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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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643RD MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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THURSDAY

MAY 4, 2017

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ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

DENNIS C. BLEY, Chairman

MICHAEL L. CORRADINI, Vice Chairman

PETER RICCARDELLA, Member-at-Large

RONALD G. BALLINGER, Member

CHARLES H. BROWN, JR. Member

MARGARET CHU, Member

WALTER L. KIRCHNER, Member

JOSE MARCH-LEUBA, Member
DANA A. POWERS, Member
JOY REMPE, Member
GORDON R. SKILLMAN, Member
JOHN W. STETKAR, Member
MATTHEW W. SUNSERI, Member

DESIGNATED FEDERAL OFFICIAL:
CHRISTOPHER BROWN
DEREK WIDMAYER

ALSO PRESENT:
ALI AZARM, IESS
STEVE BLOSSOM, STPNOC
KEVIN COYNE, RES
VICTOR CUSUMANO, NRR
CANDACE DE MESSIERES, NRR
ROB ENGEN, STPNOC
CJ FONG, NRR
MIRELA GAVRILAS, NRR
FELIX GONZALES, RES
WAYNE HARRISON, STPNOC
SHANA HELTON, NRR
RAJ IYENGAR, RES
JOSHUA KAIZER, NRR*

ERNIE KEE, STPNOC
PAUL KLEIN, NRR
MICHAEL MARSHALL, NRR
DOMINIC MUNOZ, Alion*
MICHAEL MURRAY, STPNOC
ROBERT PASCARELLI, NRR
OSVALDO PENSADO, SWRI
LISA REGNER, NRR
DREW RICHARDS, STPNOC
ANDREA RUSSELL, NRR
SELIM SANCAKTAR, RES
MICHAEL SALAY, RES
RAY SCHNEIDER, Westinghouse*
WES SCHULZ, STPNOC
ASHLEY SMITH, NRR
STEPHEN SMITH, NRR
MARK THAGGARD, RES
ANDREA D. VEIL, Executive Director, ACRS
DON WAKEFIELD, ABS*
MATTHEW YODER, NRR

*Present via telephone

C O N T E N T S

P R O C E E D I N G S

8:30 a.m.

CHAIRMAN BLEY: The meeting will now come to order. This is the first day of the 643rd meeting of the Advisory Committee of Reactor Safeguards.

During today's meeting, the Committee will consider the following: number one, Risk-Informed South Texas Project License Amendment Request. That's GSI-191. Two, Consequential Steam Generator Tube Rupture. Three, preparation of ACRS reports.

The ACRS was established by Statute. And is governed by the Federal Advisory Committee Act. As such, the meeting is conducted in accordance with the provisions of FACA.

That means that the Committee can only speak through its published letter reports. We hold meetings to gather information to support our deliberations.

Interested parties who wish to provide comments can contact our offices requesting time after the Federal Register Notice describing the meeting is published. With that said, we also set aside ten minutes for spur of the moment comments for members of the public attending or listening to our meetings.

Written comments are also welcome. Mr.

Derek Widmayer is the designated Federal Official for the initial portion of the meeting.

The ACRS section of the US NRC public website provides our charter bylaws, letter reports, and full transcripts of all our full and subcommittee meetings, including slides presented at the meetings.

We have received no written comments or requests to make oral statements from members of the public regarding today's sessions. There will be a telephone bridge-line. To preclude interruption of the meeting, the phone will be placed in a listen-in mode during presentations and the Committee discussion.

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

I also want to make you aware that this meeting is being webcast with the ability to view our presentation slides on the web. If you're on the bridge line and want to do that, you can dial -- you can connect through the NRC's public meeting website, and click on the link.

It seems to work well, and the sound when

I tried it, is better than the sound on the bridge line. If you have any problems, please call our office.

At this point, I'm going to turn the meeting over to Professor Corradini to lead us through the discussion on the South Texas issue.

VICE CHAIRMAN CORRADINI: Okay. Thank you Mr. Chair. So, for the members, this is, I guess, the culmination of, I'm sure Lisa and Steve will tell us, I can remember at least a few years. A hand full. Maybe two handfuls of years in discussing GSI-191 with a risk informed methodology.

To remind the members, we had meetings back in 2012, '14, '15, and then two recently which culminated in the staff's SE. Which essentially goes through and analyzes and I think confirms, what STP has suggested is their approach for risk informed.

So, I'll turn it over to Lisa. No. I'm sorry. Excuse me.

MEMBER SUNSERI: Yes. And Dr. Corradini, before you do that, may --

VICE CHAIRMAN CORRADINI: I'm sorry, I forgot to turn to --

MEMBER SUNSERI: Yes. So, due to some prior associations, I find that I need to recuse myself from the deliberations on this topic. Thank you.

VICE CHAIRMAN CORRADINI: Okay. Thank you Matt. I forgot to turn it over to you. I forgot. Shana?

MS. HELTON: Thank you. I'll just give some brief opening remarks before handing off to Lisa Regner.

To provide the Committee some additional background. In 2010 the Commission directed the staff to consider a risk informed method for closing GSI-191. The Commission direction included specific direction to be creative and innovative.

That led in 2012 to what is now known as closure option 2B for generic letter 2004-02. Which is the potential impact of debris blockage and emergency recirculation during design basis accidents at pressurized water reactors.

South Texas Project was the pilot plant for exercising this option for closure of the generic letter. The lessons learned from this pilot effort have already benefitted and influenced the remaining plants using the risk informed closure option.

We recently received a submittal from Vogtle and expect preliminary closure documentation for review from at least one additional plant in the next few months.

I'd like to take a minute to thank some key staff, two of whom are sitting in front of you at the table here. Lisa Regner is the Doral Project Manager. Steve Smith is our technical expert in deterministic debris analysis. C.J. Fong, who is sitting in the audience, and Candace Pfeffercorn-De Messieres, have been instrumental on the PRA analysis.

There are a number of other people who have been involved, some of whom are on the phone, who have been instrumental to the effort as well. Technical staff who are not presenting today in the Division of Engineering, and the Division of Safety Systems.

It's taken a lot of people for us to get here today. And a lot of good efforts on behalf of the staff. And additionally, I'd like to commend the South Texas project staff and NRC contractors for being creative and collaborative in addressing the challenges that inevitably arose along the way.

The many meetings, audits and site visits conducted over the last few years were critical in ensuring mutual understanding. And led to the hybrid method you will hear about today.

And finally, I'd like to express appreciation to the ACRS members past and present who have led us where we are with many thoughtful insights. And we've presented to the committees and subcommittees several times throughout the review process over the past four years or so.

We've really taken the ACRS insights to heart. And I think what you see in this staffs' SE is a reflection of how we've addressed the ACRS concerns to date.

With that, I'd like to turn things over to Lisa Regner. Thank you for the opportunity to be here today.

MS. REGNER: Thank you Shana. Good morning. I moved it so I wouldn't forget. Sorry.

Thank you Shana. Good morning. I'm Lisa Regner, the Project Manager for the South Texas Project pilot -- licensing. I'm the Licensing Project Manager for the South Texas Project.

As Shana stated, we're here today to present the staffs' results of a pioneering action to risk informed compliance with the regulatory requirements for emergency core and containment cooling considering debris.

Like most pilot projects, this review was a series of stops and starts and course corrections. But over the past year we've been able to make several leaps forward. My goal for the first few minutes is to share that voyage with you.

But before I start, I would also like to acknowledge the NRC team, both present and absent, that got us here. This is the best group of professionals that I have ever had the pleasure to work with.

And I would also like to acknowledge the South Texas Project Nuclear Operating Station staff and contractors. They are responsive, hardworking and intelligent professionals. And big-hearted Texans as well. For the agenda --

CHAIRMAN BLEY: And what?

(Laughter)

MS. REGNER: For the agenda today, we'll cover the background of the risk informed option to address GSI-191. And an overview of the STP review project.

You'll hear from the STP team. And then the staff will get into the fun technical aspects of the review. For the first few minutes however, I want to cover a bit of the history, the methods used by the licensee and the staff, and the remaining actions for us to complete this action.

So, in terms of the background, Shana gave you a real nice, high level quick look. I'm just going to go just below the surface.

And the first -- the staff first identified concerns with debris and containment as far back as the 1970s. Over the last 40 years, important efforts have been made by the industry in analysis, testing, and redesign.

And installation of upgraded design features for sumps and strainers to try to resolve the issue. Progress has been made. But, not enough to close the generic safety issue for our plans.

In fact, new concerns were identified along the way. Such that closure of the generic letter associated with GSI-191, involves licensee to -- licensees to demonstrate compliance with 10 CFR 50.46, which, as I had said before, is the emergency core cooling system performance criteria considering debris and containment as well for both strainer impacts and in-vessel effects.

And I'll talk a little bit more about that. That those two separate issues to resolve GSI-191. Because I want to make that clear for everybody that hasn't been involved in the subcommittee discussions.

In 2010 the Commission directed the staff to provide options to resolve GSI-191 and close the generic letter, 2004-02. And consider risk informed opportunities to do so.

And the staff developed three options, as Shana mentioned. Option one provided the traditional closure method based on existing models, using deterministic methods and providing near term closure. In fact, all 18 of those plants choosing option one, have been closed.

Option two, was to provide -- uses a graded approach based on the amount of insulation in the plant. And provides licensees with two paths allowing longer term closure.

The first aspect of option two is mitigative measures as well as either 2A, term 2A, the deterministic option, which allows for refined in-vessel testing. And then 2B as you know, is a new methodology employing risk information. Which STP choose.

VICE CHAIRMAN CORRADINI: So just a clarification. Under 2A, how many plants do you expect?

MS. REGNER: Twenty-eight units.

VICE CHAIRMAN CORRADINI: Under 2A? Okay.

MS. REGNER: Yes, sir.

VICE CHAIRMAN CORRADINI: Okay.

MS. REGNER: Okay. Any other questions?

(No response)

MS. REGNER: Option three, plants may use deterministic methods for strainer impacts. And a risk informed resolution for in-vessel impacts. And that's two units that are -- want to use that.

In addition to STP's use of option 2B, mitigative measures and risk informed evaluation, this slide also provides some of the other plants that have submitted their intent to use option 2B.

VICE CHAIRMAN CORRADINI: So, just to clarify. So under 2A, 28. Under 2B, the list is here?

MS. REGNER: Correct.

VICE CHAIRMAN CORRADINI: Okay.

MS. REGNER: Yes. Okay. So, was far as now --

VICE CHAIRMAN CORRADINI: How many under three? Two?

MS. REGNER: Two plants.

VICE CHAIRMAN CORRADINI: I'm sorry. I missed that. Excuse me. I'm sorry.

MS. REGNER: Yes. Two plants under option three. Okay?

MR. SMITH: I'll just say option three, the one plant that's two units that came in under option three is probably not going to actually use option three. So they maybe 2B.

They just came in and they were doing it. They proposed something that was different then was in the SRM. So, we're probably not going to accept them as option three.

So, they're probably going to have to change. But that's what they've declared.

MS. REGNER: Thanks Steve. So now focusing a little more on the South Texas review specifically, they -- the STP pilot project began six years ago in late 2010, early 2011 when STP formally submitted its intent to use a risk informed option to resolve GSI-191.

The staff hosted several public meetings to discuss STP's risk informed approach. In fact the STP method was developed enough that a description was included in the staffs' proposal to the Commission on GSI-191 closure options.

Shortly after the Commission's approval of the closure options, of the three resolution options, STP submitted its licensing action request for review. Now, originally the STP request was fully risk informed, providing the estimated change in risk, without removing debris generating material in containment.

And the most detrimental debris of concern is fibrous insulation on piping and components. The STP approach attempted to characterize the physical behavior of debris generation and transport over a full range of conditions using a platform called containment accident stochastic analysis or CASA Grande.

And I'm sure the STP staff will go into a little more detail then I intend to for this platform. The CASA Grande platform developed by STP's contractor Alion, is designed to model up to 50 different parameters to compile a spectrum of time dependent results for many thousands of postulated accident sequences.

Ultimately, CASA Grande can provide the change in risk for the actual 3-D modeled STP plant compared to the postulated clean plant without debris generating material.

Now, the full risk informed model proved to have too many uncertainties for the staff. And an alternate method was developed. But I'll talk about that in just a minute.

As expected, this project has become one of the most resource intensive, risk informed reviews undertaken by the staff. For example, experts from five divisions, 14 branches in the Office of Nuclear Reactor Regulation have contributed to this safety evaluation or supporting document such as the environmental assessment, which was issued yesterday for publication in the FRN.

And that will show up in about two weeks in the Federal Register Notice. And that is associated with the exemptions that STP has requested.

We posted over 40 public meetings. And asked more than 400 questions. Although many of those questions answered by STP have been superseded when they submitted their alternate methodology.

MEMBER POWERS: Lisa, you identify this as a resource intensive for the agency. And -- but you have about ten more plants that propose to do this.

Have you attempted to identify areas where improved methods and technology, say generated by research, could reduce the resource intensity of these reviews?

MS. REGNER: Absolutely. I would probably turn to either Vic or Steve if they'd like to provide, you know, details of that.

MEMBER POWERS: Yes. I mean, I ask because it might help us identify where research resources could be focused and what not. And in lieu of, I mean, if you have a quick response, that's fine.

But it would be useful perhaps to document it in a memorandum or something like that. What -- the general areas where some additional research intensity could reduce these resources.

Because you do have ten plants looking to do this. And I don't think you want to have five years and 43 meetings and six thousand pages of RAIs going out if we can with just a little bit of research effort give you -- put in your hands, better technology for doing this.

MS. REGNER: Um-hum. Well, obviously the overall project -- the overall project itself, there were many, many lessons learned on uncertainties and correlations that the staff just won't accept.

And so that information is out there. There have been several plants that have followed in the public meetings, so they already know that they don't have to go down that road.

So just in simple terms, in those terms, we have --

MEMBER POWERS: Well, yes. This new -- there's no question there's some learning goes through -- anybody that goes through the first of a kind operation will learn something that other people will take advantage of.

But, my question is more pertinent to the agency.

MS. REGNER: Right.

MEMBER POWERS: And how it goes about doing its work to review these materials. The idea being if there's improved technology that would just make it easier and more efficient to do the mechanics of the work that -- and at the cost of doing a little research, we ought to do it.

MS. REGNER: Um-hum.

MEMBER POWERS: In preference to research that's not so useful or something like that.

MR. CUSUMANO: Yes. If you don't mind. This is Vic Cusumano. I'm over -- way over here.

The efficiencies that we expect to gain are going to come from, in my opinion, the WCAP 17788 that we're looking at. The vast majority of the follow up plans if you will, are going to rely on that WCAP.

So we're going to have one methodology that they all can use. South Texas Project was basically a one off of that. Which is why it was so resource intensive.

So, just having a topical report that all the plants can use and in addition to that, as we develop it, we fully expect to have a model submittal worked out with industry. So it will be a standardized submittal and as much as possible, a standardized response.

So yes, efficiency has been forefront in our mind with this many plants to go. Both on Two Alpha and Two Bravo. We did leverage research to some degree during this. Especially as it related to some of the issues in the core that weren't being modeled all that well.

And Trace was a big help to us there. Thanks to Dr. Steve Pajoric (phonetic) in large part.

MEMBER KIRCHNER: May I ask a question just to -- your choice of words in the first bullet make me ask almost that fully risk informed would lead one to believe what?

MS. REGNER: Yes. All it provided was a change in risk. In considering the effects of degree. Okay?

So those -- and now -- and that's where the CASA Grande, you know, instead of developing a threshold that they, you know, when I get into this hybrid methodology as Shana called it, that develops a threshold that showed that the majority of the possible breaks meet the deterministic criteria. Which is our traditional acceptance methodology.

Okay? However, when they first developed this, they compared their existing plant with a clean plant. Okay? And then analyzed the debris and generation -- debris generation in transport, right, for all of the possible breaks to determine the risk change.

Maybe I'll let --

MR. FONG: Yes I was -- maybe I can weigh in on that a little bit. CJ Fong from the Division of Risk Assessment.

If you look at the original submittal, it treated a lot more of the variables probabilistically. But it also considered a lot more sequences.

And the newer version of the submittal uses, in certain cases, more conservative screening values, point estimates for some of the variables. And screens a lot of the sequences deterministically based on test data.

So, they're both risk informed. It's just that the scope of the original submittal in terms of risk information was wider.

MEMBER KIRCHNER: Okay. Thank you. No, it's just I think what we heard in previous meetings makes a lot of sense. You use both techniques to advantage.

I just -- the implication is it's not so well risk informed when you go to option 2A or 2B or whatever. I just -- I'm just reacting to the choice of words. Thank you.

So is the new method called CASA Pequeno?

(Laughter)

MS. REGNER: We should use that. We'll do. As expected, -- we talked about that.

The NRC conducted 13 audits at various sites to support the review. Including several in 2008 and '09 to observe STP specific testing.

More recently the staff conducted audits to observe piping layouts, quantity of insulation and debris flow paths in STP containment. As well as thermal-hydraulics and risk audits. These were invaluable in understanding the details of the new methodology and resolving concerns.

So the licensee's methodology, and again, this is the post-methodology change, the -- well, and let me go back to the original risk informed approach did have merit.

It became apparent however that there was just too much uncertainty even with plant specific testing requested by the staff to support some of the statistical distributions. The STP model was not able to show accurate modeling in several areas such as head loss, chemical effects, debris transport timing, and others. There were issues as well with epoxy coatings and the in-core analysis of debris impacts.

The licensee's methodology change was the significant turning point. As you said, that was when they did leverage the deterministic testing that they had done, to simplify the process.

This new process was termed risk over deterministic, or RoverD. And it provided the benefits of simplifying the staff's review and reducing a significant amount of uncertainty.

This graphic created by STP provides a good overview of the key elements which the licensee will discuss in more detail. The first element, the deterministic test data incorporates the licensee's plant specific testing.

As I said, they used staff approved

methods. And the staff observed some of the testing. This testing established the debris threshold for the emergency core cooling and containment spray systems. Both of which rely on the containment sumps to remain functional through long term core cooling.

CASA Grande used it a more limited scope then before. Evaluated several thousand break scenarios to determine the amount of debris generated and transported for each break size, orientation, and location in containment.

These calculations and -- so that's the CASA Grande calculated degree for individual breaks. These two were then tested.

So, if above the threshold it went into the risk informed analysis. If below the threshold, it met the traditional deterministic acceptability bin.

It's important to distinguish between the two separate evaluation segments. However, to be addressed in GSI-191, first the impacts of debris clogging at the sump strainer, and second, the debris impacts to the in-core thermal-hydraulic environment.

The evaluations overlap and deterministically dispositioning a majority of those breaks. And both in-vessel and strainer segments use risk to disposition all large hot leg breaks. Those are breaks greater than 16 inches.

Where they diverge is in the deterministic methods for each. For the strainer evaluation calculated debris amounts from CASA Grande are compared to deterministic testing thresholds, as is represented here in the graphic.

For the cold leg in-vessel evaluation, the debris amounts are compared to previously established debris thresholds in NRC approved guidance. That's WCAP 16793.

For the hot leg in-vessel evaluation for small and medium sized breaks, the licensee choose to use a calculational platform. And that's RELAP5 3-D.

Since RELAP5 3-D has not been approved for this application, the staff's safety evaluation, as you probably saw in enclosure two, provides our review of the plant specific simulations using RELAP5 3-D. And this will be discussed in a little more detail following STP's presentation.

I do what to note however that the staff's SE did not approve the evaluation methodology for generic use. It only looked at simulations specific to STP.

Slide seven is the staffs' methodology.

And what we were comparing against, our acceptance criteria, obviously is 50.46, the ECCS performance criteria. Everything else is guidance associated with our evaluation in NEI 0407, the WCAP I mentioned earlier, Reg Guide 1.182 and Reg Guide 1.174, which provides the criteria. It's based on the Commission's safety goal policy statement for risk.

The technical specification is relatively simple. It recognizes the debris assessment specifically for the emergency core cooling and containment spray systems.

The new shut down action statement provides for immediate compensatory actions if debris is identified in containment in excess of the analyzed amounts. And requires the system to return to operable status within 90 days.

The structure of the staffs' SE uses the five key principals of risk informed regulation. Which is from Reg Guide 1.174.

This guidance provides an acceptable method to assess the impact of licensing basis changes using risk. And provides consistency in areas where risk is used in regulatory decisions.

It specifies a method that compliments the deterministic approach and supports the NRC's traditional defense in depth philosophy.

MEMBER SKILLMAN: Lisa, will we have an opportunity to discuss how the operators will interpret the new technical specification?

MS. REGNER: Sure.

MEMBER SKILLMAN: Will it be with the staff or with the -- with STP?

MS. REGNER: I guess we'll let STP start. And if they can't answer your questioning, we can talk more.

MEMBER SKILLMAN: Thank you. Thank you.

MS. REGNER: The staff methodology again, this is a graphic, a simple graphic that shows you the five key principals of risk informed regulation. Which is how we structured the SE.

The -- let's see, -- and again, this whole review was specific to the effects of debris on 50.46 as specified in the generic letter.

The primary guidance documents to show compliance with 50.46 ECCS -- oh, I'm sorry. I went backwards. Excuse me.

This slide again, provides a graphic of the five key principals. The first key principal in white involves meeting existing regulations or modifying the rules through rule making.

Key principals two and three, the blue boxes, leverage deterministic criteria. And for this review, involved the staff from Division of Safety Systems and Division of Engineering.

Key principals four and five, the tan boxes, -- oh, risk expectations. And appropriately involve staff from the Division of Risk Assessment.

The NRC staffs' technical presentation following South Texas Project reflects the structure of the SE by presenting each key principal. These are not our final conclusions, but a summary of the staffs' preliminary results for your consideration.

Concerning remaining actions, the staff is completing the legal review to finalize the concurrence process for the licensing action requests. The final environmental assessment was issued yesterday as I said.

Once we address the ACRS items of concern, the staff will be ready to issue the final decision sometime this spring. And this concludes the background and overview portion of the staffs' presentation.

Unless there are questions, we will return following the licensee's presentation.

VICE CHAIRMAN CORRADINI: Questions by the members?

(No response)

VICE CHAIRMAN CORRADINI: Okay. Why don't we switch then -- who's up front? Who's going to kick it off? Wayne or Mike? Fine. Sorry. Excuse me. Go ahead.

MR. MURRAY: So, good morning. I appreciate the opportunity to address the ACRS again. I'd like to second Lisa's -- I'm Mike Murray, Regulatory Affairs Manager at South Texas Project.

I'd like to second Lisa's comments of the collaborative nature of the effort that we've all put in. We've had a lot of industry expertise that's been involved with it.

And interacting with the NRC staff is always in a collaborative nature. The audits went very well. Were very beneficial to all of us on that.

And also I'd like to appreciate the ACRS for the interactions and challenging questions as we've gone through it. So much appreciate it for that opportunity.

So I'll let the rest of the team introduce themselves, starting with Ernie.

MR. KEE: Ernie Kee, South Texas --

MR. HARRISON: Wayne Harrison, South Texas

Project.

MR. KEE: I'm sorry. Ernie Kee, South Texas Project.

MR. HARRISON: Wayne Harrison, South Texas project.

MR. SCHULZ: Wes Schultz, Mechanical Engineer at South Texas.

MR. MURRAY: We have some other members from South Texas Project that would like to introduce themselves.

MR. RINKEREL: Good morning. My name is Dave Rinkerel. I'm the Executive response for the effort.

MR. INGAN: Rob Ingan, Manager of Engineering Projects at South Texas Project.

MR. BLOSSOM: Steve Blossom, GSI-191 Project Manager, South Texas Project.

MR. RICHARDS: Drew Richards. I'm STP Licensing Engineer.

MR. MURRAY: And on the phone we have Don Wakefield with ABS if we should need his -- assistance from him. And Dominic Munoz from Alion. They're available as well. Next slide, please.

So meeting purpose, review and overview, which was the request of the process we went through and the technical information. And we'll also describe the risk informed treatment of debris and the current risk over deterministic, RoverD methodology. Next slide, please.

So in our agenda what we'll touch on so you can see where we're planning to drive through this, is we will look at the STP GSI-191, generic letter 2004-02, option two bravo. We'll general overview of the evolution of the application. Which Lisa had covered a good bit of it as with -- on with the opening.

And then general overview of the RoverD methodology. Testing, deterministic element of it. Determining and the determination of the governing brake size. How we went about that.

And in-vessel effects. Quantitative results and regulatory implementation. And that's -- I think that's where we'll be able to answer your question on -- in that area, so.

And then I'll do the closure comments on it. So with that, I'll turn it over to Wayne Harrison to start carrying us through the first part of the presentation.

MR. HARRISON: Thank you Mike. As Lisa mentioned, South Texas Project, Unit Two, we have a large amount of fibrous insulation in the reactor coolant system. And in order to meet a deterministic threshold value for containment debris loading, the amount of debris generating contributors in the plant design would need to be significantly reduced.

This would result in a real burden with respect to occupational dose. Primarily with respect, it would be almost 88 rim per cycle, per unit or for that cycle for removing and banding insulation. Compared to a -- like a 20 rim normal cycle dose.

So we're talking about real dose to real people to do this modification. The cost was not insignificant at about 55 million dollars.

And I'd also point out that as Lisa had mentioned also in hers, her presentation, that we had already taken a number of mitigative actions, significant mitigative actions. Primarily the replacement of our three original 155 square foot strainers with three new 18 hundred and 18 square foot strainers.

And in addition to the generic letter of 2002-02 -- 2004-02 response, South Texas Project has been very proactive with respect to the initiating event, the pipe break frequency. We replaced the steam generator safe in wells with alloy 690 material when we replaced our steam generators.

We overlaid the pressurizer wells with alloy 690 material. And in Unit One we just finished doing the mechanical stress improvement process on the reactor pressure vessel butt wells, or the nozzle wells.

And we will be doing that in our coming outage on Unit Two. So we've been very proactive in our activities.

Lisa covered most of -- going onto seven.

VICE CHAIRMAN CORRADINI: Wait, before you go -- can you just go back? I know you've said it. But I can't remember when. When did you replace the strainers from 155 to 1818?

MR. HARRISON: Wes, do you --

MR. SCHULZ: I believe it was in 2006.

VICE CHAIRMAN CORRADINI: Yes. I was guessing it was a while ago.

MR. SCHULZ: Yes. It was.

VICE CHAIRMAN CORRADINI: This was part of the industry's --

MR. SCHULZ: In 2006 and 2007, we replaced them.

VICE CHAIRMAN CORRADINI: But this was part of an industry, this is not just you all?

MR. SCHULZ: That's correct.

VICE CHAIRMAN CORRADINI: Okay. Fine.

MR. HARRISON: That's correct.

VICE CHAIRMAN CORRADINI: And then the third bullet, I remember that this one I had not caught. And one of our consultants caught it.

When was that replacement made? Do you know?

MR. SCHULZ: You know, I --

MR. HARRISON: The Marinite replacement, Wes.

MR. SCHULZ: That was about two years after the strainers. So, it was about 2009 time frame.

VICE CHAIRMAN CORRADINI: Okay. All right. Thank you.

MR. HARRISON: Thanks Wes. Lisa covered most of this. So, I'm not going to go through the detail of the original.

As she said, the original application was a full risk informed where CASA Grande basically provided conditional failure probabilities to the PRA. And with respect -- in that there were issues with the modeling as Lisa described.

And so we reduced the complexity. So we

determined that we could take advantage of our good deterministic testing that we had performed, and apply the tools that we had with CASA Grande, which does other then -- which does a good bit of analytical work for us.

And to show that there was a small risk of having a break that generates more debris then what our deterministic test showed. And that's what we ended up with.

And that's a bounded analysis that we used, RoverD. And Ernie Kee is going to describe that. Just give a brief overview of the RoverD analysis.

MR. KEE: So this is Ernie Kee. And we'll move to slide nine.

And Lisa's already described this in some detail. I might mention here that the basic deterministic test data were WCAP 16793 for in-core hot leg breaks -- oh, cold leg breaks. And we have the strainer tests of 2008 that Wes Schulz will go into in detail later.

I might mention also that CASA Grande, it basically is a flexible analytic framework that can be used in a -- to support a full risk informed mode. To your question, Dr. Kirchner.

So earlier, in fact at the end of 2011, I would say that full risk informed also implies maybe best estimate inputs into the -- into a PRA. And we used that CASA Grande framework in that mode to supply those inputs.

And so it's more of a best estimate. For instance we, based on previous testing, didn't take into account chemical effects that both NRC tests. And then later we confirmed that we could reasonably ignore that kind of effect on a best estimate basis.

So we use CASA Grande now in what we call a deterministic mode. Where we obey all the guidance that was mentioned in the IO-407. These kinds of guidance in that application.

And then we basically generate many, many scenarios. And as Lisa also mentioned, some of these meet deterministic criteria, others don't. And the ones that don't, we put into the category of risk informed.

So we've kind of devolved this problem into two kinds of categories. Either risk informed or deterministic. And that was the main simplification that came about with this RoverD. Slide ten then.

So I think this has been said. That the RoverD framework simplifies this very complex risk assessment of many thousands of scenarios by using deterministic test data and bounding analysis.

So we've talked about in-core analysis. And for example a hot leg breaks, we assume a full and complete blockage of the core and core bypass. Which is a theoretically bounding approach to that in-core behavior.

We bound them -- we create bounds on uncertainties to make the assessment tractable, reviewable, and easily understood. I think that's the kind of a core idea here that we have an understandable way to view this, this risk.

And the PRA from the South Texas Project along with CASA Grande was actually used quite extensively in this project. Primarily now we have the ability to look at the success frequencies.

For example, different thousands and thousands of configurations of equipment for example. And so that's how we use the PRA for that. And to understand the largely release frequency, given a core damage frequency. And now we're on slide 11.

So at a high level we ensure the tested fine fiber which we've ensured bounding all other types of debris species, is either met or if it's not, we assign it -- we assign those scenarios that don't meet that criteria to risk inform.

And those that are in the risk informed -- the scenarios that land in the risk informed category, we do some additional analysis. But we basically assign those to core damage.

So it's again, kind of a bounding concept. And finally, we confirm the containment integrity is maintained for defense in depth. Even though we don't expect to get there, but should we get there, we check to make sure that the concerns raised in GSI-191 don't lead to a containment failure.

And with that, I'll turn it over to Wes Schulz.

VICE CHAIRMAN CORRADINI: So, I didn't know where to bring this up. But since you brought up containment.

So, other members in particular who know the system better then I sometimes, reminded me that South Texas, I'll use the word unique, but has essentially the containment fan coolers. Which if containment spray failed, containment fan coolers could provide the appropriate heat decay removal function that allows bullet three to be maintained.

MR. KEE: Yes, sir.

VICE CHAIRMAN CORRADINI: Is that a fair

--

MR. KEE: Yes, sir. They're independent if you will, from those kind of concerns that are --

VICE CHAIRMAN CORRADINI: Okay. So this doesn't apply to you, but eventually we're going to get around too somewhere if we have time, how what you do can be applied to others.

But this containment fan cooler attribute is unique to South Texas. It doesn't apply across the board in many of the PWR containments.

MR. KEE: I can't make a claim for other plants. But I think the --

VICE CHAIRMAN CORRADINI: I don't remember it's --

MR. KEE: Many have a similar.

MEMBER STETKAR: In my experience --

VICE CHAIRMAN CORRADINI: The only one I remember is Zion.

MEMBER STETKAR: It's neither unique to South Texas nor is it ubiquitous throughout the fleet. Some plants --

VICE CHAIRMAN CORRADINI: In between.

MEMBER STETKAR: Have them, some plants don't. So, the plants that don't, have containment fan coolers that would be available during these types of scenarios.

The conditional large release probability would be, I don't want to say guaranteed, but it would be very close to one given core damage and sump plug in.

VICE CHAIRMAN CORRADINI: Right. Okay.

MEMBER STETKAR: So that it's -- you need to look -- they're fortunate because they do have it. So therefore there's a distinct numerical separation between the amount of -- if all of the sumps plug, they still have a lot of margin before they can reach containment failure.

VICE CHAIRMAN CORRADINI: And that was one. The second things, because you can guess the member who reminded me of all this, you have three sumps. That also, I won't say use -- he was very good at not unique versus ubiquitous.

But three is not necessarily what I'd see in others. I don't remember any plants with four. But I remember, and I've been reminded that most plants have probably two sumps.

Is that fair?

MR. KEE: Yes, sir.

VICE CHAIRMAN CORRADINI: Okay. Okay. Fine. The only reason I'm bringing that up is, how this, which I'm -- somebody used the word hybrid, which I liked, this hybrid risk informed approach is, to me, quite interesting.

But then it's applicability very much determined, is a function of what equipment and geometry it's applied to.

MEMBER STETKAR: Now just, I'd say that the approach is applicable to any plant with any configuration with any number of trains. The complexity of applying it becomes more difficult as I have larger numbers of trains and let's just say interesting configurations of the sumps. Interesting hydraulic configurations now of the sumps.

VICE CHAIRMAN CORRADINI: Well, I think John said it better then I. But thanks. Let's keep on going. I just wanted to make sure your third bullet that I had it in my mind.

MR. SCHULZ: Good morning. This is Wes Schulz. I'm going to talk about a deterministic element of our RoverD approach. And also a head loss -- the strainer head loss testing.

Let's go to slide 13. And that's a photo inside containment. South Texas is called the high fiber plant because we have fiberglass insulation over virtually all of our insulated piping and equipment inside containment.

You can see from the photo here we have -- fiberglass blankets on the piping. The trade name is NUCON.

On the right-hand side there is -- in the back, is the reactor coolant pumps. So the equipment has fiberglass insulation on it also. The stream generators have fiberglass insulation on them.

In the foreground there is, I think that might be my head. I was showing this slide to my wife. I told her that's definitely me. But to make her happy, but okay. Let's go to slide 14.

This is a picture of our original design strainer, we mentioned that. Put some scale on this, the top of the strainer is about four and a half feet off the floor.

Underneath that strainer is a sump pit. The pits are below floor level. And we have three individual sump pits, one for each train.

The strainer consists of perforated plate on four sides. The hole size is a quarter inch. Let's go to the next slide.

Fifteen shows our new strainers, which we installed like the rest of the industry, in response to the generic letter. These -- we have 20 strainer modules per sump. You can see them here.

Again, the scale of this, the top of the strainer is about two and a half feet of the floor. And again the sump pit is underneath, is below this.

The bottom right picture shows some protective grading in front of the sump. That's not for hydraulic reasons. That's more just to protect it during outages when we're moving equipment and material in that area. So we want to protect our sumps there.

And the whole size for these is .095 inches. A much smaller hole size for the new designed strainers. Let's go to slide 16.

This is a figure from our CAD model. We developed this CAD model through the years. It's become very sophisticated.

We modeled the configuration of all the piping inside containment with the associated insulation type and the insulation thickness. We modeled in the equipment, equipment supports, equipment insulation.

And we show where the coatings are, the type of coatings. We've got the structural steel in there. We also modeled in the concrete walls, the floors, the grating of the floors.

A very sophisticated picture of our physical configuration at South Texas. And this picture shows the zone of influence for a 31-inch pipe break. That you can see that. Get some perspective on what the ZOI for this 31-inch pipe break is.

Let's go to slide 17.

MEMBER MARCH-LEUBA: So wait. Can you go back in there? So some of us are not familiar with the whole methodology.

You will assume that all of the insulation in that sphere goes down the drain?

MR. SCHULZ: Yes. All that insulation in that sphere is affected by the pipe break. And we would apply the transport rules for moving that debris to the sump.

I mean, --

MEMBER MARCH-LEUBA: And that transfer rule is two percent of it? Or five percent?

MR. SCHULZ: Well, it depends on, for a given size there's some -- we get large pieces, we get small pieces, and we get fine. So, depending on how far away it is from the break, there's a certain amount of insulation.

And each insulation has its own transport type. But for our purposes, for our RoverD approach, we're looking at fiber fines, which readily transport. They're the small fibers that will clog stuff.

So, that's what we'll do.

VICE CHAIRMAN CORRADINI: But to answer Jose's question, the testing in 2008 determined, was meant to determine what gets there versus what settles out on the way there.

Is that a fair point?

MR. SCHULZ: Well, the 2008 test showed that for this amount of fiber fines, we had an acceptable strainer head loss. So, transport didn't really enter into it.

Did I answer your questions?

MEMBER MARCH-LEUBA: Yes. But that's not the transport. That is how much fiber gets into the mesh.

MR. SCHULZ: The actual mesh. Yes.

MEMBER MARCH-LEUBA: But in other monitor, you run these calculations? Did you transport five percent of the mesh? Or the fibers? Or none --

MR. SCHULZ: Oh, no. Like over 95 percent of the -- all goes yes. About everything goes.

MEMBER MARCH-LEUBA: Okay. So you use 95 percent then.

MR. SCHULZ: Yes.

MEMBER RICCARDELLA: And was that established by testing? What -- you said there was different amounts depending on how far you are from the break.

How was that established?

MR. SCHULZ: That was established by a test, right. Alion did some testing for that. Which the staff reviewed and --

VICE CHAIRMAN CORRADINI: But not the 2008? Not the 2008 testing? Separate testing.

MR. SCHULZ: How we -- for a given -- for the zone of influence, how much large pieces, small pieces, fine fibers are generated. That's the standard that there is.

MR. RICCARDELLA: And for different types of insulation too, you said?

MR. SCHULZ: Yes. It would apply too different. Particularly ZOI for the particular debris type. This is for NUCON insulation. There's another one for coatings.

There's another one for our Marinite and another one for Insulite. And those are established in the guidance that we've evolved.

VICE CHAIRMAN CORRADINI: And part of the reason I went back to one of the bullets that Wayne had mentioned, in the calculations done, Marinite is still considered.

MR. SCHULZ: Yes. I'll address that. Yes. I'll get there.

VICE CHAIRMAN CORRADINI: Okay. Fine. I just wanted to make sure to connect it back to the bullet that what had been removed but is still considered in the analysis.

MR. SCHULZ: Right.

MR. KEE: This is Ernie Key. I just wanted to make sure that we understand. That zone of influence that is indeed spherical, but we do -- CASA takes care of this.

If there's a big concrete wall in the way, it doesn't go through that. But the pipes themselves are transparent to that. So, but --

MR. SCHULZ: Yes. So there's wall shadowing but not equipment shadowing.

MR. KEE: Correct.

MR. SCHULZ: Okay. Right. Okay. Let's go to slide 17, Wayne.

So this is the -- a deterministic evaluation followed the NEI guidance approved by the staff. And we submitted that in the format to -- as the staff put out this content guide.

So these are the elements of our response, of our deterministic evaluation. And these -- our break selection initially for the test we had back in 2007, we had three or four, we actually had four different large breaks.

And we took the maximum, looked at the debris for each break. And we took the maximum amount for a particular debris type for our test. We tried to use a bounding amount in our test that would overlap that.

But again, that's sort of immaterial going forward. Because the test is just the test for that amount of debris. And we used that as a benchmark going forward.

Now, as Ernie mentioned, break selection, we have breaks at thousands of places and thousands of different size breaks at those locations.

VICE CHAIRMAN CORRADINI: But just to repeat so that -- for the members that weren't at the April 5 subcommittee. These tests determine a threshold limit that above which strainers have -- are challenged, below which strainers are not challenged.

MR. SCHULZ: Yes. It was the limit that was successful. We didn't test to a --

VICE CHAIRMAN CORRADINI: I understand.

MR. SCHULZ: That's official. That threshold mark --

VICE CHAIRMAN CORRADINI: That's a threshold limit.

MR. SCHULZ: That's a measure for success.

VICE CHAIRMAN CORRADINI: Okay. Thank you.

MR. SCHULZ: We assume it fails after. Everything about that fails.

So, the same thing with debris generation. Well, we talked about the different COIs. We have all that in our current analysis for that.

Debris characteristics per the NEI guidance, latent debris, we did a walk down to confirm that the number we used from the NEI guidance was in fact bounding.

Debris transport, again, in our current analysis we assume all the fiber -- most of the fiber finds are readily transported. The particulates readily transport.

Head loss, vortexing, that positive suction head, we evaluated the debris head loss on the strainer to show that the strainer would perform adequately. And that it was acceptable.

Coatings evaluation and all of these

elements are in the NEI guidance. We assume that coating fail particulates in a couple of locations, we assumed there were chips.

And we went -- used the REI process to evaluate this with the staff. So they were acceptable with the -- once we had that dialog.

Screen modification, knowing modifications, hardware modification, we did when we installed the new strainers. And then we removed the Marinite insulation on the reactor vessel nozzles.

Marinite is a trade name for calcium silicate. Which is a very problematic insulation. So, we took that and removed that.

And put the NUCON fiberglass insulation on the reactor vessel nozzles. So those are the only hardware modifications that we did for this paper.

Upstream effects, we looked at choke points. Count all the break flow, and it came straight flow, get to the strainers. And that was acceptable. And we have a very favorable lay out for that.

Downstream effects, we looked at our components and our safety injection system and our containment spray system to make sure that the insulation fibers and the particulate debris is acceptable. These components again, perform acceptably with this debris being recirculated.

MEMBER SKILLMAN: Did you make any changes to your equipment?

MR. SCHULZ: No. We did not.

MEMBER SKILLMAN: Okay. Thank you.

MR. SCHULZ: We didn't have to. We showed everything was acceptable.

MEMBER RICCARDELLA: Excuse me, but on the break selection, does your model include different frequencies for different break sizes? Or how's that --

MR. SCHULZ: Yes. Ernie you want -- about here?

MR. KEE: Yes, sir. We basically interpret the NUREG-1829 frequency tables for -- based on the size of the break that we estimate what that frequency is.

MEMBER RICCARDELLA: Yes. I was part of the 1829 effort. Thank you.

MR. SCHULZ: Last thing is debris source stream. We based our evaluation on information in the plant. And we want to make sure that that's valid, still valid going forward.

So we have programs, instituted programs to help maintain that. We have a containment closeout inspection. Which is really a surveillance at the end of an outage.

We'll go through and make sure we remove all the items that don't belong there. And we'll look for loose debris and remove that from containment.

And then our design change process, we've added steps in there to make sure that when we evaluate changes we need to address -- we want to maintain the insulation types. And the cooling types, the same as the analysis. If not, we have to justify them with respect to the analysis.

And we also looked at an addition of metals, including aluminum. Which is a contributor to a source for our chemical particulates.

So it's in our design change process to evaluate future changes to make sure that our evaluation here is still valid.

MEMBER SKILLMAN: So Wes, what do you do when you close out your outage? Do you go in there with mops and brooms and take out every tiny little piece of stuff that the carpenters left behind when they were making scaffolding?

MR. MURRAY: Let me answer that. This is Mike Murray. I'm on the -- I actually have to sign off the surveillance for an area. So I can tell you exactly what we do.

So during the outage, and when we get into closeout, we have a phase that's call the containment cleanup. And at that point we have coordinators that are basically working with teams that do, in some cases, mop, clean up.

And then I go through three times during the later part of an outage. For example, from my area, and I'll look for every nit to pick. Tie wraps, anything. I'll look in cabinets, I'll look in stanchions. I look everywhere, behind things.

We then identify any punch list items. They get cleaned up. And then prior to mode four entry, we go through, there's a team that has different areas, area managers we're called.

And we go back through containment the last time before we actually enter mode four and sign off the surveillance. And if there's any outstanding items, there's a punch list that has to be taken care of before entry into mode four.

And that is -- and then our resident inspector goes behind us and takes a look. And gives us feedback in anything that he may have found that we may have missed.

And we take that into our lessons learned for future outages.

MEMBER SKILLMAN: Thank you, Mike. Thank you.

MR. SCHULZ: Thanks Mike. Yes. Let's move onto slide 18. And talk about our strainer head loss testing.

This was conducted in July 2008 at Alden Labs up in Massachusetts. We used a -- one of our spare strainer models. So we had a full size strainer module there with the design flow for the module.

Debris loading was based on two trains in operation. So we're going to just scale the debris for this one module based on the total debris load. And then we scaled it -- for the two trains, we scaled it down.

And Alden constructed the flume channel to emulate the approach velocity and turbulence that we would expect in -- inside containment. Let's go to --

VICE CHAIRMAN CORRADINI: I don't remember. I remember the meeting. I don't remember. So did Alden do it so that -- because the way you describe these things, here's the sump, here's all the strainers.

So did Alden do them like this? Or like

this?

MR. SCHULZ: They did both ways.

VICE CHAIRMAN CORRADINI: They did?

MR. SCHULZ: Yes. I mean, they picked a velocity field near the strainer so some's coming this way, some comes that way. So they --

VICE CHAIRMAN CORRADINI: And they took the lowest value that's --

MR. SCHULZ: They did a bounding thing. Yes. To make sure.

VICE CHAIRMAN CORRADINI: Okay. Right. I didn't remember. So this threshold limit which you're eventually going to talk about, is from all these kind of here versus here as the lower limit of those.

MR. SCHULZ: Right.

VICE CHAIRMAN CORRADINI: Okay.

MR. SCHULZ: And they tried to simulate that in our test. To make sure we got a close velocity.

VICE CHAIRMAN CORRADINI: Okay. That's fine. I just couldn't remember.

MR. SCHULTZ: Let's go to slide 19. This is the debris we used during the test. We had our fiberglass debris. We had the fiberglass fine and had the smalls.

We had particulates from our Microtherm insulation. Microtherm is another trade name, brand name, of a high density microporous insulation. We have very little of it.

But it's another problematic insulation. We use that in pipe penetrations through concrete walls. So we do have some of that.

And we had the Marinite particulates, the Marinite calcium silicate. We used that in our test. And we had latent dust and dirt particulates also.

The chemical debris, we used -- I didn't talk about that earlier. We need -- was based on very conservative calculation from -- through Westinghouse methodology.

We used 30 days of containment spray to maximize the amount of chemical precipitants. And there's some precipitants, there's another way of -- two ways of doing it is a minimum pool volume and a maximum pool volume.

And depending on that, the particular precipitants would -- they'd be maximized. And again, we picked the worst of each from the two different ones. So we get a maximum chemical debris loading.

And our coatings, we represented the

different various types of coatings that we have inside the plant. So that's for the strainer debris.

Let's go onto slide 20.

MEMBER MARCH-LEUBA: So when you talk about coatings and you talk about flaking of paint and that flake blocking the strainer? Or is there a --

MR. SCHULZ: Right. Within the zone of influence, the particulates fail -- the coatings fail as particulates. You know, 10 micron particulates.

And our unqualified coatings, we assume they all fail not matter where they are. Whether they're inside the ZOI or outside the ZOI. They're still --

MEMBER MARCH-LEUBA: More then flakes. You're talking about grains of sand that fall out.

MR. SCHULZ: Right. Small particulates. Yes.

MEMBER MARCH-LEUBA: Yes.

MR. SCHULZ: and in one case we had chips. And we -- in one location we treated -- that was again with the staff, we agreed on the methodology for treating those particulates.

MEMBER MARCH-LEUBA: And those typically go into the vessel? Or they stay on the strainer?

MR. SCHULZ: Well they can -- they go to

the strainer. They can be captured on the debris bed. They form a debris bed on the strainer.

But some fibers and some particulates pass through. And that's what our downstream effects look at, both in-vessel and for other confluence too.

So some of it does get through the strainer, yes.

MEMBER MARCH-LEUBA: All right.

MR. SCHULZ: So back -- on slide 20, we do a reconciliation of the debris using the tests. And these test amounts were calculated back in 2007.

And since then, we did use Marinite insulation in the test. However, we removed it so we had the test had more -- extra Marinite in it.

We also -- subsequent analysis showed that the Microtherm insulation that we used in the test was much more then what actually transport. On the other hand we showed that some of our coding particulate debris, more is calculated to transport compared.

It was under the predicted amount compared to what we had in our test. So we that credit here with -- from the Microtherm and the Marinite. And we had a debit from our particulates.

And we did a reconciliation of that to the -- again, reviewed by the staff in the REI process, to show that they compensated. And so that's how we reconciled our test amounts of debris.

Let's go to slide 21. So the results of our strainer head loss testing, the debris preparation and the conduct of the test were acceptable to the staff.

The staff witnessed the test back in July 2008. We had a debris bed formed of a large quantity of particulate with the chemical loading and it did not show the need for additional thin bed testing.

So the test showed about half the head loss was due to the chemical debris. The other half was due to the fibers and particulates.

We added them in sequence. We had particulates and the fibers first. The test took about two and a half days.

We had them in batches. We added the fibers and particulates in batches. And then we added the chemicals. And about half of the head loss was due to the chemical precipitous.

Although the test shows that it was acceptable. And it shows we have satisfactory performance of our equipment up to the level of the amount of debris that we tested.

So that's our benchmark going forward.

And also this eliminated the need for head loss correlation. Our very first fully risk informed approach we used a head loss correlation to calculate debris loss.

But we -- that was getting too many questions from the staff. And there was too much uncertainty there. So we're actually using actual test results was about that uncertainty if you were us a correlation.

MEMBER MARCH-LEUBA: And sorry to ask these questions this late in the review, because I wasn't here. But, on the testing will you use two foot flow into the channel?

Because in real life, you will have two inches. And those two inches will clog. And then the level will rise. And you'll have clean strain then that will clog. And then --

MR. SCHULZ: Oh. Yes. We had full level. Yes. To do what we had.

MEMBER MARCH-LEUBA: Yes. So that's fairly conservative. I would expect most of the fibers to deposit on the first two inches. Right?

MR. SCHULZ: No. We don't turn the pumps on until we get the level above the strainers. So --

MEMBER MARCH-LEUBA: Oh. You have to

cover it completely?

MR. SCHULZ: We have a containment spray and safety injection takes selection from a fueling wash storage tank. And that tank gets depleted and gets -- ends up on the containment floor and fills that up.

When that tank's nearly depleted, then we switch over to recirculation mode. And then we -- then the flow goes through the strainer then.

MEMBER MARCH-LEUBA: Oh, they explained it --

MR. SCHULZ: So those will, yes. There's about -- the minimum amount is about ten inches above the strainers when we turn the pumps on.

MEMBER MARCH-LEUBA: And is there --

MR. SCHULZ: At least one of the pumps.

MEMBER MARCH-LEUBA: Then is there enough time for it to settle into the floor?

MR. SCHULZ: Yes. There is some set -- maybe some settling to it. Yes. We didn't query that too much in our transport. But we have to --

MEMBER MARCH-LEUBA: Okay.

MR. SCHULZ: So, the strainer head loss testing established a benchmark of this quantity of debris, the various debris types resulted in a successful strainer operation head loss testing.

And now Ernie is going to continue with how we use those. Our benchmark.

MR. KEE: And so this is Ernie Kee. We can move to slide 23.

So we've already said this that RoverD scenarios begin with a break at a particular location. At each location there's many thousands of scenarios examined.

And this was for the purpose of finding a certain value. Which I'll address here in a minute. But, so we look at break sizes and orientation of the break to find a particular one we're interested in. Or to exclude as deterministic that whole location.

I mentioned already that we'll use CASA Grande in the deterministic mode to calculate the generation transport and erosion to the floor pool. And this is where we get the limit of -- we look to see if all that material that gets into the floor pool is greater than or less than what Wes has described in his test.

And if it's greater than that amount in terms of fine fiber, which is the most readily transported debris, then if it's more then what Wes has tested, then we relegate those scenarios to the risk-informed category.

I might mention -- now I can't recall who asked this question. But, about the particulate being cat -- was that you Dr. March-Leuba?

MEMBER MARCH-LEUBA: Call me Jose.

MR. KEE: Yes. I'm sorry. That debris is actually counted twice by virtue of the way the tests are done. So, we put all the particulate in Wes' test. And collect it as it will be collected on the screen.

But then also the WCAP at 16793 test, which we used for the cold leg break criteria for adequate core cooling, it uses all the particulate in its test as well. And all the chemicals.

So it kind of -- those things get counted twice. And that's an area of conservatism or uncertainty quantification.

And then scenarios that have to meet the deterministic criteria upstream, as Wes talked about that downstream. Wes talked about that in-vessel and reactor containment building integrity criteria. Or else they're categorized as risk informed.

So we're moving onto slide 24. Of these scenarios that we put in the risk-informed category, and this goes to Dr. Riccardella's question about how do we determine these frequencies.

So this is one of the great simplifications we make in the RoverD analysis. We only have to look at -- well, you could look at many. But what we've done is we only look at one of those -- at each location, we only look at one of the scenarios that's in the risk informed category.

What one is that? That's the one that by carefully looking in a very fine mesh for the break that just exceeds the break size. It just exceeds the amount of debris that was tested.

So, we only have to look at one. Because any other break will create more. You could think of it that way. In fact, it's another conservative assumption in terms of risk estimation that we make because of how we do that.

I don't think we need to get into that. But the point is, we've degenerated this into one scenario that just exceeds the break, the tested deterministic amount, and we find the highest frequency by virtue of the smallest break size on that scenario.

MEMBER RICCARDELLA: Wouldn't a larger break create more debris? And be more of a concern?

MR. KEE: Absolutely.

MEMBER RICCARDELLA: But it's a lower frequency? Is that what you're saying?

MR. KEE: It's a -- well, those larger breaks are also in the risk informed category. And so --

MEMBER RICCARDELLA: Oh, I see. That's fine.

MR. KEE: Exactly right that it would have a much lower frequency as they become larger and larger. So we're looking for the --

MEMBER RICCARDELLA: And this is the determinant.

MR. KEE: A risk estimate. Yes.

MEMBER RICCARDELLA: All right. All right. I've got it. Just a comment.

You know, there's another big program going on now called, oh, what is it? XLRP? You know XLPR, or extra low probability of rupture which is a big quantificative -- quantified study that try to replace leak before break and come up with, you know, more accurate pipe break frequencies then what's in this.

There's NUREG-1829 which was kind of an expert solicitation effort. But, I don't know. That's -- we're not there yet. But just so you're aware, there is more research going on in that area.

MR. KEE: Yes, sir. We've kind of looked into that in a very early time of this -- addressing this problem. And tried to look at probabilistic fracture mechanics. And what kind of causal models are out there for these breaks.

And there's not much information. I think it's still the best source today is this NUREG-1829.

MEMBER RICCARDELLA: I'm sure.

MEMBER SKILLMAN: Ernie, before you go on, that second bullet. Kind of building on Pete's question, relative to the break size.

Fetch smallest break size. Why isn't that fetch the most probable break size? Or the most likely break size? Regardless of its orientation?

MR. KEE: Actually, that is the most probable, based on the way these curves are developed. The smaller the break size the greater the frequency estimate is for currents.

So, there actually can be larger breaks -- I don't know if we want to get mired down. But there actually can be larger break sizes that don't fail this tested amount of material. We know that.

MEMBER SKILLMAN: Yes. I can understand that.

MR. KEE: Yes. So -- but we still -- still we look for the smallest one. So that's a --

MEMBER SKILLMAN: Because it's most likely.

MR. KEE: Yes. And it produces a conservative estimate.

MEMBER SKILLMAN: The greatest amount of fiber.

MR. KEE: Yes. For the risk. So we want to know are -- we want to have a good sense that we've bounded the risk shall we say.

MR. HARRISON: The smallest break that exceeds the tested amount.

VICE CHAIRMAN CORRADINI: And anything above that is assumed to also fail.

MR. KEE: All those -- yes. We assume they all fail.

VICE CHAIRMAN CORRADINI: Could go larger ones and a different orientation would be --

MR. KEE: Might not.

VICE CHAIRMAN CORRADINI: Master tool might not.

MR. KEE: They may not.

VICE CHAIRMAN CORRADINI: Okay. Very good. Thank you.

MR. KEE: Yes, sir. So why don't we -- we're on slide 24 now.

VICE CHAIRMAN CORRADINI: Next bullet.

MR. KEE: Oh. Wayne changed the slide on me.

MR. HARRISON: No, I didn't.

(Laughter)

VICE CHAIRMAN CORRADINI: Very crafty Wayne.

MR. KEE: So we look at all these smallest breaks at each location. And we interpret -- we say we interpret the NUREG-1829 tables to arrive at the total frequency across all the risk informed scenarios.

And we assign that frequency to core damage. So, that's another kind of area that is a conservative estimate if you think in terms of PRA. There would be other failures along the way that could occur.

And then we use the probabilistic risk assessment from the South Texas Project. The same one we developed early and like we were talking about in 2011, to get the change in large early release frequency.

Now I already said that this -- the

containment integrity is maintained independent of this -- the concerns raised in GSI-191. The concerns raised in GSI-191 do not lead to containment failure.

But of course since the core damage frequency increases, these other kinds of scenarios that exist, you know, for isolation and so forth, those are still out there that are in the PRA. And so we -- there's an increase in learn from that effect.

So, we need to this -- we're moving to slide 25. And we need to look at in-core reactor vessel scenarios. We've talked about all the others so far.

And in this case, in this vessel situation, we look at two kinds of breaks, hot leg breaks and cold leg breaks. Because they're different in terms of how the debris flows into the core and the driving head.

So looking first at the hot leg break scenarios, and I've already mentioned this, we're using peak clad temperature. Which is a common acceptance criteria for success.

And in this case, the theoretically bounding assumption that we make here is that for all the fuel assemblies and the core barrel bypass channel are blocked fully at the time of sump recirculation switch over. We do take credit for the build up to 15 grams of per fuel assembly that's in the WCAP 1693 testing.

MEMBER MARCH-LEUBA: So Ernie, is there any flow path for water to get into the reactor core after you block those?

MR. KEE: Yes. There is. And there's flow paths that have large enough holes in them. For instance, the upper head spray flow and then of course the flow through the other intact loops of steam generator can come around eventually and come into the top of the core.

MEMBER MARCH-LEUBA: They'll go from up in the hot leg?

MR. KEE: Correct. Correct. And so we show that we meet the thermal hydraulic requirements with this kind of a limiting of function. And so we move to slide 26.

MR. HARRISON: And those were the RELAP5 3-D analysis that Lisa was talking to. And we don't use the 3-D function as I recall.

MR. KEE: That's correct. Although we have looked at that.

VICE CHAIRMAN CORRADINI: But just to clarify for the members, you -- I don't think you said it for the out of -- for the strainers. But this large of a break is already covered from a failure standpoint because by that time the pipe diameter that would have caused suctions and strainer blockage is smaller than the 16 inches.

MR. KEE: Yes, sir.

VICE CHAIRMAN CORRADINI: So you don't -- in some sense this is included in the failures that already would have occurred by just blocking the strainers?

MR. KEE: Yes. So we're looking at the ones that we said passed deterministically through the strainers. And now do they pass in-vessel? Yes, sir.

VICE CHAIRMAN CORRADINI: Okay.

MEMBER MARCH-LEUBA: So every single break that passes deterministic -- that is accepted deterministic from the strainers will also be acceptable for in-core.

MR. KEE: That's what we have to make sure. Yes, sir.

MEMBER RICCARDELLA: And then those get eliminated from your eventual probabilistic assessment, PRA assessment? Is that the -- is that the way it works?

They're acceptable deterministically, then you don't consider them?

MR. KEE: That's correct. All those scenarios that meet the -- to Wes' test level. Tested them out of the fiber level. And also are acceptable in-core in-vessel.

MEMBER RICCARDELLA: Okay.

MR. KEE: They have to meet all these deterministic criteria. Yes, sir.

MEMBER RICCARDELLA: And that's 16 inches for a hot leg break?

MR. KEE: Yes. When we look at 16 inches and below.

MEMBER RICCARDELLA: Sixteen inches and below.

MR. KEE: They all pass. Yes, sir.

MEMBER RICCARDELLA: Okay. And then for things greater than 16 inches, you're going to include that in the probabilistic, in the PRA model?

MR. KEE: Yes. We did that. Yes, sir.

MEMBER RICCARDELLA: Got it. Got it.

MR. KEE: So moving to slide 26 now. So the cold leg break only has a driving head of the down comer of course, the excess water from the ECCS just flows out the break in this case.

And because of that characteristic, you

accumulate much less fiber on the core. And we analyzed that with a mass conservation three linear different -- well actually they're nonlinear.

But anyway, three differential equations for the strainers, the core and the pool with how much is allocated all throughout. And then we check to make sure that the -- we look at many scenarios and check to make sure that we don't exceed the WCAP 16793 criteria, 15 grams per fuel assembly.

And in fact we couldn't conjure up a scenario that came even close to that level.

MEMBER MARCH-LEUBA: Okay. So what you're saying is that for all the cold break -- cold leg breaks, none of them are produce enough fiber in the core to block it.

MR. KEE: That are below the deterministic criteria. Yes, sir. So both hot leg and cold leg are acceptable from that point of view in-vessel.

MEMBER RICCARDELLA: And what is it that makes the cold leg less critical? Cold leg breaks less critical?

MR. KEE: They're not less critical. But --

MEMBER RICCARDELLA: Create less debris.

MR. KEE: It's the same amount of debris

in the pool. But what happens is, the -- all that flow that's being bypassed right out through the cold leg, the broken cold leg, goes right through and gets restrained through the strainers.

It gets strained again --

MEMBER MARCH-LEUBA: It then runs right through the filter.

MR. KEE: Yes. And we've done --

MEMBER MARCH-LEUBA: So you've -- the water pump get into the cold leg leaks out, goes back to the filter again. And that's all. The filter has a chance for catching it again.

MEMBER RICCARDELLA: I see.

VICE CHAIRMAN CORRADINI: The flow -- can I try it a different way? Also the flow from through the core is reduced since it's not a hot leg break, it's a cold leg break.

So whatever is there, a lot of it doesn't get transported to the core. It gets transported back out the cold leg break. Back to the strainer.

MR. KEE: That's even a better -- yes. And it's governed by the decay heat level. Which doesn't demand near the -- we have a huge amount of flow from our emergency core cooling system.

MEMBER RICCARDELLA: So it's less critical from the standpoint of in-vessel effects. It's not -- no different for the strainers. It's still a concern to the strainer.

MR. KEE: Yes, sir. Yes. It has to pass. Yes. So, we're moving onto slide 27 now.

This question came up with regard to single train assumption in our last -- in the subcommittee. So some people weren't here.

And the question is regarding taking into account the extra amount of flow that you realize in the containment spray system in single train operation. And whether or not that extra flow under the assumption of half the debris load criteria, which we adopted for single train, if that would be acceptable.

And the answer is no, using the -- using that half amount. So we turned back to the PRA. And we asked again, remember we've done all these different equipment configurations already.

So we know the answer to this as to what the level of risk is. Which was when you add up all the scenarios that represent the single train configuration, they're like on the order of 18 to the minus 9.

So, these kind of -- in terms of success.

And these kind of scenarios contribute very little to the total problem where we're looking at 18 to the minus 7 numbers.

So moving onto slide 28.

MEMBER MARCH-LEUBA: So then go back to the last bullet. You said the single terms then it could be added to risk informed integrity.

But how do you do it?

MR. KEE: No. Well, just add 18 to the minus 9 to 15 to the minus 7.

MEMBER MARCH-LEUBA: You did add it, and it was inconsequential.

MR. KEE: Yes. It's so small. Yes. So now we're on slide 28.

And this just summarizes all the information that we've talked about so far. Basically the measures we took at South Texas when we replaced the strainers and insulation and so forth, did achieve the desired result that we have a very low expectation for any kind of problems related to the concerns raised in GSI-191.

And these risk estimates include a significant safety margin. We talked about that and Wes has talked about that.

Also, we maintained defense in depth. So even in the unlikely event that we see something like this happening, we're relatively sure that the containment is not going to be conditionally failed due to the same cause.

We've done, as Lisa mentioned, many, many different approaches of many different -- incorporated many different analyses using this best estimate and modified best estimate. And now with this RoverD approach, hybrid approach is, I think Ms. Shelton mentioned, Helton mentioned, we consistently see the same kind of low risk estimates in all these approaches.

So moving to slide 29. Just to summarize RoverD as a framework. It makes GSI-191 risk assessment understandable, easy to review.

We use conservative testing and bounding analysis to accomplish that kind of simplification. Scenarios fall into just two categories. And that -- by applying accepted test methods and bounding analysis.

We've done many supporting additional tests and analysis throughout the six-year life of this project. I think it's about six years. That confirm for example, the morphology and the quantity of chemicals and so forth that we made publically available.

It's on Adams. A lot of this information is on Adams. Or we've also published in the academic literature.

With that I'll turn it back to -- the discussion back to Wayne Harrison.

MEMBER MARCH-LEUBA: Not so fast Eric.

MR. KEE: Okay.

MEMBER MARCH-LEUBA: All these conclusions that you have here in the summary, how much do they depend on the fact that you increase the size of those filters from 155 to 1800?

MR. KEE: Oh, it's -- no, and that is -- that observation that was made in the generic letter asking us to review whether or not we expected that we would be successful with that current strainer design, I mean, that was instrumental in everything we've done since then.

We've changed our design change package approach to make sure we control this aluminum and nefarious kind of debris types. And then Wes replaced these strainers with these enormous things to accommodate large amounts of debris.

MEMBER MARCH-LEUBA: I don't think you take enough credit for the positive steps you took to make your plant safer.

VICE CHAIRMAN CORRADINI: I think I know where Jose is going. But I think just to put it in context, I think the PWR community, the population as a whole, not just South Texas, made these equipment changes early up front.

And then I'll get confused about the right name, the generic letter -- I'm sorry, I've got it wrong. But, a second communication concern about downstream in-vessel effects, et cetera, then led to.

So I think South Texas took the initiative to look at a combined holistic look at the problem. That's my kind of way of thinking about it.

But the population of all the plants had to do the first thing in terms of strainer change out.

MR. KEE: Yes, sir. And that led to these lowered risk estimates for the -- for problems.

MEMBER RICCARDELLA: It would be interesting to see what these Delta CDF and Delta LERF values would be if you went through the full analysis with the old trains. Did you do that?

MR. KEE: We've done that actually. I believe that was -- that maybe in one of our applications. I think it is.

And it maybe the supplement two.

MEMBER RICCARDELLA: I don't recall that.

MR. KEE: We've done that.

MEMBER MARCH-LEUBA: Your critical break size will go down to one or two inches instead of 16.

MR. KEE: Yes. And downstream effects are -- in other words, these in-core effects with these larger holes would bypass much more debris into the --

MEMBER MARCH-LEUBA: You might still make a risk informed argument that you're okay. But you wouldn't feel comfortable with a one inch break critical effects.

MR. KEE: Yes. I'm not sure. Yes. It was pretty high I recall, for the number.

MEMBER MARCH-LEUBA: But where there is a 13, 14-inch break, critical break size with conservative assumptions, it makes you feel very comfortable. It does to me.

MR. KEE: Yes, sir. Yes, sir.

MEMBER BALLINGER: To pick up on what Pete's mentioned, I was pouring through 1829 to try to find out what probabilities of failure are for the larger pipes. I mean, the XLPR program is going to basically eliminate those as failure modes, I think.

So, I don't know which ones dominate the CDF here. I don't remember.

MR. HARRISON: All break lower, it's about 13 inches basically.

MR. KEE: Twelve point eight.

MR. HARRISON: Twelve point eight.

MR. KEE: And above.

MR. HARRISON: Yes.

MR. KEE: See that's not surprising.

MEMBER RICCARDELLA: I wouldn't try to predict what -- it's not surprising either. But I wouldn't try to predict what's going to come out of it.

MEMBER STETKAR: It's probably safe to say the numbers will be different.

(Laughter)

MEMBER BALLINGER: However, you can assume leak before break.

VICE CHAIRMAN CORRADINI: Okay, Wayne? Wayne?

MR. HARRISON: I will talk about -- this is Wayne Harrison. I will take about the regulatory implementation starting on slide 31.

And basically our regulatory information -- implementation included three elements. We had a debris specific action, promote three and above, and our emergency core cooling system and containment spray system technical specifications.

Those are the only two systems that we have that have any dependencies on the emergency sumps. We have a set of updated final safety analysis report changes that help us as a licensee in our implementation process.

We will change -- for instance, I'm going to talk just briefly about the exemptions field to change the description of how we comply with the regulations using the risk informed process.

We will explain the program. What we just went through here, we will have an appendix in Chapter 6 of our USFAR that explains the analysis that we went through and the rules of engagement for that analysis with our plant.

And it also has elements of change control that you've seen in the draft rule, 50.46(c), and the things that are analogous to 50.59 that we can't change without prior staff approval.

Because there's some of these things that are fundamental to our methodology that we -- that would require prior staff approval to change it. And that's outlined in our USFAR changes as well.

We have requested exemption to four of the regulations, 50.46(a)(1), the other properties portion of that. Basically the basis for all these exemptions is that we're using a risk informed methodology instead of the deterministic methodology that's implicitly if not explicitly required by the regulations.

So, we can -- the basis is that we can achieve the underlying purpose of the rule without using the deterministic method that the risk informed still achieves the basic purpose of the rule.

And I mentioned the burden associated with it, but the basic reason is that we can achieve the rule's purpose using the risk informed approach.

General design criteria in 35 for emergency core cooling system, general design criteria in 38 for containment heat removal, and general design criteria in 41 for containment cleanup. And they all have the same basis.

I'll go on and talk about the regulatory information -- implementation of the technical specification change. Because I think you had a question on that. And I can -- we can talk about that.

This is what the emergency core cooling system tech spec looks like. The containment spray system is very similar to this.

And this is in addition to the already -- the existing action statements that we already have. But we have a statement for -- with less then the required flow paths operable solely do the potential effects of LOCA generated and transported debris that exceeds analyzed amounts perform the following, and that's to implement action to -- action for a -- compensatory actions.

And then has a 90-day action statement on that. You'll notice that there are a couple of things that are important here. One is this only applies -- it can only be applied for effects of debris.

If it's something else that's broke with that strainer or that emergency core cooling system or containment spray system, you have to apply the other tech spec actions.

MEMBER MARCH-LEUBA: But does the tech spec tell you how to evaluate this?

MR. HARRISON: The basis will. And we -- no. This tells you what the requirements are. And we don't -- but we get additional guidance to the operators in the basis for the technical specifications, where we will explain what we mean by debris effects.

So they will, for instance, say well I've got a tarp over the strainer. Is that a debris effect? Well no. a tarp over the strainer is something worse then the debris effect. Okay?

But if it's something out there in the containment that's covered by Wes' analysis, well that's debris. And you can go assess that in accordance with the analysis.

Now, we do this in accordance with our corrective action program in how we address degraded, non-conforming conditions. Nothing's really changed in that program.

There's a guidance in that that we are currently applying for debris. And this will actually -- so the operators you see -- and it's in that procedure and it's also in another procedure that says the kind of debris and the quantities of debris that they're -- that they have when they do those walk downs Mike's talking about.

Well, they're doing this in accordance with a walk down procedure. And there's guidance in that procedure on the amounts of debris that you can have.

As part of our implementation of this, Wes is going to go in and polish those. Or update those debris instructions. I think he's probably going to make it simpler.

So the process that we've always been using is essentially the same process we're going to use. If they find something, they're going to make an immediate operability determination and a prompt operability determination necessary in accordance with that procedure.

Engineering will get involved in the evaluation, is that in conformance with our -- what we did in this evaluation. And make a determination is there too much debris or is it okay?

And there's margin in the calculations that Wes has. And there can be some room for evaluations. No -- really no different from any other engineering assisted operability determination.

MEMBER KIRCHNER: Pragmatically though, you wouldn't have this kind of occurrence except after you've gone to shut down and refueling and maintenance, right?

MR. HARRISON: That would be --

MEMBER KIRCHNER: Or are there other -- I mean, where I'm going with this question is, during the normal operation of the plant, when the containment's buttoned up, what's going to change the status?

MR. HARRISON: Probably the only way you would have that occurring, and Wes can chime in, is that if something comes out and you say, oh, my gosh, I found a calculation that we didn't account for this debris. Or we found this piece of thing that came in here that has this insulation on it or this other debris source that we didn't -- it was a calculational error that we --

MEMBER KIRCHNER: That would be in the engineering department, not in the operation of the plant.

MR. HARRISON: Yes. That's right.

MEMBER SKILLMAN: Yes. That's where I was going when I introduced this topic an hour or two hours ago.

Mike described the close out, the cleanup, and I presume, you secure the door. You lock the containment.

MR. MURRAY: So this is Mike Murray. There's one thing I thought about that I didn't add. Is once we get the containment closed out, there's also a procedure for every entry into containment after it's closed.

I think that may close the gap there. And in that procedure, you have to account for what you take in. You have to account for what you bring out.

And you also have to keep your -- one of the requirements is you look for other things that someone else may have missed.

And that is a briefing that happens with every entry team after we've established containment integrity. Which was the surveillance when we signed it off. And operations accepts that surveillance.

Containment integrity is then set. The door interlocks are in. And then you go into the process of every entry team has to be accountable for what's taken in and brought out.

MEMBER SKILLMAN: Here's what prompted my question. Presuming that we're on watch. All of us have gone one. Let's go to nuclear one.

It's three o'clock in the morning on a 24-month fuel cycle. We're out four hundred days. We're doing great.

And at three o'clock in the morning one of us says, you know, I forgot. I think we're inoperable because there's something down there that just struck me would render that one strainer inoperable.

What do you do? And the way this surveillance is written or the way this tech spec is written, potential effects of debris generated and transported.

And it just seems to me that that is -- first of all, I understand the words. But it's certainly vague for a practical operator.

You're sitting there on -- at power. And you're saying, I think something's wrong down on that strainer.

But if you're like most PWRs, you do not do in power containment entries.

MR. HARRISON: We do.

MEMBER SKILLMAN: Do you do it often?

MR. HARRISON: Absolutely.

MEMBER SKILLMAN: Yes? You don't go into the primary shield.

MR. HARRISON: No.

MEMBER SKILLMAN: Okay. So you're on the operating flat or you're in a shielded area. But this probably not something that's done a lot.

MR. HARRISON: No. And remember, you know, we're -- this is something that is -- that's not debris -- that's not unique to debris.

You know, that could be anything that is -- becomes an operability question. And that's where you go back to your procedures.

You go back to this degraded non-conforming condition procedure. You go back to your basics of the presumption of operability.

That it's the shift manager's call. And it's based on his presumption of operability. And there's a process you go by that operations will drive.

And we have confidence in that process. It's worked for us for a long time. You know, a very -- depending upon the level of assurance that shift manager has with what he believes is in that containment, he may say well, I'm going to enter this action statement.

And then they can go do a confirmation. They determine that it's okay. They can exit the action statement.

MEMBER SKILLMAN: That's fair enough. I understand what you're saying. And I understand the authority risks with the shift manager to make the call.

And I understand you can also go too operably degraded and continue to operate. So, I really do understand that.

I was just intrigued at what could be the interpretational vagueness of this wording.

MR. HARRISON: And we -- and I'm glad you brought that up. Because we will get -- we hadn't got -- we will train operations on this.

And we will sit with them in their classroom, here's this new tech spec change. And they reviewed this by the way, also. So this shouldn't be news to them. They were part of the review.

But there will be questions from the various shift managers and license operators. And we will respond to those questions. It might result in some additional clarification in the basis.

MEMBER SKILLMAN: Yes. Okay. Okay. Thank you. That's fine.

MR. MURRAY: We've answered those questions I think that they're -- are you all ready -- are you ready to move on?

MR. HARRISON: The only thing that I was going to add to this was with -- I think there was some question with compensatory actions.

And those are the -- our typical risk management compensatory actions that might mean we'll -- if we know what the debris is, we can remove the debris or take action that would prevent transport of the debris.

We can defer maintenance that would affect availability of effective mitigation systems. We can increase the frequency of leak detection monitoring and brief operators.

There are a number of risk management actions that we can take as compensatory actions. And the 90 days is based on the very low likelihood of a break that would challenge our -- or exceed our tested amount.

And that also gives additional time if we do find that to have -- if, you know, we need additional regulatory relief from the Nuclear Regulatory Commission, we would have additional time to do that.

But it's analogous to what was done previously with the containment -- I mean, the control room cleanup system.

So, with that, I'm done and ready to turn it over to Mike.

MR. MURRAY: All right. Change the slide, please. Next. Mike Murray. I'd like to do the conclusion statement on it.

From -- the RoverD process incorporates all aspects of the debris. It allows closure of the generic letter of 2004-02. It has deterministic testing, debris generation and transport, core effects and risk informed evaluation.

All being said, we have confidence that we've established through our extensive analysis, debris transport modeling, testing and risk analysis, that the change in risk is a result of the LOCA generated debris meets the risk acceptance guidelines established in REG Guide 1.174.

VICE CHAIRMAN CORRADINI: Okay. Questions by the members? Otherwise, we're scheduled for a break. So they can change out, staff will come up. Questions?

(No response)

VICE CHAIRMAN CORRADINI: Okay. We'll take a break. Come back here at 10:30.

(Whereupon, the above-entitled matter went off the record at 10:18 a.m. and resumed at 10:30 a.m.)

VICE-CHAIRMAN CORRADINI: Okay, let's get back in session, and staff will begin. Lisa, I guess you're up.

MS. REGNER: Yes, sir.

VICE-CHAIRMAN CORRADINI: Green light. Yeah, you're fine.

MS. REGNER: At this point, the staff will go into more details. The safety evaluation, starting with Key Principle One of the five Key Principles of

Risk-Informed Regulation. The proposed change meets current regulations, unless it is explicitly related to a requested exemption or rule change.

The proposed change does not meet current regulations, since the NRC has interpreted the regulations in 50.46 as requiring a deterministic approach to show compliance. In 2012, the staff proposed a change to 50.46 to allow licensees to use risk without meeting an exemption from the use of the deterministic approach.

The rule change would allow licensees, on a case-by-case basis, to use risk information, risk-informed-alternatives to assess the impact of debris. In March 2016, the staff submitted the final rule and is awaiting commission vote.

In the meantime, since the rule-making has not been promulgated, STP requested four exemptions, which Wayne talked about. So I don't intend to go into those again, unless you have questions.

MEMBER MARCH-LEUBA: I do. It may be a novice question, and you may be the wrong person to ask. But why is this a change? The reactor hasn't changed. You haven't done anything but re-analyze an erroneous analysis that you did before.

MS. REGNER: That is true. However,

again, ECC -- your question is, Nothing is changed, why do we need --

MEMBER MARCH-LEUBA: The reactor yesterday was just as good as the reactor tomorrow.

MS. REGNER: Correct.

MEMBER MARCH-LEUBA: Exactly the same reactor.

MS. REGNER: Correct.

MEMBER MARCH-LEUBA: Nothing has changed.

MS. REGNER: Correct.

MEMBER MARCH-LEUBA: Only we found out there was a bad calculation on the record, and they're redoing it.

MS. REGNER: Correct.

MEMBER MARCH-LEUBA: So maybe, as I said, maybe it is a logic question. I would consider a change if I want to replace my insulation, I want to put in new insulation. Then it's a change. But they didn't change anything.

MS. REGNER: Right. It was a series of accidents, both here and abroad, that brought the --

MEMBER MARCH-LEUBA: Could this have been handled through a Part 21, for example? I mean, I have an analysis on the record.

MS. REGNER: Right.

MEMBER MARCH-LEUBA: Which I now know to be incorrect. And I'm fixing it.

MS. REGNER: Right, right. There are alternative ways. However, this, and I don't know as much history as Steve does. But from a regulatory standpoint, there usually are alternative ways to handle things. In this case, since ECCS is such a safety significant part of our design bases --

MEMBER MARCH-LEUBA: I'm not in any way insinuating that we're wasting our time.

MS. REGNER: Right, right. No, no, no, understood.

MEMBER MARCH-LEUBA: Absolutely, we need to do it.

MS. REGNER: You are right, and ultimately, people keep asking me what's changed, what's changed. Nothing has changed but the paperwork.

MEMBER MARCH-LEUBA: The analysis of record has found something incorrect, and we are fixing it.

MS. REGNER: Right, right. Now, there has been a lot of work in terms of replacing the strainers and in terms of replacing insulation. The licensees have taken, as I said, a lot of effort. However, this exercise right here that we're talking about today is ultimately a paperwork exercise. A really, really expensive paperwork exercise.

Yeah. And when I was doing the environmental assessment, there are no design -- it was a hard concept to get across. Why are you doing an environmental assessment for no changes to the plant?

So it didn't need to be done. The environmental assessment did not need to be done. But we did it because this is a significant departure from the way we've done business before for compliance with 50.46. Does that help?

MEMBER MARCH-LEUBA: Yeah, as I say, this is probably a question for the lawyers more than for you.

MS. REGNER: Oh. Do we have any lawyers here?

MEMBER MARCH-LEUBA: No, no, we don't want them.

MS. REGNER: No, we don't, but. So I won't go into the exemptions any more, unless there are questions. And again, but I will say that if the staff grants these exemptions, the only departure is for STP to use the risk-informed methodology to show compliance considering the impacts of debris.

At this point, I'd like to introduce Mr. Steve Smith to talk about Key Principles Two and Three.

MR. SMITH: All right, next slide. All right, for the deterministic part of the review, we concentrated on the colored slides on the right. Two and three is defense in-depth and safety margins. And we also have to input to the risk evaluation, so that's block four.

I'm not going to take too much time, I don't want to repeat things that have already been said. I'm just going to kind of go over the major differences between how STP did things and how things were done in the past, because that might be of interest. And if you have any questions, please ask. Next one.

This slide we've seen, and this just shows where the deterministic review went into the test. It went into how the debris amounts are calculated, and then we determine if the scenario is acceptable or not.

The next slide, safety margins and defense-in-depth is the staff's evaluation of that is detailed in the SE. And we had input from a lot of branches on the safety margin and defense-in-depth, and we appreciated that help.

I'm not going to talk about this in detail, but if there's any questions about safety margins or defense-in-depth, I can try to answer those. All right, next slide.

This talks about the debris source term. And the way that STP calculated the debris source term was in accordance with staff guidance, and it's conservative guidance. First, they performed a test with a known amount of debris under which the strainer performance was acceptable. And they talked about that.

Then they performed evaluations for thousands of break scenarios and compared the amount of debris from those scenarios with that which was tested. And if any scenarios ended up with more debris, then that was tested. Then that scenario was assumed to go to core damage and had an impact on risk.

The way typical evaluations for this are done, for example, like the option 1 plans or the options 2A plans, they would go and they would identify two or three breaks that were the most likely ones to create the largest amount of debris, or the most problematic combination of debris, and they would put all that debris in a test and show that the test was acceptable.

So in STP's case, they couldn't show that. So then they had to, anything that was more than what was acceptable was considered to be core damage.

The way we've talked about this, and they did calculate those amounts of debris using the CASA Grande code, which automated it, made it so that it could be done in a reasonable amount of time. Otherwise, we'd probably still be cranking out calculations.

VICE-CHAIRMAN CORRADINI: So just to stop you. You said something that I've been trying to understand. So the difference between and 2A and 2B for strainers is strictly the comprehensive nature of the analysis that STP used, versus picking what are obviously bounding break sizes at particular locations in containment.

MR. SMITH: Right.

VICE-CHAIRMAN CORRADINI: That's really the difference between 2A and 2B.

MR. SMITH: 2A plants, every break would be deterministically okay.

VICE-CHAIRMAN CORRADINI: Fine.

MR. SMITH: 2B plants, some breaks might not be deterministically okay.

VICE-CHAIRMAN CORRADINI: All right, thank you.

MEMBER RICCARDELLA: But isn't what South Texas is doing is some combination of 2A and 2B, in effect?

MR. SMITH: Yeah, we lump them in the 2B category. If the plant had one scenario that was not deterministically acceptable, we would lump them in the 2B category, yeah.

The other thing we talked about a little bit was for partial breaks, they chose the break orientation that created the largest amount of debris, sort of. Anyway, they chose the orientation that produced the smallest break, which is the most likely break to occur.

So that cranked their risk values up, you know, higher than they normally would be. So we thought that was a good margin that they had there.

And then the last thing I'll say about this slide is that we had some good help from Southwest Research. They independently verified the calculations, they exercised the CASA Grande software to make sure it was working properly, and gave us a lot of confidence in our evaluation.

This slide 19 is about debris transport. I'm not going to talk about the strainer debris transport. That was all done with CASA Grande. The debris transport for in-vessel effects was done separately from CASA Grande. So what they did was they tested how much debris could get through the strainer under various conditions.

They chose a conservative value from those tests. And then what they did was they took the amount that could get through the strainer, and then they calculated where that might go. How much is going to go to the vessel, how much is going to go out the break, how much is going to go to the containment spray.

And what they found was that in general, the fewer pumps that were running would have more fiber reach the core for a cold leg break. Because the lower the flow, the more of that flow has to go into the core, so the more debris is going to go into the core. If you have a lot of pumps running, there's a lot more flow going out the break.

And what they found was that as far as debris amounts, is that under design basis cases, the debris amount is very low. And then if you start getting into some other cases, they did one case where they had only HHSI, high head safety injection, pump running.

And that increased it a little bit, but it still was much lower than the values that we found acceptable when we did some evaluation for WCAP-16793. And again, Southwest did independent calculations to verify that they had calculated these transport amounts correctly.

Okay, then we talked about the impact of debris in the strainer. One thing I did want to say, I thought maybe during the discussion with South Texas there was a little bit of, I don't know if it was well understood how much of the debris that's generated actually gets to the strainer.

So it's actually about, of all the debris that's generated or even knocked off in whole pieces, about 20-30% of that debris is going to be fine debris that's going to get to the strainer. And that accounts for the amount of debris that's generated as fine. And then when it gets in the pool, some of that is considered to be eroded.

Now, in the very first phase, the amount of debris that got to the strainer was way higher. Because we said 40% would be generated as what we call small fines. And that had to be treated as fine.

And then 90% of whatever was left would erode. So almost all of it got to the strainer. But there's been testing done since then that gives us more realistic values.

So as far as impact of debris on the strainer, what they did was, when they did their test, they came up with a head loss for the amount of debris.

They put all the debris and all the chemicals on, and then they evaluated various aspects of the strainer or the pumps, net positive suction head, structural vortexing, and also flashing. And they found that at load, the strainer would pass when you considered staff guidance and you did things in accordance with staff guidance.

ACRS had previously questioned that South Texas had done a test before this with a lot more debris in it. And ACRS had questioned why the head loss was so much higher than the second test. And it was basically just the amount of fiber that was in the test.

And if we look at the next slide, we can see in the circle, I tried to fix this up since the last meeting, in the circle are all the breaks that fail, all the weld locations that produce debris amounts great enough to cause the strainer to have too much debris on it, okay.

And the one, most of them are, the ones that are about 400 pounds of fiber fines, are all loop piping breaks. So big breaks. And these are all double-ended breaks, these are not the single end breaks, because the whole chart would be filled up and we wouldn't be able to see anything.

The one small, the one break that's just above the 200 line, that's the pressurizer surge line break. And that's the only non-loop piping break that can cause an excessive amount of debris on the strainer. And all the other breaks below the 200 pound line are, those all would result in a deterministic success.

Now, the arrow up on the upper left side --

VICE-CHAIRMAN CORRADINI: Just for some of the members, so that's literally near the, that is the threshold limit that was --

MR. SMITH: About 200, it's just below 200 is the threshold limit. And then, and that's what the test, that's what the 2008 test was done with, just below 200 -- 191, 192 pounds. If you see the arrow up on the left, that's what the original test -- so it had about almost four times as much fiber in it. And that's why that first test was unsuccessful. Yeah.

Okay, and then we want to talk about the impact of debris on the in-vessel. And this is only talking about the cold leg break. We'll get into the hot leg break in a minute.

But for the cold leg break, the amount of debris reaching the core at STP is much lower than the amounts that we found to be acceptable when we did a review on WACP-16793.

That WCAP did not evaluate the potential need for changes in boric acid precipitation calculations due to debris effect.

And the problem that could occur when you have debris in the core, at the inlet of the core, most plants credit both the core water volume and the volume below the core inlet, like it's called the lower plenum in the PWR, as a mixing volume for, basically to reduce the concentration of the boric acid.

So if you have debris at the core inlet, it could split those two volumes up. You might get a very high concentration in the core and a low concentration in the lower plenum. So that's a concern.

The staff doesn't have any kind of information on what kind of debris amount would actually cause that segregation between the lower plenum and the core. We think the amounts of debris that South Texas has would not cause that issue.

We're still searching for a good answer for boric acid. And the Option One plants are going to have to address this in the future as well.

So basically, what we're saying is that we're not going to say that it's bad, but we're not going to say that it's good. And we're going to expect STP to come back and evaluate boric acid precipitation with the effects of debris at some later time.

The Option One plants are also going to have to come back and do that. And we think the answer to that will be in WCAP-17788. We're not sure, but we're still looking at that.

VICE-CHAIRMAN CORRADINI: So I guess I didn't catch that in the, what you just said, I didn't catch in the subcommittee meeting.

MR. SMITH: Okay.

VICE-CHAIRMAN CORRADINI: Maybe I missed

it. So how is this going to be resolved, or is the resolution to be determined?

MR. SMITH: The resolution is to be determined. We think that WCAP-17788, which is the new in-vessel for the two alpha plants, which is under staff review at this time, addresses it.

We haven't said it, we haven't come to the point where we're agreeing with what the PWR Owners Group is saying yet. You know, we have some RAIs on it. So we have to see how those RAIs are responded to before we can write an SE on it.

MEMBER POWERS: The boric acid precipitation issue is supersaturation in boric acid, or is it chemical reaction with all the junk?

MR. SMITH: Supersaturation.

MEMBER POWERS: And you don't worry about precipitation by precipitating Loctite and things like that.

MR. SMITH: The other chemical effects are evaluated separately. We haven't attempted to try to combine the two phenomena and, you know, evaluate them together.

MEMBER POWERS: And the reason for not combining the two effects?

MR. SMITH: The reason to have to do it?

UNIDENTIFIED SPEAKER: The reason you're not doing it, I think he's asking.

MR. SMITH: Well, I have Paul Klein here from the audience to hopefully jump it.

MR. KLEIN: Paul Klein from NRR. The reason we're not concerned boric acid precipitation is designed not to happen. So the hot log switchover time is designed to flush the core out before we get to the point where the boric acid reaches the point where it will precipitate.

MEMBER POWERS: Yeah, but I think what I'm asking is you're now going to sweep in to your sump liquid a lot of goethite and a variety of other stuff, a little calcium, magnesium. Other things that just love to glomp onto a borate ion and drop out of solution in a nice flocculate precipitate that's nice and soft and pushes through any hole that it can. Why wouldn't you recognize that?

MR. KLEIN: I think I maybe I misunderstood your question, Dr. Powers. I was talking about global boric acid precipitation in a cold leg break scenario, where you concentrate it over a period of time until you reach a point where it begins to precipitate.

I think, if I understand you correctly,

you're more considering local effects, perhaps deposits on fuel rods of borates and other species that are dissolved in solution?

MEMBER POWERS: I mean, you can expect this stuff in, you're going to have certain, I mean, you put a lot of junk down into the sump. Now you're into recirc mode, you're putting a lot of junk into your core along with boric acid.

Zinc just loves to precipitate out of zinc borate. Iron loves to precipitate out of ferric borate. Calcium likes to drop out of calcium borate. Why wouldn't you recognize that?

MR. KLEIN: That is considered as part of the analysis of the LOCA DM part of WCAP-16793.

VICE-CHAIRMAN CORRADINI: So that is being considered now.

MR. KLEIN: It is considered as part of the --

VICE-CHAIRMAN CORRADINI: That's part of the LOCA DM calculation?

MR. KLEIN: LOCA DM, yes.

VICE-CHAIRMAN CORRADINI: Okay.

MEMBER POWERS: Thanks, Paul.

MR. SMITH: Let's see, did I finish this one? Yep, I think I did get to the bottom. So I think we're now going to move on to the thermal hydraulic part of the presentation.

MS. REGNER: Dr. Kaizer, are you able to hear us? Josh Kaizer? Dr. Josh Kaizer's supposed to be on the phone --

VICE-CHAIRMAN CORRADINI: Let's make sure, yeah, I hear it start crackling. Something's going to happen. Josh, are you out there?

MR. KAIZER: I am. Can you hear me?

VICE-CHAIRMAN CORRADINI: We can now.

MS. REGNER: Thanks, Josh.

MR. KAIZER: Excellent.

MR. SMITH: Okay, so Josh isn't here. He's at ASME's main V&V conference, making a presentation. And one of the V stands for Vegas, because it's in Vegas. I don't know what the other V stands for.

VICE-CHAIRMAN CORRADINI: Verification. Are you really in Vegas?

MR. KAIZER: I am.

MR. SMITH: And the other reviewer, Reed Anzalone, he isn't here today because he's home taking care of a newborn baby. So therefore, I get to present these slides.

VICE-CHAIRMAN CORRADINI: But we can call upon them as needed.

MR. SMITH: Yes, we can call on Josh. Let's go to the next one. Okay, the goal of the thermal hydraulic evaluation was to determine whether the LTCC, long-term core cooling, model used by South Texas was acceptable.

When they started doing the evaluation, they determined that large breaks, greater than 16 inches, were too complex to evaluate, and they were not evaluate with the evaluation model, the RELAP5.

So the evaluation only focused on the long-term portion of the event, which requires the evaluation of fewer phenomena. And then the phenomena that are evaluated are a lot less complex than the typical LOCA evaluations. So this reduced the complexity of the evaluation and made the review much more reasonable.

And then the next slide shows what portion of the in-vessel evaluation was done by the long-term core cooling evaluation model. And that small and medium, medium is less than 16-inch, I know that's not a typical medium, but that's what we called for the long-term core cooling evaluation model, that's what where that evaluation model came in.

The big, greater than 16-inch, hot leg

breaks were risk-informed, that means they were assumed to go to core damage, just so that they didn't have to do a RELAP5 evaluation of those. And the cold leg evaluations were all done by RoverD by comparing with WCAP-16793.

One thing I think I heard perhaps during the STP discussion was it was asked if all the greater than 16-inch breaks already went to failures due to debris effects. And they do not.

There are some hot leg 16-inch breaks that do not result in a debris failure. So they added a few breaks, I think maybe five for the two train case, to the core damage bin. I don't know if that was clear during the earlier discussion.

VICE-CHAIRMAN CORRADINI: Well, actually, I said it, and I probably said it wrong. So what I thought was occurring was that when I'm larger than 16 inches, I would have already covered that failure from the strainer failures. That's what I thought I was trying to say, but that didn't --

MR. SMITH: But you don't. That's incorrect. There's a few greater than 16 inch hot leg breaks that don't generate enough debris to cause the strainer to be considered to go to core damage. So it added a few large breaks to the risk.

VICE-CHAIRMAN CORRADINI: Okay, then, actually, I'm going to, don't move your slides, but I want to take a step back. Because you guys did a bounding analysis, which I think properly, in notes to myself and how I'm trying to inform the members. But in the staff's bounding analysis, you took some number, I think 12 point whatever.

MR. SMITH: Twelve point eight.

VICE-CHAIRMAN CORRADINI: For the two train and operations. And anything above that, you said, Let there be a failure.

MR. SMITH: That's right.

VICE-CHAIRMAN CORRADINI: And that's your bounding analysis.

MR. SMITH: Correct.

VICE-CHAIRMAN CORRADINI: Okay, and that delta CDF and delta LERF was all, of course, higher but still -- so do I have that correct?

MR. SMITH: Yes, you do.

VICE-CHAIRMAN CORRADINI: Okay.

MR. SMITH: So that analysis would have accounted for all those breaks, all the breaks that we're talking about. And then the other part of this slide is that the review concentrated on the criteria which came from WCAP-16793, for a maximum PCT and a deposit thickness.

VICE-CHAIRMAN CORRADINI: All right, look what I wrote.

MR. SMITH: And the next slide, this is just a summary of the conservatisms and simplifications that were used in the evaluation model. For the first one, the full core blockage, it's unlikely that it would occur, especially at the early time of five minutes. Very unlikely that it would occur at that time, or that it would be full core blockage.

The other thing is that the flow through the barrel baffle region was ignored. That would have allowed more coolant into the core. Other than, what was assumed was it just spilled over the steam generator tubes. There's other paths that the coolant could take that were just assumed not to be available.

VICE-CHAIRMAN CORRADINI: And just to ask there, because I think you guys made a point of this. It's just too hard to evaluate what the flow was. So because it didn't affect the end result, it conservatively was ignored. But for sure, there'll be some flow through the first two, the second and third bullet. Correct?

MR. SMITH: Yes.

VICE-CHAIRMAN CORRADINI: It's just too hard to know what it is?

MR. SMITH: And actually, looking into the future, 17788 is trying to quantify some of those flows through the barrel baffle region to credit them.

VICE-CHAIRMAN CORRADINI: Okay, fine. Thank you.

MR. SMITH: So that's a later topic I'm we'll be back talking about.

MEMBER MARCH-LEUBA: What I think you have told us is not hard to get an estimate of what those numbers are, it's how to validate that estimate. And that would have been a very large effort to validate.

MR. SMITH: Yes. And the flow, the trouble is that the flow -- those are relatively, I'm not going to say high resistance, but higher resistance flow passed through the barrel baffle.

So the flow actually doesn't, it's complex because the flow doesn't actually start going through those until you start getting debris built up at the core inlet. So it's a, you know, it's a dynamic problem.

VICE-CHAIRMAN CORRADINI: Okay, thank you.

MR. SMITH: They biased the key input parameter conservatively, and they also used a counter current flow model. It was a conservative model, it was implemented conservatively to reduce the flow through the core. So the implementation maximized the effects of the counter current flow, which would tend to reduce the flow through the core.

And the bottom line is this ended up in a simplified hot leg break simulation, which the staff was able to review and find to be acceptable. Is there any questions on the thermal hydraulics? Josh, I think you're off the hook.

Now I'm going to turn it over to CJ for the risk portion of the presentation.

MR. FONG: Thanks, Steve. My name's CJ Fong, I'm a team leader for risk-informed licensing at NRR.

Candace De Messieres, who you heard it from at the subcommittee meeting, couldn't be here today, but I did want to take a moment and thank her on the record. She was instrumental in producing not just these slides, but also the safety evaluation. So, certainly appreciate her help. Next slide, please.

As you heard mentioned earlier today, this was a very integrated review. So I'm going to talk about Principles Four and Five, which talk about risk and long-term performance monitoring. But of course, a lot of topics Steve just mentioned really fed into how we calculated that risk.

And then on topic five, or Principle Five rather, what we wanted to monitor and what portions of the licensee's analysis, that was really a team decision. So, again, these are presented as kind of five individual or independent principles, but in this review, the concept was that we did a very integrated, team-focused approach. Next slide, please.

So a little background. This is sometimes affectionately referred to as the most famous figure in risk, figure 4 from Reg. Guide 1.174. What we did here was we wanted to rely on the existing framework for risk-informed changes to a plant's licensing basis. So you see this stair step figure here, this defines how much of an increase in risk is acceptable to the staff.

And so for a GSI 191 evaluation, we look at the delta CDF, which is the CDF from the as-built as-operated plant, as realistically as possible with all the debris that's there, minus the CDF from a plant, a hypothetical clean plant, where debris would not present a challenge.

And of course, there's a corresponding

figure for LERF. I didn't show it here, but the risk thresholds are a factor of ten lower. Next slide, please.

So the major areas reviewed by the staff, again, kind of a recap. We looked both at the increase in risk with a delta CDF, and also the plant's baseline risk, in order to do a comparison with our acceptance guidelines. And so very comprehensive review.

We looked at the initiating event frequencies; what plant configurations were in place, in other words, the pump combinations; how the licensee identified the breaks to be evaluated; scenarios; what hazard groups were in play. A whole litany of things here.

I have highlighted two in red that we're going to talk about in a little more detail on future slides. And we also looked at the licensee's base PRA, and of course we leveraged Reg. Guide 1.200, and the results of previous looks at South Texas's PRA.

And so for the most part, the base PRA evaluation the staff did was to verify the CDF results and LERF results presented by the licensee, i.e., the X-axis.

Although I'll point out that some portions of the base model were used, both in identifying which sequences to look at it and also the percentage of time that the licensees was in, for example, a two train operation versus one train. So that base model did feed into the delta CDF calc a little bit. So maybe I should have had a dash there or going up there, but kept it clean.

And then, of course, we took a long, hard look at the delta CDF and delta LERF. And to do that, all the bulleted items that you see here were evaluated by the staff.

MEMBER STETKAR: In fairness, CJ, it did affect the conditional probability of being in a single train configuration versus not a single train configuration. But it also affected something that Dr. Corradini brought up before the fact, the likelihood that those containment fan coolers are running.

MR. FONG: Sure.

MEMBER STETKAR: Which substantially affects their delta LERF calculation.

MR. FONG: Yeah.

MEMBER STETKAR: And that came directly out of your base PRA model also.

MR. FONG: Absolutely. So yeah, it's,

like I said, you could argue there should be another arrow there, but just trying to illustrate the concept. Yup. Next slide, please.

So the really, the key assumptions that we noticed South Texas made in order to calculate the increase in risk, as was mentioned earlier, they relied on NUREG 1829, LOCA frequencies. And if you read NUREG 1829, there's a bunch of those. There's 25 year, 40 year, PWR, BWR, there's different aggregation schemes. It's not kind of a one-stop shop, it's more like a large menu.

So the staff took a long, hard look at which frequencies were actually used, and we noted that there's been a lot of discussion about aggregating the opinions of experts in different schemes, mainly geometric arithmetic.

And of course, we're aware of the ACRS feedback on that topic, both in this case, and also with 50.46(a) proposed rulemaking. So I think it was important to note that the licensee presented their results using both schemes, which is an approach recommended by the ACRS on several occasions, and the staff agreed with that.

Number two, you heard earlier a little bit about how the licensee took plant-wide LOCA frequencies that are provided by NUREG 1829 and allocated them onto individual locations.

So you'll get, for example, a value out of NUREG 1829 that says, Large break low frequency at a PWR is 5E minus 6. But it doesn't tell you how to allocate that onto individual breaks.

So South Texas did that using what we call the top-down approach, which assumes that the likelihood of a break is a function only of its size and doesn't look at location-specific factors, for example, what kind of weld it is, what shape it is, that sort of thing.

And there is some information in the literature that suggests that that can make a difference. And we'll talk about how that was addressed in a future slide.

The third topic is what South Texas called their continuum break assumption. And what they did was, they did consider partial breaks under this model. And they assumed that a break, for example, a partial break of six inches, has the same likelihood as a complete break of six inches.

Again, if you read NUREG 1829, there's some kind of qualitative thoughts on that. It's difficult to assign a numeric value to that.

And so what South Texas also did was consider an alternate assumption, which was that only double-ended guillotine breaks occur, or only complete breaks. And again, the staff felt that, lacking a clear-cut consensus approach, exploring these two sort of ends of the spectrum was a reasonable solution. Next slide, please.

MEMBER RICCARDELLA: So they did that as sort of a sensitivity study, and determined that it didn't affect the result?

MR. FONG: Right. They did both cases. It's kind of interesting, because you really can't say that one's more conservative than the other. It kind of depends on what scheme you are in and, you know --

MEMBER KIRCHNER: CJ, given what they were using it for, wasn't location more important that?

MR. FONG: Yeah, they always look at all the locations, right. For each location --

MEMBER KIRCHNER: Debris source term is the real issue here.

MR. FONG: It is, yup.

MEMBER RICCARDELLA: But they didn't consider, well, if something's a dissimilar metal weld that has known susceptibility to a mechanism.

MR. FONG: Right.

MEMBER RICCARDELLA: They didn't treat that differently than another on that doesn't have that concern, right?

MR. FONG: Exactly. And there are other approaches out there, albeit not endorsed by the NRC staff, but in academic literature, for example. It'd say, Hey, if it's a dissimilar metal, well, you should bump it up by a factor of ten.

Or if it's been recently, you recently had a weld overlay, you can reduce it by a factor of ten. So there's some thoughts out there on how to do that, but there's not a clear-cut consensus approach.

VICE-CHAIRMAN CORRADINI: But geometry dominated, and then if I understand this -- I still want to get to how they tried to do it compared to your bounding calculation. Geometry dominated, they came in said, If I had a double-ended at some scale, 12 point whatever, anything like that bigger would have failed, even though geometrically, it may not have produced as much debris.

You came in and then did a bounding on top of that and said, Okay, I'm not really sure about whether it's how you weight these, so anything above the 12 point whatever failed. And now did that the delta CDF or delta LERF enough to be concerned. And the conclusion was not.

MR. FONG: Right, and we'll show you numerically what that looks like in a second, Dr. Corradini.

VICE-CHAIRMAN CORRADINI: There are two kind of check calculations, one by them, one by you.

MR. FONG: Right, and then another way to look at it is we both felt that the 12.8 double-ended guillotine of the pressurizer surge line would be a core damage scenario under the conservative assumptions analysis, etc.

If you start going up and looking at bigger breaks, they have a number of them where they said, There's no core damage here because of where it's located. As a conservative assumption, we said, No, anything bigger than that goes to core damage. So we're kind of bounding all that with our approach.

VICE-CHAIRMAN CORRADINI: Okay, thank you.

MR. FONG: Next slide, please. So here's what we did, and this has been shared with the ACRS on a couple different occasions. We just assumed, conservatively, that the portion of risk attributable to debris, or delta CDF, is equal to the frequency of the smallest what we're calling critical break size.

Critical break means it can produce and

transport debris in excess of the tested amount. And it's not shown here, but we looked at this under the two train and single train configurations and used the appropriate waiting factors.

The graph I'm going to show you uses the arithmetic mean, but in our safety evaluation, we present several other cases as well. Next slide, please.

So what you saw at the subcommittee, we'll kind of work from the bottom up, there's a range of licensee-reported values. As I mentioned, they looked at arithmetic mean, geometric mean, continuum break, double-ended guillotine. And depending on what combination of assumptions you want to look at, those kind of light blue diamonds, thanks Lisa, indicate the increase in CDF.

The bounding calculation in our safety evaluation, using the method I just described, Dr. Corradini, is the blue diamond here. So it does make a difference, but still, substantial margin to the acceptance guidelines.

And then, in response to some of the discussion at the follow-up subcommittee, which is I think on April 18 or around then, I wanted to include -- we'd already performed this calculation, our consultant had already performed it.

But I want to include what happens if you assume that for the single train case, any break two inches and up goes straight to core damage. And that's the orange diamond you see there. So again, there is certainly an increase, which makes sense. But it's not something that challenges the acceptance guidelines. Next slide, please.

So to summarize, we looked at the licensee's base PRA and determined that it was of the appropriate scope level of detail and technical adequacy for this application. We looked at the approach they used to quantify that portion of risk attributable to debris, that was not initially in their base PRA, of course.

And we compared the calculated risk to the acceptance guidelines in Reg. Guide 1.174 under a variety of different assumptions, and we performed our own bounding calculation and sensitivity, and confirmed that the acceptance guidelines are met. Next slide, please.

Moving on to Principle Five, which is long-term performance monitoring, so we took a look at the risk assessment provided by the licensee. Of course, that's kind of a snapshot.

And so the concept behind the Fifth Principle in Risk-Informed Decisionmaking is you want to make sure that there aren't changes to the plant down the road that could erode the safety gains that were made or could undermine the assumptions that went into the risk calculation.

MEMBER STETKAR: CJ, I'm, sorry, can you back? You said something that caught my attention, and I want to make sure that I understand it. Can you go back to that picture that you showed on, that picture there. What is that orange diamond, that sensitivity thing? Can you explain that again?

MR. FONG: Yes, that's a sensitivity calculation provided by Southwest which assumes that, for the two train case, 12.8 is still the threshold. But for the one train case, instead of the nine inches and change, it goes down to two.

MEMBER STETKAR: Thank you for clarifying that, because when you first said it, I got the impression that you assumed that any two-train break, two-inch break went to core damage regardless of --

(Simultaneous speaking.)

MR. FONG: That'd be off scale high, I believe.

MEMBER RICCARDELLA: But when you say --

MEMBER STETKAR: It wouldn't be off scale high, it would be on the order of a couple times ten to the minus five. Off this scale.

MR. FONG: Yeah, yeah.

MEMBER RICCARDELLA: But when you say goes to core damage, that doesn't go to core damage with a probability of one, does it?

MEMBER STETKAR: Yeah. That would, the assumption is, that break generates enough debris that you would plug the strainers, that you would not have -- if you only had one train available, one set of pumps running. That's all you have, the other pumps all failed because they didn't have electric power or they, you know.

MR. FONG: Out for maintenance, whatever.

MEMBER STETKAR: Out for maintenance or, you know, whatever. If you only had one train running and you had a two-inch break, that would be the core damage frequency.

MR. FONG: Correct.

MEMBER STETKAR: Because you would plug the strainer for that one train.

MEMBER MARCH-LEUBA: No, it's a conditional probability of core damage where you need to multiply times the frequency.

MEMBER STETKAR: That's right.

UNIDENTIFIED SPEAKER: The frequency of the two-inch break.

MEMBER KIRCHNER: But that's the frequency of the two-inch break.

MR. FONG: Don't forget it's kind of a weighted value, right. Because the likelihood of being in a configuration where only one train's available is small.

Can we go back to the future here? Okay, so performance monitoring, as I said, it's a process to ensure that long-term, the assumptions made in the risk analysis remain valid. So the information provided by South Texas is that they're going to update their analysis every 48 months.

The staff looked at the procedures and controls they have in place to ensure that debris is prevented or mitigated if it's discovered. You heard about the tech spec this morning, and also the design control process.

There's also a provision that the NRC would be notified if the acceptance guidelines are exceeded. And that's all spelled out in the SR, again, as you heard this morning. Next slide, please.

So to summarize these last two principles, we feel that, you know, just kind of in plain English, the licensee provided a credible calculation for risk that met the acceptance guidelines. We acknowledged there were some portions of their analysis that didn't use a consensus method.

We felt they addressed that adequately, either through credible, reasonable, alternative calculations, or the staff's bounding calculations and sensitivity calculations.

And we took a look at their performance monitoring approach and determined that it was consistent with our guidance in Reg. Guide 1.174. Next slide. That's it.

MS. REGNER: So this is a summary.

VICE-CHAIRMAN CORRADINI: Green light. That's right, you're no worse than some of us.

MEMBER POWERS: Actually, you just provide an exercise for him.

MS. REGNER: Just trying to make you feel good, doctor.

VICE-CHAIRMAN CORRADINI: Thank you. I need everything I can get.

MS. REGNER: This is a summary of some of the major topics that were discussed during the subcommittee and where questions were answered by the

South Texas project. I didn't necessarily want to go into them unless there are questions.

MEMBER STETKAR: Lisa, are those supplements now on the docket with this application?

MS. REGNER: Both the email response is publicly available on the docket.

MEMBER STETKAR: Okay, but the second?

MS. REGNER: The second was submitted under oath and affirmation as a formal document, and that is recognized in our SE, correct. And there -- right. Okay, any other questions about these topics?

So in summary, STP acceptably evaluated the impact of debris, appropriately considered both risk and deterministic aspects. Most of the break scenarios are addressed using the traditional deterministic methods.

Their long-term core cooling evaluation, in this case that's the in-vessel thermal hydraulic analysis, those simulations are conservative. They meet the acceptance criteria. Their debris analysis meets the key principles of risk-informed regulation. And their probabilistic risk assessment results show that the change in risk is very small.

What questions do you have on what the staff has presented?

VICE-CHAIRMAN CORRADINI: Thank you, Lisa. Any questions from the members? They haven't been shy. Okay, so no more questions from the members. I'll thank the staff. And I think now we want to turn and see if there's members of the public, either inside the room or the phone line, and --

UNIDENTIFIED SPEAKER: Bridge open.

VICE-CHAIRMAN CORRADINI: Thank you. So are the members of the public, if you could please give us your comment. First identify yourself, please. Going once, going twice. Okay, why don't you close the line. Okay. I'll turn it back to our chairman.

CHAIRMAN BLEY: Thank you, Mr. Corradini. We will go off the record at this time until 12:45, when we'll reconvene. Members, wait. Off the record.

(Whereupon, the above-entitled matter went off the record at 11:19 a.m. and resumed at 12:44 a.m.)

CHAIRMAN BLEY: We are back in session. At this point, I'll turn it over to Dr. Rempe to take us through the work on our Consequential Steam Generator Tube Rupture. Joy.

MEMBER REMPE: Colleagues, today we are going to receive what I believe will be our final briefing on the report Consequential Steam Generator Tube Rupture Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes, or NUREG 2195.

This report documents results from a multi-year and multi-disciplinary effort that the staff completed to address the user need concerning CSGTR phenomena. The last time we met about this topic as a full committee was way back in May 2013. But since that -- really.

Since that time, we've had several meetings with the subcommittee on materials, metallurgy, and reactor fuels. And during our last meeting, which was December 2106, the members of the subcommittee that were present agreed that this effort was ready to be presented to the full committee for comment.

And I believe today that we're going to be starting by hearing from Kevin Coyne from the Office of Research.

MR. COYNE: Okay, thank you, Dr. Rempe. My name is Kevin Coyne, I'm with the Office of Nuclear Regulatory Research, and am their branch chief responsible for this effort since it kicked off in late 2009.

Thank you again for the opportunity to brief the full committee this afternoon. We're looking forward to the briefing and any additional comments we get from the committee.

Just to go through some very brief history, I believe Raj Iyengar will go through some of it in his presentation. But just to put this in context, this work was initially kicked off after the agency closed the Steam Generator Action Plan in December of 2009.

The focus of this effort was to take all that we learned from the Steam Generator Action Plan and focus on developing a better and more integrated and traceable approach for assessing the risk from consequential steam generator tube rupture events.

Needless to say, we had several challenges along the way. Most notably, we had diversion of staff to address higher priority work, such as the Fukushima follow-up, and we had some budget challenges. So the schedule for this work became much longer than we had initially assumed, and the scope of the work had changed.

So when you look through some of the user need documents, maybe some questions of how the work today looks as how it was initially envisioned. And it's because we kind of dynamically re-scoped things as the regulatory picture evolved over time.

Despite those efforts, there are several advancements that we came out with through this effort. We have a much better and improved understanding of how to characterize steam generator flaws in the steam generators based on operating experience, and the ability to align that flaw characterization to an actual operating plan, which is a significant advancement over what we had with the previous approach.

We have a full analysis of the combustion engineering steam generator geometries to complement the previous work we did on Westinghouse steam generators. We have a much better integration of the thermal hydraulic work, the CFD and MELCOR work that underpins the PRA analysis.

And improved validation of the structural modeling we use in our steam generator tube rupture calculator that helps us calculate the probabilities.

So we've talked about some of this in previous subcommittee meetings, and I won't go through that anymore. And we'll discuss portions of that as we go through the briefing today. But with that, and my remarks, and turn it over to the staff. And again, appreciate the opportunity to brief the committee today.

VICE-CHAIRMAN CORRADINI: Kevin, can I ask a question?

MR. COYNE: Sure.

VICE-CHAIRMAN CORRADINI: Maybe I'm, I want to understand. So assuming all of this is good, and it seems quite good to me, where is this going to be used? How is this going to be used? I'm vague on that.

MR. COYNE: So this is one of the things that was one of the big evolutions in our thinking when this work started. So if we go back to 2009, we had initially presumed there may be issues with the combustion engineering designs, with the geometry of the steam generator and how they would respond these severe accident conditions.

However, the staff hadn't fully documented the technical basis for that concern. So the thinking at that time was that this work would lead to a much better technical basis to support a potential backfitting analysis, for CE plants or at least a generic communication of stressing the need to keep water on the secondary side of the steam generators to prevent creep failure.

So that was before the Fukushima accident. As we progressed with the work in the Fukushima follow-up, the Flex Initiative came through. And in large part the Flex Initiative did for the steam generators what we were hoping this effort would do.

So the regulatory purpose of this work evolved from that, and so we have NRR here today to speak to their regulatory view and how this would be used. But right now, our focus is on developing improved tools to underpin the significance determination process. So some of the things we use for assessing steam generator issues --

VICE-CHAIRMAN CORRADINI: Can you stop there? Just help me, significant determination for current operating plan?

MR. COYNE: Yes, yes, the ROP.

VICE-CHAIRMAN CORRADINI: Okay, okay. All right, and that currently is the main point of application.

MR. COYNE: Right. In addition to making sure we fully document the staff work completed today, that would be one of the potential outcomes of this effort.

VICE-CHAIRMAN CORRADINI: Okay. The reason I'm asking the question is, for example, would it be used as part of an audit tool when the staff is looking at PRAs from other plants? Like APR 1400?

MR. COYNE: It's a good question. It could certainly inform that. And one of the initial purposes, and it's still useful to support this, is when we first had the user need from NRR, one of the questions, in addition to reactor oversight, was the evaluation of the SAMDA analysis for license renewal.

And so these types of scenarios come up in the SAMDA analysis. And the staff lacked an independent assessment tool to really look at some of the consequential steam generator scenarios that were coming up in the license renewal review. So part of the purpose of this was to provide an updated and independent evaluation of those methods.

MEMBER STETKAR: Okay, let me just say, from my perspective, Mike, I would hope that the staff would use it in that context. Because in particular, for anybody under Part 52, they are supposed to do a level two assessment of large release frequency.

VICE-CHAIRMAN CORRADINI: Right.

MEMBER STETKAR: As part of their PRA that's audited by the staff. And if this issue was not addressed at all, that could conceivably be an omission from the scope of their level two analyses.

Because this isn't, you know, it's not going to affect core damage frequency, but it could conceivably have a measurable effect on conditional large release frequency. So in terms of auditing, at least, you know, a question I would ask is, Have you considered it, and if not, that's not so good.

VICE-CHAIRMAN CORRADINI: But you wouldn't go so far as to change?

MEMBER STETKAR: I've got it for now. So for right now, significant determination, possibly PRAs for those that have geometric similarities to what you're considering.

VICE-CHAIRMAN CORRADINI: Any PRA should address it.

MEMBER STETKAR: Right, but it might not be important, for some plants, depending on their design, how many, you know, all of the other factors that feed into this. But it ought to be addressed. In terms of staff auditing, in terms of big picture things.

MEMBER REMPE: But this topic did come up in our review of the APR. And yeah, so I think it is important to have the staff make sure that all of the staff is aware of it. During our last subcommittee meeting, we did mention that this work is being used as part of the level three PRA?

MR. COYNE: That's correct. And notably, the tools that we developed as part of this work were applied for the global project.

VICE-CHAIRMAN CORRADINI: Thank you.

MR. IYENGAR: Okay, thank you, Kevin. I think Kevin already walked through some of these steps. When you start in 2009 and went through some tough times because of re-prioritization of our work. But we pulled through, and we engaged with you in a full committee meeting in 2013.

Since then, we prepared a NUREG, and we came and talked to you in 2015. But I want to tell you interim, we always had some informal meetings with Dr. Rempe and some of her colleagues present here. And then since then, we have prepared the NUREG. We came to you, we briefed you on the public comments. Yes?

MEMBER STETKAR: Just for the record, informal meetings with members of the ACRS have no bearing on the committee's deliberations. I just want to make that clear on the public record.

MR. IYENGAR: Right. Well, I'll come back to it, why I mentioned that. So now, I think after the meeting late last year, we had kind of a path forward with you.

And then we have certain things that we wanted to address in front of the committee, full committee, particularly focused on some thermal hydraulic issues that Mike Salay would be giving an overview on. Because other things that I guess have been satisfactorily addressed in the NUREG.

Before I turn over to Mike, I just wanted to mention that throughout that six-year period since 2009, or a seven-year period, we have been very fortunate to have the guidance and support from you, and the feedback has been very important. And we've had differing priorities. Some of us have actually taken on other assignments or other responsibilities in the agency.

But I wanted to highlight this, a single commitment of a staff is because we are still here. All of us are in a very different, some, like Kevin is in a different office and a different branch. But we are still here because we want to make sure this happens and this is closed satisfactorily.

It's very important, and just shows, this is probably only one of the many hundreds of examples you have seen of staff commitment. And throughout the process I also, I'm thankful to the ACRS committee members, in particular Dr. Rempe and Dr. Dana Powers.

I want to highlight they've been guiding us and prodding, if I may say, throughout the process, and ensuring that we are here at this stage. Thank you all very much. Mike.

MR. SALAY: Good afternoon, I'm Mike Salay, and I'll talk about thermal hydraulic overview and some response to the questions about the thermal hydraulic analysis.

Hadn't seen this before, so I will talk about CSGTR scenario description, the TH analyses that were done, briefly go over the method, and just a bullet on the experimental basis, and then discuss some of the differences between CE and Westinghouse plants.

The scenario that we're looking at is the station blackout. It's a low probability event, and combined with a loss of feed water to steam generators. The reactor inventory boils off, the system is at high pressure, and it starts heating up. And something in the RCS is going to fail.

It's either going to be the tubes or something else in the RCS. If the tubes fail, it can provide half-efficient products to the environment, bypassing the containment. If something else fails, the containment can contain what is released.

And so here you see Westinghouse, a fast Westinghouse scenario for -- oh, it's the PDF. Okay, for a fast Westinghouse scenario. This actually was an animation, but, oh well. Yeah, so you start with a full system, then you lose offsite power, the secondary starts draining down.

Your primary is lost only through a pump leakage, if any. When your secondary side inventory depletes, then your primary inventory starts boiling off. Then when your primary sedimentary hits the top of the U tube, your recirculation, the primary, of water, stops.

And the whole thing boils off, and you release, you develop a steam recirculation with, if your loop seals are closed, then the recirculation pattern goes up through the hot leg, through the steam generator tubes, back through the steam generator tubes, and back.

This occurs within four hours, and this is for a situation where AFW fails immediately. And this is from NUREG 6995, this work. And more likely scenarios involve operator actions to delay or prevent this from happening. So it will delay failure time.

And here the temperature traces for the

hot leg, hottest tube and average tube for the scenario. It shows steam generator dry-out, where we start delaying super-heated steam, and it shows the temperature difference between the hot leg hottest tube and your average tube. And then, your oxidation accelerates and temperatures rise rapidly before failure.

And there are a few points on the RCS that are of special interest. You have, it should be noted that you have different materials with different oxidation and melting temperatures. Another important aspect is your wall thicknesses.

Your thermal response time is quicker for lower thickness materials. Your steam generator tubes would provide a path for fission products containment if they fail to bypass containment, are very thin, five hundredths of an inch. Whereas your hot leg's about two and a half inches.

There's a bunch of other points of interest. I've listed them before and just put them on a slide here. And the situation that we're looking at is the so-called high dry low scenario, where your primary side of your steam generator is at high pressure. Your secondary side is dry and at low pressure.

MEMBER KIRCHNER: Do you have any sense for how strong any validation of this recirculation of the superheated steam in the steam generator?

MR. SALAY: Yeah, the --

MEMBER KIRCHNER: Have there been any experiments?

MR. SALAY: There've been the Westinghouse one-seventh scale experiments.

MEMBER KIRCHNER: I'm just curious how much the flow might stagnate rather than recirculate.

MR. SALAY: In? Well, if you have buoyant.

MEMBER KIRCHNER: No, I understand the thermal hydraulics. I'm just curious how strong this effect might be in the actual condition that you're describing.

MEMBER POWERS: Pretty effective.

MEMBER KIRCHNER: It is pretty effective?

MEMBER POWERS: Yeah, all the questions you have running through your mind like crazy right now arose when it was first proposed. And Westinghouse did their one-seventh scale with sulphur hexafluoride, and they tried, to the limits that you can, to scale things properly. They got pretty healthy flows, and stable flows.

Now, you ask what are the potentials. Well, clearly they didn't have the kind of radiant heat transfer that you would have in these accidents here, which is going to affect things.

MEMBER KIRCHNER: Well, I'm thinking of the pressurizer sitting there too, as to how that heats up.

MEMBER POWERS: Yeah, there's limits to how they do things, and I think they did have some simulation of the pressurizer. It's one of those tests that's kind of receded into the folklore of reactor safety. So it's a bit of a struggle to find things, but I think I do actually have the topic report if you want to see it.

VICE-CHAIRMAN CORRADINI: The assumption, though, in the cartoon is the loop seal is closed.

MR. SALAY: Here you're looking at two flow paths. There's one where the loop seal's closed, and one where your loop seal's open.

VICE-CHAIRMAN CORRADINI: But in terms of the analyses, the cartoon calculation you showed before, that's assuming it was closed.

MR. SALAY: The loop seal's closed, yeah.

VICE-CHAIRMAN CORRADINI: So --

MR. SALAY: If the loop seal opens, you

know, there's a --

VICE-CHAIRMAN CORRADINI: And the analysis you guys are doing considers either path?

MR. SALAY: No, we only look at closed. If the loop seal's open, you get, I mean, you're going to get a release. And was the assumption with Westinghouse, because your tubes get to much higher temperatures.

VICE-CHAIRMAN CORRADINI: No, I understand that. What I guess I'm kind of asking is, is there any deterministic, or how do you know if it's open or closed? That's what I guess I'm getting at.

MR. SALAY: Well, they looked at it in NUREG CR6995, they looked at it in quite a lot of detail. And I do go over it a little bit, but it's basically say they look at it as detail. Lots of parameters, they came up with little parameter maps to, when it, they calculated it to clear or not.

And so they came up with maps. And when it would and wouldn't clear, they point out that there were lots of things that would affect behavior that wasn't even in there, such as bypass leakage area between the downcomer and the upper core internals. Because you need to have a sealed lower head and a sealed loop seal to, it has to come up.

And the flow, if you have enough leakage, you don't have a pressure differential to, but --

VICE-CHAIRMAN CORRADINI: You don't have a pressure differential to blow it out.

MR. SALAY: Yeah.

VICE-CHAIRMAN CORRADINI: Well, the reason I'm asking the question, one is when we did small break LOCA spectrum analysis for Westinghouse, and I guess I can't say anything more than that in this meeting, the loop seal was forced to clear because that made it worse, right.

MR. SALAY: Yeah.

VICE-CHAIRMAN CORRADINI: In this case, I'm just trying to understand it. So the other question is, so Westinghouse is in the topical, they actually had, not criteria, but regions where it was clear and not clear. And so is it a race with the loop seal clearing also with, in terms of the hot leg heating up and opening, versus the tube?

That is, the prediction is both the mostly likely scenario is, in a Westinghouse design, is that I start this natural circulation, the loop is not cleared, I overheat somewhere near the pressurizer surge line, and I pop a hole. That's the likely scenario.

MR. SALAY: Yeah.

VICE-CHAIRMAN CORRADINI: And then how can I, in a Westinghouse geometry, get the loop seal to clear early because of the bypass? That's what I can't remember.

MR. SALAY: Well, if they had no bypass at all, I think Chris mentioned. I wasn't involved in that.

VICE-CHAIRMAN CORRADINI: Oh, okay.

MR. SALAY: This was in the steam generator action plan.

VICE-CHAIRMAN CORRADINI: Okay, how would you stabilized it.

MR. SALAY: But, no, it would have caused clearing almost every time. So if you have area, you don't get the DP, because --

MEMBER MARCH-LEUBA: Yeah, but first principle, without revealing any proprietary information, same as the bypass, it's clearing one seal. Once you clear one seal, you release the pressure in the upper plenum.

So if I were going to bet on something, it's at most one we clear. Because once one clears, the pressure in the upper plenum releases, and there is no pressure to push that wire on the other two loops or one loop. So there will be always one or two loops that will be closed. If there is sufficient bypass where you were pointing on core value then none will clear.

VICE-CHAIRMAN CORRADINI: Well, that's what I was thinking.

MEMBER MARCH-LEUBA: Yeah, that's a possible situation too.

MR. SALAY: This is one of the things that was deferred. We started looking at how we'd look at it and how we'd adjust the model to make, check the model deck to verify how it'd be. But then we never ended up doing that work.

MEMBER MARCH-LEUBA: Yeah, but my first principle modeling is that that drawing that you have in there, you don't need to put the dotted line in the middle. That's what you really have in the reactor. Half of it is clear, the other half is not.

(Laughter.)

MEMBER POWERS: Except that if you've got one loop that's clear, you're going to have such a ferocious flow through there, it's going to pop on the open line.

MEMBER MARCH-LEUBA: I mean, if you are going to pop, I'm not planning, but I would assume it's possible.

MEMBER POWERS: It don't take much.

MEMBER MARCH-LEUBA: If it's possible and likely to clear it, you should assume it cleared, unless you know better.

MR. SALAY: They also notice some effects with the pump suction elevation and nodalization would affect it. So they included that there was considerable uncertainty in whether it clears or not.

MEMBER MARCH-LEUBA: And your models, how long does it take to boil that? I mean, you have really, really hot steam out there. How come is that water staying liquid there? How long does it take to boil?

MR. SALAY: Yeah, this, well, had the previous, this actually was a little step through each step. So, you got dry-outs, steam generator dries out in 100 minutes.

MEMBER MARCH-LEUBA: That's the steam -- not the steam. I'm talking on the loop seal. The loop seal is leaking water in there and you have steam --

MR. SALAY: Leaking water, you have hot steam here, and then you have heat losses to the environment.

MEMBER MARCH-LEUBA: In your models, they never boil?

MR. SALAY: In ours, they didn't, and I'm not sure about the previous analysis. And I think the loops, in some analyses, the loop seals, in the ones we were using, the loop seals, were net condensing. But, okay.

Anyway, the full loop seal, full loop natural circulation occurs if your loop seal has been cleared. It's a severe challenge to your tubes. And I mentioned some of these things.

And the other scenario is the counter current, natural circulation. And if you're looking at, see whether you'd have a bypass, you'd expect the bypass in the full loop seal.

So you look at the counter current natural circulation to see if you get a bypass in those scenarios. And these are the analyses. So we ended up analyzing these rather than the others, unless you're actually looking at releases.

MEMBER KIRCHNER: What's your sense of how the pressurized and surge line play in this scenario?

MR. SALAY: Well, there is actually water level. It actually goes up and down. It actually went down, and then starts bubbling up and then gets full and then I can't remember. I have actually an animation that I wasn't planning to show, but I could show.

MEMBER KIRCHNER: Is that a heat sink effect that the pressurizer level goes up and down?

MR. SALAY: Well, you're having steam come up through here, and so it keeps it up. And so if you're, I mean, it depends how much your loss is. Could be heat losses, could be. I mean it was one of things, again, that we would have liked to look at, but.

MEMBER KIRCHNER: Okay.

MEMBER MARCH-LEUBA: I don't know, it's hard to believe that you have steam hard enough to melt the steel and keep water liquid.

VICE-CHAIRMAN CORRADINI: I think the point that Mike is getting at is, is that you've got your saturated water down here at the given pressure, but you're cooking it and it never sees that way down in the loop seal. You see it --

MEMBER MARCH-LEUBA: No, no --

MR. SALAY: He's talking about how do you still have water in the pressurizer.

MEMBER MARCH-LEUBA: You're bubbling superheated steam through that liquid in the pressurizer, and the liquid remains? I mean, you're not going to evaporate all of it soon.

VICE-CHAIRMAN CORRADINI: But the pressurized empties. I don't remember this station blackout scenario, but I think the pressurizer empties --

MR. SALAY: We had it. It would go up down. Yeah, and I think in Three Mile Island, it was water --

MEMBER POWERS: There was water all the time in TMI. In the pressurizer.

VICE-CHAIRMAN CORRADINI: It wasn't this long cooking it at these temperatures, was it?

MEMBER POWERS: No.

VICE-CHAIRMAN CORRADINI: Okay.

MEMBER POWERS: TMI was a wet plant. There was water going in nearly all the time.

MR. SALAY: So TH, there are two analyses. There is the Westinghouse analysis that was performed for the steam generator action plan. And this is documented in NUREG 699 -- CR 6995.

They did perform some CE analyses, but it didn't receive the same level of attention. And so they didn't update some of the models the way they updated their Westinghouse models.

Babcock and Wilcox plants were not analyzed because vigorous natural circulation flows were not expected. So both Westinghouse and Combustion Engineering TH analyses were used for the current work, and we did the CE TH analysis under this project.

We used a system code, MELCOR, and CFD code. CFD predicts the spatial flow and temperature distributions.

Your system code predicts the whole overall transient behavior. It uses the CFD results from both modeling and the results on the transient results can also be combined with those of the CFD to develop a transient spatial temperature distribution for your steam generator tubes.

So the CFD analyses were validated against Westinghouse one-seventh scale experiments, and they built up from those models to prototypic steam generator geometries. Multiple sensitivity studies were performed at the CFD.

And so here you see both CFD and system analyses. That's from NUREG 1922. So you have a CFD analysis, and then you have a much coarser nodalization in your system code, and you sort of have to transfer information from one to the other. They used parameters that were based on previous hand calculations.

And the CFD provides the hot leg flow rate by the use of the discharge coefficient based on the approved number of correlations. Inlet plenum mixing amount, hot tube fraction, recirc ratio. And it also provides the distribution of temperatures entering your tubes.

And this is shown by, given by a distribution, and it's normalized temperature. And there are differences between C and Westinghouse plants. In CE, there's considerably less mixing of the hot gasses before it reaches the steam generator tube inlets. This is because there's a lower hot leg length to diameter ratio.

And some CE plants have shallow inlet plena. There are few other effects in the CE plant analyzed. The hot leg comes in normal to the plates separating the steam generator plena. And whereas it comes in at an angle from Westinghouse, which adds additional mixing.

MEMBER REMPE: Mike?

MR. SALAY: Yeah.

MEMBER REMPE: For some reason, in our subcommittee meeting, I thought when we discussed how representative the example plans were of the whole fleet, that we heard back about ten percent responded back. But the geometries are about the same.

And now I see the words some CE plants have shallower inlet plena. Have you identified some CE plants that differ?

MR. SALAY: I know Chris asked around, and he's the one who answered that question. I can't, Chris Boyd is not here.

MEMBER REMPE: He did answer at the meeting. I just am puzzled because I thought you guys said, Well, as far as we know, but we only have ten percent. And now do you have knowledge that says --

MR. SALAY: No, no, I don't have any additional knowledge.

MEMBER REMPE: Okay, thank you.

MR. SALAY: Yeah, so under certain conditions, I mean because your temperatures are nearly as hot as your steam, the gas temperatures that your steam generator tubes are seeing are nearly as hot as the gas temperatures of the hot legs you're seeing. There is a potential for unflawed tubes being ruptured before the hot leg.

And since you have a lot of unflawed tubes, multiple unflawed tubes could potentially reach the failure condition at the same time.

And so here you see the Westinghouse model 51 and CE inlet plenum showing about the plume diameter. And you have about a one and a half plume diameters before you hit the tube sheet for CE, where you have four and a half plume diameters, hot plume diameters, before you hit the tube sheet. So there's more opportunity for mixing there.

And here you see CFD results for both CE and Westinghouse. And if you look, you can see at the hottest temperatures reaching the Westinghouse steam generator at about 0.6.

Whereas, the hottest for the CE are about one. And this is normalized temperature relative to the temperature difference between what the hot leg sees to the cold side of the steam generator.

So one means that the steam generator tubes are seeing hot leg temperatures. And so for the MELCOR CE calculations, the objectives were to provide thermal hydraulic results for CE plants to calculate failure using the CSGTR calculator and finite element calculations.

It's also to provide some scoping component failure calculations and to calculate some fission product releases, although that was de-scoped.

On the analyses, so for each case, you had to run for each event, you had to run two analyses. One for scoping, to see what MELCOR would predict to fail, and one in which component failure was suppressed to allow the other codes to, to provide input for the other codes to calculate.

And that short-term station blackout, long-term station blackout where the auxiliary feeding water were assumed to operate for four hours. And there were variations assumed, operator action and relief valve behavior and variation on reactor coolant pump seal leakage. Yeah.

MEMBER KIRCHNER: During the transit, what do you assume happens at the reactor coolant pumps seals when you're in the steaming configuration? Is there any loss of pressure there that's measurable? Or do you make a very conservative assumption that the seals block any steam release?

MR. SALAY: Essentially, it modeled the flow path. So, and with a flow area that would match the expected seal leakage rate and allowed the code to calculate. So it didn't --

MEMBER KIRCHNER: No, I understand seal leakage when you have water. I'm thinking now you're in a steaming condition. So what are the seals doing?

MR. SALAY: Yeah, so we just take the same hole and assume gas leakage from there. So --

MEMBER KIRCHNER: That doesn't help depressurize the system at all?

MR. SALAY: So, yeah, you would lose some gas. That's calculated by the flow solver.

VICE-CHAIRMAN CORRADINI: But to answer your question directly, everything is leaking. PRVs are opening, this thing is leaking. And so it's kind of sitting there.

UNIDENTIFIED SPEAKER: It's just chattering away.

VICE-CHAIRMAN CORRADINI: There is a surface station blackout analysis with MELCOR actually shows the seal leakage starting off at its prescribed ring then getting saturated. Pops up, water exits, steams comes in, and it kind of goes back down again, as Michael suggested.

MR. SALAY: And there was also an uncertainty analysis to determine the impact of thermal hydraulic uncertainty on failure timing. And here is some example of the MELCOR CE results, which this is RTP leak, reactor coolant pump seal leakage sensitivity on both pressure and temperature. And so that's what they look like.

MEMBER REMPE: So let's go back to that one just for a minute. As I recall in our subcommittee meeting, the reason that you are, what motivated that analysis was the comment from the PWR Owners Group, where they said, Hey, you've made the wrong assumption about the seal leakage.

And, again, my recollection was that you said, Well, okay, let's assume we don't have any leakage, and what happens.

MR. SALAY: Yeah, that's exactly.

MEMBER REMPE: And that was the reason you did this analysis and why you went forward with the same conclusion.

MR. SALAY: Yes.

MEMBER REMPE: Okay, thanks.

MEMBER BALLINGER: Do you assume that the seal leaks but remains intact?

MR. SALAY: Yeah.

MEMBER BALLINGER: Because the seals at least contain some stuff that'll handle pretty high temperatures, but some stuff which will erode very quickly. And so you end up with parts of the seal which are basically gone, and parts of the seal surfaces that are still there.

And so I'm guessing that you would get a lot of flow after a while. A lot of those seals are -- but they assume that they remain intact. They leak but remain intact.

VICE-CHAIRMAN CORRADINI: I'm not familiar with their analysis, but the MELCOR analysis for SOARCA has three levels based on time and temperature, if my --

MEMBER BALLINGER: Okay, because the graphite's going to be gone.

CHAIRMAN BLEY: If they get up to 500 GPM, that's going to add another seal there.

UNIDENTIFIED SPEAKER: That's about right.

CHAIRMAN BLEY: I don't know what they get.

UNIDENTIFIED SPEAKER: 500 GPM is an O seal.

MR. SALAY: Yeah, I think we just use a single seal leakage, and then yeah, the PWR Owners Group, they said you really wouldn't get any, or much, and I don't know. Do any of you have anything to do add or no? About seal leakage.

MR. AZARM: Yeah, I do. First if I might confirm what Mike said, we basically assumed an initial clearance of the seal, the spacing tack doesn't grow as the accident goes on. That's a question or assumption when you are in the regime of severe accidents and you are dealing with high temperature. As mentioned, lot of seals are going to fail.

I think we also, we haven't done the analysis but we talked about the a lot. The timing of failure becomes really important. So you know, like as Dr. Corradini was saying, you can have three phases, but it is important when is the threshold of those phases.

Because when your seal fails catastrophically, we are worried about the loop seal to get cleared. And if that happens, basically it's end of the run for us.

So, no, the analysis has been done, we have done quite a bit of thinking about it. We haven't done what SOARCA has done, and I'm not even aware of it, but we do understand the timing of seal failure plays an important role. And might be expecting high temperatures, but it was not within the scope of this analysis.

MR. COYNE: Mike, if I, Kevin Coyne from the Office of Research. If I could add when we re-scoped this work several years ago to try to finish it more efficiently, and I know this always dangerous to say in the severe accident space, but we tried to bias some of the assumptions towards minimizing the likelihood of CSGTR issues for the CE plant.

In other words, if we changed the assumption, the presumption would be it would get worse if we move the assumption.

And I know it's hard to necessarily hit that sweet spot in the severe accident modeling, but this is one case where loop seal clearing for the CE plant is just going to make the consequential steam generator tube rupture probability higher than what we calculate if we assume the loop seals are intact.

Now in the end, the temperatures the tube sees wouldn't change dramatically between the loop seal intact and the loop seal cleared. Is that correct?

MR. SALAY: Yeah, and so some engines in possible works --

MEMBER MARCH-LEUBA: Can you rephrase that? I mean are we talking about the loop seal or the pump seal?

MR. SALAY: The loop seal. He was mentioning the loop seal.

MEMBER MARCH-LEUBA: He was mentioning the loop seal. You weren't mentioning the pump seal.

MR. SALAY: Initially, yeah, we switched seals.

MEMBER MARCH-LEUBA: So if the pump seals or doesn't fail has not consequence that you can see with any certainty?

UNIDENTIFIED SPEAKER: We haven't done that.

MEMBER MARCH-LEUBA: But if the loop seal is open, then you have a lot more consequences. It's bad. Loop seal clearing is bad, pumps it clean, inconsequential. Is that correct?

MR. AZARM: I don't think I said that.

MEMBER STETKAR: In general, if the pump seals fail, the loop seal is going to clear it.

MEMBER MARCH-LEUBA: So that's really bad.

MEMBER STETKAR: That's not a good day.

MEMBER MARCH-LEUBA: Why would it clear?

MEMBER STETKAR: Because the pump's at the bottom of the loop seal.

MEMBER MARCH-LEUBA: No, no, no.

MEMBER STETKAR: It's a low pressure spot, and it'll flash it out through the seals.

MEMBER MARCH-LEUBA: If you say so.

MR. SALAY: Another factor is that your temperatures in CE are already that hot, so the loop seal clearing isn't going to make that big of a difference because you can't really increase the temperatures relative to the Westinghouse --

VICE-CHAIRMAN CORRADINI: I think the reason we're asking, or some are asking all these questions, is that these can change your timing. So I want to go back to what Kevin said to make sure I understood it.

So at least with the CE plant, or maybe for both Westinghouse and CE, you chose a set of conditions which would delay steam generator tube rupture, or enhance its timing compared to the hot leg creep rupture. That's what I didn't understand.

MR. SALAY: To paraphrase, I think Kevin is, we looked at Westinghouse and CE, tried to make what's the worst Westinghouse can get and what's the best CE can get, sort of take that.

VICE-CHAIRMAN CORRADINI: Okay, all right, so I did hear it right. What the best CE did.

MR. COYNE: With the presumption that if the reality was different from the assumption, it would only make the probability of the CSGTR get worse.

We were trying to get kind of a lower, these are dangerous terms for me to use, but a lower estimate of what the probability would be or a best estimate assuming optimistic --

VICE-CHAIRMAN CORRADINI: What if you say that, I was just asking it again because I didn't think I heard that, so I.

MR. SALAY: Would it be bad, even with the optimistic assumptions?

VICE-CHAIRMAN CORRADINI: Okay, you said it right. I'm with you, thank you.

MEMBER REMPE: To go back, just to make sure I understand in my mind, a long time ago when they did the Westinghouse plant, they did have a situation where you had a seal leakage rate and you increased it for the pump.

But I believe for your MELCOR calculations, you just left it at 21 gallons per minute until you did this thing for the PWR Owners Group. Is that a true statement?

MR. SALAY: We left it at the hole size that would give 21 gallons per minute.

MEMBER REMPE: Okay, so you did not increase the hole size.

MR. SALAY: We did not.

MEMBER REMPE: And so that's why your answer might have seemed a little fuzzy to some folks.

I was trying to clarify that. Thank you.

MR. SALAY: So it was interesting, but a lot of it was determined to potentially not be worth the effort. But some of the things that could be done in more detailed spatial temperature distribution, assessment of TH factors that impact relative failure timing, analysis of loop seal clearing.

Look at water hold-up in the steam generator and flooding counter current flows is known, so water was also held up in the previous steam generator action plan calculations. And a detailed evaluation of fission product release.

I mean, this analysis focused on thermal hydraulic input, not fission product release. So we didn't re-run cases when they failed solely for purposes of extracting of the fission product release behavior.

Now I'll go over some of the questions that were asked since the last meeting. So the recent questions were, temperature distributions, the impact of loop seal clearing. I'll sort of reiterate what I already said. And expected impact of models that have been subsequently added to MELCOR after this analysis, and a little bit on the TH uncertainty analysis that was done.

And so here you see the Combustion Engineering tube sheet inlet temperature distribution. This was calculated using fluent, anti-fluent, and what it shows is the normalized temperature distribution in terms of percentage of tubes of the whole bundle.

There are five data sets captured at the transient CFD calculations. And there are five data sets, so this gives you an indication of how the size of the plume changes and moves around.

One thing I should mention is for the Westinghouse, your distribution looks like this, where the maximum is about 0.5, 0.6. So here the hottest tubes for Westinghouse around up here, at 0.5, 0.6.

Whereas for CE, your hottest tube temperatures are essentially the hot leg temperature. It's about 1.5-2% of eight thousand-some tubes. So it's about 160 tubes up here are about as hot as the hot leg, the basic temperature is about as hot as the hot leg.

When your loop seal clears, in Westinghouse, you go from this distribution to something more that looks like what's on the screen now. And so your hottest tubes become as hot as hot leg.

MEMBER MARCH-LEUBA: Sorry, Mike, I'm confused. That represents five different calculations?

MR. SALAY: The same calculation, but it's a transient calculation.

MEMBER MARCH-LEUBA: This is a histogram.

MR. SALAY: It's a histogram, so the first one is some period within the calculations. It's as it's increasing in temperature.

MEMBER MARCH-LEUBA: So it's the same calculation.

MR. SALAY: It's the same calculation, but it's a transient calculation. But it's different snapshots of the same calculation. So --

UNIDENTIFIED SPEAKER: Of each?

MR. AZARM: I'm sorry, I shouldn't say, but think about this a time-dependent calculation. Do you remember that plume he was showing you --

MR. SALAY: The plumes moving around.

MR. AZARM: That thing is moving, and the number of tubes within that tube is moving. So he has taken five snapshots.

MEMBER MARCH-LEUBA: Five snapshots.

MR. AZARM: You know, between 0.2 to 0.25, you see five lines. In each of those snapshots, that's what I saw.

MEMBER MARCH-LEUBA: If I take the left line of every one of those histograms, that will be one of the --

MR. AZARM: One, yes.

MR. SALAY: Next one is ten seconds later, the next one is ten seconds later, the next one's ten seconds later, and the next one's ten seconds later. And so it gives an indication of how much it changes over time.

MEMBER MARCH-LEUBA: You have significant number on the high end, which is --

MR. SALAY: Yeah, I mean, that's important. And also concerned, at least in CFD analysis, was whether, for the Westinghouse CFD analysis, to what extent are your same tubes hot, and are the same tubes hot at different times.

So another thing about loop seal clearing, it was pointed out that many studies out there conclude that loop seals would clear before core damage. Do any scenarios indicate that, and discussed it earlier.

I mean, it was looked at in steam generator action plan and documented in NUREG 6995. They generated parameter maps that were based on pump seal leak rate, feed water operation time, number of PORV openings, and number of PORV opens, and when operator action started.

It was also found that it was sensitive to nodalization and core bypass area. And they concluded that uncertainty still remains on loop seal clearing.

The loop seals did not clear in any of the calculations we did for CE. However, it is a shallower. The initial scoping work, we did initial scoping work, but analysis of loop seal clearing was one of the things that was cut during the work scope reduction. So we never actually looked in detail.

It's also more important for Westinghouse plants because, again, it takes that normalized temperature and brings it nearly as hot as hot legs. Whereas for CE, you've got some tubes always at that high temperature.

And another question was that their models were added and were used in SOARCA. And these models were added after the version of the code used in this analysis. And the question was, Do these changes affect the conclusions of the study? And those analyses were done for CSGTR were done in 2011, 2012, there was some recalculation in early 2013.

MELCOR-186 was used for the CE CSGTR

analysis. There were different MELCOR-2 versions used in different SOARCA analyses. So the Surry SOARCA analysis, they did a comparison, they formed a comparison, well, they compared 186 and 2.1 timings.

And we can see here station blackout, station dry-out. It's a few minutes off, but the fission product gap releases were at the same time hot leg creep failure within a minute, even though the calculation was nearly four hours.

And I mean, all the way to accumulator injection, you're pretty close. So major event timing for that version, you wouldn't expect any changes for CE, since it's the same version. There have been some further model changes subsequent to that analysis. And the ones that could potentially affect behavior are the upgrade, update to the dry-out model and update of the declension models.

Both of these models, they affect behavior when you're reflooding or when the accumulators kicked in. And for interest for this analysis, we're looking at the situation before accumulator injection. And so we don't expect that these updated models would significantly affect or alter the report conclusions.

And there was a request for more information on certainty analysis. So the impact on certainties on thermal hydraulic-caused uncertainties was looked at upon initial decreation (phonetic), essentially how much do uncertainties in TH affect failure timing releases.

This is, people put a calculator, Ali was asking. And so Sandia performed uncertainty analysis on an early station blackout models prior to addition of hot tube final flows and prior to addition of heat structures for the tube sheet.

So they used an average hot tube for steam generators, it's a stress multiplier of two. Expect results to be reasonably representative of failure timing variation resulting from TH variations, but we don't necessarily expect the actual values to be representative.

So the analysis performed by sampling from the hydraulic parameters, and observed the effect on absolute and relative failure timing using a Monte Carlo sampling, 100 realizations. It was analyzed with the STEPWISE (phonetic) code.

MEMBER STETKAR: Mike?

MR. SALAY: Yeah.

MEMBER STETKAR: A hundred samples from any kind of reasonable analysis won't give you any sense of what the uncertainty is. So I'm curious what anyone learned from that exercise.

MR. SALAY: Well, just giving the results. And so --

MEMBER STETKAR: It is 100, that's not a typo somehow, that it was really 1,000 or 10,000?

MR. SALAY: I think it was 100.

MEMBER STETKAR: 100.

MR. SALAY: And these were the parameters. They were sampled, the discharge coefficients for PORV and SRV, the oxidation rate constant. The mixing parameters that were applied from the CFD. So, what if the CFDs off a little bit? Heat transferred multipliers for the outer tube wall and RCS containment heat transfer.

And these were distributions that they got out that with a 95% confidence interval for hot leg, and for the tube and hot leg. And this is the distribution for the relative failure timing in both.

MEMBER REMPE: But can you confirm that it is 100 is all you did for sampling or what was done?

MR. SALAY: I'm pretty it was 100.

MEMBER POWERS: I think you can be positive it was 100, not 1,000.

MEMBER MARCH-LEUBA: One hundred MELCOR calculations?

UNIDENTIFIED SPEAKER: MELCOR calculations, yes.

MEMBER MARCH-LEUBA: Of a thousand, two thousand seconds each? They were not a thousand.

MR. SALAY: The short-term station blackouts take about a week to run each.

MEMBER MARCH-LEUBA: I would have run only 59 myself. Which is the minimum number you're supposed to do.

VICE-CHAIRMAN CORRADINI: He's, 59.

MR. SALAY: And these are what they found to affect the failure timing. It was the RCS to containment, heat transfer multiple, see our oxidation rate, recirculation ratio in order of importance. Didn't look at, although it would have been better to look at also was the relative, what impacted the relative failure timing.

So these are the standard deviations for absolute failure timing. About six minutes, nearly seven minutes for steam generator absolute failure timing. Eight and a half minutes for hot leg absolute failure timing. And if you take the difference of the two distributions, well not the difference of the distributions, the difference in timing actually. The distribution of the difference in timing, and it's about seven minutes.

MR. COYNE: Mike, can I interrupt for a second? On the uncertainty analysis you just covered, was that, is SOARCA uncertainty analysis?

MR. SALAY: No, no, this was done at the beginning of the CSGTR project.

MR. COYNE: Okay, just wanted to clarify. Thanks.

MR. SALAY: Yeah, so we did perform MELCOR calculations for CE plant with replacement steam generators and provide that input to the CSGTR calculator and find out element component failure analyses.

Relative temperature increase rates and relative component failure time between steam generator tubes and other components is more important for releases than absolute failure time.

Some work was deferred because of limited resources, and many of these, the benefit was determined not to be worth the expense. And we received and incorporated useful feedback from both the ACRS and the public.

MEMBER MARCH-LEUBA: Can you go back to slide 38? All right, so am I reading this correctly that the tubes break before the hot leg?

MR. SALAY: Again, this was done before the models were completed, and I think that one of the major things that hadn't been added is the heat structures of the tube sheet. So it seems there is some heat loss there.

MEMBER MARCH-LEUBA: So with improved models, this would not be --

MR. SALAY: No, no, there were some cases where you did get, but most of the cases it was the hot leg that failed first.

MEMBER MARCH-LEUBA: I mean, if I look at that, some tubes fail at time 20, some hot legs fail at time 19. But those are different parameters here. Use the same parameters, the tube always fails before?

MR. SALAY: In this one, yeah. This is the difference of failure times. For this set of analyses, yes, the tube always failed first. But for the final analyses, it wasn't.

MEMBER MARCH-LEUBA: Maybe not 100% probability, but 25.

MR. SALAY: The point was to get how much variation in TH would affect them. So how much uncertainty in timing would this give them, for the failure calculator.

MEMBER MARCH-LEUBA: But no matter how you change parameters of conductants to containment or this and that or all the parameters you change, the tube always failures earlier.

MR. SALAY: For this one. But for the later analyses, it was the other way.

MR. COYNE: Mike, is it a true statement that you're using the MELCOR calculation to determine hot leg and tube failure for this calculation?

MR. SALAY: For this one, yeah. And this was also earlier too. Even for MELCOR with subsequent analyses, it went the other way when you added heat structures.

MR. COYNE: So these are simplified correlations that predict tube failure in hot leg and MELCOR. The actual -- for the project and the results presented in NUREG 2195, we used the steam generator calculator, which takes a thermal hydraulic output from MELCOR and represents the actual flaw distribution in the tube.

So it's a more realistic manner of doing that. What is the relative timing between hot leg and tube failure?

MR. SALAY: And this, yeah, just gives how much variations in TH would affect the relative and absolute time --

MEMBER KIRCHNER: Since he brought it up, may I ask, the flaw distribution, did that change the time to failure significantly?

MR. SALAY: For?

MEMBER KIRCHNER: Within the band of these kind of results you're showing, did the factoring that added level of fidelity, so to speak, into the calculations, did that have a significant effect?

MR. COYNE: So I'll start, and then I'll immediately turn to Ali Azarm, but one of the, I think, giveaways on this graph for me was this MP=2. So previous work used a pressure multiplier to represent degradation in the tubes.

And to be honest, it's very hard to correlate a pressure multiplier to an actual observed flaw characterization you get from a steam generator tube. So it was practically hard to work with, particularly if we were going to use this for, say, a STP-type determination for a particular plant. So it's hard to speak for the relative timing.

But the more recent work that Ali Azarm had led was to use operating experience to come up with distributions to characterize the flaw distribution. And then we can use that within the steam generator calculator to get a more, what I think it is a realistic prediction of the failure.

MEMBER KIRCHNER: Yeah, but Ali, when you did that, did you see a marked impact on the time to failure within the scope of the overall?

MR. AZARM: Let me first clarify, if I may, if you look at your NUREG document, I believe there's a bunch of graphs, like 7-25, 7-26, that has the distribution or the probability of public failure graph.

And then it has the graph for leak area. Because when we do the PRA for us, the two failures wasn't good enough. We needed to accumulate leak area of three centimeters squared, six centimeters squared, etc.

So yes, those graphs have been generated showing more or less similar behavior that with high likelihood that you're going to fail first. And you will get those leak areas. Now, the only problem with those graphs in NUREG is that it accounted for no uncertainty from thermal hydraulic.

I don't want to get -- for us, most of the system uncertainly, model uncertainty, it's epistemic. So basically, the graph stays the same, the confidence bound will be added. But right now, your NUREG gives you the blue curve, but not the confidence bound that is coming from the thermal hydraulic.

So the picture, you know, the glass is half-full. You have the uncertainty from material, from all other stuff, but you do not have the uncertainty from thermal hydraulic.

MR. SALAY: This is a scoping calc that doesn't include the flaw distribution. It just assumes a single large flaw.

MEMBER BALLINGER: In answer to your question, the answer is no. If you look at figure 5-6 in 2195, it shows predicted versus observed time to fail for flawed and unflawed tubes. And they lay on top of one another, the scatter is --

MEMBER KIRCHNER: It's almost like putting too much information into the estimate, given all the other variables and uncertainty.

MEMBER BALLINGER: At those temperatures, the creep rate is so high.

MEMBER KIRCHNER: Yeah, exactly. So I'm not a metallurgist, but I'm just saying.

MEMBER BALLINGER: The creep rate is so high that --

MEMBER KIRCHNER: That defect history is probably matched right away at these high temperatures.

MEMBER BALLINGER: Because you don't get cracks.

MEMBER POWERS: Not your problem there.

MEMBER BALLINGER: Not.

(Simultaneous speaking.)

MR. IYENGER: I did skip Selim's slide, I am sorry. I apologize for that.

MR. SANCAKTAR: That's okay. So after the fact, let me quickly say couple of closing things. The first bullet is already I guess obvious that we had multiple branches and fields involved in this, with its benefits and challenges.

And as we said in the past, most of the work was done in-house. PRA work was contracted out to IESS eventually, although it started without our vendor, it basically transferred to IESS.

Okay, and this we talked about, Fukushima, seven years. There was one more thing I wanted to mention. Not this, oh, here. The two bullets at the bottom.

So the next actions we have are have the draft NUREG go through NRC technical editing process, which turns out to be rather hefty. And then we will send it, the edited version, to NRC publishing, and then cross our fingers that it will go through the competing other publications and finish, hopefully we will finish it in the calendar year, not the school year but the calendar year this year.

And I keep mentioning this, because it's very dear to me, we are trying to get this grandfathered format-wise, because there is a new NUREG format.

MEMBER POWERS: God help us.

MR. SANCAKTAR: I don't know what else will happen. So we will ask for mercy and we grandfather this and use the existing format. Otherwise, we'll introduce all kinds of new challenges into the process.

So that's our expectation, and that's pretty much the scope that relates to this.

MEMBER REMPE: The one thing I didn't hear you discuss but you did discuss at the last subcommittee meeting is the guidance document that will be generated after this NUREG is done. And so could I have your --

MR. SANCAKTAR: Yes, I asked, and as I mentioned at the time, what form they are interested in seeing it. And they at that -- sorry. They indicated interest in a RAS section --

MEMBER STETKAR: Just for RASP? What is

RASP?

MR. SANCAKTAR: That's a very good question. Risk Assessment Standard --

UNIDENTIFIED SPEAKER: Risk Assessment Standard --

MR. COYNE: Standardization Process. It's a name that had meaning. He's said LOCA. No, I know what it is, but it's not like LOCA for the public. This is a public meeting.

MR. SANCAKTAR: Oh, yeah, you're right, sorry.

MR. COYNE: Risk Assessment Standardization Project.

MR. SANCAKTAR: Section, a technical description of RASP handbook, so that they can take it and put it in their format of more guidance for the actual practitioners. And I already put together, I distilled the PRA portion of this to about 50 pages. So I prepared something which I thought was pretty, which I hoped to be useful in a practical way.

And so we are going to see how the response will be, if that's a satisfactory format and detail, then we'll have that document that will be consistent with this NUREG.

MEMBER REMPE: Thank you. So are there

any other questions from members?

At this point, I believe we should ask if there's any comments that the members in the audience or the public would like to provide. And if so, either come up to the microphone and state your name, or, assuming that the line is open --

UNIDENTIFIED SPEAKER: It's open.

MEMBER REMPE: Thank you, please state your name and provide your comment. And not hearing any comments --

MR. SCHNEIDER: Well, hang on. Hang on. I just wasn't sure who's actually there. This is Ray Schneider from Westinghouse. I do have one question to ask. When you did the, I noticed the plot for the heat trays where you basically, it looks like you started to -- there's feedback on the line.

But when you started the calculation, it looks like the finite element or the finite difference calculation, whatever, was done assuming some kind of boundary condition on the detailed model --

MEMBER REMPE: Ray, I need to stop you right now because, one, you're breaking up a little bit. But two, you do need to realize that this is not a question and answer period. This is an opportunity to provide comments, okay.

MR. SCHNEIDER: Well, right, so I guess the, okay. I guess the comment is it doesn't look like the model considers the detailed upper head and upper plenum models for the combustion design, which may have resulted in mixing. Which would then change possibly the distribution that you're getting in the hot leg, which may make the feeding of the steam generator plenum a little different.

MEMBER REMPE: Thank you for that comment. Are there any other members of the public that would like to provide a comment? And hearing none, I'd like to turn it back to the chairman.

CHAIRMAN BLEY: Thank you very much. Thank you, John. Thanks, sorry for the -- at this point, we're almost on schedule. We're a little bit early. We are going to go off the record for the day.

(Whereupon, the above-entitled matter went off the record at 1:57 p.m.)

South Texas Project
Generic Safety Issue 191
Risk-Informed Resolution
Background and Overview

Lisa Regner, Senior Project Manager

Office of Nuclear Reactor Regulation
Division of Operating Reactor Licensing

Agenda

- Background
- Overview
- Licensee Methodology
- Staff Methodology
- Remaining Actions

Background

Generic Safety Issue 191

- Identification of safety issue
  - Sump Strainer Impacts
  - In-Vessel Impacts
- Staff developed three options, approved by the Commission

Option 1 *Compliance based on approved models*

Option 2 *Mitigative measures and alternative methods*

A. Deterministic – refined in-vessel testing

B. Risk-informed - STPNOC pilot

Option 3 *Different treatment for suction strainer and in-vessel effects*

- Other plants that plan to use Option 2B:

  Calvert Cliffs, Vogtle, St. Lucie, Diablo Canyon, Point Beach, Turkey Point, Palisades, Callaway, Wolf Creek, Seabrook

Overview

South Texas Project Review

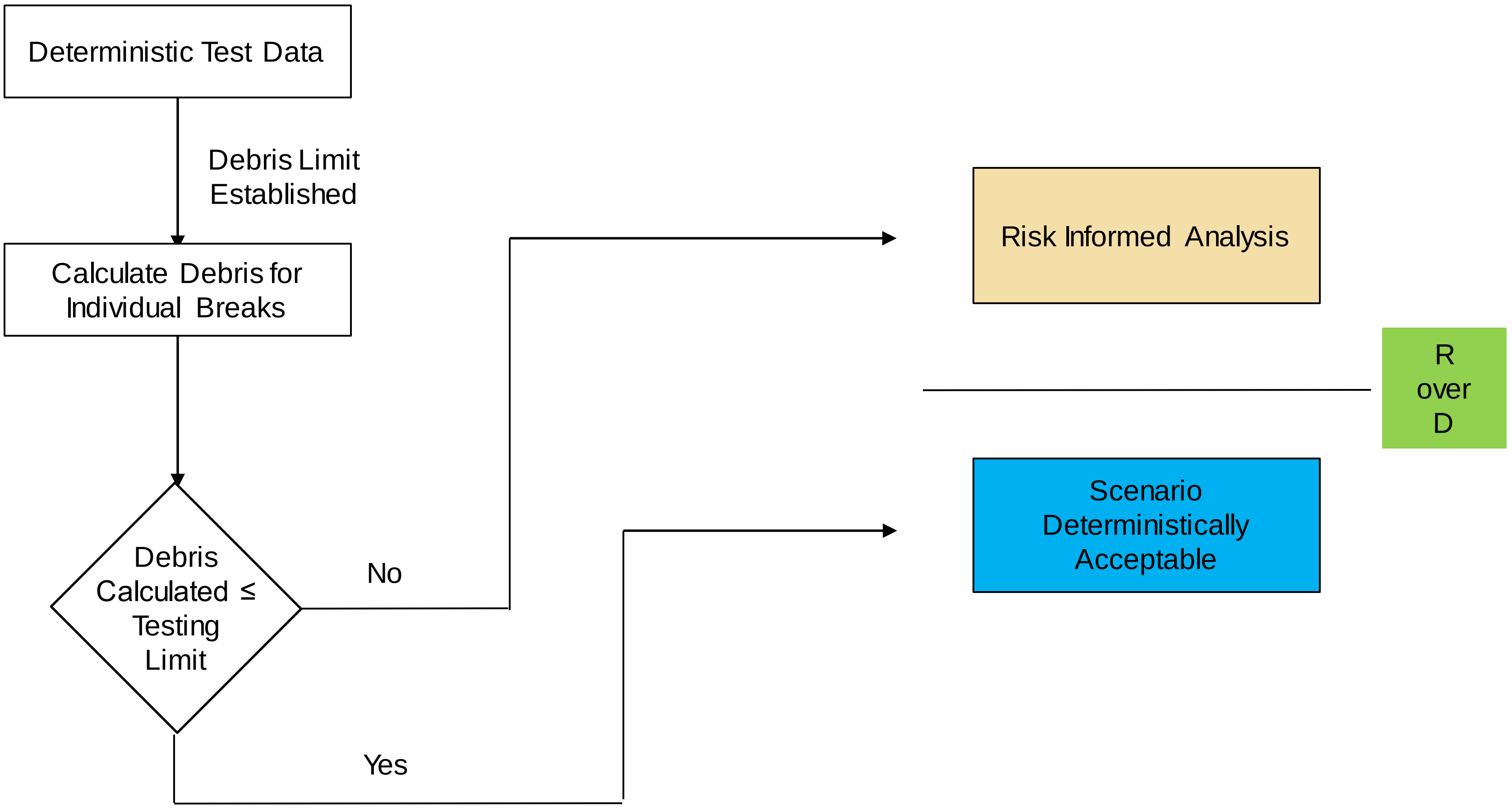
- Original request fully risk-informed
- CASA Grande
- Requests for additional information
- Public meetings
- Audits

Licensee Methodology

Risk over Deterministic (RoverD)

- Problems with original submittal
  - Uncertainties with head loss, chemical effects, debris transport timing, others
  - Epoxy coatings contributions
  - In-core thermal-hydraulic analysis
- RoverD was the significant turning point
- RoverD uses deterministic testing and analysis combined with probabilistic risk analysis (PRA)
- Reduced uncertainty in original submittal

Risk over Deterministic Methodology

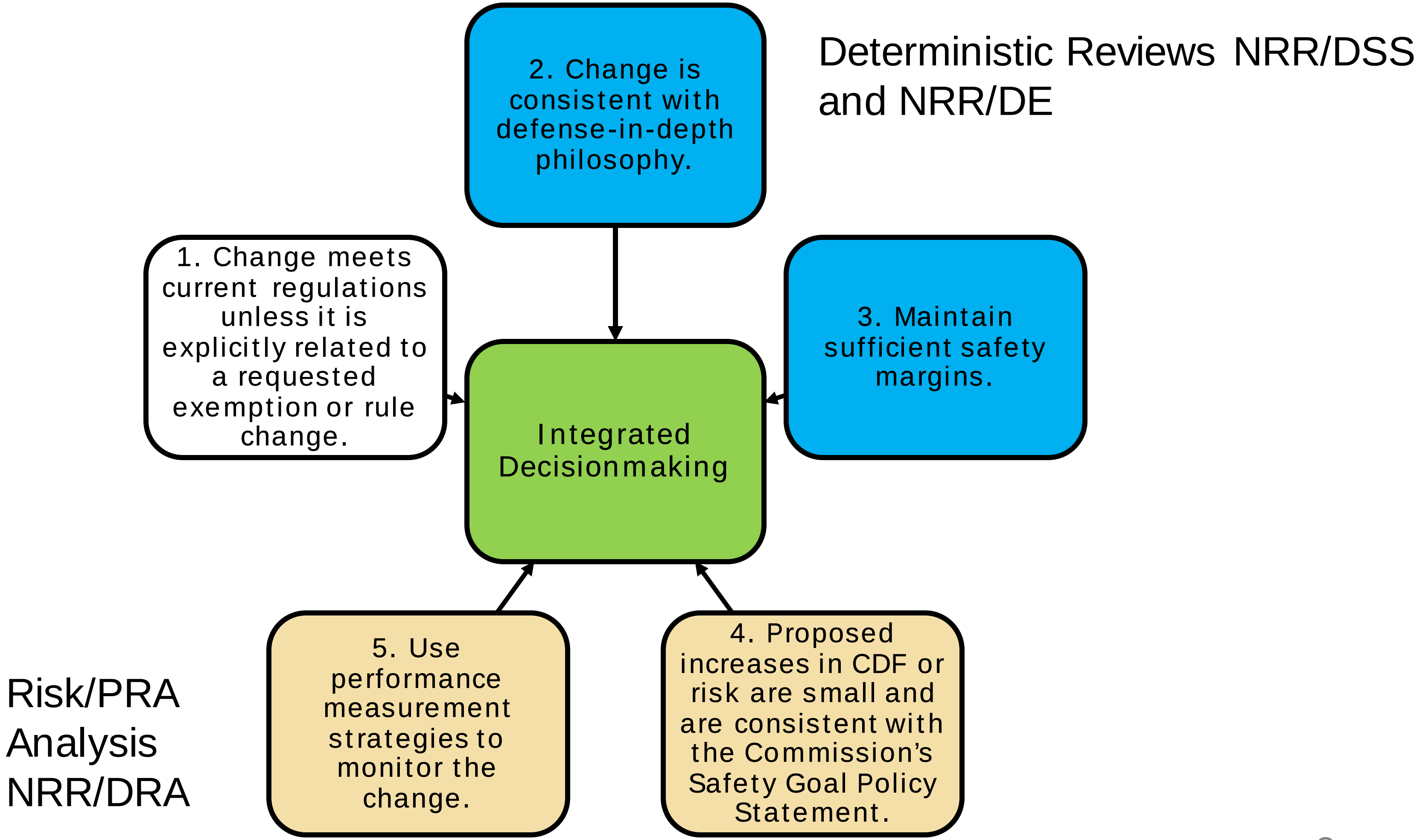


Staff Methodology

- Bases of review
  - **10 CFR 50.46** ECCS Performance Criteria
  - **NEI 04-07** "Pressurized Water Reactor Containment Sump Evaluation Methodology,"
  - **WCAP-16793** "Evaluation of Long Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid,"
  - **Regulatory Guide (RG) 1.182** "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident;"
  - **RG 1.174** "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decision on Plant-Specific Changes to the Licensing Basis"
- Technical Specifications change
- Structure of the staff's safety evaluation
- 5 Key Principles of Risk-Informed Regulation

Staff Methodology

Five Key Principles of Risk-Informed Regulation



Remaining Actions

- Complete concurrence process
- Resolve ACRS comments
- Coordinate issuance of final decision with internal and external stakeholders
- Issue final decision

STP Nuclear Operating Co.
Risk-Informed Approach to Generic Safety Issue-191 and Closure of GL2004-02: Assessment of Debris Accumulation on PWR Sump Performance

ACRS Full Committee Meeting

May 4, 2017

Introductions

- Introductions, Speakers
  - Mike Murray, Manager Regulatory Affairs, STPNOC
  - Ernie Kee, Risk-Informed GSI-191 Technical Team Lead, STPNOC
  - Wes Schulz, Design Engineering, STPNOC
  - Wayne Harrison, Risk-Informed GSI-191 Licensing, STPNOC
- Additional STPNOC Attendees
  - David Rencurrel, Senior Vice President, Operations, STPNOC
  - Rob Engen, Engineering Projects Manager, STPNOC
  - Steve Blossom, Risk-Informed GSI-191 Project Manager, STPNOC
  - Drew Richards, Licensing, STPNOC

Meeting Purpose

- Brief overview of history of STPNOC's risk-informed GSI-191 application
- Describe the risk-informed treatment of debris in the current "Risk over Deterministic" (RoverD) methodology and present results of the RoverD analysis

Agenda

- STP GSI-191/GL2004 Related Actions – Wayne Harrison
- General overview of the evolution of the STPNOC licensing application - Wayne Harrison
- General overview of the RoverD methodology – Ernie Kee
- Testing and deterministic element of RoverD – Wes Schulz
- Determination of governing break size and description of process for risk quantification – Ernie Kee
- In-vessel effects and thermal hydraulic analyses – Ernie Kee
- Quantitative Results – Ernie Kee
- Regulatory implementation – Wayne Harrison
- Closure – Mike Murray

STP GSI-191/GL2004 Related Actions

General overview of the evolution of the
STPNOC licensing application

Wayne Harrison

STP GSI-191/GL2004-02 Related Actions

- STP Units 1 & 2 have fibrous insulation on RCS
  - Large burden associated with insulation removal
    - Real occupational dose
    - Cost
- Actions taken
  - Replaced original three 155 ft^2 strainers with three new 1818 ft^2 strainers
  - Weld mitigation (overlay, replacement SG welds with low PWSCC susceptibility material, MSIP)
  - Replaced Marinite® insulation with NUKON

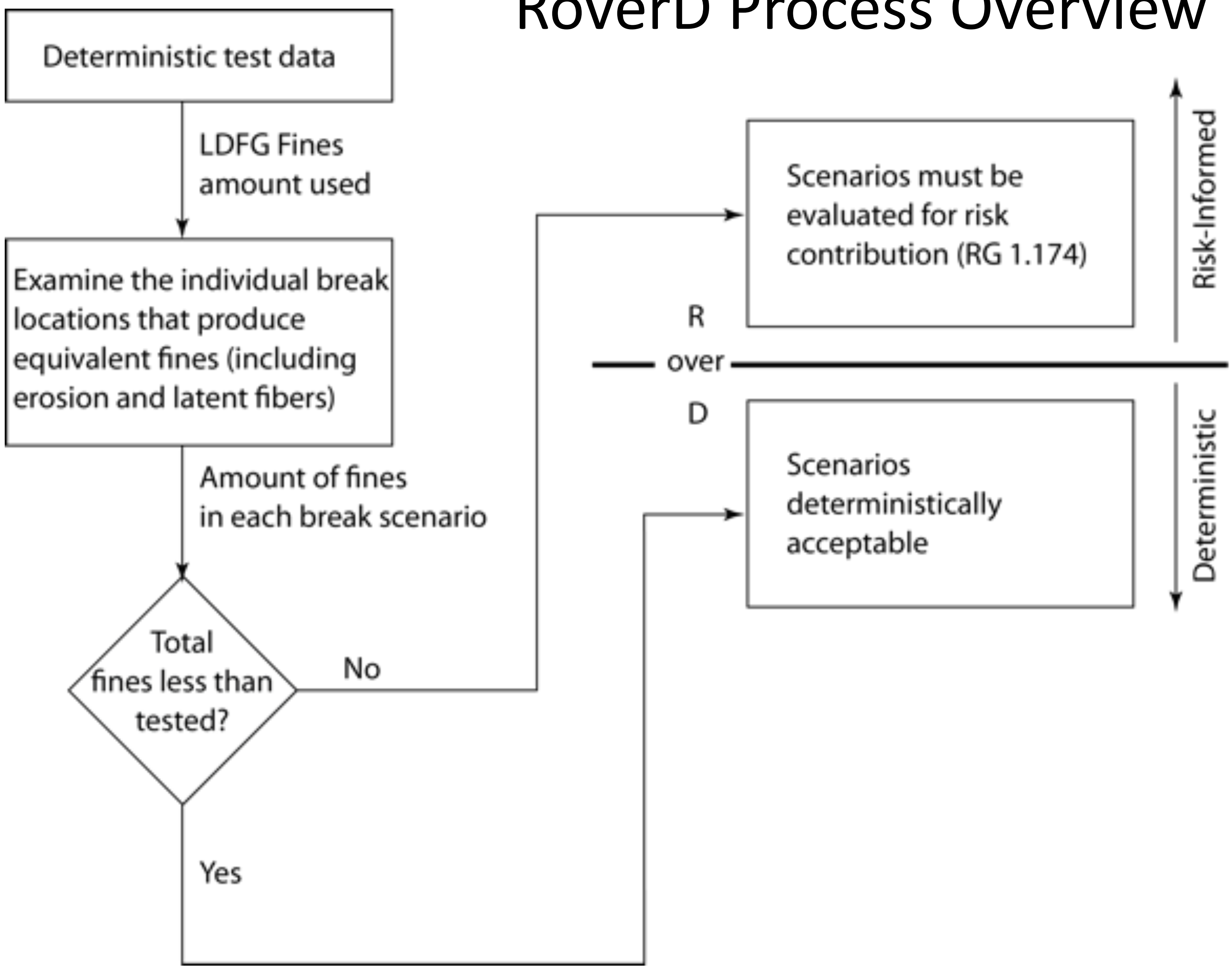
STPNOC Licensing Application

- January 2013: STPNOC requests exemptions from regulations that would enable the use of risk-informed methods where deterministic methods were previously required.
- Comprehensive model of debris generation and transport phenomena
- Coupled thermal hydraulic analyses
- Conditional failure probabilities input to STP PRA
- In order to reduce the complexity and scope of scenarios to review, in December 2014, STPNOC began a RoverD approach to bound uncertainties

General overview of the RoverD methodology

Ernie Kee

RoverD Process Overview



General Overview of RoverD Methodology

- RoverD simplifies complex risk assessment by using deterministic test data and bounding analyses
- Bounds on uncertainties make the assessment tractable, reviewable, and easily understood
- The STP PRA used to supplement the RoverD assessment with a few, easily understood evaluations

High Level Overview

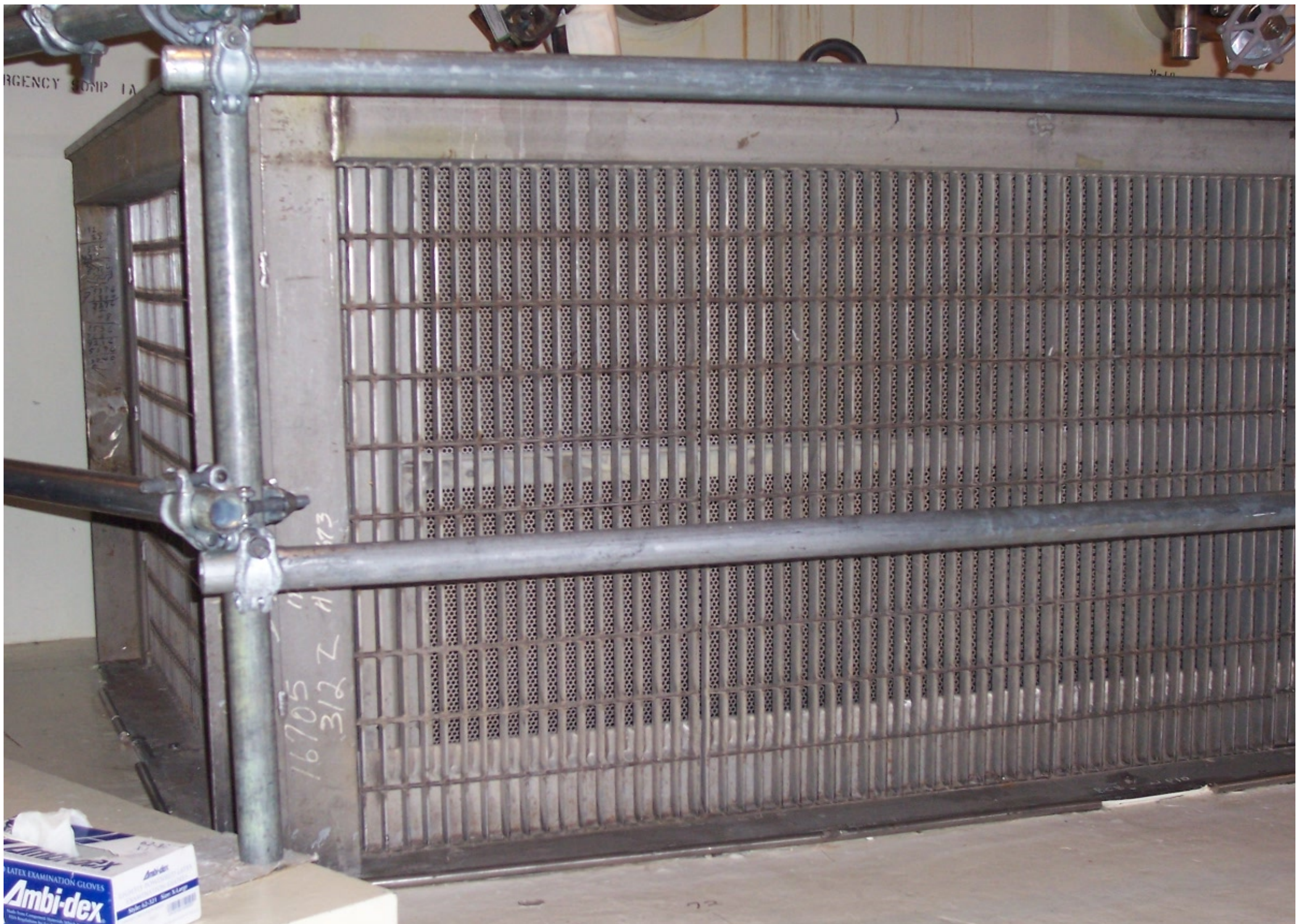
- Ensure tested fine fiber amounts bound all tested debris species on filter screens and fuel assemblies
- Assume scenarios that exceed tested fine fiber amounts lead to core damage and assess risk
- Confirm containment integrity is maintained for defense-in-depth

Testing and Deterministic Element of RoverD

Wes Schulz

Insulation

Original Strainer (155 Sq. Ft.)



New Strainer (1818 Sq.Ft.)

Zone of Influence for 31” Pipe Break

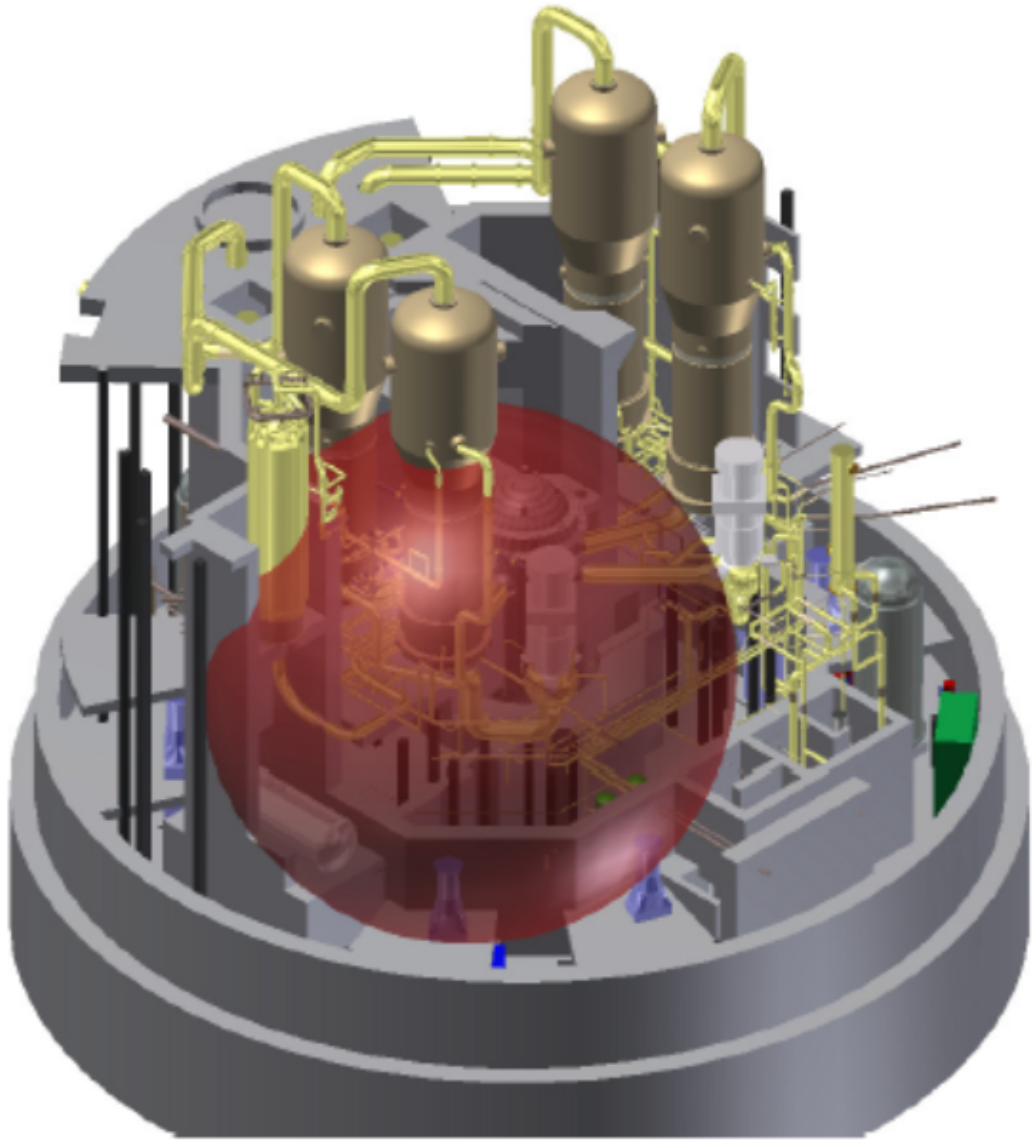


Figure 5.4.1 – Illustration of 17D Nukon ZOI for a 31” DEGB

Elements of Nov 2007 GL2004-02 Response Content Guidance

- Break Selection
- Debris Generation (ZOI)
- Debris Characteristics
- Latent Debris
- Debris Transport
- Head Loss and Vortexing
- Net Positive Suction Head
- Coatings Evaluation
- Screen Modification Package
- Sump Structural Analysis
- Upstream Effects
- Downstream Effects
- Chemical Effects
- Debris Source Term

Flume Test Description

- July 2008 flume testing at Alden Research Laboratory to satisfy GL 2004-02
- One full-size STP strainer module at design flow (20 modules per sump)
- Fiber, particle, and chemical loads scaled for 2 trains (out of 3) operating
- Flume channel designed to emulate approach velocity and turbulence

Strainer Head Loss Test Debris

Debris forms included in test:

- Low Density Fiber Glass fine and small fibers
- Particulates, Microtherm®, and Marinite® board particulates, latent dust and dirt
- Chemical precipitates representing 30 days of containment spray operation
- Coatings, zinc, epoxy, polyamide primer, alkyds, baked enamel

Reconciliation of Debris Used in Test

- Calcium Silicate (Marinite®) insulation was used in test. However this insulation type has since been removed from the containment building
- Subsequent analysis showed that the Microtherm® test amount exceeded that calculated to transport
- Subsequent analysis showed that the amounts of coatings particulate debris calculated to transport were under-predicted compared to the test
- The tested amounts of Marinite® and Microtherm® were shown to compensate for the under-prediction for the coatings particulates

Results

- Debris preparation and introduction procedures acceptable to NRC Staff
- Debris bed that formed with large quantity of particulate in combination with chemical load did not show need for additional thin bed testing
- Approximately half of head loss was due to chemical precipitates
- Successful test satisfies failure concerns up to the level of the tested debris loading
- Direct comparison of break spectrum to test results eliminates need for head-loss correlation

RoverD Risk Element and Results

Ernie Kee

Risk Element of RoverD

- RoverD scenarios begin with a break at a particular location (many thousands of scenarios are created)
- Use CASA Grande to deterministically calculate debris generation, transport, and erosion to the RCB floor pool
- Scenarios must meet deterministic criteria (upstream, downstream, in-vessel, RCB integrity criteria are met) or be categorized as risk-informed

Risk Element of RoverD

- Scenarios that introduce more fine fiber than tested are assigned to the risk-informed category
- Fetch the smallest break size among any at the location to be used for risk estimates
- Interpret NUREG 1829 for total frequency of risk-informed scenarios and assign to core damage to determine ΔCDF
- Calculate ΔLERF from PRA assessment

HLB PCT Simulations

- Assumes core fuel assemblies and core barrel bypass channels are fully blocked
- Thermal-hydraulic simulations show core cooling requirements are met for largest HLB break that is deterministically acceptable (16”)

CLB - Core Fiber Analyses

- Uncertainty in core fiber buildup in CLB is assessed using bounding limiting analyses for strainer flow and RCB floor pool fiber concentrations
- The worst case fiber buildup cases show that the industry bounding fuel fiber cooling test (WCAP 16793) criteria are met

Single Train Assumption (ACRS Subcommittee Question)

- With CS and ECCS pumps running, the total strainer flow in the strainer is 7220 gpm, or about 200 gpm more than for the tested, 2 train flow
- Using the deterministic test, full flow on a single strainer is not bounded.
- Single train scenarios can be added to the risk-informed category or screened based on risk evaluation

Risk Assessment Results

- Measures taken by STPNOC minimize the risk of concerns raised in GSI-191 (ΔCDF - 1.50E-07, ΔLERF - 3.75E-10)
- Significant safety margin is included
- Defense in depth is maintained
- Results from different approaches consistently show minimal risk from the concerns raised in GSI-191

Summary

- RoverD is a framework that makes GSI-191 risk assessment understandable and easy to review through use of conservative testing and bounding analyses.
- Scenarios fall into two categories by application of accepted testing methods and bounding analyses
- Many additional supporting tests and analyses help support the conclusions and are publically available on the docket and other academic literature

Regulatory implementation

Wayne Harrison

Regulatory Implementation

- Debris-specific action for Mode 3 and above ECCS and CSS Technical Specifications
- UFSAR changes
- Exemptions to permit use of risk-informed approach instead of prescribed deterministic methodology

Regulatory Implementation

ECCS Technical Specification change (CSS similar)

With less than the required flow paths OPERABLE solely due to potential effects of LOCA generated and transported debris that exceeds analyzed amounts, perform the following:

1. Immediately initiate action to implement compensatory actions,

   AND

2. Within 90 days restore the affected flowpath(s) to OPERABLE status,

OR

Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Closure

Mike Murray

Conclusions

- The RoverD process incorporates all aspects of the debris issue
  - GL2004-02 closure
  - Deterministic (testing for fiber and chemical effects)
  - Debris generation and transport
  - In-core effects
  - Risk-informed evaluation
- RoverD meets RG 1.174 acceptance guidelines with defense in depth and safety margin

Staff Review of STP GSI-191 LAR

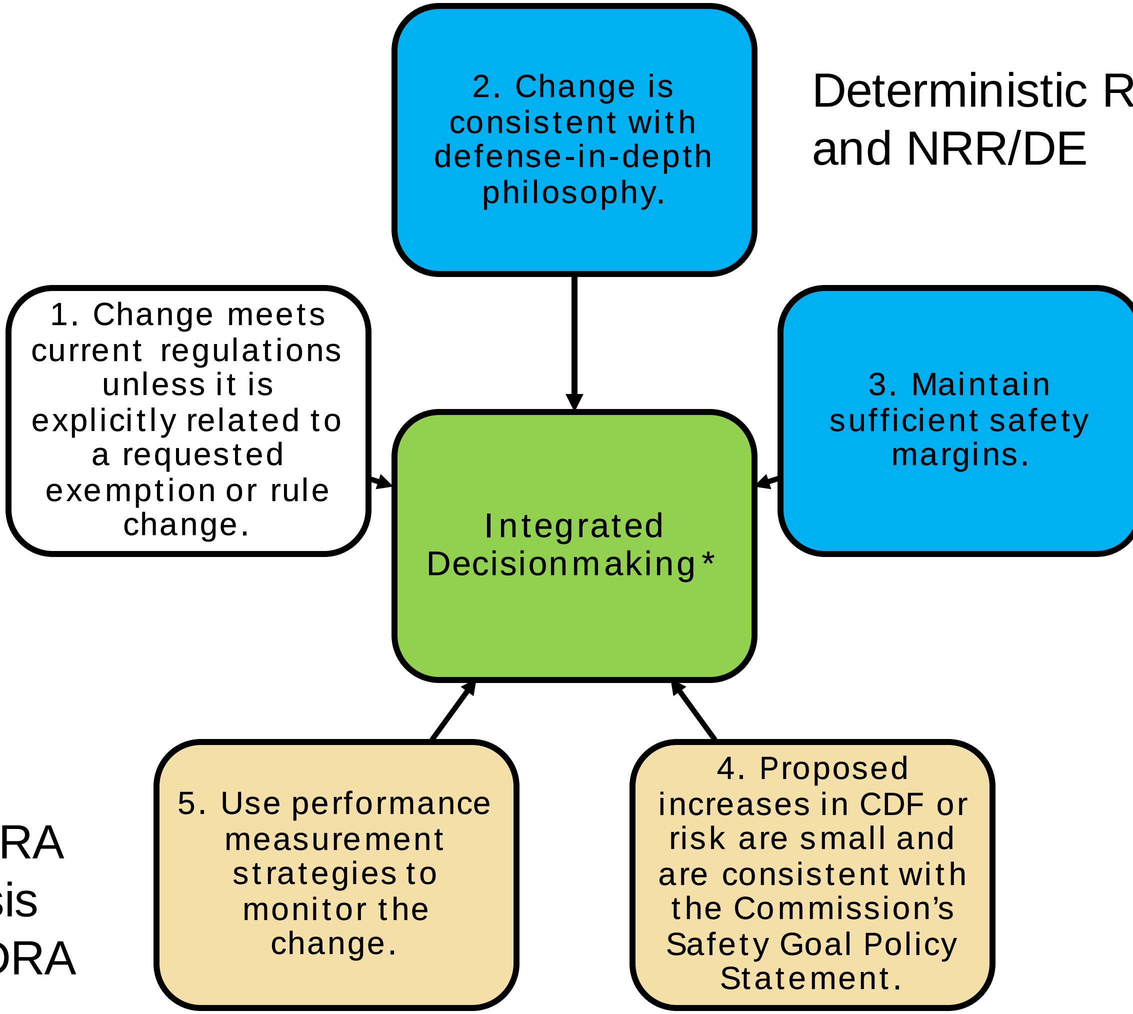
Principle 1: Meets Current Regulations

Lisa Regner, Senior Project Manager

Office of Nuclear Reactor Regulation

Staff Methodology

Deterministic Reviews NRR/DSS and NRR/DE



Risk/PRA Analysis NRR/DRA

* Principles of Risk-informed Integrated Decisionmaking from Regulatory Guide 1.174, Rev. 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis" (ADAMS ML100910006).

Principle 1
Risk-Informed Regulation

"The proposed change meets current regulations unless it is explicitly related to a requested exemption or rule change."

- 10 CFR 50.46c rulemaking status
- Exemptions requested from use of deterministic analysis method
  - Acceptance Criteria for emergency core cooling systems (ECCS)
  - General Design Criteria associated with ECCS, containment heat removal, and containment atmosphere cleanup

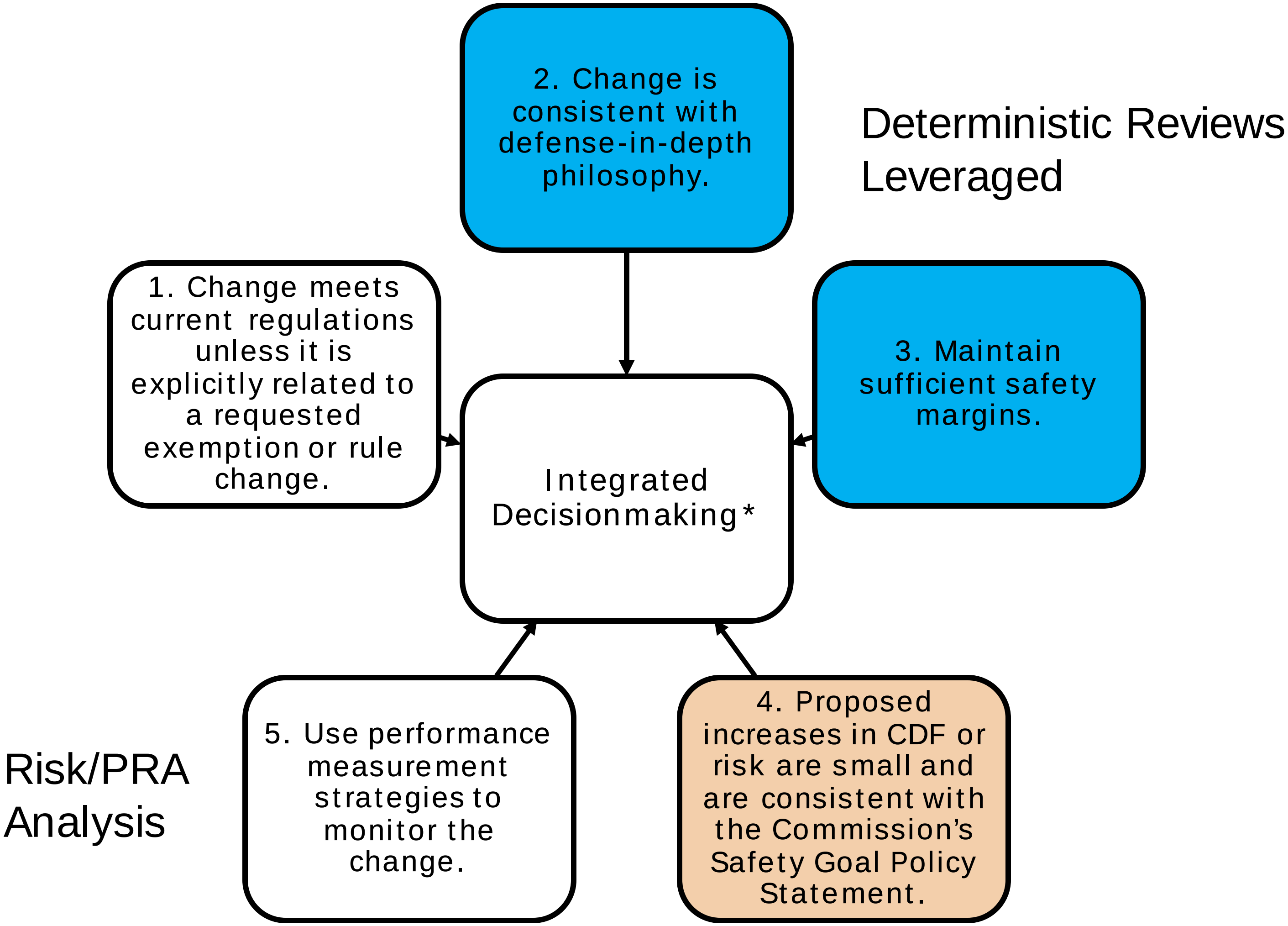
Staff Review of STP GSI-191 LAR

Principle 2: Defense-In-Depth
Principle 3: Safety Margins

Steve Smith, Senior Reactor Systems Engineer

Office of Nuclear Reactor Regulation

Integrated Decisionmaking



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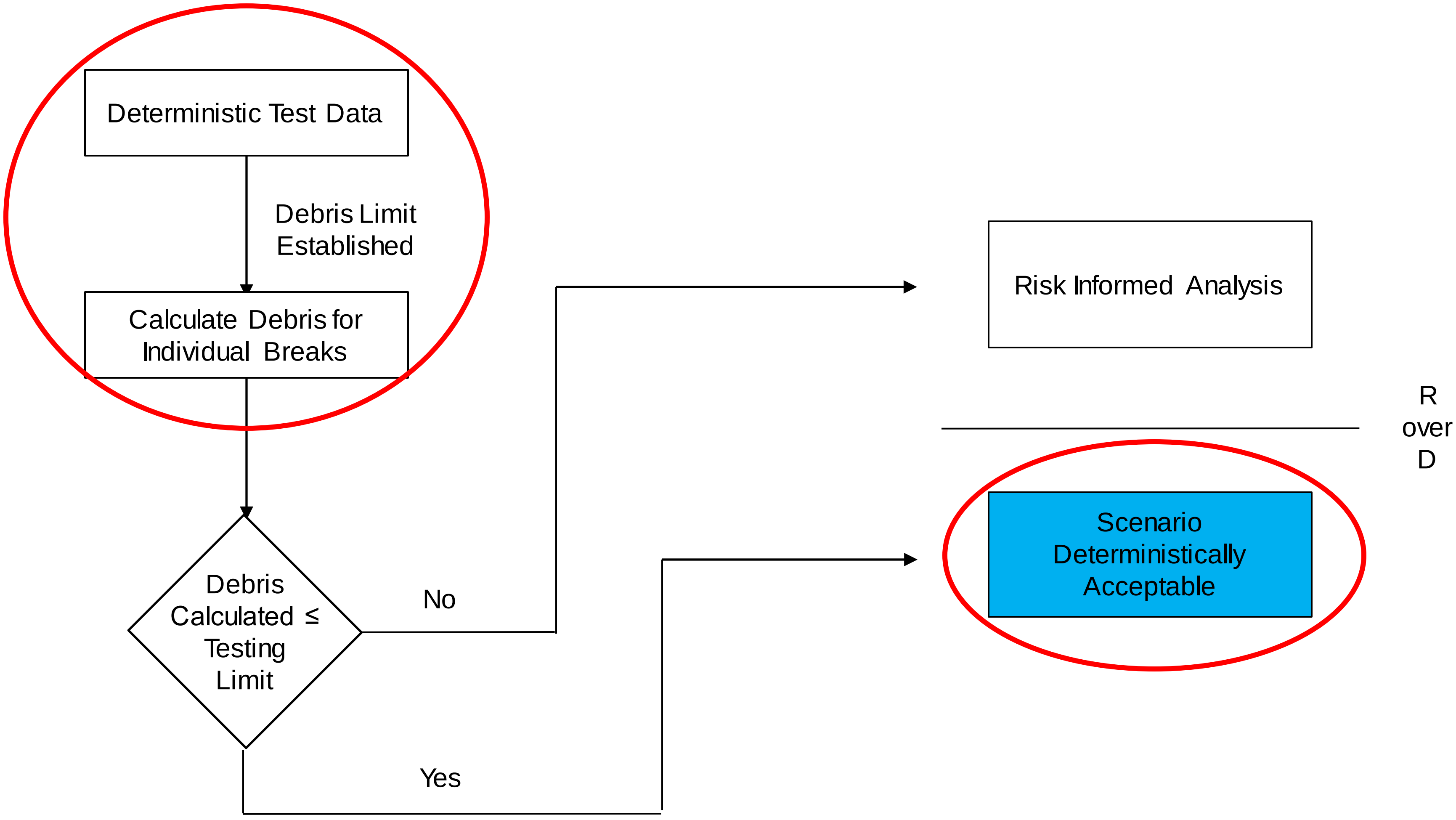
Risk over Deterministic Methodology



Principles 2 and 3
Safety Margins
Defense-In-Depth

- Licensee met guidance of RG 1.174 and listed significant Safety Margins and Defense-In-Depth (DiD)
- Safety Margins include construction and inspection per industry codes and the use of licensing basis values when assigning strainer failure criteria.
- DiD includes actions identified that are taken in response to the loss of the normal ECCS function. DiD also includes verification that balance is maintained among prevention and mitigation, redundancy is maintained, barrier independence is maintained, etc.

Principle 4
Deterministic Inputs to Risk Analysis

- Debris Source Term
  - Used NRC approved guidance for all areas
  - Calculations performed in CASA Grande
  - Differences from typical deterministic evaluations
    - For partial breaks, all weld locations evaluated for multiple orientations instead of focusing on the limiting large break
    - Double-ended guillotine break (DEGB) source term uses the same method as typical deterministic calculations
  - Source term calculated for each break and compared against tested amount
  - The most conservative orientation was selected for partial breaks at each weld location
  - Assumptions and calculations independently verified by SwRI

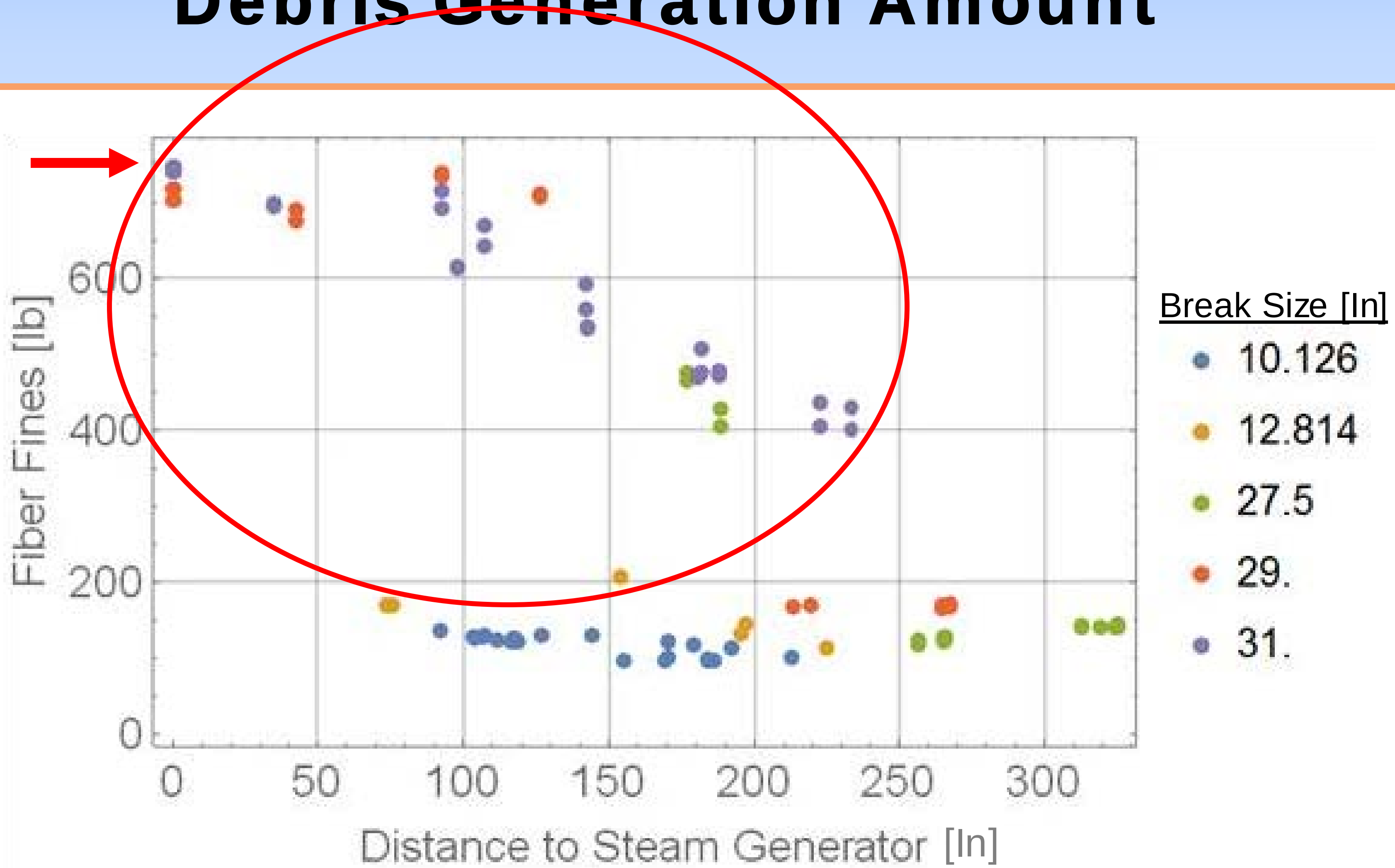
Principle 4
Deterministic Inputs to Risk Analysis

- Debris Transport – Strainer Evaluation
  - Used NRC approved guidance implemented via CASA Grande
- Debris Transport – In-Vessel Effects
  - Fiber penetration determined via testing
  - Used conservative bypass values from testing
  - Calculated fiber amounts arriving at the core for cold-leg breaks considering varying plant states (pump combinations)
  - Determined fiber amount reaching the core is small in all cases (2 g/FA design basis, 4 g/FA 1 LHSI, 7 g/FA 1 HHSI)
  - Calculations independently validated by SwRI

Principle 4
Deterministic Inputs to Risk Analysis

- Impact of Debris – Strainer
  - Strainer evaluated at tested debris load/dP for net positive suction head, structural, deaeration, vortexing, and flashing
  - Testing and evaluations were performed using staff approved guidance
  - Testing shows that increasing fiber amounts results in greater head losses
    - Majority of breaks were bounded by 2008 test results
    - Some breaks generate much larger debris amounts

Principle 4
Debris Generation Amount



Principle 4
Deterministic Inputs to Risk Analysis

- Impact of Debris – In-vessel – Cold-Leg Break
  - Debris amounts low enough to permit adequate cooling flow to the core based on WCAP-16793 findings
  - Boric Acid Precipitation not resolved by the LAR because staff has no basis to conclude that any amount of debris will not reduce mixing with the lower plenum
  - Previous staff conclusions indicate that the STP debris amounts do not result in a significant impact to boric acid precipitation (BAP) timing conclusions currently assumed by STP
  - Licensee to address BAP for the CL break at a later time

Staff Review of STP GSI-191 LAR

Deterministic In-vessel

Steve Smith
Joshua Kaizer, PhD
Division of Safety Systems
Office of Nuclear Reactor Regulation

In-Vessel Deterministic: Review Goal

Goal

- To determine if the LTCC Evaluation Model (EM) provided credible results which could be trusted for reactor safety analysis.

Solution

- All large breaks were treated with risk (removes the need to model complex phenomena)
- Focus only on “long term” portion of the event (removes the need to validate complex phenomena associated with blowdown, refill, reflood)

In-Vessel Deterministic: Review Scope

Break Size	Hot-Leg	Cold-Leg
Small	**LTCC EM**	RoverD
Medium (< 16”)	**LTCC EM**	RoverD
Large (≥16”)	Risk Informed	RoverD

Criteria (WCAP-16793)

1. Max PCT < 800 ºF* - LTCC EM (SRP 15.0.2)
2. Deposit thickness < 0.050 inches

* Preferably, not above saturation (reduces complexity)

In-Vessel Deterministic: Summary

Conservatisms / Simplifications

- full core blockage
- ignoring flow through the barrel-baffle region
- ignoring flow through the holes between the barrel-baffle region and the core
- biasing key input parameters conservatively
- using a conservative counter current flow limitation model and core modeling

Simplified hot leg break simulation

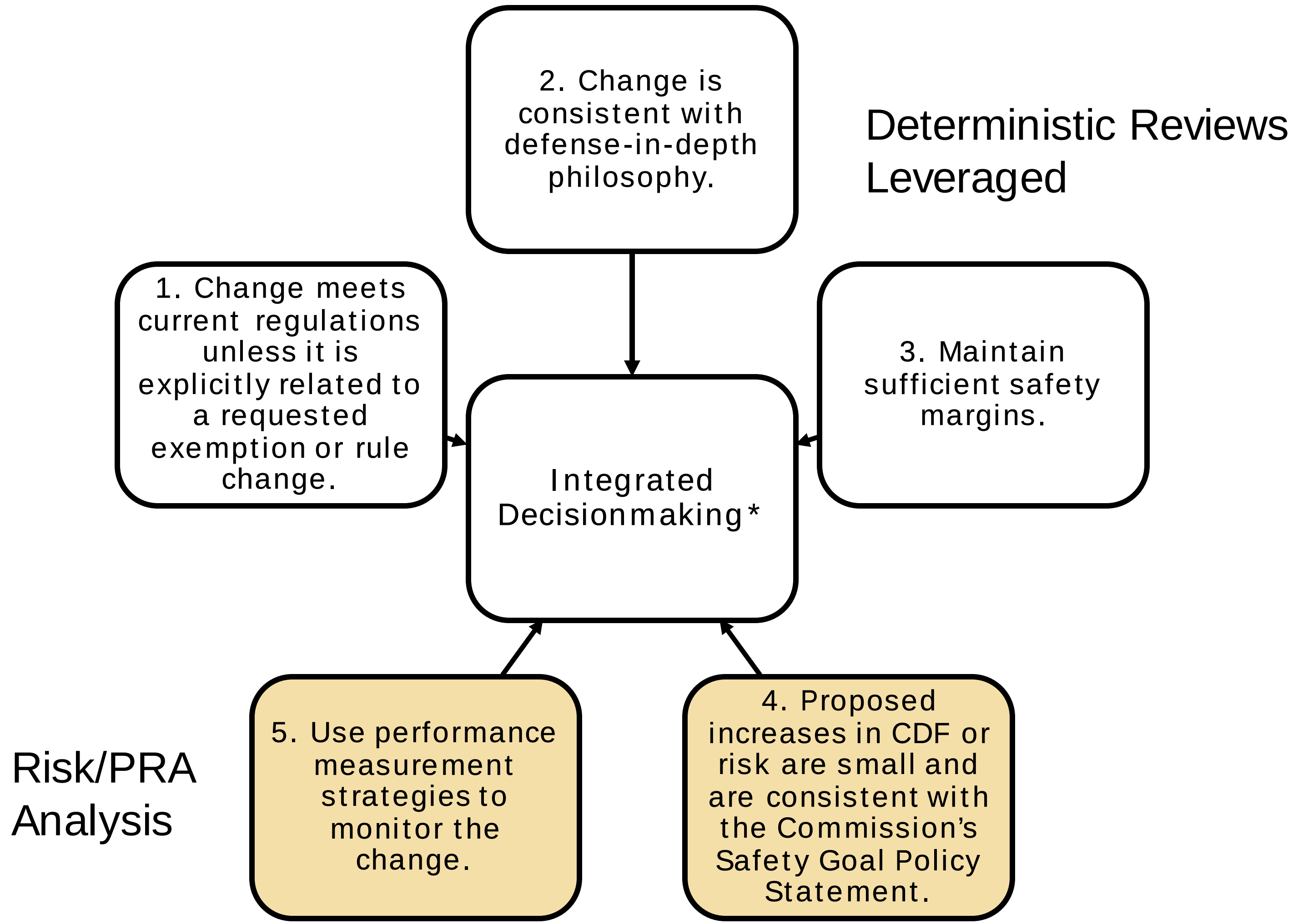
Staff Review of STP GSI-191 LAR

Principle 4: Risk
Principle 5: Performance Monitoring

CJ Fong, PE, Team Leader
Candace Pfefferkorn de Messieres, PhD, Reliability and Risk Analyst

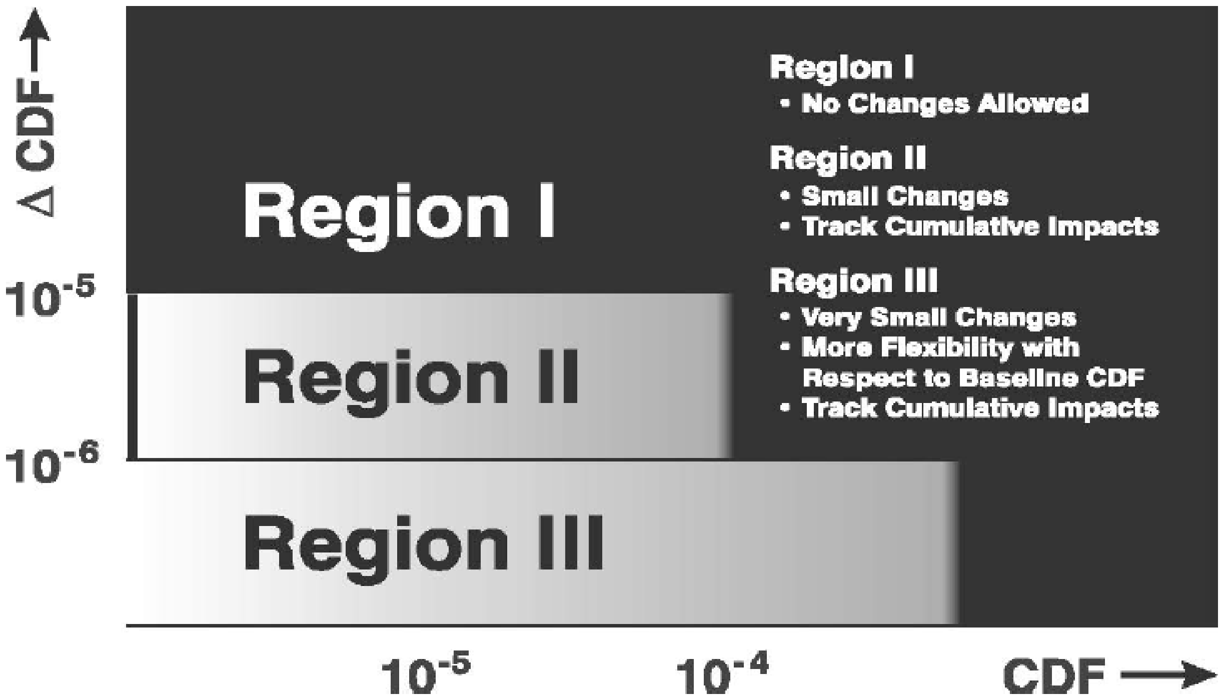
Office of Nuclear Reactor Regulation
Division of Risk Assessment
Risk Informed Licensing

Integrated Decisionmaking



* Principles of Risk-informed Integrated Decisionmaking from Regulatory Guide 1.174, Rev. 2, "An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-specific Change to the Licensing Basis" (ADAMS ML100910006).

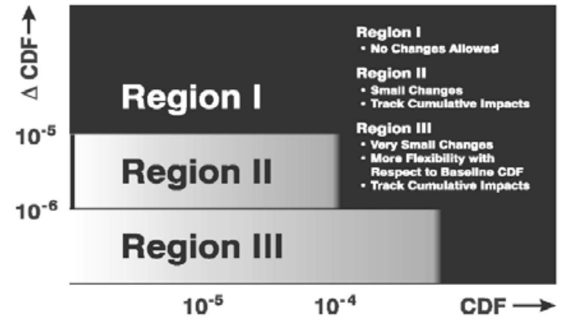
Staff relied on existing framework in RG 1.174



$$\Delta CDF = CDF_{\text{with debris}} - CDF_{\text{no debris}}$$

Major areas reviewed by the staff

- Was risk attributable to debris (ΔCDF, ΔLERF) calculated in an acceptable manner?
  - Initiating Event Frequencies
  - Plant configurations (pump combinations)
  - Break selection
  - Scenario development
  - …
  - Sensitivity and Uncertainty Analyses



- Is the base PRA model acceptable?
  - Scope
  - Level of detail
  - Technical adequacy

The STPNOC Systematic Risk Assessment Key Assumptions

1. Considered both the geometric and arithmetic mean aggregation schemes
2. LOCA frequency allocated to various break locations according only to break size (e.g. "Top down")
3. Considered complete vs. partial breaks.
   - In the "continuum break" assumption a complete break of a given size in one pipe is equally as likely as a partial break of the same size in a larger pipe.
   - In the "DEGB only" assumption, only complete, DEGBs were evaluated.

Staff Performed a Bounding Calculation to Evaluate all Key Assumptions

- Staff applied the conservative, upper bound approach presented to the ACRS during discussions on draft RG 1.229*

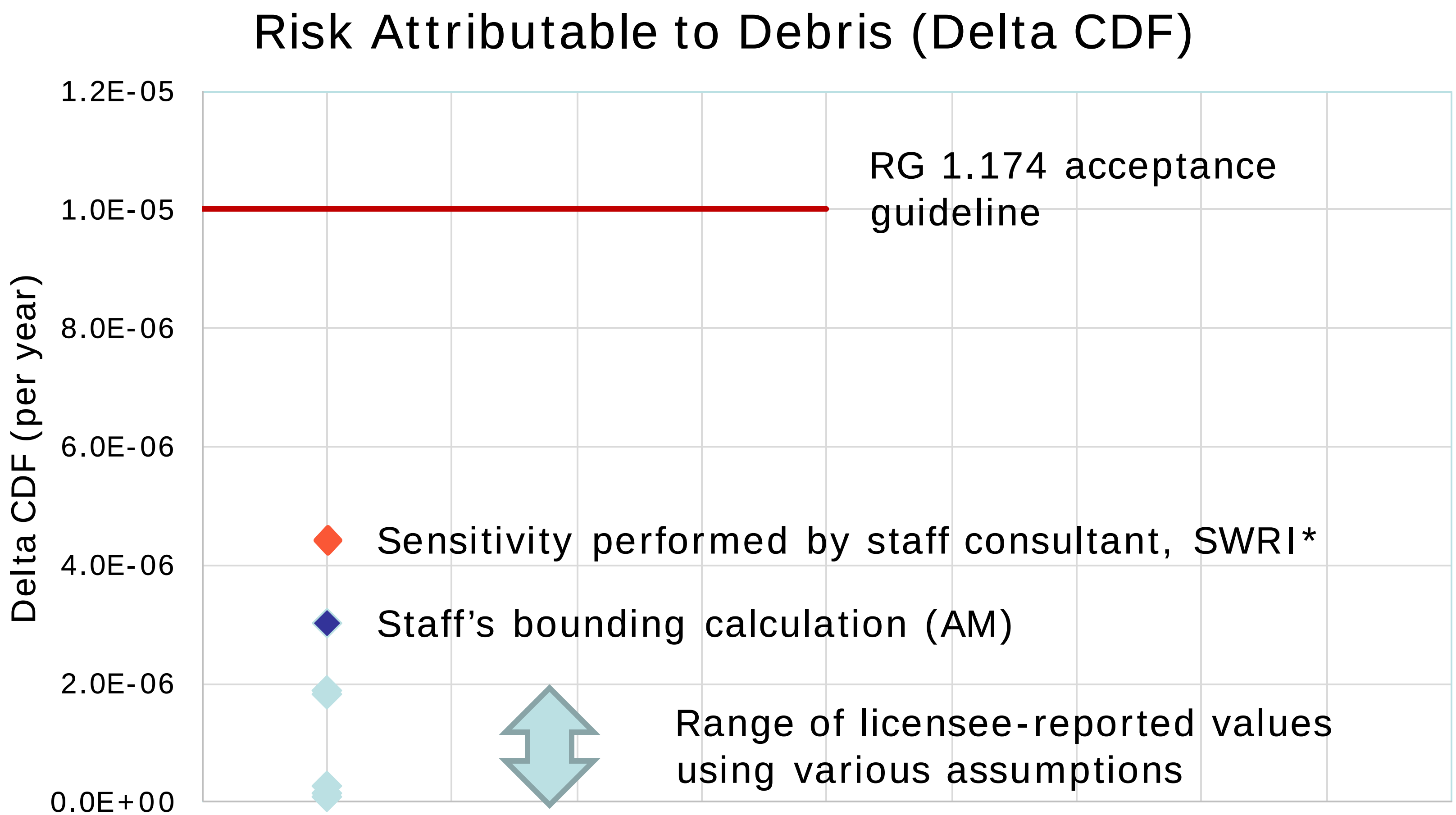
$$\Delta CDF_{debris} = f(x_{min})$$

$$x_{min} = \text{smallest critical break size}$$

$$f(x_{min}) = \text{exceedance frequency using arithmetic mean}$$

* ACRS Meeting April 7, 2016 (ADAMS ML16110A150)

Staff Explored Various Models and Assumptions when Evaluating Risk



**assumes core damage for all breaks ≥ 2 inches for one train case*

Principle 4
Summary of Key Criteria

- The licensee PRA is of the appropriate scope, level of detail, and technical adequacy.
- The risk-informed approach used by the licensee to address the effects of debris on long-term core cooling is consistent with approved practices.
- The increase in risk meets the risk acceptance guidelines as defined RG 1.174.

Principle 5
Performance Monitoring

- Risk analysis reviewed/updated every 48 months
- Procedures/controls have been developed to prevent/mitigate debris in containment (e.g. new TS and programs)
- NRC is notified if acceptance guidelines exceeded
- STP licensing basis (UFSAR) will specify key methods and assumptions that impact results

Principles 4 and 5 Summary

- STPNOC appropriately identified the scenarios that contribute to the increase in risk due to debris (ΔCDF_{debris}, $\Delta LERF_{debris}$)
- There is a lack of consensus for some assumptions in STPNOC's risk calculations
- Bounding calculation addresses lack of consensus and provides confidence that risk is within acceptance guidelines
- Performance monitoring approach is consistent with NRC guidance

ACRS Subcommittee Topics

- Containment Spray System flow rate for single train operation and resulting net positive suction head for sump pump
- Reason for delta CDF decrease for continuum break model when critical break size decreased
- Expected primary pressure transitioning from Mode 3 to Mode 4 and required number of ECCS trains

Overall Summary

- STP acceptably evaluated the impact of debris
- STP appropriately considered both risk and deterministic aspects in the submittal
- Most break scenarios are addressed using conservative deterministic methods
- STP's LTCC evaluation method and simulations are conservative and meet acceptance criteria
- STP's debris analyses meet the key principles of risk-informed regulation
- STP's PRA results show that the change in risk is very small

Questions?

A Probabilistic Risk Assessment of Consequential SGTR (C-SGTR) for a Westinghouse and a Combustion Engineering Plants

With Thermally-Treated Alloy 600 and 690 Steam Generator Tubes

U.S. NRC/RES, IESS presentation to ACRS

May 4, 2017

Purpose and Background

• NRR User Need Request "Developing Analytical Bases and Guidance for Future Risk Assessments of Consequential Steam Generator Tube Rupture (C-SGTR) Events" issued December 2009

  - Requested development of improved analytical bases and guidance for probabilistic risk assessments of C-SGTR events

• Subsequent to an April 2011 ACRS sub-committee briefing, NRR Management requested RES to restructure project to focus on near-term deliverables and to allow for an incremental approach

- Informal meetings with lead ACRS member for C-SGTR issues (Dr. Rempe) held January 2012, January 2013, and April 2013

Purpose and Background - 2

- ACRS full-committee meeting in May 2013
- Staff prepared a draft NUREG-2195
- ACRS Sub-committee briefing in April 2015
- Since the last meeting:
  - ACRS member comments reviewed and addressed (ML16315A250)
  - Draft NUREG-2195 processed and issued for public comment (ML16134A029) – May 2016
  - Public comments reviewed and addressed (ML16315A251)
  - NUREG-2195 revised (ML16315A253)

Recent Work and Path Forward

- ACRS Subcommitte meeting held on December 2016
- ACRS member comments were addressed and NUREG was further revised (ML17082A324)

- Next actions in the project are
  - Have the draft NUREG 2195 go through NRC technical editing process
  - Send the edited version to NRC publishing
- Expect to publish NUREG 2195 in the calendar year 2017

Outline of today's presentation

- Today's presentation focuses on current status and Thermal Hydraulic aspects of the C-SGTR project
- Presentation contains 3 sections:
  - Current status of C-SGTR Project
  - Thermal Hydraulic Overview of C-SGTR
  - Overview and proposed resolution of comments on thermal hydraulic work

C-SGTR Project

Pilot Risk Assessment of Consequential SG tube rupture (Pressure Induced/Creep Rupture) for a Westinghouse and a CE plant consisting of three elements

- Deterministic based Element
  - TH evaluation (MELCOR/RELAP) – informed by CFD
  - Finite element Analysis (Abaqus)
- Performance based Element
  - Failure probabilities (Calculator)
  - Flaw Characteristics/Statistics
- Risk-Informed Element
  - Simplified CDF
  - Conservative LERF

C-SGTR Project (2)

- Involved work scope by 3 RES divisions including 4 branches
- T&H and structure/materials related studies were mostly done in-house; PRA work was contracted out
- During its current work period of 7 years, the project competed for resources with other projects, including Fukushima-related ones.

Severe Accident-Induced Steam Generator Tube Rupture (SGTR)

Thermal Hydraulic Overview of CSGTR

Michael Salay
NRC – Office of Nuclear Regulatory Research

Consequential Steam Generator Tube Rupture (C-SGTR) ACRS Briefing

May 4, 2017

Topics

- CSGTR Scenario Description
- TH analyses
- Method (CFD & System Code)
- Experimental Basis
- Differences Between CE and Westinghouse Plants

The Station Blackout

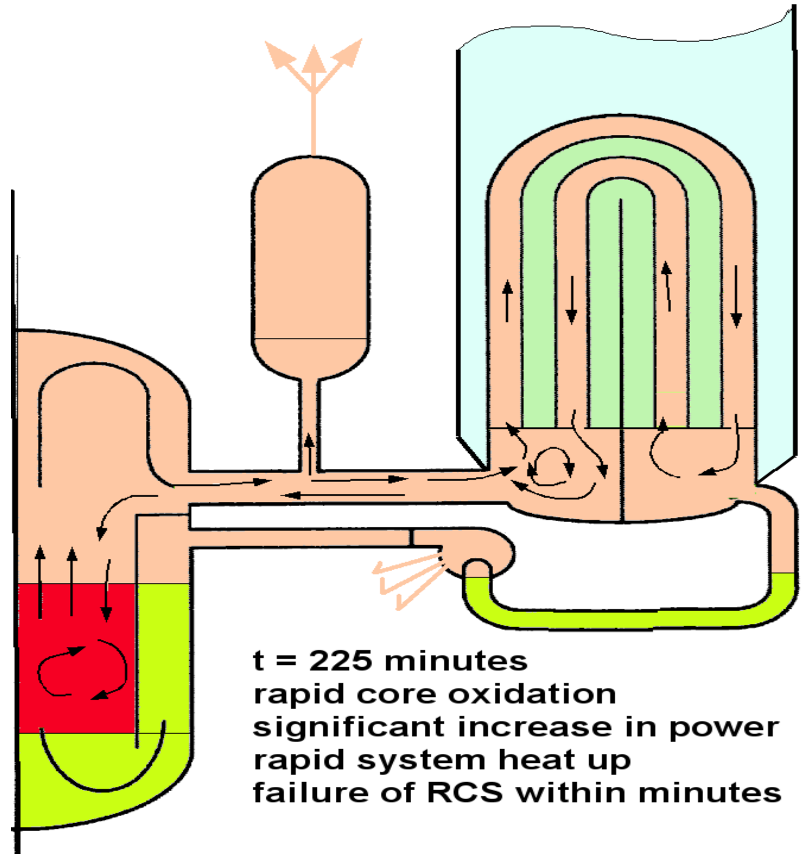
- A low probability station blackout event with immediate or subsequent loss of feed water to the steam generators.
- Reactor inventory boils off resulting in fuel damage and high temperature and high pressure conditions within RCS.
- Failure of the RCS boundary is induced by these conditions.

- If SG tubes fail first, then a flow path is created that bypasses the containment
- Failures of other RCS components (hot leg or surge line), RCS blow down into the containment
- Determining SG tubes failure is important in consequence analysis

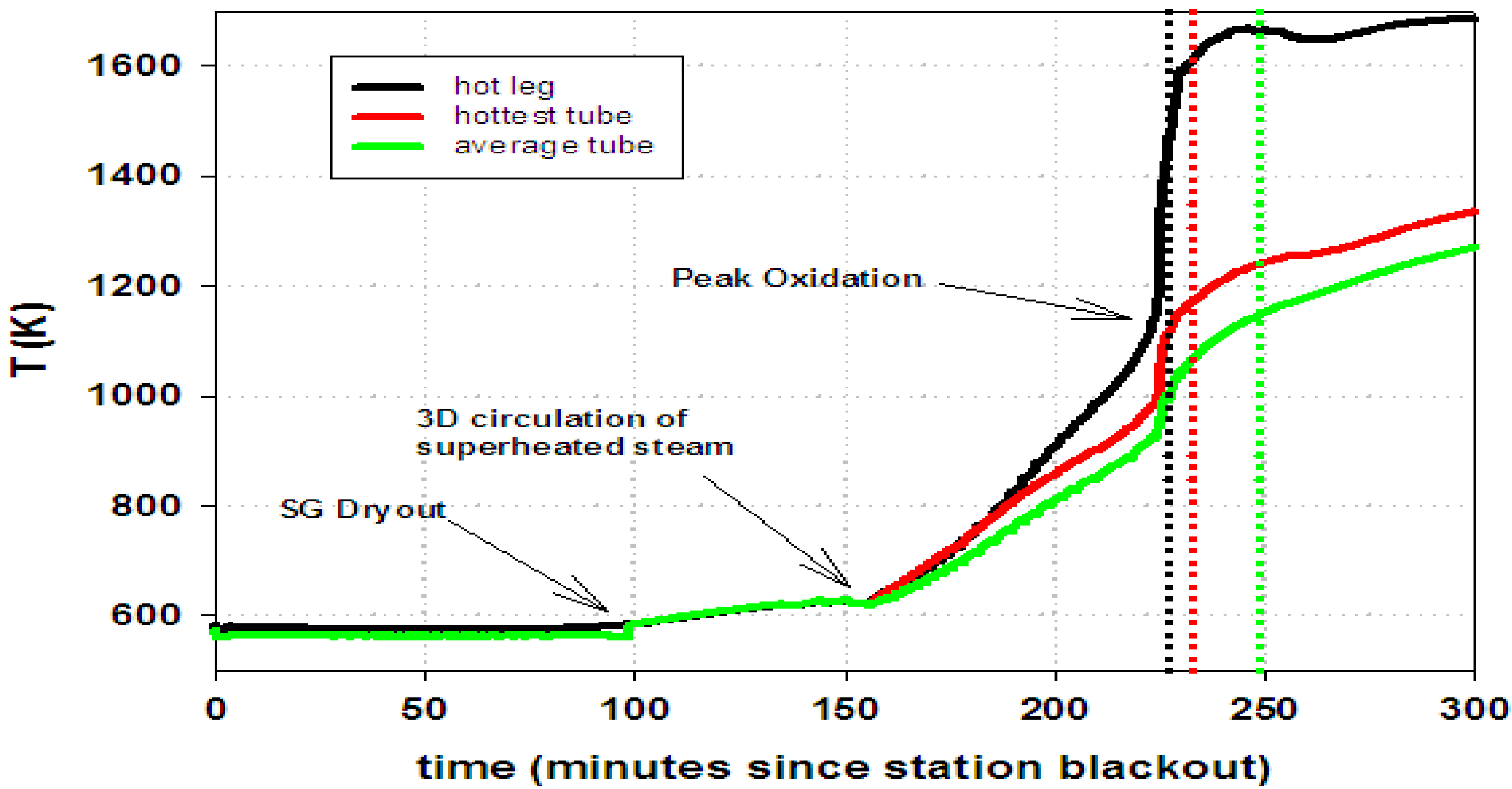
A Fast Westinghouse Scenario RCS failure within 4 hours

- loss of offsite power, failure of diesels, and failure of auxiliary feedwater systems
- primary inventory lost through reactor coolant pump seals. Secondary side boils off
- secondary side dry, primary inventory lost through safety valve cycling and pump seals
- loop natural circulation stops as primary inventory falls in SG tubes.
- natural circulation of superheated steam begins as inventory falls below hot leg. Core and system heat up.
- Core uncovers, core oxidizes and produces significant power, system heat up accelerates and induced failure is predicted for RCS components.

- More likely scenarios involve some auxiliary feedwater or operator actions that significantly delay the failure time.



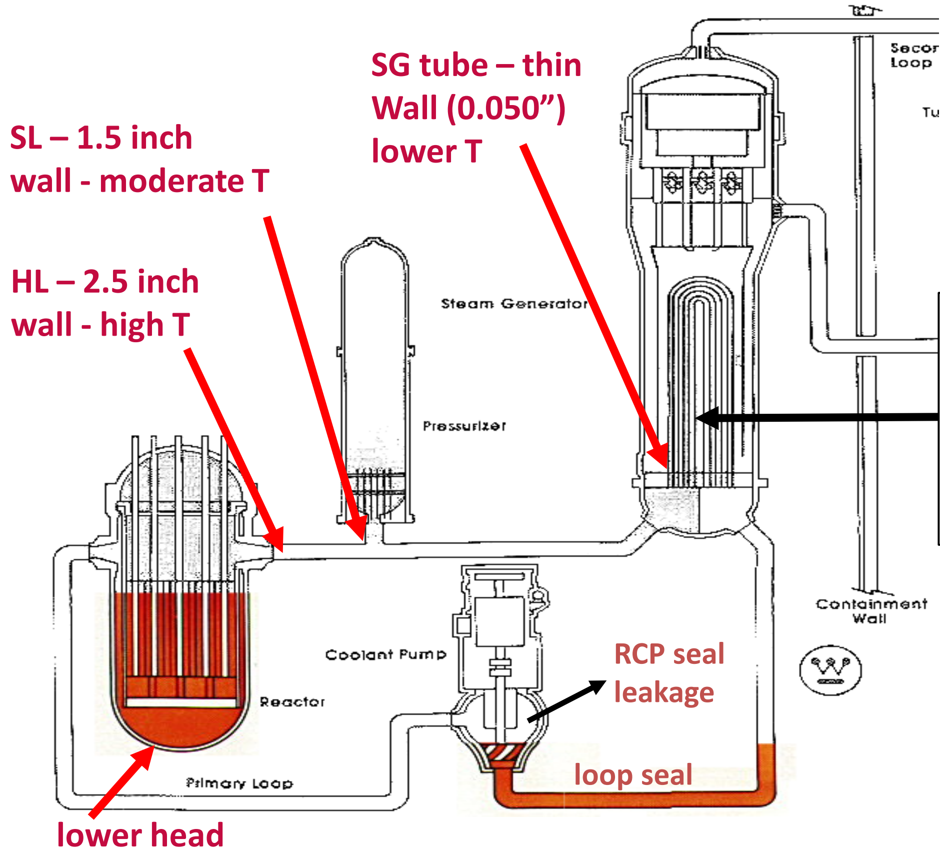
RCS Structure Temperatures – Fast Westinghouse Scenario



RCS Points of interest and modeling considerations

Different oxidation and melting temperatures

Wall thickness indicative of thermal response times



SG tube ruptures provide a path for fission products to bypass containment.

Rapid temperature rise and pressure difference leads to induced failure.

- **failure location affects consequences**

RCS Points of interest and modeling considerations

- Pressurizer draining and surge line orientation
- Primary relief valve behavior

- Shell heat loss
- SG depressurization

Different oxidation and melting temperatures

Wall thickness indicative of thermal response times

SL – 1.5 inch wall - moderate T

HL – 2.5 inch wall - high T

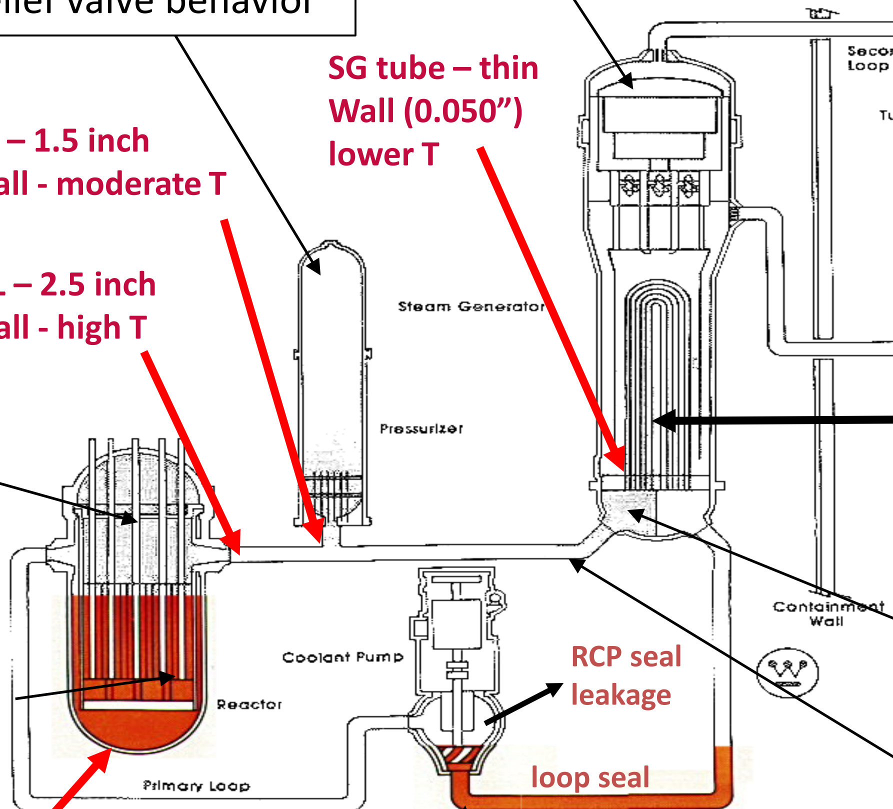
SG tube – thin Wall (0.050”) lower T

- Tube heat transfer
- Secondary flows
- Tube mass flow fraction
- Leakage
- Tube plugging

SG tube ruptures provide a path for fission products to bypass containment.

Natural recirc core bypass flow

- Core oxidation rate
- Core blockage
- Nodalization
- Natural circulation
- Instrument tube failure



lower head

- Inlet plenum mixing
- Recirculation ratio
- Plume T distribution

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- Radiation modeling
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Rapid temperature rise and pressure difference leads to induced failure.

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High-Dry-Low

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High Pressure

*** no significant leakage to reduce pressure**

Secondary Side

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*** Loss of water allows tubes to heat up**

Low Pressure
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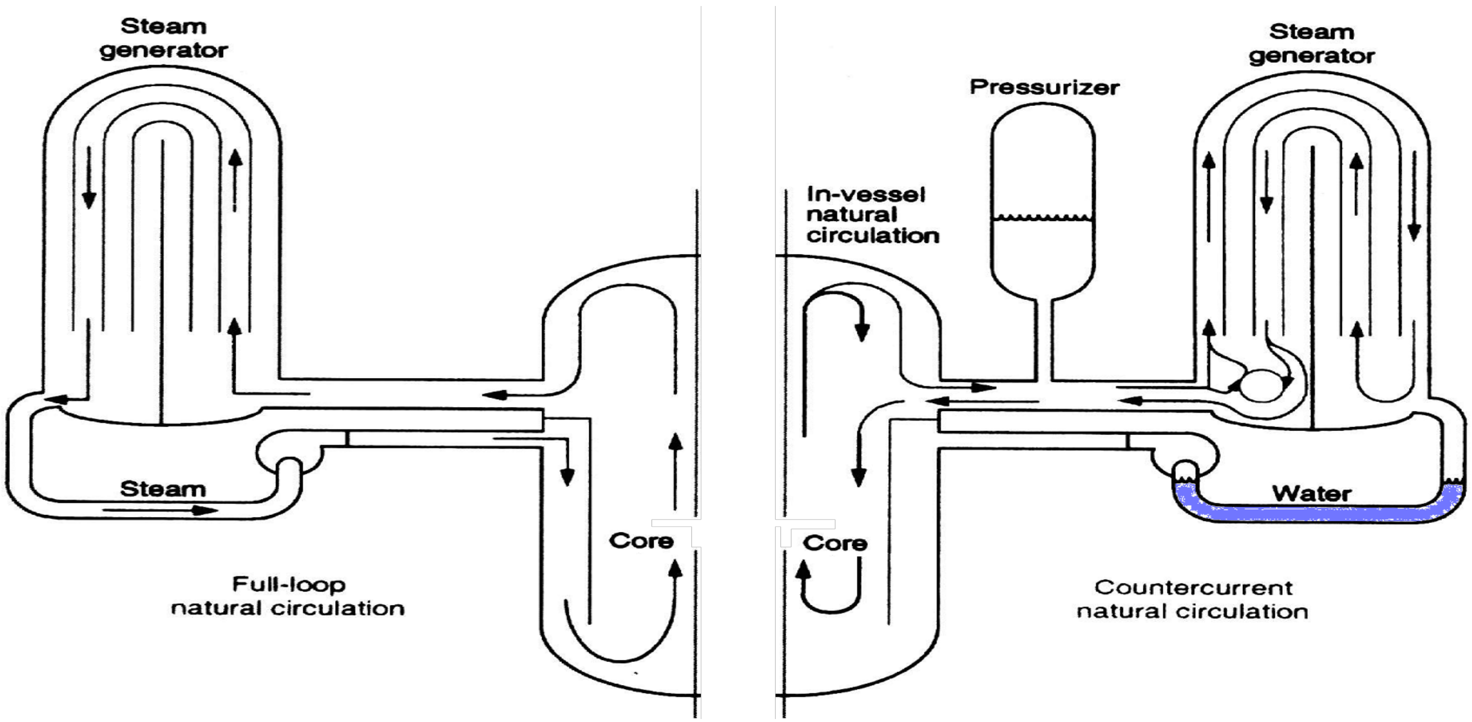
SG tube wall

Two Flow Patterns - PWRs with U-Tube SGs



full-loop natural circulation **Counter-current natural circulation**

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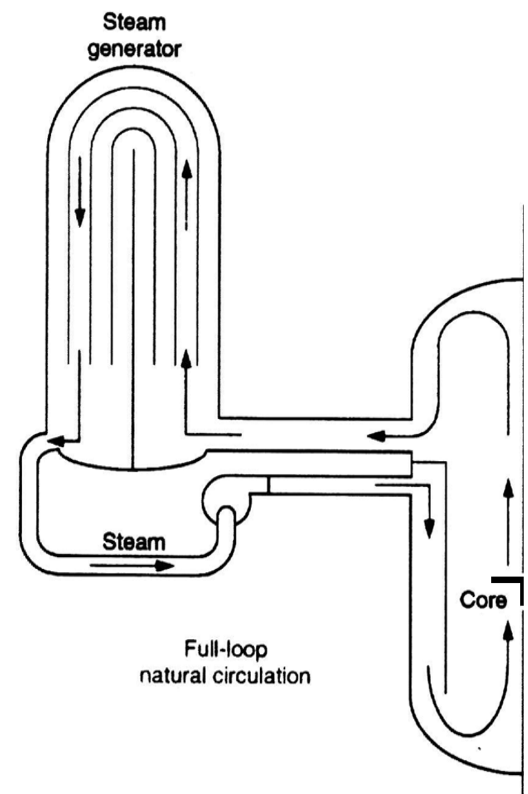

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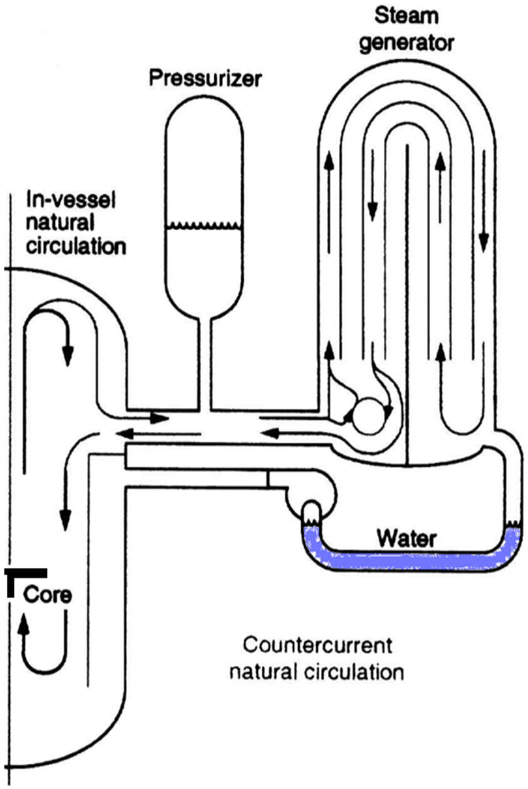

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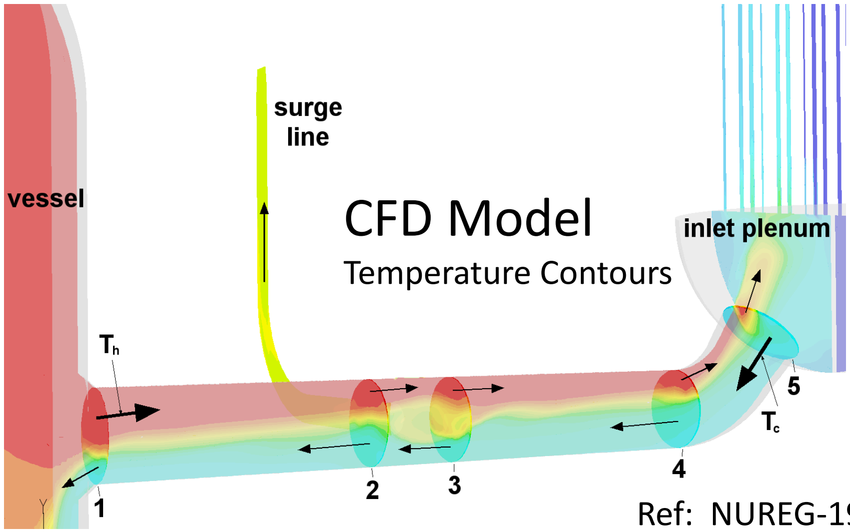

Ref: NUREG-1922

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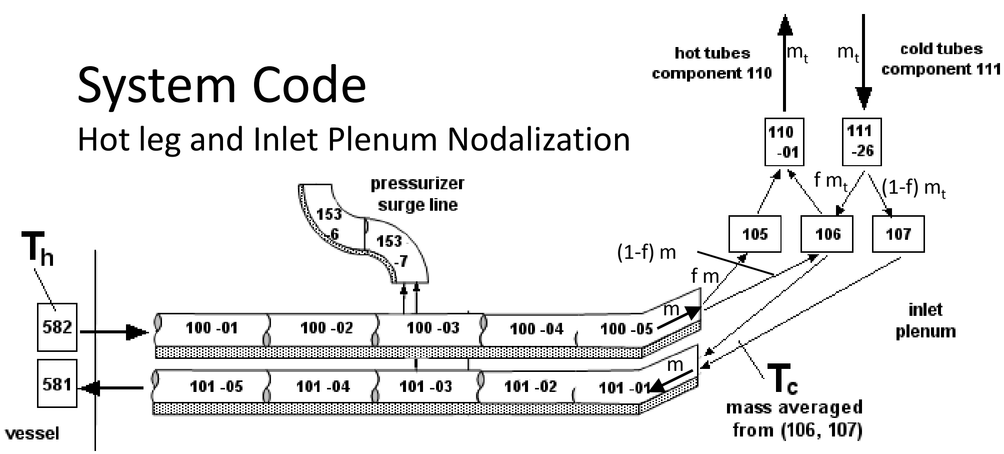

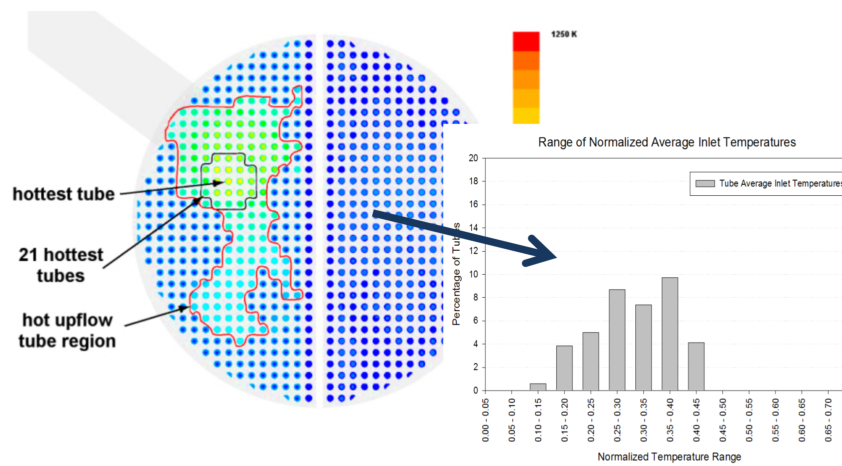

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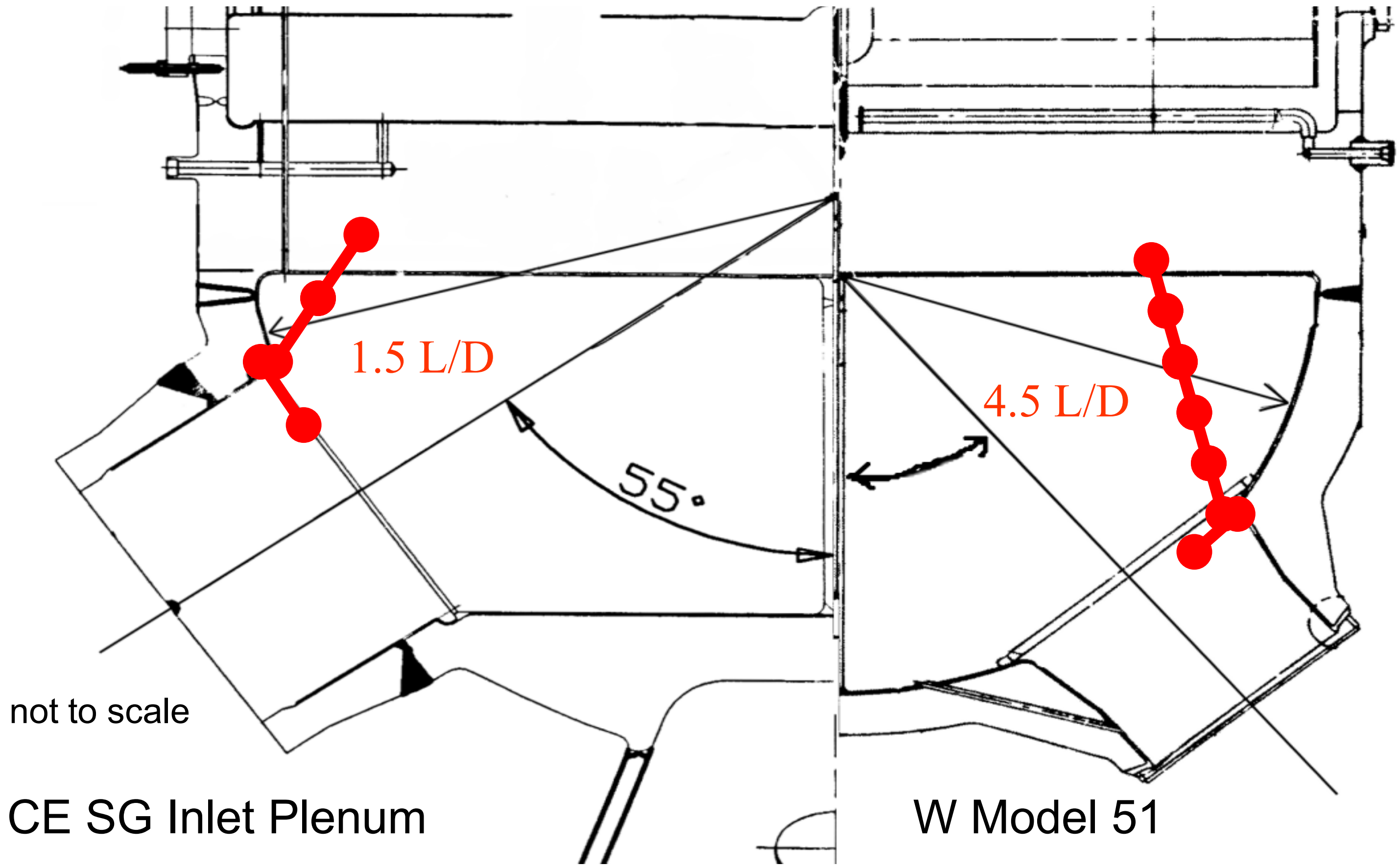
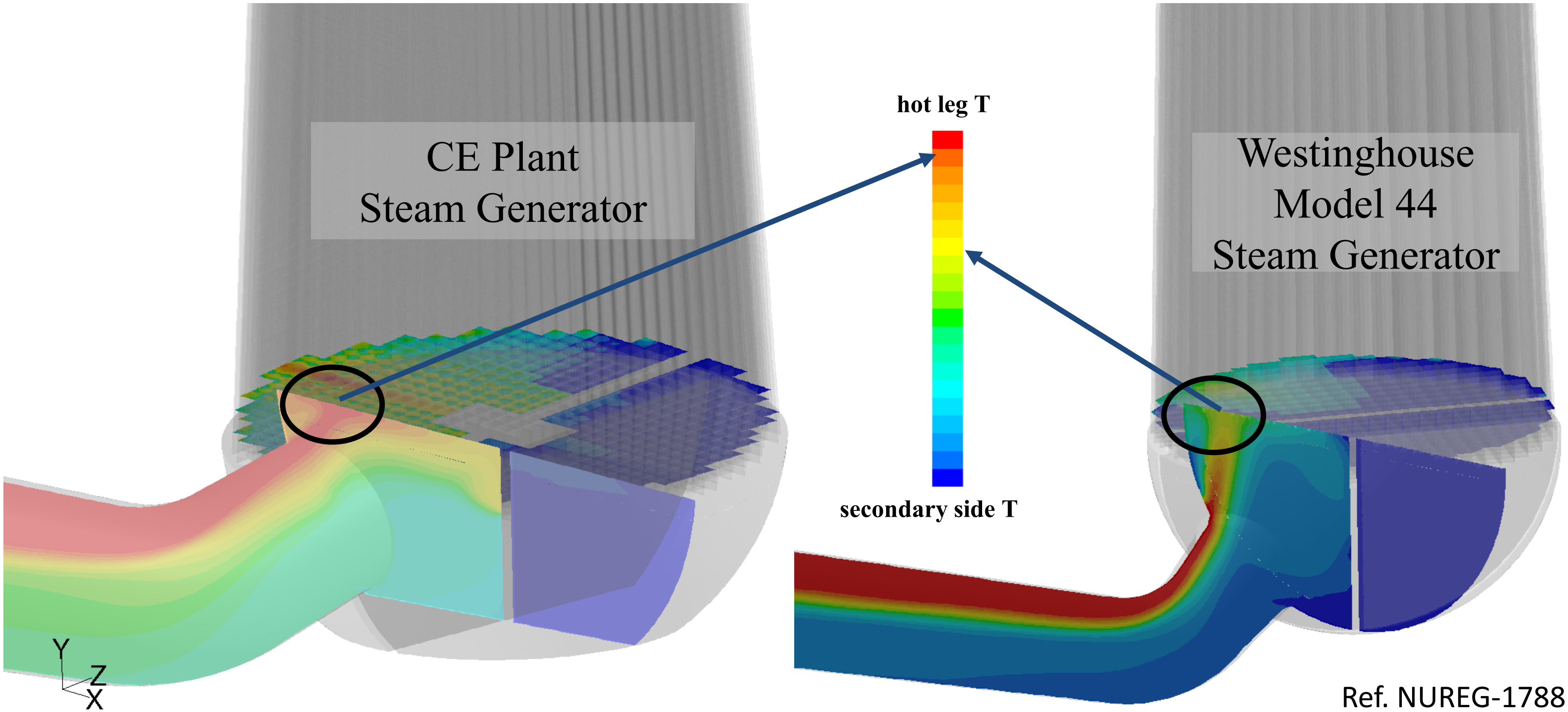


CFD Predictions - Westinghouse and CE (hottest tube region circled)



Ref. NUREG-1788

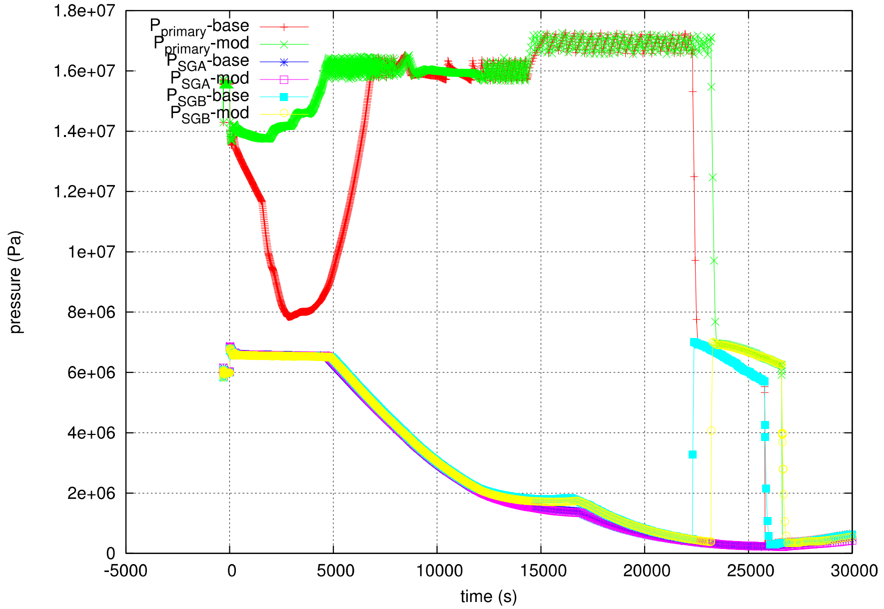
(temperature contours on vertical centerline plane of hot leg)

MELCOR CE Calculations

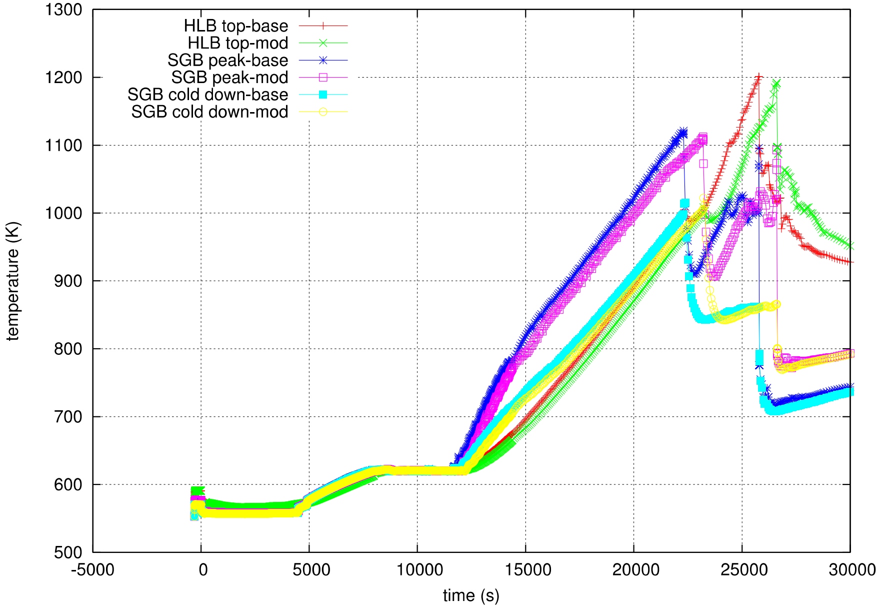
- Objectives:
  - Provide TH results for CE plants to be used to calculate component failure using:
    - CSGTR Calculator
    - Finite Element calculations
  - Provide scoping component failure calculations
  - Provide FP releases
- Analyses (scoping and component-failure-suppressed calculations)
  - Accidents
    - Short terms station blackout (stsbo) (AFW fails to start)
    - Long term station blackout (ltsbo)
  - Variations
    - Operator actions/relief valve behavior
    - RCP seal leakage
- Comparison against RELAP5 CE calculation
- Uncertainty analysis to determine impact of TH uncertainty on failure timing

Example MELCOR CE results

Impact of RCP seal leakage sensitivity



Effect on system pressures



Effect on Loop B structure temperatures

Possible future CE TH work

- Interesting but deferred work because of resource limitations
  - More detailed spatial temperature distribution
  - Assessment of TH factors that impact relative component failure timing
  - Analysis of loop seal clearing
  - Water hold up in SG, flooding / counter-current flow
    - Water also held up in previous SGAP calculations
  - Detailed evaluation of FP release
    - Current focus on TH input, not FP release
      - Didn't rerun cases to solely extract FP release behavior

Severe Accident-Induced Steam Generator Tube Rupture (SGTR)

Overview and Proposed Resolution of Comments on TH Work

Michael Salay
NRC – Office of Nuclear Regulatory Research

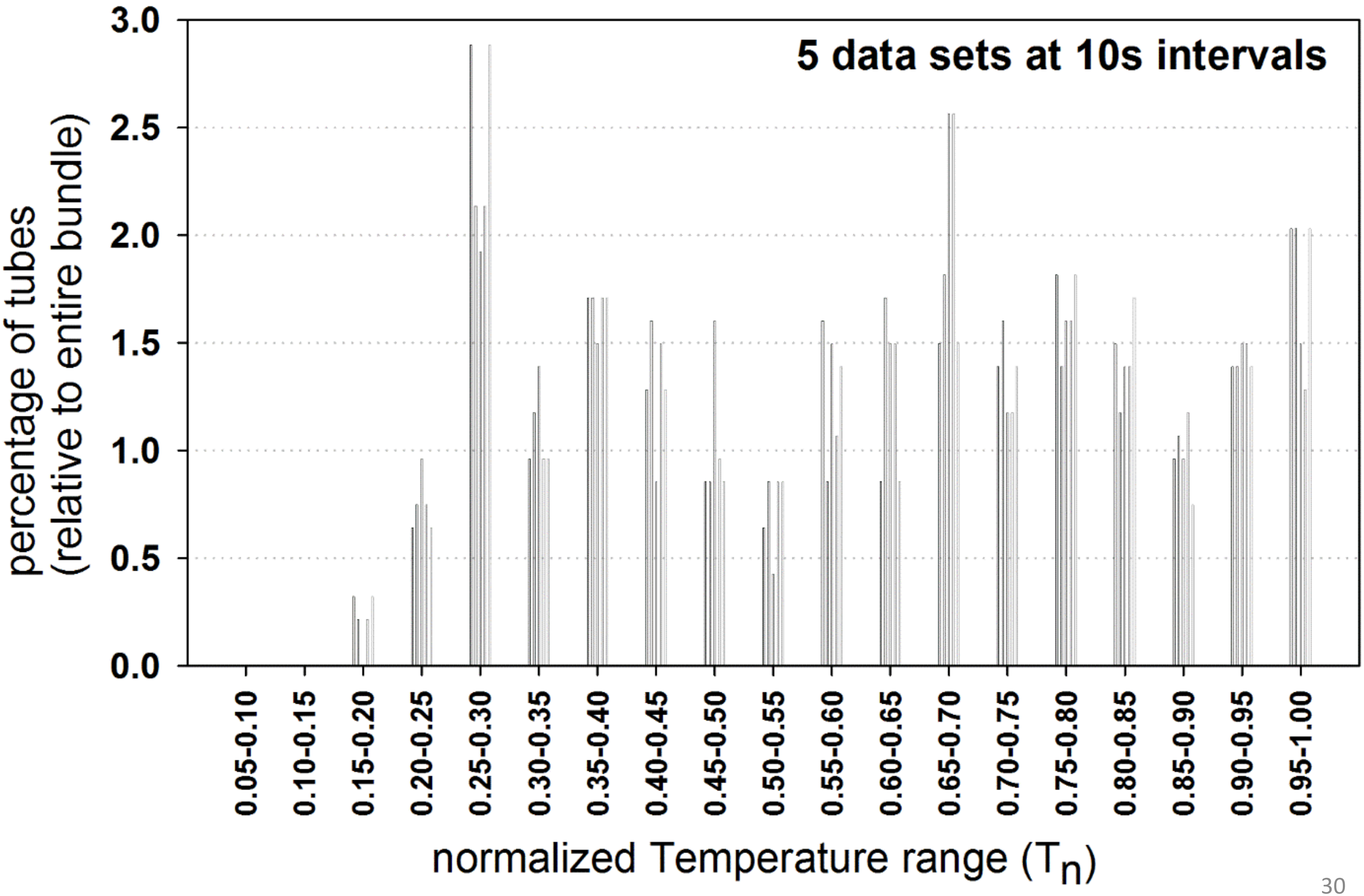
Consequential Steam Generator Tube Rupture (C-SGTR) ACRS Briefing

May 24, 2017

Recent ACRS questions

- CE tubesheet inlet T distributions
- Impact on Loop seal clearing
- Expected impact of models in later versions of MELCOR
- Uncertainties in TH analyses

CE tubesheet inlet T distribution



Loop seal clearing (1/2)

- Q: Many studies out there conclude that loops seals would clear before core damage. Do any of scenarios indicate that?
- Loop seal clearing was studied extensively for Westinghouse for SGAP and several mechanisms studied and documented in NUREG/CR-6995
  - Generated parameter maps for conditions under which loop seals would clear or stay intact
    - f(RCP pump seal leak rate, TDAFW operation time, number of PORV opened, time of operator action)
    - Found to also be affected by core bypass area and nodalization
  - Concluded that uncertainty remains regarding whether loop seals would clear.

Loop seal clearing (2/2)

- Loop seals did not clear in any of the MELCOR CE calculations
- Initial scoping work for loop seal clearing CE built upon the SGAP analyses
- Analysis of loop seal clearing eliminated upon work scope reduction - Issue not explored fully for CE
  - Loop seal clearing is more important for Westinghouse plants because this clearing exposes SG tubes to gases nearly as hot as those in the hot leg
  - For the CE geometry studied, the hottest gases entering SG tube bundle are nearly as hot as those in the hot leg
    - Loop seal clearing not as important

Impact of Updated MELCOR Models (1/3)

- Q: MELCOR models used in the SOARCA analysis resulted in significant changes in timing of events. Do these changes affect the conclusions of this study?
- Most TH analysis work done primarily in 2011 and 2012. Some cases recalculated in 2013.
- MELCOR 1.8.6 used for CE CSGTR analysis
- Different MELCOR 2 versions used in different SOARCA analyses

Impact of Updated MELCOR Models (2/3)

- SOARCA Uncertainty Analysis for Surry short term station blackout compared event timing of MELCOR 1.8.6 to MELCOR 2.1
- Major event timing very close up to and beyond component creep failure
- No significant change expected to CE MELCOR calculations models

Event times for Surry STSBO (hh:mm)

Event	1.86	2.1
SBO	00:00	00:00
SG dryout	01:16	01:14
Start of fuel heatup	02:19	02:20
RCP seal failure	02:45	02:47
First FP gap releases	02:57	02:57
HL creep rupture failure	03:45	03:45
Accumulators start	03:45	03:46

Impact of Updated MELCOR Models (3/3)

- There have been phenomenon model updates in MELCOR subsequent to Surry analysis:
  - Update to Lipinski debris bed dryout model
  - Update to quenching models
- Both of these models affect behavior during reflood
- The CE analysis concerns system behavior at high pressure before accumulator injection
- The use of the updated models are not expected to alter report conclusions

CE TH Uncertainty Analysis (1/6)

- Additional detail on MELCOR uncertainty analysis requested
- The impact of uncertainties in TH considered upon initial deck creation: “How much do uncertainties in TH affect failure timing and releases?”
- SNL performed TH uncertainty analysis on early stsbo model with deck prior to addition of hot tube and other modifications
  - Used Average Hot tube for SG with stress multiplier of 2
  - Expect results to be representative of failure timing variation resulting from TH variations
  - NOT necessarily expected to be representative of component or relative failure time
- Sampled TH parameters and observed effect on predicted absolute component failure timing and relative SG-tube-to-RCS-component failure timing
  - “simple” Monte Carlo sampling
  - 100 realizations
  - Analyzed with STEPWISE regression software
- TH uncertainty analysis parameters chosen based on those in NUREG/CR-6285 and NUREG/CR-6995:

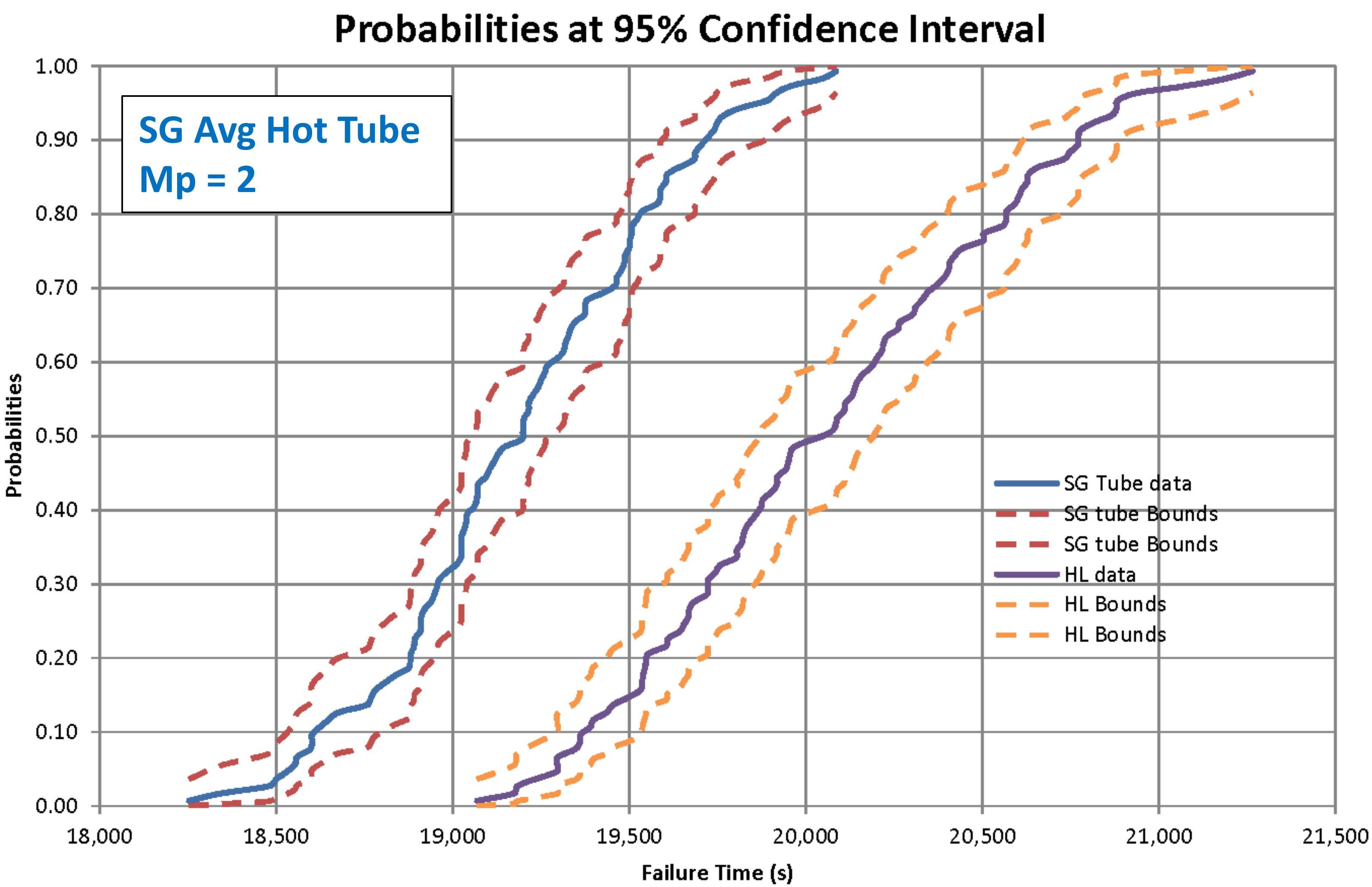
CE TH Uncertainty Analysis (2/6)

Sampled Parameters

Parameter	Range
PORV and SRV Valve discharge coefficients	0.7 - 1.1
Zr oxidation rate constant	14.8 -44.4
Mixing parameters	
Cd	0.064 - 0.0863
Recirculation ratio	1.1 - 1.4
SG tube outer wall heat transfer multiplier	0.5 - 1.5
Hot leg wall emissivity	0.3 – 0.9
RCS to containment heat transfer multiplier	2.8075 – 8.4225

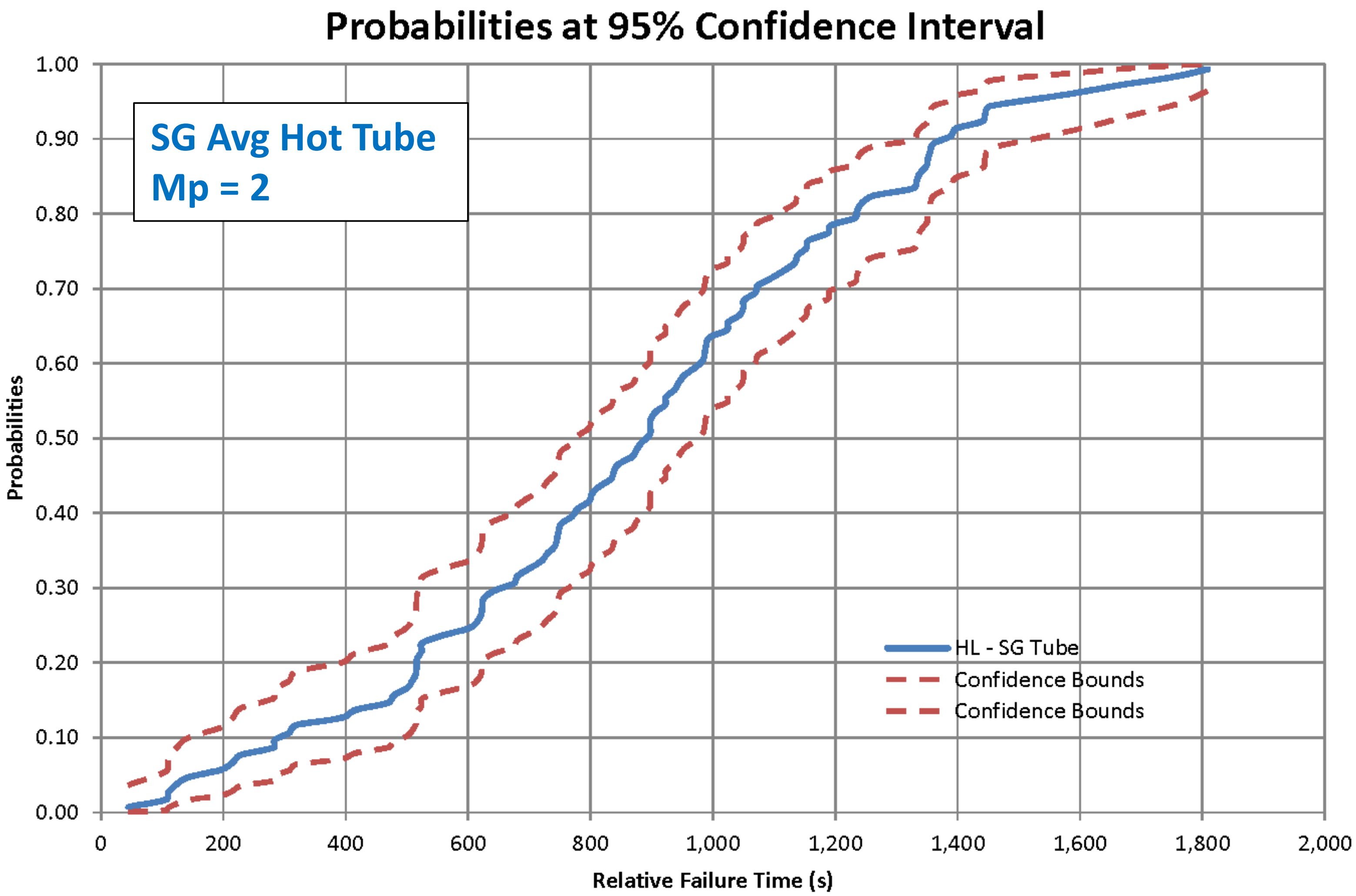
CE TH Uncertainty Analysis (3/6)

Component failure time distributions



CE TH Uncertainty Analysis (4/6)

Relative component failure time distribution



CE TH Uncertainty Analysis (5/6)

What affected absolute failure timing?

R^2

Parameter (SG)	SG tube		Parameter (HL)	Hot Leg
RCS to Cont HTC mult	0.585		RCS to Cont HTC mult	0.784
SG wall HTC mult	0.279		Mixing Cd	0.113
Zr oxidation rate	0.06		Zr oxidation rate	0.049
Recirc. ratio	0.023		SG wall HTC mult	0.014
Mixing Cd	0.012		Recirc. ratio	0.01
PORV/SRV discharge	0.002			

(Higher numbers indicate greater importance)

What affected relative failure timing?

CE TH uncertainty Analysis (6/6)

- Distribution of failure timings resulting from TH variation uncertainty analysis had standard deviations of:
  - $\pm$400 s (6 min 40 s) – SG absolute failure timing
  - $\pm$511 s (8.5 min) – HL absolute failure timing
  - Approximately $\pm$420 s (7 min) – relative SG-to-RCS component failure timing

CE TH Conclusions

- MELCOR calculations for a CE plant with replacement SGs provide input to CSGTR calculator and finite-element component failure analysis
- Relative temperature increase rates and relative component failure timing between SG tubes and other components more important for releases than absolute failure time
- Some work was deferred because of limited resources
  - Benefit determined to not be worth the expense for the project
- Received and incorporated useful feedback from ACRS and public

A Probabilistic Risk Assessment of Consequential SGTR (C-SGTR) for a Westinghouse and a Combustion Engineering Plants

With Thermally-Treated Alloy 600 and 690 Steam Generator Tubes

U.S. NRC/RES, IESS presentation to ACRS

May 4, 2017

Purpose and Background

- NRR User Need Request "Developing Analytical Bases and Guidance for Future Risk Assessments of Consequential Steam Generator Tube Rupture (C-SGTR) Events" issued December 2009
    - Requested development of improved analytical bases and guidance for probabilistic risk assessments of C-SGTR events
- Subsequent to an April 2011 ACRS sub-committee briefing, NRR Management requested RES to restructure project to focus on near-term deliverables and to allow for an incremental approach
- Informal meetings with lead ACRS member for C-SGTR issues (Dr. Rempe) held January 2012, January 2013, and April 2013

Purpose and Background - 2

- ACRS full-committee meeting in May 2013
- Staff prepared a draft NUREG-2195
- ACRS Sub-committee briefing in April 2015
- Since the last meeting:
  - ACRS member comments reviewed and addressed (ML16315A250)
  - Draft NUREG-2195 processed and issued for public comment (ML16134A029) – May 2016
  - Public comments reviewed and addressed (ML16315A251)
  - NUREG-2195 revised (ML16315A253)

Recent Work and Path Forward

  - ACRS Subcommitte meeting held on December 2016
  - ACRS member comments were addressed and NUREG was further revised (ML17082A324)
- Next actions in the project are
  - Have the draft NUREG 2195 go through NRC technical editing process
  - Send the edited version to NRC publishing
- Expect to publish NUREG 2195 in the calendar year 2017

Outline of today's presentation

- Today's presentation focuses on current status and Thermal Hydraulic aspects of the C-SGTR project
- Presentation contains 3 sections:
  - Current status of C-SGTR Project
  - Thermal Hydraulic Overview of C-SGTR
  - Overview and proposed resolution of comments on thermal hydraulic work

C-SGTR Project

Pilot Risk Assessment of Consequential SG tube rupture (Pressure Induced/Creep Rupture) for a Westinghouse and a CE plant consisting of three elements

- Deterministic based Element
  - TH evaluation (MELCOR/RELAP) – informed by CFD
  - Finite element Analysis (Abaqus)
- Performance based Element
  - Failure probabilities (Calculator)
  - Flaw Characteristics/Statistics
- Risk-Informed Element
  - Simplified CDF
  - Conservative LERF

C-SGTR Project (2)

- Involved work scope by 3 RES divisions including 4 branches
- T&H and structure/materials related studies were mostly done in-house; PRA work was contracted out
- During its current work period of 7 years, the project competed for resources with other projects, including Fukushima-related ones.

Severe Accident-Induced
Steam Generator Tube Rupture (SGTR)

Thermal Hydraulic Overview of CSGTR

Michael Salay
NRC – Office of Nuclear Regulatory Research

Consequential Steam Generator Tube Rupture (C-SGTR) ACRS Briefing

May 4, 2017

Topics

- CSGTR Scenario Description
- TH analyses
- Method (CFD & System Code)
- Experimental Basis
- Differences Between CE and Westinghouse Plants

The Station Blackout

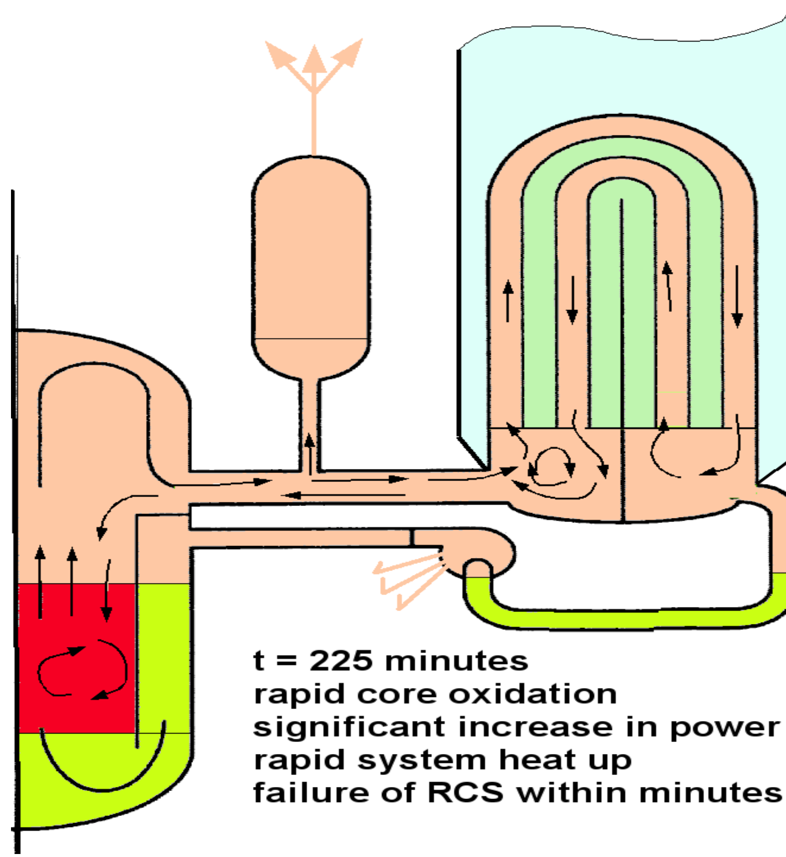
- A low probability station blackout event with immediate or subsequent loss of feed water to the steam generators.
- Reactor inventory boils off resulting in fuel damage and high temperature and high pressure conditions within RCS.
- Failure of the RCS boundary is induced by these conditions.

- If SG tubes fail first, then a flow path is created that bypasses the containment
- Failures of other RCS components (hot leg or surge line), RCS blow down into the containment
- Determining SG tubes failure is important in consequence analysis

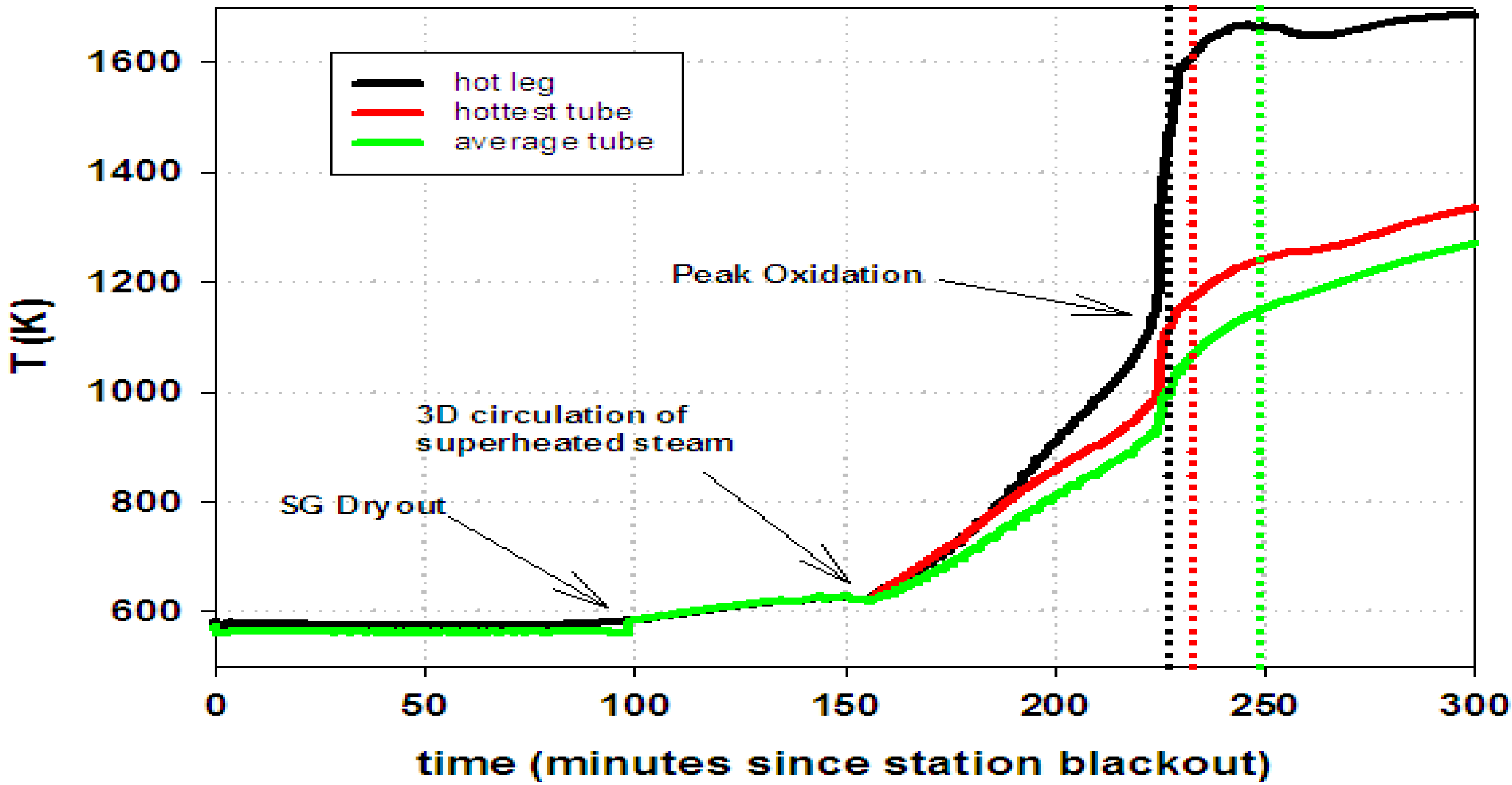
A Fast Westinghouse Scenario RCS failure within 4 hours

- loss of offsite power, failure of diesels, and failure of auxiliary feedwater systems
- primary inventory lost through reactor coolant pump seals. Secondary side boils off
- secondary side dry, primary inventory lost through safety valve cycling and pump seals
- loop natural circulation stops as primary inventory falls in SG tubes.
- natural circulation of superheated steam begins as inventory falls below hot leg. Core and system heat up.
- Core uncovers, core oxidizes and produces significant power, system heat up accelerates and induced failure is predicted for RCS components.

- More likely scenarios involve some auxiliary feedwater or operator actions that significantly delay the failure time.



RCS Structure Temperatures – Fast Westinghouse Scenario



RCS Points of interest and modeling considerations

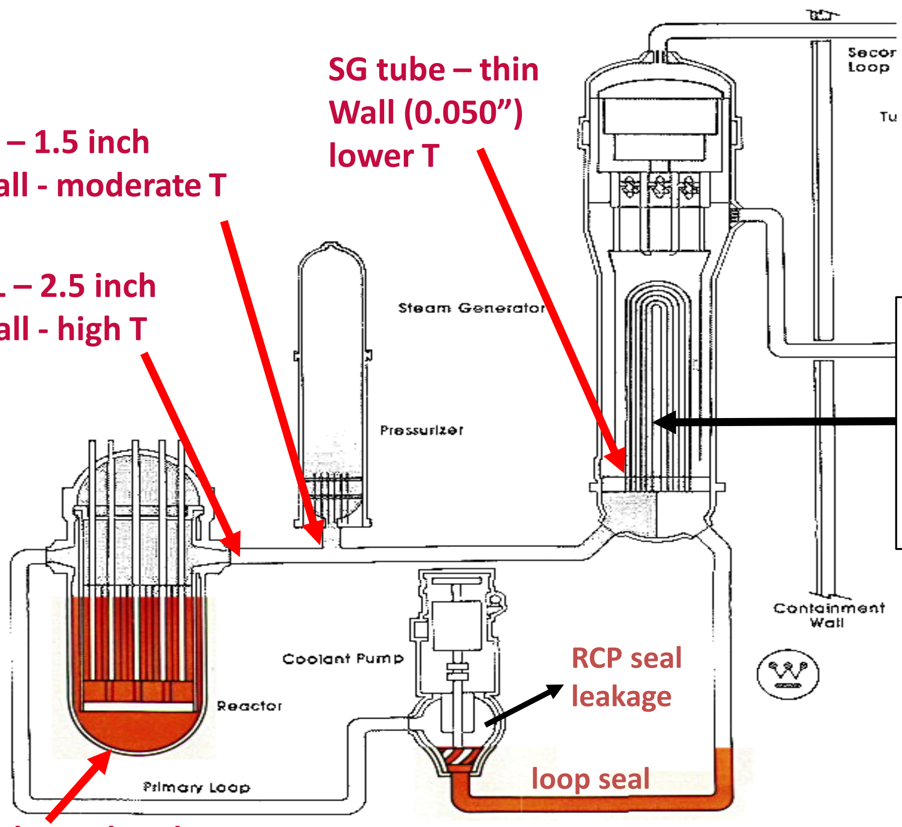
Different oxidation and melting temperatures

Wall thickness indicative of thermal response times

SL – 1.5 inch wall - moderate T

HL – 2.5 inch wall - high T

SG tube – thin Wall (0.050”) lower T



lower head

SG tube ruptures provide a path for fission products to bypass containment.

Rapid temperature rise and pressure difference leads to induced failure.

- **failure location affects consequences**

RCS Points of interest and modeling considerations

- Pressurizer draining and surge line orientation
- Primary relief valve behavior

- Shell heat loss
- SG depressurization

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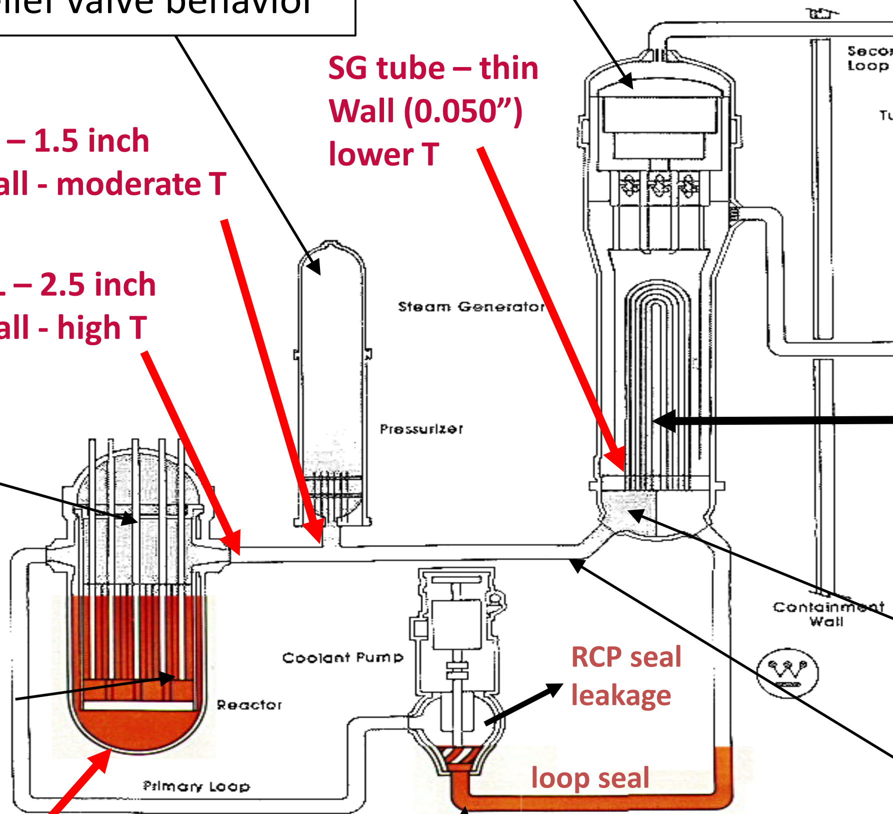

lower head

- Tube heat transfer
- Secondary flows
- Tube mass flow fraction
- Leakage
- Tube plugging

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Natural recirc core bypass flow

- Core oxidation rate
- Core blockage
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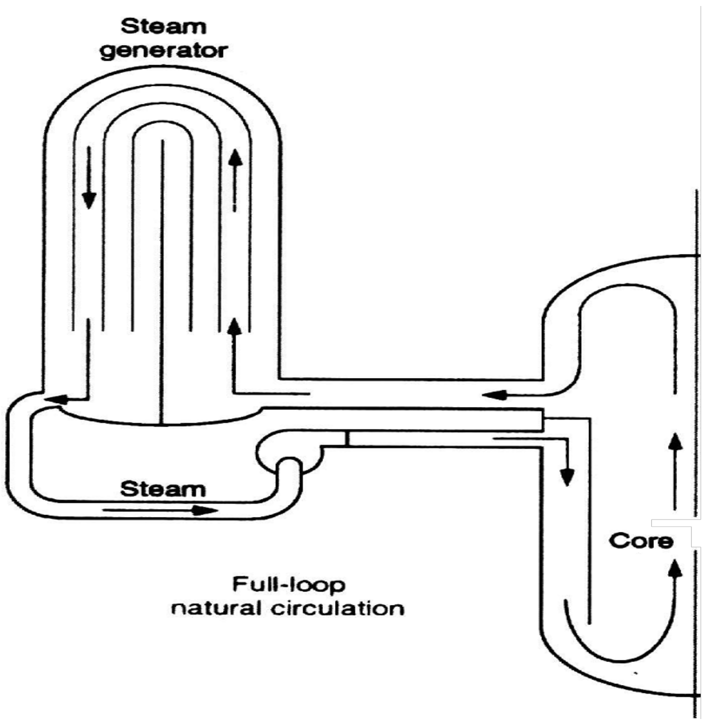

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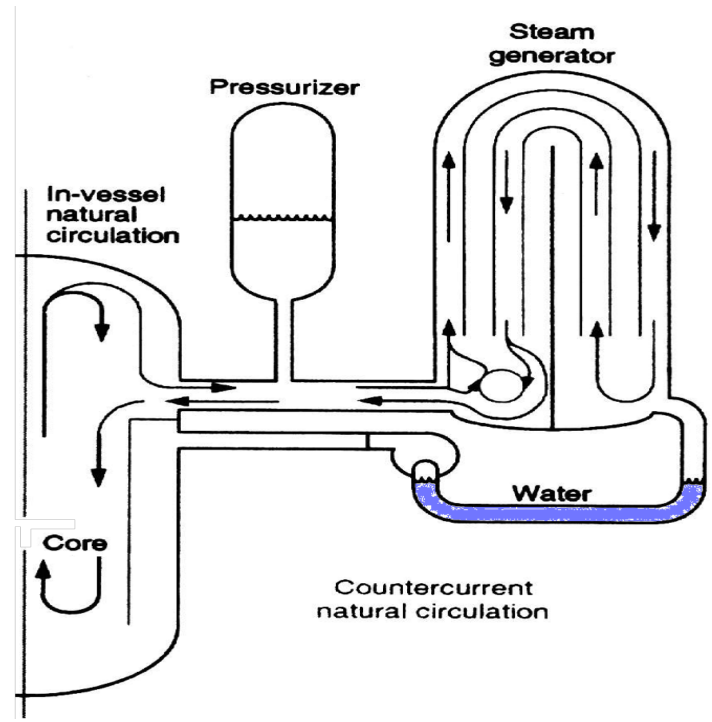

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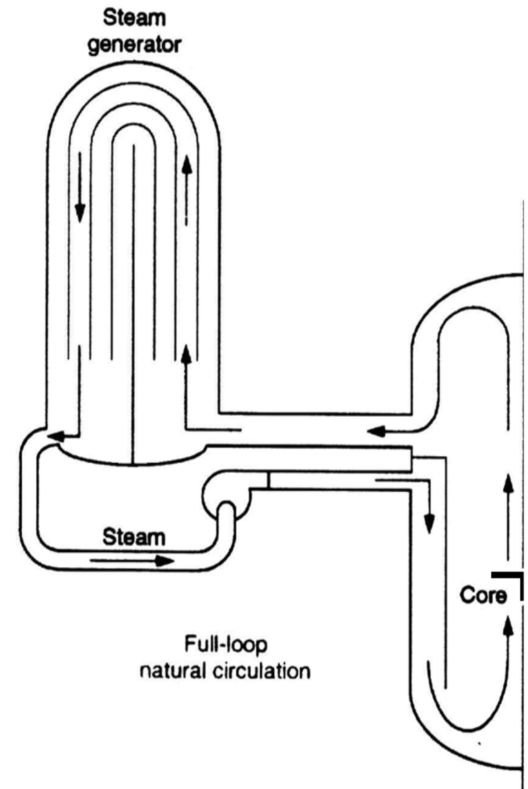

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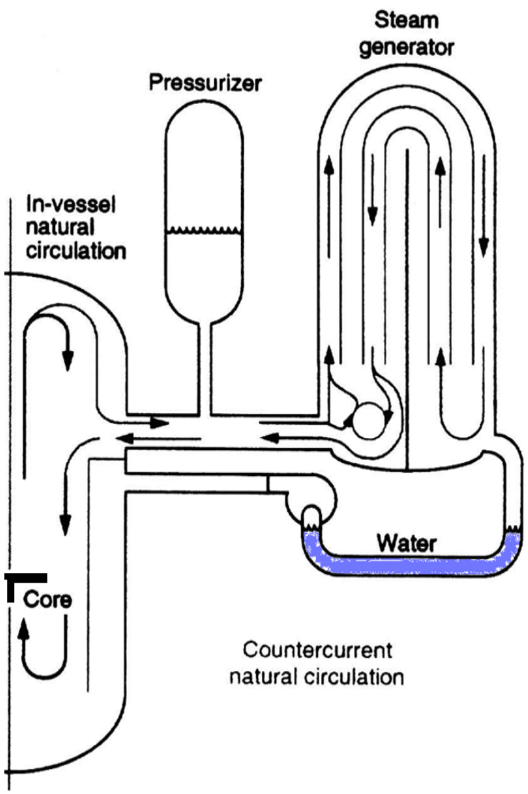

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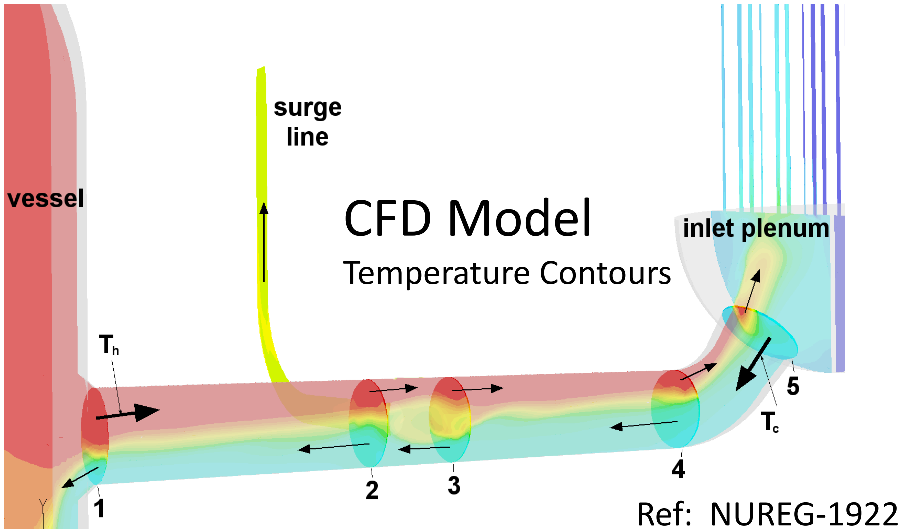

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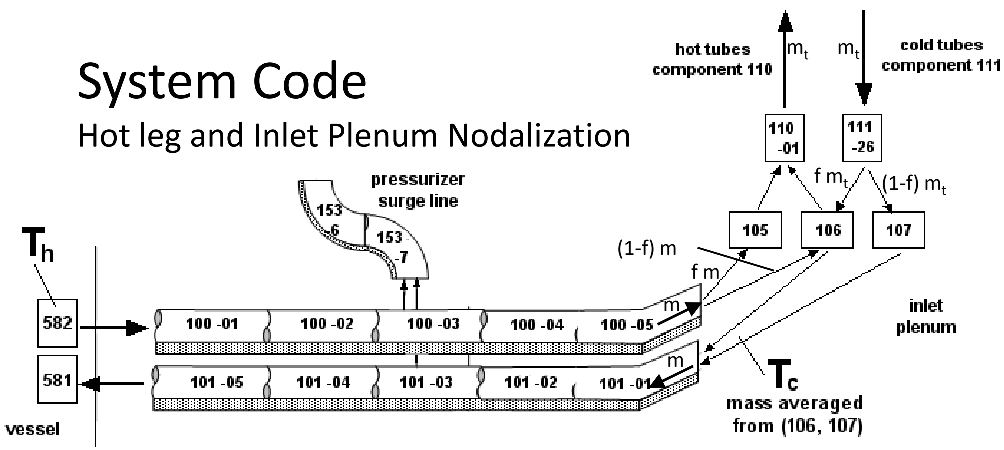

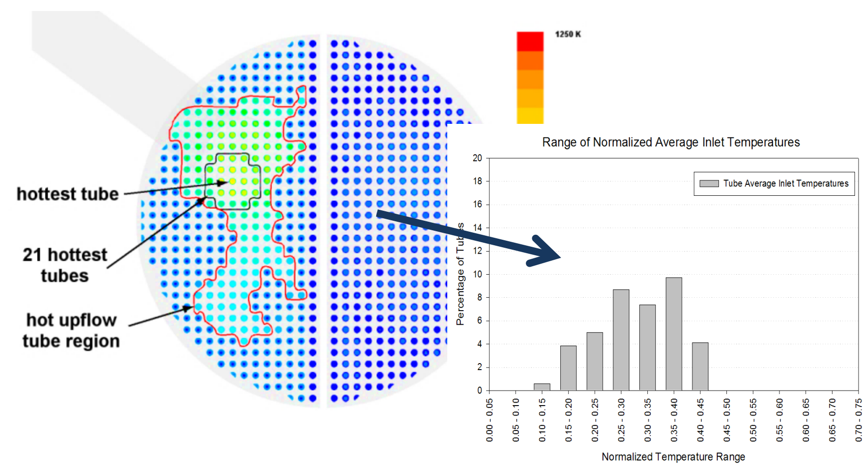

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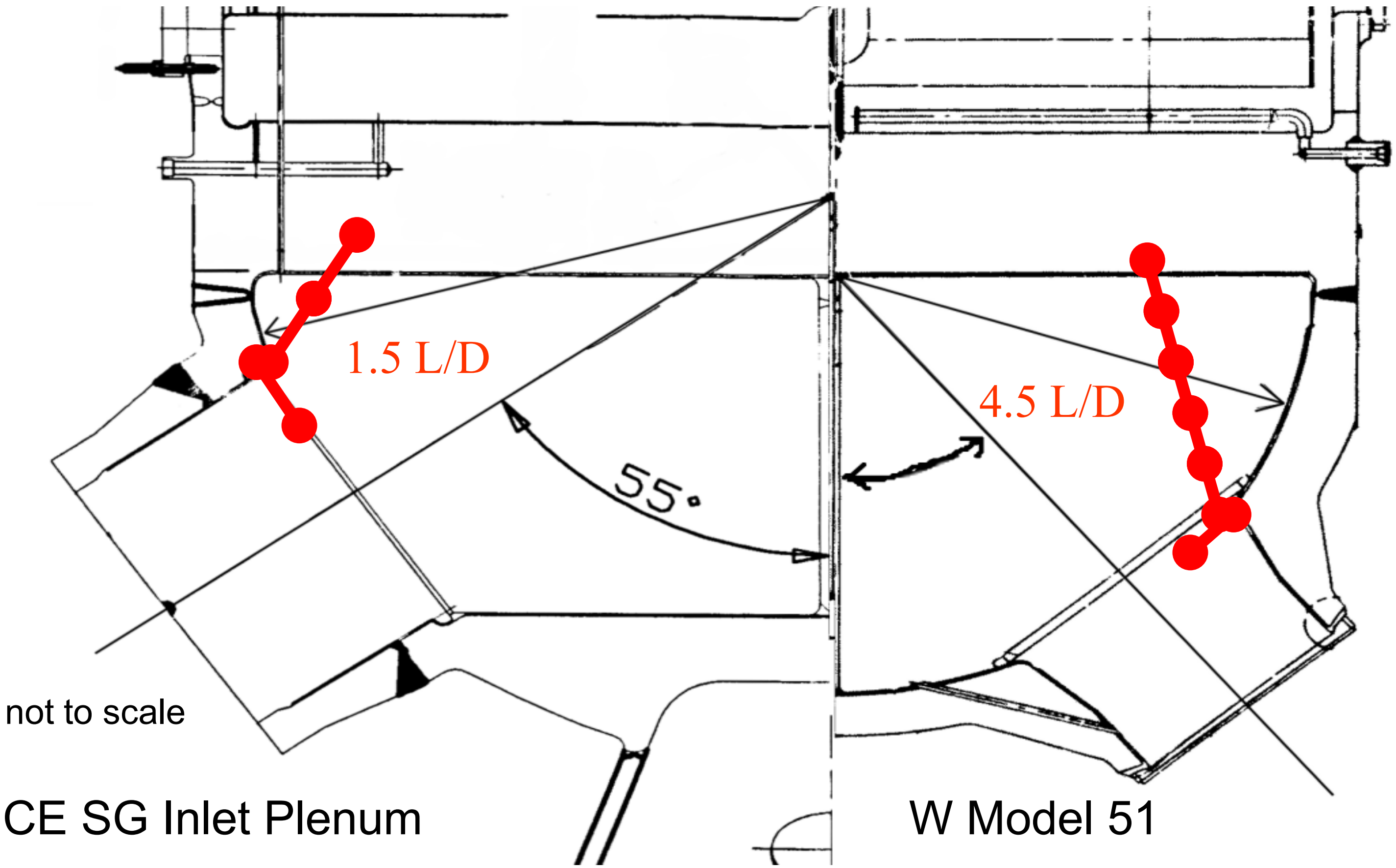
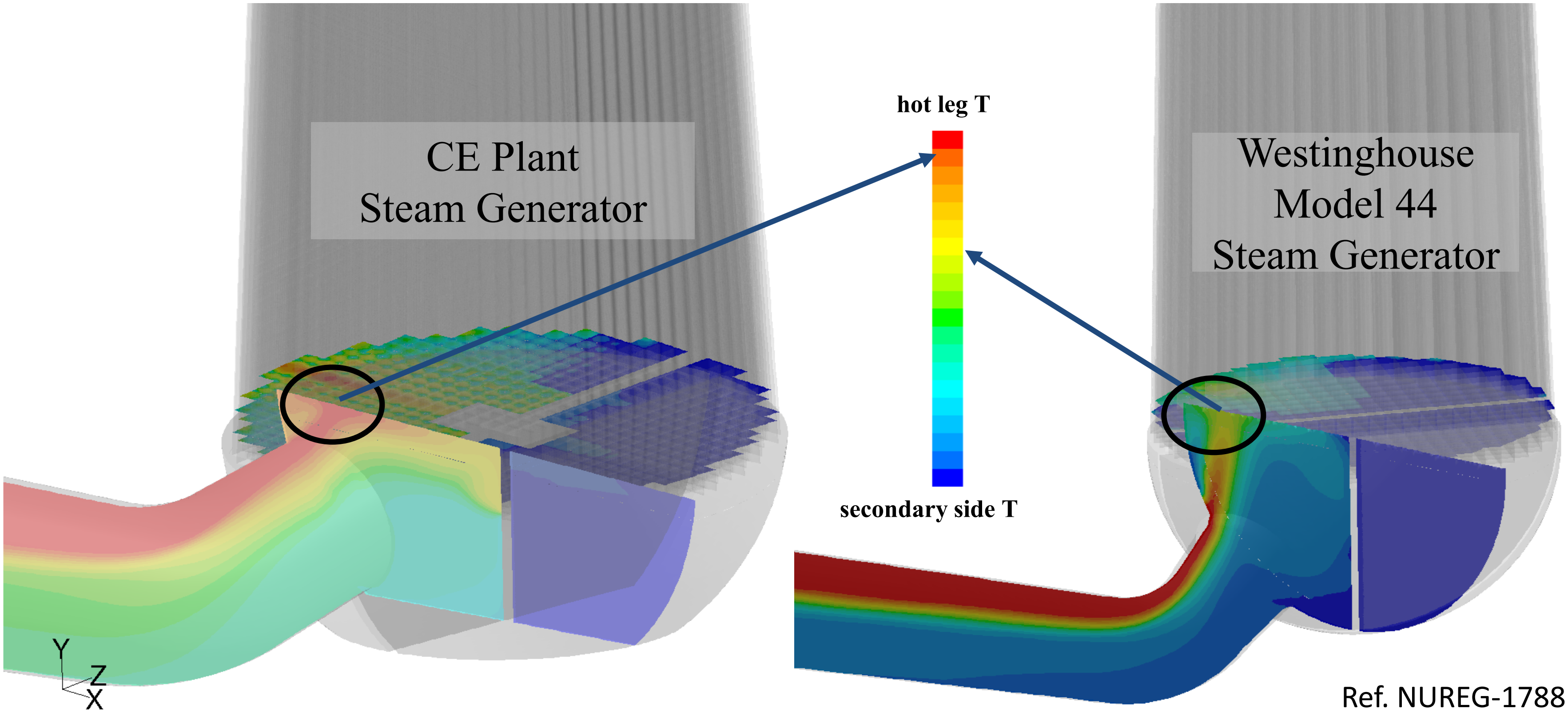
- Unlike for the rupture of a flawed tube, multiple unflawed tubes could potentially reach the failure condition nearly simultaneously resulting in a rupture large enough to depressurize the RCS sufficiently fast to prevent failure of other RCS components.

The CE inlet plenum (compared to W model 51)



CFD Predictions - Westinghouse and CE (hottest tube region circled)



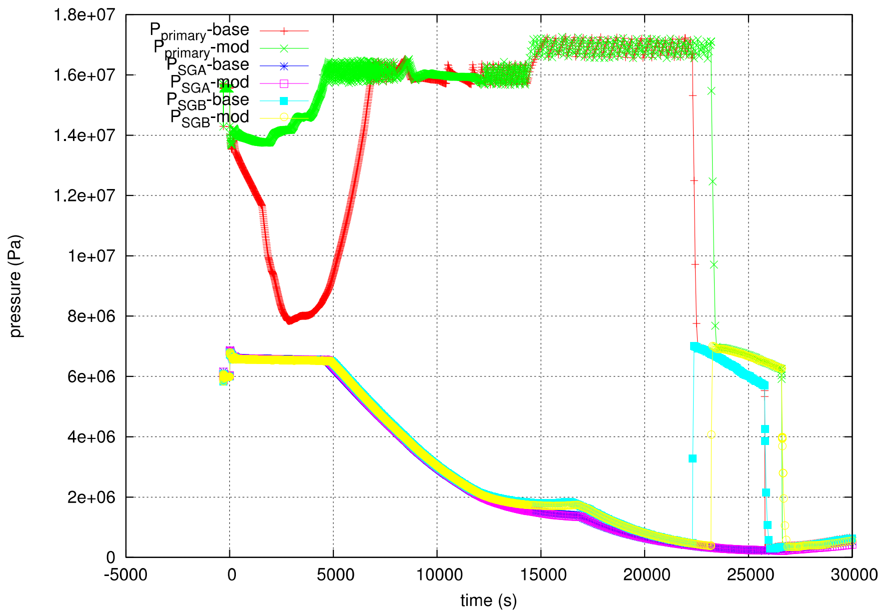
(temperature contours on vertical centerline plane of hot leg)

MELCOR CE Calculations

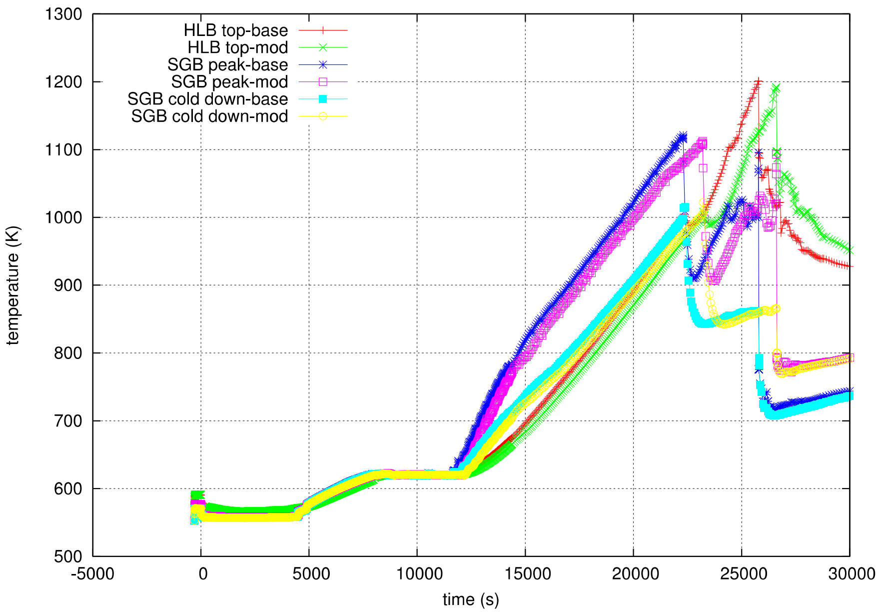
- Objectives:
  - Provide TH results for CE plants to be used to calculate component failure using:
    - CSGTR Calculator
    - Finite Element calculations
  - Provide scoping component failure calculations
  - Provide FP releases
- Analyses (scoping and component-failure-suppressed calculations)
  - Accidents
    - Short terms station blackout (stsbo) (AFW fails to start)
    - Long term station blackout (ltsbo)
  - Variations
    - Operator actions/relief valve behavior
    - RCP seal leakage
- Comparison against RELAP5 CE calculation
- Uncertainty analysis to determine impact of TH uncertainty on failure timing

Example MELCOR CE results

Impact of RCP seal leakage sensitivity



Effect on system pressures



Effect on Loop B structure temperatures

Possible future CE TH work

- Interesting but deferred work because of resource limitations
  - More detailed spatial temperature distribution
  - Assessment of TH factors that impact relative component failure timing
  - Analysis of loop seal clearing
  - Water hold up in SG, flooding / counter-current flow
    - Water also held up in previous SGAP calculations
  - Detailed evaluation of FP release
    - Current focus on TH input, not FP release
      - Didn't rerun cases to solely extract FP release behavior