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PUBLIC NOTICE
BY THE
UNITED STATES NUCLEAR REGULATORY COMMISSION'S
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NOVEMBER 6, 1996

The contents of this transcript of the proceedings of the United States Nuclear Regulatory Commission's Advisory Committee on Reactor Safeguards on NOVEMBER 6, 1996, as reported herein, is a record of the discussions recorded at the meeting held on the above date.

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

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARD (ACRS)

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JOINT MEETING

MATERIALS AND METALLURGY

AND

SEVERE ACCIDENTS

SUBCOMMITTEES

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WEDNESDAY

NOVEMBER 6, 1996

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ROCKVILLE, MARYLAND

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The Subcommittees met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 8:30 a.m., Robert L. Seale (Chairman, Materials and Metallurgy Subcommittee) and Mario H. Fontana (Chairman, Severe Accidents Subcommittee) presiding.

COMMITTEE MEMBERS:

ROBERT L. SEALE, Chairman, Metals & Metallurgy

MARIO H. FONTANA, Chairman, Severe Accidents

COMMITTEE MEMBERS (continued):

THOMAS S. KRESS
DANA A. POWERS
WILLIAM J. SHACK

ACRS STAFF PRESENT:
NOEL F. DUDLEY

ALSO PRESENT:
JACK STROSNIDER
JOSEPH DONOGHUE
STEVE LONG
JEFF GORMAN
ROBERT PALLA
RAY SCHNEIDER
CHARLIE TINKLER
ROBERT JONES

A-G-E-N-D-A

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

(8:31 a.m.)

MR. JONES: Joe Donoghue will be speaking for the staff.

CHAIRMAN SEALE: The meeting will now come to order. This is a meeting of the ACRS Joint Subcommittee on Materials and Metallurgy and Subcommittee on Severe Accidents. I am Robert Seale, Acting Chairman of the Subcommittee on Materials and Metallurgy.

The ACRS members in attendance are Mario Fontana, Chairman of the Subcommittee on Severe Accidents, Tom Kress, Dana Powers and William Shack.

The purpose of this meeting is to continue discussions with representatives of the NRC staff, the Nuclear Energy Institute and EPRI to gather information concerning the technical approach used in developing the proposed risk-informed performance-based rule and regulatory guide associated with steam generator tube integrity.

The committee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions as appropriate for deliberation by the full committee. Noel Dudley is the cognizant ACRS staff engineer for this meeting.

The rules for participation in today's meeting

have been announced as part of the notice of this meeting previously published in the Federal Register October 12. The transcript of the meeting is being kept and will be made available as stated in the Federal Register notice. It is requested that the speakers first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

We received no written comments or requests for time to make oral statements from members of the public.

During the June 12-14, 1996 ACRS meeting, the committee heard presentations by representatives of the staff in the Nuclear Energy Institute on this matter. Today the subcommittee will hear from the staff concerning the technical basis for the proposed rule.

We will begin the meeting. I guess Joe, you'll start things off for the staff. Is that correct?

MR. DONOGHUE: Yes, sir.

CHAIRMAN SEALE: Mr. Donoghue.

MR. DONOGHUE: At this time, yes. I am coming through okay?

My name is Joe Donoghue. I am in Reactor Systems. What we wanted to do this morning was to step through the example calculation that we've done to assess severe accident risk of tube failure.

To start with, let me show you the list of things I'm going to talk about. I wanted to start off just with a general discussion of what contributes to tube rupture risk. Then our emphasis again, the severe accident contribution. These bullets go through the thought process and significant portions of the analysis, the example analysis that we've done.

What we plan on doing is having the people here that are key players in each of those areas, so that if we get into a lot of detail discussion, they can p .ch in. In the interest of time, we weren't planning on having separate presentations. I'll try to go through as much of that material as I can.

Back in June, we basically said that there's two ways to have tube ruptures, either spontaneously or induced by some means. We said at the outset that spontaneous failures, we didn't expect the frequency to change under the rule.

Under induced failures, we had listed pressure and thermally induced failures and contributors. There was a question from the committee about the potential for mechanically induced failure. I'll address that in the next slide.

Just to repeat a couple of things from June, when we looked at pressure induced failure, there's a

chance in an ATWS you get to a high pressure that could challenge the tubes, but the frequency of that event is low enough that we don't consider it to be a significant problem.

Secondary depressurization is a somewhat different matter. Main steam line break kind of accident you heard discussed yesterday could lead to a bypass situation which was assessed in NUREG-0844. The risk there was in the low to mid 10 to the minus six range for a bypass. We think that we understand that risk, it was based on the five percent pressure induced conditional failure for tubes, which is what you heard talked about in the reg guide. So again, we don't expect that to change under the rule, that frequency.

MEMBER POWERS: Could I come back to this assumption that the rule has no impact on the spontaneous failures?

MR. DONOGHUE: Yes.

MEMBER POWERS: That seems a remarkably conservative position. I mean that says that spontaneous failures never occur in monitored tubes.

MR. DONOGHUE: What we are saying is that we don't have a reliable way to assess the impact that the provisions of the rule are going to have on --

MEMBER POWERS: If you take them both as

randomly occurring, why can't you come up with a reasonable estimate?

MR. DONOGHUE: Well, we understand what the history has told us of spontaneous rupture. We know that people have voluntarily in different degrees in different plans, voluntarily implementing measures that go beyond the current requirements.

But it is difficult for us to tell exactly what those different inspection methods or different leakage monitoring methods for instance, contribute to the ability to prevent spontaneous tube ruptures, because a lot of the times, these seven U.S. or nine internationally tube ruptures came from things that people didn't expect to happen under the rule, although the inspection requirements are tightened up.

People are still looking for things that they have had in their generator, some kind of degradation that they have already seen, and not necessarily looking for any type of degradation anybody has ever seen in the industry. It's just not practical to do that.

So it seems -- it is somewhat conservative. You would hope that some of these provisions would help prevent ruptures more at some plants maybe that haven't done as much as they could, but in general, it's hard to assess what the overall risk impact would be.

MEMBER POWERS: If a licensee came in with an assessment on this, you would accept it?

MR. DONOGHUE: Say that again, sorry.

MEMBER POWERS: If a licensee came in with an assessment and said my rule is going to affect the spontaneous failure frequency from unknown mechanisms --

MR. DONOGHUE: Well, you saw yesterday that there's a limit that we placed that's based on historical precedent. That will be evaluated. If it's a lower number, and it seems like it was done in an acceptable way, yes.

It's just that for generator purposes, we didn't see a way to say that it was going to be a benefit.

MR. LONG: This is Steve Long with the Risk Assessment staff. What we are really trying to do here is not spend a lot of effort talking about the spontaneous rupture frequency.

The industry on the other hand, did submit an analysis to us or at least a study, let's call it, where they assume that the rule would have the spontaneous rupture frequency. They took credit for that.

We see a lot of things going on here that we don't think we can assess accurately enough to make that kind of a conclusion. For instance, the rule would allow detective flaws to remain in service, which we're not

doing now. We also have acceleration of degradation in some plants which people are trying to figure out how to control, but which frankly, they are not controlling as well as they hoped they would.

So there are trends where we may be allowing the spontaneous rupture to go up. There are also factors which may be trying to bring it down which is trying to improve the quality of the inspections.

We just don't see any way right now of saying if we think it's going to go up or down, and we don't think it will move very much right away, and we didn't spend much more effort on it. I think that's really what Joe is trying to say.

MEMBER KRESS: The only real way to do that I guess is let time pass, see how it affects it in reality.

MR. LONG: Yes. There is simulation modeling, which is something we really haven't done. One member of the staff is looking into it.

But in order to project the affect of the distribution of flaws that we think we're going to end up with, we have to do a lot of things that include projecting the initiation rate of flaws, the growth rates of flaws, the ability to detect and remove the flaws before they grow to certain sizes, and then figure out what we've missed.

How many of those grow to the point where they would rupture if they were at normal operating pressure and temperature? How many would rupture if they got an abnormally high delta P? And how many would rupture if they got an abnormally high delta P and an abnormally high temperature? That's beyond where we have been able to do the modeling right now.

CHAIRMAN SEALE: I was going to ask this later, but since you brought it up, I will ask you now. Yesterday we made considerable of the fact that there's only one mode or one mechanism of failure, one type of flaw, if you will, that you feel is characterizable to the degree that you can really know enough about the crack in order to exercise judgement as to whether or not you might not play it, okay.

MR. LONG: You are talking about the actual cracking of the tube support blades?

CHAIRMAN SEALE: Yes. I mean the rest of them you are saying you can't really characterize the depth of penetration, so you are going to have to plug it anyway. Right?

MR. LONG: At this point, and when we get into the discussion a little bit further, you will see that we have had to make some estimates of what the flaw distributions are because we don't have measurements.

CHAIRMAN SEALE: Okay. Now let me ask you this. What fraction c= the flaws you detect are of that variety?

MR. LONG: It is going to vary greatly from plant to plant. There's a very large number of flaws. The example that the Office of Research, Dominion Engineering put together, I'm trying to remember. Jeff can shake his head, but I think the majority of the flaws were actually under the tube support blades. The second largest number of them were in freespan sludge pile area.

We don't believe that the flaws that are under the tube support blades are going to be able to rupture under the scenario as we look at it. They are constrained. So the studies you will hear about shortly really disregarded those from the burst calculations. We were unable to deal with the circ. cracks which are a smaller number yet, but still perhaps significant.

So we have some examples, but we can't say we have comprehensively covered the area here.

CHAIRMAN SEALE: Well, are you going to tell us eventually how many steam generator tubes you think you might be able to "save" if you go to this rule?

MR. LONG: I don't know that saving -- I don't think we will probably ever be able to tell you that. I think saving may not be the right answer anyway.

CHAIRMAN SEALE: Delay. Yes.

MR. LONG: I think the industry probably has some estimates of that, but we don't.

CHAIRMAN SEALE: I would be interested in knowing that. I sometimes wonder if the banquet is worth the price, depending on what those numbers turn out to be.

I'm sorry. Go ahead.

MR. DONOGHUE: The next statement is sort of wrong. Let me try to clear it up. Previous studies where people actually thought about severe accident tube rupture risk and I should have added the words thermally induced, really said that the tubes wouldn't be challenged. Of course as I mentioned before, NUREG-0844 did consider containment bypass from a pressure induced rupture leading to core damage.

So this statement should be -- I'll change it on the next presentation I give.

MEMBER KRESS: Is that because it would fail the hot leg first and depressurize?

MR. DONOGHUE: Well, hot leg or surge line. Some other component would fail earlier. Although there were some studies done I think subsequent to NUREG 1150, where people, I think expert panels, tried to estimate what the effect of flaws on tube survivability would be. It wasn't really a thorough assessment or as thorough as

been done now.

Their conclusions were that they didn't expect to be a large contribution to bypass risk.

So since then, we have come up with I don't know if it's a better understanding of degradation mechanisms, along with a combination of new degradation coming up. You have heard about that before. But also, we have developed some more information that we can use to understand high temperature to performance. We'll talk about that later on this morning.

The understanding has led to a couple of questions when we started thinking about this. We wanted to make sure we understood what the predicted reactor coolant system conditions were, especially of the tubes. That's something that although people had calculated these things before, hadn't really dove into the uncertainties involved. The committee has considered that at length, I know.

But also the other parts of the reactor coolant pressure boundary response are things that I will talk about, but we really haven't quantified them for this risk study. I'll explain that later on.

As I mentioned, mechanically induced tube failure wasn't something we spent a lot of time on. These are the reasons why. I think Dr. Catton was asking about

this in June.

What we went and looked at were two events, North Anna and Mihama, where circumferentially cracked tube could have impacted other tubes and caused cascading failures if the adjacent tubes were weak enough or impacted.

In North Anna, there were tubes adjacent to the failure site that were in service. So here is in both of these events, were good tests of this theory to see if tubes that were pressurized and in service could have failed if a tube started whipping around, so to speak.

In the follow-up report in the North Anna event, the eddy current examination didn't report any damage to tubes adjacent to the break site.

Now in Mihama, they did a lot more extensive studies after the fact. After we went in I think with video cameras, we're able to look at the tubes adjacent to the failure.

What they basically said was that although there was some evidence of contact, deposits on the outside of these tubes adjacent to the break shows that there was contact, there wasn't any denting. It didn't seem that there was any significant reason to think that circumferentially cracked tube or broken tube was going to cause subsequent failures there.

The latest information that we have seen is associated with the sleeving campaign that occurred in Maine Yankee, where they did some calculations to justify that they were within design basis when they were operating with the four, I think it was four, circumferential cracks that they found. The significance of the circumferential cracks that they found.

They showed that the impact in jet impingement forces for a broken tube down at the tube sheet, which would be a pretty bad case, because you have a large length of tube that's able to move around, and you also have other tubes adjacent to it that were circumferentially cracked. They were able to say that the forces were smaller than what they said were the shear load margins for a cracked tube. So they even tried to figure out what a cracked tube would be able to withstand on that situation.

CHAIRMAN SEALE: Do you have any information on the in-service history of Mihama? That's been five years since that event.

MR. DONOGHUE: I think those steam generators were taken out of service. Is that correct?

MR. STROSNIDER: This is Jack Strosnider. The Mihama steam generators were replaced.

In fact, it's very interesting. The one that

the failed tube was in is now in a museum. You can actually go in and look at those tubes. But they have been replaced with a new generator.

CHAIRMAN SEALE: I see. Thank you.

MR. DONOGHUE: So now I wanted to step into the thermally induced tube rupture analysis that we have done.

Before I really talk on this slide, I just want to set us up to mention that when we started this exercise, w again, we understood what was done in the past, but we were taking what we thought was a new look focusing on the tubes and we tried to get as much information as we could in a relatively short amount of time.

What we found is that there was enough uncertainties and plant specific and design specific features that we're going to have to talk about later on this morning, that we have called this not a generic analysis that covers all PWRs. What we're saying is that this is an example analysis and assessment of one plant design using that specific information.

We tried to branch out in some ways to see what other designs and other plant information, how it might influence what we found. But as I will say later on, there's more work that one would have to do to be able

to cover all PWRs. But this gives us insights that we think we need for the basis for the rule.

Okay now, a couple of pages passed page four here, is the event `e. Now I think Noel Dudley has given out a separate page so that you don't have to flip back and forth every time I refer to it. But it is in your package, I just gave out a separate one for reference.

The first part of that event tree starts out with what we are calling Hi/Dry events. These are events for Surry now, this is our example plant. It's predominantly station blackout. I'll talk about that later.

But right now, when I say Hi/Dry, that means an elevated primary secondary differential pressure. The steam generator is at a dry core uncovery. This means that we have a station blackout with a loss of auxiliary feedwater. That's important.

You need, to have a temperature challenge, you have to have the steam generators dry. If you are able to supply secondary site water, the steam generators will remain cool enough that we don't expect there to be a challenge or a significant challenge. Of course you need the high differential or some elevated differential pressure.

Now as you step through the event tree, you'll see that there's questions on the primary and secondary integrity that are asked. They are asked more than once. But we start out with early questions, early in the event tree. If you look on there, it's top events A through E. I'll put the event tree up in a minute.

But we were asking questions about whether or not the primary system stays in tact, if there's other failures such as a stuck open PRV or a seal LOCA for a coolant pump.

In a secondary system, if the steam generators remain in tact again, you don't have the high delta P. But if you have a failure of a secondary side valve or if the operators are directed in APGs to manually depressurize, those kind of things are factored into the event tree.

MEMBER POWERS: I look at this event tree and I see a variety of questions that are answered with a 50:50 split.

MR. DONOGHUE: Right. I'll get to that in a couple of pages, if you don't mind holding on.

Go ahead and ask it now. I mean I'll just --

MEMBER POWERS: Well isn't that a mark of trying to decompose rare events down in a way that you have no business doing?

MR. DONOGHUE: Okay. What you are referring to is in here, right?

MEMBER POWERS: Right.

MR. DONOGHUE: Top events G and H, where we are -- there's not a lot of information that was available in 1150 to tell us what the PORV or secondary side safety valve reliability would be under these conditions.

Now we have done some sensitivity studies. I'm going to talk about closer to the end of the presentation where we saw some information at EPRI or some assessments that EPRI had done discussing primary and secondary system valve reliability. We used those as sensitivity studies to see what the impact there would be.

But this is based on basically a lack of information to really say one way or the other.

MEMBER POWERS: I could put an infinite number of questions in between G and H where I have no information, and take a 50:50 split on them, and presumably drastically affect the predictions on the ends of those cut sets.

MR. DONOGHUE: As I said, we tried to do sensitivity studies to see what other information would do to change the results, just based on changing these assumptions. I'll try to point that out later.

MEMBER POWERS: My question is, if you don't

have any information, why are you decomposing the tree down this way?

MR. LONG: This is Steve Long with the Risk Assessment Branch.

First of all, we inherited a bunch of 50:50 splits from the NUREG 1150. Some of them are implicit on the lefthand side of the tree. Things that we have put in there that have 50:50 splits are in there so we can do sensitivity studies.

I think I understand where you are going, that if you put in enough 50:50 splits with uniform probability distributions, it's essentially useless for a parametric uncertainty study.

We don't believe that we are in a position to do a parametric uncertainty study without further information. However, by putting the structure there, it allows us to do sensitivity studies where we ask what happens if. It just makes it an explicit question in the tree. We think the ones that are there are critically important. We didn't want to just ignore them.

Without those questions, we don't really know what the answer is. With the questions there, we can explore potential ranges for the answer, we still don't know what the answer is until we get the right number for the splits.

MR. DONOGHUE: And it's possible under the rule that if these are sensitive parts of the event tree that affect risk, that a licensee could take actions to just gain information or actually try to correct what might be deficiencies in the reliability of those components. Then we can use this to assess the impact of those actions.

CHAIRMAN FONTANA: I don't know if I missed something, but what was RC-3, which you don't have any of?

MR. DONOGHUE: Okay. I'll explain those. Those are just release categories.

CHAIRMAN FONTANA: Yes. I see.

MR. DONOGHUE: RC-3 was left over from a previous use of this event tree and matrix, that's all.

CHAIRMAN FONTANA: Okay.

MR. DONOGHUE: The last bullet on page four. We included pressure induced rupture probabilities. Let me flip back and forth on you again, under at the top of event F.

Because during these events, before you even get to the high temperature condition, you are going to have a pressure challenge. So you'll see values here which are pressure induced rupture, but this is not -- I don't want you to confuse this with the pressure induced rupture event leading to core damage. See, we have

already started out with core damage here. So this doesn't include all of the assessment that was done in NUREG-0844.

Now as I said, the event tree kind of -- it asks some questions about primary and secondary more than once. We have the full core uncovery or at core uncovery, it says, but the numbers from A, from top events A to E, are the questions we're asking early. We begin to later questions about primary and secondary integrity on the righthand side of the tree.

MEMBER KRESS: That -- excuse me. That quantification of the probability event would be specific for a specific steam generator, because you have to know the flaw distribution. So that is why you say this is not generic or one of the reasons this is not a generic --

MR. DONOGHUE: Well, I'll be talking about flaw distributions later on. But you would like to have a plant specific flaw distribution to understand the risk at a plant. The flaw distributions that we have are what we're calling representative.

In some cases, you can get some more information from one plant and another, and you don't necessarily want to penalize a plant where replacement generators. So it's hard for us to -- when we get to that discussion, you'll see I think the examples we use don't

necessarily cover every possibility either. As other people I think that were present or are present, they can talk about the flaw distributions in detail.

Understand that a specific flaw distribution will change not only this question, but also the end result here where we actually ask ourselves what's the chance of temperature induced tube failure. Because the flaw distribution plays in both of those top events.

As I was saying, the late questions that we're asking are whether or not PORV safety valve is affected after core uncovery from the high temperatures.

And then besides asking the question earlier on about safety valve reliability on the secondary side, we found that we didn't have any information to really tell us what's going on with the MSIV integrity. They are not in tech specs, they are not included requirement for MSIV leakage.

It's possible that, just some anecdotal evidence that we've seen, that even though you think the generator is isolated, that over the period of time that we are considering for this kind of event, that the steam generator could become depressurized. That's something that we have included in the event tree, but we have really no information to tell us what the chance of this happening is.

Again, if this is something that we in the end think is very important to know, I think people would, might be inclined to try to get some information to tell us one way or the other if we should discount it.

Again, the final branch, the final top event on the event tree, I think I have already kind of pointed to, but this is where -- this top event I, for temperature induced tube ruptures, where all the flawed tube failure model, thermal hydraulics assessments, information feeds into finding out what the numbers are or the estimates of the numbers that go into that, into those branches.

We've already had -- let me talk a little more about the event tree itself. You see that we start out again, this is core damage, Hi/Dry condition. The numbers that we had, I'll talk about the quantification in a little more detail. But again, for Surry, what you are seeing on here, the 1150 numbers are represented up to -- other than E, using the representative flaw distribution. There's two that I will discuss. One that we're calling the research flaw distribution. The other one, the NRR flaw distribution, for their sources.

We used what we think is the slightly more conservative flaw distribution. But I'll talk about the problems we have with each of those distributions in a minute.

What you see is that there's a partitioning for seal LOCA, for whether or not the relief valve was stuck open, the SOR was stuck open relief valve. Then we asked questions about whether you have the steam generators old or just one depressurized. I'll talk about that more in a second.

Now here, just to point out some of the labeling. When we try to characterize the conditions that the plant would be in under each of these end states, we used some thermal hydraulic analysis that I'll mention in a few minutes. These designators just help us keep track of which event we're using. I'll discuss those later on. The 3Rs, the 6N and so forth, are just thermal hydraulic events designators, just to keep us straight.

As was asked earlier, these release categories, RC-3 is meaningless for us here. The RC-1 is the release associated with a tube failure and a path through the secondary.

The other, RC-2, which there aren't many of, is a little differential pressure situation, but where you could have some leakage to the environment. I'll pick one example. Here, where you have the primary in tact, but when you ask questions about the secondary integrity, you could depressurize the steam generator at that point. Then you could have a path in the environment through tube

failure.

Again, I would say keep a loose copy of this handy, because the next few pages still refer to it.

I have talked about Surry, but make sure we weren't out in left field just by taking one plant as our example. 1150 information from Sequoyah was looked at, although not explicitly included in the event tree. We also took a look at the IPE data base, at least for a few of the -- at least for some of the information from the first couple of top events.

You see what we came in with as entry frequency. This is mostly station blackout. There are some other contributors from other events, loss of feedwater and DC bus. But as I say for Surry, the majority of the contribution is from station blackout.

MEMBER POWERS: This is the contribution to the Hi/Dry scenario?

MR. DONOGHUE: Right. Then again, when we went and looked at the IPE data base, we saw that there was a range for this number. I think as low as 10 to the minus seven range to as high as 10 to the minus four for some plants.

What I want to point out early on here is one reason why we're calling this an example calculation, we took a number that we thought was reasonable, but it

doesn't represent all the plants.

The end results here should be considered over a range. We didn't go and do a study for one plant at one end of the range and the other. But we want you to understand that the results are going to be covering a range.

That's important, because later on when -- the reasons why we are putting things into the rule and reg guide is predicated on not understanding.

I just wanted to point out the plant specific differences also come up when we are trying to look at the particular, the quantification of the particular branches in the event tree. You see, for example, the RCP seal LOCA contribution varies just between Surry and Sequoyah.

We have considered for sake of the sensitivity studies I'll talk about later on, the differences between CE and Westinghouse plants are significant there. The integrity of the seals is something that is going to vary.

Then here for the PORV or safety valve failure, it doesn't vary much for between Surry and Sequoyah, but understand that other plants -- there are RC plants out there without PORV, so you are just relying on the safety valve integrity. There is some information that's been developed by industry on that. Try to factor some understanding of that into our sensitivity studies.

Here's a large difference that we've seen, just between Surry and Sequoyah. The secondary system integrity, which isn't again -- this is important. We see that the vast majority of the events that are going to lead to thermal challenges of the tubes have a depressurized steam gererator.

You can just see between our example plant Sequoyah, there's a huge difference there. When Bob Palla, a man who spent most of this time on this event tree, went and looked, he found that there were instructions for the operators that vary between the two plants on how to handle the secondary side depressurization or failure or an ADV during one of these events that affected these frequencies.

So this just points out again, it's going to happen over and over again during this presentation, that there's a lot of plant-specific and design-specific dependencies here that we found as we tried to do this example analysis.

We have talked about the .5 assumption that's in the event tree. I will just point out that the industry has submitted for our information a report depicting safety valve, although it's not PORV I don't believe, but safety valve and secondary side safety valve liability.

We have some questions about what's in there, but again, we tried to use that information, at least to put into our sensitivity studies, to see what the effect would be. We would have to really understand a lot more of the basis for that kind of information before we would accept it.

Right now, we're just -- there's really little information, for example, on PORV operability under the effects of these superheated steam situations.

MEMBER SHACK: Does it make a dramatic difference though? Is it critical?

MR. DONOGHUE: Well, it sure can, yes. The steam side safety valves, for example, if they basically never fail, you'll see in I think some of the sensitivity studies that can help out a lot, by reducing the chance that you are going to have a high differential pressure.

MSIV leak rates again, I mention that earlier in another page that there's just not information for us to make an assessment on whether or not isolating a steam generator is going to save you here. I just need some information by MSIV integrity that we don't have.

I have already mentioned what we tried to do to characterize the end states. When I say a limited variety of scenarios, what I mean there is that we had the benefit of thermal hydraulic studies that covered what we

thought we going to be the important risk contributing scenarios.

There are a couple of questions now after we've gone through the evolution here of putting numbers into the event tree and understanding tube failure probabilities, to obtain a bypass number, where there's other questions that have come up. For example, one I'll mention is RCP seal LOCA contribution to the risk, where we don't understand the effect of opening up a PORV during one of those RCP seal events. That's something that we would like to have included in the event tree, and we used the information we had to as best we could characterize it.

But by and large, the events that we tried to use, and I'll list them in a few pages, are suitable for most of the branches on this event tree.

I'll talk about the temperature induced tube rupture calculation later on. When I talked about the event tree, I talked about the -- here we ask a question about whether or not the operators depressurize all the generators, or whether you have a failure that depressurizes one generator. We included that and have probabilities of pressure induced temperature, induced rupture based on that.

Also, as I mentioned, you have RCP seal LOCA

which has a couple of effects. I'll mention them later on. But that was included. A lot more work has been done just recently because we had some surprises by RCP sealed LOCAs, the magnitude of their contribution here.

The end result, when I get to results, I'll just let you know now, this is what we come with in our example calculation. It's very small, but it's very big as far as significance. We see a low to mid 10 to the minus six number for our example plant for containment bypass under this situation.

Remember that the frequency coming in for the Hi/Dry events should be considered over a range, meaning that the end result should be considered over some range. Without going through an analysis, I don't think anybody should ever say it's going to change proportionately based on just putting in a new beginning number here, but the plant-specific and design-specific features would have to be integrated into an assessment like this to get an understanding of what this range really is.

But enough said. I think just remember it's over a range.

Any questions?

CHAIRMAN FONTANA: Would the RCP seal LOCA always include loop seal clearing?

MR. DONOGHUE: No.

CHAIRMAN FONTANA: It doesn't?

MR. DONOGHUE: No. What I'll say is that --

CHAIRMAN FONTANA: I may have missed something.

MR. DONOGHUE: Well, there's been some work to understand that, but right now what we know is that for the Surry calculation, there's a potential of clearing one of the loop seals. The Russian roulette here is whether or not it's to seal on a depressurized generator.

If the steam generator remains pressurized, it's not going to have as large a tube channel.

MR. PALLA: let me just clarify. This is Bob Palla with the staff. We did assume that we always cleared one loop seal. We did not assume that we knew where it was. We assumed that it could occur in any of the three generators. We looked for whether the cleared loop seal might be coincident with the same branch, the same loop that may have been depressurized on a secondary side.

For example, if we had a scenario where we just depressurized one steam generator, it would be a one in three chance that you had a cleared loop seal in that scenario.

We looked at the probabilities of thermally induced rupture for each of the three loops under various

combinations.

CHAIRMAN FONTANA: Thank you.

MR. DONOGHUE: Now we'll be talking about the event tree somewhat later on, but I think we've gone through the phase where we are flipping back and forth. I am going to move onto a short discussion about the thermal hydraulic analysis that we used.

The reason I think I am able to keep this somewhat short is that the committee has heard quite a lot about thermal hydraulic analysis and their basis. I just wanted to point out the cases that we used in the risk results that you are hearing today.

Again, the base case, station blackout, loss of auxiliary feedwater. Then we move into that same scenario where you have a steam generator depressurized, or all steam generators depressurized, and then the RCP seal LOCA case with one steam generator depressurized. Then another case where the PORV fails open, but the PORV failure is inserted at the time where operator guidance would have an operator depressurize a primary. So it could be considered PORV open by the operator also, if guidance included that with a steam generator depressurized.

Just I promised I wouldn't flip back and forth, but just real quickly just to show you so you are

not confused. The R or the N corresponds with, for example 6N is NRR-6 up here.

This is just, again, these are the scenarios we had available to us to use over the summer when we were trying to quantify the event tree and get the tube failure probability numbers generated.

I'll talk later about some studies that research has done to understand the sensitivity of the thermal hydraulic results. We understand that there's some changes to the results from some of these cases. We'll have to -- we have taken a quick look at some of that information, but that's going to have to be incorporated into our risk study.

But as I say down here, jumping around on my slide already, the results appear -- the results that we used for this risk study appear somewhat conservative compared to what newer information may be telling us. But we just needed to take a closer look at it to see what the implications really are.

When we do the tube failure analysis, I'll point this out in a little more detail. But what we did was estimated tube conditions, but the failure probabilities is calculated relative to the failure probability of other components, other major components in the RCS, namely the surge line and the hot leg.

SCDAP/RELAP, as you understand, I think calculates a creep damage index for those components, for the tubes, to surge on a hot leg. What we did was use Larson-Miller modeling for that creep failure to try to understand the failure potential of those components relative to one of the --

MEMBER POWERS: Is Larson-Miller an appropriate thing to use for the large section?

MR. DONOGHUE: For the hot leg?

MEMBER POWERS: Yes.

MR. DONOGHUE: The way it was done in SCDAP arguably could be done better. There has been some discussion of the staff on ways to do a creep failure calculation using more than just one thickness of the pipe. But for our purposes, since we're not considering really exotic materials for the hot leg or the surge line, and the information is well developed, we think that using Larson-Miller is an adequate way to do this. I don't think we are going to gain a lot by --

MEMBER POWERS: Is Larson-Miller really well developed for failures? Are these relative to the high temperatures for thick section steel that's not in a uniaxial loading?

MR. DONOGHUE: Well, we had some information from previous studies at high temperatures that was used

to -- and I think only show you later on the tube unfortunately. But there were uncertainty bounds put on the Larson-Miller parameters for both the surge line and the hot leg at these temperatures, I think over a temperature range. It seemed like it was -- I can give you some more information on that because we do have some more detail studies that were done to characterize Larson-Miller for us. I don't have the material with me.

MEMBER POWERS: I guess I would like to see it because I know that in looking at the BWR dry well failure, which is actually a less difficult case than this, that elaborate effort was made to improve upon the Larson-Miller formulation simply because they didn't have a good data base at the temperatures where failure would occur. They worried a lot about it not being in a uniaxial load.

MR. DONOGHUE: That second point has definitely been discussed, but just to talk about the Larson-Miller modeling by itself, I think we have some information I could show you to at least tell you what temperature range was considered.

Now as far as what the loading on the particular components was, the SCDAP/RELAP certainly doesn't get you that far. It is something we have discussed in our NUREG that we are putting together to lay

out all this analysis.

There is at least a qualitative discussion of that. I think we leave it open for a further analysis.

I'll talk later on about the sensitivity studies that were done which are very recent studies on the thermal hydraulic results. That just gives you an outline of what was used to characterize the end states of the event tree.

Getting into some of the details that give us the conditional tube failure probability itself, one of the key parts of this is the modeling of tube failure. This is part of the analysis which we feel the best about. There's been a lot of work done here, very recent work, but it seems like it's quite reliable.

It's based on high temperature testing of machine flaw tubes under pressure. When I say here that we're using the original thermal hydraulic analysis, as I mentioned earlier, we have done thermal hydraulic analysis and when questions came up, re-analyses were performed. At the time that the tube testing was being done earlier this year, there was a question on what should be used.

I think when Dr. Muscara was here in June, he showed you two temperature ramp rates, one generated by our contractor, another one from a report submitted by EPRI. I think we can still say that the model can handle

a wide range of temperature and pressure variation.

The latest results on the thermal hydraulic analysis show us a somewhat lower temperature, a slightly different ramp rate, and I think right now people are assessing to see if there is any kind of significant change to what the tube failure model would tell you, but I think the first indications are that the testing that was done covered a wide enough range of temperatures and heat-up rates that we should be able to cover even the latest thermal hydraulic analysis. They weren't drastically different, is the bottom line, of what we had before.

The test results that went into this model that we're using for our study only considered axially oriented cracks. There was a question earlier about did we consider other degradation types. The tube failure model as it stands did not, although there is work under way to characterize and model circumferential failure.

What was found was that under the temperature ramp rates at pressure ramp rates that we saw, the creep failure mechanism was dominant. Earlier on, when we first started doing this work before testing was done, a flow stress model was used. The evaluation and test data showed us that that model wasn't really valid under the conditions we were seeing.

Also some unflawed tubes were included in the test matrix. That showed that they could withstand the kind of conditions that we saw from the thermal hydraulic analysis.

I'll show you on the next page, the very detail of what I am saying here, that the model includes the uncertainty of the effects of the crack and the material availability on creep failure predictions.

CHAIRMAN FONTANA: I don't know if I missed something, but that thermal challenge from the thermal hydraulic model, what sequence did you say that was? That wasn't one of the cleared loop sealed?

MR. DONOGHUE: Oh, no, no, no. That's right. This was the station blackout case with one steam generator depressurized. That was the ramp rate that was used.

CHAIRMAN FONTANA: Okay.

MEMBER SHACK: And it only stands in the sense that it doesn't fail first?

CHAIRMAN FONTANA: Yes. Right. Correct.

MR. DONOGHUE: If you hold it up at a long enough temperature, it will eventually creep fail.

I think I tried to say that caveated with understanding that we're considering the event of interest, not necessarily getting to that condition and

holding it for more than several hours.

I tried to put up enough information to explain the tube failure model, but not enough so we're getting in trouble I think.

The basic prediction, this is in SCDAP/RELAP, integrates the time of the event, ratios into the time, to predict the time of failure for the component which is a function of the temperature, there's time of rupture right there. Function of the temperature and the Larson-Miller parameter.

The Larson-Miller parameter in this equation is given right here. It's slightly different from what's in SCDAP/RELAP. This is what we from the latest assessment of pre-failure information for the tube material was shown as -- it was compared to what SCDAP/RELAP predictions would be, which are not to be too different.

This includes, I'll point out right away here, this includes variability. This is at the 95 percent bound for the material variability. You will see that includes a stress, a hoop stress for the tube right there. So that's where you see time to rupture is a function of stress. It's also a function of this multiplication factor, M sub P, which I show right there, crack magnification.

This was derived originally for, and that's why I put it up, another crack magnification factor right here. This is a through-wall crack magnification factor. It's 100 percent through-wall crack, will effectively increase the stress this way.

Now a partially through-wall crack, this is what was used to characterize that effect. The variability from the test results is included by this number right there, 1.06.

CHAIRMAN FONTANA: And the alpha is some fudge factor?

MR. DONOGHUE: I originally had it on here. Yes, it's on the next page I just -- it's called the crack depth parameter, but it was put in there based on the test information that was generated.

MEMBER POWERS: Twenty percent uncertainty at the 95 percent confidence level in the Larson-Miller parameter really impresses me.

MR. DONOGHUE: Again, I could show you the basis for that information. You know, that gave us these, both of these equations as a matter of fact.

MEMBER POWERS: Is that a precision for a particular data set?

MR. DONOGHUE: I remember it was data over several, I think from several sources.

MEMBER POWERS: Several sources? And we only have a three percent uncertainty?

MEMBER SHACK: Yes. It's sort of amazing.

MEMBER POWERS: That's amazing.

CHAIRMAN SEALE: I would say they were impeccable.

MEMBER POWERS: I would say that we could ignore that uncertainty.

MEMBER SHACK: Oh no, no. It makes a big difference.

MR. DONOGHUE: Well, for the ramp rates --

CHAIRMAN SEALE: I mean if you look in creep tests, that may look small. But it's a big effect.

MEMBER POWERS: But when we don't know the event tree to beyond one significant digit at best.

MEMBER SHACK: Well, you know, I'm a national lab nerd. In my laboratory, that's a big effect. Maybe at the end of a probability risk analysis it's not so big, but it doesn't conflict with your intuitive expectation that when you run a creep test, you get a wide range of results.

MR. DONOGHUE: And it is a good point to make that. I mentioned that when I first started talking about the tube failure model, this is the part we feel best about. You will see other components of this analysis

where there's still a wide range of uncertainty that we're considering.

The last thing I wanted to point out here, I think I already have. You know, the basic Larson-Miller model in SCDAP is for non-flaw tubes. These components were added to understand, to include the effects of cracks.

On the next page, I already flashed it up there just to find some of the variables. I don't need to put that up there again.

There's a lot more that could be said about the tube failure model. I think you saw a presentation in June where the actual test results in time were put up there. There has been some testing done since then, including isothermal tests. As I said, a model of circ. cracks has been put together but not included in this risk assessment. When that is more developed and there is more testing done, it's possible to include that in the same kind of study.

Any more questions? Any other questions about the tube failure model?

Now going from what we considered to be a tight part of the --

MEMBER POWERS: In the sense that a lot of this is going to boil down to a race between the failure

of the tubes and the failure of other components, does a similarly sophisticated model apply to them, the other components?

MR. DONOGHUE: The surge line or hot leg?

MEMBER POWERS: Yes.

MR. DONOGHUE: To include the effects of flaws?

MEMBER POWERS: Flaw distributions, the Larson-Miller parameter and things like that, you use SCDAPs, raw result.

MR. DONOGHUE: Okay. What was done for the hot leg in the surge line, was a study to -- I unfortunately didn't bring again. A study was done to give us the uncertainty in the Larson-Miller equation for the hot leg and the surge line. So we are able to show where we think the prediction would vary over the material variability, but we did not include flaws. So the magnification factors are not included for --

MEMBER POWERS: In the Larson-Miller parameter that you used for these others was similarly accurate?

MR. DONOGHUE: I don't recall what the numbers were. I think the hot leg had the widest variability, of what I remember. The surge line and tubes were very tight.

MEMBER POWERS: Yes, it would really be the

hot leg that you would think would be the broadest, simply because tubes are easy to understand.

MEMBER SHACK: Well, it's also the hot leg material is not something one normally studies.

MEMBER POWERS: That's right.

MR. DONOGHUE: Again, there's information I could get to you later that tells you --

MEMBER POWERS: Yes. I would like to see that.

MEMBER SHACK: It's significantly broader than the hot leg and the tubes.

MEMBER POWERS: It would be interesting to see what you have done there.

MR. DONOGHUE: I think I lost a page. Just a second.

Now from a higher confidence to a lower confidence part of our analysis, flaw distributions as I mentioned, were put together on a representative basis. We're not trying to use a flaw distribution that necessarily characterizes Surry or any particular plant.

They were based on operational inspection information and you will see the examples when we get to them in a second there. But we have to understand two things, is that to get a plant-specific crack sized distribution, these distributions are frequencies of

certain sized cracks, the inspection capability would have to be somewhat upgraded from what it is now. The plants below a certain through-wall depth, licensees really can't tell the population of the flaws, which kind of sizes. So some of the information in these flaw distributions is somewhat artificial.

As was mentioned earlier, different plants have different types of degradation it's dealing with. So putting up a flaw distribution that might be weighted towards axial cracks rather than circumferential cracks doesn't necessarily represent plants that are affected more predominantly by IGA or circumferential cracks.

The two distributions, again I labeled them from the offices that generated them, are different, are quite different in that there are six separate distributions characterizing a distribution under the research model.

Under NRR's, several types of degradation were folded into one distribution for axial cracks only. There are two specific flaw lengths that were used, a half inch and an inch. Under the research flaw distribution here, the functions were given as continuous over flaw length and depth.

Now I'll throw the --

CHAIRMAN FONTANA: Was this deliberately done

so you have two parallel activities or did it just happen?

MR. DONOGHUE: Well, the question about flaw distribution is not easy to answer. The first example that we tried, that we started using left us with a lot of questions that I will get to in just a second. But a parallel effort was undertaken in the Office of Research.

What we tried to do was not supplant one flaw distribution with another, but trying to consider both of them because both have strengths. We are going to have to say at the end of this analysis, that the best thing to do is have a plant-specific flaw distribution if you can generate it.

That's a tough question to answer though. Like I say, the inspection capability doesn't necessarily get you there.

CHAIRMAN FONTANA: But you tried the NRR one first, the one on the righthand side?

MR. DONOGHUE: Right.

CHAIRMAN FONTANA: I see.

MR. DONOGHUE: Right. That was the first one we had. We knew the work was underway with the research, as they call it, distribution.

We have shifted to use the RES study, does use the results from -- I'll show you exactly which part of the results from this research distribution, for reasons

that some people might dispute, but that's the one that we're using at the moment, although we haven't done away with the NRR distribution. We still want to understand why we have the differences between them.

MR. LONG: This is Steve Long with the Risk Assessment staff.

Part of this was just trying to deal with the information that we could get at the time. This is essentially the fourth shot at the distributions. There were three that came from NRR. It helped to have two different thought processes going on, because this is a very uncertain area and it gives us a little bit of a chance to see the sensitivity from difference in the perspective of how you would generate a flaw distribution. I think he has some graphs that will give you an indication that that was really worth doing.

MR. DONOGHUE: Since I mentioned we were concentrating lately on the research flaw distribution, I'll just show you what the degradation types were.

One from circumferential cracks, TTS top of tube sheet, that's stress corrosion cracking on top of tube sheet. ODSCC at tube support plate dents. Freespan ODSCC. IGA and stress corrosion cracking of the hot leg sludge pile. Axial ODSCC at tube support plates, and then flaws from loose parts.

For a second there, I thought I indicated the wrong ones that we used in the study, but I think it's right. Again, we were able to characterize axial cracks so we don't include circumferential cracks.

Earlier on it was pointed out that the freespan cracks are the largest part of the population of flaws. Then we knew that the temperatures in the tube bundle for these situations was going to be highest, right in the hot leg sludge pile area. So those two parts of this research distribution are the ones that were actually used for our RES study.

Now I'll put up a couple of graphs here. It's maybe unfair to put this one up before I put up the actual example of the research distribution, but so be it.

This is a comparison of the two distributions. What was done was the two of the six were force fit, just represent flaws greater than one inch. That is what you see here. You see number of cracks versus the through-wall depth.

One thing that's obvious is that there's some large disparity here for shallow cracks. The basic discussion goes that with this distribution, the assumption is made that you really can't protect those cracks below about 40 percent through the wall, and that that number is conservative obviously.

Down here, this above 20 percent, the probability of detection was somewhat more favorable for this characterization, and we're able to see some smaller than 40 percent through-wall cracks. But this is probably not realistic either. It seems like it's probably a lot lower than it should be.

But what I want to point out for our purposes here of understanding thermally induced failure, what we are really worried about are flaws between about 60 and 80 percent, in that range.

We don't see a big contribution from the small flaws. So we are not too worried about that difference for our purposes.

You can see that there is a bigger population of these flaws of concern for the research distribution, which is one reason that I pushed us to use it. I will show you some specific numbers for the tube failure probabilities based on these later, and show you why they make sense.

It makes a little more sense right now to use that distribution, but the difference is in the shape here. The gamma distribution versus this kind of exponential function. We have to understand the bases for these distributions are somewhat different. We have to look into more why those differences come about to really

decide on which is the best way to characterize the representative flaw distribution.

But what was comforting was that there wasn't a large disparity in the area that we were concerned about.

CHAIRMAN FONTANA: Roughly how many samples are we talking about here?

MR. DONOGHUE: I would have to go back and look at the report, but the author is back. He might be able to tell you.

CHAIRMAN FONTANA: On the order of thousands or hundreds or tens?

MR. DONOGHUE: Oh, okay. I think the research distribution was based on a large thousands of tube samples or inspection data points. Is that right?

MR. GORMAN: This is Jeff Gorman, Dominion Engineering. The freespan indications were based on large scale rotating probe inspections of Palo Verde 2, and the numbers of tubes inspected I think were in the order of 4,000 tubes per steam generator. The number of flaws detected, if my memory is correct, was in the order of 600 to 1,000.

The number of tubes that were pulled and destructably examined were smaller to verify the NDE results.

But the data that were presented are based on the rotating probe inspection results, so in the order of 600 to 1,000.

CHAIRMAN FONTANA: Thank you.

MR. DONOGHUE: I just wanted to show you this is straight out of the Dominion Engineering report. This is just a freespan defect distribution. You can see again, it's a gamma distribution. I don't want to spend a lot of time on it. It's just to show you the raw data that got folded into that last slide I showed you. This whole report is going to be a reference in the NUREG that we are putting together.

Now what had to be done was to take the continuous distribution of length and understand where the break points were for certain lengths, because that's where the tube failure probability calculation is done, in different bins. I'll explain that in a minute.

But you can see above an inch, there's a relatively small fraction of flaws in these distributions that we use. We used bins that were from one inch and greater, between a quarter inch and one inch. Those are the flaws of concern for these situations. This just gives you an idea of how they would bend and what the contributions were.

Now using the flaw distributions and the

thermal hydraulic analysis results that gave us an understanding of the creep rupture potential, a program was put together to calculate tube failure probability.

You already know how the event tree was put together, and that we had to include information for pressure and thermally induced failure. But what was done with this model was to actually take the information -- I'm trying to find a slide to show you what we start with here. SCDAP/RELAP calculates that creep damage index.

This is just an example. This is just one of the cases. This is not in your handout. Just a backup to help illustrate. Over the event time, Larson-Miller is -- SCDAP/RELAP is calculating the Larson-Miller creep index, the integral that I put up earlier.

Eventually, the conditions are right that the component is going to experience creep failure when you get to one for that damage index.

What was done was the program calculates the probability of failure over time. I'll show you another slide in just a few minutes, but this is what we start with. We start with the creep damage index. We have to understand what's going to change it, whether the variability of materials is going to change one way or the other to change this prediction, the thermal hydraulic analysis results have some kind of variability that could

change these, and so forth.

So that's in essence what the tube failure probability calculation is doing, is understanding how the relative -- how these creep predictions for the hot leg, the tube, or the surge line to the tube predictions would change over time.

MEMBER POWERS: Could you explain a little more about the legend on that figure?

MR. DONOGHUE: Okay. This particular --

MEMBER POWERS: I mean in SM equals two.

MR. DONOGHUE: Right. This particular figure was showing us the effect of a crack that would have a stress multiplier to keep changing our variables. I'm sorry. But stress multiplier, M sub P that we had of two. That represents I think a 60 percent through-wall half-inch crack, something on that order.

Then we have the creep index, the creep damage index that was produced by SCDAP/RELAP or R/S, somebody used RELAP/SCDAP as an abbreviation there.

Sc you see what was actually generated by SCDAP/RELAP in the dotted or dash lines, and then what program that we used that calculated the effect of the crack on that tubes is a solid line.

MEMBER POWERS: And what you conclude from all of this is that almost no matter what you do, these things

are hitting their creep damage index over a period of perhaps 20 minutes, 10 minutes?

MR. DONOGHUE: This is I think in this case it was about 20 minutes. If I put up the -- this is a space of 20 minutes, roughly. That was pretty consistent with the other cases that we saw. Exactly.

If one of these was skewed off, and for example, when vessel failure was considered, that was at such a different disparate time from these other components, that this kind of analysis really was not warranted.

But since these things are pulling around on top of each other, this kind of calculation for tube failure probability is really needed.

MEMBER POWERS: It's just very difficult to believe that you can calculate something in a severe accident to plus or minus 20 percent -- I mean 20 minutes.

MR. DONOGHUE: Well, I am just showing you here is the effect of one variable.

MEMBER POWERS: It's a very illuminating graph, yes.

MR. DONOGHUE: As I say, there are several components. There's nothing about flaw distribution in here yet. This is just an assumed flaw on one tube.

MEMBER SHACK: But what Dana is pointing out

is you go from the unflawed tube to a fairly severely cracked tube, it's 20 minutes.

MR. DONOGHUE: Right. I am not including any variability of thermal hydraulic analysis results. The uncertainty analysis that have just been done is not part of this.

MR. LONG: This is Steve Long again. If you can think of the lines there that were originally calculated by RELAP/SCDAP as fuzzy areas, as some sort of distribution in probability, and then what we do is try to overlap those distributions and ask what's the probability, at least as far as we know, in the variation of these various lines, that essentially the tubes would fail before any other part of the pressure boundary. That's the technique that we used.

Then we have to do that for tubes that have different sizes of cracks. Then we have to ask how uncertain are we about the effect of those cracks on the strengths of the tube. We are starting to get into that kind of result now.

MEMBER POWERS: I understand. I think this is an extremely illuminating graph on what the range of uncertainty is here. I mean you find all the failures occurring in roughly the same time as small variations in parameters move their relative positions around a lot it

seems.

MR. LONG: Well, for instance, the curve that goes up to one earliest, the one on the left for the surge line, if you include the materials variability in there, there's just a few minutes. I think it's three or four minutes maximum, difference between the five percent probability that you will have failed and a 95 percent probability you would have failed.

So that the tubes and the surge line look relatively distinct, even when you put the probability distributions on them, as far as materials uncertainty is concerned, materials variability.

Now if you start adding things that you brought up earlier about well, the hot leg may have a bend in it that its stress isn't calculated, that's true. There's some systematic problems with this.

But just in terms of the materials variability, they do look fairly distinct until you start putting cracks in them. When you move the cracks, move the lines with the cracks, you start getting different amounts of overlap. Eventually the crack will, at least with materials uncertainty, clearly fail before the surge line if the crack is large enough.

MR. DONOGHUE: I just want to throw this up to illustrate something Steve just said. This was one of the

earlier plots that was done once we had some understanding of the variability in the Larson-Miller parameter.

For the tubes, you can see when we get up near one, it varies over several minutes where the surge line is fairly tight over just a couple of minutes.

Again, there's several ways of doing the variability study. This was done earlier on when we were just considering the material and the crack variability. But when we tried to include all the other things from the model as I'll explain in a second, tube failure probability gets quite complicated.

I think I am on page 19, after a lot of flopping around.

I think we have talked a lot about the top part of this page. What I wanted to point out here is that again, flaw distributions were used from what we are calling the research distributions, and the calculation is done at bins and length and depth of bin. I will explain that in a minute. But understand that what we tried to do here was leave the ability for other flaw distributions to be considered at some point. I think the model is put together so that we're not locked in to any particular flaw distribution, but as information is updated, it can be included.

At the end of the presentation, I will be

talking a little bit about sensitivity studies that were done using this probability calculating tool over a range of tube temperatures to understand the effect of variability uncertainty in the creep magnification factor. We did it at one different RCS pressure to see what the difference would be from a PORV to a safety valve set point on the RCS. Then we did is using, as I said, both flaw distributions.

Okay. We have talked a lot about these points up here already. The point to me here is that the two length bins were picked for particular reasons. The crack length for -- the critical crack length for normal temperature and normal differential -- I'm sorry, differential pressure at the PORV setpoint was one break point.

The next was for critical length for the temperatures associated with core damage. That's where the flaw distribution was broken into greater than one inch for this critical length, and between a quarter inch and one inch for those lengths. So those are the length bins that we are talking about. Then the through-wall depth bins were over five percent increments.

So the creep model that we talked about earlier is used at each flaw sized bin and it's done for each thermal hydraulic case. Take the results from a

thermal hydraulic case, and you take those creep damage indices. You do it at each of the length and depth bins.

CHAIRMAN SEALE: Joe, you are going to be showing some detailed results here for the next few slides. Is this a good point to take a little hiatus for about 15 minutes?

MR. DONOGHUE: I think so.

CHAIRMAN SEALE: Okay. We'll come back in 15 minutes.

(Whereupon, the foregoing matter went off the record at 9:58 a.m. and went back on the record at 10:15 a.m.)

CHAIRMAN SEALE: All righty. Proceed.

MR. DONOGHUE: Just before the break I had finished mentioning how we had binned the flow distributions for this calculation. And what's being done -- and somewhat restating, I think maybe, what I've already said -- was that the probability of another component -- and the other components being a surge line or hot leg -- probability that they would not fail prior to the tubes at some time when the creep damage would equal one -- integral I was discussing before -- that's what's being calculated.

Now, a little more detail on what's in this calculator tool. Monte Carlo method is used to account

for the probability over a range of Larson-Miller parameters as I mentioned, and not only for the tubes, also for the -- well, for the surge line and hot leg, I think that's included as well. The tube wall thickness and the tube diameters, the variation there is included, and then the bins as I mentioned.

Now, on the event tree -- which I won't flip to, I'll just mention that -- pressure-induced top event -- there's a limit load analysis that's performed independently, using the conditions for main steam line break, the high differential pressure condition, but without a temperature challenge. And that -- I think I say down here later on, basically by doing that earlier in the event tree we remove those flaws from any consideration for temperature challenge.

Calculation -- figured out the probability of failing a tube ahead of the other components, and again for each bin -- I keep saying that -- but then, we have to make sure that we don't use the flaw distributions for cracks greater than 100 percent through wall.

As I was showing you earlier, that plot of flaw distributions, the actual functions go beyond the hundred percent point. And one way you could consider that is to take everything that's out beyond a hundred percent and stack it up at 100 percent.

And we don't think that would be realistic. It would just tell you that you have a much higher probability of failure then you really ought to. So basically, I think we're truncating at the hundred percent point, those functions for use in these calculations.

As I said -- oh, this is a different point -- the flaws expected to burst in normal service are removed from the distribution. So I think greater than 89 percent through wall, I think that's the through wall for a one-inch crack. I could be wrong on those numbers. But flaws that are going to fail spontaneously are removed. That kind of makes sense.

And as I said, anything that's going to fail under the pressure challenge is removed earlier on, so that we're only considering the failure probability of temperature-induced rupture.

Now, some of the results -- let me start by saying, before I get any barbs about the number of figures in these numbers -- it does not indicate the accuracy of the calculation. They were simply carried to prevent us from having larger roundoff errors when we start combining all these results. So I'll just -- I should put it up there in big red letters, I guess.

But you can see, first thing we did -- I'll just lay out how this -- again, have the descriptors for

the thermal hydraulic analysis that we used for the event tree. Then, first that pressure-induced calculation was performed and as you can see --and you have two pages that look almost the same here -- this one uses the research flaw distribution that I have been discussing, and then that sensitivity study that I mentioned where we do two different flaw distributions, came in handy.

I won't keep this up here too long; I just wanted to show you the comparison of the -- you see over there for pressure-induced under the research distribution, somewhat higher values than you have here for the NRR distribution. And this corresponds -- well, this is pretty close to what we're allowing under the Rule for pressure-induced rupture. And it seems realistic to keep that -- that's one reason why we wanted to keep the research distribution into consideration.

And then when you see, for temperature-induced rupture you see quite a difference, but I just wanted to point out that the pressure-induced difference kind of drove us to consider -- one reason to consider using a research distribution.

Now, a couple of things to point out here. The bracketed "1.0" is not a calculated value. We've mentioned earlier that if we had a clear loop seal with a depressurized generator, that's going to lead to higher

temperatures in the tubes. I think we've had -- you've heard that discussion before.

If you have a one-directional flow rather than the counter-current flow, it's going to lead to a greater tube challenge. And in this case if you have that happen, the temperatures get high enough where you're almost certain to have a tube failure if the steam generator is depressurized.

You can see the difference is much -- it's in the weeds here, two percent -- if you have an intact steam generator with a clear loop seal. So that's where I was saying -- using the term earlier, Russian roulette -- if you have only one steam generator depressurized, you have that RCP seal failure, even a clear loop seal, but the question is, where. It's going to be the depressurized generator or not?

The other point to make is that, if all these generators are depressurized for a station blackout case, you have a relatively high, but not drastically unacceptable, tube failure probability. And then the other contributor is up here. I think is a -- well, correct me if I'm wrong; this might be a -- I think this .17 is a combination of the numbers below it -- no, it's not.

This is -- oh, thank you, okay. This is just

a seal LOCA with a single steam generator depressurized. And that's relatively high and discussion on that is -- get's back to the thermal hydraulic results where you have a -- not considering a loop seal clearing, you have accumulator injection causing a pressure spike. It gives you a high tube temperature, very momentary high tube temperature that could lead to tube failures.

MR. LONG: Joe, this is Steve Long with the staff. Let me just clarify what's going on under that case 9 up there. It's essentially a combination of three of those four probabilities listed there. We essentially, in the thermal hydraulic case, had one steam generator that was depressurized, that didn't have a clear loop seal -- that's the 17 percent.

We had another one that had a clear loop seal but the steam generator was pressurized, and that gave us something down in the noise, and when we averaged -- excuse me, when we combined that -- and plus a third steam generator with a zero failure probability because it was pressurized and remained relatively cool because the loop seal didn't clear.

So those are three steam generators we effectively added up for none of them having a failure, and then took the complement to get a total failure probability, or one tube somewhere in all -- any of the

three stream generators. So the number you'll see for that case combined is about 19 percent, I think.

The thing in the brackets we didn't really have in the thermal hydraulic case, and that was the combination of the clear loop seal and the depressurized generator, but the temperatures are extremely high. I think I saw 1308 K as a peak temperature in the pressurized generator with a clear loop seal on -- we just assume that's going to fail if it was depressurized instead.

And when we've tried to put this in the event tree we had to take all the possible combinations of where that depressurized generator might be. So those are the four constituents that you add to different degrees in different parts of the event tree for a seal LOCA case.

MR. DONOGHUE: Thanks. Just to show you again the NRR average distribution -- the values for temperature-induced failure are much lower than we saw with the research distribution, and that reflects again, on the difference in the area of concern here. You can see again, the research distribution is somewhat higher, flawed tube population than the NRR.

I think I've laid out -- and I'll go back to, just to kind of recap how -- what we've gone through so far; in fact, one of my first slides. We discussed the

event tree, at least laid it out for you and showed you what went into quantifying it and how it was structured to some extent.

The quick discussion about thermal hydraulic analysis results, the basis for the tube failure model, the flaw distributions that we considered, some of their limitations, and a quick discussion on the tube failure calculation -- that put these three pieces together to get a number for tube failure that's going to go into the event tree.

Now, you already saw a result from the event tree for containment bypass; that low- to mid-10^{-6} number, for example. And the next thing I wanted to do is just discuss some sensitivity studies we've done on these different components of the calculation, and then discuss what we consider to be some of the key issues that remain and limitations for our work.

I'll start with several sensitivity studies that were done using the flaw burst calculation probability. We did a sensitivity over a range of tube temperatures, and this was done earlier on when there were a lot of discussions going on about the uncertainties involved in the thermal hydraulic analysis.

And right now there's some work that's in-hand that looks like it's going to resolve, or at least

address, a lot of those issues. But what we did -- and this again, sorry is not in your handout -- what was done was, in over a range of temperature this offset -- this is Kelvin temperature -- is an offset from what the SCDAP/RELAP would get the tube temperature for hot tubes would be.

And we offset it 60 degrees below that, up to 90 degrees above that value for these different cases. For the base case, station blackout, loss of auxiliary feedwater -- with all the steam generators intact. And then the next case with the steam generator depressurized, and then we used another flaw distribution on that particular case. And then the last one is the RCP seal LOCA scenario with the generator depressurized.

And what you see is, for the no offset, you see the basic results that -- the big contributor here is the RCP seal LOCA, up above 10 percent tube failure probability. And then the base case that we've used for this study up till now -- the case 3-R we were discussing there -- is between five and ten percent tube failure.

And you can see as the temperature increases the failure probability is higher, but at some point we reach -- at 70 degrees offset, when we add 70 degrees to the temperature calculated -- you see that the smaller flaws become bigger contributors.

And then this green curve, using the NRR flaw distribution, takes off. And that's where I was showing you earlier, the flaw distribution, NRR flaw distribution is a flat line below 40 percent at a high value -- that's where we're not sure it's realistic. It counts -- when you get to higher temperatures small flaws are going to be more of a concern, and this becomes such a large contributor we're not sure that's realistic.

But by the same token, you saw how low the research distribution was for small flaws. So these values may be somewhat different in reality. We think that, at least for purposes of doing a sensitivity study on our risk estimate, that we've picked originally a 70 Kelvin degree temperature offset just for giving us some idea on how sensitive these results would be.

I understand now -- and I'll talk a few pages -- about the recent thermal hydraulic sensitivity analysis that show that the temperature with some certainly is going to be offset by something much less than 70 degrees. Somewhere in the 40, I think the 50-degree, the Kelvin degree range is where you can say the maximum offset ought to be. So that makes us feel a lot more comfortable of something at 70 or above.

So I think I've given you a picture of the temperature sensitivity for tube failure probability. And

then we did creep model -- the multiplier, the M_p variability -- and again, another one that's not in your handout -- just to show you that, using the research flaw distribution you can see that -- or what I'm saying here is that it's about equivalent to a ten degree, or added ten degree error.

If I take, for the 95 percent error --or variability in M_p, that's up around -- well, it's says .06 -- that gives you something less than .1 probability of more failure. And if I go to the graph I just had up, this is where the ten degrees come from.

For the base case, with a 3-R, it's the red line. If I look at something just below ten percent, about eight to ten percent, that tells me it's something around ten degrees. That's where that statement comes from. So it looks like the flaw stress multiplier is not a large contributor.

And the other sensitivity that was done was for pressure sensitivity; just adding a hundred pounds to the output from SCDAP. It was equivalent to about a 15 degree temperature difference and failure probability.

Now, there were further sensitivities done on the event tree, or accident progression event tree, or APET a lot of people call it. And these are -- this work is quite recent; very recent, I'll say, and we still are

evaluating. But we knew --these were the kind of sensitivity cases we wanted done.

And basically they fall into secondary, integrity, primary. system integrity -- including RCP seal LOCAs -- and then a temperature offset is discussed. And we can -- as I said, this was quite conservative, this 70K offset; at least it appears it was quite conservative to use that. And that's something we'll have to update with more recent information as far as thermal hydraulic uncertainties go.

But one could argue whether or not you should do a sensitivity considering perfect integrity or absolutely no integrity under these kind of assumptions, but this is not supposed to represent necessarily, the effect of the Rule, that the Rule provision would have. It's just to give us some insight into the event tree sensitivity.

Questions on that page? There's a lot of text on there.

Now, this is probably quite exceeding the limit for putting up slides that aren't in your handout, but I do think you have this page separately in case you wanted to take notes on it. This is some, as I was saying, very recent work so it didn't even make it into your handout. But, results from some of those sensitivity

studies that I just listed, I'll discuss here.

Now, what we saw from the base case assumption is that you see about a 20 percent chance of containment bypass given the core damage -- the Hi/Dry frequency. So you're seeing about a -- a reduction by five. Now, the majority of the events that lead to bypass -- it's not the majority of the Hi/Dry frequency but it's -- the majority of that bypass probability was in RC-1, that large release category.

I think I mentioned that earlier. It's the direct release to the environment via some failed secondary component when you have a tube rupture. One thing that's important to point out is that the event tree here, the analysis that we've done, doesn't make a distinction between a single tube rupture or multiple tube ruptures.

There's been some discussion in the staff about what the potential is for a failed or failing tube to lead the tube ruptures, and what we're saying is that if you have a tube that has ruptured in this sequence, you have containment bypass whether it's a single or a multiple tube rupture. It's all binned together.

MEMBER POWERS: When you assign things to RC-1 category, have you looked in detail at the transport from the site of the rupture, to the release to the

environment?

MR. DONOGHUE: I know there has been analysis done using the Melcor source term, and I believe it accounts for hold-up fractions in the steam generator and so forth. So I think it's been done in some detail.

MEMBER POWERS: What I'm interested in is that, if you look on the secondary side, depending on where your rupture is, the fission products would have to flow through a forest of tubes and bins and a variety of things. It's not 100 percent obvious that they could make it through all that tortuous path; it's not obvious that they wouldn't. And certainly in the older codes there was just no attempt to model that, and I wondered if that was still the case.

MR. DONOGHUE: Well -- I'm sure I'll be corrected if I'm wrong here -- but I believe that there were assumptions made for hold-up and the position in the steam generator. I just don't recall what they were. That information we can get to you. But for purposes of categorizing things on this event tree, it wasn't necessarily on the value of the release; it was just basically, do you have the release -- certain release path.

MEMBER POWERS: I understand. But I mean, you understand that these bypass events are not typically a

frequency-dominant issue; they show up as a risk-dominant issue because of the magnitude of the source term associated with it.

MR. DONOGHUE: That's why -- yes, right. I mean, this number that I'm calling a result for this example analysis, this 10^{-6} number, is, yes. It's not necessarily a stratosphere itself, but we understood that. That is of concern because --

MEMBER POWERS: But if the source term is wrong, they're overly conservative; then maybe we're going through a huge amount of effort that's not merited.

MR. DONOGHUE: Right, but you know, my presentation stops short of discussing that.

MEMBER POWERS: I understand that. I was going to push you to say, let's go on and look at the risk now.

MR. DONOGHUE: Well, I think you heard me say yesterday, we're putting in words in the Reg Guide that are -- we're trying to make sure risk is consistent with the subsidiary safety objectives, and the measure that we're using -- I guess, maybe to avoid arguments about anything more than containment bypass frequency -- is the containment bypass frequency.

MEMBER POWERS: But what happens if a licensee comes in and says -- conservative values for the frequency

of a bypass accident, but then I looked in depth at the consequences of that accident -- and this is not a hypothetical example -- I happen to know that there are at least contractors for some of the utilities that are in fact, looking at this issues, and they came in and made it a risk argument. Would you be in a position to evaluate a risk argument? This is --

MR. LONG: This is Steve --

MR. DONOGHUE: I think I could say we'd be in a position to evaluate it; however, what we've said all along here is that, this comparison to the subsidiary safety objectives has to be done with an understanding of the defense-in-depth effect -- the effect on the defense-in-depth that you're having.

I mean, we're talking about containment boundary essentially; talking about the tubes in this situation. So that's why we focused on the potential -- the containment bypass frequency -- the potential to fail that boundary and less on the release consequences as far as understanding what we had to have the Rule do for us.

MEMBER KRESS: But if a licensee came in with that kind of analysis, you would entertain it as a -- if you had a rule based on, strictly the frequency of the failure, and they came in with a subsequent analysis like Dana talked about -- even though they exceeded some

criteria in the rule on this frequency, you might say it's okay, you guys are all right, you don't have to reduce that frequency?

MR. LONG: This is Steve Long. Let me add a couple of concerns here. We already have an analysis like that from NEI that tried to draw a distinction between the number of tubes that ruptured and whether or not if one ruptured and then the reactor pressure boundary gave way somewhere else, it would cut down the driving force and drastically reduce the release.

The problem is, once we get to the point where we think we have a significant failure of the steam generator tube at very high temperatures, we're really not sure what that's going to do to adjacent tubes. And trying to predict the progression of the accident sequence after that is pretty tough. If you have impingement of very hot gases on adjacent tubes, we're not sure what that leads to.

So just what's getting into the steam generator is already uncertain, and how much gets into the steam generator, of course, will have something to do with the temperatures of the steam generator's surfaces, secondary side reach. I think some -- I'm not sure if Victoria calculations have been done on the secondary side -- it's in the back of the room either.

There are codes that could address that, but you're getting into some very researchy areas to try to draw good solid conclusions; things that we didn't feel we could do for the study, and things that we didn't feel we could really credit from industry right now. What we'd really be doing here would be saying, we don't really need containment for this kind of an accident in the form of physical barrier. It could be more or less in the form of a filter.

And we've had those arguments years ago, also, but we were talking about much more effective filters.

MEMBER KRESS: Would you comment on the defense-in-depth arguments you're making?

MR. LONG: Well, people have discussed things like sand bed filters for venting containments in the past. And I think probably you can generate a good enough filter that that might be a viable option, but we're not so sure what kind of a filter we have here in a steam generator secondary. Until we could really have some confidence there I don't think we want to do away with the requirement that the boundary stayed physically intact.

MEMBER KRESS: So I interpret your answer to my question being no, you'd probably --

MR. LONG: No.

MEMBER KRESS: -- give much credit for

attenuation calculation of fission products because of the large uncertainties associated with that kind of calculation?

MR. LONG: I guess what I'm saying is, I know that the industry -- and I think we have been doing some work to try to investigate that and see what kind of effects we see -- but it's very difficult to reach a conclusion where we think we could, in a short period of time, allow the industry to go ahead with the Rule that we're talking about here.

It didn't seem to be a very quick way of solving the problem, let me put it that way; however, if we got into a cost benefit discussion on the benefit of avoiding some accident where we did --

MEMBER KRESS: You almost have to do that in a cost benefit, I think.

MR. LONG: We have to, but how conservative we have to be is another matter.

MR. DONOGHUE: I'll just point out that I guess, one way of answering your question is by pointing to the outline of the section in the Reg Guide discussed briefly yesterday. I mean, that lays out what we think is an acceptable way to approach a risk assessment here.

And the two options again were: do a PRA to understand what your initiation frequency here would be,

and if that's low enough then you're okay because even if you challenge the tubes you're not going to -- if you're below that 10^{-6} number to begin with, you're not going to exceed the safety objectives.

The other option is to plow through a calculation like this, including the tube failure kind of modeling, understanding your flow of distribution, etc. So that you know, they're given that option but nowhere do we really say, show us how small your release is; get below some certain number. We aren't able to that.

MEMBER POWERS: But I mean, it seems to me, from a personal point of view it seems a lot easier to attack the source term than try to understand flaw distributions in pipe that I can't see.

MR. DONOGHUE: It might be easier to attack flaw distributions but it's not easy for us to say how large a release is acceptable, necessarily, unless the PRA implementation guidelines end up spelling it out in detail. But that's not a foregone conclusion from what I understand.

Something to say, Charlie?

DR. TINKLER: Yes. Charlie Tinkler from the Office of Research. I just wanted to clarify -- we did determine the off-site consequence using Melcor which gave some credit for deposition on the secondary side, but not

a great deal. It did not take credit for the, all the additional deposition that might be attributable to the presence of the tube bundles, separators, dryer, etc., etc. We only took credit for it as a volume with a residence time, and things like that.

There was some credit given, but greater attenuation of the source term was not credited in the Melcor calculations.

MEMBER KRESS: Do you use the standard dies, 1465?

DR. TINKLER: No, we actually --

MEMBER KRESS: Or did you let Melcor --

DR. TINKLER: We actually let Melcor calculate the source term, and as you heard in other presentations, we did calculations with Victoria and Victoria did produce a lower source term than Melcor -- for a couple of reasons. But for these off-site consequences --

MEMBER KRESS: -- mostly.

DR. TINKLER: No 1/p dependency and things like that. But we did use the Melcor calculated source term and a limited attenuation of the off-site release due to deposition on the secondary side.

MEMBER POWERS: At least when we have studied dryers and separators in the upper internals of BWRs we have found they can be a fairly significant attenuator.

DR. TINKER: I wouldn't dispute that.

MEMBER KRESS: But I don't think I would describe a steam generator as (indiscernible).

MEMBER POWERS: Well, I can't disagree with you that they're different, but you have a significant attenuation here and -- I mean, the potential of some attenuation -- and I'm wondering why wouldn't an applicant licensee in the face of an awful lot of work say gee, this is an easier way to get around the skin-the-cat; just to go after the consequences rather than trying to understand flaw distribution.

MEMBER KRESS: I think he would if he thought that staff would accept that as a reasonable argument, to not fix (indiscernible), talk about (indiscernible) fix.

MEMBER POWERS: It seems like we ought to be in a position to evaluate the legitimacy of the arguments that he puts forward.

MEMBER KRESS: Well, I think there is some credence though, to this concept of there may be more uncertainty there than in these calculations, and I think that's -- I think the uncertainty drives whether or not you treat this as a defense-in-depth concept or not.

MEMBER POWERS: There clearly is a very big defense-in-depth argument here, because in contrast to many other parts of the RCS, we do not have another

physical barrier beyond this one, and so there is a very big defense-in-depth question. The question then boils down to the one that Mr. Long articulated so well. Is that filter a barrier or not?

MEMBER KRESS: Yes. Well, I think it is; it's just how much confidence you can have in its ability to --

MEMBER POWERS: Well, you won't have any consequences at all if you don't look at it.

CHAIRMAN SEALE: It's a bit of a barrier.

MEMBER KRESS: That's true.

MR. DONOGHUE: Okay. I think I left off at this point. Yes, the balance of the events where we talk about RC-2 on the event tree, were again, due to partially or low -- partially depressurized or a low pressure RCS with an intact secondary; however, we're not sure about the MSIV integrity. We make an assumption there that we're going to depressurize the secondary and possibly have a release path.

I mentioned the RCP seal LOCA contribution here, and we think it's a large contributor depending on - - it does depend on again, which plant you're talking about -- RCP seal leak model varies from designs, but for our example, it's 70 percent again, of the containment bypass frequency. It's not -- I'm sorry, it's 50 percent of the total of the bypass frequency. Am I saying that

right?

So we see it's a large contributor and that's why we spent a lot of time looking at this, and I'll just divert for a moment from the slide. When we first saw the high tube failure probabilities under this case, a lot more work went into looking at the RCP seal LOCA branch on the event tree.

Because originally it was put in the event tree thinking it was going to give us a benefit, that the depressurization from the seal leak would actually help us and avoid tube challenges. But for the reasons I mentioned earlier, it does lead to a challenge to the tubes, somewhat higher than we thought it would be.

So we're going to need to look a little more closely at some things that are used to pick RCP seal leak; its frequency and the magnitude that you're going to see; how it progresses through the event -- how the leak actually may change during the event.

The sensitivity studies I showed you just a minute ago that list the sensitivity studies were done -- and you can see a variation for the conditional containment bypass frequency changing from a very small number, two percent, up to -- you can see 30 percent for that 70K increase.

Now again, I think this is conservative, so I

think we're seeing something from below ten percent up to the order of maybe, if we use some maybe more realistic offset for the temperature, maybe in the 20 or so percent range for containment bypass.

I say right up there, digital sensitivity studies are planned on the event tree -- I'm not going to go into detail on what those might be. We have a list that we're considering to augment what -- a list I showed you a minute ago.

Now, another piece of recent work were sensitivity studies done using the thermal hydraulic modeling, and these results have not been factored into the risk conclusions I've been talking about today. We just haven't had the chance. Some of this stuff is still being documented, as a matter of fact.

But I at least wanted to point out that some work has been done to address the questions that have come up before this committee, and we're going to first look here at some of these results as telling us -- at least for looking at th revised base case that we have used before.

It tells us that what we used before might be somewhat conservative; however, I'll point out here that the original base case we had, had a tube split -- this 35/65 is the hot upflow in the tube bundle versus the cold

return flow to the inlet plenum.

Well, if we decide that it's more representative to use what was seen in the transient tests -- the mock-up, the Westinghouse mock-up, included steady state and transient tests -- the transient tests showed a more even split between hot and cold tubes -- if we think that's more representative and we switch to a base case that uses that modeling, not only do we have lower tube temperatures but we have more tubes being affected by whatever that hot tube temperature is.

So the first look at this tells us that it's probably a wash as far as the tube failure probability. You might have lower temperatures but you have more tubes being affected. But we need to do a detailed look at that, at these conditions, going through the calculation that I've been talking about, and really see what the result is.

MEMBER POWERS: Is the sensitivity of tube failure or tube temperature, linear? Because the number of tubes is linear -- is a linear effect here on the probability of failure.

MR. DONOGHUE: Well, just keeping the tube -- I believe in these cases the number of tubes being affected is the same, in all these cases.

MEMBER POWERS: So what you can argue is that

it's essentially linear in the load differential temperature region, and it may become non-linear at very high temperatures, or something like that.

MR. DONOGHUE: Well, unfortunately, the flow distribution plays into this --

MEMBER POWERS: Yes, I know.

MR. DONOGHUE: -- and mucks it up. But you can see it's a relatively -- it's a straight enough line over the temperature ranges we're talking about. The temperature offset range we're talking about. It will correct.

Just to give you some background on what was just completed of these sensitivity studies. The SCDAP/RELAP5 model was updated with a mixed convection heat transfer correlation and that was used to rerun this case 3. That came up with somewhat -- I'll put up a slide in just a second that will show you some of those results.

And then also what was called case 6, which was not used in our risk study originally because we thought that this tube bundle split was more representative of a base case, but we are now considering that this may be more representative, so we might be switching again as I mentioned a minute ago.

So both of these cases were used in a sensitivity study, but the emphasis was put on what might

be our next base case, and these are the different sensitivity studies that were run, including changes in the heat transfer correlations, heat exchange in the hot leg between the twc flows, and then applying the five percent limits for those three mixing model parameters: the mixing fraction, the flow ratios, and the percentage of tube bundle.

I think you've heard the discussion about mixing models to some extent, and this -- I'll show you the next slide -- these are just -- I tried to put up both civilized and barbarian units here. And you can see for the base case, the hot tube if you recall, about 987 I believe, was the number for the original case.

You don't see too much of a change, but you can see if we switch to another base case it does drop the tube temperature somewhat and as I mentioned earlier, it's a wash probability-wise, because of a larger number of tubes affected.

Then you can see, you don't see a large difference until you apply all those mixing model parameters at their limits, which may not be -- is probably not -- realistic, because although we're trying to -- the purposes of doing this study were to understand some synergistic effects, that might be -- or the effect of synergistic effects -- it's not clear at all that you

would ever have a situation where you had all three of those mixing model parameters at the same extreme. There's compensating effects there that are going to prevent that from happening.

But you can see that the temperature offset is going to be something less than 70 degrees. We were much too conservative when we picked that number, but we're rather say we were too conservative than not conservative when we did that.

And we'll be looking at this information to decide what the real temperature sensitivity ought to be that we use in the sensitivity study we do on the risk study.

Now, that was the synopsis I intended to give you on the sensitivity studies that we have put together. The next couple of pages, I just wanted to discuss where we think the soft spots are, and what key issues we've dealt with or are still dealing with.

I'll start with probably, the hardest part of this whole analysis to deal with, the flaw distribution. I've already mentioned several times the difficulty in specifying a flaw distribution for a certain plant. The representative flaw distributions themselves have, as you saw when we contrasted the two distributions that we use in our examples, have certain things that we need to

understand. You know, what's the effect for really small flaws?

But for that one piece of the flaw distribution, thermal hydraulic analyses are showing us that if the temperatures don't get too high, which appears to be the case, the small flaws are probably not a big concern. So I think the emphasis is really going to be on the shape of the distributions in that area, above 50 or 60 percent through wall where you really see the thermal risk coming from.

The other parts of this analysis when we try to understand, for example, the thermal hydraulic results much better, helps us key in on the other parts of the analysis that we may need to understand. It's really clear here, the flaw distribution questions are somewhat more focused now than they were just a few months ago, but there is work to do there if we're going to try to get any particular flaw distribution from a plant.

Event tree quantification. We mentioned failure frequencies. Right now in there are essentially guesses from 1150 for primary/secondary components to some extent. Sensitivity studies that were done on a list there try to account for either no integrity or perfect integrity on those components to see what the effects would be, and they're not huge by -- well, I don't want to

put up another slide right now, but -- but we saw the overall changes from 2 to 30 percent, and it's probably 2 to 20 or 25 percent for containment bypass, so even there we don't see a large contribution. But it's still going to have some effect.

Besides the quantification of the event tree, the event tree itself could change for different plant designs, and also different plants will probably have different contributors to the events of concern for the Hi/Dry scenario.

I did mention that for Surry, besides station blackout, loss of feedwater transients and the DC bus loss, are small components. Those things could change for other plants and those might have to be things that would modify the event tree relative to a station blackout if they're really big enough contributors.

Another point that should be made is that, everything I've talked about as far as RCP boundary failure goes, is hot leg surge lines and tubes. It's apparent to a lot of people that there's other parts of the reactor coolant system that could fail under these high temperature, elevated pressure conditions.

Now, in a study that's been done, the staff did take a look at that but not in any quantitative way. It's basically a discussion of how hot things will get and

not understanding the materials and their design, chances, judgment, whether or not they're going to fail or not under these conditions -- but that's not something that's been factored into the risk study. Again, it's a discussion after you crank through the numbers.

And just incidently, things like Manway gaskets, I think some valve bonnets -- the seals on valve bonnets -- the things that are going to be of particular interest if we're going to look at other places and the reactor coolant pressure boundary may fail ahead of the tubes under these kind of challenges.

Okay, we've talked a lot about thermal hydraulic results before today and a little bit today, and basically I think we're feeling a lot more comfortable than we did just a few months ago as far as the analysis that we've used.

I think the modeling issues with this latest work are going to be addressed, and it looks like there's a pretty convincing case that's made for -- not too drastic offset that could be applied to the basic results.

But one thing that has to be understand is that again, plant design specific features are going to affect those thermal hydraulic results -- you know, change some things relative to the one we use for the sample analysis.

Staff has done some other analyses with the CE plant. There are other analyses out there people have done for things like DCH studies and so forth for other U tube designs -- other PWR designs -- and the risk study would be expanded a lot by trying to factor in all those different kinds of designs, but when you look at the frequencies that we started with, the range of frequencies from the IPEs that we looked at, we already know there's a range that the results could fall into.

I think the best that could be done at this point is if one were to try to do this on a plant-specific basis, you not only have to understand of course, the flaw distribution for that particular plant, but you're going to have to understand the thermal hydraulic response for that particular plant under these kind of events.

Now, although I said the tube performance model itself is probably the component of the analysis that's of the highest confidence, I will say that a couple of things -- a couple of questions are left. What we're considering here is burst for these tests. That's the failure model considering creep failure leading to tube burst.

The change of leakage, maybe high leakage or leakage at these high temperatures impinging on other tubes and causing other tube failures as I mentioned

before, is not considered here, and when we did the risk analysis we basically said, anything that's going to burst is going to be a containment bypass. But there could be other failure modes that could be considered that could lead to the bypass situation.

Also, it's specific to axial cracks. And again, I'll just repeat that the circumferential cracks -- we haven't included in the risk study -- there is a flaw distribution out there for them, and there is work ongoing to understand response to cir cracks under these situations, but it's just not matured enough to the point that we could fold it into this kind of analysis yet.

And the last point on here is the creep failure prediction is dependent on understanding the temperature and pressure histories for a particular event. I don't think this is a large concern, but we do have to understand that when people do different analyses, as I mentioned early on today, we did see a different temperature history for the tubes from our analysis that our contractor did versus what was presented by an industry analysis.

There is an effect on the creep potential for the tube, and you can argue that the failure model could be altered somehow if you had a drastic change in the temperature and pressure history under those conditions.

That's all I prepared for today. If there's any more questions I have lots of backup material and lots of people --

CHAIRMAN FONTANA: Looks like a nice job to me. One thing that comes to mind. It may be worthwhile estimating the fission product attenuation in the secondary system, particularly if you have the break in a steam line. The hottest part of the steam generator probably would be somewhere near the inlet plenum of the steam generator, so if you get a leak there, if the tube burst there, two things would happen.

You'd have a high velocity out of the tube burst itself, but then it would impinge on the surrounding tubes and that gives you a lot of surface area, and since most of the fission products will be as aerosols in particulates, you get some attenuation there. And then the flow as it goes up through the tubes and spreads out will probably be kind of slow, so you get more attenuation. Then it has to go through the steam separator region.

So you know, it may be worth trying some back of the envelope calculations and see what kind of attenuation you might be getting.

MR. DONOGHUE: Well, I agree that it might be interesting to do that, to understand the release, but

from the aspects of trying to put together a Rule that addresses the risk, we wanted to make sure that we didn't lose defense-in-depth, and saw that having a focus on the containment bypass potential, just the potential of having a release path was what was really important to us.

CHAIRMAN FONTANA: Yes, I wasn't suggesting changing what you had, it's just a little better understanding of the fraction of the release that wouldn't get to the environment would be interesting.

CHAIRMAN SEALE: Are there any other comments that anyone -- any on the Committee wishes? Any questions?

(No audible response.)

Anyone want to make any comment? Go ahead.

MR. SCHNEIDER: Ray Schneider, ABB/CE. I guess a lot of the work that you're doing, a lot of the numbers in your APET are based on the NUREG 1150 logical structure, which is our first look at severe accidents and our first understanding as to how the plant may cope with severe accidents.

Over the past years plants have been -- or at least the industry has been asked to deal with severe accident management, looking at the issues. There's a general understanding that high pressure RCS scenarios are bad for a number of reasons, not just the steam generator

tube rupture issue.

And the guidance in most of the severe accident -- or maybe all the severe accident guidelines or at least most of them, at least ours -- is to basically do whatever steps you can to basically do two things: one, get water into the steam generator, and the other one is to depressurize the primary side.

And I see all this discussion, all this analysis, tremendous amounts of detail, tremendous amounts of sensitivity, and absolutely no consideration for the fact that the industry is aware of these issues and is trying to implement strategies and procedures to mitigate them, which should be included into the logic structure and so would basically eliminate probably 80 percent of your tree. Or at least put the significant part of your tree into the low probability range.

MR. DONOGHUE: Well, let me say that, you know, the Reg Guide that we've written up gives people the opportunity -- the licensees the opportunity to use updated information in the PRA to tell us that the event frequency itself was going to be low.

You know, we used information that we thought was -- maybe is somewhat dated -- but we thought was pretty reliable from past studies, and we understood from the onset that there was other information that people

could use to update, based on operator actions, like you're saying, EPGs and so forth, to effect that initiating frequency.

MR. SCHNEIDER: Actually, my concern is probably more of, there are two NRC programs: one to implement accident management guidance to reduce the risk of these events, and the other to demonstrate that the risk of these events is higher and we totally ignore the other piece of it.

And shouldn't there be kind of a meshing like -- you know, you have the same person kind of working on both ends of this task -- shouldn't there be a meshing of the fact that you anticipate accident management's guidance to basically be beneficial to the industry and reflect that in your Rule?

MR. DONOGHUE: Well, as a matter of fact, we had the people that were working on -- and he's right there.

MR. PALLA: Bob Palla with the staff. Let me just clarify that. This analysis that you've just heard about was based on 1150 results. We looked at the information coming out of the Level 1 plant damage status and we basically backed out from that, you know, the fraction of these events that occur with stuck-open relief valves, seal LOCAs, depressurized steam generators, etc.

We recognized that implicit in those numbers is some credit for operator actions, but that the credit at the time of 1150 would not have extended to severe accident management guidance. Now, what one would do if, you know, if we had more time we might have pulled a model off -- IRRAS-based model and perhaps added additional branches to the event tree to represent depressurization of the primary or operator actions on the secondary side.

But we did not do that. We realized that that's -- it's possible to improve the rigor of this analysis. One thing I want to just mention, though, is that there would be some difficulty in assigning probabilities to the operator successfully carrying out these actions, because accident management guidance are not as prescriptive as emergency operating procedures.

And frequently when one deals with issues about quantifying the reliability of an operator action they look to see just how explicit the guidance is to give some sense as to how much can I count on that action.

And the guidelines are more general in nature so there's, number one, a question about the human error probabilities that one would assign, and then furthermore, these are events -- at least for Surry -- we're looking at events that are dominated by station blackout, and it's a combination of early station blackouts that involve loss

of feedwater from, you know, basically at the initiator, as well as late station blackout.

And these -- you get into some very plant-specific questions as to whether the support systems that you would need to depressurize -- even if you have an accident management guideline that tells you to depressurize, you may not have the available support systems. I mean, your power operator relief valves will require air plus electrical power.

So these guidelines may be fine, and these are good objectives to depressurize, but one would have to go into the very details of the plant's capabilities to actually use those depressurization features, and that's one of the reasons we didn't try to do it, is because we wanted to get finished.

But licensees in a more complete analysis would want to look at their specific support systems and determine whether their guidelines, their accident management guidelines would provide the competence that they could depressurize.

MR. LONG: This is Steve Long. Let me remind you of one more thing. We did do a thermal hydraulic case to see what would happen if we opened a PRV and that bottled at about the time that the EOPs would say -- or rear accident management would say to open it. And it did

show in effect, it was beneficial and we did do -- or at least we're doing a sensitivity study -- I've forgotten which, if it's done yet -- to see what happens if that is maximized for --

MR. DONOGHUE: That was on -- yes, that was --

MR. LONG: So we have looked at the effect. The problem is what Bob was talking about. What's the probability of being able to succeed in getting that effect?

MR. SCHNEIDER: Well, I understand, but you have on one hand you're trying to -- you're pretty confident you can identify the flaw distribution to within a few percent, but operator actions aren't quite as clear.

MR. LONG: I wouldn't say that.

MR. SCHNEIDER: I know; I'm overstating the position. But the point is that you really have an issue that I think is equally as important as any other issue on the table. And I haven't heard the words accident management, you know, guidance, and recovery actions, and stuff like that, significantly discussed in the issues.

And that's the one message you want to get across to the industry I think, is that this is an important issue and you want to make sure that we respond to it, not just in calculations, but in tangible action should an event occur.

MR. LONG: I understand.

CHAIRMAN SEALE: That is a good point. The influence of emergency or accident management activities on the motivation of the operators to do that job appropriately and so on, could very well influence the human factors; that is, the success of actually doing it. So the relationship needs to be drawn somewhere to help make that point fairly obvious.

Any other questions or comments?

(No audible response.)

Well, the Committee is concerned about having enough time and enough opportunity to really prepare -- I'm sorry. Thank you very much, but don't leave yet, please, okay? About having enough time and an opportunity to really do justice to the efforts that have been put forth by both the staff and the industry in preparing your presentation so that we really reflect the best we can in the letters or whatever messages we may see fit to send, either to the EDO or to the Chairman and the commissioners.

For that reason, we would like to now try to discuss any issues that -- the things that the members of the Committee have identified as being important to them, and use that then, as guidance for what it is you might wish to say tomorrow, either the staff or the industry,

when you meet with the full Committee. And then from that we can also synthesize the kinds of things we might wish to include in any formal communication.

Now, one other thing. The presentation on this subject is scheduled from 8:45 to 10:45 tomorrow morning. We have received a request from Dr. Hopenfeld to present the features of a differing professional opinion he filed back in 1994 on this subject, and we estimate that that's going to take somewhere around, oh, 20 minutes or so. We would like to reserve some time for the Committee, after we hear everyone.

So doing all that manipulation and all, it looks like the staff should count on perhaps, 45 minutes tomorrow to do its discussions, and so we might keep that in mind when we start trying to identify issues, both members of the Committee and your response to it. And I guess I think the industry might want to have perhaps, 15 minutes to summarize your position as well.

So with that background, I'd like to ask the members of the staff -- or the members of the Committee, to identify the things they might think would be appropriate to present to the Committee as a whole.

Dr. Powers, do you have some comments?

MEMBER POWERS: The Rule that's been put forth here is extremely complicated. It has a lot of elements

in it, and it's diverse in its nature. And quite frankly, I still haven't gotten the length and the breadth to know why each one of these major blocks is there.

But it strikes me that, the first and foremost thing is to present to the Committee as a whole, the major blocks. There's material on characterization of the tubes, there's material on monitoring, there's material on risk analysis. And probably at no greater level of detail than that, present those blocks and say, here's what we're trying to accomplish with each one of these blocks.

I think that would be useful in an abbreviated presentation of just 45 minutes, to kind of cast things in that format.

Then there are lots and lots of detailed things about each one of those blocks that I have to admit, I still haven't gotten my hands around, and I think they just invite protracted discussions --

CHAIRMAN SEALE: We've noticed.

MEMBER POWERS: -- on those things, because you know, for the life of me this .05 business and .025 and 1 x 10 to the minus -- there are a lot of probabilistic criteria that I don't understand. If they can help me with a sentence that wouldn't get Dr. Apostolakis launching off into epistemic analiatory -- which I think is impossible -- I'd appreciate them helping

me but I -- in defense of our schedule, maybe they should help me offline.

CHAIRMAN SEALE: We're not sure we want to provide that challenge.

MEMBER POWERS: However, to help me, they must avoid the words epistemic analiatory.

CHAIRMAN FONTANA: He's not here to defend himself.

MEMBER POWERS: And that's why I bring it up, because his defense is withering as well. That would be the first bit of guidance that I would make.

Then I think, Mr. Sheron's introduction on this subject where he laid down -- here are the issues and the areas that I'd like to get some guidance from the Committee as a whole -- was a very helpful insight to this that got elaborated as we listened to some remarks made by representatives of the industry.

And to the extent that they could articulate those to issues they had, which I believe, if my notes are correct, one was that there were concerns that the Reg Guide might be too prescriptive.

On the other hand they were concerned about the amount of flexibility they ought to grant to the industry because they feel like they come in very much after the fact on this. I think it's important to

articulate the area they'd like to have some advice on.

And finally, I think that it would be useful for them to articulate very carefully, but at still a fairly broad-brush level, how it is that risk gets factored into this. And I say at a broad-brush level because there's lots and lots of detail here, and each one of those details invites lots of questions about what particular probability distribution you used, or how you broke down the event tree, but I would say more at a philosophical level how risk has been factored into this, and whether it is really risk or it is really core damage frequency that's been factored into this.

There was, at a point in the presentations yesterday, a comment made that -- I wrote it down in the notes but quite frankly, it eluded me as I listened to the presentation -- was that they had chosen to put their conservatism in the source term calculation rather than in the failure calculations.

Well, quite frankly, it looks to me like there's lots of conservatisms in the failure calculations and I'm not terribly wild about putting unquantified conservatisms in on things in the midst of calculations. But I'd like to understand that just a little bit better because I think there's more meat than that comment elicited from me at the time it was made.

Those are the bits of advice I have.

CHAIRMAN SEALE: It's rather comprehensive, I'd say.

MEMBER POWERS: It is one of the most diverse Rules that we've ever had presented to us, and quite frankly, let me echo Dr. Fontana's comment that I was quite impressed with today's presentation in the length and breadth of their attempts to try to understand the phenomonology here in some sort of a probabilistic context. Quite impressive the amount of work they've done on this.

And equally well, I think it's quite impressive the amount of work they've done in all parts of this Rule. This is a monumental effort that they've undertaken, and we should spend some time on this.

CHAIRMAN SEALE: Yes, I think that it's -- earlier the comment was made that they went so far in their analysis and then they had to stop, or they knew when to stop, or whatever -- or they had to make a decision. And it's very clear that you had that problem repeatedly, because you had such an incredible number of alleys that you could start running up with your calculations and never get back out of the mess if you didn't do that.

MEMBER POWERS: And that is an area that they

should think about in formulating a presentation with the Committee, is frequently we heard -- well, we just didn't have time to do this, or this is something we're fixing to do, or maybe one of these days we'll get around to doing this -- is just how close are we?

Are we really ready to get this thing out on the board, are these leftovers that important or unimportant, or have we really got it sorted out before we send out for public comments or can we do it after? Some sort of a context to know where we are.

CHAIRMAN SEALE: Dr. Shack, do you have any comments?

MEMBER SHACK: Yes, I guess Dana alluded to it, but let me just emphasize it again. I think, you know, one of the things that's going to come up is this question of, what level of review is going to be given to what level of documents? And the staff has told us that the current plan is not to review the detailed documents, the methods in a sense.

And I guess I'd just like some words about why they feel that's the right way to -- you know, Jack mentioned that, you know, they were drawing this box to try to describe acceptable methods in the Reg Guide. What are the advantages of doing that versus a simpler Reg Guide and then review the procedures, and why you seem to

opt for one approach over the other.

Then of course, the -- and again, the question of how the .05, you know -- or, how the performance criteria, whatever number one happens to pick, is going to be, you know, related to risk. Something that just sort of has to be -- come up again.

CHAIRMAN SEALE: Anything else? Dr. Kress?

MEMBER KRESS: I think I'd like to have a better -- on how one arrived at the acceptable probabilities for one tube failure versus two tube failures versus three or more.

MEMBER POWERS: With or without epistemic analiatory in it?

MEMBER KRESS: Yes, without. Please. I thought a little better -- more discussion on the factor of 3, -- factor would be useful. The discussion we had today I thought was very good. I would like to see a repeat of a lot of that, some of that, plus --

CHAIRMAN SEALE: At least a summary, a summary.

MEMBER KRESS: A summary. But particularly, I'd like to hear more about the reasoning behind not looking at fission product transport and the site specific qualitative health objectives, as opposed to conditional containment failure probability as an acceptance criteria.

Why that would not be also an acceptable approach.

And I think that's all I had on it.

CHAIRMAN FONTANA: Well, after three smart guys have talked there's not a whole lot left to say. The thing is though, in trying to read the Rule, you see that you're trying to patch on to a -- what's originally a deterministic, if you want to call it that -- sort of exposition, and trying to patch on some risk-based thinking onto it.

And the impression I get is, you end up with something like a 3-hump camel, which will probably get you across the desert but probably would have pretty nasty disposition, so you don't know when he's going to spit at you.

I would like to see that thing kind of reduced down to essentials if it's at all possible, with as much application of risk-based thinking as possible, which -- well, that goes far enough.

CHAIRMAN SEALE: Okay. Well, I, as I went through here I've tried to identify in my own mind the issues that Dr. Apostolakis would have raised if he had been sitting there, and I won't use the magic words. But certainly the question of how risk gets factored in, that Dana raised; the question of the level of review of the documents that Dr. Shack raised; and Dr. Kress's point

about the acceptable probabilities for single and multiple tube failures.

Those are all issues I think, that Dr. Apostolakis would have added. I guess the other comment I would make would, perhaps, be more addressed to the industry and that is that -- and I certainly concur with the comments that have been made here -- but I would think that it would be worthwhile for the industry people to briefly outline the approach that they took for the Rule, because at least formally it seemed to me, there was a little bit more of a risk-based initial approach to setting the case up, if you will. I think you know what I mean.

And I think that would be very useful to the members of the Committee to see that somewhat different approach. Other than that, I think we know what we want to do as far as the presentations tomorrow are concerned.

Any questions? Going to be able to do it?

MR. STROSNIDER: No, I don't -- this is Jack Strosnider from the staff. I don't think we have any questions. The issues are clear; we'll have to see if we can cover it in 45 minutes.

CHAIRMAN SEALE: Okay, fine.

MR. STROSNIDER: We'll make an attempt.

CHAIRMAN SEALE: I understand. It's a tough

one. Any questions or comments from the industry at this point?

(No audible response.)

I have one other thing to ask. Yesterday at some point someone talked about the possibility of this representing a backfit kind of activity. And whether or not that's the case, at some point -- and it may be premature today or tomorrow -- but at some point it seems to me that it would be very worthwhile to hear if anyone has done a risk benefit assessment of what this Rule does.

And that's now in the broader sense of, you know, how many -- you know, what is the impact on tube availability and what are the potentials for saving inspection times and all of that sort of thing? Or what are the costs of increased inspection times, and so forth.

Did you want to comment on that?

MR. STROSNIDER: Yes. This is Jack Strosnider. Just a comment on -- that analysis is being performed. I think it's probably middle of December before it will be complete. It has to be performed as part of our promulgation of the new Rule. We have to determine under what parts of the backfit Rule you would do these different things.

So the work is in progress. I'm afraid it probably is a little early for us to lay out all the logic

there because it's still being done.

CHAIRMAN SEALE: I can understand that. On the other hand, I think it would be very helpful to us in assessing what's involved here and what the traits may be, for us to hear about that when it is available.

MR. STROSNIDER: Yes, I guess, as I indicated, we'll be looking at whether these things or backfit is under compliance, or safety enhancements or cost beneficial -- that will be included in the package that comes around for office review and concurrence, which the Committee will also a get a copy of. And I guess my only comment there is, the Committee would have to decide whether reading that package is adequate or if you want to hear from us again in our presentation.

CHAIRMAN SEALE: Very good. Are there any other questions that we need for the record, at this point?

(No audible response.)

Well, we'll dismiss this session. I understand there's another Subcommittee meeting this afternoon.

(Whereupon, the ACRS Joint Meeting was concluded at 11:40 a.m.)

C E R T I F I C A T E

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of:

Name of Proceeding: ACRS JOINT MEETING: MATERIALS AND METALLURGY AND SEVERE ACCIDENT SUBCOMMITTEES

Docket Number: N/A

Place of Proceeding: ROCKVILLE, MARYLAND

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

Irene Gray

IRENE GRAY
Official Reporter
Neal R. Gross and Co., Inc.

INTRODUCTORY STATEMENT BY THE CHAIRMAN OF THE
MATERIALS & METALLURGY AND SEVERE ACCIDENTS JOINT SUBCOMMITTEE
11545 ROCKVILLE PIKE, ROOM T-2B3
ROCKVILLE, MARYLAND
NOVEMBER 5-6, 1996

The meeting will now come to order. This is a meeting of the ACRS Joint Subcommittee on Materials & Metallurgy and Severe Accidents.

I am Robert Seale, Chairman of the Subcommittee.

The ACRS Members in attendance are:

Ivan Catton, Mario Fontana, Thomas Kress, Dana Powers, and William Shack.

The purpose of this meeting is to hold discussions with representatives of the NRC staff, the Nuclear Energy Institute (NEI), and the Electric Power Research Institute (EPRI) to gather information concerning the technical approach used in developing the proposed risk-informed, performance-based rule and regulatory guide associated with steam generator tube integrity. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions as appropriate, for deliberation by the full Committee.

Noel Dudley is the Cognizant ACRS Staff Engineer for this meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the Federal Register on October 21, 1996.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register Notice. It is requested that the speakers first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

We have received no written comments or requests for time to make oral statements from members of the public.

(Chairman's Comments, if any)

During the June 12-14, 1996 ACRS meeting, the Committee heard presentations by representatives of the staff and the Nuclear Energy Institute (NEI) on this matter. Today the Subcommittee will hear from the staff concerning the technical basis for the proposed rule.

We will proceed with the meeting and I call upon Brian Sheron, NRR, to begin.

RESULTS FROM APET ANALYSES

- Under base case assumptions, ~1 in 5 chance of containment bypass, given core damage with high primary pressure and dry SGs

- Majority of bypass events (90%) assigned to large release category (RC-1 on APET)

  - Direct release to environment via open ADV or MSSV
  - No delineation in APET between single and multiple tube ruptures due to inability to assure integrity of adjacent tubes

- Balance of events (10%) have lower releases due to partially depressurized RCS, intact (but leaky) MSIVs, and secondary side holdup/deposition (RC-2 on APET)

- SBO with RCP seal LOCA is dominant contributor to thermally-induced SGTR frequency (70%), and total containment bypass frequency (50%)

- Conditional containment bypass probability found to range from 0.1 to 0.4 in preliminary sensitivity analyses (to be updated)

  - minimum of 0.02 for optimal secondary side integrity
  - maximum of 0.3 for 70K increase in temperature history

- Additional sensitivity analyses planned

Proposed Steam Generator Rule
Safety Analysis

ACRS Materials and Metallurgy Subcommittee and Severe Accidents Subcommittee

November 6, 1996

Joseph Donoghue, DSSA/SRXB
Office of Nuclear Reactor Regulation
(301) 415-1131

AGENDA

- Contributors to Induced SGTR Risk
- Accident Progression Event Tree
- Thermal-Hydraulic Analysis Results
- Flawed Tube Failure Model
- Flaw Distribution
- Tube Failure Probability Calculation
- Results/Sensitivity Studies
- Key Issues/Limitations

CONTRIBUTORS TO INDUCED SGTR RISK

- Spontaneous and induced tube failures contribute to SGTR risk

  - Spontaneous failures often due to unknown mechanisms
    Assumption: No impact by rule on spontaneous failure frequency.

  - Induced Failures:
    - Pressure - ATWS (low frequency),
      Secondary Depressurization
    - Thermal - Severe Accident
    - Mechanical - Impact from spontaneously failed tube

- Previous studies concluded that severe accidents would not challenge tube structural integrity (e.g, NUREG 1150, DCH studies)

- Development of new degradation mechanisms and additional insights in high temperature tube performance

- Significant questions: Understanding of predicted RCS conditions and RCPB response during severe accidents

MECHANICALLY-INDUCED TUBE FAILURE

- Potential for a circumferentially failed tube to cause subsequent failures

- Two events involving circumferentially failed tubes:
  - North Anna - 1 July 17, 1987 Mihama - 2 February 9, 1991

- Each failure was a complete circumferential break due to high cycle fatigue

- North Anna:
  - Tubes adjacent to failure site were in service
  - Eddy current examination, no reported damage to tubes adjacent to break

- Mihama:
  - Most tubes adjacent to failure were in service
  - Extensive examination of adjacent tubes: contact had occurred with broken tube; contact traces on tube deposits without any denting of the tubes.

- Maine Yankee Circumferential Cracking:
  - Calculations submitted showing that impact and jet impingement forces would be less than the shear load margin for a cracked tube.

ACCIDENT PROGRESSION EVENT TREE (APET) STRUCTURE

- Treats subset of core damage events: "Hi/Dry" events
  Elevated primary/secondary ΔP and dry SGs at core uncovery

- "Hi/Dry" events sorted by RCS and secondary system condition at core uncovery

  - Primary system:
    Intact (high P)
    Stuck-open PORV/SV
    RCP Seal LOCA

    Secondary system:
    Intact (all SGs at pressure)
    Stuck-open ADV/MSSV
    Manual Depressurization

- Pressure-induced rupture of flawed tubes at normal temperature (PI-SGTR) evaluated since not considered in baseline PRA model

APET STRUCTURE

- Additional mechanisms for late primary/secondary depressurization also evaluated

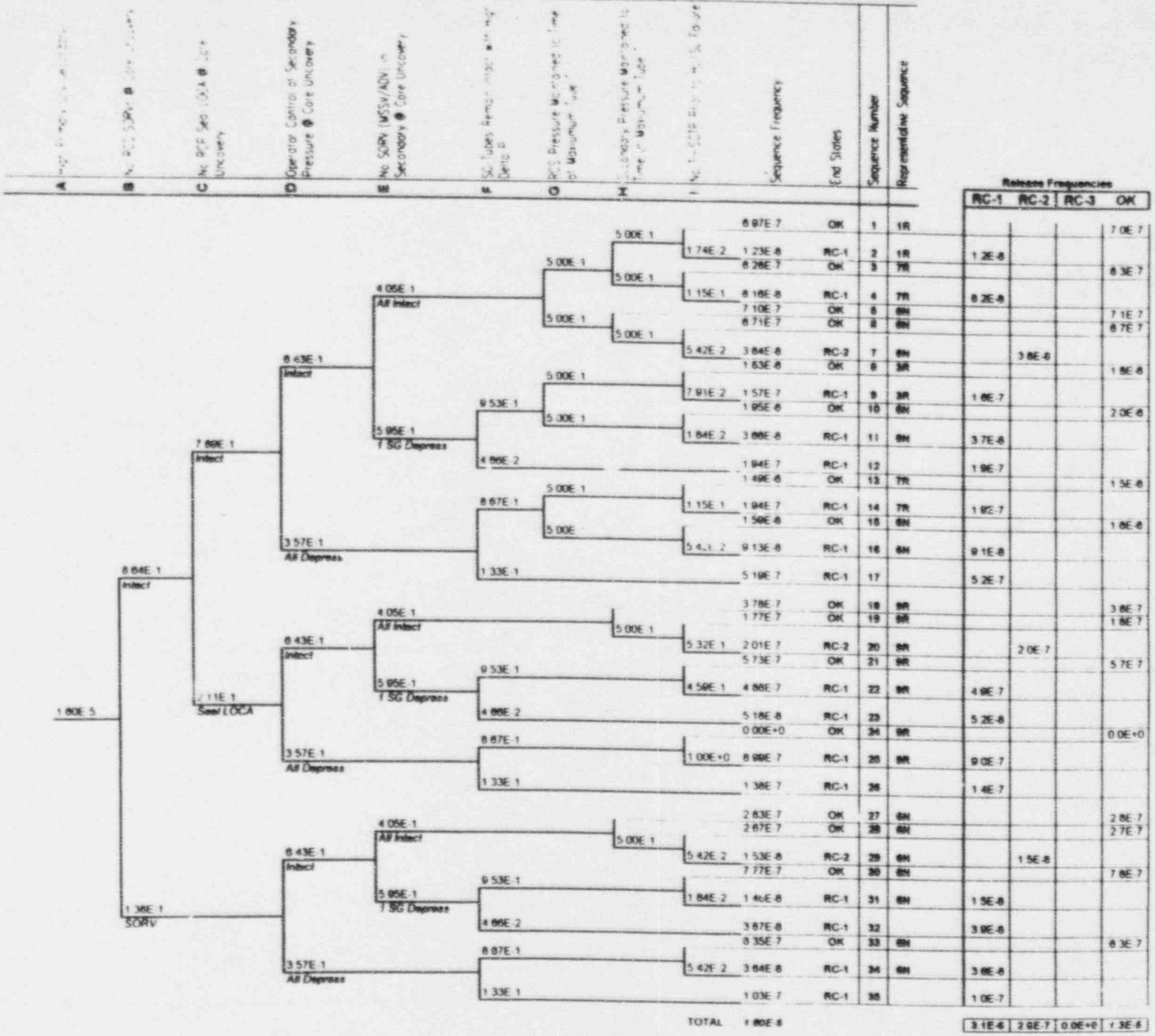
  - Stuck-open PORV/SV following core uncovery and heatup

  - MSIV leakage following SG dryout (not considered in past PRAs but suggested by anecdotal evidence)

- Potential for thermally-induced tube rupture (TI-SGTR) prior to HL/SL failure evaluated as final event on each APET branch

B..E CASE: **Reference Plant**

ACCIDENT PROGRESSION EVENT TREE



APET QUANTIFICATION

- Top events quantified based on NUREG-1150 analyses for Surry and Sequoyah, and survey of IPE database
- APET entry frequency ≈ 2E-5/reactor-year
  - NUREG-1150 plants dominated by LOOP events, with balance from loss of feedwater transients and loss of DC bus
  - IPE survey suggests leading contributors highly plant-specific
- Primary system status (early):
  - Intact in majority of events (70% for Surry, 50% for Sequoyah)
  - RCP Seal LOCA contribution where RCS failure occurs with dry secondary (20% for Surry, 50% for Sequoyah)
  - PORV/SV failure a minor contributor (10% for Surry, 3% for Sequoyah)

APET QUANTIFICATION

- Secondary system status (early):
  - Probability that 1 or more SGs will be depressurized is significant in Surry-1150 (75%), but minimal in Sequoyah-1150 (5%)
  - Attributed to differences in ADV dependencies and procedures for SG depressurization during an SBO
- Likelihood of late primary and secondary system depressurization assigned scoping values of 0.5 in base case and assessed via sensitivity analyses, due to lack of information on valve performance
  - Pressurizer PORV/SV reliability under repeated cycling and severe accident temperature conditions
  - MSIV leak rates (MSIVs not included in Appendix J leak rate test program)

APET QUANTIFICATION

- A single accident sequence was selected to represent the family of sequences defined for each branch
  - Limited variety of scenarios analyzed
- Probability of TI-SGTR quantified based on stand-alone calculations
- Probabilities of PI-SGTR and TI-SGTR for each branch adjusted to account for number of intact/depressurized SGs (and loop seal clearing in RCP Seal LOCA events), and imported into APET
- Final APET quantification provides the frequency of containment bypass due to induced SGTR resulting from core damage events

THERMAL-HYDRAULIC ANALYSES

- Used SCDAP/RELAP5 results from Surry model
- Cases used in safety assessment:

Case Designation	Case Description
RES 1	Base Case: SBO, loss of AFW
RES 3	One SG depressurized (failed relief valve)
RES 7	ALL SGs depressurized
RES 9	RCP Seal leaks, one SG depressurized
NRR 6	PORV fails open, One SG depressurized

- Calculations performed to estimate tube conditions at time of predicted RCS pressure boundary failure and relative times to failure for major RCPB components (hot leg, surge line, tubes)
- Results appear somewhat conservative based on recently completed sensitivity studies

TUBE FAILURE MODEL

- Model based on high temperature testing of machine-flawed tubes using original thermal-hydraulic analysis results
- Only axially oriented cracks considered
- Results indicate that creep failure is dominant mechanism
- Indications were that unflawed tubes would withstand thermal challenge
- Includes uncertainty in effect of crack
- Includes material variability effect on creep failure prediction

TUBE FAILURE MODEL

Creep Failure Prediction

$$\int_0^{t_f} \frac{dt}{t_R(T, m_p \sigma)} = 1$$

Time to Rupture

$$t_R = 10^{\frac{P_{lm}}{T} - 15}$$

Crack Magnification

$$m_p = \frac{1 \quad \alpha \frac{a}{mh}}{1 - \frac{a}{h}} (1 \pm 0.06)$$

Larson-Miller

$$P_{lm} = (23.2 \pm 0.7 - 2.4 \ln\sigma) \times 10^3$$

Crack Magnification (through-wall)

$$m = 0.614 + 0.481\lambda + 0.386\exp(-1.25\lambda)$$

Normalized crack length

$$\lambda = [12(1 - \nu^2)]^{\frac{1}{4}} \frac{c}{\sqrt{R_m h}} = 1.82 \frac{c}{\sqrt{R_m h}}$$

TUBE FAILURE MODEL

Where:

a	crack depth
c	semi crack length
σ	hoop stress
P_{lm}	Larson-Miller parameter
R_m	mean radius of tube
T	temperature
α	crack depth parameter
m	magnification factor for through-wall cracks
ν	Poisson's ratio
h	tube wall thickness

FLAW DISTRIBUTIONS

- 'Representative' - based on operational data and inspection information
- Inspection methods not capable of defining a plant-specific crack size distribution
- Plant-specific attributes
- Two approaches developed: - Each uses three categories of distributions

<u>RES</u>
- 6 Degradation types
- Continuous function of length and depth

<u>NRR</u>
- Axial cracks only
- 2 specific flaw lengths

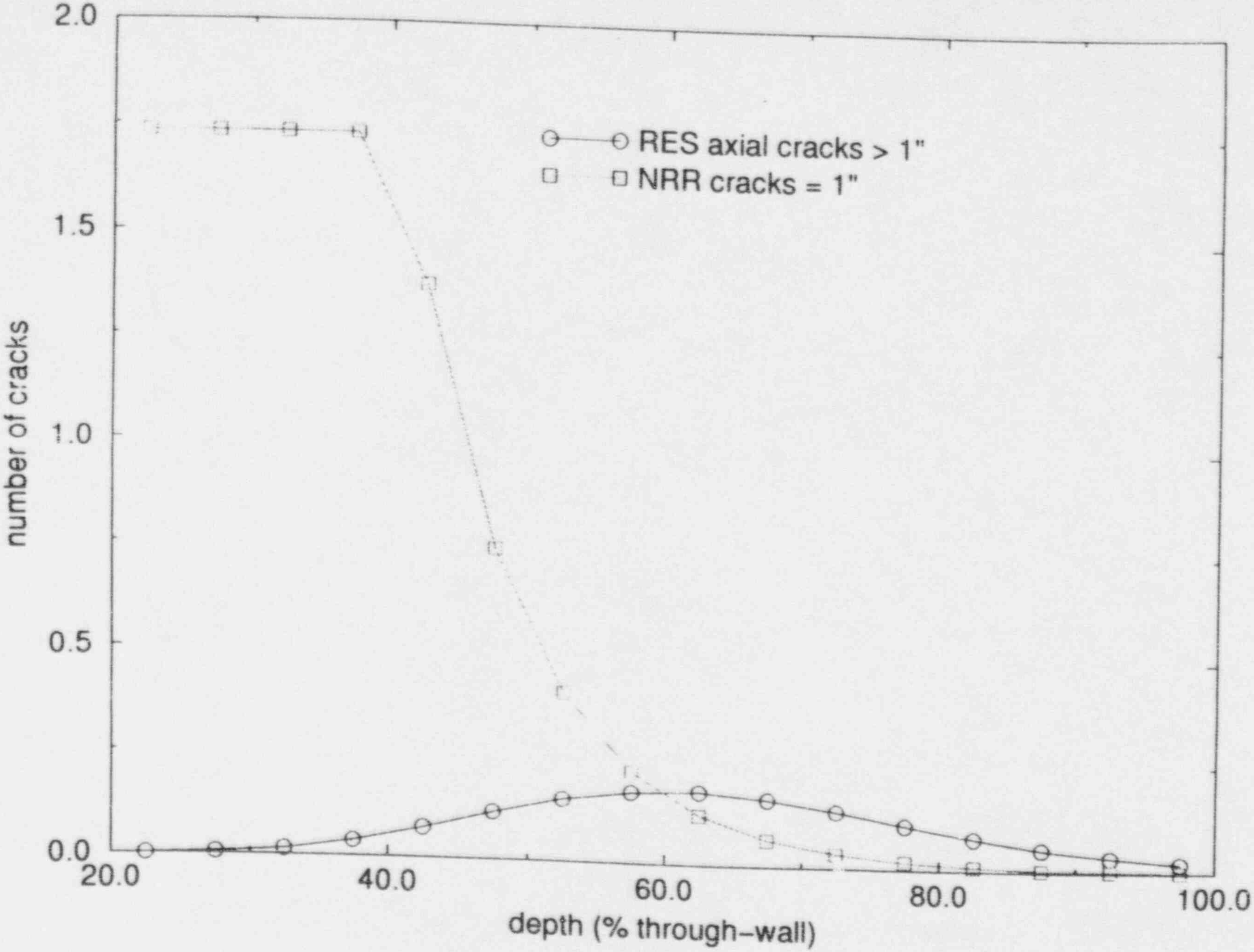
- Creep failure calculations for axial cracks only
- Neither distribution realistic for shallow cracks

FLAW DISTRIBUTIONS

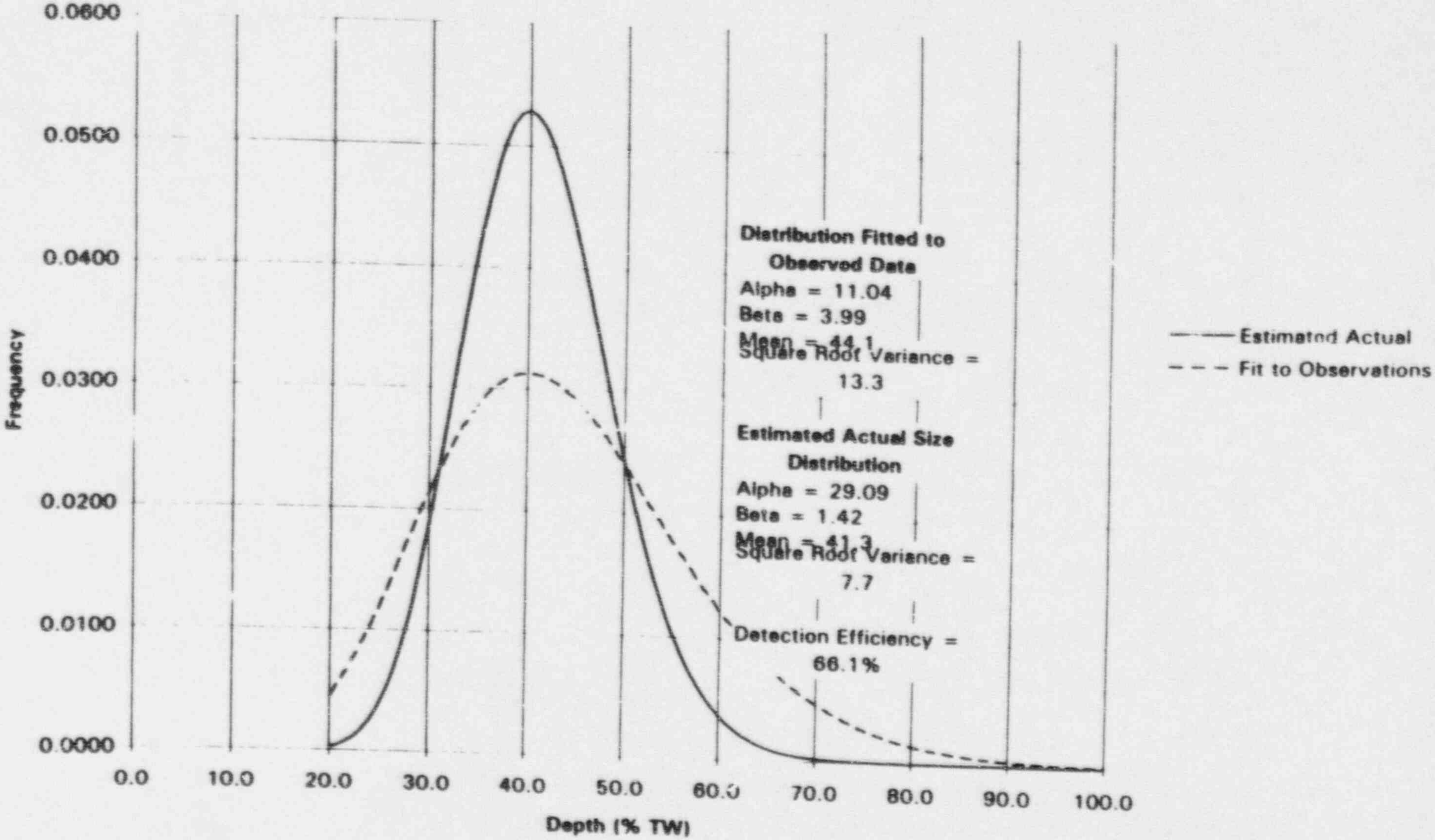
- RES distributions developed for six types of degradation
  - \- Circumferential SCC at TTS
  - \- Circumferential ODSCC at TSP dents
  - ⇒ Freespan ODSCC
  - ⇒ IGA/SCC in hot leg sludge pile
  - \- Axial ODSCC at TSPs
  - \- Flaws from loose parts

⇒ Indicates used in risk study

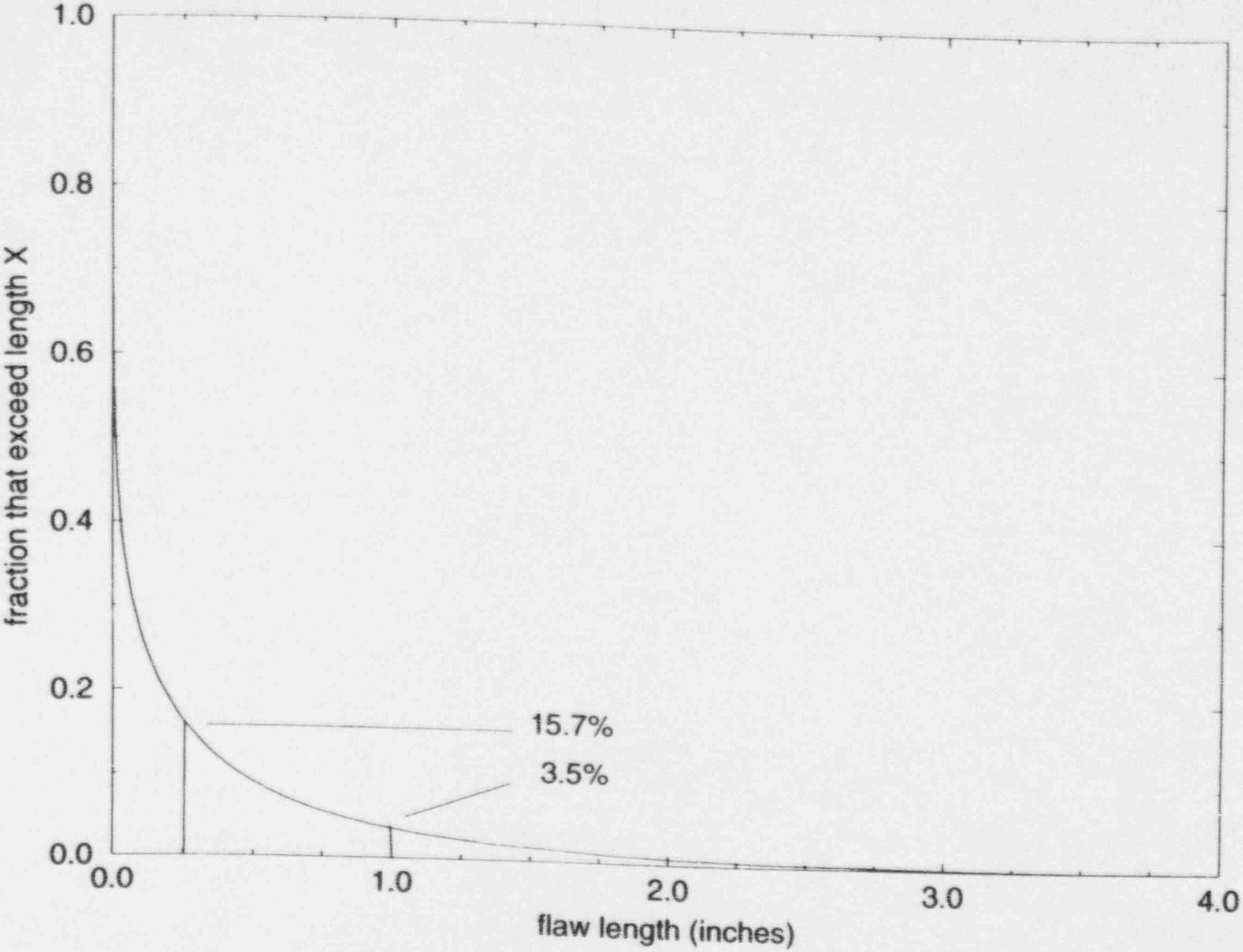
Comparison of NRR "Average" and RES "Moderate" Distributions



RES DISTRIBUTION **Free Span Defect Depths**



RES Flaw Length Distribution



FLAWED TUBE FAILURE PROBABILITY

- Event tree uses split fractions for pressure- and thermally-induced tube failure
- Split fractions change depending on predicted thermal-hydraulic conditions
- Probabilistic models developed for flawed tubes, hot leg and surge line creep failure
- Calculation uses flaw distributions
- Sensitivity studies:
  - Tube temperature
  - Creep model magnification factor
  - RCS pressure
  - Different flaw distributions

FLAWED TUBE FAILURE PROBABILITY

- Failure probability of surge line or hot leg versus tubes as a function of time
  - Used SCDAP/RELAP5 temperature/pressure results
  - Included distribution of Larson-Miller creep characteristic for materials
- Flaw distributions divided into 2 length bins
  - Exceeding critical length at normal temperature and ΔP at PORV setpoint
  - Shorter flaws exceeding critical length at temperature associated with core damage
- Flaw distributions divided into depth bins spanning 5% thickness
- Creep model for axially cracked tubes used to calculated creep damage index for each of 32 flaw size bins for each thermal-hydraulic case
- Probability that another RCPB component would not fail prior to the tubes based on time at which flawed tube creep damage index equals 1.0

FLAWED TUBE FAILURE PROBABILITY

- Monte Carlo method used to average flaw bin failure probability over range of:

  - INCONEL 600 Larson-Miller Parameter
  - Tube wall thickness
  - Tube diameters
  - Crack length and depth bins

- Limit-load analysis basis for pressure-induced tube failure calculation under normal temperatures

- Probability of tube failing first in each sequence found using results of failure probabilities for each bin combined with population of flaws in each bin

- Flaw distribution not used above 100% through-wall depth

- Flaws expected to burst in normal service were removed from distribution

- Flaws expected to burst under elevated pressure at normal temperatures were removed from creep calculation

FLAWED TUBE BURST PROBABILITY - RESULTS

RES "Moderate" Plant Flaw Distribution

Sequence Designator/Description	Pressure Induced	Temperature Induced
RES-1 No SGs Depressurized	0.0	0.0174
RES-9 Seal LOCAs - Depressurized SG	0.0549	0.1671
Depress. SG, clear loop seal	0.0549	[1.0]
Intact SG, loop seal intact	0.0	0.0
Intact SG, cleared loop seal	0.0	0.0207
RES-3 One SG depressurized	0.0549	0.0791
NRR-6 Pzr PORV open, 1 SG depress.	0.0549	0.0184
RES-7 All SGs Depressurized	0.1646	0.1197

FLAWED TUBE BURST PROBABILITY - RESULTS

NRR "Average" Plant Flaw Distribution

Sequence Designator/Description	Pressure Induced	Temperature Induced
RES-1 No SGs Depressurized	0.0	0.0092
RES-9 Seal LOCAs - Depressurized SG	0.0059	0.0774
Depress. SG, clear loop seal	0.0059	[1.0]
Intact SG, loop seal intact	0.0	0.0029
Intact SG, cleared loop seal	0.0	0.0108
RES-3 One SG depressurized	0.0059	0.0390
NRR-6 Pzr PORV open, 1 SG depress.	0.0059	0.0054
RES-7 All SGs Depressurized	0.0175	0.0547

FLAW BURST PROBABILITY - SENSITIVITY

- Temperature sensitivity: considered wide range due to earlier questions regarding thermal-hydraulic uncertainties.

- Distribution sensitivity

  - RES distributions yield higher burst probabilities except for temperature errors $\geq$ 70K

- Creep model flaw stress multiplier sensitivity:

  - 95% confidence level $\approx$ +10K temperature error

- Pressure sensitivity:

  - 100 psi $\approx$ 15K temperature difference

APET SENSITIVITY CASES

	Sensitivity Case	Represents
1.	No late SG depressurization due to MSIV leakage	Potential impact of confirming MSIV leakage integrity
2.	Lower early SG depressurization probability (0.05 based on Sequoyah 1150)	Plants without procedures/capability to manually depressurize in SBOs. Better MSSV reliability as claimed by EPRI
3.	Cases 1 and 2 combined	Optimal secondary side performance
4.	Probability of late primary depressurization = 1.0	Potential impact of providing AC-independent depressurization capability. High probability of PSV failure for liquid cycles as claimed by EPRI
5.	Probability of late primary depressurization = 0.	PORVs continue to reseat under repeated cycling or fail with small leak area. Operators fail to manually depressurize using PORVs.
6.	No RCP seal LOCAs	Reduced contribution of RCP seal LOCAs at plants with AC-independent seal cooling systems. Lower probability of RCP seal LOCA with Byron Jackson pumps (CE plants)
7.	Increase temperature histories by 70K (RES flaw distribution)	Impact of upper bound temperature estimate on TI-SGTR probabilities

THERMAL-HYDRAULIC MODELING SENSITIVITY

- RES recently completed sensitivity calculations using the Surry SCDAP/RELAP5 model

- Cases used:

RES 3	One SG depressurized (failed relief valve), 35/65 (hot/cold) tube bundle split
RES 6	One SG depressurized (failed relief valve), 53/47 (hot/cold) tube bundle split

- Sensitivities:

RES 6A	+20% heat transfer correlation (HL, SL, upper plenum, SG tubes)
RES 6B	-20% heat transfer correlation
RES 6C	+30% heat transfer correlation (Entrance to HL, SL, and SG tubes)
RES 6D	Increased heat transfer correlation (SG tubes only)
RES 6E	Radiation heat exchange in hot leg
RES 6F	Lower 5% limits on mixing model parameters

THERMAL-HYDRAULIC MODELING SENSITIVITY

Selected structure temperatures from Surry SCDAP/RELAP5 calculations

Case	Time difference Surge line to tube failure (min)	Structure temperature (K/°F) @ surge line failure time	
		surge line	hot SG tube
3	17	1259/1806	973/1291
6	21	1254/1797	957/1263
6A	22	1260/1808	938/1228
6B	20	1258/1804	964/1275
6C	21	1259/1806	944/1239
6D	20	1255/1799	957/1263
6E	25	1256/1801	937/1227
6F	13	1257/1803	1007/1353

KEY ISSUES/LIMITATIONS

- Representative Flaw Distribution
  - Significant uncertainties remain in characterizing distributions
  - Inspection capabilities do not provide plant-specific distributions based on flaw size

- Event Tree Quantification
  - Failure frequencies for pressure relief components in RCS and secondary are not well known
  - Plant-specific configurations and procedures could affect structure and split fractions

- RCPB Weak Points
  - Only qualitative treatment for potential of other component failures (other than hot leg, surge line, tubes)

KEY ISSUES/LIMITATIONS

- Thermal-Hydraulic Results
  - Plant- and design-specific factors could affect thermal-hydraulic response
  - Resolution of modeling issues

- Tube Performance Model
  - Based on high temperature tube burst tests
  - Does not consider other failure modes (failure due to leakage)
  - Specific to axial cracks
  - Creep failure prediction dependent on understanding temperature and pressure time histories