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The issue was really that we are not really certain -- and I don't know who we is, but someone is not really certain that the MAAP results are valid for these analysis.

And we want you to confirm the validity by some checks with your design basis codes, and that is really the story that we are trying to pursue here.

CHAIRMAN APOSTOLAKIS: The core makeup times, and I am sure that I don't understand you. You can have zero, or one, or two?

MR. SCHULZ: Yes.

CHAIRMAN APOSTOLAKIS: Okay. So in some cases, and let's say you need -- you decide that your best case is that you need one of the two. Now, you are using a code to do the calculations and so on, and you say, gee, I have uncertainty here.

MR. SCHULZ: Uncertainty?

CHAIRMAN APOSTOLAKIS: Uncertainty in the result, and that in fact it is one that you need.

MR. SCHULZ: Okay. In terms of the core cooling?

CHAIRMAN APOSTOLAKIS: Yes, the core cooling capability. So I don't think that you went back and did what Dr. Ransom suggested, to look at perhaps the correlations that you have used for other models in the code and assign uncertainties, and you didn't do that?

MR. SCHULZ: No.

CHAIRMAN APOSTOLAKIS: But what is not clear to me is why did you do? I thought -- is it that you are saying that instead of assuming one core makeup time at a certain flow rate, I will have something less than that, and prove that it is still adequate, or do you do something else?

MR. SCHULZ: I did something else, and I think it would be better to in the last half of this presentation --

CHAIRMAN APOSTOLAKIS: If you are going to address it later, that's fine, but this question is unclear to me, and it is not clear to me how it was handled. But I know that it was not handled the way that some academic in the clouds would do it.

MR. SCHULZ: Yes, I agree with you. Hopefully the last part of my discussion will clarify that, and if it doesn't -- but right now what I was trying to talk about here is the success rate criteria analysis done with MAAP.

And we had considered this to be a success rate for the AP-600 with this longer core uncovery for AP-1000.

CHAIRMAN APOSTOLAKIS: So it was considered a success, and I think that comes back to Mr. Rosen's question.

MR. SCHULZ: Yes.

CHAIRMAN APOSTOLAKIS: It was a success, even though you uncover, you know, 2 or 3 feet of the core, because the temperature never reached --

MR. SCHULZ: Yes.

MEMBER ROSEN: And there is no fuel damage?

CHAIRMAN APOSTOLAKIS: And there is no fuel damage?

MR. SCHULZ: Yes.

MEMBER SIEBER: But they didn't calculate the temperature, right?

CHAIRMAN APOSTOLAKIS: Well, they said they did.

MR. SCHULZ: We got temperatures out of MAAP. They are not as precisely calculated as we do for design basis analysis. But it gives you a good feeling for if you are going to have damage in the core, and core melting.

MEMBER SIEBER: But they had enough margin, right?

MR. SCHULZ: Yes.

CHAIRMAN APOSTOLAKIS: So the criterion then for core damage is not core uncovery?

MR. SCHULZ: That is correct.

MEMBER ROSEN: But there is still plant cooling going on, right?

MR. SCHULZ: Yes.

MEMBER ROSEN: And in that circumstance, when you have uncovered the top, there is steam cooling going on?

MR. SCHULZ: Yes.

CHAIRMAN APOSTOLAKIS: So what is the order of magnitude of the duration of the uncovery in order to see some problem? I mean, Terry mentioned that it is about 300 seconds in those other problems. If it was a thousand seconds, would that have a problem?

MR. SCHULZ: Three is two kinds of issues. One is that there are relationships between depth and timing. Obviously if you have a large LOCA and you completely uncover the core very early in the transient, things heat up rapidly.

If you only uncover a little bit of the core much later, things heat up very slowly. That is one issue. So you can calculate based on depth, timing, duration, what the peak clad temperatures are.

MEMBER RANSOM: But wouldn't it be a good idea to use COBRA-TRAC and see if it predicted any heat up?

MR. SCHULZ: Well, this is why we look at -- it is not a good idea to try to do that for 500 transients.

MEMBER RANSOM: Well, I know that, but --

MR. SCHULZ: And that is why we use MAAP for these hundreds of events, okay? We did do benchmarking against -- MAAP results against NOTRUMP, and using LOFTRAN to calculate peak clad temperatures for those same transients.

And to ensure that MAAP was reasonable/conservative relative to the design basis codes.

MEMBER SIEBER: You actually have to try out part of the core in order to get core damage, right, as long as you have vapors going through there?

MR. SCHULZ: I can't really answer that question. You may need more than just the vapor.

MEMBER SIEBER: Okay.

MR. SCHULZ: But again there is times and durations; timing after a shutdown, and depth and duration of uncovery, all relate to that.

MEMBER SIEBER: Okay.

MEMBER ROSEN: But then you said that you also went back with the low margin risk sequencing presumably with your better codes?

MR. SCHULZ: That's right, and I will be talking about that in the last part of my presentation.

CHAIRMAN APOSTOLAKIS: So which one is a better code?

MR. SCHULZ: For the small break LOCAS, we repeated the analysis with NOTRUMP, which is what we used in the design basis analysis for our justification, with then being successful.

This is the Category 2 o f these events, the se same as the previous one, except that instead of requiring ADS-4 and gravity injection, we are using a couple of twos and threes, and an RMS pump injection.

So this is a mixed slice of active system operation, and look at the same spectrum, depth duration, again is a little better than AP-600, and we think that this is successful. You see here that this is again a spectrum of breaks.

And for very little ones, we get a little bit of uncovery after ADS, and for the bigger breaks, the break plus this ADS, Stage 2 and 3 get the pressure down fast enough that RNS injection happens relatively quickly, and the core stays covered.

Now I would like to talk about the manual ADS cases. This is with one accumulator and no core makeup tanks. The previous cases were with the opposite.

We are requiring a passive RHR to be available to bide the operators time to 20 minutes at least to do the manual ADS. Again, we look at the same spectrum of break sizes, and we got as good or getter performance than AP-600.

MEMBER SHACK: Do you have some emergency operating procedure that tells --

MR. SCHULZ: Yes. To do what?

MEMBER SHACK: To manually blow the valve.

MR. SCHULZ: Yes. Yes, the way we end up evaluating operator actions is in accordance with our emergency procedures. The operators have to have procedures, and they have to have indications of instrumentation or whatever.

And then we use that to figure out how much time, and then based on that time, reliabilities and probabilities of the operators actually doing that in that time or calculating.

This is the spectrum of break analysis and what you tend to see here is that initially you get no uncovery, but afterwards, you tend to get some. And what happens in this case is the accumulators don't run very long because of their nature.

Core makeup tanks run like 20 minutes all the time, and accumulators, it is variable depending on how fast the pressure goes down. And so what you tend to see is gaps between the end of the accumulator injection and the beginning of IRWST injection, which results in some core uncovery.

The passive RHR operation is beneficial right in this area here. What is happening in these cases is the break big enough to start challenging core uncovery, but not big enough to get down to accumulator injection.

But with these bigger breaks the pressure comes down fairly rapidly just because of the break and accumulators start injecting. So you don't get an early core uncovery. You get more of a late core uncovery.

This is looking at the 3-1/2 inch break case, which is probably the most critical from a passive RHR operation and operator timing. And you can see that the AP-1000 with the passive RHR is considerably better than AP-600.

AP-600 we did not require the passive RHR to be available. So it was not in the success criteria, and so we didn't include int in this analysis.

If we had, it would have significantly improved this early, and this thing is due to the fact that you have no makeup from your core makeup tanks, and the break is not big enough to get you down to accumulator injection, and so you just sit there for 20 minutes or so with no injection.

Once ADS goes off here, then the accumulator injection -- this is an accumulator wire mass, and so the accumulator is not draining at all, and then once ADS goes off, it empties pretty quickly.

And then sometime a little later, the IRWST injections starts. So again the AP-1000 performance, we get no core uncovery early. We get a shorter core uncovery later.

MEMBER ROSEN: Now, this is an analysis artifact, this core uncovery early before 20 minutes, because in reality operators would have enough information would they not to manually initiate ADS?

MR. SCHULZ: They would. Okay.

MEMBER ROSEN: In other words, they would not let the core go uncovered like that. They would see this all happening, and they would have adequate time to say we are not going to let that happen.

MR. SCHULZ: Right.

MEMBER ROSEN: And they intervene and mess up your analysis in saving their plant.

MR. SCHULZ: Right, they would, but what we are doing is we are saying is that the operator could be delayed, or he could wait as long as 20 minutes and still be okay.

MEMBER ROSEN: That's what I m saying. It is an analysis artifact. We impose a restraint on the operator, who really isn't there, and who really would not be there.

MR. SCHULZ: Oh, we are not saying that the operator should wait. Certainly not.

MEMBER ROSEN: When you say core mixture level is that a collapse level, or --

MR. SCHULZ: It is a mixture level and not a collapse level.

MEMBER RANSOM: If you mean a mixture level and it actually declines much above the top of the core, then you do dry out presumably the upper part of the core.

MR. SCHULZ: Not with a mixture. There is still a mixture going through the core. So as long as the mixture level is above the top of the core, the core is not going to heat up.

MEMBER RANSOM: Is there a flow through the core?

MR. SCHULZ: Oh, sure, yes.

MEMBER RANSOM: What, the pumps are running?

MR. SCHULZ: No, you are venting out the break. There is not really significant flow. There is steam being generated, which is going up through the core.

MEMBER RANSOM: Steam cooling.

MR. SCHULZ: Well, not use steam cooling. The steam is carrying water with it, and so there is water also going.

MEMBER RANSOM: Well, you said the mixture level is down about six feet below the top of the core, and that would imply --

MR. SCHULZ: That is AP-600, first of all, and this is AP-1000.

MEMBER RANSOM: It would be a lot more meaningful to calculate core temperatures and then show those, and it would answer the question do you damage the core or not.

MR. SCHULZ: Yes, we could present that.

MR. CUMMINS: This is Ed Cummins again. I think that most people are skeptical of MAAP calculated core temperatures, and that is why we don't show them.

MR. SCHULZ: The fourth class is again the same as the last one, with one accumulator, one core makeup tank, but with pump injection, and no stage 4 and 2, stage 2 and 3.

MEMBER ROSEN: Is this still the same 2-1/2 inch break?

MR. SCHULZ: Well, we look at a spectrum in all four categories, from .5 up to 8, and in this case we get no core uncovery at any time for any of these breaks. So this is not so challenging with the RNS pumps.

So I would now like to move on to large break LOCA success criteria. For cold leg breaks, the success criteria is two accumulators, just like the design basis, the DCD analysis. So initially we actually didn't do a special PRA analysis for AP-1000.

But we eventually noticed that the success criteria also requires that we consider no containment isolation, which is a little more conservative, and would tend to increase PCP above the design basis, numbers which were already pretty high, 21 something degrees.

So we reanalyzed the event using COBRA-TRAC, which is our design basis code, and running it the same way because it was easier to do that, and we calculated an even lower temperature.

Now, the reason that went down was that we assumed the availability of off-site power for 10 seconds, which we thought from a probability point of view was justifiable. The chance of losing off-site power that quickly we were not worrying about.

MEMBER ROSEN: Well, I agree a hundred percent, but that is not the standard analysis. The standard analysis, you take off-site power off the instant of the break.

MR. SCHULZ: Right, which is appropriate DCD analysis. Now this is PRA analysis. So what I am saying is that we should use this in the DCD. I am just saying that for the PRA that we didn't make that super conservative assumption.

MEMBER ROSEN: Now, for the PRA, you could just leave off-site on, period, because there is almost no instances of SCRABS, for instance, or loss of an energy source from a plant causing an off-site power loss. I mean, it has happened, but not usually.

MR. SCHULZ: And all I am saying is that

for this analysis, it is only important as far as how it affects the large break LOCA until you trip your reactor coolant pumps, which we automatically do in an S signal, plus the time delays.

So it really only has to run like 10 or 15 seconds, for off-site power to be available, and after that, we could lose it and it won't affect this result.

CHAIRMAN APOSTOLAKIS: Now, what does without uncertainty mean?

MR. SCHULZ: When you look at the DCD, the methodology for large break LOCA includes a calculation of DCD, and then it separately accounts for plant uncertainties, and it adds up a number that is in the AP-1000 case something like 230 degrees, which would get added to this if you wanted to look at with uncertainty.

CHAIRMAN APOSTOLAKIS: So 1850?

MR. SCHULZ: Yeah, and so when you look at the T&H certainty evaluation that we did for large LOCA, we put that uncertainty, we added that on. But for the success criteria --

CHAIRMAN APOSTOLAKIS: Well, the 2200 degrees that is not a best estimate is it for the failure criteria?

MR. SCHULZ: You absolutely have to stay below that.

CHAIRMAN APOSTOLAKIS: But that is a regulatory requirement.

MR. SCHULZ: Yes.

CHAIRMAN APOSTOLAKIS: But in terms of uncertainty --

MEMBER ROSEN: If you did a realistic estimate, it would be more, but not a whole lot.

CHAIRMAN APOSTOLAKIS: Okay.

MR. SCHULZ: We also did a spurious ADS for large LOCA, where we opened all four stage four valves at the same time after the initiating event. We used one out of the accumulators, and we analyzed this with COBRA-TRAC, and we got a very low PCT, and hot leg breaks just tend to be a lot less severe than cold leg breaks.

You don't get that flow reversal and initial heat up, and the core cools down much better at the end of blow down, and so there is a lot more space and temperature to heat up before you get into trouble.

ATWS analysis. The first thing to thin about here is AP-1000 has what we call a low boron core, which means that the beginning of core life just after refueling the maximum boron concentration is probably a thousand PPM, instead of 11 or 1200 PPM.

This gives us a more negative moderator temperature coefficient, which makes it easier to ride out the transients. The current AP-1000 is able to ride out transients over about 98.5 percent of the core life, or the UET is 1.5 percent.

We have analyzed the two cases, and shown them in the PRA. One of them is the beginning of equilibrium core cycle, which has an MPC that is at least minus 12.5, and we also looked at the first core, which tends to have less negative MPCs, and about 40 percent of life, we have got about minus 10, and at this point we bump up against the pressure limit post-ATWS.

So I think these are the peak pressure transients, and this is the beginning of like equilibrium core cycle, which stays below 3000 psi. The first core cycle goes right up to 3200 psi, and this is actually psia and the limit is psig. So this is right at the limit.

We have some discussions with the staff going on whether 98.5 percent is enough, or whether we need 99 percent or something, and we are still talking to them about that.

Okay. Now I would like to get into the T&H uncertainty stuff. We have already talked a little bit about this, and hopefully the rest will paint a real clear and easily understood explanation. We have provided evaluations that are actually in a response to the RAI, where we went through and implemented a process like we did on the AP-600, which I am going to explain here.

The whole process is trying to calculate the high risk, low margin cases from a probability point of view, and we have used the MAAP success criteria analysis to pretty much tell us when we get core uncovery, and any time we get core uncovery, we are considering that to be a low margin case, no matter what the temperature is.

We take the event trees that Selim showed you that we did for the core melt level one analysis, and we expand them to include intermediate failure cases. Well, not failure, but success equipment availability cases.

And then we connect those expanded event tree branches to whether they are low margin or high margin success paths. In the end, we think we have bounded about 98 percent of the core melt sequences with the conservative T&H analysis we have done.

CHAIRMAN APOSTOLAKIS: Again, you say you have expanded the --

MR. SCHULZ: Let me show you. Just hang on.

CHAIRMAN APOSTOLAKIS: You have to explain what that means.

MR. SCHULZ: I was qualitatively doing that. We ended up from this identifying the limiting analysis cases, which were three small LOCAs, two large LOCAs, and two long term cooling cases, and if we analyzed these seven events with DCD codes and methods conservative with Appendix K --

CHAIRMAN APOSTOLAKIS: These are high risk and/or low margin?

MR. SCHULZ: That's right. That's right, and it showed successful core cooling for those cases.

CHAIRMAN APOSTOLAKIS: Okay.

MR. SCHULZ: We pretty much talked about this, and let me go on here to this, and hopefully this will help you. What you see on the left is a kind of event tree structure in the PRA, when you are just trying to figure out whether the core melts or it doesn't melt.

You are not trying to differentiate anything else. So what we do when we expand the event trees, is we take, for example, the core makeup tank, and instead of it being just zero or one, it could also have two tanks available, and so that is what we do here.

So we have zero, one or two, or two. So actually these three things, zero, one, or two. And the same for the accumulator, zero or one. Now, what we do is that we then look at the end points, and we try to figure out, well, is that like a design basis case?

Well, in this case it is design basis, and we have got two accumulators and two core makeup tanks, for a medium LOCA. That is what we would normally have for a design basis.

We also called this design basis in the sense that we have analyzed DVI line breaks with one core makeup tank, and one accumulator, because the other two spilled, and so we consider this to be design basis.

This case here has no accumulators, but two core makeup tanks. We have put this into these categories that are UC are like uncovery. They are low margin.

So the okay ones are high margin in our terminology, and things where we put UC something is a low margin case. And it can have two core makeup tanks, or if we take this one and expand it just like this, and so this tree really grows.

CHAIRMAN APOSTOLAKIS: So UC-3 does not exist on the left.

MR. SCHULZ: That's right. It is a subset of one of these, and you can't figure out the probability of UC-3 here, because this one only takes the extreme failure conditions.

CHAIRMAN APOSTOLAKIS: So the logic, and again at the high level, is that we are getting into a little bit of trouble by going with the minimum from a success criteria point of view. So let's look at the actual case where I need only one CMT, but I really have two.

So there are some cases perhaps where I will get both of them?

MR. SCHULZ: That's right. And when we do expand these trees, we go through and calculate the probabilities of all of these different branches.

CHAIRMAN APOSTOLAKIS: Which again is an expansion of the probability that you have on the left.

MR. SCHULZ: That's right.

CHAIRMAN APOSTOLAKIS: And what does that do now? Is that a bounding case?

MR. SCHULZ: It provides you more detail.

CHAIRMAN APOSTOLAKIS: More detail.

MR. SCHULZ: In terms of the probability, and what we ultimately want to do is figure out what is the highest probability of getting these UC things. These we really don't care about so much, because we are saying there is T&H uncertainty really with these guys. We are not getting core uncovery, and we have lots of margin for cooling.

CHAIRMAN APOSTOLAKIS: But all you are doing -- if I expand the middle one there -- yes, that one, and if I expand that one, I will end up with --

MR. SCHULZ: You will end up with three more branches like this.

CHAIRMAN APOSTOLAKIS: Exactly, and one of the sequences will be what I have on the left won't they?

MR. SCHULZ: Yes. In fact, it will be --

CHAIRMAN APOSTOLAKIS: But what happens now is the probability is lower?

MR. CUMMINS: The whole objective of this is to find out which of the uncovery cases have some impact on the PRA.

MR. SCHULZ: Yes, I understand that.

MR. CUMMINS: So you are looking for the ones that are risk important. We are going to find a whole bunch of uncovery cases, some of which have some PRA value, and some of which don't, and we are going to throw away the ones that don't.

MR. SCHULZ: Let me continue here a little bit. I think it will become clearer. This is just a listing of how we group the different okays and these UC categories with sort of different kinds of equipment being available.

CHAIRMAN APOSTOLAKIS: So can you point here to the sequences that correspond to the ones that you had on the left in the normal case in the slide before?

MR. SCHULZ: Oh, the normal case?

CHAIRMAN APOSTOLAKIS: The way that you do the standard PRA.

MR. SCHULZ: Well, the standard PRAs don't relate to these. They are just okay, period. They are all mushed together. We don't differentiate. The success paths intend to be extreme, in terms of that they have multiple failures in them, and you can't differentiate this, and you can't get this detail out of the PRA level one event trees. They are not that detailed.

In the expanded event tree, we have used all these detailed branches to differentiate between uncoveries, and --

CHAIRMAN APOSTOLAKIS: Let's go got he previous slide, and maybe that will help. You are doing -- and you call it normal, too, your normal event tree. And you have core melt when you have no core makeup tanks and no accumulators, right?

MR. SCHULZ: That's right.

CHAIRMAN APOSTOLAKIS: And you have core melt because you have uncovered the core and for a period there is no high pressure injection?

MR. SCHULZ: Right.

CHAIRMAN APOSTOLAKIS: Now, when I expand the tree, what happens to that sequence, the 00 sequence?

MR. SCHULZ: The 00 sequence will be a core melt still.

CHAIRMAN APOSTOLAKIS: It will still be there?

MR. SCHULZ: It will still be there.

CHAIRMAN APOSTOLAKIS: Have I bounded it in any way?

MR. SCHULZ: What do you mean by bounded?

CHAIRMAN APOSTOLAKIS: Well, I mean I have a frequency here--

MR. SCHULZ: We are not trying --

CHAIRMAN APOSTOLAKIS: You are doing it to the others, to the successes.

MR. SCHULZ: It is the successes.

CHAIRMAN APOSTOLAKIS: You are doing it to the successes.

MR. SCHULZ: Yes.

CHAIRMAN APOSTOLAKIS: So now I take the first success from the bottom, where I don't have a CMT, but I have one accumulator.

MR. SCHULZ: Yes.

CHAIRMAN APOSTOLAKIS: And somebody says, well, how do you know the accumulator is good enough and so on, and that is what you are addressing now?

MR. SCHULZ: Eventually. Right now I am trying to calculate probabilities of these intermediate states, and then I am trying to figure out --

CHAIRMAN APOSTOLAKIS: But you will still have a sequence on the right that says no CMT and one accumulator?

MR. SCHULZ: That's right. It will be here and have a certain probability.

CHAIRMAN APOSTOLAKIS: So that is what

bothers us now and we have to try to find out what to do with it.

MR. SCHULZ: Yes. The next question. So we expanded eight event trees in AP-600, and we didn't take all 26, okay? We looked at ones with ADS actuation, and not the ones without ADS actuation. Now for AP-1000, we expanded five trees.

Now we lost the intermediate LOCA because it does not exist in AP-1000, and we added the spurious ADS, because that did not exist on AP-600. But we didn't do the small LOCA transients with ADS to rupture with ADS that we did do in AP-600.

And the reason for that is that these three events, expanded event trees, did not produce any limiting risk important cases. They all came out of the other events, and generally what happens is that these events result in later ADS actuations, so that the timing of uncovery is later, and it is delayed. So it tends to be less severe.

So we looked at five event trees that we expanded, and this is just a summary of that, and what we did in AP-600 and what we did in AP-1000, and as an example, this is a DVI LOCA, and you actually are seeing half of it here. The other half is on the next page.

And so you see the full thing here, and you were asking about, for example, one -- well, one of the characteristics of a DVI LOCA is that you lose half of the systems.

So you don't see two core makeup tanks or two accumulators anywhere here.

CHAIRMAN APOSTOLAKIS: Zero, one.

MR. SCHULZ: Zero, one, and so that looks a bit more like the normal event tree. However, in ADS land, you see a lot of intermediate states. And then we go over and we plug in what these end-states are; okays, okays, and there is a core damage, and there you start seeing some uncoveries, and uncoveries.

Now, all of these events here are with containment isolation, which is the first question on the tree. The next page is without containment isolation, and the same story. So after we set this tree up, we calculate it and then we sum up the potential core damage events that were treated as success in the base PRA.

So these are all the UC, these low margin, coolant recovery things. If you calculate all of those, and we don't worry about core damage.

CHAIRMAN APOSTOLAKIS: So sequence number six, is a sequence of appearance of the normal event tree?

MR. SCHULZ: No. It would be a subset of one of the ones. It is covered by and bounded by the normal base line PRA.

MEMBER ROSEN: I would say included in.

MR. SCHULZ: Included in. It is included in there, but it is a subset of one of the branches.

CHAIRMAN APOSTOLAKIS: Success branches.

MR. SCHULZ: Success branches, yes. So we end up calculating all these intermediate success states, and we move them all into a big table, and we sort them, and figure out which are the most probable ones, to try to figure out this is the bottom half of that same tree.

Now, where do we draw the line? Which ones are -- you know, we have this big table from higher probability to very, very low probability situations. So we -- okay, this is still before that.

When we talk about large release, we didn't really calculate it like we do in the base PRA. We used a constant 6 percent of the core damage events, and this is with containment isolation now, and we go to large release, and the same thing that we did with AP-600. Here we talk about the criteria.

So we basically take any of these potential core damage events which were a success on the baseline PRA, and we say that all of those that are within one percent of the total core damage frequency for AP-1000, we will consider to be risk important.

MEMBER ROSEN: Give me that again. Within one percent?

MR. SCHULZ: Yes.

MEMBER ROSEN: Meaning?

MR. SCHULZ: That they are greater than or equal to one percent of --

MEMBER ROSEN: Of 2.4 E to the minus 7.

MR. SCHULZ: Yes.

MEMBER ROSEN: In other words, anything greater than 2.4 E to the minus 9?

MR. SCHULZ: Yes. We will consider those to be low margin, because all of these are low margin, risk important sequences, and we should consider them in the T&H uncertainty.

CHAIRMAN APOSTOLAKIS: Risk important?

MR. SCHULZ: They will be risk important - -

CHAIRMAN APOSTOLAKIS: I thought these were successes?

MR. SCHULZ: They are successes in the base PRA, but there is a question about --

CHAIRMAN APOSTOLAKIS: But here they are successes.

MR. CUMMINS: Excuse me, but the question is a MAAP success a real success, and our answer is, well, I don't know. We will have to prove it with our DCD code. Well, rather than do this a hundred times, we are trying to figure out a way to do it 5 or 6 times, and so we are going to explain how we pick the 5 or 6 winners out of the hundred in order to run your DCD code and prove that MAAP predicted correctly.

CHAIRMAN APOSTOLAKIS: You do that later, but at this stage --

MEMBER SHACK: He has first got all the ones with uncovered, and so they are by definition low margin. How he is sort of looking at the probability that he will actually get one of those, and he is going to pick the most frequent ones of those, and so those become his dominant sequences.

MR. SCHULZ: And some of those sequences are 3 or 4 orders of magnitude less than the core melt frequency, and so --

MEMBER ROSEN: But the dominant sequences, I am sure that you are confusing George. When you said that, he went into outer hyper drive. This is simply a technique for Westinghouse to be able to pick important sequences, even though they were successes, to do some detailed calculations to show that the temperatures with steam cooling do not exceed or do not cause core damage.

CHAIRMAN APOSTOLAKIS: So a success means that you may have some uncovery for a while, but the temperature --

MEMBER ROSEN: The temperature stays low enough that the uncovered portion of the core, that the fuel, although it gets hotter than you would like it to, it never gets so hot that it is damaged.

CHAIRMAN APOSTOLAKIS: Okay. And then you are looking at those, and you have their frequency occurrences.

MEMBER ROSEN: Right.

CHAIRMAN APOSTOLAKIS: This frequency is not part of the base line DCD.

MEMBER ROSEN: No, because these are successes.

CHAIRMAN APOSTOLAKIS: but now you are saying that I arbitrarily will consider those success sequences that have a frequency and look at all of them and decide whether I should move them down to failure.

MR. SCHULZ: No, what we are considering is the potential core damage, and we are going to look at them very closely from a T&H point of view.

CHAIRMAN APOSTOLAKIS: So a very negative review and so you are going to look at it?

MR. SCHULZ: Yes.

CHAIRMAN APOSTOLAKIS: And to convince yourself if it is a success?

MR. SCHULZ: Right. This is one page of about four of the total sequences that come out of expanded event threes, and you can see for each of them the sequence CDF.

Now, this is a potential, and these were all success in the base PRA. So this is potential. So obviously this is a 10 to the minus 7 kind of sequence. So that is more than a core damage.

So the ones that are boxed in here are ones that meet the one percent criteria. So you see that you are starting to get down below 2 times 10 to the minus 9 here.

And we looked at large release as well against core damage, and we picked up a few large releases down here. Here you can see what kind of failures went along with these sequences, just for your information.

In addition, now there is 13 of these cases, and there is none on the lower pages, and so you see all of the risk important low margin sequences here, 13 of them.

MEMBER ROSEN: Because you sorted it by --

MR. SCHULZ: Right, they get lower, and lower, and lower as you go down.

MEMBER ROSEN: -- the most important.

MR. SCHULZ: That's right.

CHAIRMAN APOSTOLAKIS: Okay.

MR. SCHULZ: Now you also see on the right, and I am getting a little bit ahead of myself here, is that we selected seven cases to analyze; five of them short term, and two of them long term cooling cases.

And you see here two columns; short term, long term, cooling. And these letters relate to one of the cases that we did analyze. So we think that we have analyzed with these seven cases more than -- and you see these cases here, and these two cases, for example, are not. They are 10 to the minus 9, and 10 to the minus 11 cases.

And it happens that in order to or instead of analyzing 13 cases, we smooshed them into 7 cases, and because of that, we ended up with a little bit of conservatism, which then covers a few more cases. There are 102 cases here total, and 13 of which are risk important, are cases bounded by 56 of the 102, which ends up being 98 percent or so of the risk of the plant, are bounded by these conservative T&H analysis cases.

Now, let me show you which cases those are. This is the 13 pulled out of that big table just to summarize for you how much they would contribute to core melt and large release if they were core damage, and obviously you don't want that to happen.

It also shows you what we call the residue, and if you take all of the cases that aren't in these 13, how much does that add up to be compared to these cases.

So these cases add up to be 10 to the minus 6, and these other cases add up to be 10 to the minus 8. So they are small relative to the total. So we ignored those other cases, although again we covered many of them off.

Here are the seven cases that we picked for candidates for the detailed T&H analysis. Three of them are small LOCAs, and two large LOCAs, and short term, and then two long term analysis.

And you can see here which equipment availability we selected, and this indicates which of the dominant cases are bounded by them. So for these first two, and for example, no core makeup tanks and accumulators, one of them actually has two accumulators. They both have four stage fours.

MEMBER ROSEN: That means four fails stage fours.

MR. SCHULZ: No, four working.

MEMBER ROSEN: Oh, four working stage fours?

MR. SCHULZ: That's right. All of these cases rely on passive systems only. We did not include in the expansion of threes any active systems because the issue of T&H uncertainty seems to be focused on passive system performance, and this whole issue of low Dts, and uncertainty, and newness of passive systems, and so again, just like AP-600, we did not expand active system branches, only passive system branches.

So all of the success criteria here and equipment availability is passive system.

MEMBER ROSEN: Yes, but what does this table mean now? It says CMT, zero.

MR. SCHULZ: That is available. Those

CMTs are available, and one accumulator is available, and this is available equipment.

MEMBER ROSEN: Available equipment. Okay.

CHAIRMAN APOSTOLAKIS: You keep talking to risk guys. So I am dying to go to the meat of it.

MEMBER ROSEN: He is preparing us.

MR. SCHULZ: This is very similar to the previous page, and it shows you the code that we used to analyze each of the cases, and as I said, we used NOTRUMP for the small break COBRA-TRAC for the large breaks, and the COBRA-TRAC long term cooling model for the long term cooling. These codes were run like they were in the DCV analysis.

CHAIRMAN APOSTOLAKIS: This is now considered what, a conservative analysis?

MR. SCHULZ: Yes. Appendix K, decayed heat, and limiting plant parameters and limiting --

CHAIRMAN APOSTOLAKIS: And the argument that you are making is that if I show that even with these conservative analyses, this is a success, that I don't have to worry about Dr. Ransom's concern about the uncertainties? That is the essence of your argument.

MR. SCHULZ: It is bounds of uncertainties.

CHAIRMAN APOSTOLAKIS: That is the essence?

MR. SCHULZ: Yes.

MEMBER RANSOM: Going to an Appendix K type approach.

MR. SCHULZ: Yes.

CHAIRMAN APOSTOLAKIS: Which is admittedly conservative though.

MR. SCHULZ: Yes, it says that they are not so important.

MEMBER ROSEN: And this covers most of the risk of the plant. okay.

MR. SCHULZ: So you can see from this that A and B get no core uncovery, even with these conservative analysis. C does get core uncovery, and the PCT is like 1500 degrees or 1600 degrees. Large break LOCAs, and I have actually talked about these, but these are with the DCD uncertainties.

So that if large break LOCAS were done not Appendix K, but the best estimate, DCD type analysis with separately calculated uncertainties.

CHAIRMAN APOSTOLAKIS: Let me understand the first two. You are saying no core uncovery.

MR. SCHULZ: Yes.

CHAIRMAN APOSTOLAKIS: What did you have

originally with NOTRUMP?

MR. SCHULZ: Well, this is with NOTRUMP. With MAAP?

CHAIRMAN APOSTOLAKIS: With MAAP.

MR. SCHULZ: Well, what I showed you was more limiting cases. The cases that I showed you would be, for example, disbursement would be no containment isolation, and this would be the same, and this would be the same, and but it will be 3 ADS. So I didn't show you one of these cases.

CHAIRMAN APOSTOLAKIS: You didn't?

MR. SCHULZ: I mean, we typically didn't analyze such cases. In our MAAP analysis, we were looking for the limiting cases. So we didn't analyze cases which had more things working.

Now, we did that on AP-600 just to make sure that more things didn't make things worse, and it doesn't. So when we did AP-1000, we didn't look at more things with MAAP, because we were focusing on the limiting success rates area.

MEMBER SHACK: This is one of the sorted sequences, which means that MAAP's end state was uncovered, right?

MR. SCHULZ: That we would say that it was either uncovery, or potential uncovered, because we didn't necessarily analyze with MAAP some of these lesser sequences for AP-1000. The next three slides show you the results of the events A, B, and C, or cases A, B, and C, showing the RCS pressure, core mixture level, and then you see here that we just barely dip to the top of the core.

The accumulator, which there are no core makeup tanks, is supposed to inject both -- just before ADS was off for 20 minutes, and then injects very rapidly after that until it empties. Then IRWST starts up some little time after the accumulator is empty.

But the core mixture level is popped back up again, and doesn't dip below the top of the fuel throughout that. So again NOTRUMP, Appendix K, analysis.

CHAIRMAN APOSTOLAKIS: All of the sequences that you analyzed, did you declare them a success or did you find some problems?

MR. SCHULZ: Yes. In some earlier cases where we hadn't, for example, put the passive RHR in, when we first started trying to do this, and it didn't work. So then we backtracked and changed the success criteria so that it would come out to be successful.

And in all seven cases that we have now

performed are all successful, with the seven cases that we have analyzed that cover 98 percent of the risk.

CHAIRMAN APOSTOLAKIS: So this kind of analysis made you change some success criteria?

MR. SCHULZ: Yes.

MEMBER ROSEN: And require passive RHR?

MR. SCHULZ: Yes, that was the only real change that came out of this, but it did. This is Case B, which is a CMT line break case, two accumulators, no core makeup tanks, 4 out of 4 ADS with containment isolation being effective.

And everything is very good on this case, and not that challenging. In this case we do get core uncovery, and this is a DVI LOCA, one core makeup tank, and no passive RHR. 3 out of 4 ADS, no containment isolation.

So we get near the top of the core here, and then as the core makeup tank empties about in this time frame here, then we don't get injection from the IRWST 4 sometimes, and so we deplete the inventory from the reactor, and then we start getting injection, and we get some recovery here.

And we analyze the peak clad temperature for this and it is 1570 degrees. So again we said that is okay from a core damage point of view.

MEMBER LEITCH: I'm not sure I am reading that right. Is that minus 18 feet, or is it minus 4 feet?

MR. SCHULZ: Well, the mixture level is on an absolute scale and the top of the core is 20.5 or something feet.

MEMBER LEITCH: Okay. I see.

MEMBER ROSEN: This is a revelation.

MR. SCHULZ: It is about two feet.

MEMBER ROSEN: Maybe the light is beginning to dawn on me, and maybe the for the old guys who run BWRs. We have always thought of containment as a good thing to protect the public's health and safety, in the sense that if you had an accident that stuff doesn't get out and get to a potential member of the public.

Here it does that function, too, but it is much more important because it makes these, and without the ECCS may not work in certain cases. So that is another whole deal that is new in the sense of these passive plants. Now maybe some BWRs need to back pressure to have enough MPSH. They need some credit for it.

But this is the clearest demonstration of what happens when you don't have containment isolation, and in this case, you are going to have a whole lot more core damaging events than if you did or you would have in the other kinds of plants, and which don't rely so heavily on containment to provide the back pressures.

MEMBER SIEBER: There is a number of current generation BWRs that need some containment pressure needed to take credit to get MPSH adequate for --

MEMBER ROSEN: Yes, mother nature was telling us that there is some other function for containment other than directly protecting the public's health and safety, because it does show up in some BWRs, and in some PWRs. But here it is much clearer. Just an observation.

MEMBER SIEBER: You could accomplish the same thing without containment and not that you have it, you can use it. Otherwise, the plant just gets taller and taller.

MR. SCHULZ: I am not going to show you the large break cases. I have already really talked about them, but what I would like to do now is talk about the long term cooling case, and the one that is the most interesting there is the one with the failed containment isolation.

And so what we have done is an analysis that looks at the largest single penetration, which is an 18 inch H back line staying open indefinitely. We assumed the BWR LOCA, which is in our opinion the most limiting LOCA because it results in a lower initial water level and containment.

That X is about two feet, and I forgot to write that down, but what that means is that if you had a non-DVI LOCA, including any large LOCA, the initial containment water level would be two feet higher.

So you would have a lot more inventory that you could lose out the break, out the hole in the containment, before you would challenge core cooling and a recirc long term mode. So that waw the events that we looked at.

And what you will see following here is some analysis that shows hat with passive containment cooling operating, with the water cooling going on, that the containment leakage is terminated in about 2-1/2 hours.

For that 2-1/2 hours, you have leakage going out of the containment. After that 3-1/2 hours, there is essentially no more leakage.

MEMBER ROSEN: And no more driving head?

MR. SCHULZ: Right. And what has happened is that a accumulation of decayed heat has dropped and the PCS performance improved, and the reason why it has improved is that during the leakage, the leakage has taken air, as well as steam, out of the containment.

And taking the air out of the containment increases the partial pressure of steam, and increases the temperature of the mixture in containment at these low pressures. And allows for better heat transfer through the containment.

And as a result, you end up with PCS performance going up, and decayed heat coming down, and about 2.8 hours out, you end up terminating the leak out of containment. During that time, you lose about .3 feet of level in the containment, which is not very much.

And then we did a COBRA-TRAC analysis to show that with this reduced level and atmospheric pressure that we are still okay. This shows you what is going on in containment in this event. The IRWST level is dropping as it injects, and in fact spills.

The PXS-B is the room where the PXS valves are located and where the break is located, and so that is a separate volume in containment and it behaves differently from the bulk of the containment. There is a drain line out of the bottom, but it tends to get overwhelmed by the blow down from the break.

So it tends to fill up faster as you see than the containment, which is this solid line, is the main containment. Then eventually the main containment becomes the highest level and it is driving the recirculation flow back through and into the reactor coolant system.

You see down here the containment leakage, and it is higher early, and then in about 10,000 seconds or 2.8 hours, it drops to about zero. Containment pressure goes up to about 10 psig for something, and then it drops to atmospheric pressure in that same time period.

This code here is a little confusing, in that it shows the decayed heat level on the dotted line which seems to be above the PCS, and that is above the PCS heat removal, and so you are saying why is it matching decayed heat.

Well, the PCS heat removal is what is actually going through the shell and it doesn't count other places that heat can go. So if you look at this whole time frame, the water going into the reactor is somewhat subcooled, and it is fairly highly subcooled in this early time frame.

And even out here it is still somewhat subcooled. So the PCS doesn't see that. It just sees how much heat is going out from steam generation basically that is going out through the wall in the containment.

It also doesn't see how much heat is going into concrete or steel inside a containment. Now, you see in the end here that things are coming together as the subcooling goes away and as the passive heat sinks and saturates.

Okay. This is just a summary of the T&H uncertainty analysis. We had calculated the probability of the low margin sequences, and the selected risk important low margin sequences, the important ones.

And the defined seven bounding cases, and five short and two long term. And we analyzed all those cases using DCD codes and methods, and for all of them have shown successful core cooling.

And that by doing that, we have bounded 98 or 99 percent of the risk of the plant with those conservative analysis. Any questions? No?

CHAIRMAN APOSTOLAKIS: Very good. Thank

you very much. You finished early. Okay. We are going to break for 20 minutes. We will be back at 3:22, which is a mean value and Selim will tell us what the high bound will be.

(Whereupon, at 3:03 p.m,, the meeting was recessed and resumed at 3:27 p.m.)

CHAIRMAN APOSTOLAKIS: All right. Now we will hear from the NRC staff, Mr. Saltos.

MR. BURKHARDT: Yes, and before Nick gets started, Dr. Apostolakis, I would like to make a few comments. I am Larry Burkhardt, the NRR AP-1000 project manager.

As Mike stated earlier in his opening comments, we obviously do have an established schedule, and our next milestone is to issue the draft safety evaluation report in June of this year. So as you can imagine, we are in the midst of our review looking at the RAIs and all the other material that Westinghouse submitted.

And consequently what you are going to hear here is not final, but we would like to give you a snapshot of where we are in our review. So with that said, this afternoon you will be hearing from Nick Saltos on the level one PRA, and Walt Jensen on PRA success criteria, and Marie Pohida on the shutdown

PRA.

And one more comment. There are three different groups of slides, copies of slides, going around. So I hope that everybody has a copy. With that said, I will turn it over to Nick Saltos to about the level one PRA review.

MR. SALTOS: Good afternoon. This is Nick Saltos from the NRR, the Probabilistic Safety Assessment Branch, and I am going to talk about major objectives in the process of the PRA review, and also talk about the major issues of the level one PRA review.

The major objectives of the PRA review are to ensure the quality of the PRA, and commensurate with its intended use, such as gaining insights about the design, and support the design certification processes.

MEMBER KRESS: You know, if Dr. Wallis was sitting here, which he isn't, he would ask you two questions I'm sure. The first one would be how do you measure the quality of the PRA, and the second one he would ask is how do you know when the quality is commensurate with its intended use? Have you got some gauges or criteria that --

MR. SALTOS: Yes, we have some generic

means I would say to do that. By evaluating the models and assumptions, and data, used in the PRA and comparing with other PRAs.

MEMBER KRESS: But in terms of the ASME quality standards would you call it a 2, or a 3, or a 1, or what?

MR. SALTOS: Yes. I see that there is compatibility there, but this work is based on the AP-600.

MEMBER KRESS: So that was before we thought about that.

MR. SALTOS: But I don't see that there is a conflict there with those criteria. The emphasis of course is on PRA modeling of novel features, like passive systems and the ITAAC. and (inaudible) for major contributors to risk, and features that contribute to reduce risk with respect to operating the reactors.

And areas of uncertainty that have to be addressed, and defense in depth to mitigate specific initiating events. Support the design and most of the PRA support of the design is done at the pre-application stage, but still we have to ensure that the PRA is valid to do that.

At that stage the PRA was used to define capabilities, and to introduce features to reduce or eliminate vulnerabilities. Quantify the effect in terms of risk of the new design features, and select a manual alternative features operating strategies, and design options.

And the use of the PRA design certification process, and then we go to the second bullet, and this is a major objective of the PRA, and a proper interpretation and use of the results for decision making in the certification process, such as identified design and/or operational changes to address weaknesses, and identify certification requirements, such as ITAACS, which stands for inspections test analysis and acceptance criteria.

And these requirements will be the ones that will be used to ensure that any future planned reference in the AP-1000 design will be operated in a manner that is consistent with important PRA assumptions.

Another area that the PRA is used in the certification process is to determine the appropriate regulatorial oversight for non-safety systems, and what Westinghouse calls defense in depth, and systems that are not safety related, like the normal RHR start up flood water system.

And first of all, determine if oversight is needed, and if it is needed, what system is better to have in terms of risk reduction to have this oversight, and what is the appropriate level of oversight.

And it is used also to determine the significance. PRA results are used to determine the risk significance of raised uses, and the focus of the most important uses, and the use of less important issues.

CHAIRMAN APOSTOLAKIS: Maybe we can skip to the next slide.

MR. SALTOS: Okay. The major issues from the review of the PRA level one power operation is the thermal-hydraulic uncertainties and success criteria, and Westinghouse talked extensively before. Another reason is the fire induced --

CHAIRMAN APOSTOLAKIS: Let me understand this. It is a major issue because you have reviewed what they have done and you don't agree?

MR. SALTOS: Well, we have not reviewed Westinghouse's response extensively yet. We are still reviewing those forms. But we had a request for additional information on this issue when we received their submittal to the PRA.

So this is one of the major issues that has to come to a close, because that impacts success criteria, and it can impact the risk and impact the requirements for the certification requirements, like risk and ITAACS.

CHAIRMAN APOSTOLAKIS: And what was the issue?

MR. SALTOS: I will talk next about that. Another issue is fire-induced spurious actuation of ADS squib valves, and another issue is that the identification of certification requirements, such as ITAACs and RTNSS, that result from major differences and design differences with respect to AP-600, because our list of AP-600 certification requirements that forms the starting point.

However, some certification requirements could change according to the resolution of some of the outstanding issues.

CHAIRMAN APOSTOLAKIS: I think this is what Mr. Schulz just described to us, right?

MR. SALTOS: Yes, more or less, but there might be some additional clarification from our point of view if you are interested in hearing that. When we start with this issue, we are talking about passive systems that rely on small driving forces, such as gravity, to perform primary functions. Such driving forces are small in comparison to those with pump systems that we use in the care and operation of power plants.

The uncertainty now in the valves of such driving forces as compared to a best estimate computer code, thermal-hydraulic analysis, can be of a comparable magnitude to the predicted values themselves.

So when the thermal-hydraulic uncertainties are concerned, some success accident sequences may actually not be a success and lead to core damage. So it would be converted from success to core damage.

CHAIRMAN APOSTOLAKIS: Could you be a little more specific? What kind of uncertainties?

MR. SALTOS: We are talking about decayed heat, for example. That has a mean aloe, and if the decayed heat is higher than what is assumed in the best estimate that could make a big difference in the thermal hydraulic analysis results about reaching the core uncovery and in terms of 2200 degrees.

CHAIRMAN APOSTOLAKIS: And it is not related to what you say there, passive systems rely on small driving forces?

MR. SALTOS: Yes, they are talking about all the thermal hydraulic parameters in the plant, and parameters that go into the thermal hydraulic analysis.

So for some accident sequences with frequency are high enough to impact the results, but which are not predicted by best estimate thermal hydraulic analysis code to result in failure, in core damage, may actually lead to core damage where these thermal hydraulic uncertainties are considered.

MEMBER LEITCH: Nicholas, presumably this is an issue that has been raised in RAIs and responded to, and --

MR. SALTOS: This is a different issue. I am going to have in my next slide and say what exactly it is.

MEMBER LEITCH: The current status of this, okay.

MR. SALTOS: Okay. This issue was addressed in the AP-600 PRA by the risk-based bounding approach, which Westinghouse described also, which uses conservative assumptions for key thermal hydraulic paraments.

It involves the identification of lower thermal-hydraulic margins, risk significant accident scenarios. When we talk about risk significant, we do not mean that they are risk dominant or risk important means.

We do not want them to cause core damage because if we (inaudible), then the results would be impacted, and therefore the inside would be impacted. In that sense, we will call them risk significant.

So this process involved the identification of low thermal hydraulic margins risk significant accident scenarios, and then the use of design basis accident computer codes like NOTRAM for small LOCAs, for example, to bound the thermal hydraulic uncertainty.

Such an approach relates to the impact of the thermal hydraulic uncertainties, to changes in the success criteria. The success criteria become or demand more equipment to be available, and therefore the risk would also change.

And when Westinghouse admitted the PRA, they told us that no sequences beyond -- there were not sequences beyond those that are defined in the AP-600, are classified as low thermal hydraulic margin risk significant on the grounds that the two designs are similar.

And the staff requested the use of a

systematic approach and/or additional analyses, as was done for AP-600, to support this argument. And Westinghouse submitted this approach that was presented before about blowing out the event trees, and basically what they do is what we consider as success sequences.

Every success sequence can be a success having one accumulator, or two accumulators, or one CMT, or two CMTs, or taking credit for a passive RHR, or not taking credit for a massive RHR based on the best estimate of thermal hydraulic codes.

Now what they did is that looking at some minimum availability system sequences. For example, one accumulator or no accumulators, and that is one key to success, and they do those calculations with a more conservative design basis accident analysis code, and this bounds (inaudible) flow rates, and (inaudible) and other initial parameters.

CHAIRMAN APOSTOLAKIS: So when you say the staff requested the use of a systematic approach, is that go beyond what was just presented to us, or is that --

MR. SALTOS: With that system analysis.

CHAIRMAN APOSTOLAKIS: So what you are not asking for is additional analysis.

MR. SALTOS: Well, no, we are asking for the systematic approach, and we review that, and if we agree with that, we are not going to ask for anything else. But we are still reviewing that.

CHAIRMAN APOSTOLAKIS: But when you say additional, it is not additional to what was presented to us.

MR. SALTOS: No. This RAI went to them before.

CHAIRMAN APOSTOLAKIS: Well, that should be clarified. Okay.

MR. SALTOS: The staff believed at the time that the difference in the thermal hydraulic parameters, et cetera, can affect plant response for PRA scenarios involving multiple failures, and potential system interactions.

And in addition, whenever the PRA changes for examining event frequencies and success criteria couldn't have changed the risk significance of the sequence. It would have changed the frequency that they calculated to determine if the sequence was risk significant or not.

And Westinghouse submitted a systemic approach that we requested and it is under staff review. Another issue is that in the AP-600 PRA at-power fire CDF is the dominant contributor to the at-power fire CDF by fire-induced spurious actuation of ADS explosive valves, which lead to a medium LOCA.

And 85 percent of the CDF comes from spurious actuation of ADS explosive valves. In AP-600, the significant uncertainty in hot short probability was addressed by sensitivity studies and design certification requirements.

And what the requirements are that are shown below are use controller circuit requiring multiple shorts of actuation; and routing ADS cables in low voltage cable trays and using redundant series controllers located in separate cabinets.

And provisions for operator action to remove power from the fire zone. This would have the degree of probability of having multiple shorts and therefore have spurious ADS squib valve actuation.

What was not considered then was that one hot short may not always be independent events, and that cable-to-cable interactions cannot be excluded. In the AP-600 certification, it was assumed that this hot shorts in two different cables wold be independent and would not cause the other.

However, the staff since the AP-600 certification, have conducted studies in SANDIA and

EPRI, which indicate that spurious actuations from cable-to-cable interactions, conductors from separate cables could come into close proximity to each other, are credible and likely for some cable types.

So the NRC asked Westinghouse and is working with Westinghouse on that, to see if the ADS cables are routed in the same cable tray, or a common enclosure, and analyze the effect of cable-to-cable interactions, and/or assess the need for additional design features, beyond AP-600, to prevent fire-induced detonation of explosive valves.

And the staff is interacting with Westinghouse to resolve this is.

MEMBER ROSEN: Now why if this is an issue on AP-1000 is it not an issue on AP-600?

MR. SALTOS: Because at the time we did not have those studies from SANDIA.

MEMBER ROSEN: I understand that, but --

MR. SALTOS: Well, I think that is it. More information since then.

MEMBER ROSEN: Well, now that you have the information isn't there some way to reflect it in AP-600?

MR. SALTOS: If we find out that this is important, we should.

MEMBER ROSEN: Yes, it seems like. The staff has to make some sort of special findings, I think.

MR. BURKHARDT: It potentially could be any number of issues that could cause us to revisit the design that is already certified, including the AP-600. One of the things that I am sure that we would do is assess the safety significance of that issue, and the likelihood of someone actually referencing a design.

I mean, the practicality, we have to deal with the human resource issue about these evaluations, and again consistent with this risk significance of the issue, we would deal with that. Another way to do that is just as you referred to.

MR. SALTOS: We might have some additional requirements about routing of cables, for example.,

MEMBER ROSEN: Well, since AP-600 and AP-1000 are not plant sized and built, if it is a backfit, it is a backfit of a design, and not of a facility that is out there operating.

MEMBER LEITCH: This fire induced operation is assumed to occur on one ADS valve?

MR. SALTOS: Well, if one ADS valve opens, you have a medium LOCA. If more than one, you have a large LOCA. But it is less likely to happen in two than one. The concern is for one based on frequency.

MEMBER LEITCH: Is there not a location such that a fire could cause all four valves to open?

MR. SALTOS: For the AP-600, based on (inaudible) cable interaction, or in other words, one short per cable causes a short in the next cable, which would be a multiple hot short, and would spuriously open the valve.

MEMBER LEITCH: But isn't there some point back in the circuit where there is a common signal?

MR. SALTOS: Well, that is why we have these requirements that I talked about here, that they are trying to prevent that. If the cables are routed that way, and the plant is built according to these requirements, that would not be very likely.

CHAIRMAN APOSTOLAKIS: So you can't have a hot short or a series of hot shorts that create a large LOCA. Is that what you are saying, or are you making the condition being in a different phase?

MR. SALTOS: Yes.

CHAIRMAN APOSTOLAKIS: Yes what

MR. SALTOS: Well, in terms of frequency, it will be much more and you would have to have many hot shorts.

CHAIRMAN APOSTOLAKIS: Even if the cable is on the same tray?

MR. SALTOS: Well --

CHAIRMAN APOSTOLAKIS: I mean, you just mentioned common cause failure.

MR. SALTOS: At the time, we didn't consider that if a (inaudible), and has another one next to it, we would consider that the hot shorts in those two cables would be dependent. So the probability that one would fail, the probability of the other failing, they don't have any common cause.

CHAIRMAN APOSTOLAKIS: But now you consider that --

MR. SALTOS: It is now time to figure that out.

CHAIRMAN APOSTOLAKIS: And that can lead to the opening of one valve, and I think that is the question from Mr. Leitch.

MR. SALTOS: Yes.

CHAIRMAN APOSTOLAKIS: And the question is --

MR. SALTOS: That you have more than two hot shorts.

CHAIRMAN APOSTOLAKIS: You have to have 3 or 4?

MR. SALTOS: Yes, 3 or 4.

CHAIRMAN APOSTOLAKIS: And you declare those as very unlikely?

MEMBER LEITCH: I am not concerned about multiple hot shorts. What I am concerned about is there a location where one could postulate a hot short that would open all the valves?

CHAIRMAN APOSTOLAKIS: A single hot short?

MEMBER LEITCH: A single hot short. I am picturing that at some point the circuit must be common to all four valves, and then you have got a cable going out to each and every valve, but at some point I would think that there is a commonality there. Is that not the case?

MR. CUMMINS: Maybe I can help. The ADS valves, each are in two pairs, and one pair that we have four actuation divisions. So one pair is actuated by both A and C actuation divisions; and the other pair is actuated by B and both B and D actuation divisions.

So the two valves are in one steam generator compartment, and the other two valves are in the other steam generator compartment. I don't know absolutely the answer to your question, but I would believe that it might be possible to actuate two of them, but it is not possible to actuate four of them.

MEMBER LEITCH: Okay. Thank you.

MR. SALTOS: The other outstanding issue is certification requirements. As I said before, an important objective of the AP-1000 PRA review is to use PRA results and insights to identify certification requirements.

And this is done by identifying important safety insights, related to design features and assumptions made in the PRA, and use such insights to support certification requirements, such as ITAACs, TS, D-RAP, and COL action items.

And to support the process used to determine appropriate regulatory treatment of non-safety systems. The identification of certification requirements requires integrated input from uncertainty, importance, and sensitivity studies.

And based on that we, we performed sensitivity studies to see how important is the issue, and do an importance analysis also to identify the importance of the issues.

And based on all this integrated results from this important sensitivity analysis, we decided what kind of certification requirements are important that we will to at future plants that we will have to achieve.

CHAIRMAN APOSTOLAKIS: Were you here this morning when we discussed the PRA?

MR. SALTOS: Partly.

CHAIRMAN APOSTOLAKIS: Were you here when we discussed the issue of common cause failures?

MR. SALTOS: Yes, I was.

CHAIRMAN APOSTOLAKIS: So that could be one of those?

MR. SALTOS: Yes. Yes. The common cause failures, you cannot do a PRA basically if you do not use common cause failures. You have to start with some number.

CHAIRMAN APOSTOLAKIS: The issue was can you do a common cause failure analysis on a generic basis.

MR. SALTOS: We do a generic basis, yes.

CHAIRMAN APOSTOLAKIS: And are you saying a requirement is that when you do the plant specific PRA to pay particular attention to it?

MR. SALTOS: Yes, you have to have a starting point. If they build the plant at the beginning, you have no information, plant specific information, and the staff will start with this.

So the safety for the human reliability

analysis that you talked about, we did sensitivity analysis and taking actually all the -- assuming that the operator would not do anything, and the risk increased, but it didn't increase that much like the operator in the plants.

The other sensitivity analysis we did was that we increased the operator and error probabilities, the human error probabilities, by a certain factor, and we saw that it didn't make much different; or if it did make any difference, that was part of our integrated process of defining sites and requirements for the design, like training procedures or whatever would be necessary.

Although I don't think that for AP-600 and also for AP-1000 that human errors are not as important as operating (inaudible).

MEMBER KRESS: As I recall, they assumed that the operator wouldn't do any of its required actions, CDF increased by a factor of 60.

MR. SALTOS: Something like that.

MEMBER KRESS: How do you decide whether that is okay, or that is --

MR. SALTOS: Well, it is not okay. It is an insight, and it tells you that this design does not rely on operator accidents as much as operating accidents.

CHAIRMAN APOSTOLAKIS: But this is based on the operator actions that have already been identified.

MR. SALTOS: Yes.

CHAIRMAN APOSTOLAKIS: Have you looked for possibilities of errors of commission?

MR. SALTOS: Well, I guess that was a long time ago, and we based this review on AP-600, and we didn't look for additional, unless it was due to some differences in the design.

CHAIRMAN APOSTOLAKIS: But now we come back to your earlier point that now we may have new information.

MR. SALTOS: Well, I don't think we have any new information that would change the results.

CHAIRMAN APOSTOLAKIS: There are NEUREGs where your colleagues on the staff compiled errors of commission in operating reactors. Wouldn't it be worthwhile to look at some of those and look at the general conclusions that your colleagues reached and see whether any of that would be applicable here?

Because, you know, I understand and appreciate raising the probabilities to one of identified human errors, but that would also be an additional investigation that would give us this warm feeling that these are better machines.

I mean, the NEUREGs exist and they have executive summaries, too, if you don't want to read the whole thing, and they say this is what has happened the last 15 years for such and such a reason.

And with a focus being on the NRC Commission, and that is part of the ATHENA effort, and the Office of Research.

MR. SALTOS: We considered some errors of commission at the time, but --

CHAIRMAN APOSTOLAKIS: On the AP-600?

MR. SALTOS: Yes, I am talking about the AP-600. But that involves the way of going against the procedures, and doing something that you are not supposed to do. It is not very easy to quantify probability anyway.

CHAIRMAN APOSTOLAKIS: And the rest of it is? Come on. You are talking about passive systems, and you are talking about all sorts of things here. And you can do a qualitative analysis.

MR. SALTOS: Yes.

CHAIRMAN APOSTOLAKIS: Like over there, I think one of the errors is throttling the high pressure injection system, and here can that happen?

Could they be asked to intervene and do that? I think that this kind of qualitative analysis would be useful.

MR. SALTOS: I think we did some of that for --

CHAIRMAN APOSTOLAKIS: You don't have to tell me that you have done it. Do you agree to do it?

MR. SALTOS: We asked for that, and we did not -- we don't find any mechanism that the operators would so something, and it was very likely to do something that would pose --

CHAIRMAN APOSTOLAKIS: But people are very creative and that is what I am saying. If you go back to the actual experience, you might see something where you say, gee, I didn't think of that, but it can't happen here because.

MEMBER KRESS: Is it the fact that they are only considering one operator in the control room change your perception of what the human error probability might be, rather than having a team of 2 or 3 operators? Is one person more likely to have a human error than if you have a team looking at the thing?

MR. SALTOS: Sure. Absolutely. It could make some change, of course, but I think that was included in the methodologies that were used to assess the human error probabilities.

MEMBER SIEBER: When we say one operator, that sort of misleading. There is an operator, but there is also a licensed supervisor.

MR. SALTOS: Yes.

MEMBER SIEBER: And the licensed supervisor is telling the operator what to do, and so the interchange between the two has a tendency to reduce the human error.

MEMBER KRESS: Or increase it. I mean, I am going to do what my supervisor tells me, whether I think it is right or not.

MEMBER ROSEN: No, I don't think so.

MEMBER SIEBER: Well, you are a different guy than me.

MR. SALTOS: But the important thing that we found --

MR. CORLETTI: Nick, excuse me, this is Mike Corletti from Westinghouse. On this subject of human errors of commission, for AP-600, one of the issues that was raised by the ACRS was to address issues of adverse system interactions, and we prepared a topical report on that, where we did the systematic approach of system interactions. Included in that evaluation was a qualitative assessment of the effects of human errors of commission, and as part of one of the RAIs that we received for AP-1000 was to repeat that systematic assessment, which we have included, and we just submitted quite a bit of information, and it probably has not been looked at yet.

But we have gone through that same -- it is a qualitative assessment of human errors of commission for AP-1000.

CHAIRMAN APOSTOLAKIS: Well, that is all that I am asking for.

MR. SALTOS: That is part of the PRA though.

MR. CORLETTI: It is no part of the PRA. It is part of the adverse systems interaction and evaluation. It is part of what we submitted.

CHAIRMAN APOSTOLAKIS: Yes, but you can go to the PRA and if you judge that some of them are credible, look at the LOCAs and ask yourself what happened.

MR. CORLETTI: It was written by Selim, and so it is part of our PRA, but it is not an official part of the PRA as far as it was not submitted with the PRA.

CHAIRMAN APOSTOLAKIS: Well, the staff

could do that.

MR. SALTOS: Well, yes, we concentrated our review to the differences, and this was not a difference between the AP-600 and the AP-1000. We have done some for AP-600, and that was seven years ago. I don't remember the details.

But I don't remember coming up with some scenario that would be very likely.

CHAIRMAN APOSTOLAKIS: And I agree. All I am saying is that just in the fire case, you argued that there is this additional information now that came from EPRI and maybe there exists additional information from the ATHENA project.

All you have to do is pick up the phone and ask for the report, and look at them, and evaluate it.

MR. SALTOS: The only difference is that the spurious situation was a big issue for AP-600. The human error probabilities and human error analysis was not that important.

CHAIRMAN APOSTOLAKIS: And you may conclude again that --

MR. SALTOS: We changed the human error probability by a factor of 10, and it would make a difference in the CDF by 11 to 50 (sic) percent or something.

CHAIRMAN APOSTOLAKIS: Again, it is the errors that they have already identified and what I am talking about is new errors.

MR. SALTOS: Okay. Yes.

CHAIRMAN APOSTOLAKIS: Are you saying that you don't want to do it?

MR. SALTOS: No, no. We will look at that in the future.

CHAIRMAN APOSTOLAKIS: Very good. That is all that I am asking. Why are we arguing here, just because of the national origin? Thank you, Bill. You pay attention, I see.

MEMBER SIEBER: I sure would like to go back to the question of the ADS, because I don't understand it.

CHAIRMAN APOSTOLAKIS: Of course.

MEMBER SIEBER: If I look at Westinghouse slide 16, that is a schematic of sorts, and they chose the ADS, and it seems to me that there is two valves on each train, and two trains on each route; is that correct?

MR. SALTOS: Yes.

MEMBER SIEBER: And then someplace else I heard that it is a DC system that is ungrounded. So if you have a significant hot short, with two valves in a series with different control systems and an ungrounded DC system, I don't know how you can get a single hot short and make that train operate. Maybe somebody can explain that.

MR. CUMMINS: It is not related to the ungrounded DC system. The valves are actuated with the PMS, which is an AC system, which came from DC power.

MEMBER SIEBER: Yes, but this is way back, the PMS>

MR. CUMMINS: The PMS does the arming.

MEMBER SIEBER: That is the logic end of it, right?

MR. CUMMINS: Right.

MEMBER SIEBER: And that is still DC and the output of the PMS.

MR. CUMMINS: There is no DC. The PMS runs on AC.

MEMBER SIEBER: Yes, the input.

MR. CUMMINS: The power to actuate the squib valves comes from the AC power of PMS. In some kind of charge capacitor comes conceptually way, and then also closes a switch conceptually way, both with AC power.

MEMBER SIEBER: But these are all signal strength, as opposed to tower strength?

MR. CUMMINS: Right.

MEMBER SIEBER: I mean, they are high on (inaudible) and global recert.

MR. CUMMINS: Right.

MEMBER SIEBER: And they are physically separated, right?

MR. CUMMINS: I believe that we agree with elements of that, and I think we are still under discussions with the staff as far as what are design really is, and whether this is an issue. I think the issues that have been raised in the industry reports are related to these hot shorts to ground, which don't really apply to this application.

MEMBER SIEBER: Well, maybe as a way to help me out, we are going to talk about this stuff at another meeting sometime, and maybe somebody can come back after they have looked at the wiring, and look at the physical locations, and explain to me how many shorts you actually have to have to make these systems operate. More than one.

MR. CUMMINS: That is what we would like to do. We have experts in this and I think we believe actually that it is essentially impossible to -- we are way lower in probabilities to do this, but let the expert explain it to you, and not me.

MEMBER SIEBER: And I would like to believe whatever the truth is, and I think you have to look at the circuits and the spacial relationships.

MR. CUMMINS: Yes.

MEMBER SIEBER: Okay. You have answered my question.

MR. SALTOS: There are several outstanding issues that have the potential to either individually or collectively to affect PRA results, and change certification requirements. with respect to AP-600, such as written requirements, for example. Examples of such issues are initiating event frequency changes.

For example, for large LOCAs, we talked about this this morning. The initiating event frequency changed by a factor of 50 or so. Maybe it is based on the NRC's contractor report, but I don't think that it is the NRC's position.

And additionally it includes more uncertainty, and uncertainty also has to be considered in the decision making process. And the same thing for the steam generator and tube router, and the PRHR-TR, and while the tubes and the number of hidden areas increased, the frequency decreased.

Another issue is the late containment failure modeling issue, which has to do with the passive containment cooling, and if there is no water cooling available, the success criteria for just air cooling are not -- we are not sure that the containment would survive and it is possible that containment failure would occur, and how that would impact core damage in the long term.

Westinghouse agrees with us with uncertainty as a sensitivity standard, and that the core damage frequency would decrease by 29 percent. Therefore, it is not big.

But on the other hand, for the (inaudible) of non-safety system failure persists when we don't credit the non-safety systems, this might be much larger than 29 percent.

And another issue that we have been discussing about is the common cause failure probability of explosive squib valves, which I related to safety injection line breaks, when one line is gone and you have just one line.

The common cause failure probability was calculated as 2 of 4 valves that are in the line that is not available anymore, instead of 2 of 2. And this makes quite a bit of difference in the results.

And what I am saying here is that if you combine the impact of all of this outstanding issues, some results might change and some of the conclusions could change regarding the certification requirements with respect to AP-600, of course, and also of course RTNSS.

CHAIRMAN APOSTOLAKIS: AP-1000.

MR. SALTOS: Well, we started with AP-600 and basically unless there is some difference because of the design difference, and they impact the PRA more, or the same, and we start with a list of certification requirements that we have for AP-600, and update that to reflect the changes, and the impact on the PRA.

CHAIRMAN APOSTOLAKIS: Is it a true perception of mine that you really are not dealing with any show stoppers? You are dealing with it down to the detail level, imposing additional requirements, and this and that, but you don't have an issue that might say, no, this is unacceptable, and you guys go back to the design boards?

MR. SALTOS: Well, yes, that is my feeling on this. Yes, I don't feel we have any, but we have to do this to make sure that we might help some important issue.

CHAIRMAN APOSTOLAKIS: Absolutely. You are doing your job, yes. Is that it?

MR. SALTOS: Yes, and we received a response from Westinghouse on this issue and it is under review, and we are working on this. This concludes my presentation. Any other questions to me?

CHAIRMAN APOSTOLAKIS: Thank you very much.

MR. BURKHARDT: This is Larry Burkhardt again, and the next staff reviewer or presenter will be Walt Jensen, discussing PRA success criteria.

CHAIRMAN APOSTOLAKIS: Who is Ms. Marie Pohida?

MR. BURKHARDT: She is to my left. She will be discussing shutdown PRA after Walt.

MR. JENSEN: I am Walt Jensen, and I work in the Reactor Systems Branch of the NRR, and I have been looking at the thermal hydraulic basis for the PRA to see if things are to be a success.

CHAIRMAN APOSTOLAKIS: Let me say something here.

MR. JENSEN: Sure.

CHAIRMAN APOSTOLAKIS: Were you here when they made the presentation on the thermal hydraulic --

MR. JENSEN: Yes, I was here.

CHAIRMAN APOSTOLAKIS: We are trying to save some time because we have an extra thing that we have to take are of as a supplement. Would you please not repeat things that we have had already and go directly to what you feel are your important points.

You don't have to tell us again how they did it,and so --

MR. JENSEN: No, I will not go into that. I am going to go very fast if you don't mind.

CHAIRMAN APOSTOLAKIS: Yes.

MR. JENSEN: I will move right along, and as you said, we have had a lot of discussions about the MAAP code, and we haven't -- we viewed the MAAP code, but it has been accepted as a tool to use as a scoping analysis.

Westinghouse benchmarked MAAP against their licensing codes for AP-600, and the results were about the same, but there were some differences in the defined structure of the sequence and the timing of when the systems actuate. But the overall conclusions were about the same.

We requested justification that the AP-600 benchmark using MAAP are valid for AP-1000, and Westinghouse promised to provide that to us. The minimum success paths, and these are the low margin sense of equipment that Terry talked about, and a lot of these were identified using MAAP and we think that they should be benchmarked against the licensing codes.

We asked for a sensitivity study for AP-600, and Westinghouse instead chose to use the bounding approach, and they used the same approach for AP-1000.

MEMBER KRESS: Why did they use MAAP? Was it because it runs so much faster than these licensing codes that they can run a lot more data and less failure?

MR. JENSEN: Yes, sir, I think it runs in just a few minutes, where I know it takes RELAP, and we have to run that all night to get the same sequence. So you are going to run 500 sequences and you would never get through using RELAP.

And we feel that all the limiting success paths that it would identify with MAAPS, and it would be verified with the licensing code. Westinghouse, of course, feels that the ones that are of very low risk are important for the PRA and don't need to be (inaudible) with the licensing codes, and we are reviewing the risk of the low margin. And we agree with Westinghouse that they are indeed of low risk.

CHAIRMAN APOSTOLAKIS: It is interesting that you do this for the thermal hydraulic analysis, but not for other elements of the PRA.

MR. JENSEN: I can only speak for the thermal hydraulics. I really don't know what is done in the rest of the PRAs.

CHAIRMAN APOSTOLAKIS: It is a loaded and unfair question and you handled it very well. You say we have reviewed MAAP4, but they are doing it, and Mr. Saltos just told us that we are using the PRA insight, and so is all of this allowed because PRAs are not formally required by the regulations?

MR. BURKHARDT: It is formally required.

MR. JENSEN: Well, we have done some review and it has been benchmarked against the licensing codes, and we have a pretty good feel about it. But we just would like to see the end states to be verified by the licensing code.

CHAIRMAN APOSTOLAKIS: But there is a slight conflict though, because the licensing codes are currently conservative, and the PRA is supposed to be at least, right?

MR. SALTOS: This is Nick Saltos. Let me see if I can answer that. Because of this (inaudible) and the magnitude of the uncertainties, not addressing and having the conservative success rate criteria is the equivalent of having some addtional failures in the PRA that would increase the CDF.

CHAIRMAN APOSTOLAKIS: You are doing fine, you know.

MR. SALTOS: Because some of them would be much more significant in other areas, and the success criteria in the PRA are a very important part of the PRA, and if the success criteria is best estimate, then basically you don't have a good PRA.

CHAIRMAN APOSTOLAKIS: But on the other hand, here is this agency spending a few millions of dollars developing the ATHENA methodology for human error analysis, and they have convinced this committee that there is such a thing as an error forcing context, and that it could be very important. And how we are about to certify a design, and we don't even mention it that there is such a thing as an error forcing context.

And I don't know. Are there any error forcing contexts here? Was the NRC wasting its time and money when it was sponsoring that major project for years? I don't know. I mean, we seem to live in parallel universes. I am not complaining, even though it sounds like I am complaining.

But I think this committee at some point -- I am planting a seed now, Mr. Jensen. This committee at some point has to face that problem. I mean, the Office of Research is not a separate pipe there that is empowered to add to the others, and the real work doesn't require what they are doing.

I mean, for years now we have been hearing about the error forcing context and I am perplexed. Do we have any error forcing context here? I never even heard the word. So let's go on.

MR. JENSEN: Well, perhaps we are hearing a conservative PRA because of the bounding approach that Westinghouse has taken.

MEMBER SHACK: Let me just say the large break LOCA is treated by a best estimate code, right? And the small break LOCAs are treated by an Appendix K type code; is that correct?

MR. JENSEN: That's true. WCOBRA-TRAC for --

MEMBER SHACK: And you would include your uncertainties in your analysis reports?

MR. CORLETTI: Right.

MR. JENSEN: Okay. The purpose of this slide is to say we have benchmarked some of the NOTRUMP PRA calculations, and PRA bounding calculations with RELAP, and these had numerous failures and which resulted in some (inaudible) in the second case, and for a fairly extended time, but Westinghouse checked or calculated the peak cladding temperature of around 1500 degrees, and we calculated less.

But to me this shows the robustness of the plant design for small break LOCAs, and that all these failures can occur and still (inaudible).

And this is just a sample of a comparison between RELAP and NOTRUMP.

Well, it is amazing, the same results. This is just impressive, but the passive systems are operating on about the same sequence, and the controller is decreasing the pressure. So this is very gratifying.

We did one comparison with MAAP, which is not such a limiting scenario, and it only fails one accumulator, and one of the four ADF4s, and it does consume containment isolation failure, which it just imposes the atmospheric pressure on the steam within the reactor and so the ADF4 is effective in relieving the steam from the reactor system.

Now, we don't get such a good comparison with MAAP, and the MAAP calculation is a lot higher than RELAP, until all of a sudden ADS4 comes on somewhat earlier than RELAP, and the pressure just goes dropping like a stone as you see.

Again, ADS-4 actuates earlier than MAAP, and a lot more flow of course puts the pressure higher when ADF-4 does actuate. And the break flow is about the same idea, but in MAAP undergoes sudden changes between high and low, which I believe is simplifying assumptions used in the code that switch the quality from a two-phased mixture to a separated flow, and that does it very abruptly in there.

So basically the conclusions from the staff audit calculations are NOTRUMP and RELAP, you get about the same answer, and they show predicted for one case, and both codes predicted brief periods of core uncovery, which were within acceptable limits.

MEMBER KRESS: So you are saying then that RELAP results are in your mind a good representation of the codes that they are going to use, so that your comparison of RELAP and MAAP gives you some indication what they might get when they do their comparison?

MR. JENSEN: I think so and when they compare MAAP to NOTRUMP, they are going to get about the same results that I get with RELAP, because RELAP and NOTRUMP seem to be getting equivalent results.

MR. JENSEN: And then we have reviews still going on, and we have unresolved issues. Westinghouse claims a failure with one of the ADS4,and they can withstand containment isolation failure, and still cool the core and long time cooling, and we have asked for that to be verified.

This is one of the scenarios though, and that i believe Westinghouse says is very low risk and the risk is so low that it is inconsequential. So we are working with that.

We are looking at the scenario where the 18 inch vent line opened in the containment, and the containment is not isolated and a LOCA occurs, is there still enough water contained within the containment building to keep the core cool, and Westinghouse analyzed that with MAAP.

Again, we would like that to be verified with one of the licensing codes, like WGOTHIC, and we are wondering about maybe some entrainment might occur from a larger break and it might get carried out of the open vent, and they are going to respond to that.

MEMBER KRESS: Where is the vent? Is it physically in containment, or are you just postulating any kind of event?

MR. JENSEN: I don't know. I don't know

whether Westinghouse postulated that.

MEMBER SIEBER: Usually you can't get entrainment unless you are pretty close to where the surface is.

MR. CUMMINS: That is how we are going to answer the question. It is assumed to be an HVAC vent, the largest existing design penetration on the containment.

MEMBER KRESS: ADS-4 discharges a sonic velocity choke flow, and how does the containment pressure influence this, in terms of whether it is isolated or not?

MR. JENSEN: Well, at first there would be choke flow, but then later the flow would become non-choked.

MEMBER KRESS: But that would be way out at the end of the thing wouldn't it?

MR. JENSEN: It is my understanding that the reason --

MR. CORLETTI: It becomes unchoked below a hundred psi as far as the reactor coolant system pressure.

MEMBER KRESS: Well, that sequence is over.

MR. CORLETTI: For large breaks, during

the PCT for a large break, I think you are right, the pressure is higher. For the key area that we were thinking of, which was for small break, was at the time of IRWST injection, and it isn't choked at that point.

MR. CORLETTI: I don't think it takes very long for pressure to get below a hundred psi once we go to stage 4 ADS.

MEMBER KRESS: Yes, it comes out of there pretty fast.

MR. JENSEN: Then Westinghouse has used the AP-600 analysis to justify some of the success paths, and we have asked that these be verified to be applicable to AP-1000 and they are going to provide us with that.

And then last of all, we are reviewing the risk significance of the unbounded cases and the expanding event trees to see if we agree with the risk, and if it is success to have these low risk paths to be unbounded by analysis with the licensing codes. And that concludes my presentation.

CHAIRMAN APOSTOLAKIS: Good job.

MEMBER KRESS: Looking at the ADS4 results, compared to RELAP for a couple of these cases, it looks like in my mind that the MAAP 4 is conservative from the standpoint of whether or not the core gets uncovered, compared to RELAP, and would that be your assessment?

MR. JENSEN: I don't know. Both of these --

MEMBER KRESS: The pressure stays up higher, for example, in this.

MR. JENSEN: The pressure was higher.

MEMBER KRESS: And to me that means that you are getting less injection coming in and probably less going out the relief valves. I don't know if that means more coming out of the relief valves and less injection coming in. That is what I would assume that higher pressure does to you, which means that the core is uncovering more.

But an auxiliary question to that is have you looked at MAAP to see why it has this difference?

MR. JENSEN: No, sir, we have not reviewed MAAP in detail. We haven't been funded to do the review.

MEMBER KRESS: It would take a pretty big effort wouldn't it?

MR. JENSEN: And I know that there are some user functions in map that the user can tune the results to get the appropriate answer, and I would suspect that Westinghouse is keying the MAAP input to pretty much follow NOTRUMP as close as they can. So this is why we sort of get the same answer with RELAP.

But there was no core uncovery in either MAAP or RELAP, and so I can't really say which is the more conservative, but as you say the pressure is higher, but perhaps if there were more leak flow, and I guess they did get about the same leak flow. It looks like suddenly that maybe the quality switches, and the voids are collapsed, and the liquid is coming out of the break. I am not sure what it is doing.

MEMBER KRESS: Yes, I don't know what causes those things. Does MAAP use the same critical flow model that RELAP does?

MR. JENSEN: I don't know. Westinghouse can pitch in.

MR. SCOBEL: This is Jim Scobel from Westinghouse. MAAP uses Henry Falsky for critical flow. I don't know what RELAP uses.

MR. JENSEN: Thank you, Jim. All right. Well, if there are no more questions, then Maria Pohida will talk about --

MEMBER RANSOM: Mr. Chairman, I don't have a question, but I do have a comment.

CHAIRMAN APOSTOLAKIS: Sure.

MEMBER RANSOM: It seems to me that there is an awful lot of subjectivity in the selection of cases that are used for bounding or checking for whether or not damage would be expected in these calculations between the, say, simplified methods and the more detailed methods.

And there are enough cases and history where subjectivity engineering judgment has been wrong to make me at least a little bit nervous about that. And I don't see why you wouldn't apply a statistical approach to something like this in a sampling, and there are very good methods for telling you how many cases you actually have to check in order to achieve a given confidence level in the result.

And that would it would seem to me to provide a lot more tighter justification for whatever reliability you want to place on such calculations. Whereas, simply choosing a few and sampling may give you a warm feeling, but it doesn't really to me at least tell me where I am at in terms of reliability of those results.

MR. SCHULZ: This is Terry Schulz, Westinghouse. I don't think I understand what you mean by subjectivity in choosing cases. I think that I or at least I tried to show you a very systematic way that we selected the low margin risk important cases, and that was not subject to engineering judgment or sampling.

MEMBER RANSOM: I think a better way would be to statically choose these cases, which gives you a method then for providing a convincing argument on what degree of reliability or confidence level you can place on those.

MR. SCHULZ: We may be able to do that.

MEMBER RANSOM: To give you an example. For example, when NASA fires a rocket, they will fire it a few times and then measure the specific impulse that they obtain, and from just a few samples, you can actually get a randomly chosen -- this would be with a solid (inaudible), and then with a high degree of probability predict what the expected performance from those additional ones would be. And they do use those such approaches.

And I would think that you could do the same thing here, unless you can by some other course, if you never depressurize the system, and you would never expect any 2-phased uncovery of the reactor vessel, and you could rule out cases like that presumably.

But if there are cases where you might

suspect that there could be core uncovery, and yet you want to use the simplified methods to explore a large number of cases, then you should be able to statistically sample that large number and benchmark them I guess against your more detailed code, and then tell a person to what degree of reliability you can rule out a possibility of core damage as a result (inaudible) --

MR. CUMMINS: Can I just clarify how we selected? We selected as low margin every case where MAAP predicted core uncovery. We didn't sample.

Every case where MAAP predicted core uncovery, we put it in the low margin bin, and then we tried to disposition that and either as significant to the PRA or not significant to the PRA. And if it was significant to the PRA, we used our DCD analysis codes.

MS. POHIDA: Okay. As I was introduced earlier, I am Marie Phida of the PRA group at NRR, and I am the current reviewer of the AP-1000 shutdown PRA. My review of the AP-1000 shutdown PRA is based on my review of the AP-600 shutdown PRA.

I issued several RAIs and many of them focused on changes from the AP-600 PRA to the AP-1000 PRA, and that includes common cause failure. the probabilities and the grouping of the high pressure gravity injection squib valves, and the high pressure recirculation squib valves.

They were put together as a single common cause failure group. This failure is risk significant, and appears in many of the dominant sequences of the shutdown PRA.

Shutdown risks for the AP-1000 design as you probably heard this morning is dominated by failures of the normal R&S system or the failure of the support systems during drain maintenance outages.

MEMBER KRESS: What CDF do you get from shutdown?

MS. POHIDA: It was 1,23, 10 to the minus 7.

CHAIRMAN APOSTOLAKIS: About 30 percent then.

MS. POHIDA: And with the bulk of that, 60 percent of that, occurring during drain maintenance operations. So because the path charged system is not available, the first three stages of the ADS valves are open, and what you have is if you were to have a loss or interruption of the residual heat removal system, or the R&S system, what you have left is gravity injection.

If it fails to go automatically, then the operator can try to initiate injection manually through the IRWST injection lines, or initiate injection through the R&S suction valves, the R&S lines.

Also, the RAI also focused on common cause failure of the low pressure recirculation squib valves, and once again since recirculation is required, following a loss of the operating train of (inaudible) during mid-loop operation, you need successful gravity injection and recirculation.

My review also focused on shorter response times for operator recovery actions, and these include containment closure, and containment closure is required to maintain long term cooling water inventory.

And specifically we have reduced times to boiling and it is now 17 minutes, and it was 17 minutes in the AP-600 design, and it is now 10 minutes for AP-1000 design. The containment closure capability is covered by the AP-1000 tech specs, shutdown tech specs.

MEMBER ROSEN: How do they have a containment open? Do they have the equipment hatch open during mid-loop operations? What are you assuming?

MS. POHIDA: I can't remember if I assumed that analysis, but that is also part of my RAIs, is basically how did you arrive at your containment failure closure probabilities, okay? So that is still part of my review.

MR. CORLETTI: If I could, I could address it. Our tech specs are to the opening of the equipment hatch to a time to boiling based on the amount of decayed heat that would be in the core.

So that for periods of time where the time for boiling would be rather short, the containment equipment hatch would have to be in place and would not be allowed to be open.

And that takes into account the decayed heat level and the inventory, and the water if it is a mid-loop operation.

MEMBER ROSEN: But operating practices say that you don't open the containment hatch while you are at reduced inventory.

MR. CORLETTI: That's right, and if you apply that criteria that would be the outcome for AP-1000. But it is based on a criteria with low -- say after a long refueling, and you were coming back up, and you wanted to go to bin loop, and you didn't have a lot of stored energy, it would take a long time for boiling.

And the equipment hatch would be allowed to be opened.

MEMBER ROSEN: Back-end mid-loops.

MR. CORLETTI: Yes.

MEMBER ROSEN: But these days back-end mid-loops occur -- the average durations are getting so short, and I don't want to go to AP-1000, but on current plants, the difference between back-end and front-end is 20 days.

MR. CORLETTI: And really the way that the tech spec is set up is that you have to ensure that you would have adequate time to close containment before you would have steaming. So if the timing is such that you cannot show adequate time, you would not be able to have the equipment hatch open.

MS. POHIDA: Okay. Well, this whole containment closure issue still is under review there, because it also impacts -- what about release, and in the event of a severe accident at shutdown, and you were for some reason unable to close your containment, what is your source term, and what is your release frequency if you will.

So that is still under review, that whole area, okay? The shorter response times also impact gravity injection, manual gravity injection. To me this issue is secondary to the primary issue, which is containment closure, because the low shutdown risk estimates that are reported in the AP-1000 PRA stem from the fact that you have automatic injection, which is much different that we currently have at operating plants.

Once you have low level in the hot leg, your ADS4 valves open up, and you have automatic injection from the IRWST. We also asked some additional questions and one was trash control during shutdown, and once again recirculation required to maintain a long term cooling water inventory, and we wanted to make sure that trash was controlled so that the common cause failure estimates for the sump strainers plugging up made sense.

There wasn't a shutdown fire or flood risk assessment that was provided to the staff and once again our concern is that during shutdown you have a lot of people moving around the plant, and you may have fire barriers that are breached or open while people are performing maintenance or testing.

So that is another area of our focus, and what I would like to say is that we have not seen any show stoppers as of yet, but we are still have not completed our review and that ends my discussion.

CHAIRMAN APOSTOLAKIS: Did you also raise all the human error probabilities to one to see what happens to the core damage frequency?

MS. POHIDA: Okay. I am trying to think. With the AP-1000 design, Westinghouse didn't report any importance analyses. Now, based on the results of the AP-600 importance analyses, there was not a tremendous change in CDF.

There was not a lot of liability associated with the automatic gravity injection.

CHAIRMAN APOSTOLAKIS: How about a larger release frequency? The containment closure issue.

MS. POHIDA: The containment closure issue?

CHAIRMAN APOSTOLAKIS: They never close it. What happens? I wonder whether the same sensitivity analysis that was done for level one power would show that even if all the humans make mistakes all the time, still the core damage frequency is low and the LERF is slow, and that applies to low power and shutdown operations.

Maybe you want to think about it. You don't have to answer now, but that is certainly something that I would be interested in knowing.

MS. POHIDA: Well, that's why I am bringing it up on the slide, because of that tech spec, which is supposed to say that you are not going to open anything up unless you can get it closed before the RCF begins to boil, no release frequencies for shutdown were reported.

And I agree with you that during my review that I need to make sure that that still makes sense, in light of that now the boiling takes 10 minutes. It is almost half of what it was in the AP-600 design.

CHAIRMAN APOSTOLAKIS: Okay.

MS. POHIDA: And that's it.

CHAIRMAN APOSTOLAKIS: Any other comments to Marie?

MR. CUMMINS: I am not sure we quite understand the containment closure. If it took -- let's say it takes an hour to close an equipment hatch, what the tech spec says is that -- and let's say it takes us as she said 10 minutes to get to boiling, you cannot open the equipment hatch until it takes an hour to get to boiling if it takes an hour to close the equipment hatch.

So you have to sort of measure your decayed heat, or calculate your decayed heat, and you can only open the equipment hatch at mid-loop if the time to boiling is longer than the time to close the containment. So in this case the time is protected by the tech spec, and I suppose you could still argue or we could always ask what happens if the operator fails to follow the tech specs.

And that is sort of beyond what we normally do in the PRA. We assume tech specs, I think.

CHAIRMAN APOSTOLAKIS: Well, you have a certain period of time and you are asking what is the probability that they will actually do it in that period of time.

And I am a little concerned about all these sensitivity studies that are so extreme. They work here and so we advertise them as look, we found the problem. We set all the human error probabilities to one and nothing happens. That creates a precedent, and what if something does happen and you do that to low power shutdown.

And then you back away from it, right? And you say, well, that was too much. I will do something else. And that makes me a little uncomfortable with the whole thing.

MS. POHIDA: Well, those importance

analyses that I was referring to for the AP-600 design, that didn't cover containment closure. That only covered the level one portion of the analysis.

CHAIRMAN APOSTOLAKIS: Anyone, you are reviewing AP-1000.

MS. POHIDA: Yes.

CHAIRMAN APOSTOLAKIS: And your final determination will not be I hope that this design is as good or better than AP-600. I mean, it would be an absolute determination won't it?

MS. POHIDA: Yes.

CHAIRMAN APOSTOLAKIS: You are using AP-600 for convenience, but it will be an absolute determination.

MR. BURKHARDT: That's correct. It will be a stand alone evaluation based on the AP-1000 design. We may discuss some differences to the AP-600, but the evaluation will based on the AP-1000 design.

MS. POHIDA: And the insights that we generate will be based on the AP-1000.

CHAIRMAN APOSTOLAKIS: Okay. Any more comments from the members? Any questions for Marie?

MR. CORLETTI: No, I don't have comments for Marie right now.

CHAIRMAN APOSTOLAKIS: From the NRC staff? No. Thank you very much.

MS. POHIDA: Thank you.

CHAIRMAN APOSTOLAKIS: And how it is back to you.

MR. CORLETTI: I think we can wrap up today's meeting. I don't think that they are that crucial, but I have some slides.

CHAIRMAN APOSTOLAKIS: How many do you have?

MR. CORLETTI: Three, but I think I will just wrap this up in five minutes. I think just in the next several slides really characterize the areas that the RAIs covered, and the RAIs related to the PRA.

And just to clarify with our answers, we did make changes to the PRA that we submitted with the RAI responses, and we collected those changes to incorporate them all into the PRA.

We expect to be able to submit the PRA with those revisions by the end of this month, the end of January. I think we have listened to the staff and the issues that were characterized I think all are in progress.

And I think we are working with them to

resolve those. I think what I have heard from this committee in regards to additional information that you might want to hear, or we want to hear, is on the ADS valve, and to me it sounds like we want to hear about the valves, the developed design features, and how it works.

And also the issue of the control and how the power to valve, and how we have attributed spurious actuation hot shorts, and I think we could handle that in the plant meeting later.

MEMBER ROSEN: But also on the valve, and not just how it works, and the design, and the likelihood of stress corrosion cracking of the seam, and other issues of vulnerability of materials, and what the reliability numbers for the valve.

MR. CORLETTI: And the basis for those.

MEMBER ROSEN: And the basis for those.

MEMBER SIEBER: And how to get explosives past the security guard.

MR. CORLETTI: I guess I would then like to open it back up to you. Is there additional items that you have heard today that you think rise to that same level that you need more information? Otherwise, I don't think I have anything else at this time.

CHAIRMAN APOSTOLAKIS: But you are not

ignoring the other minor points that we made. I mean, you just pointed out what you think are the most important.

MR. CORLETTI: Right. No, of course not.

MEMBER KRESS: On the squib valves, you mentioned that there was smaller squib valves like this out there in plants?

MR. CORLETTI: Right.

MEMBER KRESS: Has there been any experience on them being in place for a number of years, like 10 or so and then people taking them and testing them afterwards to see if they work?

MR. CORLETTI: Well, as a matter of fact, that is what the in-service testing requirements for squib valves that are in operating nuclear plants.

MEMBER KRESS: Yes, but that only goes to the point of they never shoot people with a bullet.

MR. CORLETTI: They test the charge.

MEMBER KRESS: Yes. And I worried about the charge deteriorating over time, for example.

MR. CORLETTI: Yes, they test the charge every -- it is in accordance with the ASME. Periodic testing, Terry, just like some percentage.

MEMBER LEITCH: They are in BWRs, and the same bi-liquid control systems, and which are fairly small. I would say they are on the order of one inch valves, ambient temperature, and as I recall the charge has to be replaced once per refueling outage, but what you do is you test and get a batch of charges, and you test a sampling of that batch, and if the sample works okay, then it implies that the charge is okay, and you use that particular charge.

MEMBER KRESS: They have been stored at room temperature though.

MEMBER LEITCH: Yes, in storage at room temperature.

MR. SCHULZ: This is Terry Schulz. ASME has specific requirements for in-service testing of squib valves, and I am not sure I remember the exact frequency, but for our ADS squib valves, we do not have to replace the charge every refueling outage on every valve.

MR. CORLETTI: It is a sampling.

MR. SCHULZ: So what we are doing is on a sequencing basis, like one valve every refueling outage, and then over a period of four refuelings, we replace every one of the charges over 6 to 8 years.

And when we replace that charge, we take the charge that was in the valve under the actual service conditions, and we go put it at a text fixture and actually fire it.

MEMBER KRESS: Okay. That is what I was interested in.

MR. SCHULZ: And that is in addition to other controls that they have put on when they make the charge material initially, and they do basic tests there to make sure that it is okay before you put it in, and then we do these post-service tests also.

MEMBER KRESS: What is the charge?

MR. SCHULZ: What material? I don't know.

MEMBER ROSEN: Well, my concern is more than just that the charge goes off, is that the valve works, and that it severs whatever, and locks over. I mean, just having the charge work and operates doesn't do you any good.

MR. CUMMINS: But we will cover this in our next meeting.

CHAIRMAN APOSTOLAKIS: It would really be refreshing to have a realistic estimate of the uncertainties in all of these things, and I am serious now. A factor of six, I don't think is appropriate here given all the judgments and so on, and this revelation that they are mean values, because as I read the report, it says here and there, and we are very uncertain about this thing and we put a factor of 10.

But given that your point estimate was a mean value, what you are telling me is that you are stretching the distribution on the low side. A factor of 10 doesn't mean the same thing with what it meant with the reactor safety standards, where You would go 10 up and 10 down. Now you are just pushing it all the way down.

And you may say this is a calculation and instead of 2 times 10 to the minus 7, you may find now 4 or 5 times 10 to the minus 7. But even with all these uncertainties and judgments about common cause failures of software and this and that, realistically is it a factor of 10 or 12, up and down, or up, and that is what I am interested in.

I mean, it would still give you below the goals, but it would be nice to have some sort of -- I mean, instead of using formal methods to propagate uncertainties that are not important to begin with, like failure rates, you have this realistic assessment at the end.

MEMBER ROSEN: Could I have one more word?

CHAIRMAN APOSTOLAKIS: Yes.

MEMBER ROSEN: Not on that subject, but

going back to the reliability and squib valves. What I am really trying to do is not to get you to do some sort of academic exercise and come back with some numbers that you can put up on the screen.

What I am really trying to do is get a solid feeling of the reliability of this valve on command that it will actually open, and that this is a valve that has not been built yet.

And at some point it would seem to me that it needs to be built and some component testing done of it before we -- and if it was a valve out in the periphery, sure, no. But it is at the very heard of the safety analysis of this plant, and that is my concern.

MEMBER SIEBER: There have been squib valves used in applications other than this one.

MEMBER ROSEN: Well, at this temperature, you know, and with these kinds of pressures, what I am trying to get before I say I am wiling to say, gee, I think this is great. I didn't sign off on AP-600, but I am going to have to be part of the process on AP-1000.

I want that warm comfortable feeling that I have great confidence in this valve's ability to function as designed.

CHAIRMAN APOSTOLAKIS: Any other comments from the members? Thank you, Mike.

MR. CORLETTI: Thank you.

CHAIRMAN APOSTOLAKIS: And your colleagues as well, and we will see you again tomorrow, right?

MR. CORLETTI: At 8:30.

CHAIRMAN APOSTOLAKIS: At 8:30.

MR. CORLETTI: Thank you.

(Whereupon, at 5:04 p.m., the meeting was adjourned, to reconvene at 8:30 a.m., on Friday, January 23, 2003.)

CERTIFICATE

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

Name of Proceeding: Advisory Committee on Reactor Safeguards Subcommittees on Reliability and Probabilistic Risk Assessment

Docket Number: n/a

Location: Rockville, MD

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.

Matt Needham

Matt Needham
Official Reporter
Neal R. Gross & Co., Inc.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE SUBCOMMITTEE ON
RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
ROOM T-2B3, 11545 ROCKVILLE PIKE, ROCKVILLE, MD
January 23 and 24, 2003

ACRS Contact: Michael R. Snodderly (301) 415-6927
E-mail: mrs1@nrc.gov

- PROPOSED SCHEDULE -

Thursday, January 23, 2003

	TOPIC	PRESENTER	TIME
1)	**Introduction** ▸ Review goals and objectives for this meeting	George Apostolakis, ACRS Mike Corletti, Westinghouse	8:30-8:35 am
2)	**Overview of AP1000 Design** ▸ Design Changes from AP600 ▸ Key AP1000 Key Design Features ▸ Defense in Depth ▸ PRA as a Design Tool	Terry Schulz, W	8:35-10:05 am
	**** BREAK ****		10:05 - 10:20 am
3)	**AP1000 PRA** ▸ Background / Approach / Overview ▸ Scope ▸ Level 1 PRA Internal Events At-Power, Including Uncertainty ▸ Shutdown / Fire PRA	Selim Sancaktar, W	10:20 - 12:20 pm
	**** LUNCH****		12:20 - 1:30 pm
4)	**PRA Level 1 Success Criteria** ▸ Overview ▸ Thermal-Hydraulic Analysis to Support Level 1 PRA ▸ T&H Uncertainty Assessment	Terry Schulz, W	1:30 - 3:30 pm
	**** BREAK****		3:30 - 3:45 pm
2)	**NRC Staff Presentation** ▸ Staff RAIs on Level 1 PRA and Success Criteria	Nick Saltos, NRR Walt Jensen,NRR Marie Pohida, NRR	3:45 - 5:30 pm

6)	**Westinghouse Summary**	Mike Corletti, W	5:30 - 5:45 pm

Friday, January 24, 2003

	TOPIC	**PRESENTER**	**TIME**
1)	**Introduction** ▸ Review goals and objectives for this meeting	George Apostolakis, ACRS Mike Corletti, Westinghouse	8:30-8:35 am
2)	**Level 2 and 3 PRA** ▸ Quantification ▸ Level 2 Phenomenological Studies	Jim Scobel, W	8:35-10:15 am
	****BREAK****		10:15-10:30 am
3)	**ULPU Testing Performed for AP1000** ▸ AP600 Background ▸ Test Program ▸ RV Insulation Design	Jim Scobel, W	10:30-11:30 am
4)	**PRA Importance and Sensitivity Studies**	Selim Sancaktar, W	11:30-12:15 pm
	****LUNCH****		12:15 -1:15 pm
5)	**NRC Staff Presentation** ▸ Staff RAIs on Level 2 & 3 PRA	Bob Palla, NRR Richard Lee, RES	1:15 - 2:15 pm
6)	**Westinghouse Summary**	Mike Corletti, W	2:15 -2:30 pm
7)	**General Discussion and Adjournment** ▸ General discussion and comments by Members of the Subcommittee	George Apostolakis, ACRS	2:30-3:00 pm

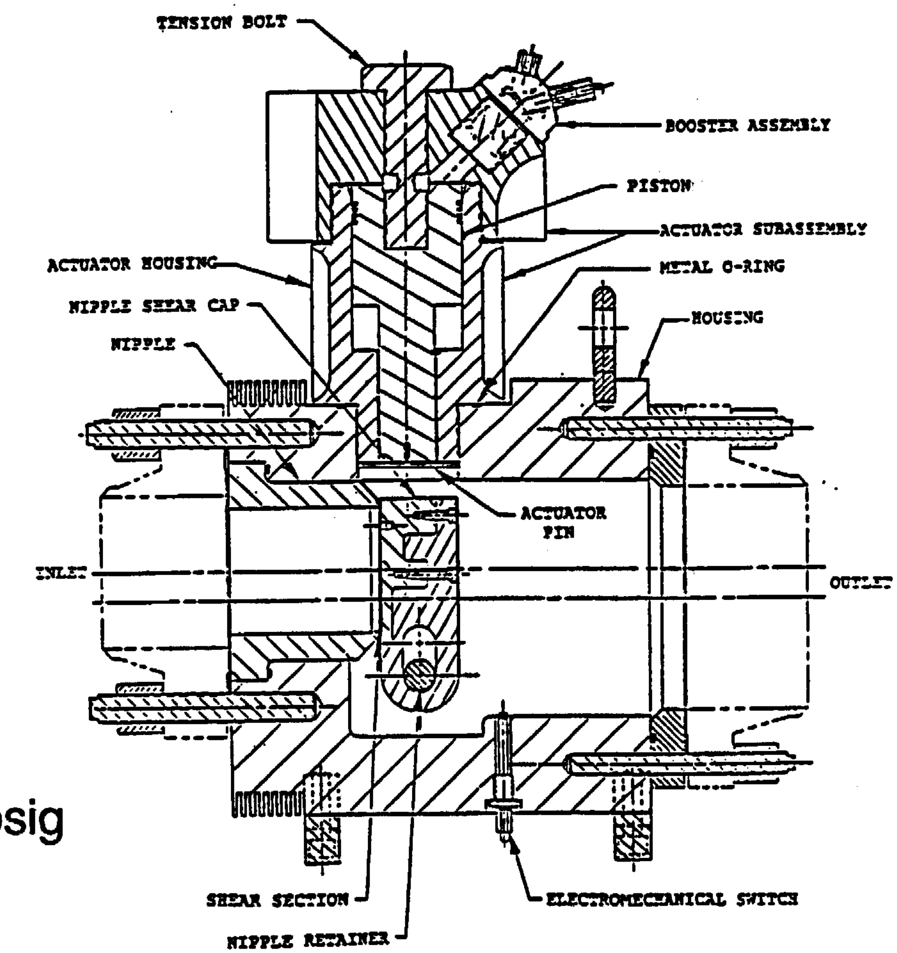
Note:

Presentation time should not exceed 50% of the total time allocated for a specific item.

Number of copies of presentation materials to be provided to the ACRS - 35.

ADS Stage 4 Squib Valve

- **Controls**
  - Two stage "arm" / "fire" circuit prevents spurious opening
  - Three ignitors provided in each valve
    - 2 wired to different PMS divisions
    - 1 wired to DAS
  - Auto opening (PMS) requires
    - SI signal (2/4) and
    - CMT low 1 (2/4) signal and
    - CMT low 2 (2/4 signal and
    - RCS pres < 1300 psig
  - Manual opening requires
    - PLS - 2 step switch & RCS < 1300 psig
    - PMS - 2/2 dedicated switches
    - DAS - 2/2 dedicated switches



STATUS OF AP1000 SHUTDOWN PRA REVIEW

Marie Pohida
Probabilistic Safety Assessment Branch (SPSB)
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

January 23, 2003

Based on Staff review of the AP600 Shutdown PRA, SPSB issued 9 RAIs on the AP1000 Shutdown PRA.

RAIs focused on changes from AP600 PRA to AP1000 PRA

- ❑ **Common cause failure of the high pressure Gravity Injection squib valves and the high pressure recirculation squib valves**

- ❑ **Common cause failure of the two low pressure recirculation squib valves.**

- ❑ **Shorter response time for operator recovery actions including:**

    - **Containment Closure (required to maintain long term cooling water inventory)**
    - **Manual Gravity Injection**

SPSB asked additional RAIs on:

- ❑ **Trash Control during shutdown**
- ❑ **Shutdown Fire/Flood risk assessment**

SPSB has not completed their review of the AP1000 RAI responses.

NRC STAFF REVIEW OF AP1000 LEVEL1 PRA INTERNAL & EXTERNAL EVENTS AT POWER OPERATION

ACRS Subcommittee on Reliability and Probabilistic Risk Assessment
January 23, 2003

Nicholas Saltos
NRR/DSSA/SPSB

AP1000 PRA REVIEW - - - MAJOR OBJECTIVES

- *ENSURE PRA QUALITY COMMENSURATE WITH ITS INTENDED USE, SUCH AS*
  - *Gain insights about the design*
  - *Support the design and certification processes*
- *ENSURE PROPER INTERPRETATION AND USE OF PRA RESULTS FOR DECISION MAKING IN THE CERTIFICATION PROCESS, SUCH AS*
  - *Identify design and/or operational changes to address weaknesses*
  - *Identify "certification requirements," such as ITAACs*
  - *Determine appropriate regulatory treatment of non-safety systems (RTNSS)*
  - *Determine the risk significance of raised issues*

AP1000 PRA REVIEW - - - APPROACH

- *RELIANCE ON SIMILARITY OF AP1000 TO AP600 CERTIFIED DESIGN TO REDUCE REVIEW EFFORT*
  - *Same system functions, spatial arrangements and capabilities*
  - *The AP1000 PRA uses the AP600 PRA as the starting point*

- *IDENTIFICATION OF DESIGN DIFFERENCES BETWEEN AP1000 AND AP600 HAVING AN IMPACT ON PRA MODELS*
  - *Major differences are due to the power uprate*
  - *Several other minor but potentially significant differences*
  - *Identification of AP1000 PRA areas for review*

- *IDENTIFICATION OF ADDITIONAL DIFFERENCES BETWEEN AP1000 AND AP600 PRAs THAT ARE NOT DUE TO DESIGN DIFFERENCES*

- *FOCUS REVIEW ON IMPACT OF CHANGES ON IMPORTANT ISSUES IDENTIFIED DURING THE AP600 PRA REVIEW*

AP1000 PRA REVIEW -- LEVEL 1 PRA MAJOR ISSUES (OPERATION AT POWER)

- *THERMAL-HYDRAULIC (T/H) UNCERTAINTY/SUCCESS CRITERIA*

- *FIRE-INDUCED SPURIOUS ACTUATION OF ADS SQUIB VALVES*

- *IDENTIFICATION OF "CERTIFICATION REQUIREMENTS," SUCH AS ITAACs AND RTNSS*

  - *Resulting from design differences with respect to AP600*

  - *Could change according to the resolution of outstanding issues*

AP1000 PRA REVIEW - - T/H UNCERTAINTY

ISSUE DESCRIPTION

- *Passive systems rely on small driving forces. The uncertainty in the values of such driving forces can be of comparable magnitude to the predicted values themselves. When T/H uncertainties are considered, "success" accident sequences may actually lead to core damage*

- *This issue was addressed in the AP600 PRA by a risk-based bounding approach which uses conservative assumptions for key T/H parameters:*

  - *Identification of "low T/H margin risk significant" accident scenarios*

  - *Use of DBA computer codes to bound T/H uncertainty*

- *Such an approach relates the impact of T/H uncertainties to changes in success criteria and, thus, to changes in risk*

AP1000 PRA REVIEW - - T/H UNCERTAINTY (continued)

REQUEST FOR ADDITIONAL INFORMATION

- *No sequences beyond those identified for AP600 are classified as "low T/H margin risk significant" on the grounds that the two designs are similar*

- *The staff requested the use of a systematic approach and/or additional analyses, as was done for AP600, to support this argument:*

  - *Differences in T/H parameters (e.g., decay heat and flow rates) can affect plant response for PRA scenarios involving multiple failures and potential system interactions*

  - *Several PRA changes (e.g., IE categories and frequencies, and success criteria) could have changed the risk significance of a sequence*

STATUS

- *Response includes requested systematic approach (under staff review)*

AP1000 PRA REVIEW -- FIRE-INDUCED SPURIOUS ACTUATIONS

ISSUE DESCRIPTION AND RELATED AP600 BACKGROUND

- *AP600 at-power fire CDF is dominated (85% or about 6.5E-7/yr) by fire-induced spurious actuation of ADS explosive valves (EVs) leading to medium LOCA*
- *In AP600 the significant uncertainty in "hot short" probability was addressed by a sensitivity study and design certification requirements*
- *Design features that prevent fire-induced detonation of EVs, such as*
  - *Use controller circuit requiring multiple shorts for actuation*
  - *Routing ADS cables in low voltage cable trays and using redundant series controllers located in separate cabinets*
  - *Provisions for operator action to remove power from the fire zone*
- *Information since AP600 certification indicates that "hot shorts" may not always be independent events and that cable-to-cable interactions cannot be excluded*

AP1000 PRA REVIEW -- FIRE-INDUCED SPURIOUS ACTUATIONS (continued)

REQUEST FOR ADDITIONAL INFORMATION

- *Hot shorts are assumed to be independent events in the AP1000 fire PRA and no cable-to-cable interactions were considered*

- *Studies since AP600 certification (SANDIA, EPRI) indicate that spurious actuations from cable-to-cable interactions (conductors from separate cables could come into close proximity to each other) are credible and likely for some cable types*

- *If ADS cables are routed in same cable tray or a common enclosure:*

  - *Analyze the effect of cable-to-cable interactions*
  - *Assess need for additional design features, beyond AP600, to prevent fire-induced detonation of EVs*

REVIEW STATUS: The staff is interacting with Westinghouse to resolve this issue

AP1000 PRA REVIEW -- CERTIFICATION REQUIREMENTS

ISSUE DESCRIPTION

An important objective of the AP1000 PRA review is to use PRA results and insights to identify "certification requirements"

- *Identify important safety insights, related to design features and assumptions made in the PRA, and use such insights to support "certification requirements," such as ITAACs, TS, D-RAP and COL action items*

- *Support the process used to determine appropriate regulatory treatment of non-safety systems (RTNSS)*

The identification of "certification requirements" requires integrated input from uncertainty, importance and sensitivity studies

AP1000 PRA REVIEW -- CERTIFICATION REQUIREMENTS (continued)

REQUEST FOR ADDITIONAL INFORMATION

- *The staff requested information, similar to what was provided for AP600, showing how PRA results and insights are used to identify "certification requirements" as well as a list of the identified requirements*

- *Differences in "certification requirements" between AP1000 and AP600 result primarily from design differences*

- *Several outstanding issues have the potential, individually or collectively, to affect PRA results and change "certification requirements" with respect to AP600, such as RTNSS. Examples of such issues are:*

  - *Initiating event frequency changes (e.g., LOCAs, SGTR, PRHR-TR)*
  - *Late containment failure modeling issue*
  - *Common cause failure probability of explosive (squib) valves*

REVIEW STATUS: Response under review

NRC STAFF REVIEW OF THERMAL/HYDRAULIC BASIS FOR AP1000 PRA

ACRS Subcommittee on Reliability and Probabilistic Risk Assessment
Meeting January 23, 2003

Walton Jensen
NRR/DSSA/SRXB

Minimum equipment requirements to prevent CD identified by Westinghouse using MAAP4

- MAAP4 used for scoping analyses to identify the limiting events trees.
- MAAP4 has not been submitted for NRC staff review.
- MAAP4 was benchmarked against Westinghouse licensing codes for AP600.
- MAAP4 results differed from those of the licensing codes because of simplifying assumption in MAAP4.
- Overall conclusions for core cooling were similar for AP600.
- Staff has requested justification that AP600 benchmarks using MAAP4 are valid for AP1000.

Minimum success paths (low margin) identified by MAAP4 are verified by bounding analyses using licensing codes

- WCOBRA/TRAC - LBLOCA and LT Cooling
- NOTRUMP - SBLOCA
- WGOTHIC - Containment

Bounding analyses are performed by Westinghouse in lieu of uncertainty analyses for the T/H parameters. Westinghouse used the same approach for AP600.

NRC staff believes all limiting success paths accepted as the basis for successful core cooling using MAAP4 should be verified using licensing codes.

Westinghouse believes that only success paths with significant risk need to be verified.

Staff is reviewing the risk significance of the unbounded success paths to determine their effect on PRA conclusions.

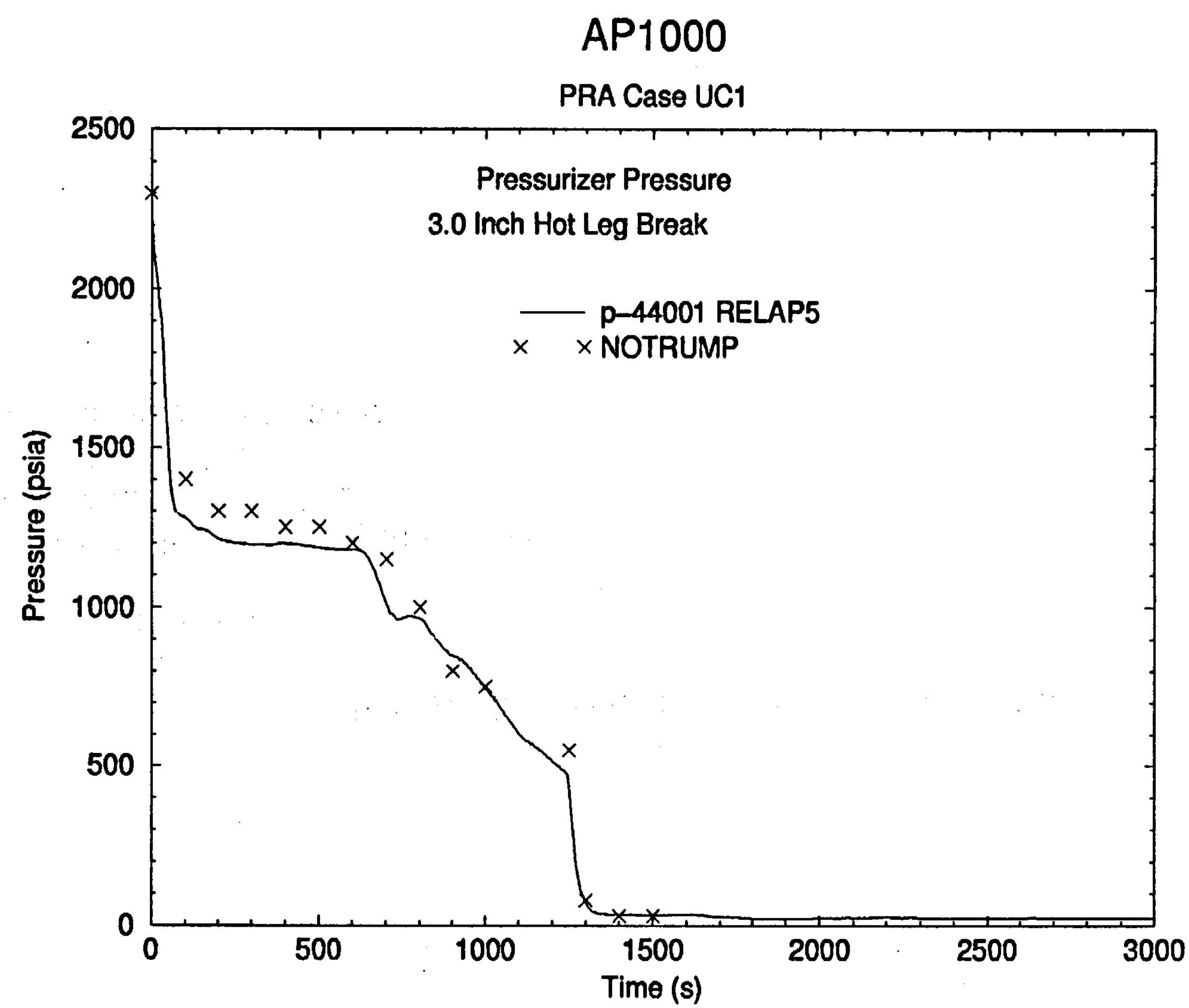
Staff Audit Calculations using RELAP5

Comparisons with NOTRUMP

- Uncertainty Case UC1 3.25 inch HLB assumes the following failures
  - Both CMT > Manual ADS-4 actuation
  - 1 of 2 accumulators
  - All ADS-1,2,3
  - 1 of 2 IRWST line

- Uncertainty Case UC3 DEDVI assumes the following failures
  - 1 of 2 CMTs > Automatic ADS-4 operation
  - Both accumulators
  - All ADS-1,2,3
  - 1 of 4 ADS-4
  - 1 of 2 IRWST Line
  - Containment isolation failed *

* Analysis extended only to initial IRWST injection. Long term cooling was not investigated



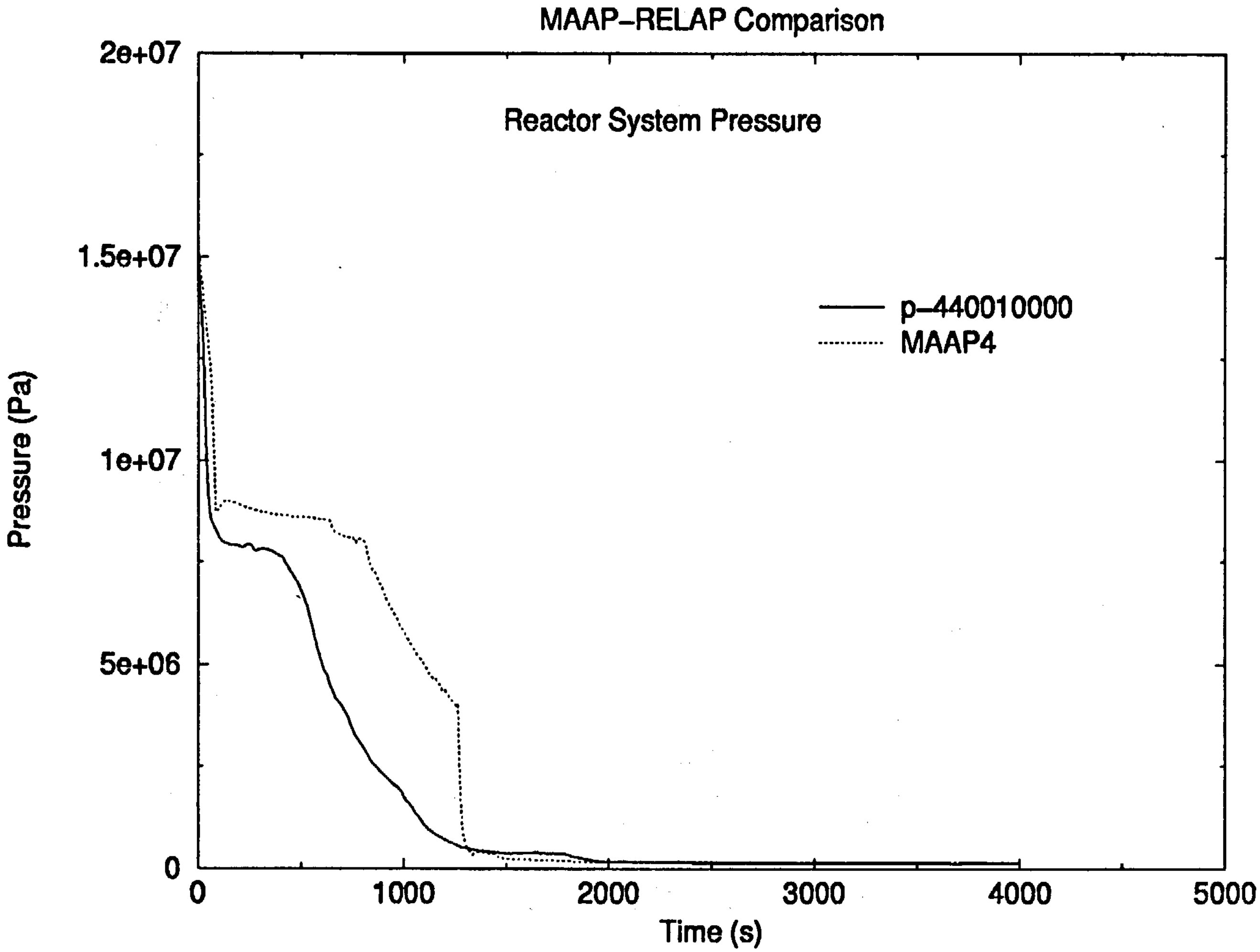
AP1000
PRA Case UC1
Pressurizer Pressure
3.0 Inch Hot Leg Break
p–44001 RELAP5
NOTRUMP
Pressure (psia)
Time (s)
2500
2000
1500
1000
500
0
0
500
1000
1500
2000
2500
3000

Staff Audit Calculations using RELAP5 (Cont.)

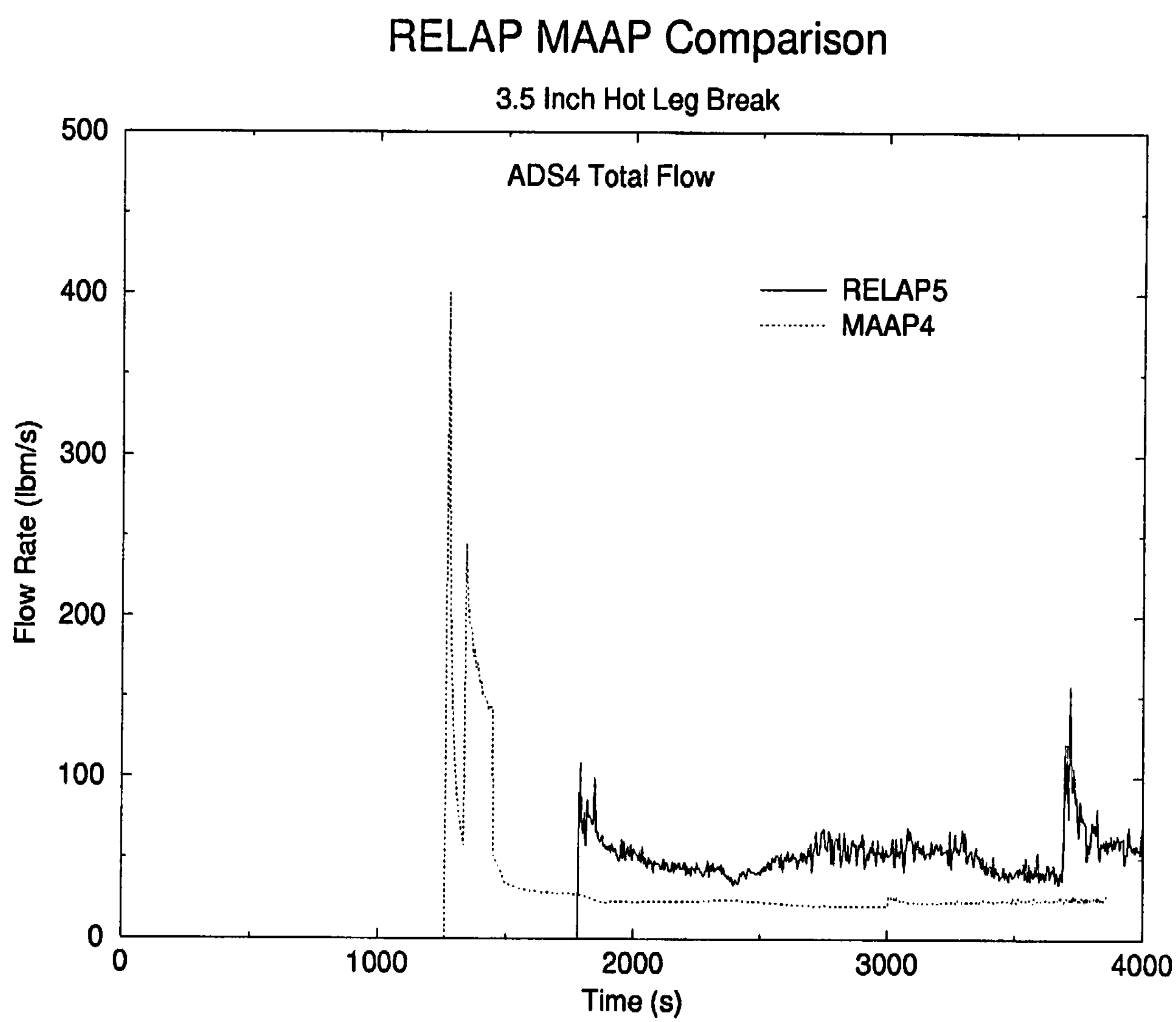
Comparison with MAAP4

- 3.50 inch HLB assuming the following failures
  - 1 of 2 accumulators
  - 1 of 4 ADS-4
  - containment isolation failure

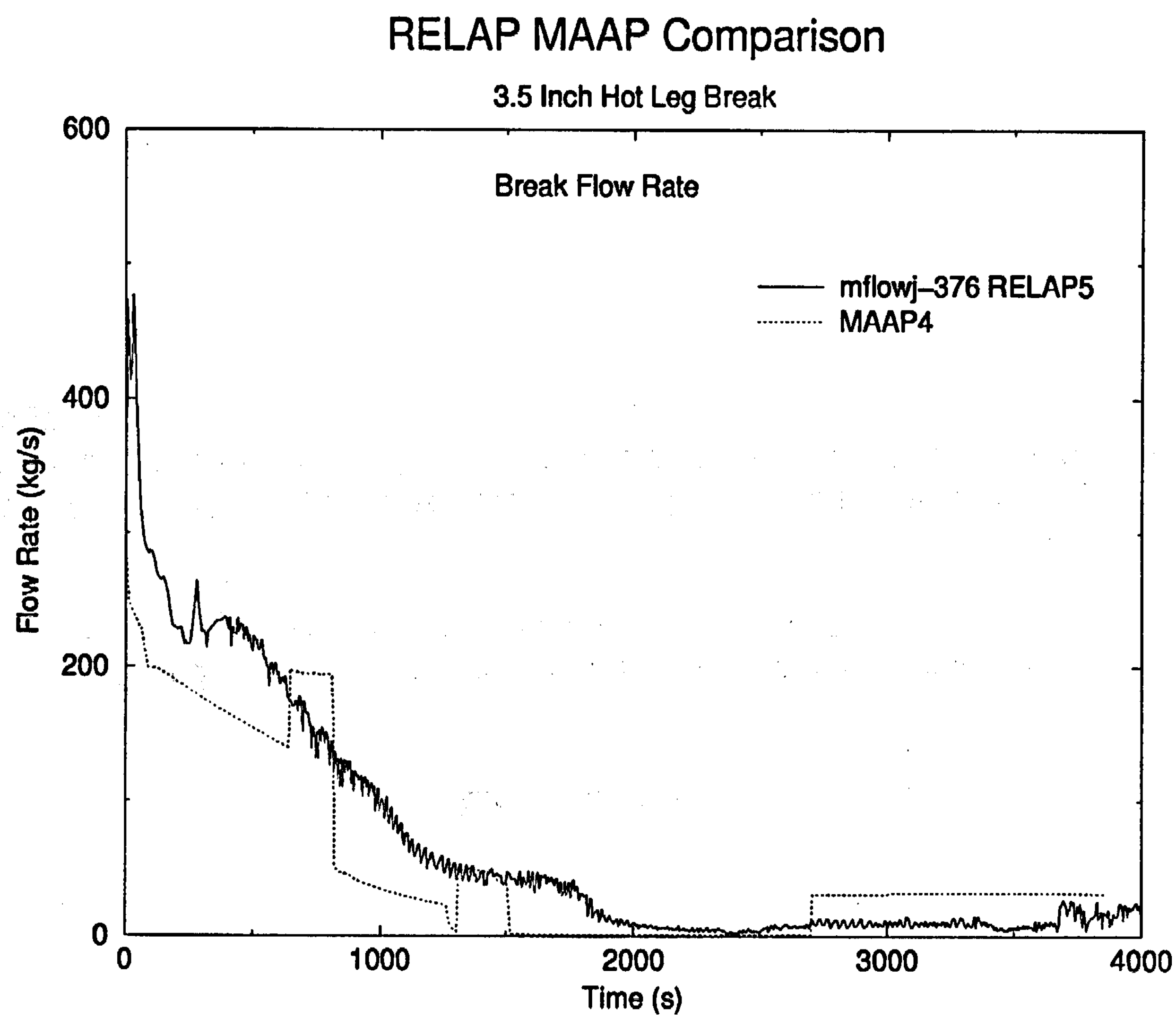
AP1000



MAAP–RELAP Comparison
Reactor System Pressure
Pressure (Pa)
2e+07
1.5e+07
1e+07
5e+06
0
Time (s)
0
1000
2000
3000
4000
5000
p–440010000
MAAP4



RELAP MAAP Comparison
3.5 Inch Hot Leg Break
ADS4 Total Flow
RELAP5
MAAP4
Flow Rate (lbm/s)
Time (s)
0
100
200
300
400
500
1000
2000
3000
4000



RELAP MAAP Comparison
3.5 Inch Hot Leg Break
Break Flow Rate
mflowj–376 RELAP5
MAAP4
Flow Rate (kg/s)
Time (s)
600
400
200
0
0
1000
2000
3000
4000

Audit Calculation Conclusions

- NOTRUMP and RELAP5 show the same general trends of reactor system response for the two cases analyzed. Both codes predicted brief periods of core uncovery which were within acceptable limits.

- MAAP4 and RELAP5 predicted different trends of pressure, break flow and ADS4 flow. Although the results were different both codes predicted the core to remain covered and cooled for the case analyzed.

Examples of Unresolved issues

- PRA Appendix A Section A3.3.1indicates success in long term cooling for 3 of 4 ADS4 and Containment Failure. WCOBRA/TRAC analysis were done for 3 of 4 ADS4 without containment failure and for 4 of 4 ADS4 with containment failure.

- PRA Section 6.3.1.5 indicates that sufficient water will be retained within the containment for long term cooling even if containment isolation fails. This conclusion has not been verified.

- AP600 analyses have been utilized to justify many of the success paths for AP1000. These need to be shown to be applicable for AP1000.

- NRC staff is reviewing the risk significance of the minimum success paths which Westinghouse has not bounded by analyses using licensing codes.

AP1000

AP1000 Design Certification Review

Westinghouse Electric Company

Presentation to

Advisory Committee on Reactor Safeguards

PRA Sub-Committee

January 23 -24 2003

BNFL Westinghouse

AP1000

Agenda

Thursday January 23, 2003

- **Introduction** — George Apostolakis, ACRS — 8:30 am
  - Review goals and meeting objectives
- **Westinghouse Introduction** — Mike Corletti, Westinghouse — 8:35 am
- **Overview of AP1000 Design** — Terry Schulz, Westinghouse — 8:40 am
  - Design Changes from AP600
  - Key AP1000 Design Features
  - Defense-in-Depth
  - PRA as a Design Tool
- **BREAK** — 10:05 am
- **AP1000 PRA** — Selim Sancaktar, Westinghouse — 10:20 am
  - Background / Approach / Overview
  - Scope
  - Level 1 PRA Internal Events At-Power
  - Sensitivity and Uncertainty Assessments
  - Shutdown / Fire PRA
- **LUNCH** — 12:20 - 1:30 pm

BNFL ACRS PRA Subcommittee - Jan 2003 Slide 2 Westinghouse

AP1000

Agenda

Thursday January 23, 2003

- **PRA Level 1 Success Criteria** — Terry Schulz, Westinghouse — 1:30 pm
  - Overview
  - Thermal-Hydraulic Analysis to Support Level 1 PRA
  - T&H Uncertainty Assessment
- **BREAK** — 3:30 pm
- **NRC Staff Presentation** — Nick Saltos - Walt Jensen - Mario Pohida — 3:45 pm
  - Staff RAIs on Level 1 PRA and Success Criteria
- **Westinghouse Summary** — Mike Corletti — 5:30 pm

BNFL ACRS PRA Subcommittee - Jan 2003 Slide 3 Westinghouse

AP1000

Agenda

Friday January 24, 2003

- **Introduction** — George Apostolakis, ACRS — 8:30 am
  - Review goals and meeting objectives
- **Level 2 and 3 PRA** — Jim Scobel, Westinghouse — 8:35 am
  - Quantification
- **Invessel Retention of Molten Core Debris**
- **BREAK** — 10:05 am
- **Level 2 Phenomenological Studies** — 10:30 am
- **Summary of PRA Results and Insights** — Selim Sancaktar, Westinghouse — 11:45 am
- **LUNCH** — 12:15 pm
- **NRC Staff Presentation** — Bob Palla, NRR - Richard Lee, RES — 1:15 pm
- **Westinghouse Summary** — Mike Corletti, Westinghouse — 2:15 pm
- **General Discussion** — ACRS Members — 2:30 pm
- **Adjourn** — 3:00 pm

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AP1000

Design Certification Schedule

Major Milestones

1. W Submits DCD Application (DCD / PRA)	3/28/02
2. Staff Issues RAI	9/30/02
3. W Provide Responses to All RAI	12/2/02
2. NRC Identify Potential DSER Open Items	2/28/03
4. W Addresses Potential DSER Open Items	4/15/03
5. NRC Issues DSER	6/16/03

W Goal is to Address All Open Items Prior to Issuance of DSER

6. ACRS Full Committee & Letter	7 / 2003

W OBJECTIVE IS TO PROVIDE THE NRC / ACRS WITH THE NECESSARY INFORMATION SO THAT A FINAL SAFETY DETERMINATION ON AP1000 CAN BE MADE IN 2003

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AP1000

W Objectives of the Meeting

- **Provide a Thorough Presentation of AP1000 PRA**
  - Level 1 / 2 / 3
  - Supporting T/H Analyses for Level 1
  - Supporting Phenomenological Studies for Level 2
- **Address All ACRS Issues Related to PRA**

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ACRS Meetings

AP1000

- **Overview to Full Committee** — **Nov. 7, 2002**
- **PRA Subcommittee** — **Jan. 23/24 2003**
- **Thermal-Hydraulic Subcommittee** — **March 2003**
  - Safety Analysis / Entrainment Issue
  - Containment cooling
- **AP1000 Subcommittee** — **April 2003**
  - Containment structural design
  - Materials
  - Regulatory Treatment of Non-Safety Systems
  - Shutdown Maintenance
- **ACRS Full Committee Meeting** — **June - July 2003**

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AP1000

Overview of AP1000 Design

Terry Schulz
Advisory Engineer
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AP600 to AP1000 Design Changes

AP1000

- **Increase Core Length & Number of Assemblies**
- **Increase Size of Key NSSS Components**
  - Increased height of Reactor Vessel
  - Larger Steam Generators (similar to W/CE SGs)
  - Larger canned RCPs (variable speed controller)
  - Larger Pressurizer
- **Increase Containment Height & Design Pressure**
- **Capacity Increases in Passive Safety System Components**
- **Turbine Island Capacity Increased for Power Rating**

Retained Nuclear Island Footprint

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Comparison of Selected Parameters

AP1000

Parameter	Doel 4/Tihange 3	AP600	AP1000
Net Electric Output, MWe	985	610	1117
Reactor Power, MWt	2988	1933	3400
Hot Leg Temperature, °F	626	600	610
Number of Fuel Assemblies	157	145	157
Type of Fuel Assembly	17x17	17x17	17x17
Active Fuel Length, ft	14	12	14
Linear Hear Rating, kw/ft	5.02	4.1	5.71
Control Rods / Gray Rods	52 / 0	45 / 16	53 / 16
R/V I.D., inches	157	157	157
Vessel flow (Thermal Design)	295,500	194,200	300,000
Steam Generator Surface Area, ft2	68,000	75,000	125,000
Pressurizer Volume, ft3	1400	1600	2100

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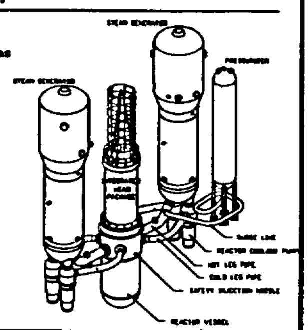
AP1000 Major Components

AP1000

- **Fuel, Internals, Reactor Vessel**
  - Similar to Doel 4, Tihange 3, S. Texas
  - No bottom-mounted instrumentation
  - Use core shroud ala W/CE plants
  - Improved materials - 60 yr life
- **Steam Generators**
  - Features from W SGs in operation
  - Size from W/CE SGs in operation
- **Reactor Coolant Pumps**
  - Canned motor pumps
    - Naval reactors, early commercial reactors, AP600
- **Simplified Main Loop**
  - Same as AP600
  - Reduces welds 50%, supports 80%
- **Pressurizer**
  - 50% larger than operating plants



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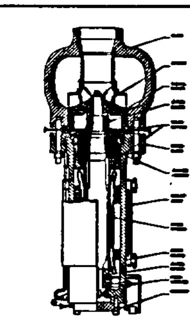
AP1000 Reactor Coolant Pump

AP1000

- **Based on Field-Proven, Canned Motor Pumps**
  - 1300 units in service
  - 12-year mean time between repair
  - No shaft seals
    - No seal injection / leakoff system
    - No seal leakage / failure
  - Water lubricated bearings
    - No oil lubricating / cooling system
  - Compact, high inertia flywheel
  - AP600 pump tests performed
    - Full size test of compact flywheel
    - Scaled hydraulics tests
    - Air-mixing tests of SG / RCP connection



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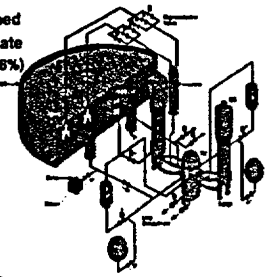
AP1000 Approach to Safety

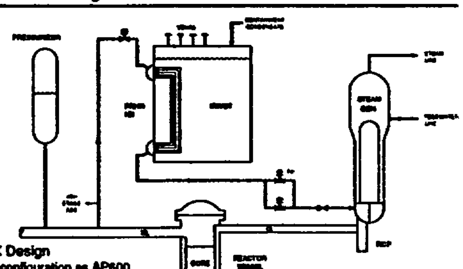
- **Passive Safety-Related Systems**
  - Use "passive" process only, no active pumps, diesels, ...
    - One time alignment of valves
    - No support systems required after actuation
      - No AC power, cooling water, HVAC, I&C
  - Greatly reduced dependency on operator actions
  - Mitigate design basis accidents without nonsafety systems
  - Meet NRC PRA safety goals without use of nonsafety systems
- **Active Nonsafety-Related Systems**
  - Reliably support normal operation
    - Redundant equipment powered by onsite diesels
  - Minimize challenges to passive safety systems
  - Not required to mitigate design basis accidents

AP1000 Passive Core Cooling System

- **AP600 System Configuration Retained**
- **Capacities Increased to Accommodate Higher Power (1933MW - 3400MW or 76%)**
  - PRHR HX Capacity Increased 72%
  - CMT Volume & Flow Increased 25%
  - ADS 4 Flow Increased 93%
  - IRWST Injection Increased 89%
  - Containment Recirc. Increased 139%
- **System Performance Maintained**
  - No core uncovery for SBLOCA
    - ≤ DVI line break
    - Large margin to PCT limit
  - No operator actions required for SGTR



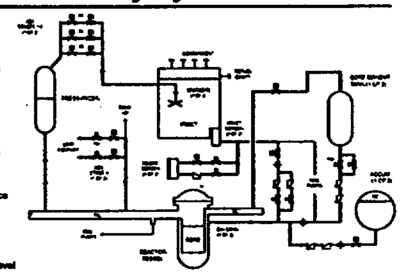
Passive Decay Heat Removal



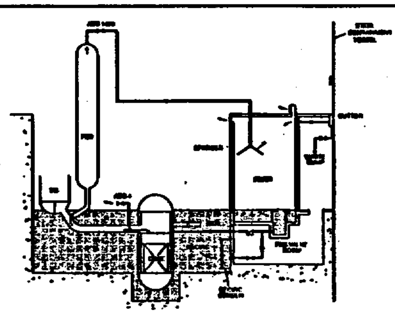
- PRHR HX Design
  - Same configuration as AP600
  - Same elevations as AP600
  - Larger pipe / valve sizes (14" vs 10")
  - Increased HX surface (more tubes / longer horizontal section)

AP1000 Passive Safety Injection

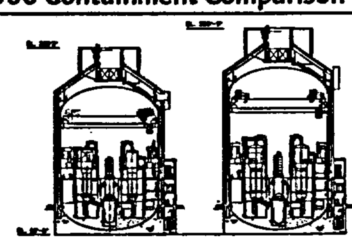
Passive Safety Injection
- Same configuration as AP600
- Same elevations as AP600
- Same Accum capacity
- Increased CMT capacity
  - 25% larger tank
  - 25% more flow
  - Same pipe, larger orifice
- Larger IRWST lines
  - 8" vs 6"
- Larger Recirc lines
  - 8" vs 6"
  - Increased cont. flood level
- Same ADS 1/2/3 lines
- Larger ADS 4 lines
  - 14" vs 10"



LOCA Long Term Cooling

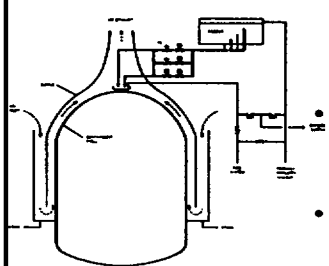


AP1000 Containment Comparison



	AP600	AP1000
Total Free Volume	100%	122%
Design Pressure, psig	45	59
Shell Thickness	1 5/8"	1 3/4"
Material	A537 Class 2	SA738 Grade B

Passive Containment Cooling System



- **PCS Water Storage Tank**
  - Provides 72 hr drain
    - Afterwards use on/offsite water
    - Air only cooling prevents failure
  - Flow decreases with time
    - Uses 4 standpipes
- **PCS Flow Rates**
  - High initial flow
    - Rapidly forms water film
    - Effectively reduces cont pressure
  - Later flows match decay heat
- **Added 3rd Diverse Drain Path**
  - Adds PRA margin
    - T&H uncertainty of cont cooling without water drain

AP1000 Safety Margins

	Typical Plant	AP600	AP1000
- Loss Flow Margin to DNBR Limit	~ 1 - 5%	~16%	~19%
- Feedline Break Subcooling Margin	>0°F	~170°F	~140°F
- SG Tube Rupture	Operator actions required in 10 min	Operator actions NOT required	Operator actions NOT required
- Small LOCA	3" LOCA core uncovers PCT ~1500°F	< 8" LOCA NO core uncovery	< 8" LOCA NO core uncovery
- Large LOCA PCT (with uncertainty)	2000 - 2200°F	1676°F	2124°F

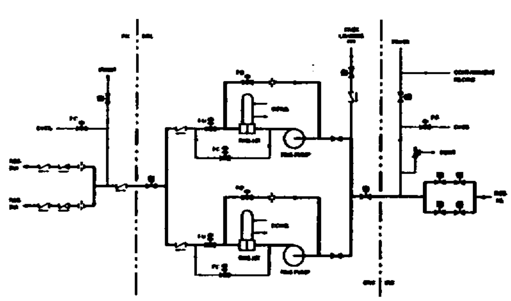
AP1000 Hydrogen Mitigation

- **Design Basis Accidents**
  - Slow long term buildup of H2
  - Uses 2 full size Passive Autocatalytic Recombiners (nonsafety)
    - No power or actuation required
  - Equipment is non-safety based on NRC / industry activities on risk-informed changes to 10 CFR 50.44 (Combustible Gas Control)
- **Severe Accidents**
  - Rapid buildup of H2
  - Uses non-safety igniters distributed in pairs around containment
  - Release paths from RCS ensure standing H2 flames located away from containment walls
    - IRWST vents changed to discharge H2 away from containment wall

AP1000 Active Nonsafety Systems

- **Active Nonsafety System Functions**
  - Reliably support normal operation
  - Minimize challenge to passive safety systems
  - Not required to mitigate design basis accidents
  - Not required to meet NRC safety goals
- **Active Nonsafety System Design Features**
  - Simplified designs (fewer components, separation not required)
  - Redundancy for more probable failures
  - Automatic actuation with power from onsite diesels
- **Active Nonsafety System Equipment Design**
  - Reliable, experienced based, industrial grade equipment
  - Non-ASME, non-seismic, limited fire / flood / wind protection
  - Availability controlled by procedures, no shutdown requirements
  - Reliability controlled by maintenance program

AP1000 Normal RHR System

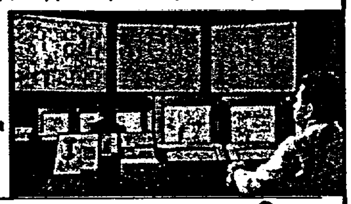


AP1000 I&C Systems

- **Control System (PLS/DDS)**
  - Plant wide non-1E system for all normal displays & controls
  - Microprocessor / software based, multiplexed communications
- **Safety System (PMS)**
  - Plant wide 1E system for all safety displays & controls
  - Microprocessor / software based, multiplexed communications
- **Diverse System (DAS)**
  - Limited scope non-1E system, PRA based displays & controls
    - Backs up PMS where common mode failure is risk important
  - Different hardware & software than PMS, no multiplexing
  - Separate sensors from PMS and PLS

AP1000 Advanced Control Room

- Compact Control Room
  - Designed for 1 Reactor Operator and 1 Supervisor
- Displays
  - Plant status / overview via wall panel (DDS, non 1E)
  - Detail display via workstation video displays (DDS, non 1E)
  - Small number dedicated displays; safety (PMS, 1E) & diverse (DAS, non 1E)
- Controls
  - Soft controls (DDS, non 1E) for normal operation
  - Small number dedicated switches; safety (PMS, 1E) & diverse (DAS, non 1E)
- Advanced Alarm Management
- Computer Based Procedures



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PMS Reliability Features

- Redundant Trains
  - 4 divisions, physically separated with improved isolation (fiber-optic)
    - Each with own independent battery-backed power supply
  - 2 out of 4 bypass logic, fail safe when appropriate
  - Different plant parameters provide functional diversity
- Extensive Verification and Validation
- Extensive Equipment Qualification
  - Environmental, seismic, EMC
- Improved In-Plant Testing
  - Built-in continuous self-testing and manual periodic testing
- West. Extensive Experience with Digital I&C Designs
  - Operating plant upgrades and new plants (Sizewell, Temelin)

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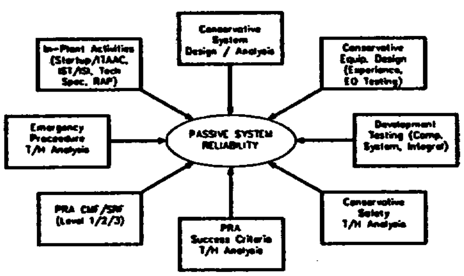
AP1000 System Reliability



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System Defense In Depth

- AP1000 Provides Multiple Levels of Defense
  - First feature is usually nonsafety active feature
    - High quality industrial grade equipment
  - One feature is safety passive feature
    - Provides safety case for DCD
    - Highest quality nuclear grade equipment
  - Other passive features provide additional defense-in-depth
    - Example; passive feed/bleed backs up PRHR HX
  - Available for all shutdown conditions as well as at power
  - More likely events have more levels of defense

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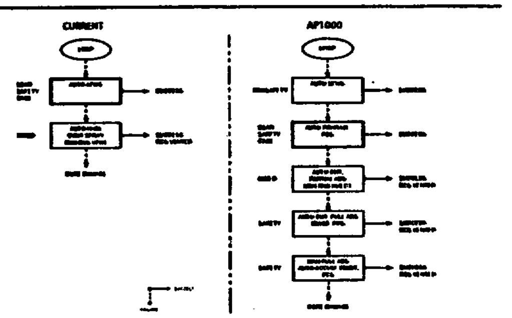
Loss of Offsite Power



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Loss Offsite Power, at Power

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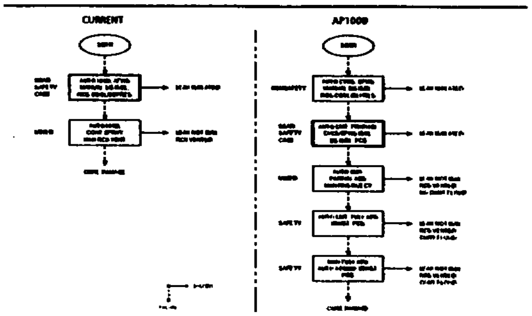
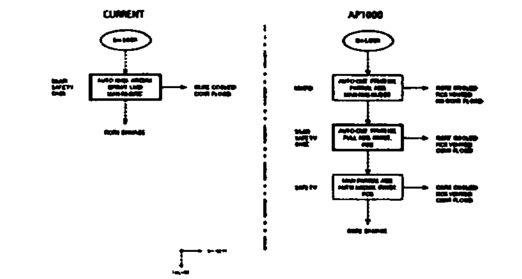
SG Tube Rupture

AP1000



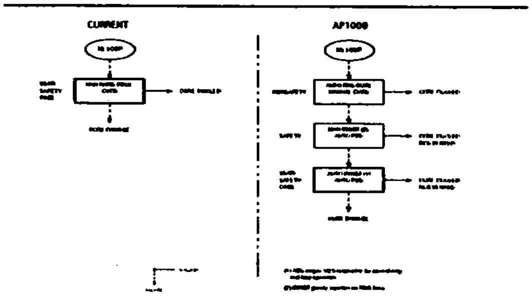
Small LOCA

AP1000



Mid-Loop Loss Power

AP1000



AP1000 PRA

AP1000

- **Westinghouse Uses PRA as Design & Licensing Tool**
  - 7 PRA major quantifications performed on AP600
    - First in 1987, final in 1997
    - Extensive interaction with plant designers
    - Extensive NRC review / comment
  - AP1000 PRA quantified in 2001
    - Started with AP600 models / analysis
  - Plant designers interact with risk analysis
    - Results reviewed, improvements made (more in AP600)
      - PRA analysis models and supporting T/H analysis
      - Plant operating procedures
      - Plant design

PRA Based Changes (AP600)

AP1000

- **Analysis Changes**
  - Accum or CMT sufficient for small / medium LOCA
  - One accum sufficient for large LOCA
  - Multiple ADS valve failures acceptable
- **Operation Changes**
  - Manually start RNS after ADS actuation
  - Require containment closure capability during mid-loop
  - Require PXS features to be available during shutdowns

PRA Based Changes (AP600)

AP1000

- **Design Changes**
  - RNS alignment valves made remote
  - 4th stage ADS valves made diverse from stages 1, 2, 3
  - Added DAS functions
  - Added redundant IRWST injection check valves
  - Added redundant / diverse IRWST recirc valves
  - Made CMT check valves normally open, diverse from accum
  - Provided logic for automatic SGTR protection without ADS

PRA Based Changes (AP1000)

- **AP1000 Analysis Changes**
  - Initiating event frequency changes
    - Larger SGs (more, longer tubes)
    - Increased number SG safety valves
    - Separated spurious ADS stage 4 and large CL LOCA
      - 2 / 2 accum required for CL LOCA, 1/2 accum required for spur ADS 4
  - PRHR HX operation needed for MLOCA without CMTs
    - Provides operators sufficient time for manual ADS
- **AP1000 Operation Changes**
  - Containment recirc MOV normally open (in series with squib valve)
  - Changed IRWST drain proceedure so it occurs earlier in core melt
  - Added Tech Spec on DAS manual controls

PRA Based Changes (AP1000)

- **AP1000 Design Changes**
  - Increased volume and injection rate of CMTs
  - Added 3rd Passive Cont. Cooling drain valve, MOV diverse to AOV
  - Incorporated low boron core, improves ATWT
  - RNS injection water supply changed from IRWST to Cask Load Pit
  - Improved IVR heat transfer via changes to RV insulation gap
  - Improved H2 vents from IRWST to keep H2 flames away from cont.

AP1000 Probabilistic Risk Assessment

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Fellow Engineer, Reliability and Risk Assessment
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OBJECTIVES

- **The purpose of the AP1000 PRA is to provide inputs to the optimization of the AP1000 design and to verify that the US NRC PRA safety goals have been satisfied**
- **As in the AP600, the PRA is being performed interactively with the design, analysis and operating procedures.**

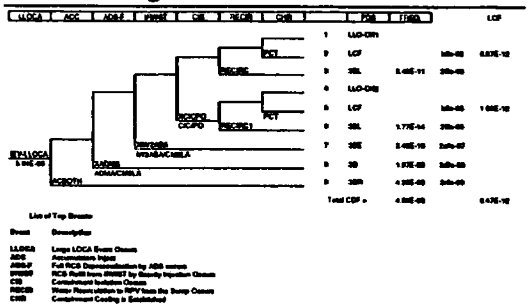
TECHNICAL SCOPE

- **Since the configuration of the AP1000 reactor and safety systems is the same as the AP600, the AP600 PRA is used as the basis of the AP1000 PRA with relevant changes implemented in the model to reflect the AP1000 design changes**

TECHNICAL SCOPE

- **AP1000 plant-specific T&H analyses are performed in order to determine the system success criteria**
- **The CDF and LRF are calculated for internal events at-power. The off-site dose risk analysis is also performed. The external events and shutdown models are also assessed to derive plant insights and plant risk conclusions.**

AP1000 Large LOCA Event Tree



AP1000 System Failure Probabilities

- Fault Tree Models are used to calculate system failure probabilities & identify minimal cutsets
- All support systems are modeled in detail
- Component random failures, human errors, tests and maintenance unavailabilities, and common cause are modeled based on standard industry practice

AP1000 PRA System Failure Probabilities

Contribution of Initiating Events to AP1000 CDF

	IEV CATEGORY	CONTRIBUTION TO CDF	CONTRIBUTION TO CDF (%)	IEV FREQUENCY	CONDITIONAL CDF
1	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2	EV-LLOCA	[illegible]	[illegible]	[illegible]	[illegible]
3	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
4	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
5	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
6	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
7	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
8	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
9	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
10	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
11	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
12	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
13	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
14	[illegible]	[illegible]	0.4	[illegible]	[illegible]
15	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
16	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
17	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
18	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
19	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
20	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
21	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
22	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
23	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
24	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
25	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
26	[illegible]	[illegible]	0	[illegible]	[illegible]
	Totals	[illegible]	100.0	[illegible]	

Comparison of AP600 and AP1000 CDF

AP1000 PRA Dominant CDF Sequences

AP1000 PRA Dominant CDF Sequences

[illegible]

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Summary of Sensitivity Analysis Results

Case Description	Results
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]
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[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]

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AP1000 PRA System Importances

AP1000 SYSTEM IMPORTANCES BY RAW (CDF) INCREASE			
PMS DC-1E IRWST-REC ADS IRWST-INJ	CMT ACC PRHR	DC-Non 1E DAS AC PLS	CAS N-RHR SWS CCS SFW DG MFW SG Overfill Protection
> 200	50-200	2-50	<2

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AP1000 PRA System Importances

[illegible]

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Importance of PMS and DC-1E Systems

- PMS and DC-1E are the most important systems (by risk increase measure)
- PMS is very reliable and redundant; its reliability is only limited by postulated CCF (such as CCF software).
- In case of a total postulated failure of PMS, the plant relies on DAS (auto or manual) and control systems (only for some transients); in this scenario, the plant CDF goes up by orders of magnitude

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Sensitivity Analyses Results

- The component, operator action, and system importance analyses provide us input for other AP1000 programs (such as RTNSS, reliability assurance program)
- The sensitivity analyses increase our confidence in the stability of PRA numerical results.

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UNCERTAINTY ANALYSIS

- The plant CDF uncertainty range is found to be 7.3 E-07 – 2.1 E-08 for the 95% to 05 % interval
- For a lognormal distribution, this would correspond to an error factor of 6, which can be considered as low for rare events

UNCERTAINTY ANALYSIS

- The mean values of the dominant accident sequence frequencies are close to the upper bound (95%) estimates;
- Among the initiating event categories, SI-LB has the highest 95-percentile CDF of 3.2E-07 /year.
- Among the dominant sequences, sequence # 07 of SI-LB event has the highest 95-percentile CDF of 2.1E-07/yr.

SHUTDOWN EVENTS

- A quantitative shutdown risk evaluation is performed for AP1000 for internal events
- The risk profiles of AP1000 and AP600 for events during shutdown conditions are almost identical
- The AP1000 Shutdown PRA has a CDF of 1.23E-07 events per year. This CDF is an 18% increase of the AP600 Level 1 Shutdown CDF of 1.04E-07 events per year

SHUTDOWN EVENTS

- The three events dominating the CDF for each plant are loss of component cooling / service water during drained condition, loss of offsite power during drained condition, and loss of RNS during drained condition
- The initiating event CDF contributions show that the initiating event importance to be similar for the two plants

SHUTDOWN EVENTS

- The twelve dominant accident sequences comprise 77 percent of the level 1 shutdown CDF. They consist of:
  - Loss of component cooling or service water system initiating event during drained condition with a contribution of 64 percent of the CDF

SHUTDOWN EVENTS

- Loss of RNS initiating event during drained condition with a contribution of 6 percent of the CDF
- Loss of offsite power initiating event during drained condition with a contribution of 5 percent of the CDF
- RCS overdraining event during drainage to mid-loop with a contribution of a 2 percent of the CDF.

INTERNAL FLOODING AND FIRE

- The internal flooding-induced CDF is estimated to be 8.8E-10 events per year for power operations
- The CDF from flooding events at power is not an appreciable contributor to the overall AP1000 plant CDF

INTERNAL FLOODING AND FIRE

- The top five at-power flooding scenarios comprise 91 percent of the at-power flooding-induced core damage frequency
- These scenarios are for large pipe breaks in the turbine building with an initiating event frequency in the range of 1.4 – 2.0 E-03 / year, leading to a loss of CCW/SW event
  - Each scenario has a CDF of 1.2 – 1.8E-10/year.

INTERNAL FLOODING AND FIRE

- Extensive fire hazards analysis review completed for AP600 subsequent to fire AP600 PRA
  - Fire separation improved
  - Fire suppression features incorporated
  - Design features incorporated to address hot-shorts
- AP1000-specific Fire PRA is performed with a resulting CDF of 5.61E-08/yr (for internal events)

INTERNAL FLOODING AND FIRE

- AP600 design features important for fire protection are included in the AP1000
  - Fire separation / fire zones
  - Systems used to achieve safe shutdown
  - Fire suppression features
- AP1000 design is sufficiently robust that internal fires during power operation or shutdown do not represent a significant contribution to plant CDF

SEISMIC MARGINS EVALUATION

- The seismic margin analysis shows the systems, structures, and components required for safe shutdown. HCLPF values are greater than or equal to 0.50g
- This HCLPF is determined by the seismically induced failure of the fuel in the reactor vessel, core assembly failures, IRWST failure, or containment interior failures

SEISMIC MARGINS EVALUATION

- The SMA result assumes no credit for operator actions at the 0.50g review level earthquake, and assumes a loss of offsite power for all sequences
- The SMA shows the plant to be robust against seismic event sequences that contain station blackout coupled with other seismic or random failures
- AP1000 structural design and seismic analysis will be discussed at a future ACRS meeting

Comparison of Low HCLPF SSCs in AP1000 and AP600 Designs

Basic Event ID	Description	AP600 HCLPF	AP1000 HCLPF
EQ-CER-INSULATOR	Failure of Ceramic Insulators	0.09g	.09g
EQ-CORE-ASSEMBLY	Core Assembly Failure (not fuel)	0.50g	.50g
EQ-CV-INTER	Interior Containment	0.60g	.50g
EQ-IRWST-TANK	IRWST Failure	0.60g	.50g
EQ-RV-FUEL	Fuel Failure	0.50g	.50g
EQ-AB-EXTWALL	Aux. Building Exterior wall	0.56g	.51g
EQ-AB-FLOOR	Aux. Building Floor	0.56g	.51g
EQ-AB-INTWALL	Aux. Building Interior wall	0.56g	.51g
EQ-PCC-TANK	PCC Tank Failure	0.56g	.51g
EQ-SHDBLD-ROOF	Shield Building Roof	0.56g	.51g
EQ-SHDBLD-WALL	Shield Building Wall	0.56g	.51g
EQ-CABLETRAY	Cable trays - support controlled	0.54g	.54g
EQ-CMT-TANKS	Tank PXS 2A/B (Core Makeup Tank)	0.63g	.54g
EQ-SG-FAILS	Steam Generator Fails	0.65g	.54g
EQ-SGTR	Steam Generator Piping (one or a few)	0.65g	.54g
EQ-ACDISPANEL	120 vac distribution panel	0.51g	.56g
EQ-DC-SWBRD	125 vdc switchboard	0.51g	.56g
EQ-DCDISPANEL	125 vdc distribution panel	0.51g	.56g
EQ-PRZR-FAILS	Pressurizer Fails	0.67g	.56g
EQ-TRSFSWITCH	Transfer switch	0.51g	.56g

Comparison of AP600 and AP1000 PRA Results

Scope	AP600	AP1000
Level 1 At-Power Internal Initiating Events	Quantification Performed CDF = 1.7E-07 Several additional cases quantified in response to NRC RAIs	Quantification Performed CDF = 2.4E-07 AP600 additional cases incorporated into the model
Level 2 At-Power Internal Initiating Events	Quantification Performed LRF = 1.8E-09 Containment Effectiveness = 89.9%	Quantification Performed LRF = 2.0E-08 Containment Effectiveness = 91.8%
Level 3 At-Power Internal Initiating Events	Quantification Performed	Quantification Performed
Internal Fire Events	Conservative (no focused PRA) Quantification Performed CDF = 8.5E-07 (internal) CDF = 3.5E-07 (shutdown)	Quantification performed CDF = 5.61E-08
Internal Flooding Events	Quantification Performed CDF = 2.2E-10	Quantification Performed CDF = 8.8E-10
Shutdown Events	Quantification Performed to Level 1 and 2 CDF = 1.0E-07 LRF = 1.5 E-08 Several additional cases quantified in response to NRC RAIs	Quantitative Evaluation Performed CDF = 1.2E-07 LRF = 2.0E-08

SUMMARY OF RESULTS

- **The AP1000 PRA results show that**
  - The very low risk of the AP600 has been maintained in the AP1000
  - The AP1000 PRA meets the US NRC safety goals with significant margin

PRA Level 1 Success Criteria

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Overview

- **Success Criteria Justification**
  - Summary of success criteria (Chapter 6 of PRA)
    - Changes in success criteria vs AP600
  - Success criteria justification
    - Based on analysis - DCD, specific PRA, or other analysis / calculations
    - Summary of PRA analysis
      - Analysis results for small LOCA, large LOCA and ATWS
    - T&H Uncertainty Evaluations
      - Calc of low margin / risk important sequences
      - T&H analysis to bound T&H uncertainty

AP1000 Success Criteria

- **Similar to AP600**
  - Similar system design, arrangement, capabilities
  - Several Changes Made to the AP1000 Success Criteria
    - Due to increase in power and other factors
- **Verified Using Same Approach as AP600**
  - Use DCD analysis where applicable
  - Perform special analysis where DCD analysis not applicable
- **AP1000 Success Criteria More Conservative / Robust**
  - Uses same or more equipment for success than AP600
    - For example, uses 3/4 ADS 4 instead of 2/4 ADS 4 (AP600)
      - Even though AP1000 ADS 4 is larger / MW
  - Reduces T&H issues / uncertainty

Success Criteria Basis

- **Provides Critical Functions**
  - Decay heat removal (core cooling)
    - Peak clad temperature < 2200°F
  - RCS inventory control
  - RCS pressure control
    - Less than emergency stress limits, < 3200 psig
  - Containment heat removal and containment isolation
    - Less than emergency stress limits, < ??? psig
  - Reactivity control

AP1000 Full ADS Success Criteria

TABLE A2.3-1 FULL ADS SUCCESS CRITERIA(1)

Event	PRHR HX - on		PRHR HX - off	
	CMT - on	CMT - off	CMT - on	CMT - off
	Accum - off	Accum - on	Accum - off	Accum - on
RCS Transients, Loss of Power, Station Blackout	None (2)	None (2)	Auto 1/2 ADS stage 2/3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
RCS Leak	Auto 3/4 ADS stage 4	Man 3/4 ADS stage 4	Auto 1/4 ADS stage 2,3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
SGTR	None (3)	None (3)	Auto 1/4 ADS stage 2,3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
Small LOCA	Auto 3/4 ADS stage 4	Man 3/4 ADS stage 4	Auto 1/4 ADS stage 2,3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
Medium LOCA	Auto 3/4 ADS stage 4	Man 3/4 ADS stage 4	Auto 3/4 ADS stage 4	(7)
Spurious ADS	n/a		Auto 3/4 ADS stage 4 (5)	
Large LOCA	n/a		Auto 3/4 ADS stage 4 (6)	

Notes:
1. Automatic ADS actuation via PMS. ADS actuation can also be performed manually via PMS or DAS.
2. Successful PRHR HX operation eliminates need for ADS.
3. SGTR does not require ADS operation if PRHR HX operates and SGs are isolated.
4. Operation of PRHR HX has no effect on ADS success criteria, use "PRHR HX - off" success criteria.
5. Spurious ADS requires 1/2 accumulators and 1/2 CMT to work.
6. Large LOCA requires 2/2 accumulators and 1/2 CMT to work.
7. No credit is given for success for this event; the time available for operator action is short.

Post ADS Success Criteria

- **Changes Made to Post ADS Success Criteria**
  - Full ADS (IRWST) >> requires 3/4 ADS stage 4
    - AP600 PRA used 2/4 ADS stage 4
    - AP1000 ADS 4 capacity has been increased by more than power
  - Partial ADS (RNS) >> requires 2 of 4 ADS stage 2 or 3
    - AP600 PRA used 1/4 stage 2 or 3
    - ADS stages 1, 2, 3 capacities not increased for AP1000
  - Requires PRHR HX for MLOCAs with only Accum
    - Provides operators more time (> 20 min) to take action
  - Requires 2/4 Cont Recirc if Cont Isol fails
    - 1/4 Cont Recirc if Cont Isol works
  - Full ADS required for large LOCAs to support long term cooling

LOCA Size Definitions

- **Large LOCA (> 9" ID)**
  - Requires 2 of 2 accum
- **Spurious ADS Stage 4 (1 to 4 ADS 4 valves)**
  - Require 1 of 2 accum and 1 CMT
- **Medium LOCA, DVI LOCA, CMT Line LOCA (2-9" ID)**
  - Only requires 1 accum or 1 CMT
  - Depressure RCS below ADS 4 pressure interlock
- **Small LOCA (3/8-2" ID)**
  - Requires PRHR HX or ADS 1/2/3 todepressure RCS below ADS 4 pressure interlock
  - CVS makeup not sufficient
- **RCS Leak (< 3/8" ID)**
  - CVS makeup is sufficient

PRA Success Criteria Analysis

• Transient (PRHR HX)	DCD, LOFTRAN
• SGTR (PRHR HX)	DCD, LOFTRAN
• Non-LOCA Feed-Bleed	PRA, MAAP4
• LOCA (Small/Med. LOCA)	PRA, MAAP4
• LOCA (Lg LOCA)	PRA, WCOBRA-TRAC
• Spurious ADS 4 (Lg LOCA)	PRA, WCOBRA-TRAC
• ATWS	PRA, LOFTRAN

MAAP4 Code Use

- **Same Approach As AP600**
  - Used for defining success criteria for LOCAs and feed-bleed cooling sequences
    - Provides integrated RCS / containment response
    - Runs fast (hours vs days)
      - Important because of large numbers of runs (hundreds)
        - Break sizes, locations, different sets of multiple failures
    - MAAP4 has been bench marked against NOTRUMP for AP600
      - NOTRUMP has been shown to be applicable to AP1000
    - T&H uncertainty analysis confirms that low margin / risk important sequences will be success
      - Uses detailed DCD codes and methods (NOTRUMP, WCOBRA-TRAC)
- **AP1000 Success Criteria is More Robust**

PRA T&H Analysis

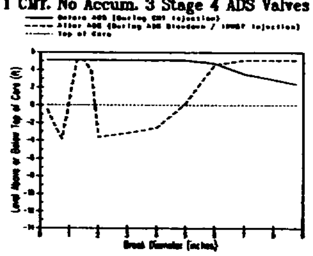
- **LOCAs and Feed-Bleed Cooling Analysis**
  - Considers many different factors
    - Initiating event, LOCA or Feed-Bleed Cooling after non-LOCA
    - LOCA size and location
    - Available mitigating equipment including CMT, Accum, RNS, PRHR HX, ADS, IRWST, Cont Recirc
  - Made use of lessons learned from AP600
    - Test results, DCD analysis, PRA analysis (both success criteria and T&H uncertainty
    - Divided into four groups of analysis
      1. Automatic ADS with CMT and IRWST gravity injection
      2. CMT and RNS pumped injection
      3. Manual ADS with Accum and IRWST gravity injection
      4. Accum and RNS pumped injection

1. Auto ADS with IRWST Gravity Injection

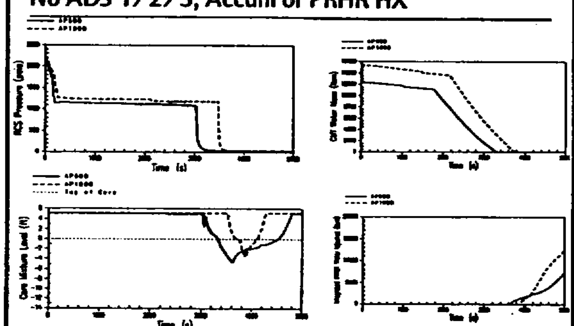
- **Limiting Success Criteria Equipment Assumed**
  - One CMT, no Accum, 1 valve path in one IRWST injection line
    - Same as AP600
  - 3/4 ADS stage 4 , no ADS stage 1/2/3, no PRHR HX
    - AP600 used 2/4 ADS 4
    - For LOCAs < 2" some ADS 1/2/3 or PRHR HX required to reduce RCS pressure to below ADS 4 pressure interlock
  - Containment isolation fails
- **MAAP4 Analysis Was Performed**
  - Break sizes 0.5" up to 8.75"
  - Core uncovery depth and duration is less than AP600
    - Increased capacity PXS, especially ADS 4 & IRWST injection
  - AP1000 success criteria verified

1. Auto ADS with IRWST Gravity Injection

AP1000 Minimum Vessel Mixture Level
Automatic ADS. IRWST Injection
1 CMT. No Accum. 3 Stage 4 ADS Valves



2" HL LOCA, 3/4 ADS4, 1 CMT, 1/1 IRWST No ADS 1/2/3, Accum or PRHR HX

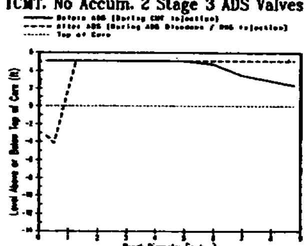


2. Auto ADS with RNS Injection

- **Limiting Success Criteria Equipment Assumed**
  - One CMT, no Accum, 1 RNS pump (SFP Cask Loading Pit)
  - 2/4 ADS stage 2/3, no ADS stage 4, no PRHR HX
    - AP600 used 1/4 ADS 2/3
  - Containment isolation fails
- **MAAP4 Analysis Was Performed**
  - Break sizes 0.5" up to 8.75"
  - Core uncovery depth and duration is less than AP600
  - AP1000 success criteria verified

2. Auto ADS with RNS Injection

AP1000 Minimum Vessel Mixture Level
Automatic ADS. RNS Injection
1CMT. No Accum. 2 Stage 3 ADS Valves



3. Manual ADS w. IRWST Gravity Injection

AP1000

- **Limiting Success Criteria Equipment Assumed**
  - One Accum, no CMT, PRHR HX, 1/1 valve / path IRWST injection
    - AP600 does not require PRHR HX, increases time for operator action
  - 3/4 ADS stage 4 , no ADS stage 1/2/3, no PRHR HX
    - ADS 4 manually actuated at 20 min.
    - AP600 uses 2/4 ADS 4
  - Containment isolation fails
- **MAAP4 Analysis Was Performed**
  - Break sizes 0.5" up to 8.75"
  - Core uncovery depth and duration is less than AP600
    - Increased capacity PXS, especially ADS 4 & IRWST injection
  - AP1000 success criteria verified

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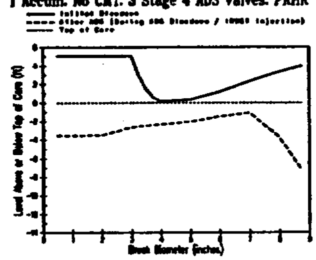
3. Manual ADS w. IRWST Gravity Injection

AP1000

AP1000 Minimum Vessel Mixture Level
Manual ADS at 20 Min. IRWST Injection
1 Accum. No CMT. 3 Stage 4 ADS Valves. PRHR



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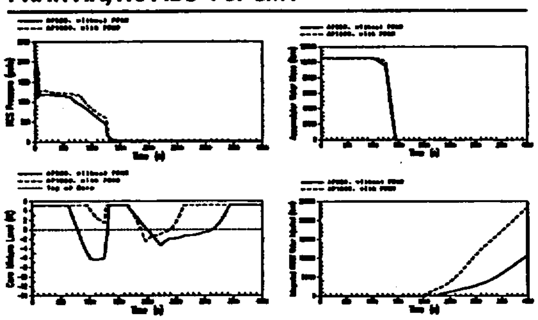
3.5" LOCA, 2/4 ADS 3, 1 Acc, 1/1 IRWST PRHR HX, No ADS 4 or CMT

AP1000



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4. Manual ADS with RNS Injection

AP1000

- **Limiting Success Criteria Equipment Assumed**
  - One Accum, no CMT, PRHR XH, 1 RNS pump (Cask Loading Pit)
  - 2/4 ADS stage 2/3, no ADS stage 4
    - ADS manually actuated at 20 min.
    - AP600 used 1/4 ADS 2/3
  - Containment isolation fails
- **MAAP4 Analysis Was Performed**
  - Break sizes 0.5" to 8.75"
  - AP1000 success criteria verified

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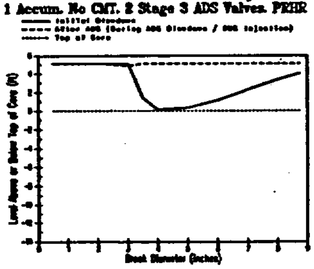
4. Manual ADS with RNS Injection

AP1000

AP1000 Minimum Vessel Mixture Level
Manual ADS at 20 Min. RNS Injection
1 Accum. No CMT. 2 Stage 3 ADS Valves. PRHR



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Large LOCA Success Criteria

AP1000

- **Large CL LOCAs**
  - Uses 2 of 2 Accum, like DCD analysis
  - Unlike DCD assumes failure of containment isolation and availability of offsite power
  - Was analyzed with WCOBRA-TRAC (RAI 720.012)
    - Calc PCT 1626 F without uncertainty
    - PCT less than DCD case because offsite power was available
- **Spurious ADS 4 Large LOCAs**
  - Limiting case is all four ADS 4 valves opening
  - Uses 1 of 2 Accum, failure cont. isolation, offsite power available
  - Was analyzed with WCOBRA-TRAC (RAI 720.010)
    - Calc PCT 833 F without uncertainty
    - Case analyzed assumed cont isol, because of margin fail cont isol will be OK
- **Both Cases Are Successful**

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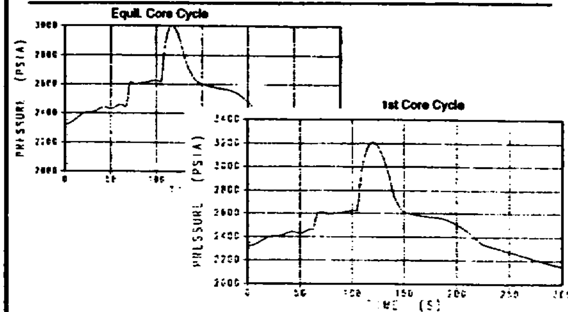
ATWS Analysis

- **Provides Very Low Unfavorable Exposure Time**
  - AP1000 has low boron core
    - MTC is more negative
    - ATWS "ride out" capability is possible for more than 98.5% of core life
      - Throughout equilibrium core cycles, peak RCS pressure < 3000 psig
      - Through 60% of 1st core cycle, peak RCS pressure < 3200 psig
      - UET < 1.5% over 40 years
- **AP1000 ATWS Analysis**
  - Analyzed with LOFTRAN
  - Equilibrium core has MTC = -12.5 pcm/F at BOL
  - 1st core has MTC = -10.0 pcm/F at 40% life

AP1000 ATWS Analysis Results



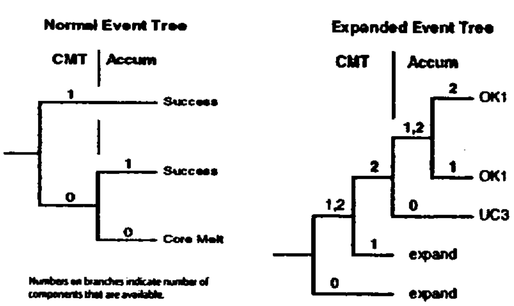
T&H Uncertainty

- **Same Approach As AP600**
  - Detailed evaluation performed (RAI 720.012)
  - Bounds AP1000 T&H uncertainty
    - Determined high risk / low margin cases
      - MAAP4 success criteria analysis used to identify low margin sequences
      - "Expanded" event trees used to identify high risk sequences
      - Bounds more than 98% of LOCA core melt
    - Identified limiting analysis cases
      - 3 small LOCAs, 2 large LOCAs, 2 LTC cases identified
    - Analyzed limiting cases with DCD codes and assumptions
      - Conservative decay heat (Appendix K), line resistances, plant parameters
      - All show successful core cooling

Expand Event Trees

- **Purpose of Expanded Event Trees**
  - Branches with safety equipment are expanded to identify the numbers of safety components that are available
    - The normal event trees only identify the minimum number of safety components that are required
  - Branches with non safety equipment are removed
  - End states changed to differentiate success paths
    - Two general classes, high margin (OK) and low margin (UC)
      - Low margin cases have core uncovery, high margin cases do not
    - More detailed sub-grouping made
      - Based on equipment available / not available
      - Supports selection of T&H uncertainty cases that are analyzed
  - Allows probability of low / high T&H margin cases to be calculated

Expanded Event Tree Example



Numbers on branches indicate number of components that are available.

Expanded Event Tree End States

1. OK1 More ADS-4 than Design Basis (DB)
2. OK2 Design Basis
3. OK3 More ADS-4 / Less ADS-1, 2, 3 than DB
4. OK4 Less ADS-1, 2, 3 than DB
5. OK5A More ADS-4 / CI fails
6. OK5B More ADS-4 / CI fails / Less ADS-1, 2, 3
7. OK6 DB ADS / CI fails
8. OK7 2 Accumulators / DB for LLOCA
9. OK8 SI line break with Auto ADS from faulted CMT
10. OK9 Loss of CMTs for smaller breaks

1. UC1 No make-up of inventory if RCS pressure greater than 700 psig
2. UC2A 1 Accumulator depletes prior to operator intervention
3. UC2B 2 Accumulators deplete prior to operator intervention
4. UC3 No rapid inventory make-up during blowdown
5. UC4 Reduced inventory make-up during LLOCA reflood
6. UC5 No make-up when ADS is actuated
7. UC6 Less ADS-4 than DBA (ie < 3 of 4 ADS-4)
8. UC7 Less ADS-4
9. UC8 No containment isolation / DBA
10. UC9 No containment isolation / reduced ADS

Which Event Trees

- Selection of Level 1 Event Trees to Expand
  - AP600 expanded 8 event trees, all with ADS actuation
    - No core uncovery in events / sequences without ADS
  - AP1000 expanded 5 event trees, all with ADS actuation
    - 3 event trees included in AP600 were not expanded for AP1000 since they did not result in limiting T&H analysis cases
      - Small LOCAs, Transients with ADS, SGTR with ADS were not expanded
      - These events did not add any limiting T&H uncertainty analysis cases
        - Some of their end states are not success in AP1000 (for example, 2 / 4 ADS 4 was considered success in AP600 but is not considered success in AP1000)
        - They tend to have more equipment available because they are more probable events
        - ADS occurs later in these events with lower decay heat

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Expanded Event Trees

Initiating Event	AP600	AP1000
Large LOCA	yes	yes
Spurious ADS 4	na	yes
Medium LOCA	yes	yes
CMT Line LOCA	yes	yes
DVI LOCA	yes	yes
Intermediate LOCA	yes	na
Small LOCA	yes	-
SGTR with ADS	yes	-
Transients with ADS	yes	-

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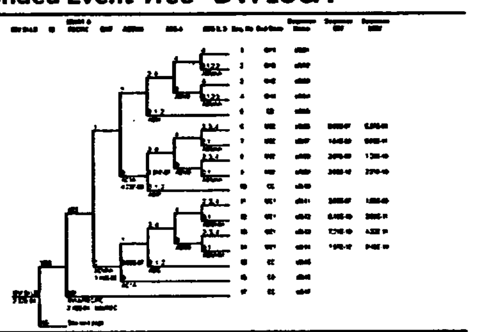
Expanded Event Tree - DVI LOCA



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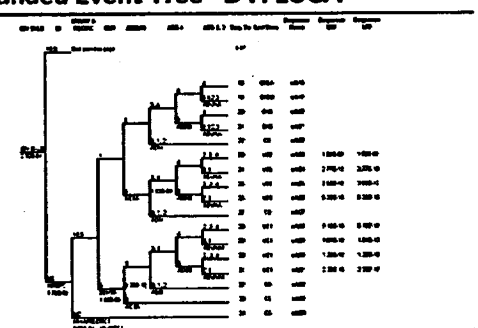
Expanded Event Tree - DVI LOCA



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Calculation of CDF / LRF

- Potential CDF
  - Conservatively assumes low margin sequences (UC) may be core damage
  - System reliabilities based on fault tree calc
    - Base PRA or special fault trees as needed
- Potential LRF
  - Based on potential core damage sequences
  - Uses constant ratio 6% for containment isol branches
    - Conservative, same as AP600

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Determination of Risk Important Sequences

- All Low Margin Sequences Are Collected
  - Includes all UC sequences
  - Sorted by CDF and LRF
  - Criteria for risk importance
    - 1% of baseline CDF or LRF
    - Residue of less important sequences must be small
      - Required to be less than twice the risk important sequences
- Results
  - 102 low margin sequences quantified in 5 expanded event trees
  - 13 low margin sequences selected as risk important
    - Covers 99.4% of risk from all low margin sequences
    - Residue of other sequences is < 6% of CDF and LRF

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Sorted UC Sequences (Top 25 of 102)

Table 3-1 UC Sequences Sorted by CDF Frequency

	End State	Seq. Name	Sequence CDF	Sequence LRF	% CDF	% LRF	CI	IRWST & RECIRC	CMT	ACC	ADS 4	ADS 2,3	Bounded By Short/Long-Term		
►►	UCS	[illegible]	[illegible]E-07	6.37E-08	371.[illegible]%	276.[illegible]%	YES	YES	1	0	4	2-4	C	FG	
►►	UC	[illegible]	4.[illegible]E-07	2.76E-08	190.[illegible]%	140.[illegible]%	YES	YES	2	1	4	2-4	E	FG	
►►	UC1	slb11	3.0[illegible]E-07	1.[illegible]E-08	128.7[illegible]%	94.[illegible]%	YES	YES	0	1	4	2-4	A	FG	
►►	UC2B	[illegible]	2.[illegible]E-07	1.73E-08	119.[illegible]%	[illegible]%	YES	YES	0	2	4	2-4	AB	FG	
►►	UC2B	[illegible]	1.34E-07	[illegible]E-09	[illegible]%	41.2[illegible]%	YES	YES	0	2	4	2-4	AB	FG	
►►	UC	[illegible]	9.12E-08	5.47E-09	37.[illegible]%	28.[illegible]%	NO	YES	2	2	4	2-4	E	G	(2)
►►	UCS	[illegible]	3.01E-08	1.81E-09	12.[illegible]%	9.[illegible]%	YES	YES	2	0	4	2-4	C	FG	
►►	UCS	lls15	8.51E-09	8.51E-09	3.5[illegible]%	43.[illegible]%	NO	YES	2	2	4	2-4	D	G	
►►	UCS	[illegible]	6.42E-09	3.8[illegible]E-10	2.[illegible]%	1.9[illegible]%	YES	YES	1	0	4	2-4	C	FG	
►►	UC2A	[illegible]	2.44E-09	1.47E-10	1.0[illegible]%	0.75%	YES	YES	0	1	4	2-4	A	FG	
	UCS	[illegible]	2.09E-09	1.2[illegible]E-10	0.87%	0.64%	YES	YES	1	0	3	2-4	C	F	
	UCS	[illegible]	1.64E-09	[illegible]E-11	0.[illegible]%	0.[illegible]%	YES	YES	1	0	4	0-1	C	FG	
►►	UCS	[illegible]	1.[illegible]E-09	1.52E-09	0.[illegible]%	7.7[illegible]%	NO	YES	1	0	4	2-4	C	G	
	UC2A	[illegible]	1.14E-09	[illegible]E-11	0.47%	0.3[illegible]%	YES	YES	0	1	4	2-4	A	FG	
	UC	[illegible]	1.07E-09	6.42E-11	0.44%	0.3[illegible]%	YES	YES	2	1	3	2-4		F	
	UC	[illegible]	8.40E-10	5.04E-11	0.3[illegible]%	0.2[illegible]%	YES	YES	2	1	4	0-1	E	FG	
	UC	[illegible]	7.77E-10	4.[illegible]E-11	0.32%	0.24%	NO	YES	2	1	4	2-4	E	G	(2)
	UC1	slb10	7.21E-10	4.3[illegible]E-11	0.30%	0.2[illegible]%	YES	YES	0	1	3	2-4		F	
	UC2B	[illegible]	[illegible]E-10	4.15E-11	0.2[illegible]%	0.21%	YES	YES	0	2	3	2-4		F	
	UC	[illegible]	6.7[illegible]E-10	4.0[illegible]E-11	0.2[illegible]%	0.21%	YES	YES	1	1	4	2-4	E	FG	
	UC2B	[illegible]	6.44E-10	3.[illegible]E-11	0.27%	0.2[illegible]%	YES	YES	0	2	4	0-1	AB	FG	
	UC1	slb12	6.1[illegible]E-10	3.[illegible]E-11	0.2[illegible]%	0.1[illegible]%	YES	YES	0	1	4	0-1	A	FG	
►►	UC1	[illegible]	5.16E-10	5.16E-10	0.21%	2.6[illegible]%	NO	YES	0	1	4	2-4	A	G	
►►	UC2B	mlo73	4.[illegible]E-10	4.[illegible]E-10	0.20%	2.5[illegible]%	NO	YES	0	2	4	2-4	A	G	
	UC2B	[illegible]	3.17E-10	1.9[illegible]E-11	0.13%	0.10%	YES	YES	0	2	3	2-4		F	

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Risk Important Sequences

Table 3.4 AP1000 T&H Uncertainty Low Margin / Risk Important Sequences

Case	Sequence	CI	IRWST & RECIRC	CMT	ACC	ADS 4	ADS 2/3	PRHR	Sequence CDF	Sequence LRF	% CDF	% LRF	Short Term	Long Term
1	[illegible]	YES	YES	1	0	4	2-4	N/A	8.[illegible]E-07	6.37E-08	371.7%	275.[illegible]%	C	F
2	sed06	YES	YES	2	1	4	2-4	N/A	4.[illegible]E-07	2.76E-08	190.1%	140.[illegible]%	E	F
3	slb11	YES	YES	0	1	4	2-4	YES	3.0[illegible]E-07	1.[illegible]E-08	128.[illegible]%	94.[illegible]%	A	F
4	mlo01	YES	YES	0	2	4	2-4	YES	2.[illegible]E-07	1.73E-08	119.[illegible]%	[illegible]%	B	F
5	cml01	YES	YES	0	2	4	2-4	YES	1.34E-07	8.0[illegible]E-09	56.7%	41.3%	B	F
6	sed06	NO	YES	2	2	4	2-4	N/A	9.12E-08	5.47E-09	37.[illegible]%	28.[illegible]%	E	G
7	mlo11	YES	YES	2	0	4	2-4	N/A	3.01E-08	1.81E-09	12.[illegible]%	9.[illegible]%	C	F
8	lls15	NO	YES	2	2	4	2-4	N/A	8.51E-09	8.51E-09	3.[illegible]%	43.[illegible]%	D	G
9	cml06	YES	YES	1	0	4	2-4	N/A	6.42E-09	3.8[illegible]E-10	2.7%	2.0%	C	F
10	mlo06	YES	YES	0	1	4	2-4	YES	2.44E-09	1.47E-10	1.0%	0.8%	A	F
11	slb23	NO	YES	1	0	4	2-4	N/A	1.52E-09	1.52E-09	0.[illegible]%	7.[illegible]%	C	G
12	slb26	NO	YES	0	1	4	2-4	YES	5.16E-10	5.16E-10	0.2%	2.[illegible]%	A	G
13	mlo73	NO	YES	0	2	4	2-4	YES	4.[illegible]E-10	4.[illegible]E-10	0.2%	2.[illegible]%	A	G
								Totals =	2.22E-0[illegible]	1.44E-07				
	Residue from UC Sequences not selected								1.26E-08	[illegible]E-10	5.2%	4.4%		
	Residue from sequences with PRHR failure								1.[illegible]E-10	9.67E-12	0.1%	0.[illegible]%		

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Bounding T&H Analysis Cases

- **T&H Uncertainty Cases**
  - 5 short term and 2 long term cooling cases are selected to bound the 13 risk important cases
  - These cases also bound 58 of the 102 low margin cases
    - Covers 99.8% of risk from all low margin sequences

Analysis Case	Initiating Event (1)	Cont. Isol.	IRWST & RECIRC	CMT	ACC	ADS 4	ADS 2/3	PRHR	Bounds Dominant Case
Short-Term Cooling									
A	RCS hot leg (3.0")	no	yes	0	1	4	0	yes	3,10,12,13
B	DE CMT balance line (6.8")	yes	yes	0	2	4	0	yes	4,5
C	DE DVI line (4")	no	yes	1	0	3	0	no	1,7,9,11
D	DE CL LLOCA	no	yes	2	2	4	0	yes	8
E	Spur ADS4 (2)	no	yes	1	1	4	0	yes	2,6
Long-Term Cooling									
F	DE DVI	yes	1/1 & 1/1	1	0	3	0	no	1-5, 7, 9, 10
G	DE DVI	no	1/1 & 2/1	1	0	4	0	no	6, 8, 11-13

Notes (1) Break sizes are effective sizes (inside diameter or orifice, not outside pipe diameter).
(2) Spurious ADS assumes all 4 ADS stage 4 valves open at same time as initiating event.

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T&H Uncertainty Analysis

- **All of These 7 Cases Have Been Analyzed**
  - Using DCD codes and methods
  - All cases show successful core cooling

Case	Initiating Event	Cont Isol.	IRWST	REC	CMT	ACC	ADS 4	ADS 2/3	PRHR	CODE	Results	Reference
Short-Term Cooling												
A	RCS hot leg (3.0")	no	yes	yes	0	1	4	0	yes	NOTRUMP	No core uncovery	RAI 720.015
B	DE CMT bal line (6.8")	yes	yes	yes	0	2	4	0	yes	NOTRUMP	No core uncovery	PRA App A
C	DE DVI line (4")	no	yes	yes	1	0	3	0	no	NOTRUMP	PCT = 1670 F	PRA App A
D	DE CL LLOCA	no	yes	yes	2	2	4	0	yes	WCOBRA-TRAC	PCT = 1[illegible] F (1)	RAI 720.012
E	Spur ADS4	no	yes	yes	1	1	4	0	yes	WCOBRA-TRAC	PCT = 10[illegible] F (1)	RAI 720.010
Long-Term Cooling												
F	DE DVI	yes	1/1	1/1	1	0	3	0	no	WCOBRA-TRAC	No core uncovery	RAI 720.013
G	DE DVI	no	1/1	2/1	1	0	4	0	no	WCOBRA-TRAC	No core uncovery	RAI 720.013

Notes:
(1) Includes DCD Large LOCA uncertainties.

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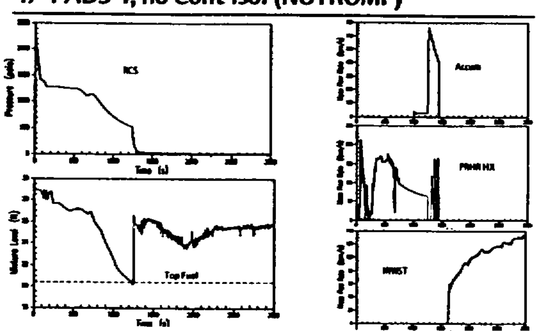
Case A, 3.0" LOCA, 1 Acc, 0 CMT, PRHR 4/4 ADS 4, no Cont Isol (NOTRUMP)



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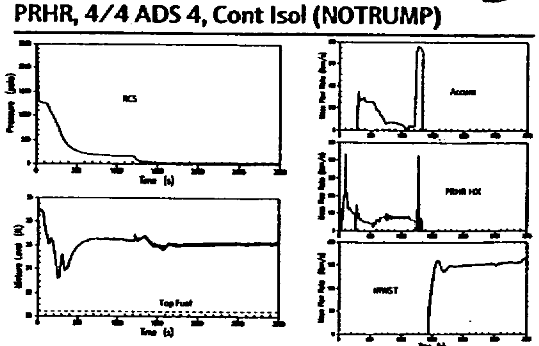
Case B, 6.8" CMT LOCA, 2 Acc, 0 CMT PRHR, 4/4 ADS 4, Cont Isol (NOTRUMP)



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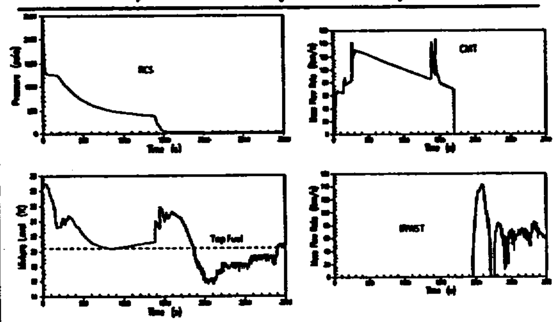
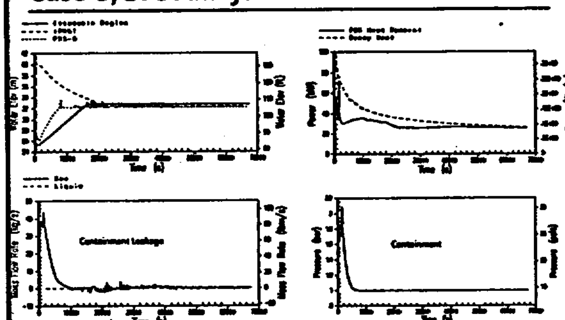
Case C, DVI LOCA, 0 Acc, 1 CMT, no PRHR 3/4 ADS 4, no Cont Isol (NOTRUMP)



T&H Uncertainty Case for Long-Term Cooling with Cont. Isol. Failure

- **Conservative / Limiting Case Analyzed**
  - Largest containment penetration is open (18" HVAC line)
  - DVI LOCA assumed to give lowest initial containment level
    - Causes flooding of PXS valve room where break is located
      - Reduces containment level by ~ x ft
- **LTC Analysis Results**
  - Containment leakage terminated in ~ 2.8 hr (MAAP4)
    - PCS is able to remove decay heat with cont. at atmospheric pressure
    - Leakage of steam/air mix removes air from containment
    - PCS heat transfer improves as partial pres of steam increases
    - Containment recirc level is reduced by ~ 0.3 ft
  - Core remains covered (WCOBRA-TRAC)

Case G, LTC Analysis with Cont. Isol. Failure

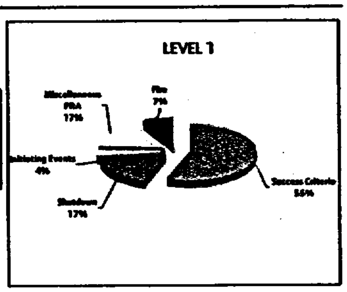


T&H Uncertainty Summary

- **AP1000 T&H Uncertainty Analysis**
  - Has calculated probabilities of low margin sequences
  - Has selected risk important, low margin sequences
  - Has defined 7 bounding T&H uncertainty cases
    - 5 Short and 2 Long-term
  - T&H Analysis has been performed on these cases
    - Using DCD Codes and methods
    - Shows successful core cooling
- **AP1000 T&H Uncertainty Is Not Risk Important**
  - ~ 99% of CDF and LRF is bounded by conservative T&H analysis

Summary of RAI on AP1000 Level 1 PRA

RAIs on PRA Level 1	
Success Criteria	30
Shutdown	9
Initiating Events	2
Miscellaneous PRA	9
Fire	4
TOTAL	54



AP1000 PRA Report Updates Included with RAI Responses

Level 1 PRA Updates

- T/H Uncertainties Explicitly Addressed
  - Expanded Event Trees
  - Additional T/H Analyses Performed
    - 99% of Success Sequences Backed-Up with DBA Analysis Models
- Operator Action Times Addressed
- Revision of PRA Chapter 6 and Appendix A
- AP1000-Specific Fire PRA Performed

AP1000 Level 2 / 3 PRA

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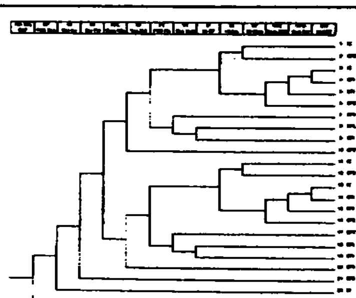
AP1000 Containment Event Tree

- Used to quantify frequency and magnitude of releases to the environment
- Essentially the same structure as AP600 Containment Event Tree

AP1000 Containment Event Tree Structure



AP1000 Containment Event Tree

- Phenomena and System Availability
  - reactor coolant system pressure
  - containment isolation
  - cavity flooding for external reactor vessel cooling
  - in-vessel reflooding
  - vessel failure
  - passive containment cooling water

AP1000 Containment Event Tree (continued)

- Phenomena and System Availability (continued)
  - hydrogen control (igniters)
  - containment overtemperature (diffusion flame)
  - hydrogen combustion (deflagration and detonation)
  - containment integrity

AP1000 Containment Event Tree

- Operator actions
  - Recovery Actions
    - depressurize RCS
    - isolate containment
    - actuate PCS water
  - Manual Severe Accident Management Actions
    - flood reactor cavity
    - actuate hydrogen control

Containment Event Tree Simplifying Assumptions

- High pressure RCS at core damage results in induced SGTR containment bypass
- Vessel failure and debris relocation into the containment results in early containment failure
  - highly conservative

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Interface with Level 1 PRA

FUNCTIONAL DEFINITIONS OF LEVEL 1 ACCIDENT CLASSES

Accident Class	Subclass	Definition
1	A	Core damage with RCS at high pressure following transient or RCS leak
	AP	Core damage with no depressurization following small LOCA and RCS leak with [illegible]
	D	Core damage with partial depressurization of RCS following transient
3	A	Core damage with RCS at high pressure following anticipated transient without [illegible]
	BE	Core damage following large LOCA with full RCS depressurization, [illegible]
	BE	[illegible]
	BL	[illegible]
	C	Core damage following vessel rupture
	D	Core damage following LOCA [illegible] with partial depressurization
6	E	Core damage following steam generator tube rupture or interfacing system LOCA. [illegible]
	L	Core damage following steam generator tube rupture. [illegible]

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Interface with Level 1 PRA

Accident Class Frequencies Level 1 PRA Results			
Accident Class	Frequency	%	Description
1A	5.0E-9	2.1	High RCS Pressure (Transient or SLOCA)
1AP	1.5E-9	0.6	SLOCA with PRHR Operating
3A	4.4E-9	1.8	High RCS Pressure (ATWS)
3BE	8.1E-8	33.4	RCS Depressurized; Gravity Injection Fails
3BL	2.4E-8	9.9	RCS Depressurized; Gravity Recirculation Fails
3BR	4.6E-8	19.2	RCS Depressurized; LLOCA CMTs and Accum Fails
3C	1.0E-8	4.2	RCS Depressurized; Vessel Failure Initiating Event
3D	6.0E-8	24.8	Partial RCS Depressurization
6	9.5E-9	4.0	SGTR or ISLOCA
Total CDF	2.41E-7	100	

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Release Categories

IC	Intact Containment	Containment integrity is maintained throughout the accident, and the release of radiation to the environment is due to nominal leakage.	Normal Leakage
BP	Containment Bypass	Fission products are released directly from the RCS to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to onset of core damage.	Large Release
CI	Containment Isolation Failure	Fission-product release through a failure of the system or valves that close the penetrations between the containment and the environment. Containment failure occurs prior to onset of core damage.	Large Release
CFE	Early Containment Failure	Fission-product release through a containment failure caused by severe accident phenomena occurring after the onset of core damage but prior to core relocation. [illegible]	Large Release
CFV	Containment Venting	Fission-product release through a containment vent line during [illegible]	Controlled Release
CFI	Intermediate Containment Failure	Fission-product release through a containment failure caused by severe accident phenomena, such as hydrogen combustion, occurring after core relocation but before 24 hours.	Large Release
CFL	Late Containment Failure	Fission-product release through a containment failure caused by severe accident phenomena, such as a failure of passive containment cooling, occurring after 24 hours.	Large Release

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Level 2 PRA Quantification Results

Table 43-8

LRF AND CONTAINMENT EFFECTIVENESS BY ACCIDENT CLASS

	3BE	3BL	3BR	1A	1AP	3A	3C	3D	6		TOTAL	%
CDF	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]	[illegible]
CFE	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]	[illegible]
CFI	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]	[illegible]
CFV	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]	[illegible]
CFL	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]	[illegible]
CI	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]	[illegible]
BP	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]	[illegible]
LRF	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]	[illegible]
	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]			
CNTM-EFF	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]	

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AP1000 Dominant LRF Sequences

CONTRIBUTION OF PDS TO LRF						
CET SEQ	REL. CAT	PDS	FREQ	% CDF	Cumulative % LRF	Sequence Description
23	BP	3A	[illegible]	1.7	[illegible]	Containment Bypass, ATWS
23	BP	6	[illegible]	1.6	[illegible]	Containment Bypass, SGTR
21	CFE	3BE	[illegible]	1.1	[illegible]	Sump Flooding Fails
21	CFE	3D	[illegible]	[illegible]	[illegible]	Sump Flooding Fails
23	BP	1A	[illegible]	[illegible]	[illegible]	Containment Bypass, Induced SGTR
10	CFE	3C	[illegible]	[illegible]	[illegible]	Vessel Failure Initiating Event
12	CFE	3D	[illegible]	[illegible]	[illegible]	Core Reflooding Fails, Diffusion Flame
23	BP	1AP	[illegible]	0.25	[illegible]	Containment Bypass
22	CI	3BL	[illegible]	[illegible]	[illegible]	Containment Isolation Fails
6	CFE	3BE	[illegible]	[illegible]	[illegible]	Hydrogen Ignition Fails, Early DDT
22	CI	3D	[illegible]	[illegible]	[illegible]	Containment Isolation Fails

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AP1000 LRF Sensitivity Analyses

Sensitivity	Result
No Credit Taken for DP Node for PDS-6	LRF becomes 2.49 E-08/yr, with a CCFP of 10.3 percent
Lesser Reliability for Containment Isolation	LRF becomes 4.05 E-08/yr, with a CCFP of 16.8 percent
Lesser Reliability for Hydrogen Igniters	The LRF becomes 2.31E-08/yr, with CCFP of 9.6 percent
Lesser Reliability for PCS	The LRF becomes 1.97E-08/yr, with CCFP of 8.2 percent
No Credit for Depressurization for High Pressure PDS	The LRF is 2.91E-08/year, with CCFP of 12.1 percent

AP1000 LRF Sensitivity Analysis

Sensitivity	Result
Set PDS-3C Vessel Failure Probability to 1.0	The LRF is 2.85E-08/yr, with a CCFP of 11.8 percent
Set 3D and 1AP Diffusion Flame and Detonation Failure Probability to 1.0	The LRF becomes 7.66E-08/yr, with CCFP of 31.8 percent

AP1000 LRF Importance Analyses

Table 43-10

CET EVENT TREE NODE IMPORTANCES

CET NODE	LRF (per year)	Containment Failure Prob	Containment Effectiveness	Node Failed in Following PDS
BASE LRF	1.95E-08	8.1%	91.9%	N/A
DP RCS Depressurization	2.91E-08	12.1%	87.9%	1A, 1AP, 3A, and 6 set to 1.0
IS Containment Isolation	2.41E-07	100.0%	0.0%	all PDS set to 1.0
IR Cavity Flooding	1.59E-07	65.6%	34.4%	3BE, 3D, 6 set to 1.0
RFL Core Reflooding	1.91E-08	7.9%	92.1%	3BE set to 1.0
VF Vessel Failure	2.85E-08	11.8%	88.2%	3C set to 1.0

AP1000 LRF Importance Analyses

Table 43-10

CET EVENT TREE NODE IMPORTANCES

CET NODE	LRF (per year)	Containment Failure Prob	Containment Effectiveness	Node Failed in Following PDS
BASE LRF	1.95E-08	8.1%	91.9%	N/A
PC PCS Failure	2.41E-07	100.0%	0.0%	all PDS set to 1.0
IG Hydrogen Ignitor Failure	6.28E-08	26.0%	74.0%	all PDS set to 1.0
DF Diffusion Flame	1.41E-07	58.3%	41.7%	3D and 1AP set to 1.0. 3BE set to 0.0535 (15% of 3BE are SI-LB events where DF may start. Diffusion flame does not apply to other PDS.
DTE Early DDT	2.16E-08	9.0%	91.0%	all PDS set to 1.0
DFG Hydrogen Deflagration	2.17E-08	9.0%	91.0%	all PDS set to 1.0
DTI Intermediate DDT	2.17E-08	9.0%	91.0%	all PDS set to 1.0

AP1000 Level 2 Conclusions and Insights

- **LRF is 1.95x10^{-8} per reactor year.**
  - Goal is LRF less than 1x10^{-6} per reactor year
- **Overall containment effectiveness (CE) is 92%**
- **PDS-3A (ATWS) has lowest CE.**
- **CE for PDS-6 (SGTR) is 57%.**
  - If all SGTR sequences go to bypass overall CE = 89.7%

AP1000 Level 2 Conclusions and Insights

- **LRF is not sensitive to the reliability of the hydrogen igniters, but if the igniters are assumed to be failed (probability of 1.0), the CE drops to 74%**
- **If the DF failure probability is 1.0 for all 1AP and 3D sequences, the CE is 84.5%. LRF increase by a factor of 4.**

AP1000 Level 3 PRA

- AP1000 specific source terms calculated with MAAP4
- MACCS2 v. 1.12 used to calculate doses
- Goal
  - Frequency of site boundary whole body dose >25 rem EDE less than 1.0×10^{-6} per reactor-year.

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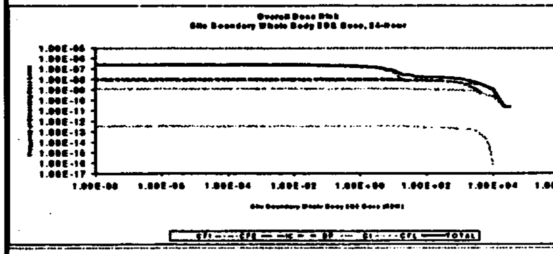
24 Hour Site Boundary Whole Body EDE Dose



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AP1000 In-Vessel Retention of Molten Core Debris

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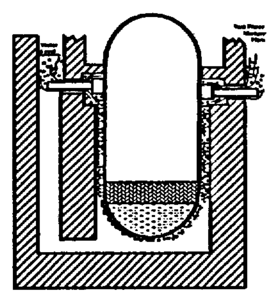
IVR via External Cooling of Reactor Vessel



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Passive Plant Features Promote IVR

- Reliable post-accident RCS depressurization
  - low stresses on reactor vessel
- No RPV lower head penetrations
  - creep failure of lower head only failure mechanism
- Reactor vessel submerged in water post-accident
  - automatic or manual flooding of cavity with IRWST water

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Passive Plant Features Promote IVR

- Core support plate sits low in lower plenum
  - lower plenum debris contacts and melts RPV internals
  - thick metal layer
  - no focusing effect of metal layer
- Reactor vessel insulation designed to promote IVR
  - standoff from reactor vessel
  - provides flowpath for cooling

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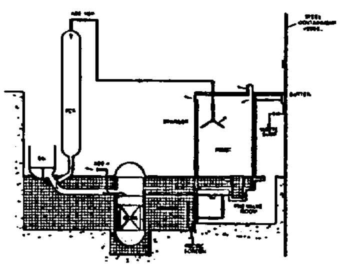
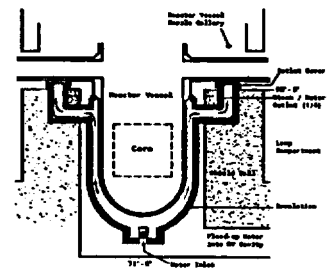
AP1000 Containment Flooding



Reactor Vessel Insulation Promotes IVR



AP600 IVR Assessment

- **Risk Oriented Accident Analysis Methodology**
  - Analysis
  - Test Program
  - Peer Review
- **DOE/ID 10460, "In-Vessel Retention and Coolability of a Core Melt," Theofanous, et. al.**
- **ACOPO test to investigate natural convection heat transfer from debris to vessel at Ra' ≤ 10^{16}**
- **ULPU test to investigate CHF on external vessel surface**

AP600 IVR Assessment

- **Exceeding Critical Heat Flux (CHF) is limiting vessel failure criterion**
  - heat flux to vessel wall < CHF is success
- **Steady-state, two-layer debris configuration presents limiting challenge to the reactor vessel**
  - metal over oxide debris bed in lower plenum
- **Large margin to vessel failure**
  - RCS depressurized
  - cavity flooded

AP1000 vs. AP600

- **Designs are similar**
- **Changes to the AP1000 that potentially impact IVR**
  - power is increased from 1933 to 3400 MWt.
  - 157 14-ft fuel assemblies.
  - core shroud instead of reflector
  - lower core support plate is 1" thicker

Implement IVR for AP1000

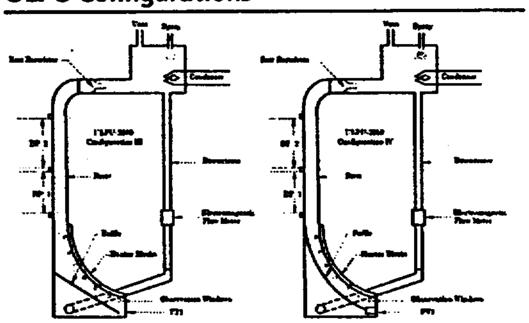
- **Increase critical heat flux (CHF) at vessel surface to maintain margin to failure**
- **Demonstrate thermal failure remains the limiting failure mechanism for increased heat removal**
- **Investigate in-vessel melt progression**
- **Demonstrate that the heat load correlations scale appropriately to the AP1000.**
- **Quantify the margin to failure**

Increase CHF

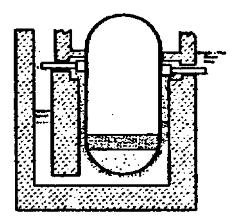
- **UPLU Configuration IV Test - UC Santa Barbara**
  - Lower Head slice geometry at full scale radius
  - Full scale simulation via power shaping
  - Models AP600 entrance and venting restriction
  - movable baffle, fixed at 90°
- **Tests Completed**
  - examine lower head baffle geometry impact
  - examine water level effects

ULPU Facility

ULPU Configurations

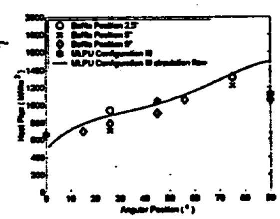


Effect of water level during IVR



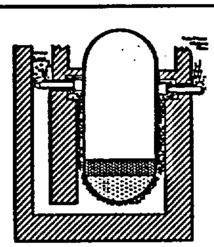
Low Water Level

Pool Boiling



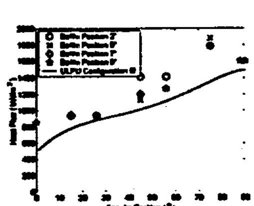
ULPU-IV Data Compared to ULPU-III correlation

Effect of Water Level during IVR



High Water Level

2-Phase Nat Circ



ULPU-IV Data Compared to ULPU-III correlation

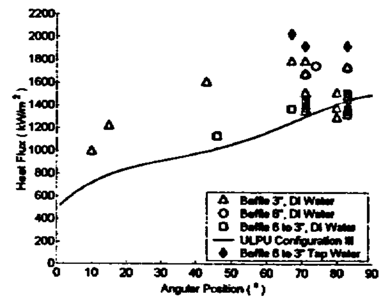
ULPU Configuration IV Conclusions

- **ULPU Configuration IV test report submitted to the NRC**
  - DCP/NRC1510 dated 6/6/2002
- **CHF can be increased sufficiently to provide margin for AP1000**
  - channel flow around lower head
  - high water level for 2-phase natural circulation
- **Adverse exit effect at top of baffle that reduced local CHF**
  - resolved by ULPU Configuration V tests

ULPU Configuration V

- Funded by DOE International-NERI Program
- AP1000 specific inlet/exit modeling
- Adjustable baffle design
- Additional aspects investigated
  - surface effects
  - water chemistry
  - exit phenomena
- Optimization of reactor vessel insulation/water circulation flow path

ULPU Configuration V



ULPU Configuration V

- Tests performed show AP1000 CHF can easily be met with margin.
- Exit phenomena is negligible
- Optimum surface is unpainted and oxidized

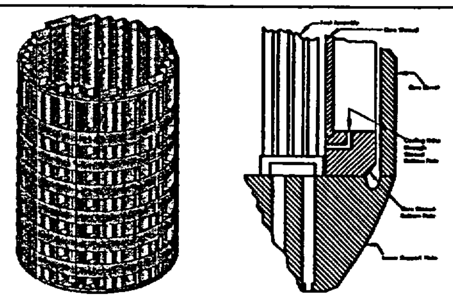
Vessel Structural Failure

- Confirm that thermal failure criterion is still limiting for increased heat load
  - large margin to structural failure
- At a bounding heat flux of 2000 kW/m², vessel thickness is 36 times the thickness required to carry dead load
- Thermal failure criterion is still limiting

In-Vessel Melt Progression

- AP600 in-vessel melt progression influenced by low power density and radial reflector
  - downward relocation to lower plenum blocked
  - sideward failure through reflector into dead ended region
  - core barrel failure
  - quickly contacts support plate to mitigate focusing effect
- AP1000 has higher power density and a core shroud instead of a radial reflector

AP1000 Core Shroud

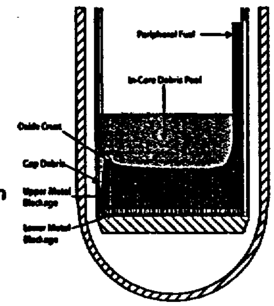


Modeling of Core and Internals Heatup

- Accident Sequence
  - fully depressurized
  - earliest core uncovery is conservative (Large LOCA)
  - no vessel reflood
  - conservatively assumed spurious ADS stage 4 opening
- MAAP4
- Finite Difference Model of core and internals
  - using uncovery timing from MAAP4
- Hand calculation of core heat up and melting

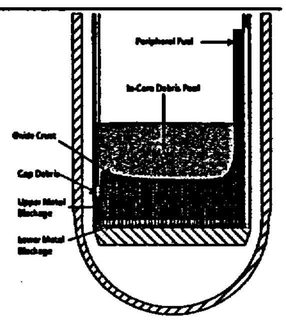
Formation of In-Core Debris Pool

- Upper core shroud melts prior to fuel melting
- Upper core barrel significantly thinned and overheated
- Most peripheral fuel assemblies initially remain intact
- Oxide blockage at ~1 m above bottom of fuel



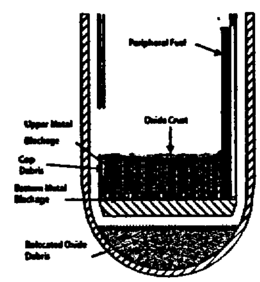
Formation of In-Core Debris Pool

- Downward relocation pathway blocked by frozen metal and oxide
- Gap between shroud and barrel fills with debris
- In-core debris pool contact with core barrel
- Core barrel fails sidewards near upper surface of pool

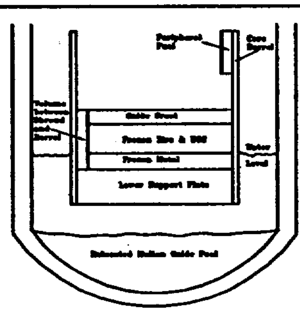


Initial Relocation to Lower Plenum

- 6.2 m^3 of UO_2 and ZrO_2
- 8 m^3 below lower core support plate
  - creep of core barrel
- Occurs at 6000 seconds
- Duration of initial relocation is ~500 seconds
  - ablation of core barrel by relocating debris



Subsequent Relocation of Debris



Subsequent Relocation of Debris

- Success Criterion
  - debris contacts lower support plate before dry out
  - mitigates focusing effect
- Debris contact occurs 6717 seconds
- Lower plenum dry out occurs at 6888 seconds
  - calculated conservatively assuming heat load from 8 m^3 of debris
- Transient debris configurations are water cooled
- Focusing effect is mitigated by inclusion of lower support plate and shroud in metal layer

RASPLAV and MASCA Tests

- **Addressed in RAI 720.047**
- **In-vessel materials testing**
- **Prototypical materials**
- **Non-prototypic conditions**
  - Rayleigh number too low
  - Heat fluxes too high
  - Ratio of masses not applicable
- **Tests do not contradict position on IVR**

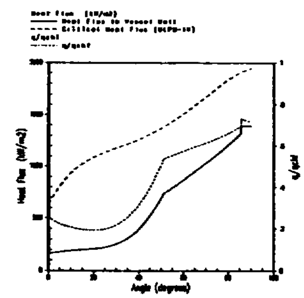
Application of Heat Transfer Correlations

- **Oxide Debris Pool Heat Transfer ($Ra' \sim 10^{16}$)**
  - to vessel wall and upward to metal layer
    - Angelini-Theofanous correlations ($Ra' \leq 10^{16}$)
- **Metal Layer Heat Transfer ($Ra \sim 10^{10}$)**
  - to vessel wall
    - Churchill-Chu correlation ($Ra < 10^{12}$)
  - from oxide layer and to top surface
    - Globe-Dropkin correlation ($3x10^{5} < Ra < 7x10^{9}$)
    - modest extrapolation for thick metal layer

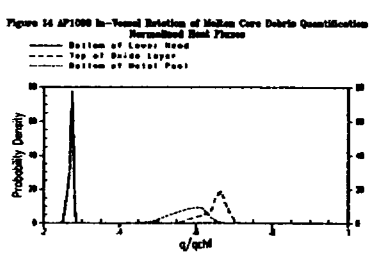
Quantification of Thermal Loads

- **Calculate AP1000 thermal loading using DOE/ID 10460 methodology**
- **Use ULPU Configuration IV Critical Heat Flux**
- **Input parameters based on AP1000 power level, geometry of reactor vessel and masses of core materials**
- **AP1000 probability distributions for uncertain input parameters**
  - fraction of cladding oxidized during melt
  - mass of stainless steel in debris
  - time with respect to shutdown (decay heat)

AP1000 Bounding IVR Calculation



Results of Thermal Load Quantification



Conclusions

- **IVR is successfully demonstrated for AP1000 with margin to failure similar to AP600**
  - CHF is increased
  - ULPU Configuration V has greater margins
- **Insulation geometry and structure are important**
  - forms baffle to direct water smoothly over lower head
- **Two-phase natural circulation is required**
  - deep flooding of the reactor cavity is needed

AP1000

AP1000 Severe Accident Phenomenological Evaluations

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AP1000 Severe Accident Studies

- Support Level 2 PRA Quantification
- SECY-93-087 Deterministic Requirements

Severe Accident Phenomena

- In-vessel fuel coolant interaction
- High Pressure Core Damage
  - Induced failure of steam generator tubes
  - High pressure melt ejection / direct containment heating
  - Melt attack on the containment pressure boundary
- In-vessel hydrogen generation
- Hydrogen deflagration and detonation
- Diffusion flame overheating containment shell

Severe Accident Phenomena (continued)

- Containment overpressure by decay heat
- Reactor vessel integrity
- Ex-vessel fuel coolant interaction
- Core-concrete interaction
- Equipment survivability

In-Vessel Fuel-Coolant Interaction

- Lower head integrity under steam explosion loads
- Steam Explosion Assessment for AP600
  - large margin to failure
- AP600 conclusions are extended to AP1000
- AP1000 conditions
  - similar debris relocation pathway
  - similar molten debris mass flow rate
  - same lower plenum geometry

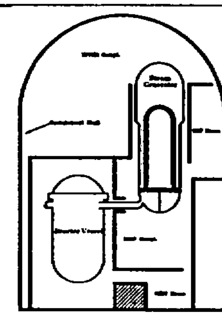
High Pressure Core Damage

- Severe Accident Issues
  - Induced failure of steam generator tubes
  - High Pressure Melt Ejection/Direct Containment Heating
  - Melt attack on containment pressure boundary
- Prevention
  - Diverse RCS depressurization capability
    - two train, four stage ADS
    - PRHR Heat Exchanger
  - High pressure core damage frequency < 5% total CDF

High Pressure Core Damage (continued)

- **Mitigation**
  - operator actions to recover ADS, PRHR
  - potential for hot leg or surge line creep rupture
  - torturous pathway from reactor cavity to upper compartment
- **PRA Treatment**
  - assess likelihood of operator actions to depressurize RCS
  - assume induced tube rupture and containment bypass
- **Success Criterion**
  - 2 of 4 ADS stage 4 valves open

HPME Debris Retention



Hydrogen Generation

- **In-vessel hydrogen generation**
  - cladding oxidation during core uncovery
- **Ex-vessel hydrogen generation**
  - prevented by in-vessel retention of core debris
  - containment pressurization during core-concrete interaction

Hydrogen Combustion

- **Threat to containment integrity**
  - locally high temperature (diffusion flame)
  - overpressure (deflagration)
  - dynamic loading (detonation)
- **Prevention**
  - low core damage frequency
- **Mitigation**
  - passive autocatalytic recombiners (PARs)
  - hydrogen igniters

Treatment of Hydrogen in PRA

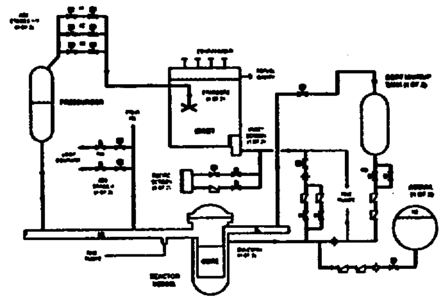
- **In-vessel releases only**
  - vessel failure is conservatively assumed to fail containment early
- **Three scenarios**
  - no reactor vessel reflood
  - early reactor vessel reflood (core relatively intact)
  - late reactor vessel reflood (core geometry lost)

Treatment of Hydrogen in PRA

- **Diffusion Flame**
  - postulated at IRWST vents, PXS compt exits
  - mitigated by ADS stage 4
  - preferential release away from containment walls
- **Success Criterion**
  - Hydrogen vented away from containment shell
    - ADS stage 4
    - IRWST pipe vents
    - PXS compartment hatches

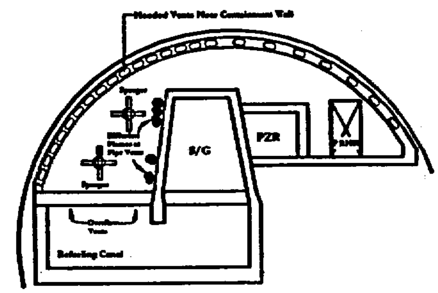
Hydrogen Release Through Stage 4

AP1000



Preferential Venting from IRWST

AP1000



Early Detonation

AP1000

- During hydrogen release from RCS
- Containment not well mixed
  - locally high hydrogen concentrations
- Mitigated by hydrogen igniters
- Deflagration to Detonation Transition (DDT)
  - no source for direct ignition
- Probabilities for early DDT based on AP600
  - RAI showed approach was conservative

Sherman - Berman Methodology

AP1000

- Assign Probability of Deflagration to Detonation Transition
  - flame acceleration
- Function of Gas Mixture and Compartment Geometry
- Detonation cell widths
  - equivalence ratio (measure of mixture with respect to stoichiometry)
  - steam concentration
- Compartment Geometry Classes

Global Hydrogen Deflagration

AP1000

- Intermediate Time Frame (<24 hours)
- Containment well-mixed
- Mitigated by igniters
- Adiabatic peak pressure calculation
- Performed for three general accident scenarios
  - no reflood
  - early reflood
  - late reflood

Global Hydrogen Deflagration

AP1000

- Input probability distributions for each scenario
  - mass of hydrogen generated (cladding oxidation)
  - containment pressure at ignition
- Containment fragility success criterion
  - probability of containment failure vs. pressure
- Probability of containment failure

Global Hydrogen Deflagration

- **Safety Margin Basis Calculation**
- **Deterministic Calculation**
  - 100% cladding reaction
  - containment pressure at 55% steam concentration
  - adiabatic peak pressure calculation
- **Peak pressure is 90 psig**
- **Containment Service Level C is 91 psig**

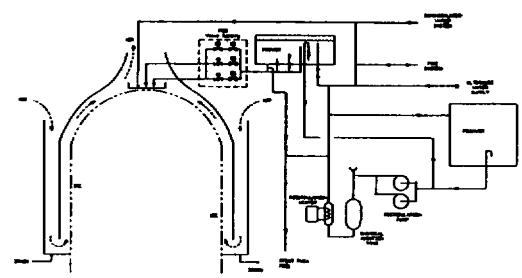
Intermediate Detonation

- **Less than 24 hours after core damage**
- **Containment well mixed**
- **Deflagration to Detonation Transition**
- **Sherman-Berman Mixture Class Probabilities**
  - calculated from hydrogen mass and containment pressure probability distributions
  - air-steam-hydrogen mixture classes
  - dry air-hydrogen mixture classes for CMT room
    - resolves uncertainty with respect to steam stratification
- **Output probability of DDT**

Containment Overpressure by Decay Heat

- **Mitigated by passive containment cooling water**
- **PCS water cooling is more reliable than AP600**
  - added third diverse actuation path
- **Success criterion**
  - at least 1 of 3 PCS actuation paths operates

PCS Water Delivery



Dry PCS Cooling

- **Dry PCS cooling is sufficient to prevent containment failure for 24 hours.**
- **Success Criterion**
  - containment fragility probability distribution
- **Nominal conditions**
  - 0.0 failure probability in 24 hours
- **Conservative conditions**
  - 0.02 failure probability in 24 hrs
    - ANS 79 decay heat + 2 sigma uncertainty
    - Outside Temperature = 115 F

Reactor Vessel Integrity

- **Vessel integrity maintained via external cooling**
  - cavity fully flooded
- **Vessel Failure Modes**
  - Global failure of lower head (hinged failure)
  - Local failure of lower head
- **Containment conditions**
  - water level at 83' elevation (loop compartment floor)

Ex-Vessel Steam Explosion

- Prevented by in-vessel retention of core debris
- AP600 assessment
  - hinged failure of the lower head
  - partially flooded cavity
- Similar vessel failure mode for AP1000
- Similar geometry
  - AP1000 vessel is closer to the floor
- AP600 conclusions are extended to the AP1000

Core-Concrete Interaction

- Prevented by in-vessel retention of core debris
- Vessel failure modes
  - hinged failure
  - localized failure
- Concrete Types
  - Limestone
  - Basaltic
- Success Criteria
  - Basemat intact for 24 hours

Core-Concrete Interaction

- MAAP4 calculation of CCI
- Minimum time to basemat failure
  - 2.8 days to melt-through basemat
- Basemat melt-through occurs before containment overpressurization by non-condensable gases

Equipment Survivability

- Identified actions to achieve controlled stable state
- Defined time frames for each action
- Identified equipment and instruments needed for each action
- Determine bounding environments (MAAP4)
- Show reasonable assurance that equipment will perform when needed

Summary of PRA Results and Insights

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Comparison of AP600 and AP1000 PRA Results

Scope	AP600	AP1000
Level 1 At-Power Internal Initiating Events	Quantification Performed CDF = 1.7E-07 Several additional cases quantified in response to NRC RAIs	Quantification Performed CDF = 2.4E-07 AP600 additional cases incorporated into the model
Level 2 At-Power Internal Initiating Events	Quantification Performed LRF = 1.[illegible]E-0[illegible] Containment Effectiveness = [illegible]%	Quantification Performed LRF = 2.0E-08 Containment Effectiveness = 91.[illegible]%
Level 3 At-Power Internal Initiating Events	Quantification Performed	Quantification Performed
Internal Fire Events	Conservative (no focused PRA) Quantification Performed CDF = [illegible]E-07 (internal) CDF = [illegible]E-07 (shutdown)	Quantification performed CDF = 5.61E-0[illegible]
Internal Flooding Events	Quantification Performed CDF = 2.2E-10	Quantification Performed CDF = [illegible]E-10
Shutdown Events	Quantification Performed for Level 1 and 2 CDF = 1.[illegible]E-07 LRF = 1.5 E-08 Several additional cases quantified in response to NRC RAIs	Qualitative Evaluation Performed CDF = 1.2E-07 LRF = 2.0E-0[illegible]

SUMMARY OF RESULTS

- **The AP1000 PRA results show that**
  - The very low risk of the AP600 has been maintained in the AP1000
  - The AP1000 PRA meets the US NRC safety goals with significant margin

Summary of PRA Results

- The total mean core damage frequency is at least two orders of magnitude smaller than those for existing pressurized water reactors
- The total plant severe release frequency is another order of magnitude smaller than that of the core damage frequency; that places such a release frequency in the range of incredible events
- A bounding analysis of the core damage due to internal fire and internal flooding events shows that these two categories of internal events are much lower for AP1000 than are calculated for currently operating plants

Summary of PRA Results

- The severe release frequency is about equal for at-power and shutdown events. The severe release frequency as a percentage of core damage frequency is 8 percent for at-power events and 17 percent for shutdown events
- The results show that the design goals of low core damage frequency and low severe release frequency have been met. The AP1000 frequencies are lower than the NRC and ALWR URD goals set for new plant designs.
- The results show the effectiveness of passive systems in mitigating severe accidents and reflect the reduced dependence of AP1000 on nonsafety systems and human actions

Most Important Level 1 Insights

- The AP1000 design benefits from the high level of redundancy and diversity of the passive safety-related systems; passive safety systems have been shown to be highly reliable, their designs are simple so that a limited number of components are required to function
- AP1000 is less dependent on nonsafety-related systems; the nonsafety-related support systems (ac power, component cooling water, service water, and air) have a limited role in the plant risk profile because the passive safety-related systems do not require cooling water or ac power
- AP1000 is less dependent on human actions; the AP1000 meets the NRC safety goal even when no credit is taken for operator actions

Most Important Level 1 Insights

- The core damage and large release frequencies are low despite the conservative assumptions made in specifying success criteria for the passive systems. The success criteria have been developed in a more systematic, rigorous manner than typical PRA success criteria. The baseline success criteria are bounding cases for a large number of PRA success sequences. The baseline success sequences, in most cases, have been defined with:
  - worst (i.e., the most limiting) break size and location for a given initiating event
  - worst automatic depressurization system (ADS) assumption in the success criterion
  - worst number of core makeup tanks (CMT) and accumulators
  - worst containment conditions for in-containment refueling water storage tank (IRWST) gravity injection.
  - Many less-limiting sequences are therefore represented by a baseline success criteria.

Most Important Level 1 Insights

- Single system or component failures are not overly important due to the redundancy and diversity of safety-related systems in the design. For example, the following lines of defense are available for reactor coolant system (RCS) makeup:
  - chemical and volume control system
  - core makeup tanks
  - partial automatic depressurization system in combination with normal residual heat removal
  - full automatic depressurization system with accumulators and in-containment refueling water storage tank
  - full automatic depressurization system with core makeup tanks and in-containment refueling water storage tank

Most Important Level 1 Insights

- Typical current PRA dominant initiating events are significantly less important for the AP1000 - for example:
  - **Reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA) event has been eliminated as a core damage initiator since AP1000 uses canned motor reactor coolant pumps which do not have seals**
  - **Station blackout and loss of offsite power (LOOP) event is a minor contributor to AP1000 since the passive safety-related systems do not require the support of ac power**
- Passive safety-related systems are available in all shutdown modes
  - **Planned maintenance of passive features is only performed during shutdown modes when that feature is not risk important**
  - **Planned maintenance of nonsafety-related defense-in-depth features used during shutdown is performed at power**

Most Important Level 1 Insights

- The AP1000 passive containment cooling design is highly robust. Air cooling alone can prevent containment failure, although the design has other lines of defense for containment cooling such as fan coolers and alternate sources of passive containment cooling water
- The potential for containment isolation and containment bypass is lessened by having fewer penetrations to allow fission product release; all normally open and risk important penetrations are fail-closed, thus eliminating the dependence on instrumentation and control (I&C) and batteries
- The reactor vessel lower head has no vessel penetrations, thus eliminating penetration failure as a potential vessel failure mode
- The potential for the spreading of fires and floods to safety-related equipment is significantly reduced by the AP1000 layout

Most Important Level 2 Insights

- The containment effectiveness for AP1000 is over 90%, which provides an order of magnitude decrease from CDF to LRF. Since the results already includes CDF sequences that directly bypass the containment, the containment effectiveness for remaining sequences is actually much better. For example, for 5 (3BE,3BL,3BR,3C,3D) of the nine accident classes studied, the containment effectiveness ranges from 89.7 to 99.8%
- Preventing the relocation of molten core debris to the containment eliminates the occurrence of several severe accident phenomena, such as ex-vessel fuel-coolant interactions and core-concrete interaction, which may threaten the containment integrity. Therefore, AP1000, through the prevention of core debris relocation to the containment, significantly reduces the likelihood of containment failure

Most Important Level 2 Insights

- A frequency of 1.0e-08/year has been assigned to the vessel failure initiating event (accident class 3C). In 90% of these events, the vessel is assumed to undergo failures that will be above the beltline: in which case the molten core could be cooled and containment would not be challenged. In the remaining 10% of the cases, the failure is assumed to be below the pressure vessel beltline, whereby the molten core would drop into the containment. In this case, it is conservatively assumed that the containment would fail. A sensitivity analysis is made where by 100% of the failures would be below the beltline. The result shows that the containment effectiveness drops to 88.2%. This change is not significant, and the assumptions behind the case are very conservative.

Most Important Level 2 Insights

- The LRF results are sensitive to failure of hydrogen ignitors. If no credit is taken for hydrogen ignitors, the containment effectiveness drops to 74%.
- However, LRF is not very sensitive to the reliability of hydrogen ignitors; if IG reliability is assumed to be degraded (0.1) across the board for all accident classes, the containment effectiveness becomes 90.5%, which is an insignificant change from the base case.

Most Important Level 2 Insights

- The LRF is dominated (53.9%) by containment failures or bypasses due to SGTR, and unmitigated high-RCS-pressure core damage sequences, classified as BP. The remaining containment failures are dominated by an early containment failure due to reactor cavity flooding failure.
- The LRF is not very sensitive to the reliability of PCS; if PCS reliability is assumed to be 0.001 across the board for all accident classes, the LRF becomes 1.97E-08, which is an insignificant change from the base case.
- The LRF is sensitive to the operator action to flood the reactor cavity in a short time following core damage. This operator action has been moved to the beginning of ERG AFR.C-1 to increase its success likelihood.

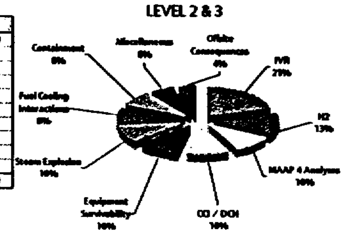
Most Important Level 2 Insights

- The potential for a release of radioactive materials to the environment is very small. This is largely due to the very small core damage frequency and very small release frequency. The containment design provides enhanced deposition of core materials that could be released in a severe accident, and the passive containment cooling system minimizes the energy available to expel such materials from the containment.
- Deterministic analyses of severe accident phenomena show that AP1000 features are effective in maintaining containment integrity

Summary of RAI on AP1000 Level 2/3 PRA

RAIs on PRA Level 2 & 3	
IVR	10
H2 generation, mixing & combustion	6
MAAP 4 Analyses	5
CO / DCH	5
Equipment Survivability	5
In-vessel & Ex-vessel Steam Explosion	5
Fuel Cooling Interactions	4
Containment (failure, isolation,..)	4
Miscellaneous	3
Offsite Consequences	2
TOTAL	48

LEVEL 2 & 3



AP1000 PRA Report Updates Included with RAI Responses

IVR (level 2)
- IVR of Core Melt Debris Analyses
- Revision of PRA Chapter 34 and 39
- Revision of DCD Section 19.39

MAAP4 Analyses (level 2)
- Severe Accident Analyses
- Fission-Product Source Term Analyses
- Revision of PRA Chapter 34 and 45
- Revision of DCD Section 19.34

AP1000 PRA Report Updates Included with RAI Responses

Hydrogen (level 2)
- H2 generation, mixing and combustion analyses
- Revision of PRA Chapter 41
- Revision of DCD Section 19.41

Equipment Survivability (level 2)
- MAAP 4 Analyses (Environment)
- Revision of PRA Appendix D

Offsite Consequences (level 3)
- Offsite dose risk quantification
- Revision of PRA Chapter 49 and 59
- Revision of DCD Section 19.59

AP1000 PRA Report Updates Included with RAI Responses

Others (level 1, 2 and 3)
- Revision of PRA Chapter 12 (IRWST CCF)
- Revision of PRA Chapter 29 (IRWST CCF)
- Revision of PRA Chapter 30 (Time window for operator action)
- Revision of PRA Chapter 35 (CET)
- Revision of PRA Chapter 57 (Fire)
- Revision of PRA Chapter 59 (Insights, Fire)
- Revision of DCD Appendix 19E (Shutdown)

Summary

- **AP1000 PRA Report**
  - Complete AP1000-Specific PRA
    - Sufficient for AP1000 Design Certification
  - Demonstrates that the AP1000 meets the US NRC safety goals with significant margin
  - Revision 1 will be issued to include W responses to staff RAI: February 2003