Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

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

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Friday, September 25, 2009

Work Order No.: NRC-3101 Pages 1-142

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARD

(ACRS)

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SUBCOMMITTEE ON MATERIALS, METALLURGY AND

REACTOR FUELS

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FRIDAY, SEPTEMBER 25, 2009

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ROCKVILLE, MARYLAND

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

SUBCOMMITTEE MEMBERS PRESENT:

DANA A. POWERS, Chair

J. SAM ARMIJO

SANJOY BANERJEE

DENNIS C. BLEY

WILLIAM J. SHACK

JOHN D. SIEBER

JOHN W. STETKAR

NRC STAFF PRESENT:

CHRISTOPHER BROWN, Designated Federal Official

SELIM SANCAKTAR

ED FULLER

KEVIN COYNE

ALSO PRESENT:

BOB LUTZ

DON FLETCHER

DAVE BRADLEY

C O N T E N T S

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

8:30 A.M.

CHAIR POWERS: Let's come back into session, we are continuing our discussion of the Steam Generator Action Plan. I see no reason to go through a laborious introduction here. I will remind people that one, we encourage comments from the audience, and if you choose to comment, please come to a microphone, identify yourself and speak with sufficient volume and clarity that we can make out what you're trying to say.

With that, we will begin our discussions of Item 3.4, Selim, the floor is yours.

MR. SANCAKTAR: Good morning. Let me see if I can get a sense of what's the best way to talk into this naturally. My name is Selim Sancaktar. I'm in the PRA branch of U.S. NRC research and I've been with them for the last five years. Before that, I worked for Westinghouse Electric Corporation in the area of PRA for I forgot how many decades.

Well, following his lead, I would like to confess first that I know nothing about materials and my T&H experience is limited to boiling eggs and making tea. So those areas were --

CHAIR POWERS: Perfectly adequate.

MR. SANCAKTAR: Those areas were extensively discussed yesterday and if necessary I have allies, hopefully behind me, where are they, they're all gone, who will help me.

Here's the outline of the presentation. I'm not going to go into the details itself, it's self explanatory. Let me make sure of the very first thing. My purpose today, my presentation today is about this document, just to make sure.

It's a 145 page report, the title of A Risk Assessment notice A Risk Assessment, not the, not the definitive, not the, it's A. Please note that it's not -- it's on purpose that it's stated that way.

A Risk Assessment of Consequential Steam Generator Tube Ruptures, March 2009. It is approximately or exactly 145 pages. So there's no misunderstanding, this is what I'm talking about. Any credit or follies in this report, I take responsibility for.

And second report that kind of supports this is we refer to it as the Sandia report. Its title is Sever Accident Initiated Steam Generator Tube Ruptures Leading to Containment Bypass Integrated Risk Assessment, February 2008 Letter Report JCNY6486. So I have references to this.

MEMBER BLEY: Just for your information, we only had the 2005 version until this morning, so we haven't had a chance to really go through the most recent Sandia report in any detail. We'll have questions about that.

MR. SANCAKTAR: Just to complete the preliminary information, I have a couple of people who were involved in the Sandia report sitting behind me and they will -- the moment I get into trouble I'll blame them and ask them questions, so they're there.

MEMBER STETKAR: Selim to interrupt you right here at the beginning, are you going to give us information about how the conditional induced tube failure probability was calculated?

MR. SANCAKTAR: Yes. I will even show you pictures --

MEMBER STETKAR: Okay. Good. Thanks.

MR. SANCAKTAR: -- like this to satisfy your curiosity since yesterday afternoon the subject came up and I made these additional slides so that we can dazzle you with color.

MEMBER STETKAR: Good. We don't have hard copies of those do we?

MR. SANCAKTAR: No, because I just made -- I made these, I made two slides afterwards, yesterday after I left --

MEMBER STETKAR: Those are a little bit difficult --

MR. SANCAKTAR: Yes. The purpose so you can't read them, but you can see that they are there.

MEMBER STETKAR: When we get to that part early or late, I'd like to have a hard copy as soon as I can get one.

MEMBER SHACK: Are these from 1570? Are these the same?

MR. SANCAKTAR: No, no. They're from, in fact, you can't find them anywhere in the record, these kind of pages.

MEMBER BLEY: Until today.

MR. SANCAKTAR: Until today. I made them -- I made the slides yesterday from the software that we have.

So I'm going to talk about what we have done most recently and based on that, I will make the recommendation that we should close Steam Generator Action Plan Item 3.5, whatever that means in this context. We already wrote a letter as RES and requested that it should be closed within the NRC organization, but what this means in the context of ACRS, I'll leave it to you.

The serious in perspective I have to give you the story. The Steam Generator Action Plant Test 3.5 contains seven subtasks and I'm listing them, three of them are here the four are on the next page. To understand this thing as much as or as little as I do, let me give you another piece of background information.

I got into this about 22 months ago, I think Jeff is also in a similar situation as I am plus or minus so many months. And before that if you asked me what SGAP stood for, I wouldn't be able to tell you except to guess SG looks like Steam generator.

At that time, the first three tasks were declared closed by RES. So when it says closed next to it, they were declared closed and there were ML numbers next to them, which were removed by powers higher than I am from this slide.

And these were basically interim reports that eventually ended up, I think one way or the other, in the Sandia report. So if you read this, it covers everything that went before that, if you get a chance to read it.

So the first three were declared closed and I'm going to emphasize the word declared, okay. And so I take -- I look at this and I look at all the tasks and the other tasks say things like calculate the frequency of containment bypass, extend them to CE plants, extend them to low-power and shutdown and then to say something about main steamline breaks et cetera.

So this is everything except the kitchen sink, maybe also the kitchen sink is in here. So this is a rather ambitious and complete set of scopes. So at that point I sat down and I took inventory.

I said first of all, what do we have with respect to PRA out to this point, what is available? This slide is not -- it's a back up slide, so you don't have it there. I had to get to this to make this story clear, okay.

It was in there, it has been moved around, but I have to use this to tell you my -- to give you a continuous and cohesive discussion. So I said, okay, what is available to us, you know, I'm here, I'm supposed to do something what do I have with respect to PRA.

Well the grandfather of all is the NUREG-1570, which is like 12 years or so old by now, something like that. It's a noble effort, it covers -- it attempts to cover everything and of course in doing that, it creates a lot of questions.

However, it's a very, in my opinion, it's a very intensive effort and it's really touched up on everything. I mean it didn't leave anything unturned, although it may not have conclusively really satisfied the stuff.

And then I found an INL report that supported that, another INL report that supported it. So these are all available in one way or the other, but they are time frame of the 1570, which is like 12 years old.

And then there was the Sandia report, which was -- which comes as the next step from NRC point of view which was pretty much done by 2005 in draft form, but it was sitting to be finished. And that's the report that finally was finished in February 2008 and is one of the references I have.

In the meantime, there is also the EPRI report, which is licensed material. So I don't know how much I can quote, but I looked at it and I formed some opinion as to other aspects of things.

MR. FULLER: If you quote anything it's to the NRC.

MR. SANCAKTAR: Good. Good. So if I make a statement to that it's not --

MEMBER STETKAR: For the record, you have to identify yourself.

MR. FULLER: I'm Ed Fuller, I'm from the NRO.

MEMBER STETKAR: You have to come to a microphone and do this.

MR. FULLER: Sorry. I'm Ed Fuller from NRO. That report and its accompanying document, the Volume 2 Specific Application to the Diablo Canyon plant was provided to the NRC about three years ago, I guess, right about the same time of that public meeting or shortly after that public meeting that was talked about yesterday.

And if you're working for the NRC or you're a contractor to the NRC who has signed non-disclosure agreements, you have access to those reports.

MEMBER SHACK: They're in ADAMS.

MR. SANCAKTAR: So after taking that inventory I also tried to form an opinion as to what else is going on, not necessarily in PRA, but other things. And I'm going to list some bullets here. Some of them may sound like what does that have to do with this, bear with me, it does eventually.

First of all, Steam generator two rupture is routinely modeled in risk assessments as an initiating event, as a random event. You know that. I mean, that's really this slide is kind of repeating what you've seen. You knew already and you probably seen a couple of times yesterday so I'm going to kind of skip this and Consequential Steam Generator means this.

Notice that, just let me call your attention to something that I consider the scope to include all Consequential Steam Generators, not necessarily limited to ones that are driven by temperature, mainly and pressure as the secondary, but maybe also as pressure as the main culprit regardless of the temperature like maybe ATWS where the primary is not -- the primary pressure is not controlled and a spike occurs or really, really a large steamline break where things drop.

At some point you may ask me what do you mean by really, really a large steamline, but I haven't fully resolved that yet honestly I'm telling you up front.

However, it can't be a small one. It can't be one that's so tiny that the plant doesn't even know that it happened. And I looked at the events in the INL database that went into steamline break database and many of them, the plant couldn't feel it.

I mean, the operators in some cases manually tripped the reactor according to the description. So there's a demarcation after which this becomes an issue and I don't think it's well -- to my knowledge it's not well defined yet.

MEMBER STETKAR: Let me ask you since you brought it up.

MR. SANCAKTAR: Sure.

MEMBER STETKAR: So I can get a feel for the rate of depressurization that you're talking about. Would, for example, if the main turbine failed to trip and one of the MSIVs stuck open, would that qualify as a sufficiently large steam release to challenge this?

MR. SANCAKTAR: Well as I told you, I'm not --

MEMBER STETKAR: Just yes or no.

MR. SANCAKTAR: Say that again, one more time.

MEMBER STETKAR: Main turbine fails to trip, so the turbines stop -- one set of turbines stop and governor valves stay open and one MSIV fails to close. So you connect the steam generator to the main condenser after the reactor trips. Is that big enough?

MR. SANCAKTAR: I don't have a sense, I don't know.

MEMBER STETKAR: Thermal hydraulics guys, is that big enough? Suppose all the turbine bypass valves stick open and one of the main MSIVs failed to close. Is that big enough?

MEMBER SIEBER: You'll see a significant change in perimeters.

MEMBER STETKAR: You'll certainly will. You'll see a pretty doggone cool down. I'm trying to get a handle on what this ten to the minus five very large steamline break really means because it has a frequency, so somebody must know how big it is.

MR. SANCAKTAR: Yes, I'll come to that when --

MEMBER STETKAR: And I'm trying to figure what other frequency it might have.

MR. SANCAKTAR: I'll come to that when we proceed.

MEMBER STETKAR: Okay. I'd like an answer to the question though about how big is big enough.

MEMBER BANERJEE: You should table the question and Don or Chris could possibly give you an idea of that.

MEMBER STETKAR: Okay.

MEMBER BANERJEE: I don't have an idea of that.

MR. SANCAKTAR: For example, I have a drawing from FSAR here that looks at that 1.4 square feet break of site power available pressure, eight-inch radius, it's about equivalent to an eight-inch radius, this break 1.4, which is 16-inch diameter. And I look at the profile and looked at how quickly dropped and how many PSI it dropped.

But still it doesn't answer the question until you also know the other side of the story, mainly --

MEMBER SIEBER: What's on the primary side.

MR. SANCAKTAR: Yes, what does it do to you. I examine this and I -- we'll come to that.

MEMBER SHACK: Just to calibrate you on the delta-T for an unflood tube is probably like 9,000 PSI.

MEMBER SIEBER: For an unflooded tube?

MEMBER SHACK: Yes.

MEMBER SIEBER: But they're looking at a very large steamline break as having a very high conditional probability of conditional Steam generator tube failure. I need to understand what that is.

MEMBER BLEY: And when you get to it later, I think what John's asking you is when you calculated the frequency of that break did you just look at pipe ruptures or did you also look at functional failures in the plant that could give you the same conditions?

MR. SANCAKTAR: Can we -- can I retain that question until --

MEMBER BLEY: You bet.

MR. SANCAKTAR: -- and put it in place before the end of my talk but can -- would you let me get a bit more because it may fit in the context better, if you don't mind.

MEMBER BLEY: Go ahead. I'll write it down so I don't forget.

MR. SANCAKTAR: So, just a couple of more points here at the risk of boring you with some of them. There are two aspects of the risk from Consequential Steam Generator Tube Rupture if it occurs due to another initiating main steamline break or ATWS and it wasn't modeled originally it could increase the CDF, it's on account of CDF if it's not modeled already.

And if it occurs after core damage due to core damage then it would not increase the CDF, concede it, we conceded the CDF, however it can increase the containment bypass, which then brings up questions like what happens afterwards and I'm trying to avoid using the word LERF, but I guess I can only stay away from it so much.

But in any case, in my mind the issue is containment bypass. You know, I'm not fixating on LERF necessarily. I look at it a little bit more broadly.

This subject has been studied by NUREG-1570 and since then extensively by the NRC, not only in PRA, but of course in materials and thermal hydraulics areas as you heard yesterday. So I looked into that and I got a very warm feeling from the thermal hydraulics aspects.

PARTICIPANT: Warm?

MR. SANCAKTAR: Warm.

PARTICIPANT: Like temperature-wise?

MR. SANCAKTAR: No, comfortable. I thought that there really, whatever is the state-of-the-art I'm not in a position to claim I know it, but they look like competent and state-of-the-art kind of activities, so I got a warm feeling from them.

Their major observations have been used, NPRA. I looked at their pre-report and I noticed that for example, they looked at a couple of CE plants and one of them had much larger Consequential Steam Generator Tube Rupture frequency than the other one.

Then I looked at it a little bit more then I realize it because one of them I think did not have a bleed mechanism or something. There was a major difference in the designs of the two plants.

So, I got a very distinct opinion that as risk is plant specific depending upon what sequences dominate at the time frame you are looking at them, even those can change in time, not simply improve the models, we change the data and so on.

And so sequences keep moving even if basically the design and operation stays the same. And I looked at all the rather extensive work done on 1570 and their pre-report and in the Sandia report and I reached a conclusion that everything has been modeled to a level of a Cadillac.

And I said what would that mean if we stated, okay, there are two aspects of it, one, what do we expect the licensee to do, you know, how much detail, how much intrusion they should go into the PRA.

And the second one is what happens if I

get a phone call from NRR and they say we received an analysis of Consequential Steam Generator, they'll give you one man month, one engineering man month and you evaluate the situation, give us an independent situation. What would I do?

Would I go into this -- can I exercise this in a just way without too much effort. So what is an appropriate level of what is the method that we should consider and the Cadillac methods that are around, I mean you can always do, draw new event trees add things to each content sequence and calculate their stuff.

That's already there, it can be done. No problem there. But what would I do if I were in the shoes of NRR analyst or if I were asked by them. So I looked at it that way from a pragmatic point of view.

It appears that I didn't give them the silver bullet because they have a user need that's coming up to do more stuff so obviously. I thought it was a close to a silver bullet but there are -- they're at the front end of our efforts at the NRC so we try to support them to the best of our abilities.

So what else happened? As you know, and you mentioned this yesterday, there were lots of Steam generator replacements which improved the materials and even if you don't trust the materials at least they gave you fresh start with a new Steam generator. So yes, less flaws hopefully. We can conceded at least that if not if you still wonder about what will happen in time with the materials.

In rest of the PRA models have matured and there were lots of changes in the plant risk profiles and our SPAR models also reflect that, especially in the last few years when we try to calibrate the SPAR models with the industry models.

As you may or may not know, we have 77 PRA models, SPAR models that represent all 104 units that, domestic units in the United States and we, the NRC spent considerable effort to see that our results are -- I wouldn't call it benchmarked because we don't assume that theirs is right; however, we look at the differences and try to justify them and we still have problems with some of the success criteria and we are success criteria might be is slightly more conservative in certain areas.

MEMBER STETKAR: I'm just curious. I was kind of surprised to learn that the SPAR models didn't have a steamline break initiating event at all --

MR. SANCAKTAR: Right. Yes.

MEMBER STETKAR: -- which raises questions about how complete the SPAR models might be. How confident are you that the SPAR models actually are fairly robust in terms of looking at conditions that could potentially challenge these induced tube ruptures, if I can call them that.

Because as noted in the Sandia report, you have to make sure that your models are set up to look for the things that you're trying to quantify. So, for example, you discovered that the SPAR models didn't include steamline breaks so you had to add a steamline break model because you wanted to go look for that potential challenge.

How confident are you that the SPAR models are fairly robust in other ways that might challenge these events?

MR. SANCAKTAR: Personally, my opinion, I'm very confident. Why? Because I spent my life in PRA and I know, I've seen things left and right. Now in an analytic sense, in a digital sense if you say did you take the 1000 points of reference, each one, they meet each one or not. I cannot argue that.

However, as we speak, the SPAR models are being peer-reviewed, you may or may not know, as we speak. The BWR one has been peer-reviewed last month and if you're interested I'm sure that kind of information can be provided to you. I wasn't involved in it, so I cannot tell too much. And the PWR one will be peer-reviewed in a week or two.

So, I don't see any big ticket item that's missing. I have done peer reviews for PWRs, I led them. I led about five or six of them and so I have a sense of what goes on where. So I feel comfortable based on my experience, but does that spar your curiosity, I don't know.

MEMBER STETKAR: Do the SPAR models contain feedwater line break events?

MR. SANCAKTAR: Separately? Yes. No.

MEMBER STETKAR: Did you quantify the frequency of a large feedwater line break event in your analysis?

MR. SANCAKTAR: It's buried in there.

MEMBER STETKAR: Oh I'm sure it is.

MR. SANCAKTAR: There is another important development, the data collection that research does and makes available to everybody recently updated initiating frequencies, power recovery and other components data so that SBO event evaluations might have, the things of dominant sequences would have -- could have moved very easily up and down.

I seem to sense a general trend that

station blackout numbers are coming down in a risk sense. And it includes the ones in the SPAR models that kind of a little bit, makes me a little bit think about it, I'm not sure if it's too premature to give credit for certain things very quickly, but they are well justified I mean with the current information.

But you never know, these things go up and down. I mean last decade we thought a large LOCA was 10 to minus six everybody stuck at 10 to minus six as an initiative event.

Then we had problems with that and they have analysis and discretion and now it's 10 to the minus five-ish if you went 20 years ago it was 10 to the minus four, so the things moved from decade to decade.

So we have to recognize that even if you don't change the plans, our understanding of or our representation of risk can change even subtly within overall numbers. We may still see if one times ten minus five, but the composition may subtly change. So we have to be careful as to how we are going to do this thing with the evaluation.

And another point that I would like to make is the industry has also documents about steam generator, how to handle steam generator events first.

I mean first you have to know how to handle steam generator as an event, what happens after it breaks.

Because our original understanding, assumption and risk space was if the steamline break -- excuse me, steam generator to break occurs it's large release, you know, everything goes out we are in the domain of LERF, but that's not necessarily true and there has been work done to point out to point out under what else conditions really it happens one way and what else conditions you have releases that are quite moderated.

And this has been studied by Westinghouse Owners Group, whatever its new name is now, PWR Owners Group and they're, I think, coming up with more guidance and so on.

The new PRA standards provide guidance on more integrated evaluation of the plants internal events, external events, events that shut down are commonly mentioned and whether there's action taken on that or not, but at least their level of attention is changing so we can certainly talk about maybe quantify reasonably -

Security issues --

COURT REPORTER: Please keep your voice up.

MR. SANCAKTAR: Okay. You assume that I have the energy to do that, you know.

Security issues prompted utilities to provide new equipment and procedures that can apply recovery from severe accidents, not only in the domain of recovering but together see core damage. But even if the core damage is conceded, maybe not get other things such as Consequential Steam Generator Tube Rupture.

And for each of the plants I looked at, I actually looked at their super-secret whatever documents or B5P stuff and see what they have, equipment and operator action logs. And one thing I noticed was I didn't see anything new there actually.

I mean, all the stuff, you have to put either water on the secondary side or depressurize the prime, I mean there aren't that many things you can do. It's not like you have 100 different things.

And many of these in the PRAs in the past, one way or the other were credited, sometimes without too much basis, like you say, oh if it happens I bring a fire pump and pump the water on the secondary and stuff like that without really any strong basis.

What these did was, they pointed out I will use this pump and it has this capacity. I'm going to take the water from here. So it makes it concrete and gives it a bit more credibility and also it strengthens the paperwork so that now we can say, oh there is a procedure for it, or at least some administrative letter or whatever does the job.

So they have this kind of stuff, but they are not really new. I mean, they were around in one way or the other.

And finally severe accident management guidelines and stuff like that is now well established, operators are more comfortable, they know about it, the probably got used to it so they don't think it's just pie in the sky and so on.

So they're -- so in general, when I look at it I didn't see any negative things, you know. I mean, things were fine. They were in the positive direction with respect to this subject, although, they didn't really were necessarily done to address Consequential Steam Generator Tube Rupture.

MEMBER STETKAR: Can I ask a thermal hydraulics question because I can't boil water for tea either, I don't know anything about materials.

If I introduce cold water into a dry steam generator, you mentioned fire pump example, as part of a impending severe accident, mitigation or severe accident mitigation measure, do I actually increase the conditional probability of a tube rupture in that steam generator compared to what it was when it was dry?

MR. SANCAKTAR: This is a very good question and that's why I told you that I don't know anything about materials, so I will not be able to give you an answer that is worth anything. But if there's anybody here or there that has an opinion or knowledge one way or the other, they're welcome to answer. I can tell you what it feels like.

MEMBER STETKAR: I know what it feels like, I'm trying to figure out whether it makes any difference because I've been wrong too many times in my kind intuitive sense of this issue.

MR. SANCAKTAR: Absolutely. That's a very good point. So --

MEMBER STETKAR: That's actually an honest question that I'm curious about.

MR. SANCAKTAR: Anybody ventures a guess? Okay, so --

MEMBER ARMIJO: John, what are you concerned about that there's something like a thermal shock or pressurization or something that would damage --

MEMBER STETKAR: Not so much a pressurization it's a thermal shock. I'm increasing the delta-T at a given pressure across those tubes and these types of -- what I'm concerned about is that these types of recovery actions are pretty typical in PRAs these days as Selim mentioned. Those sequences are success sequences in the PRA, they don't show up as core damage sequences --

MEMBER ARMIJO: I just, you know --

MEMBER STETKAR: -- at the moment so they wouldn't be captured by the process that they're using.

MEMBER ARMIJO: You're speculating. I wouldn't worry too much about the tubes, but the joints in the tube sheets and stuff like that, there could be a lot of strange stresses.

MR. LUTZ: I'm Bob Lutz from Westinghouse. I've been involved in emergency procedure work for a long time and sever accident management work for a long time. For the -- I can speak for the Westinghouse NSSS plants that feeding a dry steam generator all of the emergency procedures say to feed it slowly at first just for the very reason of cold thermal shock.

You don't want to hit it with the full

load of flow when it's dry. I believe the procedures, and I don't have them in front of me, but I believe they say to feed at 100 gpm per generator for 10 minutes or something like that before you increase to full flow. That's based on material issues.

COURT REPORTER: Speak into the mic.

MEMBER SIEBER: Okay. It's the steel components that are especially susceptible to thermal shock more so than the alloy 600 components which have a different thermal characteristic and transient response. But it's never a good idea to change temperature very rapidly and you can do that by feeding an empty steam generator. The question is whether it will be what the temperature difference will be.

MEMBER STETKAR: I guess my question stems from the fact that this guidance has been around for several years about restoring feedwater to a dry steam generator and obviously the guidance has said to feed it slowly.

I'm curious whether the folks doing the thermal hydraulic analysis and the PRA work, as we learned yesterday, it's an iterative process, have thought about these types of scenarios given the current guidance for feeding dry steam generator and have thought about whether that particular type of condition would indeed lead to measurable conditional probabilities of tube failures.

Because we've heard an awful lot about core damage sequences during station blackout leading to, you know, reflux cooling or whatever you call it these days and very high temperatures at the tube sheets, but I haven't heard much about these other types of conditions.

MR. FLETCHER: Don Fletcher, ISL. The analysis we've done, we have not looked at slow onset of aux feed. The aux feed is delivered at the full rate on or off or we control it to level, something like that. Have not considered your question.

The earlier question on steamline breaks and whether something like sticking open all turbine bypass might be worse. It seems intuitive that the pipe break, especially if you could break it upstream of the flow whether would be the worse possible condition from the viewpoint of the delta-P across the tubes.

MEMBER STETKAR: Certainly true that's the worst possible condition, I was trying to search for whether that's the only possible condition or whether there might be other challenges that are perhaps not equally severe because that particular condition is assigned a relatively high likelihood of conditional tube failure.

I'm looking for perhaps more frequent conditions that are -- might be somewhat less challenging but still might have a measurable conditional probability.

MR. FLETCHER: Well there's certainly been a lot of steamline break analysis done and I've done quite a bit of it, but mainly looking at reactor vessel PTS, cool down events, return to power that sort of thing.

There's probably a wealth of steamline break thermal hydraulic data out there that could be used to evaluate that if you want to take the pressures and temperatures and see what effect it would have on tubes, but I've not personally done any tube rupture, induced tube rupture steamline break analysis.

MEMBER BLEY: I guess the point John's getting at and we raised yesterday a few times is we're using analysis and sometimes of worse cases to bound a problem, then we're doing a PRA that's supposed to look at everything.

If that PRA doesn't look at all the cases that might influence these conditions it might be giving us a seriously wrong answer.

And that's why we're saying what if there's something more frequent, maybe less severe that could get you into some of the same troubles is if you thought about it, is it covered somewhere in the thinking that's led to the final conclusions people are coming up with. Go ahead.

MR. SANCAKTAR: Done? There is one more piece that I want to show you, PRA people are probably very familiar with this just in case you haven't seen this. This is one of the pillars of my argument, the PRA standard which I copied verbatim, indicates that in order to meet Capability 2 Category, which is what people go as a minimum, I never seen anybody who aspires to go for Category 1, so I don't know what it does.

Capability to Category 2 or 3 says thou shalt look at Consequential Steam Generator Tube Ruptures, so when NRC gets an application in some area and it has to address this issue. So I just wanted to put that as one of the points that I think is relevant.

So, what I have done is, I basically said we are the SPAR models and we can use them to get our dominant sequences which are most recently updated. Why don't I use those two to talk about a simple method, which does not exclude others from going ahead with the Cadillac approach, of course.

But I have the SPAR model, I can get the dominant sequences. It turns out that I even have external event models for 15 of them, so I can point out, regardless of the pedigree of those -- I am on Slide 11, sorry.

So I happen to even excel the event models for 15 of them, I did pretty much go through everything and get some assessment of what the risk of bypass would be for a specific plant rather than some arbitrary generic plant.

When I say specific again, the specificity is brought in by the sequences not by the other modeling assumptions. They are still very generic. I mean, I'm not making any claims about that.

(Off mic comments.)

MR. SANCAKTAR: Is this again rate for a Consequential Steam Generator or not? If it is I put it in this box, if it's not I put it in this box and then I keep doing that until I reach a point where nobody can argue that I left out too many of them. So went like to 99.9-something and if I couldn't make up my mind I put it in the possibly challenging situation box, I'm going to come back and visit those.

So the others I left out and I looked at the thinking process behind it and I tried to make it into some sort of more rule-based, you know. If you see this and this, keep it, if you don't see this and this, don't keep it, et cetera, that kind of a thing.

And you can argue that if there's a complete set is there anything else? Okay, if there's one more thing I'll throw it in, okay I'm not going to argue to you that it's complete for every plant, every time, but it's a good set and I'm being rather cautious so if I can't make up my mind I'll keep it in and discuss it further on.

So, but that process you can actually see if you have the report --

MEMBER BLEY: By the way, the bottom of your slide lists this RIS 2008-15, is that this report? I don't recognize that number.

MR. SANCAKTAR: No. What happened is very recently, like last month, we tried to issue for, especially for ACRS consumption I would say, a short 10-page report that kind of describes how all the pieces of the research work fit together, the materials, the T&H and PRA.

So it kind of gives you a roadmap and lists the documents and touches the high points, it's about 10 pages in length. And we rushed it through to get to you like a couple of weeks before the meeting. Did you get it?

MEMBER BLEY: No, we did not.

CHAIR POWERS: Well, we did.

MEMBER BLEY: We did?

CHAIR POWERS: You probably just haven't just seen it.

MEMBER BLEY: Oh, is it the letter?

MEMBER STETKAR: We got a lot of stuff.

MEMBER BLEY: Okay. I didn't recognize it for what it is then.

MR. SANCAKTAR: I'm sorry, we tried.

CHAIR POWERS: It's in the file.

MEMBER BLEY: I called it the letter.

MR. SANCAKTAR: We tried to give you a roadmap of all the pieces and things floating, probably just ask for the paperwork, I don't know if it's -- but the intent was supposed to be good.

MEMBER BLEY: I got it. I again, didn't recognize it.

MR. SANCAKTAR: So if you look at -- I don't say did you look at it now, but I'm just pointing it to you. If you look at the report I have Appendices A, B, C. A is for plant 1, B is for plant 2, C is for plant 3. And it's very repetitive. You'll see the tables repeating. A-1, A-2, A-3. B-1, B-2, B-3.

If you look at Table A-3, it's basically what I just told you. It's the list from the PRA model, it's from SPAR model, but it could have been from a PAR model or whatever, it doesn't matter. Set down, pick them up put them in Bin A or Bin B, that's all it does.

And, for example, just to give you a feel for all the stuff in there, the plant CDF is two times ten to minus five, the stuff that went into the box that is potential candidates is about nine times ten to the minus six, which is like 50 percent of the CDF.

So if somebody said that I know nothing else, I need a first approximation. This is your first approximation, 50 percent of the CDF can be put into the candidate box. It doesn't mean that these are bad guys or they're going to do -- but they're usually bad guys, those that make it into there are not the milder sequences. So I went down --

MEMBER BLEY: Are you going to go through those in some detail or is this a time to ask you about those?

MR. SANCAKTAR: I'm going to go through some of the example ones, top ones for each plant --

MEMBER BLEY: I'll wait until you get there.

MR. SANCAKTAR: -- to the level of your interest. I can spend two minutes on them or I can spend an hour on them. As we approach it, you decide how much you want to probe it and I will try to oblige.

So when I went down all the way, just to give you a feel so you know, I mean it's positive or negative, your choice, you decide. I went to 99.9 percent of the cumulative CDF, that's 132 sequences.

A couple of fine points here that I'm not trying -- I'm trying to stay out of them not to confuse the issue. First, I want to give you the big picture then I'll touch up on little points that are really devils in the detail type of stuff.

So then I took those candidate sequences which I don't know how many are here, say 10, 20, somewhere between 30 to 50 of them, then I looked at their -- what happened in those sequences and then basically the method that I suggest is a product of three factors; you multiply the candidate sequence by the factor that says, I can mitigate this sequence so I can remove the steam generator -- tube challenge without arguing with the core-melt part.

The core-melt part I conceded already. It's core-melt. Whatever you do, you're not going to recover from that. If you recover from core-melt then you should have done it before hand and not -- it should have factored into the original PRA.

So I'm not arguing with anybody's success criteria, what I'm saying here's the first number, let's say sequence is one times ten to the minus six from the PRA, whatever PRA we are looking at.

One thing you can do is, you can remove the challenge by some mitigative action or multiple mitigative actions. So that will be a factor that multiplies it.

Let's say this factor, for the sake of argument is .1 for operator actions, a set of operator actions that if you apply them you will get an order of magnitude reduction so that we'll move, you know, sequences by an order of magnitude lower. If it's two orders of magnitude, better.

The third factor is, your mitigation fails, but the tubes survive, that's the conditional Consequential Steam Generator Tube Rupture probability. So there are three numbers, frequency times probability one, times probability two.

All sequences can be looked at this way and then you can take each of the last two probabilities, mitigation and conditional steam generator tube rupture and you can either assign them screening values and then throw away the ones that are not relevant or you can spend your money and life analyzing the heck out of each one. Your choice.

But as a beginning, I can quickly hone in and eliminate those that are riff-raff, you know, that are obviously nobody would agree -- disagree that it will fall off the scale because you can think of credible mitigation or we have already determined that this has a low conditional steam generator tube rupture probability.

So my objective is to get to know as many of them as possible so we don't spend the time arguing about things that are relevant, they look relevant, they look important, they are interesting, but really, it doesn't matter for this plant if you assume that the front end is right, that the sequences you got -- I'm not arguing with that, okay. I mean, if they're wrong, they're wrong, that's a different story.

MEMBER BLEY: Something, I confess I

didn't understand, but I think I'm beginning to. This report is kind of a standalone report, your report.

MR. SANCAKTAR: Yes.

MEMBER BLEY: And you're reporting --

MR. SANCAKTAR: As much as possible.

MEMBER BLEY: -- hoarding results from the thing Sandia and others have found.

MR. SANCAKTAR: No. Right.

MEMBER BLEY: It's completely --

MR. SANCAKTAR: Yes. Yes.

MEMBER BLEY: -- separate taking your own view --

MR. SANCAKTAR: Absolutely.

MEMBER BLEY: Okay.

MR. SANCAKTAR: I tried to use pieces available that are as recent as possible like stuff from thermal hydraulics that are insights that came through the Sandia report. But it is pretty much, yes, I would say so.

So by doing this, you can pretty quickly get a sense of what the Consequential Steam Generator frequency is. And this frequency, I keep calling it containment bypass, okay. I'm not claiming it's LERF. I mean if we're going to say it's LERF, you have to do some more analysis. I mean, there's another person needed to pass that judgment or make that call.

However, as, again, first upper bound estimate you can say that, okay, let's say these are LERF, you know if you want to, but it would be a very conservative assumption. So it's a very -- that's it basically. So what do we do, let's see what else.

So I have three factors, frequency from the PRA model times mitigation failure whatever is physically possible for that particular sequence, not necessarily only for that plant, but for that sequence because I'm focusing on that sequence, the next one is the conditional core damage.

MEMBER STETKAR: Are you going to walk us through a couple of these sequences --

MR. SANCAKTAR: Yes.

MEMBER STETKAR: -- so we can kind of understand what you did you in practice.

MR. SANCAKTAR: Absolutely. There's a bunch of them.

MEMBER STETKAR: Okay. A couple of kind of interesting ones. I don't want to -- as long as you're going to do it, when you get to the appropriate time, that's great.

MR. SANCAKTAR: Yes, they'll come. So

next slide says I'll go through this and do it for internal events. So in the recovery probabilities assigned to consequential core damage probabilities for each of the sequences, calculate the sum and go through and then do it for internal floating, which is another set of sequences.

Normally, steamline breaks are covered in the PRA studies and they're in the base model and they're in the base model. SPAR models didn't have them. In the past PRAs, I included like three event trees, one for inside containment, one for outside containment, but not isolable one for outside containment but isolable, et cetera.

I mean, ordinally at least at the beginning of the PRA process people went all to the detail to this kind of detail in PRA model because main steamline break was really one of the main things in the accident -- deterministic accident analysis list, which you start from.

Now I'm going to go to some opinion, I'm going to switch to opinion, it may or not be correct, but at time passed, I think people started seeing that steamline breaks are not really driving the risk profile of the plant and especially if they are isolable, the type that are isolable. So I don't know how much detail and how many different event trees are used in many PRAs today. I don't know.

But anyway, this model, SPARs models didn't have them, so I had to supply them. So I supplied them, I gave it to them, and since I'm giving it to them I asked the first question, do you get Consequential Steam Generator or not.

So that kind of immediately diverts the sequences to the right place. Then I said, well since this kind of gets back to the original point, so the first thing I said that, okay what is it, which type of a steamline break will really challenge the tubes, what are the assumptions up to this point, how much is studied and I asked people, both our vendors and also at NRR, what is the steamline frequency -- steamline break frequency what are the details.

I went and looked at the database that calculated our steamline break frequency that each event, and I realized that there were really minuscule steamline breaks, the ones that are in our database, INL database. And typically you assign ten to the minus three number or frequency number. COURT REPORTER: Sir, please keep your voice up. You need to keep your voice up.

MR. SANCAKTAR: I checked around and the

best advice I could get was from the one of the vendors we have, they said we don't have a seriously large steamline break in 12,000 years of operation of nuclear power plants, all right, something really that is large.

Again, there are two parts to this, the initiating event and then the conditional Consequential Steam Generator Tube Rupture probability.

The number for Consequential Steam Generator conditional probability that was suggested by 1570 was that .05. I think it was more of a gentleman's agreement or expert opinion than a calculation. I don't know that, it's an opinion.

And I watched research, very eminent research, not research and another member in front of a CRS a few months ago saying that it's probably lower than that now a days. I mean they don't think it's a big fraction. But this didn't really satisfy me. I don't know the answer, whether it's .05 or if it's .005 and if you noticed I used a high number, I forgot now, it's either .5 or .4 in the study.

The reason is I'm not satisfied yet as whether to make it as a large drop would burst the tubes is well studied to a point where we can just rest it. I don't know, maybe it is. But it wasn't clear to me, I didn't have the information. So I left a needle in there.

How did I leave the needle in there? I used the fact that we didn't see any large steamline break to get small frequency which attracts attention immediately so that we don't lose it, it's there. But at the same time, it doesn't distort the final product, so I used .5-ish number. So I'm sitting at point that if either of them --

MEMBER SHACK: Which one is .5 conditional tube rupture?

MR. SANCAKTAR: Like the three factors, the first factor is the sequence number and the second factor is mitigation, the third factor is a conditional CDF.

MEMBER STETKAR: There was a number, a .5 for unanalyzed conditions.

MR. SANCAKTAR: I could have easily used -- I could have taken the easy path, okay, here is ten minus three for the front, people will see it, we'll be happy nobody will ask, then put a .05 for the Consequential Steam Generator Tube Rupture. The products come out to the same place.

I mean it's -- everything came out so

small that even if I moved things and alter the magnitude up and down, with the current information it's not going to move too much; however, I'm personally interested in finding out more about what is the real story and I wouldn't mind we pursue it if there is an interest in that from NRR people.

But however, if they are satisfied with the current state of practical knowledge it's fine. I mean, I have no --

MEMBER BLEY: Let me ask you a question because you're making a big point of keeping this around. But you only looked at core damage sequences, now if you're looking at pressure-induced tube rupture, if you have an event that causes that, that changes --

MR. SANCAKTAR: Of course.

MEMBER BLEY: -- the whole event sequence --

MR. SANCAKTAR: Yes.

MEMBER BLEY: -- diagram so you should have been looking at things that didn't lead to core damage too, the whole event tree, right?

MR. SANCAKTAR: There are such situations possible theoretically. In this case, I plugged the hole because I know where the hole was, the hole being the question wasn't asked for main steamline breaks, so it wouldn't have shown up. So I treated it.

There aren't that many events. It's not like an infinite sect that's unknown. I mean, come on, we know that it is ATWS where primary pressure goes up, it's large secondary breaks, whatever you want to call them, and then I even left the large LOCAs in, just in case.

Okay, large LOCAs were kind of considered as part of this, in the old days, but lately I never heard anybody mention large LOCA; however, I looked at one of the vendors outlines for their treatment of Consequential Steam Generator and it mentioned a section on LOCA.

So I didn't want it to disappear, I left the large LOCA in there and I was waiting for somebody to violently object to it so I could take it and nobody violently objected to it, so it stayed there. It didn't really contribute anything.

But my point is we're not looking at a totally new thing here. Come on. We pretty much know what kind of things come in. So, I caught the ATWSs, I caught the steamline breaks, why did I catch them, because I put them there and then I caught the biggest of the LOCAs, not claiming that they are bad or they will cause it or not. But I think I caught them.

MEMBER STETKAR: I guess I have a little problem with that rationale that you thought about a few things and put them in rather than having a comprehensive integrated model that systematically looked for those challenge conditions and let the model tell you what might be important.

Because we've had many examples in risk assessment where things that are very low frequency, potentially very high consequence are not the most important things to risk.

The most important things to risk are the, what I call medium-frequency challenges that aren't necessarily all that obvious. I mean, I certainly can't think of all of those a priori and list them certainly from any type of experience.

So I'm curious about why we're very, very confident that we captured all of the challenges because you've added in the steamline break that was missing and you thought about well maybe ATWS can make pressure high and put in large LOCA, which is not clear at all, but large LOCA is kind of always interesting so I better put that one in.

It seems like kind of a cherry-picking attitude toward selecting bits and pieces of things that you think might be important, which could be an okay process for the purpose of this paper, if the purpose of this paper is only to identify whether or not induced tube ruptures may be relatively important to risk.

If that's the only purpose of this paper, that might be an okay process. If the purpose of this paper is to tell either NRC staff or the industry how to do this, I think I have real problems with that process.

MR. SANCAKTAR: I understand your point and I'm not going to disagree, however I would like to refer you back to my original statements starting with this is a risk assessment of steam generators and there's absolutely no problem with what I called Cadillac methods that are already multiple times shown exercised and discussed.

Each one has its merits and I will not disagree with what you said. And it's all a matter of eventually in what context do you want to utilize things. And I told you what my perspective was, my goal was to provide an answer to some request to review from an NRC point of view, submittal quickly and see if I can hone into important sequences for that plant and put my effort into that.

But I cannot disagree with what you said, it's just a matter of, again, what the purpose is, what's the end product, what are you going to do with it. I have no intention of and in fact, you don't see it anywhere here, of saying everybody out there do it this way or not.

I'm just -- I would like to point out that I noticed many times that when we look at one subject of the bigger picture, we tend to -- it's human nature I guess to be perfect, to do the best job we can do and maybe not consider the other aspects of the issue namely resources and whether it's doable or not.

A specific example I have, without being disrespectful as if you look at the two NUREGs on how to do fire PRA that came out, I mean each one is this thick and they're very comprehensive, they give you everything.

But the world has so many other things to do. I mean we have all kinds of volumes of information being created and we have to find a way to cut through this. I think we created, and I'm telling you my opinion again, this is not a position of the organization, that when I looked at the standards last time, I counted a total number of pages.

At that time it was over 800 pages, I

think it collapsed a little bit. So we basically created a new generation of people who will learn how to understand and deal with the standards, just 800 pages of stuff has been added, and it's good stuff by the way, I mean it's very useful and so on.

But we have to -- so I apply this to intolerance, large steamline break just because it wasn't there otherwise this step should be really there. Internal fire, seismic and other hazards if they apply to the plant.

And I mean, here are the rules that kind of came to the surface, I have no claims that these are complete or anything like that. If people are interested in pursuing this kind of approach one can certainly keep making these things more -- add to them and also maybe explain it further.

So there are rules like based on initiative event. There are rules based on failed systems and so on. So you can tell people if they're not familiar with the situation too much if you see this, this and this. You can even computerize it if necessary.

By the way, all this stuff is in an Excel sheet and then there's like a local database of these probabilities for operator actions and mitigation and

Consequential Steam Generator Tube Rupture that apply to them.

So I was naively imagining was I would put this and there would be a lot of people interested so we sit down and we look at things and we put different numbers quickly and we look at sensitivities, we ask questions and so on and we try to make more hay out of it if there was an interest.

There wasn't interest so there's possible that some of the assignments I made are wrong, okay, I mean it's possible that -- because nobody really sat down and challenged that part of it. So it's possible that you can truly go, if you want you can go through and say ah-ha, you put the sequence in the wrong place, it doesn't change any of the points that I'm making.

The mitigative credits, I think all these things are, in one way or the other you've seen them appear under different disguises what you can be crediting. One thing I want to --

MEMBER BLEY: It's a nice catalogue. I guess, and you've kind of made it clear that the report is almost more your working notes on how to think about this and a tool to sit with somebody and talk through what could be important and this is a catalogue.

How likely these things are to be tried and succeed is very much a plant-specific thing and that's not -- you're not laying out an approach for doing that. You're saying the Sandia approach is perhaps one way to do that.

MR. SANCAKTAR: Right.

MEMBER BLEY: If I'm hearing you right.

MR. SANCAKTAR: Absolutely. The assignment of HEPs to these individual actions is, in my opinion, extremely plant-specific and also more than that, it's also maybe sequence specific, especially for external events.

I mean, you have to -- somebody has to really, really prove to me that after high, high, PGA seismic event they can bring the pump to that location in three hours or two hours. I mean, I would really ask for a walkdown or whatever.

So yes, HEPs, how you quantify the HEPs in here and how many of them can you string together to get a number is where I think this whole thing can, what's the word I shall use, can possibly be, let me say abused inadvertently.

MEMBER BLEY: Fair enough.

MR. SANCAKTAR: Inadvertently abused. So

I was very reluctant to go into that area explicitly. There are end different methods today available to us to calculate HEPs and from anywhere from SPAR-H to THERP to whatever it pre-uses to whatever NRC.

And some of them were suggested by the Sandia report and they calculated some numbers and I tried to stay away from stringing these numbers, especially to these plants because that would defeat my own point by assigning these numbers because I'm saying that I want proof.

I want -- if plant one comes and gives me an evaluation, I want them to prove to me why they can do ten to the minus two on these mitigation actions or so on. So I'm really reluctant to let people choose from a table some HEPs and stick them in there and then multiply them.

MEMBER STETKAR: But in your calculations, if I understand it, you've used primarily a .1 screening value --

MR. SANCAKTAR: A screening value.

MEMBER STETKAR: -- for even scenarios that might have a composite operator action, if I understand.

MR. SANCAKTAR: Yes. You're understanding is absolutely along what I intend.

MEMBER STETKAR: Okay. Which means that the operators succeed 90 percent of the time.

MR. SANCAKTAR: Yes. Don't underestimate it. We have a core damage in most cases. I mean the situation is really not that good, some things happened. So, it is one of the Cadillac methods, problems with the Cadillac methods I see is they're so comprehensive and they are so big.

You can slide in a .1 somewhere and drop a whole thing and nobody will even see it because it dropped. It's very hard to drop something here because initially you focus on them. I'm picking the big ones. I only dropped ones that are dropable with the most conservative approach.

It's very difficult to slide one in here and I really enjoy complicated and intellectually challenging models, however at the same time, I take it with a grain of salt. That includes, by the way, our model for calculating conditional Consequential Steam Generator Tube Rupture probably which is very sophisticated and complicated model.

COURT REPORTER: Sir, please keep your voice up.

CHAIR POWERS: We are now on slide 15 of a package of 45. I'm not sure we're making the progress we're going to need to make.

MR. SANCAKTAR: One of the inputs for Consequential Steam Generator Tube Ruptures risk assessment, regardless of who makes it, whether they use the Cadillac method or anything you want, the conditional -- the probably of the steam generator tubes, given the challenging RCS conditions.

If you look at 1570 there's a table there kind of gave you the basically the numbers that people came up with at that time. And if you look at the Sandia report, there's a Table 4 in there that is kind of -- I didn't score it equivalent to that other table, but it's in the same realm as that based on some key sequences that research has provided to the thermal hydraulics analysts and materials people.

In calculating those -- well, if you look at Table 4 in the Sandia report, which is not shown here, one thing that immediately jumped on -- got my attention was, I saw .4 appearing in many of the sequences.

These are really bad actor sequences so it's not -- I'm not saying it's high, but .4 to me is very close to -- if you ask me what's the probability of something and if I knew nothing about it, I would say it's not one, I can't say it's one, I can't say it's zero, so I have to say it's .5. I mean, that's the closest I can guess if you buy me coffee and want my opinion on it.

So, .4 is not far from that, so it's rather from a PRA point of view it's rather colorless. I mean it doesn't take credit too much, it doesn't give you too much credit, it doesn't punish you unless the number is really off the wall.

These numbers were -- the steam generator tube ruptures were calculated by the support in the Sandia report by using two, basically three types of input.

One input was flow distributions from Gorman report, I didn't mention it here, I'll give it to -- oh, there it is, NUREG CR6521 on page 16, for a moderately degraded steam generator.

Then the second type of input was the thermal hydraulic properties of the RCS/steam generators after the core damage. And the third type of input was what other RCS component fails when, there were basically three different inputs.

Just to show you quickly a couple of pages, one of them is this is the input from what I call the calculator or the Consequential Steam Generator Tube Rupture --

MEMBER BLEY: This is what Sandia used --

MR. SANCAKTAR: Right.

MEMBER BLEY: -- the Excel program?

MR. SANCAKTAR: Right. It is an Excel-based program. It is supported by Crystal Ball, which is just a simulator, it just randomly simulates scenarios, this is one of the scenarios and the green is the input. So for example in this case there are 23 cracks and the crack properties are shown in various columns.

So what happens is you postulate the cracks, you have a bunch of equations that govern the crack growth given pressure and temperature, which is another page, I'll show you that page too.

So you go -- you pick some random numbers, create a situation, you'll put the cracks in various places randomly, you assign them certain properties with respect to initial depth, initial size and so on. Then you propagate the temperatures and pressures and let the crack start growing.

There were a couple of assumptions here. I was not comfortable with these calculations.

MEMBER BLEY: Let me ask you something about that before you tell us what happened with you. One thing wasn't clear to me reading through the 2005

Sandia report, I haven't seen the newest one, and that was -- and I'm wondering did the people who developed the thermal hydraulics work and the materials work review that computer simulation that was put together for the PRA to see if the PRA folks were using that information in the way it was intended. Did we get a quality check on that use?

MR. SANCAKTAR: Well I'm glad I told you before that I started this 22 months ago, so now I'm going to turn back and see if anybody has the history that they can mention.

MR. BRADLEY: Hi, I'm Dave Bradley from NCIC. I've been involved in this program longer than Selim or Jeff so I can comment on this.

I don't believe the folks at Argonne, who did the materials response or Don Fletcher at ISL or the other folks at ISL ever reviewed the model itself. We got together periodically and had technical exchange meetings and I presented various interim status report.

But I'm pretty sure they haven't actually looked at the details of the model, but, you know, when I presented the status reports I was kind of outlining the approach and there was no disagreement at that point.

MEMBER BLEY: Okay. Well given the complexities we heard about yesterday in both areas, I'd be a lot more comfortable if they were confident that the work they had done is really properly represented in how it's used in the PRA. And I'm a little disappointed to hear they didn't, unless those meetings really got to the crux of it so that they were pretty confident about it. But, yes, okay, thanks.

MR. SANCAKTAR: There are two -- I wasn't comfortable with the numbers. I wasn't comfortable with the details, however, something gave me the evidence to be comfortable, namely the resulting number for the worst sequences was so high that there wasn't -- you didn't have anywhere else to go in the upper direction, you could really go down. I mean, the upper direction is one, we are getting there almost.

And the other thing is the assumption here was you have like 23 cracks that are growing in various locations and we have to assign a point where things are unacceptable. I mean, what is the definition of, what is the criteria for?

You're saying, okay, I declare bypass and suppose only one crack grows so you're spitting out one drop every 30 seconds. I mean, are you going to call that containment bypass. So the criteria suggested by Sandia is, I'm going to tell you what it is, it's also on one of the slides, I looked at it and I said, that's not unreasonable, although we can sit here and argue the heck out of it, okay.

And please correct me if I say something wrong, but they are basically, their criteria was an aggregate opening that will relieve all the RCS inventory in four hours is the point where you say this is containment bypass and that aggregate area, integrated total area not for one steam generator, not for one crack but overall.

It can come all from one or it can come from a hundred different little ones. It was calculated in the Sandia report and I adopted it, you know, it's .081 inch square. Just to put this in perspective, one steam generator tube, magic number for that is approximately .4 inch square.

It depends upon your radius and newer generators have smaller radiuses and older generators have bigger, some of them are bigger, but .4, .5 is approximately the magic number. So this number, .081, is not a one guillotine break of a steam generator. I mean it is teeny-bitsy is a total, okay.

I'm not arguing against it or for it, I'm just stating to you what was done. Once I realized that, I said okay, we can proceed, but in the meantime I would like to inform you that since the -- we have completed documentation and gave you based on the anticipated NRR need, we are actively pursing fine tuning, documenting and using as a forcing function this tool.

We have already done a lot of work and in getting this tool well understood, document what we understood and then say what else we can do with this tool. For example, I'm really, really interested in putting a very strong correlation related failure of pressure induced bursts, let's call it.

It's not putting .05 or .005, can we do it so it's also in the same realm of the depth of detail of other things we did. We want to see if we can receive more recent and possibly plant specific crack data. But on the other hand, I still don't think that -- you can only go so far with these annotated calculations, there's a point where you have to stop.

CHAIR POWERS: I'm going to declare a break and I'm going to inject now my personal view is I have lost the thread of this conversation. I have no idea where we're going and it's not clear how I'm even going to get there.

So in the 15 minutes that we have a break maybe you can figure out a strategy to get me back aligned on where we're going and how we're going to get there. So we'll resume at 10:15.

(Whereupon, the foregoing matter went off the record at 9:59 a.m. and resumed at 10:16 a.m.)

CHAIR POWERS: Okay. We're back into session, we're going to have a little recap to realign me so I'm not lost and then we'll move to some of the specific examples. I believe, Mr. Stetkar you have some specific questions.

MEMBER STETKAR: I do.

CHAIR POWERS: And we will get to those when we go to the examples?

MEMBER STETKAR: I want to -- I have one question on this spreadsheet calculation actually.

CHAIR POWERS: Okay.

MR. SANCAKTAR: Which spreadsheet?

MEMBER STETKAR: The Sandia spreadsheet that you got up there.

MR. SANCAKTAR: Oh, okay.

MEMBER STETKAR: But it's a relatively simple question that I hope someone can answer very quickly. In the report, if I read the report correctly, it says that you run the Monte Carlo analyses and that if the time to develop the critical crack size is less than the time for either the hot leg or the surge line rupture, that is considered a failure, containment bypass and you run the Monte Carlo Analysis for large numbers of samples.

The report says that the conditional tube failure probability is simply the number of those samples that resulted in a failure divided by the total number of samples that was run. Is that the actual way that that probability was calculated?

MR. BRADLEY: Hi, David Bradley again, NCIC. Yes, that's the way the probability was calculated.

MEMBER STETKAR: So if you ran 1,000 samples and 400 of them showed failure, the conditional probability would be --

MR. BRADLEY: It's .4

MEMBER STETKAR: That's wrong. It's mathematically incorrect. And I don't want to take the time -- I just want to go on record that it's mathematically incorrect because it does not account for the probability of that sample. Each sample has an associated probability that is the correct set of conditions.

So for example, sample number one might have a 1 percent probability that those conditions exist in the real world, and if it fails, there is a 1 percent probability in the real world that that is a failure. I don't want to go into a lot of the details we can talk about it later if you'd like to.

MR. BRADLEY: Yes, I would like to do that.

MEMBER STETKAR: But that process of simply counting up failure bins and dividing by the total number of samples is mathematically wrong. So if that's the process that's been used, there's kind of a fundamental flaw in that whole spreadsheet calculation.

MR. BRADLEY: Okay. Well I would like to discuss this with you.

MEMBER STETKAR: We can do that offline later --

MR. BRADLEY: Okay.

MEMBER STETKAR: -- because it will take too much time. I just wanted to make sure I understood that the process was actually done according to the way the words and the report described it and you've confirmed that, so thank you.

MR. BRADLEY: Okay.

CHAIR POWERS: Okay. Let's resume the presentation.

MR. SANCAKTAR: Okay. What I propose to do is jump to slide 36 just to recap, recalibrate ourselves and then come back and maybe go through some examples and so on, keep moving forward. This is a slide 36 and this is an attempt to recalibrate ourselves back to basics.

For the last 10 to 15 years this subject has been studied by NRC and the industry as evidenced by various reports I have mentioned. And again, yesterday you've seen some of the presentations the depth of understanding and state of the art efforts that you can judge yourself to be sufficient or not, but I think the conditions and sequences are pretty well studied and understood.

There are -- as everybody recognizes and I certainly subscribe to the same school of thought that there are large uncertainties and they really go to various disciplines, fundamental principles and a reduction of these uncertainties is not achieved yet and in fact personally I think it's going to be very, very difficult to satisfy everybody in that area.

We have here an illustrative PRA estimate of the bypass frequencies for selected plants that consider internal and external events and they use a straightforward, easily reviewable estimation methods. Systematic methods, I have to emphasize that, it's not a haphazard method and its assumptions are well stated, they are simple and they're open to scrutiny and also they're open to sensitivity analyses.

And this method is prudently, conservatively exercised on purpose, although the word conservative may be taken as the bad word in PRA, but in this particular instance it's important to consider that.

So I'm going to return back to earlier, steer you back to the beginning, but I just wanted to make sure the analysis method that are described here and outlined here is a systematic method. It has rules are well defined and can be expanded easily, its input can be changed rather quickly and the results can be studied.

MEMBER SHACK: Perhaps coming back to Dana's question again, for whom is this method intended, for what purpose is it going to be used for?

MR. SANCAKTAR: Well, I'll tell you --

MEMBER SHACK: Is this something the staff is going to use to kind of do rough estimates to check somebody?

MR. SANCAKTAR: I'll tell you what I thought of this and we'll also find out what NRRs position on it is when we see the user need. I thought I explained it before, so I'll just repeat what I said before.

When I looked at tasks, the seven subtasks, what is this for? Are we telling the licensees to do it some way, a certain way, are we telling our internal PRA people to do PRA analysis, what is this for?

And I thought that the most practical immediate objective that is of use to the NRC frontline is if they receive analyses from the licensees supporting and application and it involves an evaluation of Consequential Steam Generator Tube Rupture and we would like to quickly and efficiently in a comprehensive manner review it and then if it's limited look at its additional pieces that should have been there may not be there like external events.

Just analyze it to the point of efficiency without going into theoretical expression of the existing PRA model. So there's no say that thou shalt do it this way. If they do it this way, by the way, the utilities, just to give us an assessment, that's fine.

But the point is, if they do it this way, that assessment works, that PRA model at that time, if things change and the assessment results will be changing, should be redone. But the effort it is still manageable. I don't know, did I answer your question or not? I don't know.

MEMBER SHACK: Yes.

MR. SANCAKTAR: Maybe I'll just go to the examples, unless you want me to --

MEMBER SHACK: Can we just go to the statement on page 18, you know, I heard it before and now I see it again in print. We're going to empty out the RCS in less than four hours with .1 square inch?

MR. SANCAKTAR: Yes, that is the calculation I accepted from the previous --

MR. BRADLEY: Can I comment on that?

MR. SANCAKTAR: Yes.

MR. BRADLEY: Hi, this is Dave Bradley again. I wanted the point to be made that this calculation is simply a placeholder. We were waiting on calculations that would involve more detailed analyses such as what has been done recently with the TH runs that were done by Don Fletcher.

But this was a quick calculation of something that at the time I thought made sense, I'm not sure I believe it anymore, and so I don't want you to focus too much on that. It's nothing more than a placeholder. I think if I were doing the calculation now, I might end up with a much larger area.

MEMBER BLEY: Was that made clear in your report? I'm trying to remember and I'm not sure.

MR. BRADLEY: In the report, yes it should be clear that it's simply a placeholder.

MEMBER SHACK: Okay. So all you're really saying is that you should compute the flow through here and we'll have some acceptance number at some time?

MR. BRADLEY: The flow should be completed using more detailed analysis, as I said, similar to what Don Fletcher has done more recently. And, you know, Don looked at failure of large numbers of tubes with large area, found out that it didn't prevent hot leg failure, so it comes down to a decision about what constitutes containment bypass.

Do you get a failure that releases something, outside containment but then stops? So it's well beyond the capabilities we had when that calculation was done. So again, it's just a placeholder.

MEMBER BLEY: While you're here, you -- at the time, and I've only read in any detail the 2005 version of the report, but at that time, this idea that even if you had a tube rupture there would likely be a larger failure that would depressurize the RCS to the containment, that wasn't on the table or if it was, would you have modeled that in your PRA?

MR. BRADLEY: Even if you have a tube failure that there would likely be --

MEMBER BLEY: Right now when you do your calculations, if the tube ruptures before anything else you count it as a failure?

MR. BRADLEY: You count it as a failure, right.

MEMBER BLEY: Now what you know about the sequential likelihood of failure of the hot leg, even if you fail the tube rupture, would that have been built in --

MR. BRADLEY: But again, it comes down to the -- what your definition of a containment bypass is. Is it a LERF, is it something that's large enough and there's a long enough time period before one of the other ruptures that you get a large release outside containment or what? How do you define that?

I mean you get a release, but it may not be a very large release. And I think Selim has talked about, well it's a containment bypass maybe not a large early release. So it kind of depends on your definition at that point.

MEMBER BLEY: Okay. Good enough

CHAIR POWERS: Dave, when you made that calculation what did you assume, the speed of sound?

MR. BRADLEY: Yes, choke flow through the opening. And basically it was done, if you have no other failures, if your tube failures are enough to prevent hot leg failure then you've got a release to the environment and I just assume a period of time.

MEMBER STETKAR: But you said you think, given the current information, that critical flaw or that critical size would be probably somewhat larger than what's used in these current calculations, is that correct?

MR. BRADLEY: That would be my guess, but again it gets back to the definition of what containment bypass is.

MEMBER STETKAR: I guess that leads me to the question about what the conclusions of this risk assessment may mean, because I originally thought that the purpose of doing this was to provide some input about whether the issue of induced tube rupture could be numerically important in the sense of risk assessment.

You know, in other words, is the frequency of induced tube ruptures, you know, 100 times higher than the frequency of any other containment bypass that we currently model, is it 1,000 times lower, which means we ought not to be too concerned about it or is it comparable?

And the conclusion is from the current numbers that it's comparable, but that conclusion is strongly based on this particular size assumption because the time to grow to a much larger size I would assume would be somewhat longer than the times that they're calculating in this little spreadsheet. Is that correct from a materials perspective?

MR. BRADLEY: Let me -- Dave Bradley again. I did run some calculations, because I knew this subject would come up, where I looked equivalent of two tubes, again with some uncertainty distribution and it lowers the probability but from .4 to .2.

Because what's happening is that you have a lot of flaws that are ready to go and they're happening in sequence over relatively short period of time, plus the flows that you do have are growing. So it -- you get containment bypass according to the larger area, it just happens a little bit later, not as delayed as you would think as I thought initially. So it lowers the probability but not by a large margin.

MEMBER STETKAR: Okay. That's helpful. Thanks.

MEMBER BLEY: Let me ask you one last question and then we'll go back to Selim, and that is since we did just get the 2008 report, can you in just a couple of sentences tell us the primary differences between 2005 and 2008?

MR. BRADLEY: Formatting.

MEMBER BLEY: No substantive differences?

MR. BRADLEY: No.

MEMBER BLEY: Okay. Great. Thanks.

MEMBER SIEBER: It seems to me that this whole process is self limiting. If one tube rupture pressure starts to decline, but not very rapidly once you get the second tube or the third tube there's a susceptibility of the remaining tubes is -- becomes lower and lower.

MEMBER SHACK: The other thing, you know, I mean since Gorman report was essentially more of the Argon research, kind of hard to dump on it too much, but --

CHAIR POWERS: Not for me. Only you.

MEMBER SHACK: No, I would probably all agree it's woefully out of date and you really need a better approach. The people who are doing 97-06 assessments must come up with distributions and it would be interesting to look at their distributions and whether, you know, those are appropriate -- well, at least in our plans them acceptable, they may not be appropriate but they're regulatorily acceptable.

So that might be some place to look for a different approach to coming up with a flaw distribution, which, you know, will be certainly a consequential input to this result.

MEMBER SIEBER: I sort of gathered that the purpose of doing this kind of a report was to figure out whether you needed to study it in greater detail or not. In some place along the line somebody ought to tell us what your conclusion is.

CHAIR POWERS: Well I thought the conclusion was that it ought to be considered in the PRA. I mean, I thought the conclusions were excellent. Now how we reach those conclusions are little bit of a mystery to me, but we're going to go through --

MEMBER SIEBER: We're struggling at the moment.

CHAIR POWERS: We'll go through the examples and I presume that it will be a little clear how we got there, but I thought the conclusion was pretty straight forward. Yes, it needs to be considered.

MR. SANCAKTAR: You're getting kind of glimpses of in the details of what the small pieces of information that kind of gathered together and support the conclusion. The conclusion is not reached by one definitive killing blow, like he just made a statement that he ran another sensitive case with a bigger opening. He didn't really change things too much in a factor of two in this case doesn't really change the conclusions.

So there are all kinds of little pieces of information that accumulated. Each one looks small and it's hard to keep track of them, but --

MEMBER SIEBER: I think that's part of my difficulty is assembling a lot of little things, particularly when the conclusion is stated up front saying, you know, and then supported by bits and pieces, I'm struggling a little bit and perhaps I need to do more homework.

MR. SANCAKTAR: If it is okay with you --

CHAIR POWERS: Please.

MR. SANCAKTAR: -- I will continue from Slide 22. I didn't want anything to go through all the examples, I just put them there and I leave it to you to tell me.

But this table taken verbatim from the report and it's for one of the plants, I don't even know plant one or two, but definitely not three. So this kind of says we are dealing with Loss of Offsite Power as the first sequence in the -- from the PRA.

The sequence core damage frequency comes from the PRA. The sequence description comes from the PRA. Aux feed status is inferred from the sequence, namely the battery depletion. The SPAR models are set up so that they tell you explicitly what the assumed seal leakage is in the CDF part of the sequence.

Once you go beyond that, how the seal fails is another story and you can see there that there are basically three numbers. The first number here is the CDF frequency, this is the operator actions that will recolor aud feed in different forms and it's kind of buried in these R1 and R3, they are different recovery modes, they can be applicable.

And then conditional Consequential Steam Generator probability is assigned to this sequence as 2 percent. Basically, I used three different values for -- in this illustration for conditional consequential steam generator probability. One is severe challenges I use .4, could have been .33 or .25. It's -- I'm not selling the number, you know, it's a high number.

The other one is medium challenge, that's what this is, .02 and then one is really insignificant. Yes, there might be one, but it's insignificant and I think I used .01 rather than zero.

So like this case, the product of the three numbers, three times ten to minus six time .1 times .02 gives you the bypass frequency which is six times ten to the minus nine.

You can immediately see here that one can do lots of things with this recovery. You can easily defend, analyze and defend for a plant-specific situation operator actions that might be successful 99 percent of the time.

Here I use a screening value that operators will be successful in 90 percent of the time. I'm not adverse to people calculating and using HEPs and multiplying HEPs as long as plant-specific analysis is provided and it passes the test of common sense or prudent use of numbers.

So basically if you keep going down this way I have used this kind of approach on three plants. The first one is a four-loop Westinghouse plant by itself on a -- it's a single-unit site, so it doesn't -- I tried to eliminate the other units helping out so that it can mask certain things.

You know, here it's one unit by itself it has a sole support systems, it doesn't share anything with any other unit. It's a newer unit.

MEMBER BLEY: It's a what?

MR. SANCAKTAR: Newer. Newer generation in the sense of nuclear plants being newer of course. The second one is a two-loop Westinghouse plant. By definition it's an older unit on a single site again -- I mean, single-unit site. The last one is a Combustion Engineering plant on a newer site with multiple units.

I had external event sequences for these, that's one of the reasons, well for the first two I mean, I had external event sequences. So that's why I chose the first two for reason. The last one I didn't really, for the purpose of this study, I just threw in a few seismic events just to show the impact of external events possibly limiting credit for mitigation. Want us to be more careful in giving mitigative credits in external events.

CHAIR POWERS: When you considered external events, particularly these seismic events, your consideration was on the availability of equipment and water resources you did not address the effect of the seismic event on the tube itself?

MR. SANCAKTAR: Right. Not on the tube itself. That's correct. That's a good point, yes sir.

I took the following tables from the report just to discuss and illustrate the model and its results. This is a one sequence from plant one, loss of offsite power occurs, station blackout also occurs, which means in this case diesel generators fail and there is no other AC power source external to the site for this plant.

Auxiliary feedwater fails early. What that means, of course turbine-driven pump because motor driven pumps are already gone. AC power and diesel generator recovery in one hour fail. That's one of the standards station blackout sequences that pop out and that's it.

I mean I couldn't give mitigative failure probability. There isn't enough time. This is a challenging sequence, so I gave it the highest conditional steam generator tube rupture probability and the product is seven times ten to the minus seven. So it's like -- well, it's 40 percent of the original.

CHAIR POWERS: -- sequences where you have a mitigation probability --

MR. SANCAKTAR: Probability?

CHAIR POWERS: A mitigation capability. The operator can do something to protect the tubes. When you assess that probability, you assess that probability based on the probability of the operator doing the right thing. You don't have a criterion that says, he did the right thing, but it caused all the tubes to thermally shock or something like that.

MR. SANCAKTAR: Right. There is no --

CHAIR POWERS: You have no --

MR. SANCAKTAR: Right.

CHAIR POWERS: -- thermal shock capability?

MR. SANCAKTAR: No.

CHAIR POWERS: No.

MR. SANCAKTAR: See, those kind of --

CHAIR POWERS: You are happy?

MEMBER BLEY: No.

CHAIR POWERS: Why are you unhappy?

MEMBER BLEY: I'm not completely unhappy, but if this is a good time --

CHAIR POWERS: Yes.

MEMBER BLEY: -- I can go through a few things. I think what Selim's shown us is probably a pretty good, better than seat of the pants way to check. In fact, you know, this report's purpose is to even provide some guidance to regulators to take a look at what licensees submit and do a sanity check and then if things look a little funny to dig deeper. I can see it's value.

The report itself, if I may, is I think a long way from being that kind of guidance and for the reasons John said earlier, it's probably not guidance on how to do a PRA, it's guidance on how to take a quick check back, on page 10 of your report you give a little summary of what's coming and why.

There are just some, kind of to me, unjustified conclusions in there and funny language like the one you brought up, the realistically conservative kind of thing. There is a treatment of uncertainty you haven't gotten to yet that I don't see that it adds much.

I think uncertainty is important here. I think the SAIC report, I'm sorry the Sandia report identifies the really key areas where there's some holes that need to patched or examined on uncertainty that gives some pretty good justifications for the upper bound and the fact that the lower bound might be way off.

The things in this report on uncertainty just, you didn't provide any real basis for where they came from, they don't preserve the mean.

MR. SANCAKTAR: I agree with you.

MEMBER BLEY: They just come out of nowhere.

MR. SANCAKTAR: I agree.

MEMBER BLEY: Okay. So that's not a treatment of uncertainty from my point of view. So, my own happiness is it's not what I was thinking it was when I picked it up, how to do the PRA and get it right. It's a good first cut at how to do a reasonable check to see if they've got the important things and if they've missed anything. So that's why my look was a little funny.

The work on human factors in here is -- I mean in fact all the numbers kind of get picked out of the particular case that was done by Sandia and reasonable assumptions made about their applicability, quantification of any of the human reliability work is very class-specific, procedure-specific, training-specific, sequence-specific and none of that's picked up.

There could be cases where these numbers are wildly optimistic as opposed to being bounding values. So it's not a PRA, but it seems to me it's good to have a way to take a look at what people claim and see if they're any real gaps in their thinking and I think it would serve a real good purpose in that way with some cleaning up.

So that's what my face was about. I'm sorry, go back to where you were.

MR. SANCAKTAR: That's okay. Yes. Okay.

MEMBER BLEY: You don't seem to disagree with any of that?

MR. SANCAKTAR: Not really.

MEMBER BLEY: Okay.

MR. SANCAKTAR: This is the one sequence, it's got a very simple sequence, but makes the point one can -- if you look at this, you realize that you can only do so much in those two multiplicative factors. It kind of focuses everything so that you can see how far up or down you can go.

You can see there may be no need to have a detailed pre-processing in many of the cases. And if those cases already pretty much determine your outcome for that plant, you're done. If you're not done, then you can pursue the detail of interest, I mean to the detail necessary on the sequences of interest, like this sequence. It could argue so much for this sequence, so one much can do.

The next one for the same plant, same kind of situation, station blackout, aux feed is okay initially, okay being turbine-driven pump is running, rapid secondary depressurization okay, which is on the emergency procedures; it's kind of ironic in this case, this was brought up yesterday, 21 gpm small LOCA is postulated, it was a gentlemen's agreement.

I don't disagree with it by the way, I mean that is the right number that's accepted today. AC power and diesel generator recovery eight hours fail. Again, this will lead to as mentioned in the T&H analysis section of the NRC presentation, these are the ones that are really, the 21 gpms are the more challenging conditions for those tubes.

So the Consequential Steam Generator Tube Rupture probability is .4, maybe it's .33, maybe it's .25, but it's high. Mitigation failure probability is additional equipment that can be brought and put in and effect the B5B following B5B or actually common sense.

That probability can be, depending upon the plant-specific situation, can be anywhere from ten to the minus two to ten to the minus one. In this case there's a screening value of .1 and that sequence is one of the contributors to the Consequential Steam Generator Tube Rupture frequency.

Two-loop plant, as you notice the loop and station blackout sequences normally rise to the top. In fact, if you look at the Sandia PRA report, they focused on that with good reason.

I mean, the station blackout pretty much gives you the quickest path to losing -- getting both sides of the steam generator into unfavorable situations and preventing immediate response. So they rise to the top actually.

You'll see a similar situation here, different numbers, similar situation. But let me show you an external event on page 28. This is on the two-loop plant. This is relatively high seismic event, high-intensity seismic event and the sequence CDF is four times ten to the minus six.

And this is attributed to the large structural failures that would render an unspecified number of safety systems inoperable. So I look at the details and 35 percent of this is already attributed to bypass. I mean it's already number 3.8 minus 634 percent of it is already going out. So you can't double punish it, it's already bypass.

So out of the remaining one, mitigation failure probability is not allowed because of the intensity of the seismic event and it is conceivable that the equipment, some of the equipment is stored in places where the building itself will not -- doesn't have a ghost of a chance of surviving a seismic event like this.

In fact, I saw one where it wasn't designed for these purposes and although designated pumps were there. I treated this as an unanalyzed situation. It doesn't really change anything .5 or .4, but --

MEMBER STETKAR: Would it change much if it was one?

MR. SANCAKTAR: Not really.

MEMBER STETKAR: Not this particular sequence obviously, but large numbers of these, would it change much?

MR. SANCAKTAR: No.

MEMBER SHACK: Let me ask a question. Your conditional SGTR probabilities, you're just sort of doing those out of your head, right?

MR. SANCAKTAR: No.

MEMBER SHACK: This is coming from the Sandia calculation? You're actually doing the calculation?

MR. SANCAKTAR: No. Neither. As I mentioned, in the Excel sheet for this not for the -- there are two Excel sheets here, one of them is the calculator for the conditional probabilities, okay, leave that aside for a moment. The Excel sheet here there is a data portion where you can define different -- you can place different probabilities for different sequences.

And if you want, you can put one number for each type of sequence, you can do any detail you want. At this point, I specified three types of conditional or damage probabilities, which for sequences that are highly challenging to do in steam generators as defined by the thermal hydraulic conditions.

So I examined the output of the Sandia report for those kind of sequences, I also looked at 1570 for what kind of numbers they have for those kind of situations and then I use a number, .4, you can say came from the Sandia report as the most challenging number or you can say that it's actually very close to a number from 1570.

They're not counter to each other, I could have claimed that I'm going to pick the number from 1570 and I would have used .33 or something. So, three types of sequences. It doesn't have to be three, it can be five, it can be 75, it's just a matter of detail you want to use.

So, second one is more moderate challenges and the third one is more negligible challenges. So, for each sequence assigned either one of those attributes, either say this has a conditions that look like thermal hydraulic situation would be challenging or medium or negligible.

MEMBER SHACK: Well, yesterday and maybe I'm oversimplifying the thermal hydraulics, but to me it looked like the thermal hydraulics was saying that, you know, if the loop seal cleared, bang, the thing was gone. If the loop seal didn't clear and I had a perfect tube, I was okay.

So are these calculations driven by your estimate or your examination of when they think loop seals clear, is that how you decide a severe?

MR. SANCAKTAR: No. The loop seal clearance as --

MEMBER SHACK: Well they have some

conditions where they think loop seals will clear.

MR. SANCAKTAR: Right. Loop seal clearance, for example might come after the 440 per gpm per pump leakage and those sequences are flagged in this list. If you go down the list of dominant sequences, you'll see, you'll hit the place where it says I have 440 per gpm per pump.

So if you hit there then you can say that, oops I'm going to assign it a one if those conditions occur, so, yes one of the numbers is one that you can --

MEMBER BANERJEE: What about below that? I mean if the pump was 21 gpm or something. And there is a possibility that the loop seal will fail.

MR. SANCAKTAR: Yes, possible. You can identify that.

MEMBER BANERJEE: Did you assign some probability --

MR. SANCAKTAR: No, I didn't.

MEMBER BANERJEE: I don't know how much you can rely on these types of calculations, so you probably should.

MEMBER BLEY: Just a separate thing, you brought up the issue of the data that you have in your spreadsheet calc, one thing I noticed if this were guidance for other people it doesn't -- it says you can use the generic data you have or you could develop your plant-specific.

And it strikes me if one were to give people guidance like that, I'm not sure who the people are, you'd want to be sure that the generic, so-called generic data you have would apply to the I'll call it the worst plants, the vices where perimeters would be skewed in the worst ways such that if people decided not to use plant-specific data that would be -- if they weren't the plant they'd be getting penalized for it and I didn't see that kind of thinking evidenced anywhere.

MR. SANCAKTAR: I thought that these numbers were pretty bad already.

MEMBER BLEY: Okay. So you think it is? You didn't talk about that, fair enough.

MR. SANCAKTAR: The other one is similar too, the other sequence.

MEMBER BANERJEE: There was a question here as to what about the B&W plants, is there, I mean --

MR. SANCAKTAR: They are once-through and they were not specified in the scope.

MEMBER BANERJEE: They were not specified.

MR. SANCAKTAR: But we would be willing to consider -- I mean, there's no need to consider their once-through and SSUs.

MEMBER BANERJEE: Well the once-through being -- the issue being that you'd probably fail the steam generator tubes anyway.

MR. SANCAKTAR: Right.

MEMBER BANERJEE: For sure.

MR. SANCAKTAR: Yes, I mean it's not -- no. So if you look at this slide, you get a bunch of numbers.

MEMBER ARMIJO: Which slide number is that?

MR. SANCAKTAR: This is 33.

MEMBER ARMIJO: Okay.

MR. SANCAKTAR: You get a bunch of numbers and so one would -- I ask myself, so what's the criteria, I mean, what are we going to say. Is this a big number, a small number? What is the threshold, what are we looking at to decide on something, if these were final numbers for the sake of argument.

We cannot just look at purely the exponent, you know, it's high or low, it changes from plant -- this exponent changes from plant to plant, moreover, it changes disadvantageous, seemingly disadvantageous if you look at external events. So somebody who would conscientiously do all the possible events but has categories will have a higher CDF of course.

So what is the criteria? Well for comparison reasons one criteria that always, I thought was a pretty robust one was if your CDF is f, my pass should be, no, should be at the order of .1f. I mean if you design a new reactor and give it to the NRC for review you have a CDF of ten to the minus seven and your bypass if five times ten to the minus eight I think, there will be lots of questions.

So, in general if your CDF is f and your bypass is releasing LRF, okay, it might be LRF, L-R-F, Large Release Frequency, but if your bypass is -- if your containment is 90 percent effective and effective in holding up, leaving without passing things either through the steam generator or through the containment or through the basemat, whatever, .1.

So compared to that, this total plant CDF here is 5.8 minus five for all these events and for this illustration they calculated Consequential Steam Generator CDF is 2.8 percent of it. So even if this were purely LRF, which I'm not saying it is, it's not -- it doesn't get one attention.

Look at the next plant, plat two, older plant. The ratio is 10 percent, 10.6 percent. This is at the point where it's okay, you know it's not too much, it's no too little. So I'm trying to give you a sense of some measure, okay, whether you agree or disagree, but I'm looking for --

MEMBER STETKAR: Selim, let me just ask you a question about this too. You mentioned this is an older plant, two-loop plant and the implication is that this is worse than the newer four-loop plant.

If you go back to the first slide, one thing I noticed was that the -- for example, the contribution from fires on -- at this particular plant is quite small it's .6 percent and it seems inconsistent with general experience.

For example, fires at many plants tend to lead to core damage sequences that tend to look an awful lot like station blackouts and tend to look like some seismic events. And the fractional contribution from those events at this plant are all kind of in the same ballpark, except for the fires. The fires at this plant is really small.

If you go to the other two plants that you did, the fires and the station blackouts and the seismic all line up, which leads me to believe there might be something strange about the fire scenarios on this plant that is artificially suppressing the contribution from fires here and that the difference in the relative contributions might not necessarily be the vintage of the plant and the number of loops, it might be some artifact of the particular fire model that's in this plant or the way that it was used.

Do you have any insights on that? Do you know why the fires on this one are such a relatively small fractional contribution? Not the absolute value, the fact that they're only .6 percent rather than something in the ballpark of about 5 percent something around 10 times fires.

MR. SANCAKTAR: I totally agree with you.

MEMBER STETKAR: Okay. Thanks. I just wanted to make sure I understood it because I didn't want necessarily to be confused by the fact that the four-look plant by definition was better than the two-loop plant.

MR. SANCAKTAR: Right. One should not infer from these numbers conclusions like this is the two-loop older plant so the number is bigger than the other one.

MEMBER STETKAR: Okay.

MR. SANCAKTAR: One should absolutely not infer that.

MEMBER BLEY: Kevin's got something to add.

MR. COYNE: Yes, Kevin Coyne, Office of Research. Just on the four-loop plant, the .6 refers to the contribution to Consequential Steam Generator Tube Rupture from that group of sequences.

If you look at the CDF for internal fire, it's actually 3E-5 compared to the total plant CDF, which is 5.8E-5. So I think it's in the 50 percent range contribution from internal fire. The table might be a little misleading, but I think the .6 is contribution to bypass.

MEMBER ARMIJO: Why is that so much --

MEMBER STETKAR: That's the number that I was focusing on was the contribution to bypass. If you look at the other fractions and we've lost the slide and I've lost my slide, if you look at the other fractions from the contribution to bypasses are on the order of about 5 percent.

The contribution, the fraction of the core damage frequency from that category that goes to bypass, for example for internal events at power is 4.6 percent, for internal flooding events is 5 percent, for seismic events in Bin 1 is 4.9 percent.

All of those events usually behave pretty similarly in terms of the PRA model, but it's only .6 percent for fires. Only less than 1 percent of the fire events that go to core damage also go to containment bypass. And that's the fraction that I was looking at.

MEMBER SIEBER: If you compare slide 35, which is combustion plant to the Westinghouse plant you'll find missing pieces there. And I think that you have to look at each plant as a specific analysis and not look at plant A versus plant B versus plant C.

MR. SANCAKTAR: Absolutely.

MEMBER SIEBER: Because there are pieces that are missing, the techniques differ from plant to plant and the relative answers, I think, are similar so that one could draw a general conclusion from it, but specific comparisons just don't work on these charts, nor in the report.

MR. SANCAKTAR: Absolutely. For example, if you go to the report and look at the tables for the fire scenarios, you'll probably see the following, I'm hazarding a guess here without looking at this moment, I don't remember, but I'm going to hazard a guess.

I would hazard a guess that in one plant they put -- they punished themselves more for main control room fires, which force you out and then you can either create your own station blackout, you know, solving this station blackout or get into one and the other plant might have put less emphasis on it.

These are not done -- these are basically IPEEE-vintage scenarios and they are not done by according to a perfectly standard set of assumptions. I tried to normalize them a little bit.

MEMBER SIEBER: Yes, well there's other differences that show up. For example, two-unit sites with inter connectable diesels have a lower probability of station blackout than individual units, and there are some plants with five diesels, six diesels.

MR. SANCAKTAR: So this kind of gives us an idea about what kind of information can be skimmed out of these things and I wouldn't generalize them and immediately run and say, oh, I am going to generalize this, the whole class of plants. However, one thing I would like to point out is that external events contain a maybe more of a proportion of station blackout scenarios than internal events.

So you would expect that their contribution to Consequential Steam Generator Tube Rupture would be not something that you can ignore with it or the paragraph of pressure and qualitative, dismiss it qualitatively.

MEMBER STETKAR: One of the reasons that I asked some of the questions earlier on was, for example, if you look at line item 4 on this slide for this particular unit, you note that about 61 percent of the very large steamline break scenarios that go to core damage also result in containment bypass, however it's defined, which says that if I'm interested in looking for potential containment bypass scenarios, I better be pretty careful about looking for those depressurized delta-P type failures because they apparently have a pretty high chance of rupturing tubes, which is one of the reasons I was asking the questions earlier about how careful are we in terms of searching out those depressurized large breaks or other ways that we could have a DP-induced failure.

MR. SANCAKTAR: Yes, but --

MEMBER STETKAR: Because that is another message from this that if we believe the tube -- the steamline break analyses, those models, that is the place that I need to be pretty careful that my models are reasonably complete I would say.

MR. SANCAKTAR: Yes.

MEMBER STETKAR: Is that an appropriate

conclusion?

MR. SANCAKTAR: Well, I wanted to make sure that the steamline break didn't drop off the cliff because of a conditional steam generator tube rupture probability of .05 or .005, something small, which may be true. But I didn't get enough -- I could not accumulate enough evidence to support that.

And in fact, that's one thing I wanted to make sure that we -- we don't have to analyze it or find new information if there is already evidence somewhere if I didn't catch yet.

But this, I wanted to make sure that this stays there and that makes -- what makes it 87 percent is I use an unanalyzed conditional steam generator tube rupture probability of .5 or whatever, .4 for the highest for very large steamline break.

So now it looks like an eyesore and you can't miss it. You have to make sure that we have crossed the T's and put dots on the I's before we are totally out of the woods.

MEMBER STETKAR: Okay. Thanks.

MR. SANCAKTAR: We pretty much discussed all this, I don't think there's anything new here, just quickly go over it. I would have been very happy to satisfy the need and finish this off by saying steam generator, Consequential Steam Generator Tube Rupture is no problem for anything, just forget about it, most people will be very thankful.

Or, if I could say that this is really a big problem, you know, we needed to do something, it would have really helped everybody in one way or the other. I mean, it would have jumped to the conclusion. But I don't think -- I could not find enough basis to push for one or the other.

CHAIR POWERS: It's in the pack. I appreciate your pain here. It's, you know, it's like kissing your sister; it's neither here nor there, but it says that -- it says what the PRA standard says, you got to think about these things.

MR. SANCAKTAR: Yes, think about it. And the major contributors that is already identified, there's nothing new here, are basically core damage sequences like station blackout and so on, already core damage is considered and we are going to fail the tubes are the main contributors.

Technical recommendations, this is the next slide. Plant PRAs should address Consequential Steam Generator in their evaluation of plant LRF or Level 2 analyses on a plant-specific basis, I thought be done on a plant-specific basis if possible, referring to the existing PRA standard.

However, an in depth and intrusive modeling within the Level 1 models is not necessary in my opinion. It would be nice, but if it's costly and the plant pretty much convinces itself that there might be -- they are pretty infrequent, they can use other methods.

However, if you take a shortcut, it's a pay me now or pay me later situation. If you take a shortcut then every time you submit something you have to make sure that you evaluate it according to the latest.

I have to say this, this last bullet for various reasons, I hope that people do not take excessive credit for recovery without finally demonstrating that it's feasible and it meets the characteristics of the sequence.

There are reasons why I say this, I don't know how to -- I won't say anything else about it.

COURT REPORTER: Sir, please keep your voice up.

MR. SANCAKTAR: If you really want to look at the numbers part of this, you know, if you're focused on numbers, those people who like to focus on numbers and look at this, the numbers that you see here in my opinion are on the conservative side. Okay.

They are off, they are off to one side and you can see easily. There isn't much room left on the upper side according to these numbers where there's a lot of room with justification that you can go down the other way.

And what I suggest is in a programmatic sense, we should close this task as a part of SGAP and focus on whatever NRRs current needs are for the specific items they need from a PRA point of view, which by the way, includes things like you mentioned before a guidance document, systematic evidence document and so on.

I think that's part of their list of, shopping list. And that was in -- I think that was also mentioned in Task 3.12 originally, which wasn't really pursued at this point.

But if you -- but my personal opinion is, if you keep this open in five years you'll have another clown sitting in front of you with another PRA or whatever and we'll give you another bunch of numbers without any new insights or something.

MEMBER BLEY: Selim, can I take you back to your slide number 5?

MR. SANCAKTAR: Five?

MEMBER BLEY: And if you have to go there and if you just type five and return you'll probably get there instantly.

In your report you say the first three tasks were complete, task-T looks to me like it's complete by the Sandia report, tasks E, F and G on your list are also going to be complete. Are they complete as a result of your report that you've given us or are they complete based on other things and that's CE plants, external event initiators and consideration of pressure-induced tube ruptures.

I think you're the only one speaking on 3.5, that's why I'm asking you about that.

MR. SANCAKTAR: Yes, there is no question. Okay. You can't read this, but I can. This is the outline of the report, and you probably have the report.

MEMBER BLEY: Of your report?

MR. SANCAKTAR: Yes. I think this --

MEMBER BLEY: Your report is where these three are closed.

MR. SANCAKTAR: Theoretically, yes. Let me quickly elaborate on it a little bit.

MEMBER BLEY: I'm sorry?

MR. SANCAKTAR: Let me quickly elaborate on it a little bit.

MEMBER BLEY: Okay.

MR. SANCAKTAR: The method, yes, I talked about the methods. Application illustration, I talked about it. External events, I talked about it. Main steamline break, I talked about it and there is further discussion that I didn't talk about in section 3.3. So I claim that based on that there is no more need to pursue it.

With respect to CE plants, I claim that methodology is applicable, I showed an example. The Consequential Steam Generator Tube Rupture probabilities would change. What would they change to? I think .4 was high enough that in this case, I don't think we will get any new insights from that part of it.

The insights will come from the composition of the sequences I think more, in which I tried to illustrate. You can argue either way. You can argue that no it's not because I don't see any calculations anywhere for Consequential -- conditional Consequential Steam Generator Tube Rupture probabilities. There are no calculations that are given anywhere.

So if you look at it that way, I leave it to you. Uncertainty, part of it I put a section here, but I don't think -- it just shows what you can do, it doesn't -- I don't think it addresses because uncertainties, many of the uncertainties that are really meaty parts are buried up-front in the basic assumptions of not only PRA, but also materials and thermal hydraulics.

And, I have difficulties seeing how they would be put to bed conclusively. So I made the statement in one of the slides that we recognize it, that the uncertainty it not yet addressed. But if you said, okay, keep it open go address and come back, will you get anything new soon? I'll leave it to you. I have my opinion about it whether theirs is a way to proceed or not, but I leave it to you.

CHAIR POWERS: Are there any other questions for the speaker? I think we can move to our closing presentation then. I remind the committee we have two chores to do once the presentations are over. One of those chores is to recommend to the staff what subset of the materials that we've heard over the last day and a half should be presented to the full committee.

And my recollection is that we have about an hour and a half of presentations --

MR. BRADLEY: We squeezed that a little bit more time, it's almost about two hours.

CHAIR POWERS: So we have two hours, so it's about an hours worth of presentations. So it is a substantial editing of what we've heard.

So you might, as we go through the closing presentation, I want you to of course pay attention to the closing presentation, but you might also give thought to what -- to any guidance we can provide them on how to package all this material.

Yesterday was a full day, today was very new material that the committee had not heard before, and so how they're going to package that and what you should highlight. We owe them some guidance there because that's always a hard job on how to compact things.

MEMBER STETKAR: Can I get something on the record before we go to --

CHAIR POWERS: We would love to have --

MEMBER STETKAR: -- because I need to eat some crow here. Regarding the sampling method, I just want to make sure that this is on the record, it depends on how the simulations are run because I can actually see a way, thinking about it here, that indeed the counting method that you used could indeed come up with a correct probability.

So I wanted to make sure that that's on the record. We can talk later offline to see how the actual sampling was done.

CHAIR POWERS: You are very kind, sir. Go ahead. But kinder than I was to him in the past.

MR. BRADLEY: I don't have any summary slides, but based on the questions today with respect to the PRA, I want to emphasize the user need that has been drafted.

You know, there's no PRA out there that's bullet-proof, but closing the action plan, I think we've -- and preparing the user need, we've identified where do we want to spend our resources, where do you want to dig deeper.

So we ask that you consider that and the fact that the Steam Generator Action Plan never -- work will always continue, there will always be -- we'll always learn more and take into consideration what we've in NRR have suggested that they pursue further. That's all I have.

CHAIR POWERS: Well I think the final speaker made that point in the risk-assessment world you can continue on doing this forever. I mean, his recommendation was to focus more on what the line organization needs are. And of course, that's one we're always very sympathetic to.

Well that said about our chores, thank you David, our one chore of course is to draft some positions for the committee consideration and the other chore is to give David and his team some advice on what to present.

I think I want to go through that second chore first, and I will poll the committee. What we've heard over the last day and a half is really focused on the issue of induced steam generator tube ruptures.

We've heard from a dynamic -- I mean, thermal hydraulic information much more realistic capabilities that have been new to us, they're expansions of analyses that we've heard something about in the past.

We've been assured that the materials understanding is important and is limited by the flaw distribution that seems to be a universal truth that arises on all materials issues.

And what we've heard this morning, we've heard that induced steam generator tube ruptures are -- cannot be absolutely excluded nor are they apparently the predominant effect that the PRA standard is correct, we're not to think about these, and more importantly, that it's possible to think about them, that you can actually do something with them.

And there's a reasonable amount of plant-specific acts in the specific aspects to it. All of this has to be packaged in a succinct presentation to the committee. So I'll ask you individually and collectively, what should they highlight? What should be the focus of their presentation.

MEMBER STETKAR: If you're looking at me, first if you are.

CHAIR POWERS: Yes. Hope I'm not.

MEMBER STETKAR: The way it's been presented to us over the last couple of days has been the individual building blocks, the thermal hydraulics, a little bit on the materials and then finally the integration into the risk assessment.

I'm wondering whether a different organization, in other words, starting from the risk assessment to say well, we've -- there was one slide yesterday that showed the triangle about how all of the pieces are interrelated and the PRA was actually at the top of that pyramid because that was the driving force in terms of determining -- defining the scenarios.

I was wondering whether organizing the presentation to the committee might benefit from that perspective rather than kind of the bottom up. And I don't know, I'm just thinking. I don't know how to efficiently package it in a way that the full committee was presented with both the structure and the building blocks.

CHAIR POWERS: So, Stetkar, to the surprise of everyone on the committee thinks PRA should be the top.

MEMBER STETKAR: It was on the slide, I didn't make it up.

MEMBER SHACK: One other way to look at it though is in the questions we had the last time we were here, a large number of those questions were focused actually on the thermal hydraulics.

CHAIR POWERS: Absolutely. Absolutely.

MEMBER SHACK: And so, you know, if we're out to have the presentation address the questions that the committee raised rather than the issue itself, you know, we might, I think need to have an emphasis on the thermal hydraulics. Even though the people who raised those questions are --

CHAIR POWERS: Their collective memory is still here. John, I think if the staff were coming in and saying we have resolved this issue that your organization would be the right way to go, the staff's not coming in and saying that.

To the contrast, they said we have answered the questions posed in the action plan, we now want to move to meeting our customer's needs. And so I don't think they have things packaged together in that hierarchy that you're looking for that their customer is now asking them to do that. And do I'm reluctant to take that tack --

MEMBER STETKAR: Okay.

CHAIR POWERS: -- because I think we asked them to do work that they in fact have said we're fixing to do to follow that tack. The thermal hydraulic issues and the questions, maybe that is simply the approach to take and say we have -- we've had this action plan, you've seen most of the elements.

The elements that we have not presented to you in any great detail in the past have been the induced steam generator tube ruptures, you had these questions, here's what we have on those.

MEMBER SIEBER: I would support that, but

I think you're light in the materials area, at least from my standpoint. And if you could beef that up a little bit more particularly when you're discussing the issue of failure of the hot leg before the tubes rupture, because I think that's an area I think was just sort of like a given without much showing what level of detail that you looked into the hot leg.

I think you could beef that up, because, you know, I just felt that, you know, that was just an input. I didn't see anything that would justify.

MR. BRADLEY: The way I saw it I think it would be best to move Item 3.12, which is really the user need to the very end.

CHAIR POWERS: My reaction to that, I've actually thought about that. I don't want to get too involved in the details of the user need because I think it's still in draft form.

And the fact that the user need exists, I wouldn't hesitate mentioning that, but I certainly wouldn't pursue it in as a -- here's what I has, because it is very likely not to have everything exactly as written, you know, when it finally comes across.

I mean the existence of the user need I think is fine to acknowledge, the details of it I just would not pursue at all. I comment that it would be very nice for us, for the ACRS as a whole if once NRR is happy with the user need and RES is happy with receiving the user need, they come chat with us just for information purposes.

I think that's very useful for us to do, but it's premature. My reading of it is it's premature.

MR. BRADLEY: Okay. Right now, he asked me to prepare a draft agenda for that meeting and I just took the existing amount of times in proportion and rounded it off to the nearest five minutes and gave everybody the same amount of time and we could just all -- have all the speakers up there and do it back to back.

MEMBER SIEBER: I guess, when I was thinking about preparing for this session, yesterday's session and today's session there is a couple of approaches that appeared to me and the way I think of things is, why are we having this session in the first place and the big issue is Consequential Steam Generator Tube Rupture.

In order to be able to examine that, there are a lot of tasks, and these tasks are basically your agenda. And you figured out that you needed to do these tasks years ago and so now is an update on the status of where you are.

And I would put it down, and you already had slides that said, you know, we finished this, we finished this and we ought to close it, close it, close it. Here's one where we think we need more work, but we can close the item and open something else that provides more work.

And that's a way that some of us, at least me, the way we approach these kinds of subjects. And I think the key thing is to put it into perspective what a Consequential Steam Generator Tube Rupture is. And we had that presentation today and I struggle with parts of it, and I try to understand why I was struggling.

And of them is that first of all we left out one kind of plant that has a similar vulnerability and it's absence to me is obvious. The other thing is that we relying on SPAR models in a way to achieve consistency, but SPAR models generally even though it's a mixed bag as to what licensees do, but generally they make some concessions that some licensees cover in detail.

So the use of the SPAR models, particularly over this long period of time results in variable results for different plant types. You can't take plant A and compare it to plant B particularly well because the model changed during that period of time.

And for example, there's some types of plants that didn't have internal flooding, did not have fire analysis and three or four areas that were black, so the proportion that you attribute to the Consequential Steam Generator Tube Rupture is sort of out of whack. So you can't say well this plant's more vulnerable than that plant because the analysis is different and the analyzed parts are different. Okay.

I think that ongoing work is necessary at least to normalize it, but the point is to put it into perspective. It's not -- it's a rarer, but rather consequential, event. And the question is how much time and effort and resources do you apply to something that could be severe, but is highly, highly unlikely to occur.

It's one in 10 million or something like that per reactor-year, but it's still in the box, if you know what I mean. It's in the box where you have to consider it.

If you think through the sequence of an accident and it turns out under every emergency plan that I've seen and evacuation ends up occurring pretty early in the sequence, and probably would be accomplished before a bypass occurs based on evacuation times and all that kind of stuff.

But, evacuations, that kind of defense is a last resort well beyond what these plants should be designed to protect against. So I think, to me a -- that's how I ended up putting all that was fed in the last day and a half into my way of thinking.

I don't know whether other members think that way or not because we all have different backgrounds, but that's the way I did it. And what you end up with is an action plan as to what should be closed, what needs extra study, but still can be closed out of the plan and opened somewhere else, what needs additional work to fully understand it and then balance that against all the other work that the agency needs to do based on risk.

To me, that makes sense, but there are a lot of good things that happened. The material studies, the CFD analysis, the conclusion that loop seal clearing is the key that says it either fails or doesn't.

I struggle with it a little bit because of the very short time difference in the sequence that occurs from one to the other, and you know, that's sort of a close call in my view. And that's an area where you may want to think about it a little bit more.

But that's how I'd lay it out and I actually do think your recommendations as to what should be closed and what deserves further study is pretty accurate. And so I guess that's my thoughts on it.

MEMBER BLEY: I guess if you start with the task at hand to finish those, there's a couple things, a couple little details I'd suggest in what you'd give to the full committee. One is from your materials guys have them, it sounds silly, don't print them on blue background so people can't take notes on them, print them on a white background. It's annoying.

The thermal hydraulics guys all live in seconds. I think you could make one of the arguments he made much better if you created a slide that started at the point of maximum oxidation energy and then showed in minutes the time to the various failures.

It jumps off the page in a way you don't see in the way it was presented yesterday and we did that on our own, it really helped.

I think after you say what you've done, a little digression on what are the really key uncertainties there and what are the gaps you've still got would be very useful.

MR. BRADLEY: In all areas, right?

MEMBER BLEY: In all areas. And then kind of following on what Jack said, you know, a way to tie this together is a little bit the PRA thing, but really, how are these various pieces related and how would they affect the things we care about in terms of risk.

And we haven't seen anything that says core damage is going to go up, but we've seen that LERF could go up, but we've had a mixture of people talking about things that could really effect LERF and people were talking about any kind of a bypass, even very small ones.

So given a little thought to clarifying the difference between those things and the presentation I think would be useful to you and to us.

CHAIR POWERS: Sanjoy?

MEMBER BANERJEE: Okay. So I guess the last time we really saw this was 2004, which was before my time here. My illustrious predecessor was here, Graham Wallis and I've been looking at what's been done since that time. I'm not fully up to speed.

But clearly this is a significant contributed risk from what I understand the consequential -- oh, sorry. Oh, okay. Sorry, I didn't put it on. I guess from what I've heard today, this is one of the most significant contributor to risk or not Consequential Steam Generator Tube Rupture --

MEMBER STETKAR: I think if I can characterize, the take away that I had is that it's not one of the most significant contributors to risk, neither is it an insignificant contributor to risk. It's something that could be comparable to other containment bypass scenarios that we already quantify so it's --

MEMBER BANERJEE: Is it internal or external events?

MEMBER STETKAR: I'm just saying risk, containment, you know, offsite, I don't want to fine tune it that closely. It's large enough from the analyses that have been done and the work that's been done that you cannot dismiss it.

You can't say that this is a negligible issue that the whole subject is closed, nor should you say that it's the dominant contributor or by far and away the most important contributor to risk. It's important enough that indeed people should be quantifying it in your risk assessment and by-and-large they not.

MEMBER BANERJEE: So let me ask you --

MEMBER STETKAR: So the take away from the risk assessment is, it's important enough that you need to consider it, it's variable from plant to plant, there's a lot of uncertainty involved in it, it's not an easy topic to solve, but you do need to look at it.

MEMBER BANERJEE: So even if you took uncertainties and biased them to the side where you said that bypass became likely if the scenarios leading up to this type of incident occurred, the probability of these scenarios occurring which could potentially lead to bypass are very low you are saying, therefore, they don't contribute --

MEMBER STETKAR: It's not clear how low they are.

MEMBER BANERJEE: So imagine that --

MEMBER STETKAR: All I'm saying is that it's probably in the same ballpark as other things that we already do calculate, however well we calculate them.

MEMBER BANERJEE: Yes, imagine though that for whatever reason these scenarios always led to loop seal clearing and there was always steam generator tubes that failed before some other path into containment fail. Just imagine that for the moment. Would it be then the most significant contributed risk or it wouldn't be important?

CHAIR POWERS: It would be by definition.

MEMBER STETKAR: You can always quote the books well enough to make a point.

MEMBER BANERJEE: Well let's say that we assume that, would it then be the most significant or not?

MEMBER STETKAR: If you want to assume conditions where it becomes the most significant, then you could say it's the most significant, but it's not -- the issue is that it's -- the work that's been done is not conclusive regarding how in an absolute sense how important this issue is, in an absolute sense.

In a relative sense, what has been shown is that it is important enough that it should be considered more carefully in the risk assessment. It indeed should be quantified in a careful manner. I don't want to use the term rigorous because that's a biased term.

What the results for a particular plant from a careful assessment, accounting for the realistic uncertainties, include in terms of the absolute contribution to risk is not known right now. We don't know that.

MEMBER BANERJEE: Right.

MEMBER STETKAR: If you biased everything artificially high, indeed it could be the most important contributor. If you biased everything on the low side, you know, it can be an insignificant contributor.

MEMBER BLEY: If you think what we saw was biased high, it might be, that's 10 to 20 percent of the LERF kind of thing.

MEMBER BANERJEE: Well, let me give you an idea of where the important thermal hydraulic issues are here. The first is loop seal clearing, what are the uncertainties associated with that. Okay.

If the loop seals clear, then the possibility -- the probability that the steam generator tubes will fail, as he said a bunch of them will fail, is pretty high in this scenario and therefore you won't have a path into containment, but you'll have a path out of containment because it's not likely that the hot leg or anything else will fail because it will depressurize.

MEMBER STETKAR: That depends on how big a bunch is.

MEMBER BANERJEE: Yes. But if the loop seals don't clear it will be a bunch, it will be big enough. So there will be lots. So that I think is the first large uncertainty that needs to be addressed.

And I think a lot of work has been done to set our minds at rest and so on, but it will need to continue to be addressed in some other action probably in the future. That's the way I see it because I'm not completely confident yet without going through everything in detail as to whether what I'm saying is right or not, but to me, that seems one of the large uncertainties right now.

The second has to do with the modeling of the vessel itself. I know that there's been some way to represent this in the CFD calculations, which certainly could be adequate. The way it's done, if I understand it correctly, is taking the temperatures in the top of the vessel from the SCDAP/RELAP calculations and inputting into the CFD.

The pressure losses and all are actually calculated, this I learned yesterday from a conversation with Chris. We still need to understand how this simplified representation of the vessel coupling SCDAP to the CFD calculations leads to some uncertainty or not.

The potential for mixing and stratification in the vessel I understand was dealt with back in history. I need to take a look at that and see whether it was adequately treated or not because that can make quite a difference as well.

Then there is the issue of the sensitivity of the CFD calculations to nodelization. These mixing sort of predictions, you know, there's a lot of numerical diffusion in these types of calculations particularly with codes like FLUENT and I don't know how much numerical diffusion there is that I have to get on top of that as well.

So, we need to be sure that in the shear leg which cause this mixing, which actually is why you don't get these high temperatures and indeed, the experiment shows that as well, though I'm not quite sure without again looking at the experiments in detail, what uncertainties there are.

However, in the calculations itself, some grid independence in the mixing results will probably need to be established if not today, in the future. And of course there are also the uncertainties in the thermal hydraulic boundary conditions.

So they're a lot number of uncertainties in the thermal hydraulics and there's no reason that they need to be addressed before the next meeting or anything of this at this time.

But in future, these uncertainties are going to have to be probably addressed if this thing is going to be put to bed and say, you know, the likelihood of the hot leg failing is pretty high compared to the steam generator tube failing, because the timings are fairly close to each other.

You know, if they were like hot leg was failing say a half an hour before or something, then the uncertainties may not matter, but if the hot leg is failing pretty close to the steam generator tubes, then it could be there was massive failure of the steam generator tube before the hot leg, then the hot leg may not fail and there's a path out.

So, and a lot of this depends on the mixing, you know, and that's a tricky calculation to do and to scale up. So I think that very nice work has been done on the thermal hydraulics, especially the CFD and I should say also with RELAP.

They've answered the questions regarding counter-current flow modeling that came up in the ACRS meeting in 2004, so they've done the things that the ACRS wanted done and they've shown that this seems to be in line with the sort of phenomena they see in the experiments and some other code calculations done in other countries.

So all to the good I would say. But I think we should also be sort of straightforward and say these are uncertainties that may or may not need to be addressed, but it will be considered anyway in the future, in this area.

Okay. So from my point of view, I think we do need to present the thermal hydraulic stuff because it's pretty important. I mean, that's really what gives you the sequence of events and so on.

CHAIR POWERS: So once again we're stunned, Stetkar wants PRA on the top and Sanjoy wants thermal hydraulics highlighted. These take me aback, you have no idea how stunned I am at this.

MEMBER STETKAR: Yes, I wonder, listening to Sanjoy, I wonder if there's a way to combine in a presentation to the rest of the committee or the approach that Dennis laid out, but within each topic kind of give the top level of what we've done, you know, where we are, what we've done and also address specifically the biggest areas of uncertainty.

You know, Sanjoy mentioned several sources of uncertainty, which are the largest sources of uncertainty from your perspective and how have they either been addressed and what you've done or how should they be addressed going forward.

And within thermal hydraulics, Sanjoy had mentioned several, we didn't hear much from the materials guys, but the flaw distribution and in risk assessment, uncertainties, the mechanistic propagation of the uncertainties is not what I'm talking about.

But in terms of completeness of how we've looked at the problem within the context of the risk assessment models; how confident are we that the risk assessment models and the way that they're currently structured will capture, you know, these scenarios for example.

CHAIR POWERS: What I worry about is mission creep here. We're being asked a different question, not has the question been resolved, have they done enough to meet the needs of the action plan.

And I have a feeling if we go off on a tact of what needs to be done and how have we addressed issues that we never intended to address that we will blow through the two hours so badly that Bonaca will be moved to castigate me. I of course live in fear of being castigated.

Yes, I really think that we've been asked a finite question and I'm not willing to expand it up into how do we completely resolve this issue. Many of our questions, many of the action plans were simply to gain enough understanding that they could define how they went about solving. Dennis you had a point?

MEMBER BLEY: Well, since you've dismissed -- no, but it's

CHAIR POWERS: If you're going to tell me the PRA should be on the top, then I need to write this down.

MEMBER BLEY: The PRA just organizes all these things, well maybe that puts it on top. But I think on the uncertainty issue, it's not just what are the major uncertainties, but which ones have the potential to be important.

You guys made a really good point yesterday and in the Sandia PRA they pointed out that they have identified quite a few hundred from hydraulic runs they wanted to do for their PRA and large clumps of them they ended up not having to do because of learning that many of those uncertainties changed the timing up to the point where you get the oxidation, and as you guys pointed out, all the action happens after that.

So those uncertainties that are out in that part don't really effect the problem too much. The ones that effect everything that happens after that though get really important. So if you get to them, cataloging them a little bit under how big they are and where they matter is more important than just how big they are. That's all.

MEMBER SIEBER: I think the fundamental question is -- the fundamental question is, can the staff close the items that they're recommending that they close even with the caveat that by closing it they think they need to do additional work that will come under a different category.

And I think we have to answer that question or at least give advice to the staff, because that's why they came here. And if we don't do that, then we haven't accomplished the main purpose of what we're here for.

And I can volunteer that I think the staff is pretty correct in what they think they should close and what they think they can close, but do additional work. And so if I were to cast a vote, that would be the vote.

MEMBER STETKAR: I agree, Jack. And the reason I brought up the uncertainty stuff was not to try to, you know, drag this out for another decade was to give the staff the opportunity to demonstrate to the full committee that indeed, yes indeed these issues can be closed out despite the fact that we're aware of these uncertainties.

MEMBER SIEBER: Right.

MEMBER STETKAR: And that indeed in any follow on focused research, you know, we want to address these issues, but from the perspective of closing out the action plan, we can do that, but then we recognize --

MEMBER SIEBER: Right. I think some --

MEMBER STETKAR: -- just the issues. I didn't want to, despite Dr. Powers admonition drag this into more mission creep.

MEMBER SIEBER: I think that we, and the staff needs to answer the question that are the uncertainties so large that it effects what you would so under the plan and I suspect the answer to that in almost every area is no, we know enough about the band through which we're operating to be able to make research decisions.

CHAIR POWERS: Well, I think we've given you as much advice on your presentation as we can. I don't think we've changed your overall strategy at all. But maybe this gives you some additional insights.

What I'd like to do now is to move to this question of the draft position that we prepare for the committee's deliberation and I think I am sympathetic with Mr. Sieber's position that, yes, we agree with the staff's closeout on these items, and that we prepare a letter that says that fairly well up front that the balance of the letter, which may actually be a fairly lengthy letter, provides the kind of background that I think the commission needs to have for this particular set of issues.

You know, we've heard primarily with respect to the induced steam generator issue in our presentations here, but I would propose preparing something that provides something more than just the induced steam generator background.

And I invite the members to contradict me, or if they agree, suggest topics that should be, appear in the letter.

MEMBER ARMIJO: Well, I would like to say, you know, I'm with Jack, I think the staff has done plenty of work that you can close out these items, but clearly there are some areas of uncertainty that should guide the research.

Maybe it's in the user needs, maybe not, but we should say those, we should provide our recommendations on further work whether it's through research or some other mechanism and whether it's thermal hydraulic or materials or PRA.

MEMBER SHACK: Just coming back to the bigger picture, I mean the reason we set this whole thing up in the first place was to know whether the way we changed the regulation of the degradation of steam generators was going to create us a problem in terms of severe accidents.

I think the steam generator action plan has resolved most of the questions, that, you know, the change we made and the way that we regulate steam generators, you know, hasn't led to multiple tube failures, it hasn't led to substantial increases in risk due to severe accidents. So, I think, you know, that's the first thing and that's been done and that's been accomplished.

I think that is almost it, and now we're moving on to a better understanding of risk, but in reality this thing started out to really decide whether we were making a mistake in changing the way that we regulated the degradation of the steam generator, you know, that question's been answered. And that, in my way, is the most critical question that --

MEMBER SIEBER: You actually had the benefit of exploring beyond-design-basis events that if you go back 30 years ago when a lot of these plants were designed there was not that much analysis and thought given to it because it was, in fact, beyond the design basis and considered extremely rare. So now we have a better perspective on that today and I think that's an important perspective.

MEMBER BANERJEE: I like that, actually. What you said Bill, that should be sort of in the letter --

MEMBER SIEBER: Remind ourselves.

MEMBER BANERJEE: Yes. Because otherwise people like me get caught up in detail, we love detail and then --

MEMBER SIEBER: Well this is interesting --

CHAIR POWERS: I can't imagine that.

MEMBER SHACK: Now, I would say that part of the future work I would like would be to see

Mr. Bradley exercise his tool on a realistic flaw distribution to see just how big these contributions could, you know, I think that might help set the prioritization of all this work.

You know, how much do we have to beat these uncertainties to death if, you know, when we take a look at a realistic flaw distribution that we find in our reactors now and, you know, does it or does it not make a big difference.

CHAIR POWERS: I'm donning for you to provide us realistic flaw distribution. I understand your point. John? Are there any other comments that we need to make?

MEMBER BLEY: Are you about to close or are we going to have a general --

CHAIR POWERS: We are in the general.

MEMBER BLEY: Okay. There are a couple things I didn't say while we were going through the PRA and I wanted to get them on the table. One is if you go through the Sandia report, which now we know is kind of one the 2005 and 2008 are the same thing, but in 2005 version they really, I thought, had a good statement on uncertainty in the document itself, I haven't read what's in the new one.

And it pointed out the ranges of

uncertainty that would matter and the area that was most important, which were the epistemic uncertainties on phenomena, which is where we've been ending up.

They also did a good job of addressing plant operational issues, but, and this is close to what Bill was saying, they calculated some kind of small bypass and not LERF and they didn't address the possibility of the hot leg break after the tube ruptures, loop seal clearing, any of those new issues we've been hearing about. So I agree, I think that would be very good to pick up.

The place I diverge from some of the things we heard is I think it would be possible, based on all we've seen, to put together some pretty good guidance on how to extend a particular plant's PRA to cover these issues.

A lot of the work the Sandia guys had to do was getting familiar with the plant and not only that, things that the guy who owns the PRA for the plant has at hand. And he's got all the operators, he's got all the information on training.

So I think that could be put together and I don't think it's the massive job we've been hearing to do a Cadillac if you own the PRA and know how it's all put together. John wants to interrupt me, so go ahead.

MEMBER STETKAR: And I will, thanks. There's actually a statement in there that I highlighted, I don't have in front of me, that basically said that.

They said that they spent a lot of time getting familiar with the plant and the available model for that plant, but they were limited in resources so they needed to take some pretty quick and dirty fixes to adjust the model in a way so that they could actually understand it and quantify the results.

But they also made that same observation that they didn't feel that this was a significant effort for people who really understood the model. They explicitly said that.

MEMBER BLEY: I don't remember seeing that, but I --

MEMBER STETKAR: That comes out of people who, you know, kind of had that experience, that gee if I had built this model, I would know how to change this to look at these issues as long as someone tells them what issues they need to be sensitive to.

MEMBER BLEY: I'd make one clear statement, as of right now there is no PRA that's there that integrates what's currently known about this issue and it would be nice to have that. Nobody's done that and there are pieces of it in a number of places.

Why should people do a good job on this, well one of the reasons is, it would be a really good tool for refining plant procedures and training with respect to these issues, if after we've worked out the uncertainties it still remains something of significance. Even if it's highly unlikely, if you already know what to do if it happens, it's nice to have it in your back pocket.

On all of the issues I really agree with Jack and everyone else, I think, they can close all of these. The one place closure isn't as clean to me is on the last three issues under 3.5. I don't think those are really closed, but I think it would be okay to close them based on what we've seen if they're going to carry on some additional work to get to where we were just talking about.

The big holes for me are uncertainties and knowledge gaps and I think we've talked about those enough, I've listed a bunch. But we've heard a lot of people talk about them and addressing those would be very useful.

I think, where I said yesterday and I'd

still say it, something that could be done initially is to catalog all of the uncertainties, their types and the possible ranges and flag the ones with potential significant impact. And I think that could be done as a preliminary step and would be very helpful and I think everybody would learn an awful lot.

Every time I've seen people just do that cataloging of uncertainties, work through whether they're epistemic or random, you know, characterize them and think about what their impacts could be they learn a lot.

And right now it's anecdotal and off the tops of our heads mostly on the things we like best where we're focused and I think doing that would be an excellent first step. That's the things I wanted to say.

MEMBER ARMIJO: I just had one more thing. I finally got to where I -- what's been bothering me. In the -- since the hot leg failure is so important, if it's already been done I think in your presentation, you should show at least -- at least be able to make the statement that the level of detail in the analysis of the hot leg of temperatures, times, what fails and, you know, if that level of detail is comparable to what's gone into the tube failure analysis then I'll be happy.

But I think it's a complex component there. There's a lot -- there's different materials, I saw -- Bill sent me a very condensed little report where it shows a big balloon forming on the steel pipe. If I had to guess, that's not where it would fail. I think it would fail in the carbon steel, but, you know, I don't know if that's already been dealt with and put to bed. So if you could --

MEMBER SHACK: That's not what the analysis says.

MEMBER ARMIJO: Well, I hadn't seen the analysis Bill, that's why I'm saying, so if it's been done, that would make me a lot more comfortable. And so if you can say that, you know, we've analyzed it, we went through a detailed analysis of the nozzles and the materials and their stress rupture properties, temperatures and the failures have got to be here and it's going to be, you know, a big balloon and it bursts, I'd be -- I think that needs to be said.

Otherwise, you'd have a very detailed analysis of the tubes racing against a maybe not so well understood analysis of the hot leg pipe or nozzle area.

MEMBER BANERJEE: Plus there's convection from the outside of the hot leg which is cooling it. I don't know -- I didn't get a picture of exactly how important that radiation, is it keeping it cooler and less likely to fail?

MEMBER SHACK: We actually do a pretty good job on the heat transfer to the hot leg.

MEMBER BANERJEE: All right. Okay. I'll trust you, Bill.

CHAIR POWERS: Okay. We have --

MEMBER SHACK: Well just in this general discussion again --

CHAIR POWERS: In order to further your non-contributing to our deliberations --

MEMBER SHACK: Another point. I'm a little, you know, to me, you know, are we getting off into SOARCA land where we're studying a problem to find out whether it's really risk-significant, you know, does it have regulatory impact or are we looking for a greater understanding here.

One of the things that worries me a little bit is that the results are going to be so plant-specific, you know, I can't quite imagine every plant doing a Chris-Boyd level of CFD that's going to satisfy Sanjoy and a materials analysis of the hot leg surge lines to, you know, that might satisfy me.

And if that's what it, you know, if we're down to -- uncertainties are so controlling then I think you do have to stand back and ask yourself, you know, why are we doing this, and, you know. But, onward, onward.

CHAIR POWERS: Because we are in the interest to protect the public health and safety. Are there any other comments we'd like to make? Okay.

I think I will close this subcommittee meeting with the -- imploring the members that if you have passionately held language that you would like to see appear in the letter, I will certainly give it all the consideration that I possibly can.

I will definitely consider any contributions to the language. I'm sure that I will crib the piece of language that Mr. Boyd asked me to adopt from our 2004 letter. And with that I'll close this subcommittee meeting.

(Whereupon, the foregoing matter went off the record at 12:18 p.m.)

A Presentation to ACRS Subcommittee on
Materials, Metallurgy & Reactor Fuels

For

A RISK ASSESSMENT OF CONSEQUENTIAL STEAM GENERATOR TUBE RUPTURES

September 25, 2009

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Outline

- Purpose and Scope
- Steam Generator Action Plan Task 3.5
- Background
- Objectives of the C-SGTR PRA Work
- Analysis Method
- Illustrative Examples
- Summary
- Conclusions
- Recommendations

Purpose and Scope

- Technical work performed to address remaining subtasks in Task 3.5 of SGAP:

  "Develop improved methods for assessing the risk associated with SG tubes under accident conditions"

- Recommend closure of SGAP item 3.5

Steam Generator Action Plan Task 3.5

SGAP PRA Task 3.5 contains 7 subtasks:

a. Development of an integrated framework for assessing the risk for the high-temperature/high-pressure accident scenarios of interest (Closed 04/01/02).

b. Issue report describing improved methods and appropriate treatment of uncertainty for identifying severe accident scenarios that lead to challenges of the reactor coolant pressure boundary (Closed 06/28/03).

c. Develop logic framework for improved PRA models of the scenarios identified above, including the impact of operator actions (Closed 04/06/04).

Steam Generator Action Plan Task 3.5 (continued)

d. Using the 3.5.b methods and (c) logic framework, identify scenarios, calculate the frequency of containment bypass events at an example plant, make indicated method improvements, and document the improved methods and results.

e. Extend the 3.5.b methods and (c) model logic to include CE plants, and document them.

f. Extend the 3.5.b methods and (c) model logic to include consideration of external events as initiators, and low power and shutdown as initial conditions, and document them.

g. Extend the 3.5.d, e, and f improved methods and logic to include consideration of core damage sequences initiated by secondary depressurization events (such as MSLB design basis accident scenarios) that induce tube rupture.

Background

- SGTR is routinely modeled in risk assessments of nuclear power plants as an initiating event.

- SGTR events are also included in the set of deterministic accident analyses for PWR licensing.

- Consequential SGTRs refer to

  - SGTRs that may be caused by another initiating event (e.g. ATWS or MSLB) or

  - a severe accident condition that leads to failure of SG tube(s).

- Current work evaluates the incremental risk due to C-SGTR events that may have not been included in the base PRA

Background (continued)

- If C-SGTR event occurs due to another initiating event (like a MSLB event or ATWS), it could increase the plant CDF.

- If C-SGTR event occurs due to severe accident conditions already leading to core damage, then it could increase the plant LERF, but not the plant CDF.

- In both cases, one of the concerns is the potential increase in plant LERF due to containment bypass via the failed SG tube(s) into the atmosphere.

- NRC has been studying various aspects of the SGs and issues related to them in depth for the last decade.

- The prior work indicates that the C-SGTR risk is plant specific, depending on the dominant sequences for a specific plant; and plant-specific mitigative measures available.

- The methods used to evaluate C-SGTR risk appear to be very detailed and would be intrusive, if they were to be applied to specific plants

Other Parallel Developments

- Many PWRs had steam generator replacements, which eliminate the older SGs and materials that were subject to scrutiny;
- Industry PRA models have matured and plant risk profiles (dominant sequences and cutsets) and CDFs have changed;
- USNRC's PRA models (SPAR models) matured and were benchmarked against industry models to understand and reduce the differences, if warranted;
- USNRC's data collection and analysis efforts, culminating in NUREG/CR-6890 issued in December 2005 and NUREG/CR-6928 issued in February 2007 provide new data for SBO events and others;

Other Parallel Developments (continued)

- Understanding and modeling of SGTR events, their success criteria and consequences have been further developed by the industry, with uniform guidance document provided to the utilities by the owner groups;

- New PRA standards provide guidance on a more integrated evaluation of plant risk (including internal events, external events and events at shutdown).

- Security issues prompted the utilities to provide new equipment and procedures that would also apply to recovery from severe accidents;

- Severe accident management guidelines are well established in nuclear plants.

Objectives of the C-SGTR PRA Work

- Address Task 3.5 of SGAP by developing a quantitative risk assessment method and applying it to different plants to demonstrate how to estimate the risk from C-SGTR. The method and its application should be as comprehensive in scope as possible to address specific items in the task.

- Provide an illustrative quantitative risk assessment of consequential SGTR events (C-SGTR) based on

  - Evaluation of CDF of C-SGTR events at multiple plants with existing PRA models, and

  - Including internal and external events.

- Current work uses plant-specific PRA models that are recently updated (SPAR models) to perform a risk assessment of consequential SGTRs.

- Insights and information from the other available work is used as needed to help with this assessment.

Analysis Method

- A simple method based on the sequence results from a PRA is used:
  - Identify sequences challenging SG tubes
  - Credit mitigation (i.e. probability that the operators mitigate the conditions leading to C-SGTR) (see note below)
  - Credit conditional consequential SGTR probability (i.e. probability that SG tubes do not fail given the conditions)
  - Calculate containment bypass frequency.

- This method provides a quick and efficient way to assess the plant-specific importance of C-SGTR and focus on potentially dominant sequences for further evaluation, if necessary.

- Scope of the method does not include LERF analysis; the measure of C-SGTR importance calculated is containment bypass frequency.

Note: The NRC document RIS-2008-15 provides cautions on incorporation of mitigating strategies into both the NRC and licensee PRA models, and NRC SDP analyses.

Analysis Method (continued)

The evaluation process for the CDF of those sequences that can lead to a C-SGTR includes the following steps:

1. Evaluate CDF for C-SGTR sequences from internal events at power:

   a. List internal event sequences sorted by their CDF.
   b. Identify those sequences potentially subject to C-SGTR.

2. Assign recovery probabilities and conditional C-SGTR probabilities to those sequences identified as potential C-SGTR candidates.

3. Quantify the CDF of each C-SGTR sequence and sum the frequencies.

Repeat the same process for other event categories (whenever available):

- Internal flooding
- Large steamline break
- Internal fire
- Seismic
- And other hazards (such as high winds, external floods, etc.) as applicable to the plant.

Analysis Method – Sequence Selection

- In selecting core damage sequences that are candidates for C-SGTR challenge, two types of considerations are taken into account: initiating event and system failures. For large SLB event, the C-SGTR is modeled as an event tree node and C-SGTR core damage sequences are explicitly identified.

I. Initiating Event Considerations

- Selection rules

  - Already containment bypass? - do not select.
    SGTR - do not select.
    Interfacing systems LOCA - do not select.

  - LOOP - SBO - AC power not recovered – select
  - ATWS - select
  - Large SLB and induced SGTR - select
  - Large LOCA - select

- Since Large LOCA involves a sudden drop of pressure in the RCS creating a reverse pressure difference across the SG tubes, it is kept in the list. It is noted in the example quantifications in the report that the Large LOCA sequences would not contribute much to a C-SGTR risk.

Analysis Method – Sequence Selection (continued)

II. System Failure Considerations

- The potential for containment bypass under the high-dry-low conditions is effectively eliminated if the:
  - RCS pressure is reduced due to operator actions or primary system leakage (eliminating the high-pressure condition),
  - Feedwater flow is maintained (eliminating the dry condition), or
  - SG secondary system retains pressure (eliminating the low pressure condition).

- Selection rules

  - SBO/ AC power not recovered (battery depletion – late failure of AFW (EFW)) - select
  - AFW (EFW) and HPI fails - select
  - AFW (EFW) and feed and bleed (once through cooling) fails - select
  - AFW fails; HPI operable; RCS depressurization operable - do not select
  - AFW and HPI successful; HPR (LPR) fails - do not select
  - AFW operable; HPI fails - do not select
  - AFW and RCS depressurization operable; LPI fails - do not select

Analysis Method - Mitigative Credit

- Various potential recovery (mitigation) actions can be considered sequence by sequence, whenever feasible. The following potential mitigation actions and limitations on them are used in the report:

  - Local manual control of AFW-TDP after battery depletion in SBO

  - Operation of AFW-MDP pump(s) after AC power recovery in SBO

  - Manual depressurization of SG(s) and low pressure potable water injection (must have at least 2-4 hours available; more for some external events)

  - Operators open pressurizer PORVs at CD (must have motive power to open and keep open PORVs; must not have failed previously in the sequence)

  - No AFW recovery credited in the most severe seismic bin (IE-EQK-BIN3)

  - Limited AFW recovery credited in medium severity seismic bin (IE-EQK-BIN2)

  - For fire events with MCR evacuation: Recovery of AFW after MCR evacuation and initial AFW failure (due to operator action failures)

  - SPAR model success criteria for core damage is used as is. Additional recovery actions are NOT credited for averting core damage, but for potentially avoiding or reducing potential C-SGTR, given core damage.

(*) Limit credit when multiple mitigative actions are available.

Analysis Method – Conditional C-SGTR Probability

- One of the inputs for the C-SGTR risk assessment method is the conditional failure probability of the SG tubes given the challenging RCS conditions.
- The Sandia PRA Report describes the methodology used to calculate this failure probability based on the following plant-specific inputs:
  - probability distributions for the length and depth of each flaw and the number of flaws
  - the time-dependent pressure difference and temperature experienced by the flaw, and
  - probability distributions for the failure time of other RCS components.

- The temperatures and differential pressures required to cause SG tube rupture depend on the characteristics of any flaws that may exist in the tubes due to the postulated tube degradation mechanisms (e.g., axial or circumferential stress corrosion cracking, or damage from loose parts).
- NUREG/CR-6521, "Estimating Probable Flaw Distributions in PWR Steam Generator Tubes," provides estimates for the number of flaws of each type that would be present in the steam generators of lightly degraded, moderately degraded, and severely degraded steam generators.
- For the example plant analysis, a moderately degraded steam generator was assumed.

Analysis Method– Conditional C-SGTR Probability (continued)

- To evaluate the failure times of specific reactor system components, predictions of a plant's thermal-hydraulics response and conditions that challenge the RCS components are needed.

- Thermal-hydraulics analyses were completed using the SCDAP/RELAP5 systems analysis code, aided by computational fluid dynamic (CFD) simulations that provided local three-dimensional details of the thermal conditions.

- The staff's thermal-hydraulic evaluation focused on severe accident scenarios that resulted from station blackout (SBO) events in Westinghouse four-loop PWRs.

  Details of the predictions and methodology used by the staff in support of the research to support closure of the SGAP are discussed in NUREG/CR-6995, "SCDAP/RELAP5 Thermal-Hydraulic Evaluations of the Potential for Containment Bypass During Extended Station Blackout Severe Accident Sequences in a Westinghouse Four-Loop PWR", to be published in 2009.

- Evaluations of the Potential for Containment Bypass During Extended Station Blackout Severe Accident Sequences in a Westinghouse Four-Loop PWR" (to be published in 2009).

Analysis Method – Conditional C-SGTR Probability (continued)

- In the Sandia report, in order to calculate the conditional C-SGTR probabilities, steam generator tube and RCS component failure models were programmed into an Excel spreadsheet. Uncertainty distributions for key model inputs were then developed using an Excel add-in, Crystal Ball.

- Ideally, the aggregate crack opening area required for an SAI-SGCB would be determined by a detailed severe accident analysis. In the absence of such a study, it was assumed that

  - flow through the cracks is choked, and
  - early containment bypass occurs if the contents of the RCS would be released through the cracks in less than 4 hours.

- Using this analytical approach, the mean crack opening area for containment bypass is calculated to be 0.081 in^2. The lower and upper 90-percent confidence limits for this value were calculated to be 0.053 in^2 and 0.124 in^2, respectively.

- All T-H runs listed in this column assumed nominal secondary side leakage of 0.5 in^2.

  Leak sizes of 0.5 in^2 and greater were found to result in essentially full depressurization of the SG and therefore yield essentially the same conditional C-SGTR probability. In contrast, leak sizes of 0.1 in^2 or smaller do not result in full depressurization.

Analysis Method – Conditional C-SGTR Probability (continued)

- For the most challenging sequences for the SG tubes, the conditional C-SGTR probability was estimated to be 0.4. This probability compares well with the worst-case probabilities of 0.3 (RES), 0.2 (NRR), and 0.5 (RES bounding) given in NUREG-1570, Table 5.6 for an "average plant", as defined in the NUREG.
- An insight obtained from the supporting work is that, even if the HL failure does not occur before some degree of C-SGTR occurs, there is high likelihood that a HL failure will follow a C-SGTR.
- This would not prevent containment bypass, but would significantly reduce the flow of fission products out of the breach, thus mitigating the severity of a fission product release into the atmosphere.
- This benefit is not accounted for or credited, since the RES C-SGTR PRA report focuses on estimating the frequency of containment bypass due to C-SGTR rather than estimating large early release frequency (LERF) calculations.

Analysis Method – Conditional C-SGTR Probability (continued)

Limitation

- No new T&H or materials related analyses were made for calculation of conditional SGTR probabilities for

    - CE plants, or for

    - Pressure-induced C-SGTR cases.

- The current understanding is that

  Due to the geometry of the CE SGs, the tubes may experience higher thermal stresses. For illustration purposes, the same probabilities as Westinghouse plants are used.

- The conditional C-SGTR probabilities for MSLB events may be lower than originally estimated in NUREG-1570. For illustration purposes, a high estimate of 0.5 is used in the report.

Illustration of the Logic Used to Estimate C-SGTR Frequency for a Sequence

		Mitigation ("Recovery") to avoid or terminate C-SGTR conditions (q)						
Core damage Sequence	**AFW Status**	**Manual Second. Heat Removal (after battery depl. in SBO)**	**Portable pump flow into secondary (B5b equipment)**	**RCS Venting**	**No C-SGTR event with conditions exist**		**End State**	**Frequency**
f	**s1**	**q1**	**q2**	**q3**	**p**			**f * q * p**

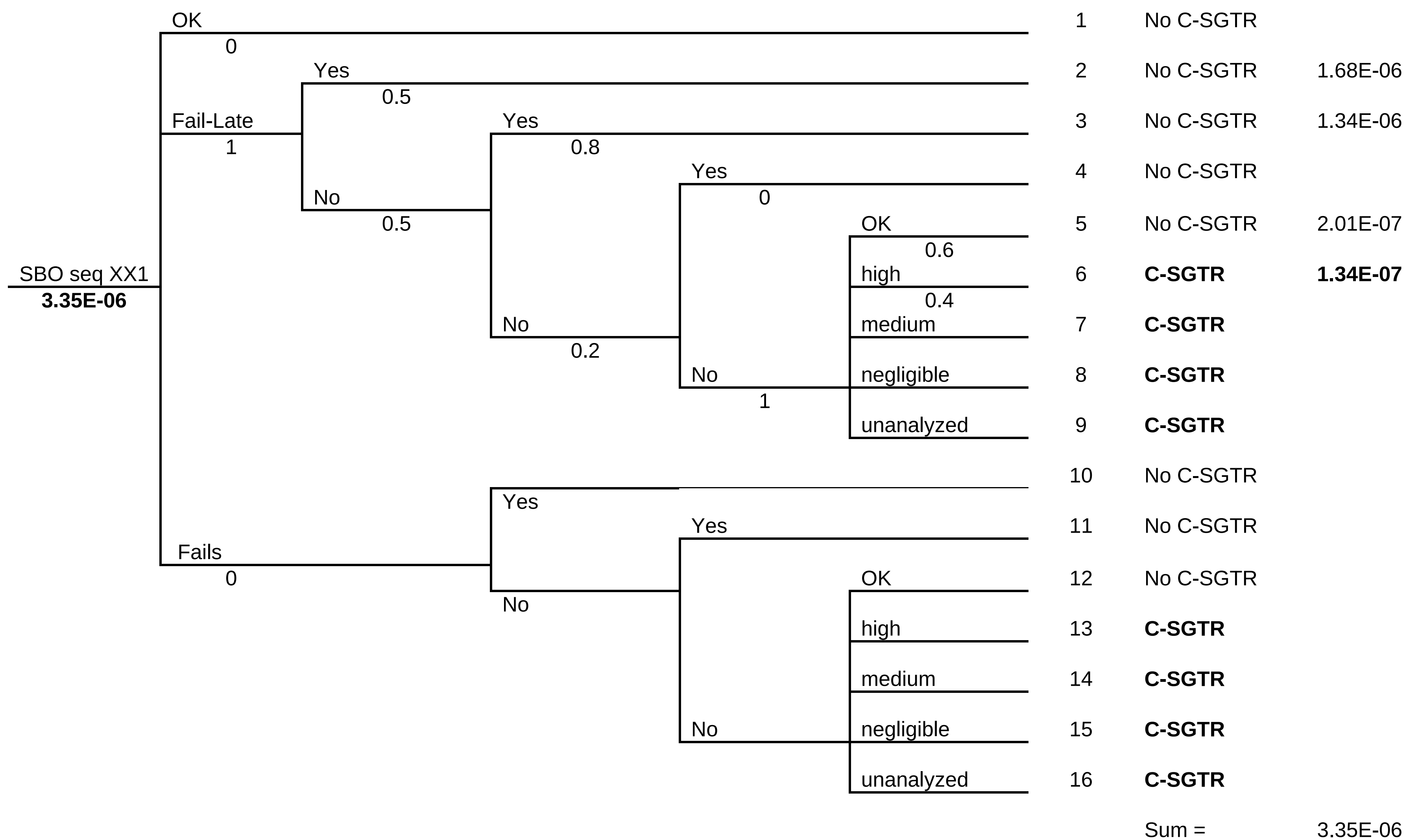

Table Illustrating the Method Used (internal event sequences)

IE Description	Sequence CDF	Sequence Description	AFW status	Recovery and C-SGTR prob.	Recovery		Condition. C-SGTR prob	C-SGTR Containment bypass Frequency
					AFW	PORV		
Loss Of Offsite Power	3.26E-06	LOOP; EPS fails (SBO); AFW TDP OK; Rapid second. Depres. OK; 182 gpm seal LOCA; Charging flow using TSC DG OK; OPR and DGR in 8 hours fail.	L-Fail	R1/R3, CC-low (182 gpm/p)	0.1		0.02	6.53E-09
Loss Of Offsite Power	1.28E-06	LOOP; EPS fails (SBO); AFW fails; ACP and DGR in 1 hr fail;	L-Fail	CC-high (AFW fails early)			0.4	5.14E-07
Loss Of Service Water System	4.54E-07	LOSWS; AFW fails; SWS recovery fails.	Fails	R3, CC-high)	0.1		0.4	1.82E-08
Loss Of Offsite Power	4.21E-07	LOOP; EPS OK; AFW fails; Feed & bleed fails.	Fails	R3, CC-high)	0.1		0.4	1.68E-08
Loss Of Offsite Power	3.10E-07	LOOP; EPS fails (SBO); AFW TDP OK; Rapid second. Depres. OK; 21 gpm seal LOCA; Charging flow using TSC DG fails; OPR and DGR in 8 hours fail.	L-Fail	R1/R3, CC-high (21 gpm/p)	0.1		0.4	1.24E-08
Loss Of Offsite Power	2.95E-07	LOOP; EPS fails (SBO); AFW TDP OK; Rapid second. Depres. OK; 182 gpm seal LOCA; Charging flow using TSC DG fails; OPR and DGR in 4 hours fail.	L-fail	R1/R3, CC-low (182 gpm/p)	0.1		0.02	5.90E-10
Loss Of DC Power BRA-104	6.59E-08	Loss of DC bus BRA-104; AFW fails; Feed and bleed fails.	Fails	R3, CC-high)	0.1		0.4	2.63E-09
Loss Of AC Bus 6	5.01E-08	Loss of AC Bus 5; AFW fails; Feed and bleed fails.	Fails	R3, CC-high)	0.1		0.4	2.00E-09
Transient	4.40E-08	Transient; ATWS; RCS pressure spike occurs.		CC-high (delta P)			0.4	1.76E-08
Loss Of DC Power BRB-104	3.31E-08	Loss of DC bus BRB-104; AFW fails; Feed and bleed fails.	Fails	R3, CC-high)	0.1		0.4	1.32E-09
Loss Of Main Feedwater	3.23E-08	Loss of MFW; AFW fails; Feed and bleed fails.	Fails	R3, CC-high)	0.1		0.4	1.29E-09

Illustrative Examples

- The quantitative risk assessment method and its application to multiple plants are captured in a technical PRA report prepared by the USNRC/RES.

- Work documented in a USNRC/RES technical PRA report, supported by a Sandia PRA report, and T&H analysis and materials analysis performed in other RES work.

- Method applied to

  4-loop Westinghouse plant 1
  2-loop Westinghouse plant 2
  2-loop Combustion Engineering plant 3

- Includes risk from internal and other hazard categories ("external events") Includes C-SGTR caused by initiating events, as well as RCS conditions after core damage sequences

- The following tables from the report are used to discuss and illustrate the model and its results.

Top 2 Dominant Internal Event C-SGTR Sequences for Westinghouse 4-Loop Plant						
Initiating Event	**Sequence CDF**	**Sequence Description**	**Cause of C-SGTR**	**Mitigation failure prob.**	**Conditional C-SGTR prob.**	**C-SGTR Frequency**
Loss of Offsite Power / SBO	1.74E-06	LOOP occurs; EPS fails (SBO); AFW fails; ACP and DG recovery in 1 hr fail.	CC-high (AFW fails early)	1	0.4	6.97E-07

Sequence 1-1. A loss of offsite power initiating event occurs and results in station blackout; AFW system fails; recovery off offsite and onsite power in 1 hour fail; core damage is postulated.

Sequence proceeds as transient or a very Small LOCA due to RCP seal LOCA without AFW and without feed and bleed. The sequence CDF is 1.74E-06/year; due to the early failure of AFW, and short time window involved, no mitigation is credited. The sequence is assigned a CC-high (high conditional C-SGTR probability) of 0.4. The estimated C-SGTR frequency (F) is

F = 1.74E-06 * 1 * 0.4 = 6.97E-07/year.

Top 2 Dominant Internal Event C-SGTR Sequences for Westinghouse 4-Loop Plant						
Initiating Event	**Sequence CDF**	**Sequence Description**	**Cause of C-SGTR**	**Mitigation failure prob.**	**Conditional C-SGTR prob.**	**C-SGTR Frequency**
Loss of Offsite Power / SBO	3.35E-06	LOOP occurs; EPS fails (SBO); AFW OK; Rapid secondary depressurization OK; 21 gpm SLOCA; ACP and DG recovery in 8 hrs fail.	R1, CC-high (21 gpm/p)	0.1	0.4	1.34E-07

Sequence 1-2. A loss of offsite power initiating event occurs and results in station blackout; Initially AFW system is operable via the turbine-driven AFW pump; rapid secondary depressurization is successful; 21 gpm per pump RCP seal LOCA occurs; recovery off offsite and onsite power in 8 hours fail; loss of AFW pump due to battery depletion is postulated (operator action for pump control is not credited in SPAR models to avoid core damage); core damage is postulated.

Sequence proceeds as a very Small LOCA without AFW and without feed and bleed. Sequence CDF is 3.35E-06/year; due to availability of long time window, mitigation credit for secondary cooling to avoid C-SGTR is credited as 0.10. The sequence is assigned a CC-high (high conditional C-SGTR probability) of 0.4. The estimated C-SGTR frequency is

F = 3.35E-06 * 0.1 * 0.4 = 1.34E-07/year.

Top 2 Dominant Internal Event C-SGTR Sequences for Westinghouse 2-Loop Plant						
Initiating Event	**Sequence CDF**	**Sequence Description**	**Cause of C-SGTR**	**Mitigation failure probability**	**Conditional C-SGTR probability**	**C-SGTR Frequency**
Loss Of Offsite Power	1.28E-06	LOOP; EPS fails (SBO); AFW fails; ACP and DGR in 1 hr fail;	CC-high (AFW fails early)	1	0.4	5.14E-07

Sequence 2-1. A loss of offsite power initiating event occurs and results in station blackout; AFW system fails; recovery off offsite and onsite power in 1 hour fail; core damage is postulated.

Sequence proceeds as transient or a very Small LOCA due to RCP seal LOCA without AFW and without feed and bleed. The sequence CDF is 1.28E-06/year; due to the early failure of AFW, and short time window involved, no mitigation is credited. The sequence is assigned a CC-high (high conditional C-SGTR probability) of 0.4. The estimated C-SGTR frequency is

F = 1.28E-06 * 1 * 0.4 = 5.14E-07/year.

Top 2 Dominant Internal Event C-SGTR Sequences for Westinghouse 2-Loop Plant						
Initiating Event	**Sequence CDF**	**Sequence Description**	**Cause of C-SGTR**	**Mitigation failure probability**	**Conditional C-SGTR probability**	**C-SGTR Frequency**
Loss Of Service Water System	4.54E-07	LOSWS; AFW fails; SWS recovery fails.	R3, CC-high)	0.1	0.4	1.82E-08

Sequence 2-2. A total loss of SWS event occurs and AFW fails. SWS recovery fails; core damage is postulated.

Sequence proceeds as transient or a very Small LOCA due to RCP seal LOCA without AFW. Loss of feed and bleed is not explicitly addressed in the model. The sequence CDF is 4.54E-07/year. It is possible that at least RCS depressurization by pressurizer PORV is available. Because of this, a mitigative credit of 0.1 is allowed, although AFW failed early. Sequence is assigned a CC-high (high conditional C-SGTR probability) of 0.4. The estimated C-SGTR frequency is

F = 4.54E-07 * 0.1 * 0.4 = 1.82E-08/year.

In addition to the above, two dominant “external event sequences for Plant-2 are discussed to illustrate the effect of such sequences.

Top 2 Dominant External Event C-SGTR Sequences for Westinghouse 2-Loop Plant						
Initiating Event	**Sequence CDF**	**Sequence Description**	**Cause of C-SGTR**	**Mitigation failure probability**	**Conditional C-SGTR probability**	**C-SGTR Frequency**
Seismic Event Bin-3 (PGA > 0.5g)	3.80E-06	Large structural failure (35.4% of the original sequence already has either containment bypass or vessel failure)	R5, CC-Unan	1	0.5	1.23E-06

External Event Sequence 2-1E. High-pga Seismic Event

Seismic events with high pga (> 0.5 g) causing failure of structures, where core damage is postulated. The frequency of this seismic event bin for plant-2 is three times the frequency of the corresponding bin for plant-1. There are large uncertainties in which equipment is failed and which equipment may be still available. Conservative assumptions may be present in the classification of this “sequence”.

The initial sequence CDF is 3.8E-06 which postulates large structural failures due to high pga seismic event. The large structural failures considered are reactor vessel, steam generators, RCS piping, containment building, turbine building, and auxiliary building.

For EQK-BIN-3, 35.4% of the original sequence already has either containment bypass or vessel failure. In order to estimate the additional containment bypass risk due to C-SGTR, the original sequence CDF is multiplied by 0.646 (1.0 - 0.354).

External Event Sequence 2-1E. High-pga Seismic Event (continued)

Since this is a high-pga seismic event, no mitigative strategy is credited (no recovery). No credit is given for B5b equipment.

Since the plant experiences high-pga event beyond design basis, the highest conditional C-SGTR probability of cc-Unan (unanalyzed case) = 0.5 is used. (For those sequences that are not studied in the Sandia report, a screening value of 0.5, which is very similar to this highest reported conditional C-SGTR frequency, is selected. Use of such a screening value in PRA acknowledges that information about this parameter is not known: however it is neither deemed to occur with certainty, nor it is deemed to be insignificant.)

Thus, the frequency of C-SGTR leading to containment bypass is calculated as:

F = 3.8E-06 * (1- 0.354) * 0.5 = 1.23E-06/yr.

Top 2 Dominant External Event C-SGTR Sequences for Westinghouse 2-Loop Plant						
Initiating Event	**Sequence CDF**	**Sequence Description**	**Cause of C-SGTR**	**Mitigation failure probability**	**Conditional C-SGTR probability**	**C-SGTR Frequency**
AFW Pump B Oil Fire Occurs	3.23E-06	LOOP; EPS OK; dedicated shutdown panel operations; AFW fails	CC(hi)	1	0.4	1.29E-06

External Event Sequence 2-2E. Internal fire events that result in evacuation of the main control room.

Control of the plant at the dedicated shutdown panel and AFW fails. Both the fire scenario frequency and the equipment that can be credited to deal with the sequence have large uncertainties and possibly conservative assumptions.

Initial sequence CDF is 3.23E-06 which has MCR abandonment and subsequent failure of AFW system.

Since AFW fails and no safety injection is credited, core damage occurs relatively rapidly. Due to the short available time for the operator actions, use of B5b equipment or AFW recovery are not credited.

High conditional C-SGTR probability (CC-hi) of 0.4 is used. This probability applies to transient or an RCP seal leak of up to 150 gpm per pump (RCS pressurized; no relief or small leak).

Frequency of C-SGTR leading to containment bypass is calculated as:
F = 3.23E-06 * 0.4 = 1.29E-06/yr.

Top 2 Dominant Internal Event C-SGTR Sequences for CE Plant						
Initiating Event	**Sequence CDF**	**Sequence Description**	**Cause of C-SGTR**	**Mitigation failure probability**	**Conditional C-SGTR probability**	**C-SGTR Frequency**
Loss of offsite Power/SBO	7.84E-08	LOOP; SBO; EFW fails; OPR and DGR in 1 hour fail.	CC-high (EFW fails early)	1	0.4	3.13E-08

Note that for Plant-3, EFW system performs the same function as AFW performs for Plant-1 and Plant-2.

Sequence 3-1. A loss of offsite power initiating event occurs and results in station blackout; EFW system fails; recovery off offsite and onsite power in 1 hour fail; core damage is postulated.

Sequence proceeds as transient without EFW and without feed and bleed. Sequence CDF is 7.84E-08/year; due to the early failure of EFW, and short time window involved, no mitigation is credited. The sequence is assigned a CC-high (high conditional C-SGTR probability) of 0.4. The estimated C-SGTR frequency is

F = 7.84E-08 * 1 * 0.4 = 3.13E-08/year.

Top 2 Dominant Internal Event C-SGTR Sequences for CE Plant						
Initiating Event	**Sequence CDF**	**Sequence Description**	**Cause of C-SGTR**	**Mitigation failure probability**	**Conditional C-SGTR probability**	**C-SGTR Frequency**
Transient Initiating Event / ATWS	4.54E-08	Transient; ATWS; RCS pressure spike occurs	(CC-high) delta-P	1	0.4	1.82E-08

Sequence 3-2. An ATWS event occurs; RCS pressure spikes beyond RCS design limit in a very short time (failure of RCS pressure relief); core damage is postulated.

In this sequence, steam generators experience sudden high pressure difference between the RCS and the secondary side. The sequence CDF is 4.54E-08/year; since the sequence develops fast, no mitigation is credited. The sequence is assigned a CC-high (high conditional C-SGTR probability) of 0.4. The C-SGTR frequency is calculated as

F = 4.54E-08 * 1 * 0.4 = 1.82E-08/year.

Plant-1 4-Loop Westinghouse single unit on a site

		Event Category	CDF	C-SGTR CDF	% of Total Plant CDF	% of Event Category CDF
1	INT	Internal Events at Power	1.96E-05	8.96E-07	1.55%	**4.6%**
2	FLI	Internal Flooding Events	3.86E-06	1.93E-07	0.33%	**5.0%**
3	FRI	Internal Fire Events	3.35E-05	2.01E-07	0.35%	**0.6%**
4	VL-SLB	Very Large Steam Line Break Events	2.49E-08	1.20E-08	0.02%	**48.3%**
5	EQK-BIN1	Seismic Events Bin 1 (PGA 0.05 to 0.3g)	2.74E-08	1.34E-09	0.00%	**4.9%**
6	EQK-BIN2	Seismic Events Bin 2 (PGA 0.3 to 0.5g)	7.90E-08	1.32E-08	0.02%	**16.7%**
7	EQK-BIN3	Seismic Events Bin 3 (PGA > 0.5g)	6.87E-07	2.99E-07	0.52%	**43.5%**
		Total Plant CDF	**5.8E-05**	**1.6E-06**	**2.8%**	**2.8%**

Plant-2 2-Loop Westinghouse single unit on a site

		Event Category	CDF	C-SGTR CDF	% of Total Plant CDF	% of Event Category CDF
1	INT	Internal Events at Power	1.17E-05	6.06E-07	1.6%	**5.2%**
2	FLI	Internal Flooding Events	1.70E-07	2.92E-10	0.0%	**0.2%**
3	FRI	Internal Fire Events	2.17E-05	2.13E-06	5.6%	**9.8%**
4	VL-SLB	Very Large Steam Line Break Events	2.17E-09	1.31E-09	0.0%	**60.7%**
5	EQK-BIN1	Seismic Events Bin 1 (PGA 0.05 to 0.3g)	6.84E-09	8.15E-10	0.0%	**11.9%**
6	EQK-BIN2	Seismic Events Bin 2 (PGA 0.3 to 0.5g)	4.89E-07	7.77E-08	0.2%	**15.9%**
7	EQK-BIN3	Seismic Events Bin 3 (PGA > 0.5g)	4.10E-06	1.24E-06	3.3%	**30.4%**
		Total Plant CDF	**3.8E-05**	**4.1E-06**	**10.6%**	**10.6%**

Plant-3 2-Loop Combustion Engineering Plant on a 2-Unit Site

		Event Category	CDF	C-SGTR CDF	% of Total Plant CDF	% of Event Category CDF
1	INT	Internal Events at Power	1.17E-05	6.06E-07	4.1%	**5.2%**
2	FLI	Internal Flooding Events				
3	FRI	Internal Fire Events				
4	VL-SLB	Very Large Steam Line Break Events	7.73E-10	6.72E-10	0.0%	**87.0%**
5	EQK-BIN1	Seismic Events Bin 1 (PGA 0.05 to 0.3g)				
6	EQK-BIN2	Seismic Events Bin 2 (PGA 0.3 to 0.5g)				
7	EQK-BIN3p	Seismic Events Bin 3p (1.0g > PGA > 0.5g)	1.96E-06	9.70E-07	6.6%	**49.4%**
8	EQK-BIN4	Seismic Events Bin 4 (PGA > 1.0g)	1.03E-06	5.00E-07	3.4%	**48.5%**
		Total Plant CDF	**1.5E-05**	**2.1E-06**	**14.2%**	**14.2%**

Summary

- Both the NRC and the industry have studied the potential C-SGTR events in detail for the last decade and a half, as evidenced by multiple technical reports in this area.
- The conditions and sequences that can lead to C-SGTR are well studied and understood.
- Uncertainties are recognized; however reduction of these uncertainties has not been achieved yet.
- An illustrative PRA estimate of the C-SGTR bypass frequencies for selected plants, considering internal and external events using a straightforward and easily reviewable estimation method based on CDF sequences and their characteristics has been completed.
- The method is exercised in a prudently conservative manner due to the existence of large uncertainties and large number of sub-scenarios and conditions.

Conclusions

- *Key insight - the fraction of CDF from potential C-SGTR sequences is lower or at the same order of containment bypass fraction that may exist for the currently postulated containment bypass fraction of internal events (e.g. 0.1 or less).*
    - *C-SGTR is not a negligible part, nor a main contributor of total plant risk. It should be considered and monitored in plant risk assessments in a manner commensurate with its expected importance for each plant.*
- *An otherwise "successful" accident sequence turning into a core damage sequence due to C-SGTR is not a major contributor to plant risk. However, an accident sequence initially progressing as core damage turning into a containment bypass sequence due to C-SGTR may not be negligible.*

Recommendations

Technical recommendations:

- Plant PRAs should address C-SGTR in their evaluation of plant LERF or level 2 analyses on a plant-specific basis, referring to the existing PRA standard.

- An in-depth and intrusive modeling of C-SGTR within the current level 1 models is most likely not necessary. Dominant accident sequences which are potential precursors of C-SGTR are well defined and identified. They can be collected and processed for their contribution to containment bypass by the users of the PRA models. Simple streamlined methods may be used to collect these dominant sequences from the existing specific PRA models.

- Care must be taken not to take excessive credit for SAMG actions, other recovery actions, and non-safety-related equipment to avoid C-SGTR, especially for external event scenarios.

Recommendations (continued)

Further comments:

- During the last decade, developments that are favorable to both the addressing of the C-SGTR in risk assessment and reduction of its potential impact have occurred:
  - The PRA standard recommends addressing C-SGTR;
  - Severe accident management guidelines (SAMG) would help reduce the C-SGTR conditions, in case of a low-frequency severe accident sequence;
  - Recent security-related enhancements of equipment and procedures may help reduce the C-SGTR conditions, even if core damage scenarios occur;
  - Many plants have replacement SGs that could reduce the C-SGTR precursors.

Recommendations (continued)

- If more credit is given to SAMG actions, and recovery actions, the fraction of containment bypass estimated for C-SGTR may be shown to be lower. The EPRI report on the subject shows an illustration of this.

Programmatic recommendation:

- Task 3.5 of the SGAP has been completed and should be closed.