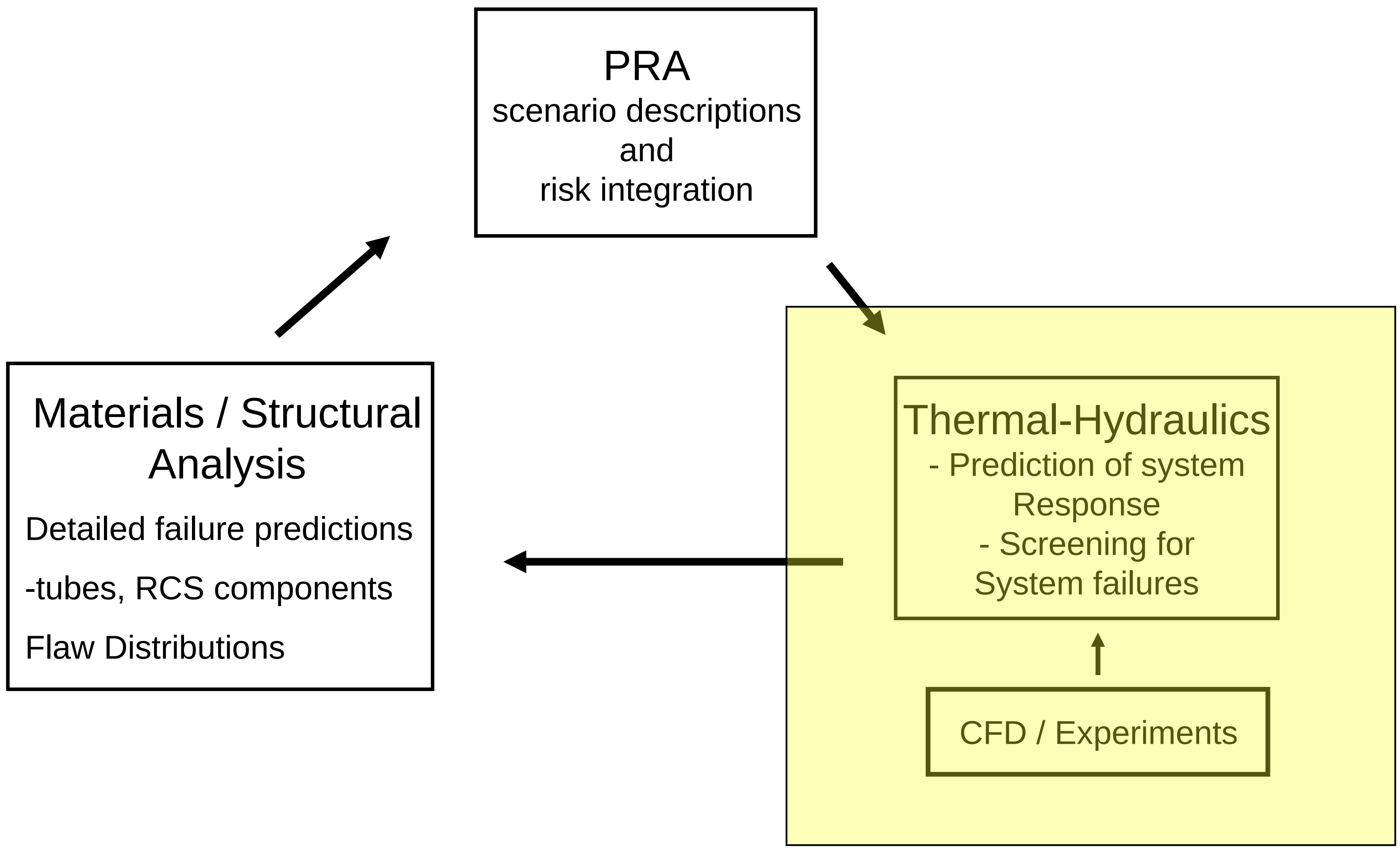
-Failure times predicted for specific RCS boundary locations.

-For high-dry-low with the loop seals filled, the HL fails first. No depressurization is modeled. Subsequent SG tube failure times are predicted. Stress multipliers are applied to determine additional stress needed to fail SG tubes prior to HL.

-The hot leg is predicted to fail even if SG tube ruptures are modeled as a break (with loop seal filled).

-Relative failure timing predictions are conservative in the SCDAP/RELAP5 model

T-H predictions Integrated with Project



Thermal-Hydraulic Issues

- NUREG-1740
  - 1D code input parameters are adjusted by comparison with experimental results to ensure consistent behavior
    - Test scaling criticized
    - Mixing may be overestimated
    - Tests did not simulate leakage
  - Sensitivity studies may not have covered the entire plausible range of variations nor did they cover simultaneous variations.
- ACRS Review – February 2004
  - requested staff to develop a model to predict counter-current flow in the hot leg using CFD
  - requested the staff to provide additional analysis of reactor coolant pump loop seal clearing to support the system code models

SGAP - Thermal-Hydraulic Tasks

- 3.4.a-g
  - **a.** perform plant sequence variations using SCDAP/RELAP5 (SR5)
  - **b.** re-evaluate system code assumptions and update model as necessary (SR5)
  - **c.** estimate tube temperature variations from 1/7th scale data
  - **d.** perform more rigorous uncertainty analysis on system level predictions (SR5)
  - **e.** benchmark CFD tools using 1/7th scale data and extend the methods to full-scale
  - **f.** estimate uncertainty due to core melt progression (SR5)
  - **g.** perform additional experiments to include the impact of inlet plenum geometry variations and tube leakage

SGAP 3.4 T/H Tasks Completed

- NUREG/CR-6995 (draft) summarizes the work that has gone into addressing these issues. This work improves our understanding of the T/H behavior of the plant and addresses key criticisms of past analyses. (covers 3.4.a, b, d, f)

- Supporting CFD analyses (3.4.c, e, g)
    - NUREG-1781 and NUREG-1788
    - NUREG-1922 (draft)

Presentation Outline

- SCDAP/RELAP5 Modeling
  - Don Fletcher
  - Information Systems Laboratories, Inc.

- CFD Modeling
  - Christopher Boyd
  - Division of Systems Analysis, Office of Nuclear Regulatory Research

STEAM GENERATOR ACTION PLAN

SCDAP/RELAP5 THERMAL-HYDRAULIC EVALUATIONS OF THE POTENTIAL FOR CONTAINMENT BYPASS DURING EXTENDED STATION BLACKOUT SEVERE ACCIDENT SEQUENCES IN A WESTINGHOUSE FOUR-LOOP PWR

Don Fletcher, Robert Beaton, Vesselin Palazov, David Caraher
Bill Arcieri, Rex Shumway (Consultant)

Information Systems Laboratories, Inc.
Idaho Falls, ID and Rockville MD

Presented

ACRS Subcommittee Meeting

Rockville, MD

September 24, 2009

Purpose of the SCDAP/RELAP5 Thermal-Hydraulic Analysis

Determine the sets of plant configurations, conditions and accident event sequence scenarios that can lead to containment bypass through induced steam generator tube failure

- **Risk affected by the order in which the reactor coolant system component structural failures occur**
  - » **Hot leg, pressurizer surge line and reactor vessel lower head failures lead to depressurization of the RCS into containment, precluding subsequent SG tube failures and containment bypass**
  - » **SG tube failures lead to discharge from the RCS into the SG secondary system and may lead to containment bypass (via release to the environment through main steam safety relief valves) but do not preclude subsequent failures of hot leg, pressurizer surge line or reactor vessel lower head**

Draft NUREG/CR-6995 (scheduled for publication in late 2009) summarizes the culmination of extensive SCDAP/RELAP5 evaluations into containment bypass

1998-2001 NUREG-1570 / NUREG-1740 Analyses – Loop seal clearing, RCP shaft seal leakage, system code limitations, consideration of SG tube leakage

2000-2002 Revised Station Blackout Analyses – Model revisions to replicate natural circulation behavior in Westinghouse 1/7 scale experiments, tube stress multipliers and hottest tube model

2003-2004 Sensitivity Evaluations – Mixing parameters, RCP shaft seal leakage, SG tube leakage, core bypass, core damage progression, event-sequence assumptions

February 2004 ACRS Review – Concerns regarding "SG power fraction" approach for hot leg circulation, loop seal clearing behavior, reactor vessel circulations and energy flows

2004-2005 Analyses to Support PRA – Evaluate RCP shaft seal leak rates, TDAFW operation, battery depletion time, SG secondary steam leakage rates and operator intervention

2005 NRC and Consultant Peer Review and PIRT Evaluation – Core axial nodalization expanded, finer nodalization near SG tubesheet, implement target hot leg discharge coefficient approach for hot leg circulation, hand calculation evaluation of loop seal clearing behavior, PIRT used to identify uncertainty study independent and dependent variables

2006 Energy Flow and Uncertainty Evaluations – Examined variations in RCP shaft seal leakage locations and rates on loop seal clearing behavior, implemented peer-review suggested model improvements, analyzed energy flows, estimated uncertainties in calculated results

2007 Public Peer Review Meeting - EPRI comments based on MAAP analyses: SCDAP/RELAP5 steam-to-hot leg wall radiation heat transfer underpredicted, assumption on hottest tube inlet temperature too conservative, not considering creep rupture failure of hot leg nozzle carbon steel safe end too conservative

2008 ISL NUREG/CR Summary Report Covering SCDAP/RELAP5 Analyses – Upgraded hot leg steam-to-wall radiation model, performed final base case and sensitivity screening analyses categorizing events into groups that: (1) lead to containment bypass, (2) do not lead to containment bypass and (3) have a potential to lead to containment bypass (depending on actual SG tube strength and distribution characteristics)

SCDAP/RELAP5 Code Description

Combination of RELAP5 thermal-hydraulic system fluid flow and heat transfer models and SCDAP core severe accident models

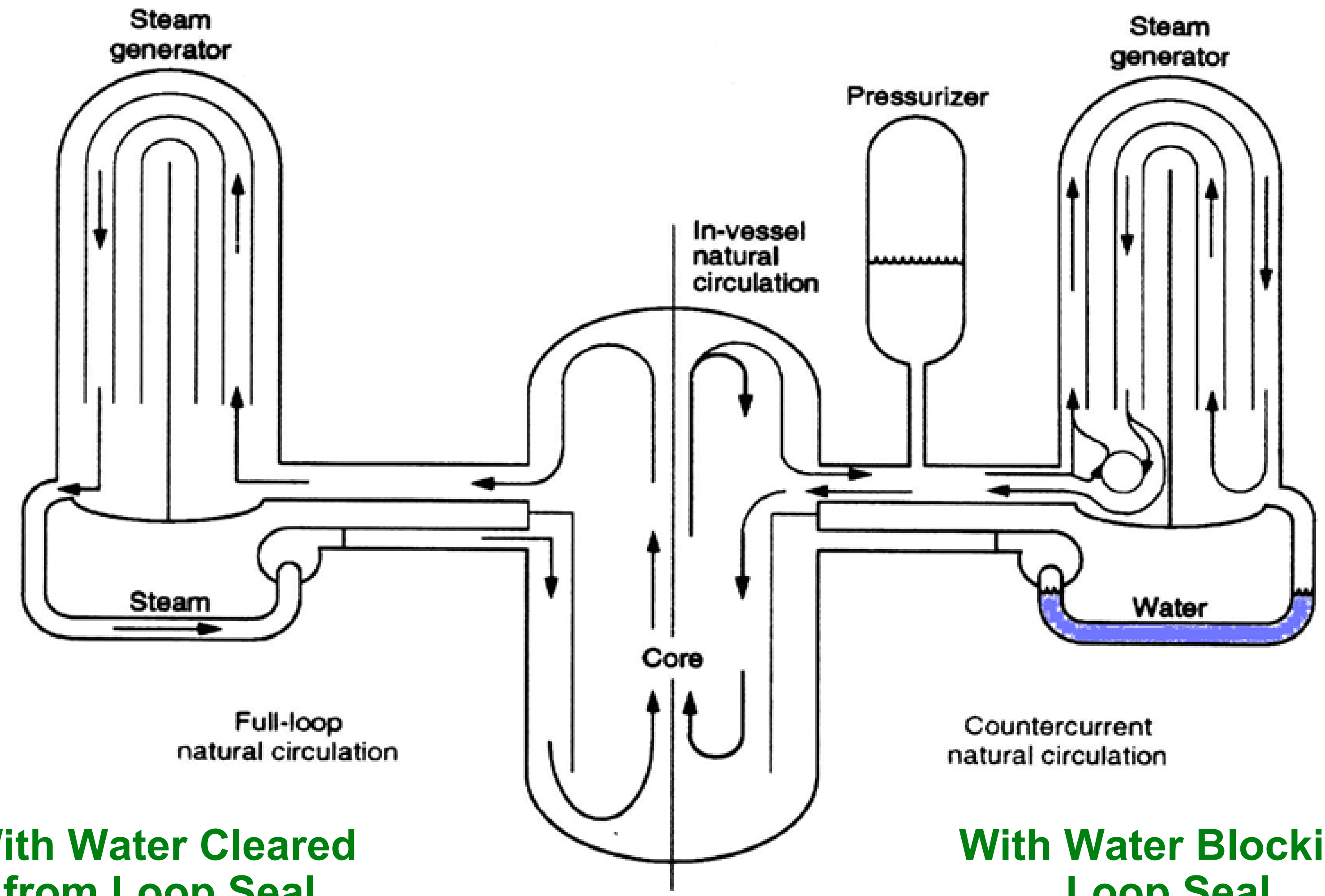
RELAP5 solves conservation of mass, momentum and energy equations using a two-fluid (steam/water), nonequilibrium, nonhomogeneous model with a noncondensible gas phase that is tracked with the steam

SCDAP models severe accident core behavior such as fuel rod heat-up, oxidation, ballooning and rupture, fission product release, melting, flow and freezing of materials, and creep rupture failure of structures

SCDAP/RELAP5 is capable of predicting buoyancy-driven flows in one dimensional geometries but lacks capabilities for modeling on a first principles basis certain multidimensional flow behavior which is pertinent for this application

To compensate for this limitation, SCDAP/RELAP5 model flow coefficients are adjusted (based on experiments and CFD predictions) to match important multidimensional hot/cool steam flow effects: countercurrent flow in hot legs, mixing in SG inlet plenum and SG tube bundle flows

Natural Circulation Flow Pattern is Determined by the Status of Residual RCS Liquid Inventory in the Loop Seals



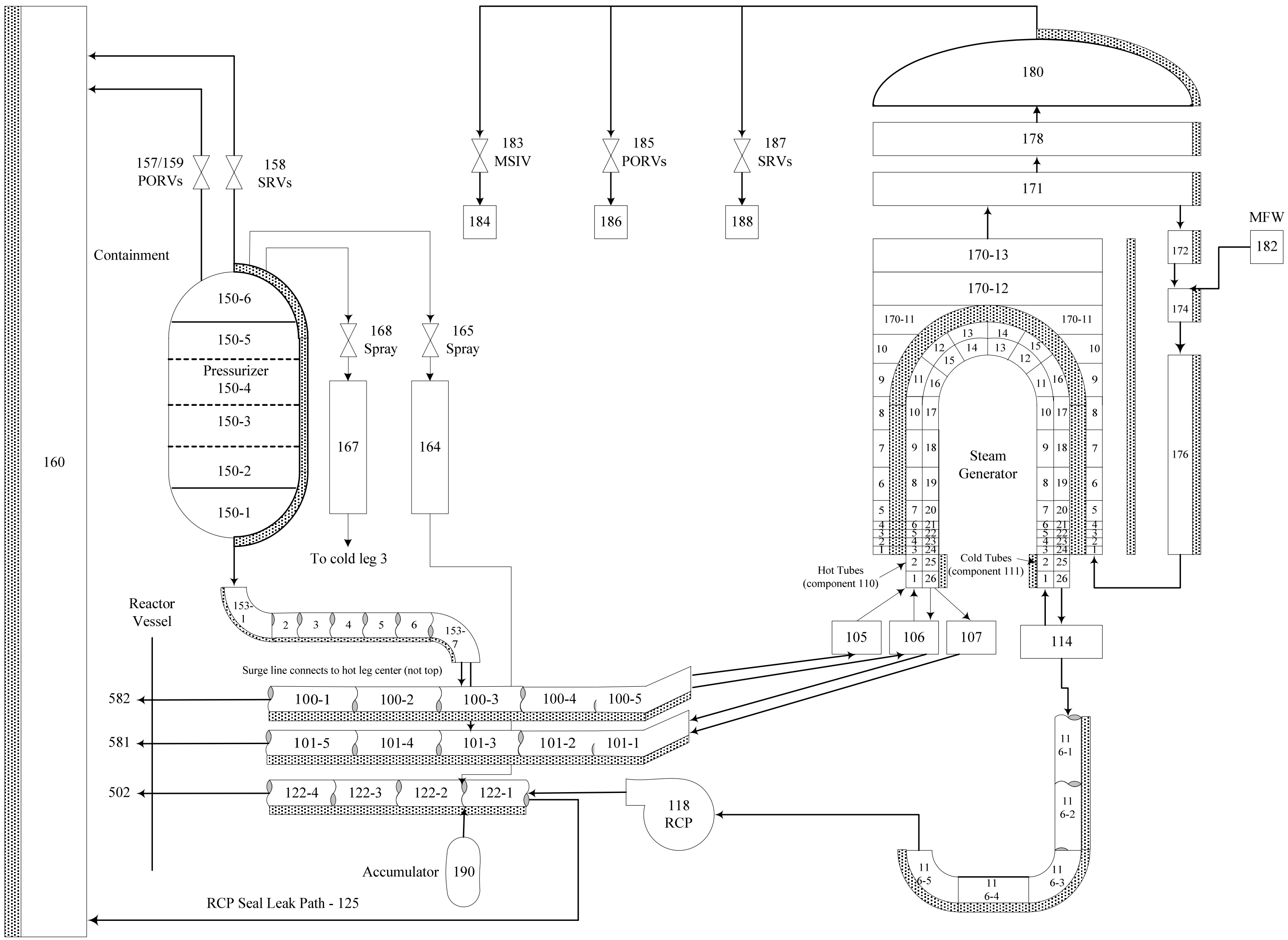
With Water Cleared from Loop Seal

(Which may result under certain specific conditions)

With Water Blocking Loop Seal

(Typical Behavior)

SCDAP/RELAP5 Nodalization for Coolant Loop Connected to Pressurizer With Provision for Countercurrent Natural Circulation - Used During Core Heat-up Period

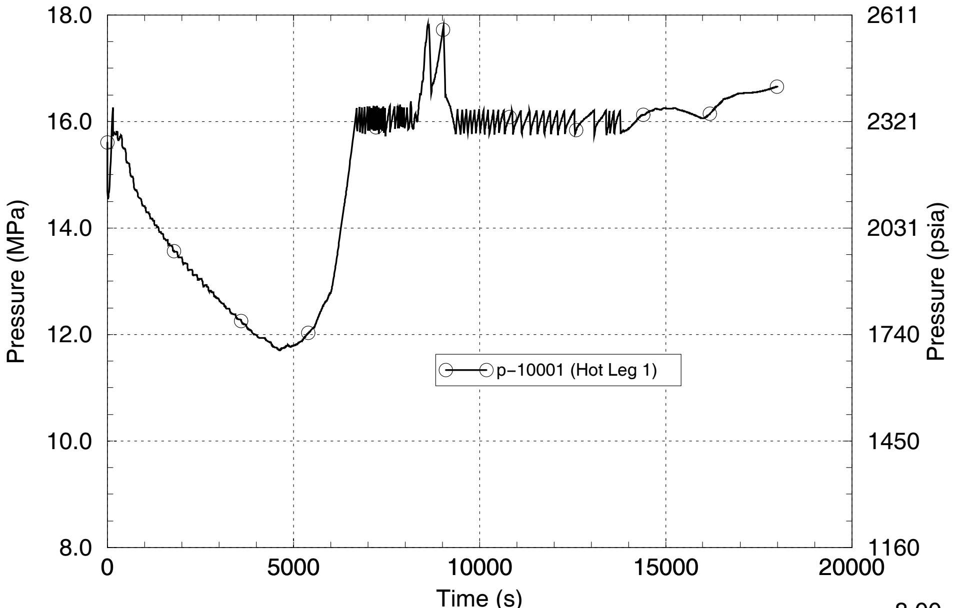


SCDAP/RELAP5-Calculated Base Case Event Sequence

- Assumes plant systems fail immediately – not considered a most-likely accident scenario
- HL, SL and tube break flow paths not directly modeled, allowing parametric evaluation of subsequent tube failures as a function of tube strength degradation

Event Description	Time (seconds)
Station blackout event initiation Loss of AC power, reactor/turbine trips, loss of all feedwater, RCP trip, RCP shaft seal leakage begins (21-gpm/pump)	0
Steam Generator 1 secondary dry-out	5,905
Steam at core exit begins to superheat (hot leg and SG countercurrent circulations begin)	9,226
Onset / peak fuel rod oxidation	10,747 / 13,566
Hot Leg 1 fails by creep rupture	13,625
Hottest SG tube creep rupture failure (SG 1, non-degraded tube strength, 1.0 stress multiplier)	13,985
Pressurizer surge line fails by creep rupture	14,140
Molten fuel pool forms near center of hottest core channel, partially blocking core flow	14,241
Average SG tube creep rupture failure (SG 1, non-degraded tube strength, 1.0 stress multiplier)	14,910
First relocation of control rod absorber material to reactor vessel lower head	15,532
End of calculation	18,000

Base Case Event Leads to High RCS Pressure and Dry SGs at Low Pressures

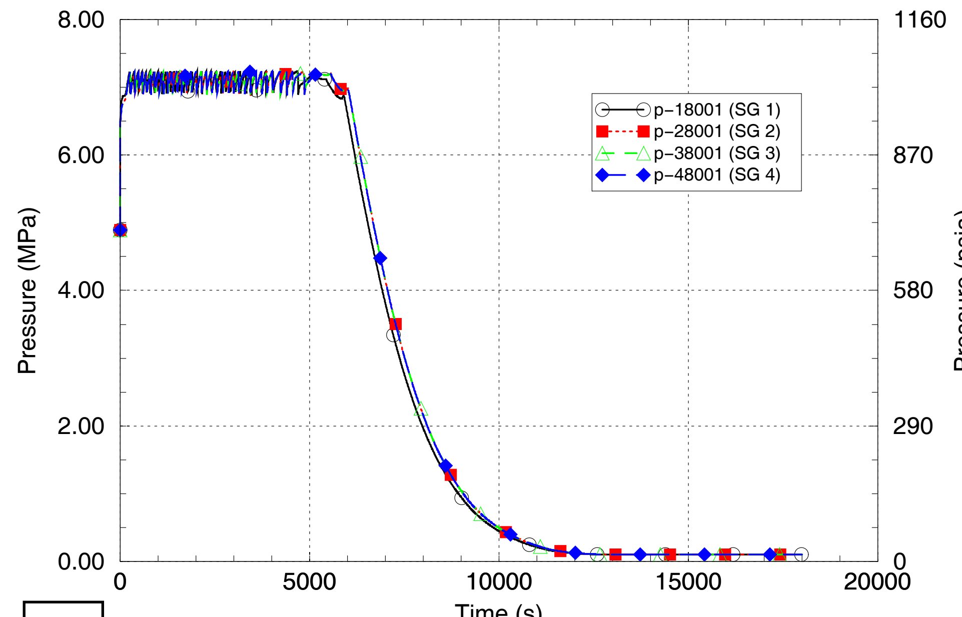


◄ **RCS Pressure**

Pressure falls until SG heat sink is lost. Pressurizer PORVs and SRVs limit the subsequent pressure excursion.

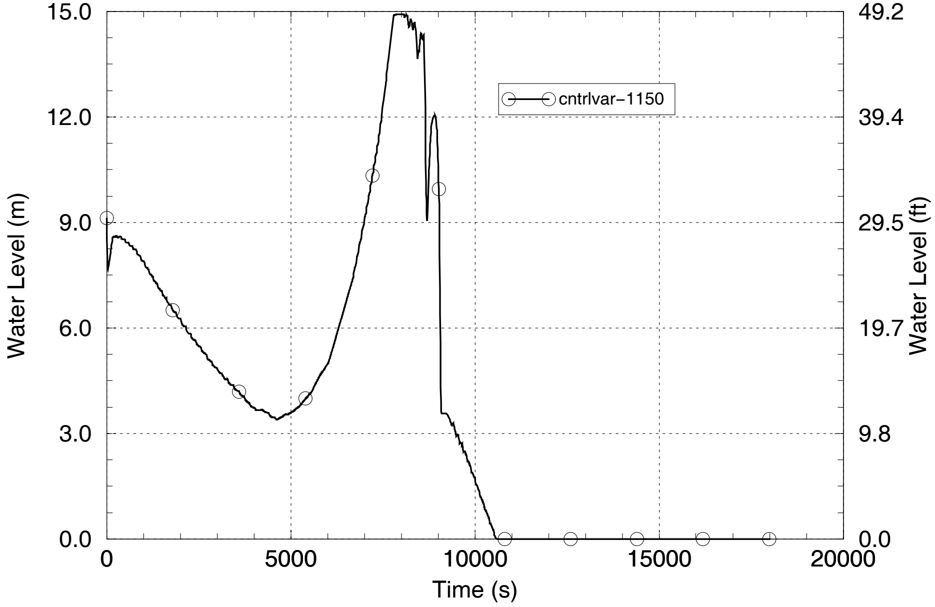
SG Pressure ►

SG PORVs and SRVs limit pressure excursion. Following SG dry-out, the steam-filled SGs depressurized by assumed 0.5-in^2 leak flow area.



Information Systems Laboratories

RCS Inventory Loss Begins After SG Dry-out, Steam-Filled RCS Heats Up; Maximum Heating Rate Experienced when Fuel Cladding Ruptures and Oxidation Power Peaks

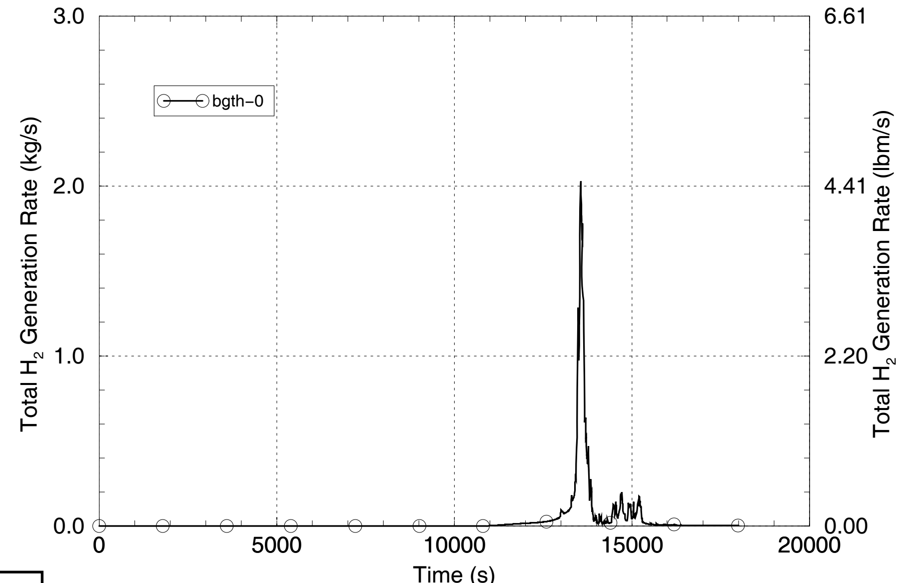


◀ **Pressurizer Level**

RCS inventory declines as fluid flows out the SRVs and PORVs into the pressurizer relief tank inside containment. Pressurizer empties due to continued PORV relief valve cycling.

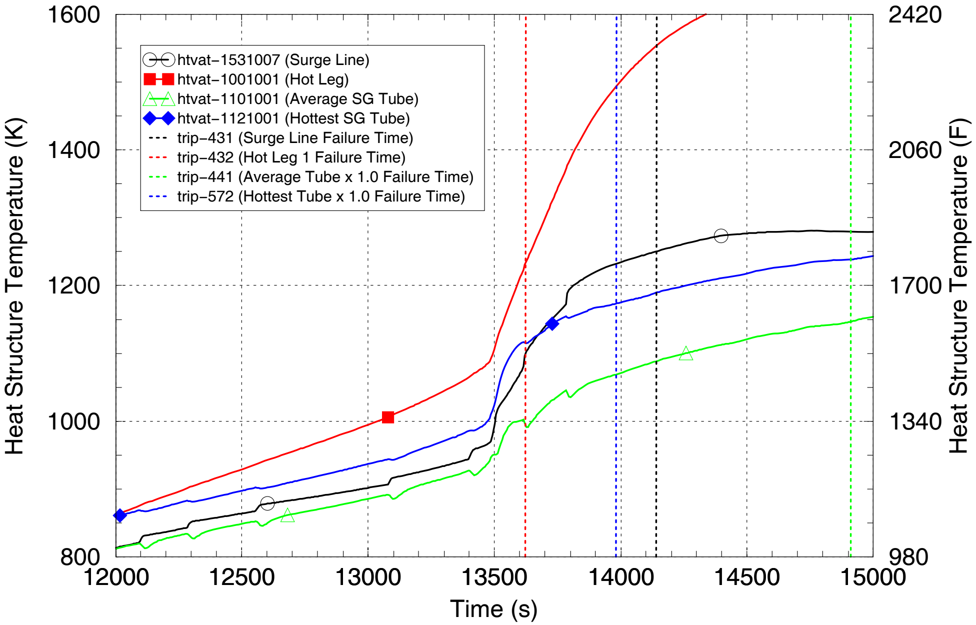
Core Hydrogen Generation Rate ▶

Oxidation rate peaks as cladding ruptures and inside surface is involved. Peak oxidation power is about 10 times the fission product decay heat, or about 9% of normal plant operating power.



Rapid RCS Steam Temperature Excursion Leads to Structural Failures

- Structure creep rupture failure affected by material, strength degradation, thickness, differential pressure and local steam temperature
- Hot steam flows from reactor vessel outward into the RCS
- Effect seen first in hot leg
- Effect in surge line is limited due to mixing effects and because pressurizer relief valves are open only part of the time
- Effect in SG tubes is buffered by time delay for flow through the hot leg and by beneficial mixing of hot/cool steam in the SG inlet plenum



Information Systems Laboratories

SCDAP/RELAP5 Base Case Calculation Results

1.0 Stress Multiplier (Non-Degraded Strength) Used for Hot Leg and Pressurizer Surge Line Structures

Effects of Stress Multiplier Parametrically Investigated for the SG Tubes Carrying Average-Temperature and Hottest-Temperature Steam

Structure	Failure Times (seconds)
Hot Leg 1 (connected to pressurizer)	13,625
Hot Legs 2, 3 & 4	13,660
Pressurizer Surge Line	14,140

SG 1 Tubes with Larsen-Miller Stress Multiplier of:	Average Tube	Hottest Tube
1.0	14,910	13,985
1.5	14,180	13,660
2.0	13,850	13,560
2.5	13,680	13,440
3.0	13,565	13,140
3.5	13,460	12,880

Average Tube in SG 1 with Stress Multiplier of 2.74 Fails Coincident with Hot Leg 1

Hottest Tube in SG 1 with Stress Multiplier of 1.68 Fails Coincident with Hot Leg 1

Key Parameter Variations Identified for the Purpose of Categorizing Event Outcomes

RCP Shaft Seal Leakage Behavior (Increases at 13 minutes and at time when pump fluid reaches saturation, ~2 hours)

Variations in Turbine-Driven Auxiliary Feedwater System Operation

Variations in SG Secondary System Steam Leakage Flow Area

Mitigative Operator Intervention (Pre-Core Damage and Post-Core Damage)

Effects Related to Opening SG Tube Rupture Flow Paths

- Tube rupture flow paths typically not modeled in the majority of the analyses in order to parametrically investigate tube-strength effects
- Sensitivity evaluation indicates that opening tube rupture flow path in the model does not significantly affect the timing of hot leg failure

Sequence Outcomes Grouped Based on Hottest SG Tube Failure Screening

Sequences <u>Resulting in Containment Bypass</u>

- Non-degraded, 1.0-stress multiplier hottest SG tube predicted to fail prior to the hot leg

Sequences with a <u>Potential for Resulting in Containment Bypass</u>

- Hottest SG tube failure margin (stress multiplier) between 1.0 and 3.0

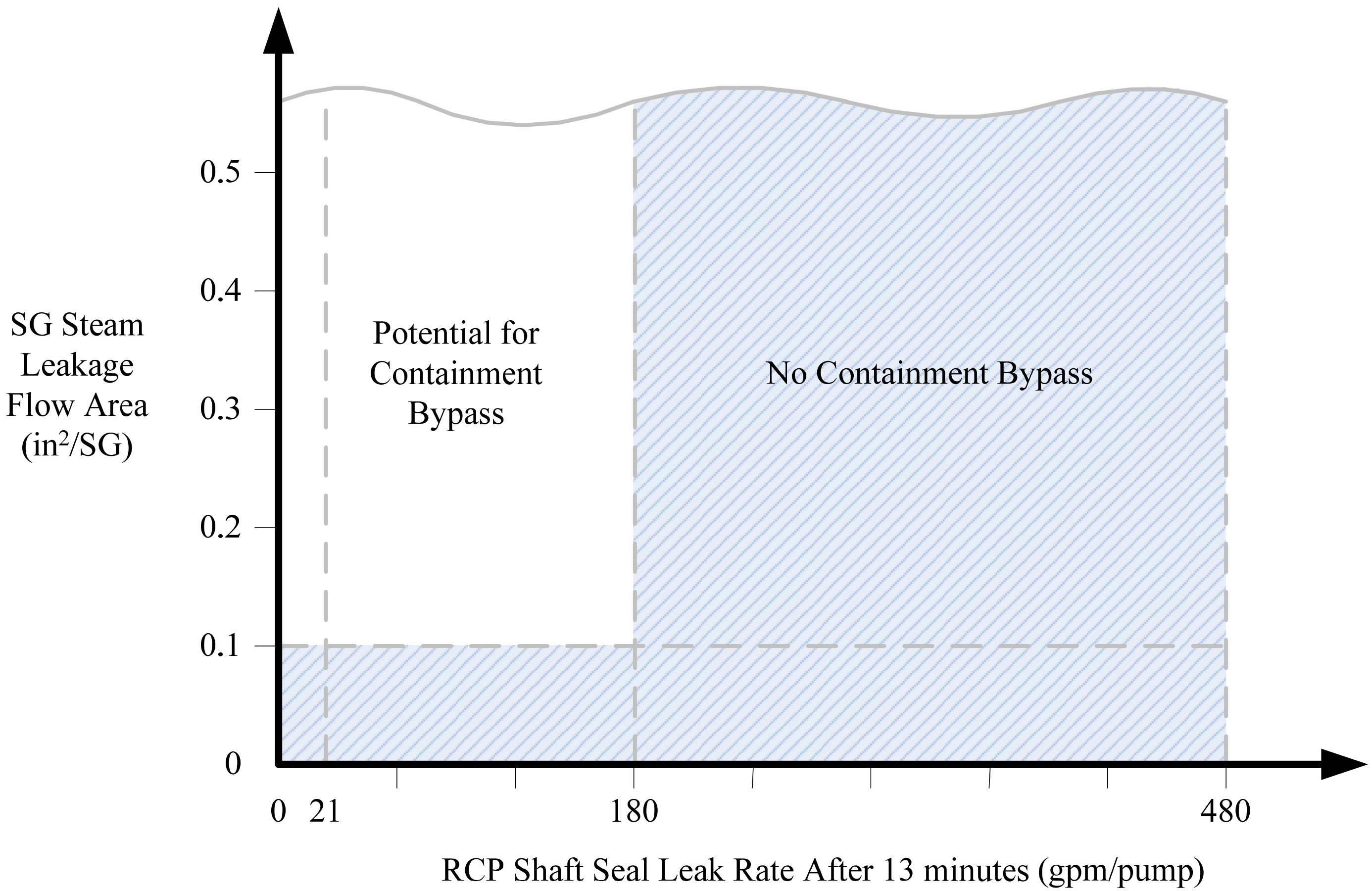
Sequences <u>Not Resulting in Containment Bypass</u>

- Hottest SG tube failure margin (stress multiplier) of 3.0 or higher

Example Containment Bypass Outcome Map, No Operator Intervention, and Variations in:

SG Secondary Steam Leakage

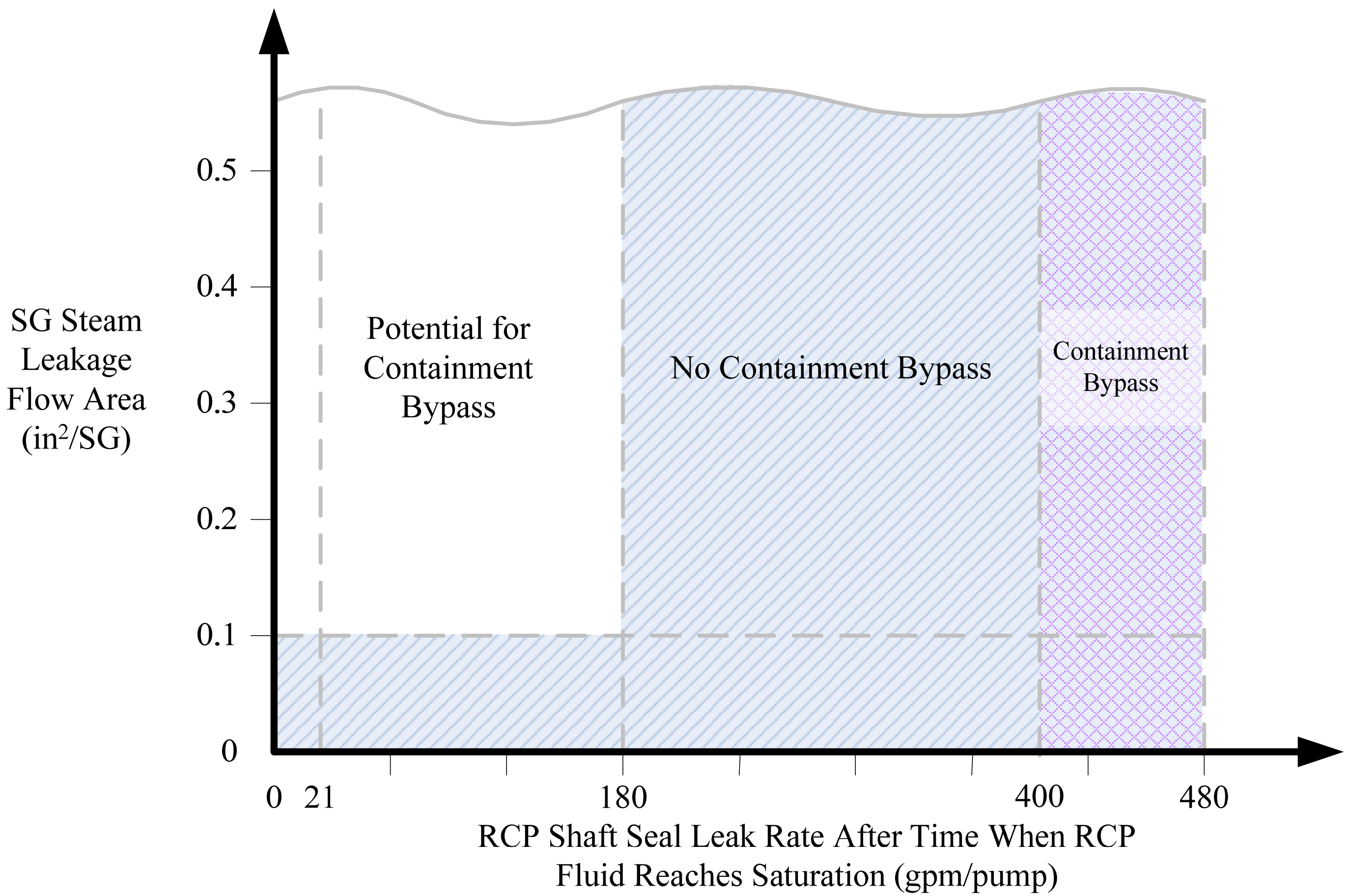
RCP Shaft Seal Leakage that Increases at 13 Minutes



Example Containment Bypass Outcome Map, <u>No Operator Intervention</u>, and Variations in:

SG Secondary Steam Leakage

RCP Shaft Seal Leakage that Increases when RCP Fluid Reaches Saturation



Key SCDAP/RELAP5 Analysis Findings

For situations where the operators are assumed to take no action:

Event sequences which assume very small leakage paths (flow area <0.1 in^2/SG) for steam to escape the SG secondary system generally do not result in containment bypass

Event sequences which assume RCP shaft seal leakage rates below 180 gpm/pump provide a potential for containment bypass

Event sequences which assume RCP shaft seal leakage rates above 180 gpm/pump generally do not result in containment bypass (exception: late increases in the leak rate to above 400 gpm/pump lead to loop seal clearing and containment bypass, regardless of other assumptions)

Key SCDAP/RELAP5 Analysis Findings

For situations where the <u>operators are assumed to take no action</u> (continued):

Event sequences in which the TDAFW system operates and continues operating do not result in containment bypass

Results for event sequences in which the TDAFW system initially operates and then later fails are very similar to the results for event sequences where the TDAFW system is assumed to never operate

Key SCDAP/RELAP5 Analysis Findings

For situations where the <u>operators use the pre-core damage strategy</u> (SG feed-and-bleed cooling at 30 minutes using TDAFW system and opening the SG PORVs):

Strategy is effective in the short term for preventing containment bypass

At a minimum the strategy significantly delays onset of RCS heat-up, thereby providing time for other recovery opportunities to be considered and implemented

In the long term, the SG PORVs fail closed when the station batteries are depleted and continued success of this strategy requires that a TDAFW water source remain available and that some capability for delivering the water to the SGs continues

Key SCDAP/RELAP5 Analysis Findings

For situations where the operators use the post-core damage strategy (depressurize RCS by opening one or two pressurizer PORVs when core exit temperature reaches 1,200 °F or 12 minutes after that time):

Opening only one PORV limits the RCS cooling, the core fails early (prior to station battery depletion) and containment bypass is avoided for either assumed operator action time

Opening two PORVs prevents early core failure and also prevents early failure of the hot leg and SG tube structures. When the PORVs fail closed after station battery depletion, the RCS re-pressurizes and re-heats, leading to subsequent hot leg and SG tube failures. The tube failure margins are significantly improved (over the no operator-intervention margins) and containment bypass is avoided for either assumed operator action time.

Information Systems Laboratories

Summary

Previous ACRS Review Comments Have Been Considered in the Current Analysis

- Improved thermal radiation modeling
- Improved method employed for determining hot leg circulation rate
- Evaluated loop seal clearing behavior
- Evaluated sensitivity to reactor vessel internal circulation rate
- Performed analysis of system energy flows
- Independent peer review of methods and results

Steam Generator Action Plan SCDAP/RELAP5 System Analysis Thermal Hydraulic Tasks are Addressed in Draft NUREG/CR-6995 (to be published in late 2009)

- 3.4a Perform plant sequence variations using SCDAP/RELAP5
- 3.4b Re-evaluate system code assumptions and update model as necessary
- 3.4d Perform more rigorous uncertainty analysis on system level predictions
- 3.4f Estimate uncertainty due to core melt progression

Steam Generator Action Plan 3.4h Potential RCS Failure Locations

C. E. (Gene) Carpenter, Jr.
Group Lead for Aging Management Issues
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Gene.Carpenter@nrc.gov

24 September 2009
Advisory Committee On Reactor Safeguards

Background

- SGAP developed to investigate concern that, during a postulated PWR severe accident, core effluents may bypass containment if failures are experienced in steam generator tubes (SGTs)
  - However, if other reactor coolant system (RCS) components fail before SGTs, containment bypass may be averted
- RES performed scoping review to determine potential failure locations, modes and time-to-failure for non-SGT RCS components during postulated PWR severe accident event

3.4h Research Overview

- NRC conducted three phase scoping study:
  - Phase I reviewed methods and models for predicting failure modes and times-to-failure, identified additional information needed for the study, and scoped RCS components that might be “weak links”
  - Phase II developed three dimensional computer models of selected components for representative Westinghouse 4-Loop plant utilizing detailed mechanical and structural drawings and included analyses of operating history of these components.
  - Phase III utilized Reactor Leak and Power Safety Excursion (RELAP5) code and Computational Fluid Dynamics (CFD) and an expanded high-temperature materials database to calculate the failure sequence of the selected RCS components.

Phase I

- November 2001 Workshop held to discuss expected behavior of non-SGT RCS components and bolted connections during severe accidents in PWRs
  - Workshop concluded it would be possible to analytically predict behavior during severe accidents of certain components
- Following Workshop, non-SGT RCS components and bolted connections were modeled to predict failure times
  - NRC initiated effort to develop improved models
  - Model included variables not addressed in previous analysis

Phase I (con't)

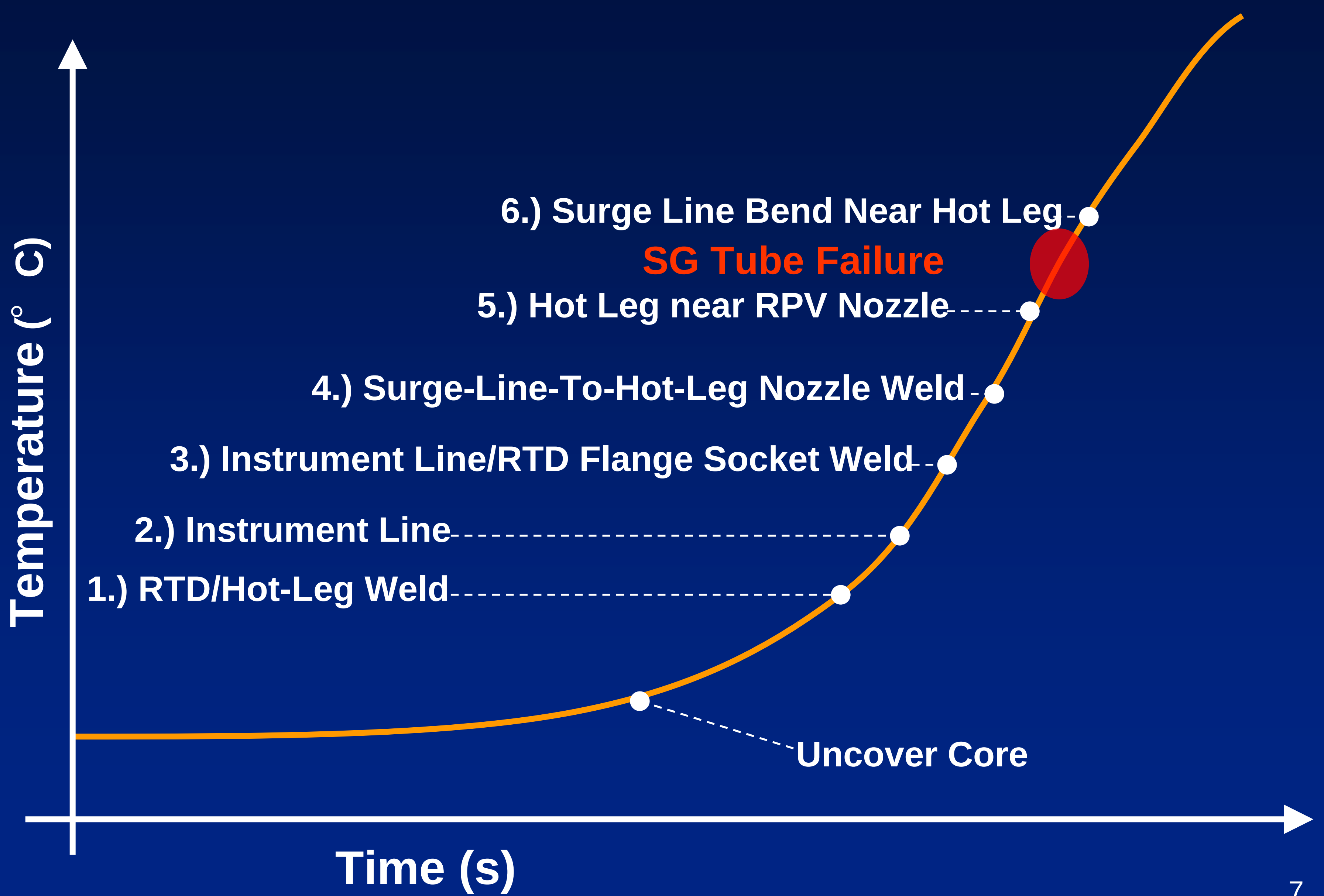
- Components selected for Phase II analysis:
  - hot leg and surge line (including nozzles and supports);
  - SG primary side manway;
  - top-dead-center resistance temperature detector (RTD) scoop that penetrates hot leg (including the welds);
  - socket weld connection of instrument line to RTD flange; and,
  - Pressure-operated relief valve (PORV) (plug–to–cage impact)

  - Reviews of operating histories of relief valves, bolted and flanged connections, and spiral-wound gaskets were also performed

Phase II

- Analysis based on Zion Nuclear Station
  - Hot leg and nozzles of Loop 4, including pressurizer and surge line, analyzed for reference station blackout (SBO) severe accident transient with "high-dry" sequence
  - Results from RELAP5 thermal hydraulic analysis of surface heat flux used as input for thermal-conduction and stress-strain analyses
  - Failure times due to tensile and creep rupture calculated with data from literature when available, and extrapolated when data were only available at lower-than-severe accident temperatures
  - Sensitivity analyses conducted to determine variability of predicted failure times due to variations of surface heat flux, thermal conductivity, creep rate, and yield strength
  - Also analyzed stress-strain response due to repeated plug-to-seat impact of typical PORV
  - Available high-temperature material properties data for components collected from literature, and temperature range over which data were not available was identified

Phase II (con't)



Phase III

- Improvements were made to thermal hydraulic modeling
  - Refinements made to surge-line-to-hot-leg connection in RELAP5 model
  - Thermal hydraulic data calculated using RELAP5 improved to account for entrance effects and flow reversals during PORV cycling
  - High-temperature materials database expanded by conducting high-temperature tensile and creep tests on stainless steel and carbon steel weldments.
- Enhancements changed calculated failure sequence
  - Resulted in hot leg failing first
  - Suggested that RCP seals could fail prior to SGTs
- Expert Workshop held to evaluate new findings
  - Agreed that seal failure could occur sooner than previously estimated and could possibly avert or mitigate containment bypass

Conclusions

- Improved models for determining time-to-failure of non-SGT PWR RCS components under severe accident conditions developed
- Times-to-failure between non-SGT (except RCP seals) RCS components were relatively close to each other
- Determined that RCP seals could fail prior to SGTs, which could avert or mitigate containment bypass
- NRR and RES looking at follow-on research

Questions?

Staff Closure of SGAP 3.10

- Not based on a specific ACRS recommended action in NUREG-1740
- NRC staff monitors plant operating experience through inspection process and reviews of results of licensee SGT inspections
- If analysis of future operating experience or research results indicates need to revisit this area, it will be considered and prioritized consistent with NRC budget process

SGAP TASK 3.1K
SGAP TASKS 3.4J AND 3.4K
SGAP TASK 3.12

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SEPTEMBER 24, 2009

ROBERT PALLA, NRR/DRA

Task 3.1k – Based on Tasks 3.1a – 3.1j, evaluate the probability of multiple tube failures in risk assessments for SG tube ARC

- Tasks 3.1a – 3.1j address physical processes that could cause SG tubes to open and leak (e.g., dynamic loads, bending stress)
- Staff concluded that loads from MSLB would not lead to additional leakage or rupture beyond that from ΔP loads alone
  - Tasks 3.1a – 3.1c: TH calculations to assess loads on TSP and SG tubes, and flow-induced vibration
  - Tasks 3.1d & 3.1e: evaluation of SG internal loading and flow-induced vibration displacement & frequency
  - Tasks 3.1f & 3.1g: crack growth calculations
  - Tasks 3.1h & 3.1j: additional TH calculations (not needed based on low loads for transients analyzed)
  - Task 3.1i: tests addressing effects of bending stresses

Task 3.1k (Continued)

- ACRS concluded that the analyses of MSLB have been completed and that SGAP Task 3.1 is closed, but this did not address 3.1k
- Objective of Task 3.1k – develop probability distribution for total SG leakage under ΔP loads alone
- Result would be used to support resolution of GSI-163 and PRA
- The need for this calculation was diminished for several reasons
  - Postulated phenomena associated with depressurization did not prove to be realistic
  - Performance-based TS provides reasonable assurance that DBA leakage will be small and well within that assumed in risk studies
  - Replacement SGs result in fewer flawed tubes left in service and fewer proposals to increase allowable leakage

Task 3.1k Conclusion

- The calculations planned under Task 3.1k are not needed to support closeout of GSI-163
- This task can be closed

Task 3.4j – Develop probability distribution for rate of tube leakage for ARC applied to flaws in restricted places

- Task 3.4i provided predicted flaw areas and leak rates from cracks under the TSP during MSLB and severe accidents
  - provided upper bound leak rates
  - showed that crevice deposits can reduce leak rates by factor of 1000
- As part of Task 3.5 SNL/SAIC developed a methodology to compute the probability of tube failure during an accident based on SG flaw distribution and RCS pressure/temperature history
- Example calculations under Task 3.5 assessed various defect types, including circumferential and axial cracks at the TSP

Task 3.4j Conclusion

- The SNL/SAIC model can be used to assess the impact of alternate assumptions or models for flaws in restricted places
- This effort has achieved the intent of SGAP Task 3.4j
- This task can be closed

Task 3.4k – Integrate information provided by Tasks 3.4a – 3.4j & 3.5 to address ACRS criticisms on risk assessments for ARC

- Specific concern: ARC that credit “indications restricted against burst”
  - Concern was specific to South Texas Project unit with stainless steel drilled-hole support plates (the only SGs of this type in US)
  - In depressurization event TSPs might move and expose flaws (no corrosion to restrict movement; tubes not clamped in place)
  - To limit displacement, tubes were expanded at various elevations
  - Staff estimated conservative leak rate of 5 gpm per burst tube within TSP region
  - Result could be included in a risk calculation but was not pursued because South Texas Project SGs have been replaced

Task 3.4k (Continued)

- Broader concern: Other SG tube integrity and licensing issues related to flaws in the free-span of SG tubes, and the ability to perform severe accident calculations in a technically defensible manner
- SGAP Task 3.5, “Develop improved methods for assessing the risk associated with SG tubes under accident conditions” was specifically intended to address this concern
- The methods and results developed through the RES effort on Task 3.5 provides insights into the risk significance of C-SGTR, as well as a foundation from which risk implications of future SG tube integrity issues might be assessed

Task 3.4k Conclusion

- Although additional research related to C-SGTR is planned, the work completed has achieved the intent of SGAP Task 3.4k
- This task can be closed

Task 3.12 – Review Insights from Task 3.5 and Assess Need for Completing Additional Regulatory Guidance

- The need for a risk-related RG on C-SGTR was identified in COMSECY-97-013 “Steam Generator Rulemaking”
  - Guidance would address how to make changes to SG licensing basis consistent with RG1.174
- Based on decision to endorse NEI 97-06 initiative in lieu of issuing a GL, proposed RG (DG-1073) was not completed
  - NEI 97-06 provisions ensure all SG tubes exhibit acceptable margins against burst/rupture for DBA
- Consistent with Task 3.12, staff has assessed the need for guidance on C-SGTR given insights from Task 3.5
- Staff concludes additional guidance and tools are still needed to support future risk assessments of C-SGTR

Task 3.12 (Continued)

- Decision rationale
  - Task 3.5 and other studies have not generically dispositioned the issue. Plant-specific PRAs should continue to address C-SGTR
  - Limitations of current work restrict its usefulness in supporting future risk assessments (flaw distributions, TH, documentation & tools)
  - Alternative methods have been developed by industry and are being used by licensees but have not been reviewed by NRC
  - Effectiveness of the peer review process in assuring technical adequacy of this PRA element is not clear

Task 3.12 Conclusion

- Additional guidance and tools are still needed to support future risk assessments of C-SGTR
  - Address acceptable approaches for modeling & quantification of C-SGTR in future risk models
  - Support staff assessments of risk implications of new licensee-proposed ARC
- Development of this guidance will be part of an RES User Need now in concurrence

User Need on C-SGTR

- Additional TH analysis
  - CFD and system code TH analyses for CE plants
  - Impact of incore instrument tube failure on C-SGTR
- Updated flaw distributions and RCS structural analyses
  - Distributions for remaining alloy 600 SGs and replacement 690 SGs
  - Finite element analyses of RCS components
- Guidance and tools for future risk assessments
  - Simplified tools and supporting documentation
  - Reassessment of $P_{C\text{-}SGTR}$ based on updated TH and flaw distributions
  - RG on RI-decisionmaking related to C-SGTR
- A document compiling/summarizing key research

STEAM GENERATOR ACTION PLAN
OPENING REMARKS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MATERIALS, METALLURGY, AND REACTOR FUELS SUBCOMMITTEE

SEPTEMBER 24, 2009

TIMOTHY J. MCGINTY, DIRECTOR
Division of Policy and Rulemaking
NRC Office of Nuclear Reactor Regulation

STEAM GENERATOR ACTION PLAN
BACKGROUND AND OVERVIEW

The SGAP, Staff Completion, Future Activities

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MATERIALS, METALLURGY, AND REACTOR FUELS SUBCOMMITTEE

SEPTEMBER 24, 2009

DAVID BEAULIEU, NRR/DPR

Steam Generator Action Plan History

- 1985-1990 NUREG 1150 studies first identify the issue of "consequential" steam generator (SG) tube rupture.
- For severe accident induced consequential SG tube ruptures, concern was that the high temperature gases created during core damage sequences could cause SG tubes to be the first component of the reactor coolant pressure boundary to fail, resulting in a potential containment bypass and the release of large amounts of radioactive material outside containment.
- NUREG-1150 quantified frequency in the low 10^{-6}/reactor-year range on the basis of expert elicitation.

SG Action Plan History (cont'd) Differing Professional Opinion

- In the early 1990s, the industry made several requests for relaxation of regulatory requirements for SG tube integrity.
- A Differing Professional Opinion (DPO) was filed involving concerns associated with this relaxation.
- Staff review of those relaxation requests identified that granting them might substantially increase the conditional probability of containment bypass during core damage accidents.

SG Action Plan History (cont'd)

- In the early 1990s, the NRC staff began a study of the effects of severe accident conditions on SG tube integrity as background information for a proposed new rulemaking on SG tube integrity.
- The results from this study, published as NUREG-1570, indicated that the risk is controlled by the current tube integrity requirements to a value that is low enough that no new rulemaking was needed.
- The DPO remained open.

SG Action Plan History (cont'd)
DPO Was Referred to ACRS for Resolution

- In 2000, the DPO was referred to the ACRS for resolution.
- After extensive public meetings and review of the issues raised in the DPO, the ACRS published NUREG-1740 to present its conclusions and recommendations.
- In particular, the ACRS concluded that the methodology being used to quantify the risk of containment bypass due to high-temperature challenges to SG tubes was "not technically defensible."

SG Action Plan (Section 3) Created to Address ACRS Recommendations

- Technical staff in NRR and RES jointly reviewed the full text of NUREG-1740 to extract the list of issues that required additional work.
- Those tasks were incorporated into a new section (Section 3) of the SG Action Plan .

SG Action Plan Tasks

- Most, but not all, SG Action Plan tasks involve severe accident induced SG tube ruptures.
- SG Action Plan also includes tasks work that involved design basis events, which addressed the potential for damage progression of multiple SG tubes due to SG depressurization. (e.g., during a main steam line break (MSLB) or other type of secondary side design basis accident).

Design Basis Event Tasks Closed – ACRS Review Complete

- The staff's work to address SG action plan items involving design basis events is complete, and;
- ACRS has previously reviewed and endorsed the closure of these items.
- Basis - Dynamic loads from such design basis events are low and do not affect the structural integrity of tubes or lead to additional leakage or ruptures beyond what would be determined using differential pressure loads alone.

SG Action Plan Status

- The staff has completed its work to close all SGAP items.
- Closeout documentation has been provided to ACRS.
- The purpose of this 2 day ACRS subcommittee meeting is for ACRS review of all SGAP items that ACRS has not previous reviewed and closed.

Desired Outcome of ACRS Review

- The staff requests an ACRS letter that finds acceptable the staff's closeout of each SGAP item that ACRS has not previously reviewed and closed which are:
  - SG Action Plan Items 3.1.k, 3.4, 3.5, 3.10, 3.11, and 3.12

Agenda

- Essentially all of the items are directly related to the work to define the risk associated with severe accident induced SG tube ruptures leading to containment bypass.
- This work involved the following technical areas of research:
  - thermal-hydraulics,
  - steam generator tube material failures;
  - reactor coolant system material failures;
  - component behavior studies, and
  - probabilistic risk assessment

SG Action Plan Closeout
Future Activities Outside of Action Plan Process

- SG Action Plan work is complete and, following ACRS review, the staff would like to close the SG Action Plan.
- NRR User Need to RES is in concurrence -- Requests specific research products to facilitate the development and review of future risk assessments involving consequential SG tube rupture events. These products will build upon analysis methods, tools, and expertise developed as part of the SG Action Plan.
- The RES work to address the NRR User Need no longer requires the level of coordination and agency focus required to implement the action plan process.
- Future work activities associated with this topic will be coordinated using other agency tools such as the User Need and the Planning, Budgeting, and Performance Management processes.