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

June 4, 2008

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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553nd MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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WEDNESDAY

JUNE 4, 2008

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ROCKVILLE, MARYLAND

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The Advisory Committee meeting was held at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 8:30 a.m., Dr. William Shack, Chairman, presiding.

COMMITTEE MEMBERS PRESENT:

WILLIAM SHACK, Chairman
MARIO V. BONACA, Vice Chairman
JOHN D. SIEBER, Member-at-Large
SANJOY BANERJEE, Member
J. SAM ARMIJO, Member
DANA A. POWERS, Member
SAID ABDEL-KHALIK, Member
OTTO L. MAYNARD, Member

COMMITTEE MEMBERS PRESENT (Continued):

JOHN STETKAR, Member
DENNIS C. BLEY, Member
MICHAEL CORRADINI, Member
GEORGE E. APOSTOLAKIS, Member

NRC STAFF PRESENT:

MICHAEL SALAY
DAVID BESSETTE
RICHARD LEE
MARK CUNNINGHAM
MARK FRANOVICH
HAROLD VANDER MOLLEN
PETE APPIGNANI
GETACHEW TESFAYE
JOE COLACCINO
BONNIE SCHNETZLER
PATRICIA HOLOHAN
SCOTT MORRIS
TIM REED
NANETTE GILLES
LOU CABELLAS
FRANK GILLESPIE
BILL RACKLEY
JAKE ZIMMERMAN
SANDRA SLOAN

ALSO PRESENT:

MARTY PARESE

JEFF TUCKER

TODD OSWALD

VIC FREGONESE

TABLE OF CONTENTS

PROCEEDINGS

(8:30 a.m.)

CHAIRMAN SHACK: The meeting will come to order.

This is the first day of the 553rd meeting of the Advisory Committee on Reactor Safeguards. During today's meeting, the Committee will consider the following: ARTIST test program; Risk Assessment Standardization Project; an overview of the Evolutionary Power Reactor, EPR, design; status of the development of rules and regulatory guidance in the area of safeguards and security; status of quality assessment of selected research projects; and preparation of ACRS reports.

The meeting is being conducted in accordance with provisions of the Federal Advisory Committee Act. Mr. Sam Duraiswamy is the Designated Federal Official for the initial portion of the meeting.

We have received no written comments or requests for time to make oral statements from members of the public regarding today's session. We have representatives of the State of Vermont on the phone bridge line listening to the discussion of the topics scheduled for today's meeting. To preclude interruption of the meeting, the phone line will be placed in a listen in mode during the presentations and Committee discussion.

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

I will begin with some items of current interest. I will point out you have a package of items of interest that has been presented to you. There are some speeches by the Commissioners of particular interest for our educators on the Committee, and an SRM on the integrated digital instrument and control test facility in the United States that you might want to look at.

I would also remind the members that we're scheduled to interview two candidates during lunchtime today. So don't run off without making arrangements to get back for those interviews.

I'm also pleased to announce the appointment of Dr. Hossein Hourbakhsh as Senior Technical Advisory for Reactor Safety. This is a well deserved promotion, and congratulations to Hossein.

(Applause.)

CHAIRMAN SHACK: We have the following four summer interns who came on board recently. All of them will be here until mid-August. Desiree Davis is a senior at the University of Maryland, College Park, studying psychology and French language and literature. Desiree is a member of the Golden Key International Honor Society --

MEMBER APOSTOLAKIS: We can't see here.

(Laughter.)

CHAIRMAN SHACK: -- and serves as the Vice President of Community Service for the University of Maryland Chapter of the National Society of Collegiate Scholars.

James Clark, III, is a senior attending Virginia Union University in Richmond, Virginia, majoring in accounting. James is a member of Phi Beta Lambda and the Accounting Club.

Kyle Thomas is a senior at the Pennsylvania State University studying energy, business, and finance, as well as economics. Kyle is actively involved in planning and organizing the 2008 homecoming celebration at Penn State.

Eric DiGiovanni is a senior at Penn State University majoring in finance with a minor in psychology. He is currently the president of Phi

Gamma Nu professional fraternity and has overall responsibility for planning and organizing the homecoming celebration.

All of you, welcome aboard.

(Applause.)

CHAIRMAN SHACK: Our first topic this morning will be the ARTIST test program, and Sam will be leading us through that.

MEMBER ARMIJO: Thank you, Mr. Chairman.

The ARTIST test program was going to be reviewed for us by a group of people from the Paul Scherrer Institute, as well as the staff. There was a mix-up in travel plans, and the PSI people will not be here this morning. So the staff will try and cover that entire scope.

The program is titled ARTIST is for aerosol trapping in a steam generator, focused on issues related to aerosols and steam generator tube rupture.

The speakers will be first Richard Lee of the staff, who will make some comments and introduce the subject, and the Michael Salay will carry the ball, I guess, both for the staff and for Paul Scherrer Institute.

Richard.

MEMBER POWERS: Before we get started, I assisted the staff in this area. So I can certainly answer questions of clarification in fact, but if I am asked to provide an opinion, undoubtedly I will. You can just discount it, as you usually do.

(Laughter.)

CHAIRMAN SHACK: Okay. With those clarifying remarks, Richard.

MR. LEE: Thank you.

Richard Lee from the Office of Research.

The office has been participating in this from ten conceptual design of this facility since 2000. We entered into a formal agreement around 2003, participation in not all phases of this experiment because they're about seven or eight phases of the program. Mike will tell you what thaw are. We only participated in the phase of regulatory significance for use for us.

And also this program, the data from there is also supposed to address one of the items under the steam generator action plan Item 3.3(a), and that has to do with getting enough information to look at the sour term attenuation in the secondary side of a dry steam generator, and that has related to the steam generator tube rupture under severe accident conditions, a lot of issues related to that that ACRS was involved with the steam generator action plan.

That 3.3(a) is how we're supposed to provide some information on that. That part has been complete. The separate facts in the experiment has been completed. So last year in November we transmitted a letter to NRR telling what our findings are, and I think Mike will tell you what it is today.

CHAIRMAN SHACK: Okay. Mike, can you just hold for a second? We have to open the bridge line.

(Pause in proceedings.)

CHAIRMAN SHACK: I think we can proceed now.

MR. SALAY: Thank you, Mike.

We'll start with NRC's findings on the ARTIST test in aerosol, retention on the secondary side of steam generators, and I'm first going to go over some background and an overview of the program and then discuss the ARTIST test program pertaining to the steam generator action plan, our major observations about the ARTIST program, modifications, whom they're developed for based on the ARTIST tests, and then I'll show some conclusions, and I guess for that we'll hear from Paul Scherrer where they will present more specifically and more detailed data and some of the risk analyses just to maybe briefly go over some of the data.

Steam generator tube rupture accidents, it's an important bypass accident. It's a design basis event. Plants are designed to cope, and they have for all events to date. I think there have been about a dozen events, and it addresses severe accident only if something else happens, which is interpreted as operator error.

Induced steam generator tube rupture is also a concern. Plants regularly operate with detectable flaws in tubes, and mostly these are stress corrosion cracking, but there's also crevice corrosion at the tube support plate where the chemistry is somewhat different. So there's a limit on flaw size at which plants are allowed to continue operating.

And in the event of a severe accident, the heat transfer from the core to the primary pressure boundary in this weakened structure, some of the vulnerable locations are the hot leg nozzle, the surge line depressurizer, and what we're interested in today is the steam generator tubes.

We currently cannot reliably predict when and where failure will occur.

MEMBER BANERJEE: This would only happen

if there was an operator error or in any case?

MR. SALAY: I think it's expected to be -- what are you asking?

MEMBER BANERJEE: The heat transfer from core to primary --

MR. SALAY: Well, if you have an operator error, but you have more heat transfer, if there is a severe accident, you have release of --

MEMBER BANERJEE: Right, but without an operator error, if it was just a steam generator rupture, would it --

MR. SALAY: Plants are designed to cope, and so without progression to severe accident. So you do get heat transfer, but --

MEMBER BANERJEE: I'm talking about the first point. Would the first point occur without operator error or not?

MR. SALAY: I don't think the temperatures will be that high to --

MEMBER BANERJEE: Won't happen without an operator error.

MR. SALAY: We're ten for ten.

MR. BESSETTE: This is David Bessette.

This is like a station blackout. Oh, sorry. This is like a station blackout scenario where the secondary side dries out and the core overheats and temperatures get very hot.

MEMBER BANERJEE: This is even if the emergency cooling works?

MR. BESSETTE: No. There is no ECCS here.

MEMBER BANERJEE: Somebody clarify it.

MEMBER STETKAR: I'll try to clarify it. The first slide, Slide No. 3, pertains to steam generator tube rupture as the first event. That was the initiating event. It can only progress to core damage if, in simplified terms, if there's an operator error. There could be a bunch of equipment failures, but now he's talking about other scenarios in which the tube rupture is a consequence of the progression of other events.

Those tend to be high pressure scenarios that are progressing in the direction of core damage. So for example, a complete station blackout is an example of that.

MEMBER BANERJEE: Okay.

MEMBER STETKAR: So that could involve operator error. It could involve other equipment failures, but these tend to be high pressure core damage trajectory type scenarios.

MEMBER SIEBER: The important point is the core's damage and the steam generator tubes are ruptured.

MEMBER STETKAR: That's right, in the second case. In the first case the tube rupture is the first thing that happens to make core damage.

MEMBER BANERJEE: But this is a chain of events.

MEMBER STETKAR: That's correct.

MEMBER BANERJEE: It's not just to --

MEMBER BLEY: The tail end of a chain of events.

MEMBER BANERJEE: And the probability of such a chain is pretty low, right?

MEMBER POWERS: No, you can't -- the issue of induced steam generator is that it may be a natural consequence of core damage.

MEMBER SIEBER: Right.

MEMBER POWERS: Okay? But it's not a bunch of events with prescribed probability. Nobody knows the answer to this right now, but it is a subject of substantial analysis.

On steam generator initiated events, we're ten for ten. There have been ten of them. The plants have coped every time, and in fact, I mean, what we've come to believe, as long as you just rupture one tube, it's very difficult for the plant not to cope because the operators have typically lots of time to interact.

As the number of tubes have ruptured to initiate the event goes up, you get to the point where there's not enough time for the operator to act. Okay? And we spent a lot of time in this Committee looking at can you get rupture of a tube propagation that cause ruptures to adjacent tubes, and no one has successfully found a mechanism for that to happen. Maybe it happens naturally, but --

MEMBER SIEBER: Who knows?

MEMBER POWERS: So really interest in steam generator tube ruptures is now focused very much on the induced variety where maybe it's all accidents progress naturally to a bypass accident. But, I mean, that's the subject of research.

Here Mike is going to talk about, okay, if you have this, what are the consequences.

MEMBER BANERJEE: So you fostered it. It occurred.

MEMBER SIEBER: Yes.

CHAIRMAN SHACK: At last the potential, as John said, every high pressure core damage sequence where you get to this point, one of these is going to fail. One of these locations will fail.

MEMBER BANERJEE: Thank you. I think I've got the picture.

MEMBER ABDEL-KHALIK: What does the first bullet mean? The transfer in and of itself does not weaken structures.

MR. LEE: Let me go back. Historically what happened has to do with the station blackout analysis that we have done. Remember all the heat transfer of the Westinghouse 1-7 scale discussion, the hot leg counter-current flow and the steam generator. If you have a loop seal blockage, you will have recirculation back, and this thing is related to that issue.

So we are looking at whether -- you see, we have done a lot of analysis looking at whether the hot leg failed first. You fail at other location and then the steam generator tube. Remember all of those exercises we have done, calculations we have done. Among those, this is sort of implying that the heat transfer weakened the structure either at the hot leg nozzles. It can be at the surge line. It could be at the steam generator tube. So there's a range of calculations. It's very high temperature.

MEMBER ARMIJO: All of this is beyond the scope of this presentation. This presentation starts with a damaged core and a ruptured tube, and you're studying a particular phenomenon that's this aerosol, transferred decontamination and whatever.

MR. LEE: Yes, and this is a dry steam generator because in one of those analyses you postulated one of steam generator secondary site. The safety valve has lifted, and that will close. So you have a drive steam generator scenario on the secondary side.

And the question here is do you get the entrainment of these aerosol and retention of fission products that release from the steam generator tube rupture to the secondary side. That's what he's looking at. That is what this experiment is about.

CHAIRMAN SHACK: But these temperatures are going like to six, to 800 C. at the peak. I mean, so these things are heating up.

MEMBER ARMIJO: But as we get into it, these experiments are conducted at low temperature, and somewhere along the line I'd like the staff to tell me that's important or not important or whatever.

MEMBER SIEBER: It's the chemical and isotopic species that are important when you're looking for the decontamination factor.

MEMBER ARMIJO: Right.

MR. LEE: We are looking at the aerosols. So the size matters and so forth. So I'm sure Mike will discuss this.

(Laughter.)

MR. SALAY: Okay. That's enough of that.

Anyway, there's a diagram of a few natural circulation flows. There are two situations to consider. One regular loop seals are intact and one regular loop seals are open. You have much freer flow when your loop seals are open. Flow can go through the core, directly through your hot leg, through the entire steam generator, back through your cold leg, and back to the core again.

However, when your loop seals are intact, there is more resistance. In the core you have flow going down and up at the same time. You have counter-current flow on your hot leg, and there's flow through some tubes in one direction. In some generator tubes the flow is in the other direction in other tubes.

MEMBER BANERJEE: The counter-current flow in this scenario is just thermally stratified flow, right?

MEMBER ARMIJO: Within one pipe.

MEMBER BANERJEE: within one pipe.

MR. SALAY: Yes.

MEMBER BANERJEE: So you've got the hot stuff going on the top and the cold stuff at the bottom.

MR. SALAY: Yeah, hot on top.

Okay. So in the event a steam generator tube ruptures, the flow would come from the hot leg into the lower plenum, through the tubes, out through a break, up through the outside of the steam generator tubes, passing some support plates, out through your separators and through your dryers, and out by some manner through secondary safety relief valve where it's postulated.

And we look at where could aerosol possibly get retained, and when your flow enters the steam generator tube, there's a contraction in the flow and aerosol can't follow the stream line, and larger ones get preferentially removed and impact on the top of the lower plenum surface.

And you can also get retention inside the tubes themselves before you reach the break. That's a turbulent deposition. It's postulated that immediately in the vicinity of the break turbulent deposition could enhance retention. It's postulated that settling could occur on the top of support plates, and we just have general attention far away from the break.

Thermophoretic deposition on the steam generator envelope was also considered.

MEMBER BANERJEE: Well, what are the particle sizes?

MR. SALAY: I'll go in a second, two or three, a few slides from here, and that's actually a subject of discussion and psi, and we don't quite agree on what --

MEMBER BANERJEE: Because turbine deposition depends very much on the size spectrum.

MR. SALAY: Yeah, I'll go over in a few slides.

And so anyway, they can settle perhaps on top of the tube support plates, perhaps better thermophoretic deposition on steam generator envelope, perhaps retained in the separators and dryers, and then you'd have another flow contraction at the safety valve.

And aerosol retention processes, the removable mechanisms are high size dependent, and for laminar flow, the dominant ones are impacting where particles can't fall. The stream line is going around the flow obstacles, a flow obstacle settling, just falling out, and interception, which just accounts for the fact that the particle isn't a point, but actually has a physical size.

And so if the center of mass goes near enough to the particle, it can interact. For example, acetylene is r-squared dependent. Internal velocity goes up with r-squared.

MEMBER BANERJEE: Settling is?

MR. SALAY: Settling is just falling out. Gravity can receive dust. It falls down.

MEMBER BANERJEE: This is in a laminar.

MR. SALAY: Yes.

MEMBER SIEBER: But the velocities are fairly high.

MR. SALAY: Yes. There are certain regions where velocities are low. So we're looking at regions with high velocity and also regions at lower velocity.

MEMBER SIEBER: Ten to the second or something like that.

MR. SALAY: There's also impaction.

MEMBER BANERJEE: Well, the reason I say this is if the flow is turbulent, settling is much slower.

MEMBER SIEBER: Yes.

MR. SALAY: Yeah, and settling. Yeah,

there are regions where it is turbulent, most definitely, and as your particle gets smaller, you get to the point where they can be moved around by the individual gas molecules and effectively diffused through them, and so very small particles get moved preferentially by fusion, and as your particles get larger and larger, they can be removed by impaction or settling or interception.

MEMBER ABDEL-KHALIK: Are we talking about the primary side of the tubes or the secondary?

MR. SALAY: Just the general. I mean, we are talking -- this is general, anywhere, but the project will be on the secondary side. The primary side, you're turbulent. Your flow is around 100 meters per second. So it's kind of fast.

MEMBER BANERJEE: So you can still get removal by impaction.

MEMBER SIEBER: Yes.

MEMBER ARMIJO: The term "bounce," is there form definition or is it just like bouncing a particle off of a --

MR. SALAY: Well, it was noticed that I think filter manufacturers, that at certain kinetic energies, the particles below a certain kinetic energy, the parties just tend to stick, but above other kinetic energies they just hit and come right off.

MEMBER BANERJEE: It depends on the elasticity of the article and whether it's a dendritic particle. What is it, I mean?

MR. SALAY: I'm just talking about what was observed, and --

MEMBER BANERJEE: It comes up with real aerosols that bounce?

MR. SALAY: yeah.

MEMBER SIEBER: Yeah.

MEMBER BANERJEE: It sounds enormous. I mean, these are what, dendritic structures or what are they?

MEMBER POWERS: It depends on which particle it looks like.

MEMBER BANERJEE: I see. If they're little, hard spheres, I can imagine.

MEMBER POWERS: And some particles are like that. Some have structure to them, and instead of bounce you get break-up when you have structure. So there are really two phenomena, bounce and break-up.

MEMBER BANERJEE: It's an interesting problem.

MEMBER POWERS: It's an intractable problem.

MR. SALAY: And then if you use like olive oil, they bounce less.

MEMBER POWERS: Have you ever noticed that professors who tell you that something is interesting, it's impossible? Why don't you ever get interesting for something that's easy?

MR. SALAY: Then it wouldn't be interesting.

Also, deposits can be re-entrained into the flow, and if particles that have a high kinetic energy cannot only come back off, but they can also knock particles that have already been there, that have already been deposited.

And one thing that these removal processes, they are size dependent, and, therefore, the removal of these particles alters the particle size distribution. The smallest ones get removed preferentially by diffusion and the larger ones get increasingly removed by the other processes, increasing with increasing size, and so you have sort of a size region about tenths of microns that are very hard to remove by any methods, and so your size distribution tends to narrow around this low tenths of micron size, what we call maximum penetration size.

If you do multiple experiments separately and calculate the retention using the same size distribution, you can't simply multiply these values together because you end up double counting the removal of double or triple or repeatedly counting the removal of the largest particles, which are the easiest to remove.

This was one of the reasons that the NRC was very interested in seeing integral tests, and it contained retention as a function of size for individual sections --

MEMBER BANERJEE: Do you have sort of an aerosol code for doing these calculations where you have a size distribution and all of these mechanisms.

MR. SALAY: Typically MELCOR does.

MEMBER BANERJEE: But it's not like a large simulation or anything?

MR. SALAY: No, we don't have it. PSI did a lot of analyses that used some DNS mostly to get coefficients, and they even modified some of their turbulence flow models to account for anti-satrophy (phonetic) near the boundary layer.

MEMBER POWERS: We spent quite a little while setting up an LES model for this particle deposition and the bend in the tube. And what you learn from that are two things. One is that I never want to see another LES calculation in my life, and that typically we calculate from conventional correlations the deposition about as accurately as you can get it from an LES calculation if there's nothing special.

MEMBER BANERJEE: Unless there's a vortex which is --

MEMBER POWERS: Yes. You do get secondary flows that come a little clearer to you physically in these LES simulations, whereas they're kind of hand weighty in the correlations that, you know, put a kink in the curve when you get the secondary flows and things like that.

But so far you have to have really complicated geometry. I'm sure Mike will talk some about flows through the separators and things like that where you've got veins and stuff like that, and we go to heroic efforts to calculate those in detail and find out the deposition is zip, you know.

MR. SALAY: You know, actually I was expecting PSI to talk about that.

(Laughter.)

MEMBER BANERJEE: Who at PSI is doing the modeling work?

MR. SALAY: Dehbi. I don't remember his first name.

MEMBER BANERJEE: Are people like Brian Smith at all involved in this?

MR. SALAY: Don't recognize that name. Dehbi was the -- D-e-h-b-i.

MEMBER POWERS: A graduate of one of the esteemed universities in America located in People's Republic of Cambridge.

MEMBER BANERJEE: I thought you were going to say in the Land of Fruits and Nuts.

MEMBER POWERS: There are no esteemed aerosol businesses in the Land of Fruit and Nuts.

MEMBER BANERJEE: I thought there was one where a guy name Abbott was at, but never mind.

MEMBER POWERS: He has some reputation in that field.

MEMBER ARMIJO: All right. Let's keep going.

MR. SALAY: And so what types of impressions were raised? What types of aerosol size would we get?

Well, a recommendation from IRSN did a survey of some ACL, PBF and PHEBUS experiments, gave this for the size distribution of the steam generator: near log normal and D-1 micron or less, a geometric standard deviation of two larger particles comprising of agglomerates of small .1 micron, highly coordinated clusters, and in two of these tests the aerosol sizes were in the maximum penetration size range, and there was a larger size distribution in the third.

MEMBER CORRADINI: So when you say survey, you mean IRSN looked at aerosols used in those tests or they generated aerosols and IRSN looked at the aerosol machs that they generated?

I don't think I understand.

MR. SALAY: "Survey" is the word that they used, and I'm using similar. They looked at the data from different experiments and micrographs and --

MEMBER CORRADINI: Okay, fine. And then the AMMD, that's aerodynamic something or other. What is that?

MR. SALAY: That's why I say I've seen it.

MR. LEE: Aerodynamic mass mean.

MEMBER CORRADINI: Okay, fine. Mass mean versus number mean versus whatever.

MR. SALAY: I have occasionally seen mass mean instead of mass median, and so --

MEMBER CORRADINI: Okay. That's fine. I

just didn't remember what the acronym was.

MEMBER BANERJEE: This must depend very much on the generation mechanism, right? I mean the generation mechanism independent.

MEMBER POWERS: I suspect when you see things like this with a sigma 2 and what not, what you're looking at is the product of a nucleation growth mechanism and then transport through some removal process that smoothed up the distribution. Because it's so close to the maximum penetration size, I suspect that you've gone through structures and whatnot knowing the tests. I happen to know that that's the case, but just looking at it you'd say, yeah, this is -- because it's not multi-modal, because it's not broad, all of the details of generation have been wiped out by getting to where --

MEMBER BANERJEE: So some sort of equilibrium, something like a Boltzman distribution, which is --

MEMBER POWERS: Something like that. It's a log-normal distribution.

MEMBER POWERS: Everything's a log-normal distribution if you plot it crudely enough.

MEMBER CORRADINI: I mean, another analogue to this is if you look at essentially particulate emissions from a combustion engine, after you go through all of the manipulations there and they pass into the whatever it is, the catalytic converter, it's essentially like that. It almost has the same general character. It may have a shift in the log normal, but it looks kind of like that.

MEMBER BANERJEE: This is something which is not near the generation point that has had the change to reach sort of equilibrium of some sort.

MEMBER SIEBER: It's the PHEBUS experiments, I think, that gives you the initial composition. Then a lot of mechanical things happen before it gets to the atmosphere, which gives you the decontamination factor.

MR. LEE: Yeah, from PHEBUS it's basically what my sighting is the size observed from looking at the steam generator surfaces. That means the generator in the core bundle and after the upper head and pipings and then go through a single tube stimulator, and they're looking at the size. That's what you're talking about.

MR. SALAY: Okay.

MEMBER BANERJEE: What are the largest particles there? I mean, I can see the mean is about a micron.

MR. SALAY: There's a distribution. So you --

MEMBER BANERJEE: Right, but it's sort of a log normal. So it is a long tail.

MR. SALAY: It is a long tail.

MEMBER BANERJEE: But what is the largest size that you have?

MEMBER POWERS: The largest size you have are samples about 20 microns.

MEMBER BANERJEE: Okay.

MEMBER POWERS: Okay? Now, in principle there are even larger particles than that, but you can't get them into a sampling device. So you really don't know too much about it. But because it's a mass median there are not very many of them.

MEMBER BANERJEE: So what you do is you do isokinetic sampling.

MEMBER POWERS: You try to do isokinetic sampling. Now, in core degradation tests, the problem is your flows are not necessarily constant, but what you want is a forgiven inlet nozzle, and people spend a lot of time designing goosenecks that are forgiving so that you get a good sampling, a good representative sample, and I would guess what did it take PHEBUS, three tries before they actually got decent samples coming in?

It takes a while. I mean it's hard to do in dynamic tests.

MEMBER ARMIJO: Okay. Just one question. We've talked about size, but does the particle density and the chemistry of the aerosol particle, do these things make any difference in decontamination?

MR. SALAY: The aerosol mass mean diameter is an indication of the size of a unit density sphere that would fall at the same rate as the particle in question. So, yes, there are shape factors. It depends on they're agglomerates.l There's questions whether they're stringy or compact, and, yes, that does affect them.

MEMBER POWERS: What we all deal with are models everywhere saying an aerosol particle is an aerosol particle, and there's really no chemistry associated with it, and if you look back on the issue, Bender is correct. It depends on the coefficient of the institution on that and on the magnitude of the Van Der Waals forces and things like that.

That's a level of detail below the resolution of any severe accident. One of the issues that's raised is do we need to go to another level of detail to model things like that, break-up and things like that, and we'd be interested in your comment on that because it's very difficult.

MEMBER ARMIJO: I imagine so.

MEMBER POWERS: I mean, for 25 years the assumption that this is a physical phenomenon and that if a particle comes in and gets close to the surface, little hands grab it and hold it to that surface dearly, and it didn't do anything else. And as we go through the discussion we'll see, well, got you, and it's how much you want to explore that approximate party. I think it's interesting.

MEMBER ARMIJO: All right, Michael.

MR. SALAY: Okay. So then the consequences of improved rupture, nuclides went directly to the environment of the auxiliary building without any attenuation from generic safety features in containment, and even though the accidents are of very low probability, they are risk dominant.

MEMBER BANERJEE: Yeah, risk dominant. That's interesting. This is actually an important thing.

MR. SALAY: And from NUREG-1150, which is risk analysis of five U.S. plants, three BWRs and three PWRs and two BWRs, two of the PWRs had significant probabilities of tube rupture and all were found to or were determined to be able to suffer induced steam generator tube rupture.

However, there was data -- data were unavailable on retention on the secondary side of steam generators and there weren't really any models available. Essentially they wanted to credit some retention, and they convened an expert panel to come up with some values.

MEMBER BANERJEE: Don't go so fast there. Expert panel to --

MR. SALAY: A source panel.

MEMBER BANERJEE: What is it?

MR. SALAY: A source panel to determine if they come up with some values. There weren't models. There wasn't any -- data were unavailable, and so hence they convened a group of experts to say, well, give us your opinion on the potential.

MEMBER BANERJEE: This seems a deterministic problem for an expert panel to be able to calculate this stuff badly.

MR. SALAY: Well, to complete their risk analyses, they had to -- they wanted to come up with an estimate because the release was --

MEMBER BANERJEE: Is this a way out every time a calculation is difficult? You convene an expert panel?

MR. SALAY: Well, and then to go back and try to get data later.

MEMBER POWERS: I mean, all of them whenever they came to an uncertainty, it's the only one they challenge, and they lacked confidence in the old source term crude package. They set up a panel, and they said, "Okay. You guys are the experts. Do your own calculations, communicate with the angels, whatever it takes to give us a distribution on what the likely outcomes are."

In the source term what were there, six, seven questions, distinct questions that they posed? You know, things like what are the release fractions, what's the transport fractions, and things like that.

You know, in principle every one of those can be calculated. They did it in the source term code package. They lacked confidence they were doing it very well, and so the issue came back. Okay, yeah. We've been spending an enormous amount of time on each one of these questions. Does it make any difference?

And of course, the conclusion was to spend some time and, of course, that led to the genesis of the first VICTORIA code and then the MELCOR code to try to do these things better.

This is just one of the questions that had eventually to do with risk. I mean, there were literally hundreds, but big panels were set up on them.

MR. SALAY: And this is what they came up with. In terms of a decontamination factor from DIA, just simply mass coming in to mass going out, and for the inlet efficiency for steam generator plenum in the ruptured tubes, they came up with a decontamination factor of two.

For the retention in tubes, they calculated a decontamination factor of ten. However, there were concerns about suspension, revaporization and glomerate break-up, and therefore, no credit was given for this.

For the secondary side, they came up with a DF of about four to six with no deposition on the opposite tube for viewer resisted by thermophoresis, and no credit was given for the steam separators and dryers because of the proprietary side of the question. There was large uncertainty --

MEMBER BANERJEE: For what reason?

MR. SALAY: They were having difficulty to get information on it was proprietary, the steam generator. The vendors were not unwilling to release information on the problem.

MEMBER POWERS: Yeah, the plant itself was willing. For instance at Surry, they were willing to give us anything. The vendor for the separators and dryers, however, objected to release a sufficient detail to do an aerosol analysis. In the end I don't think it made very much difference, but it was a challenge.

MR. SALAY: And there was a large uncertainty in these estimates, and here the risk break-up for surry, and as you notice the bypass accident, which is shown in red, is dominant for early fatalities and latent cancer fatalities.

Then industry came along and came up with an alternate retention analysis, and that was much higher. They came up with a decontamination factor on the secondary side of the steam generator on the order of 10,000 and a DF of 100 or more on the tube depending on where the break was, several tens on the secondary near the break, and about two to three far from the break. And so very different analyses.

And NRC's attention on tube rupture bypass accident is justified by risk, and there's a direct connection between risk for bypass accidents and source term attenuation on the same resized steam generators. As the attenuation goes up, the fraction on risk goes down, and the slice of the pie just decreases.

MEMBER ARMIJO: In the industry analysis, did they use different visits? How could they come up with such big differences?

MR. SALAY: There was a lot of turbulent deposition in the tube. They didn't -- I think above a certain size they assumed that deposition was constant. They didn't account for balance. They even considered that perhaps the aerosols collected a clog and --

VICE CHAIRMAN BONACA: Did they consider steam dryers?

MR. SALAY: I don't think the industry calculation did.

MEMBER BANERJEE: So there is no water in this system at all.

MR. SALAY: No, it is assumed to be dry. So they ended up with this big outstanding question: are safety resources being misdirected to an unneeded attention on containment bypass accidents because we underestimate attenuation, and this resulted in steam generator action plan Item 3.3(a), develop experimental information on the source term attenuation on the secondary side of steam generators.

And the ARTIST program came along, and this is an international project conducted by Paul Scherrer Institute, seven-phase project of which NRC participated in five. It consisted of both separate and integral tests, and you see the diagram here.

MEMBER BLEY: Is it complete now? The seven phases are all complete?

MR. SALAY: Yes. I know they did a few more tests earlier this spring, I think the most part of it is.

And the retention was measured in different locations. Each of these corresponds to a phase -- in the steam generator tube prior to reaching tube rupture, in the immediate vicinity of the break where particles could impact on adjacent tubes, on tubes far from the break --

MEMBER BANERJEE: Are those numbers in brackets --

MR. SALAY: That's how many tests were provided to us in January. I think two more tests have been done.

MEMBER BANERJEE: When you mean separate, what do you mean by "separate"?

MR. SALAY: Separate so that they take,

for example the first one is a separate test. They just take one tube and it actually -- just one big long tube and they just look at the retention. They don't actually use this facility. They just have one tube and look at the retention inside.

MEMBER BANERJEE: So there are separate effects.

MR. SALAY: Yeah, separate, and then they have a bundle only. They have a completely different facility for far field with a few different tube support plates. So they have separate facilities and then they have the whole.

And so one of the facilities was the in-tube retention. They have separate tests for in the immediate vicinity of the break, a separate test for on the tubes, between one tube's support plate and another, and also on top of the support plates, and then they have tests, the steam separate and steam dryers, and then they had combined tests with all of the components.

And the other phase is that we're going to participate on by the NRC where retention in the flooded bundle and droplet retention in --

MEMBER BANERJEE: What does flooded bundle mean?

MR. SALAY: When you're not assumed to be dry, when your bundle actually does contain water and so the aerosol --

MEMBER BANERJEE: Some amount of water or a lot of water?

MR. SALAY: They tried different submergences, and they sort of do this function of submergence.

MR. LEE: NRC did not participate in the flooded bundle part because we know that aerosol retention in water is extremely good.

MR. SALAY: Very high.

MR. LEE: So we said we really don't need to worry about that.

MEMBER CORRADINI: How do you have bypass with water there? How do you even set up the conditions? They seem counter. They seem inconsistent.

MR. SALAY: I wasn't involved in the start of the project, but that could be why we didn't buy into those.

MR. LEE: So that part that we did not produce, they will not give us those data because you can see the southern part of it. So we produce the part that are all dry.

MEMBER CORRADINI: All right. Thank you.

MEMBER POWERS: Many, many of the European plants have or contemplate accident management strategies to avoid flooding the secondary side of the steam generator. In fact, the Sizual (phonetic) plant has a hard-engineered facility which would operate on the secondary side. I don't know of any U.S. plant that has that capability, and as Dr. Lee said, it's not one that I would spend an enormous amount of time calculating. If you'd flood the secondary side, you're going to get very little aerosol through that.

MEMBER ARMIJO: If that's so effective, why don't we do it?

MEMBER POWERS: well, you have a cost associated with it.

MEMBER CORRADINI: It would be a dedicated system

MEMBER POWERS: It would have to be considered in light of the fact that the probability of one of these events is about three times ten to the minute six. Now, the consequences of it are enormous, but so -- I mean, how much money do you want to spend on a three times ten to the minus six event?

MEMBER BLEY: How much does it cost to hook up a firewall?

(Laughter.)

MEMBER POWERS: If you have water, where are you going to put it? Are you going to put it in the core or are you going to put it in the secondary side?

(Laughter.)

MR. SALAY: The ARTIST facility is based on best now plants, and --

MEMBER CORRADINI: Did the NRC staff participate in the scaling?

MEMBER POWERS: Yes.

MR. SALAY: I wasn't around at the time. So I couldn't answer.

MEMBER CORRADINI: Thank you.

MEMBER SIEBER: The answer is it's almost as bad as we thought it was.

MEMBER CORRADINI: But ARTIST is separate from PANDA or is this a subcomponent of the PANDA facility?

MR. LEE: A separate thing. This has nothing to do with PANDA.

MEMBER BANERJEE: Completely different.

MEMBER CORRADINI: But I was just curious about if it was a component of PANDA that they essentially tested separate.

MEMBER BANERJEE: I haven't seen this facility. So it must be in a separate building.

MEMBER POWERS: An orgy of stainless steel.

MR. SALAY: The facility is based on the Beznau plant. It's 365 megawatt electric, Westinghouse two-loop PWRs, 69 and 72. It's scaled for the steam generator tube rupture accident, about two centimeter tube diameter. It's approximately 120 by flow area, and the main facility or the bundle is a short and narrow bundle. The total height is 10.5 versus 17, but for the tubes it's three-something versus nine. It's somewhere on here.

MEMBER BANERJEE: What was the rationale for this? Because I can see reducing the number of tubes, but why would you reduce the height?

MR. SALAY: My guess is cost, but --

MEMBER POWERS: The height of the building that's involved is huge.

MEMBER SIEBER: Now, the Beznau plant is similar in design to Gennay (phonetic), two-loop.

MEMBER POWERS: Two-loop BWR, but pertinent to this, it has brand new steam generators.

MEMBER SIEBER: Yeah. It doesn't make any difference whether it's two, three or four.

MR. SALAY: Okay, and the main facility also contains a tube sheet and three support plates and a full-scale separator and dryer. It contains one of these, whereas the plant steam generators each contain 12, and separate effectually is they're making four of the facilities into at the break, following the break in support plates and for separator and dryer.

MEMBER BANERJEE: The surface area to volume ratio is the same, or is it?

MR. SALAY: Surface area to volume, yeah.

MEMBER BANERJEE: In rough terms.

MR. SALAY: They're the same hydraulic diameter, the same pitch.

MEMBER BANERJEE: You have the same hydraulic diameter, same velocities, whatever.

MEMBER POWERS: The critical issue in the scaling is if you have a break, you have a jet going out, there is a jet through the tube and affect the shroud that you use around the facility or is that flow dissipated sufficiently to start moving all upward. For quite a while -- and they're very quite on there. They're not bad.

Now, as far as the height, steam generators, we typically treat them as a bunch of units with the tube support plate marking the boundary of those units. All we need is a couple of those, and it looks kind of the same. The tube ruptures that we have seen are kind of uniformly distributed up and down the tubes. You're as likely to break at the bottom as you are at the top.

There was a lot of agonizing about whether you got guillotine fractures or fish-mouth fractures, and what we have learned especially from Dr. Shack is the ones that are the biggest danger are the guillotine breaks within the tube support plates. The more likely ones are fish-mouth breaks within the spans.

Okay. So you look at those things.

MR. LEE: In some of these break geometry was actually prepared. Argonne with Dr. Shack's help actually, and we should go back to --

PARTICIPANT: Operated fish-mouths.

MEMBER POWERS: You don't want to hold that against the Swiss program. They did the best they could. They saw the best offer they could for --

(Laughter.)

MR. SALAY: Okay. Test parameters for those tests are guillotine break. They used a few different aerosols, TI, titanium dioxide agglomerates, and they used two different manufacturers because they were having difficulties controlling the size. The agglomerates, they were having difficulties breaking them up, and so they took a different brand, went to a smaller size, but still couldn't reduce it to the desired size, and so they ended up going to silicon dioxide spheres, which are products as you can see, and figured in the break.

MEMBER BANERJEE: Nothing like it.

MR. SALAY: They're really neat, and they also used latex spheres. You have a titanium dioxide agglomerates and silicon dioxide spheres. These two figures are on the same scale.

And the types of concentrations they used were on the order of .0 hundredths, two-hundredths of the milligrams per meter tube, a flow rate of cold nitrogen, and some tests had steam, and the flow rates of a few tens to several hundred kilograms or hour inside a tube because I mentioned before the velocities ended up being hundreds of meters per second.

And they performed scoping tests to determine what parameter they should use before settling on them, and they also repeated some tests to determine experimental uncertainty.

MEMBER BLEY: So the highest temperature is around 100 C. or a couple hundred at most?

Do we know if --

MEMBER POWERS: The highest temperature was like 327 degrees Centigrade.

MEMBER BLEY: It wasn't steam? Did they evacuate?

MEMBER POWERS: No, they didn't. There may have been some water vapor.

MEMBER BLEY: But nothing high temperature. Do we know if that makes a substantial difference in any of this?

MEMBER POWERS: Well, your transport properties change a little bit, I suppose.

MEMBER ABDEL-KHALIK: I can understand how during a transient of this type you have high temperature on the primary line, but could you explain to me when during this transient the pressure on the primary side will be higher than the pressure on the secondary side?

MR. LEE: In the severe accident stimulator, the secondary side is very low. That's why you have very large damage index on this tube, and that's why it failed, because of the high pressure to the secondary side.

MR. BESSETTE: But the primary pressure is about the 2,500 psi?

MEMBER ABDEL-KHALIK: He's talking about a guillotine break where, in a tube?

MR. SALAY: A tube, yes. This is for a test.

MEMBER ABDEL-KHALIK: But the initial event was what? What caused the primary to lose inventory completely?

MEMBER SIEBER: Probably a hot leg break.

MR. LEE: It based on a station blackout scenario.

MEMBER ABDEL-KHALIK: So you have a pump seal failure.

MR. LEE: Yes, pump seal leaking and so forth.

MEMBER ABDEL-KHALIK: And that's how you lost inventory?

MEMBER POWERS: No. You get a secondary side bypass. So you open up, say, a relief valve. The primary side is still a full pressure. It's leaking out at 2,500.

MEMBER ABDEL-KHALIK: Oh, is it?

CHAIRMAN SHACK: And no injection. You have no feedwater.

MEMBER POWERS: The core is just melting. The temperature is high and the temperature is going to six, seven, 800 C.

MEMBER STETKAR: The primary pressure relief is through the break through the secondary relief valve. So that's a driving process.

MEMBER MAYNARD: I thought the big problem was with the secondary side being dried out.

CHAIRMAN SHACK: As Richard said, that gives you the maximum pressure across the tube. You know, you've got much higher stresses on the tube.

MEMBER MAYNARD: How are you getting it dry around the tubes?

MEMBER POWERS: What happens is that you rupture a tube. You're now putting in primary side pressure on the secondary. The secondary side safety relief valve is open,a nd it just blows the water right out of the tubes. You don't have any feedwater to make up. So you go dry, and it goes dry very quickly. Twenty minutes and you're dry on the secondary side.

In a station blackout, you can't have it around the port or you go dry. There's no natural convection of heat transfer. Until you go dry, you're not melting the core.

MEMBER SIEBER: In a severe accident you have to break that cool --

CHAIRMAN SHACK: Before you go dry on the secondary side you don't have any problems.

PARTICIPANT: There's not much liquid water left, I suppose.

PARTICIPANT: It's getting pretty hot.

CHAIRMAN SHACK: No, but I mean that's why you keep trying to pump water into that secondary side.

MEMBER POWERS: I mean, if you don't have feedwater, you're not putting any water in there.

CHAIRMAN SHACK: Right.

MEMBER POWERS: Where you're trying to put water in is to the primary side.

CHAIRMAN SHACK: No, but I mean in scenarios where you have the auxiliary feedwater, until that pump dies you're okay. Once that pump dies then you're dog meat.

MEMBER POWERS: Yeah.

MR. SALAY: Okay. And here's some of the primary measurement methods. They look at the size distributions.

CHAIRMAN SHACK: You have a half an hour left, right?

MR. SALAY: Heavy concentration, the routine mass, and from that your decontamination factor. They sampled at the inlet and outlet for all tests, and occasionally you had other locations. They determined the size distribution with cascade impactors or low pressure impactors and optical particle counters. Concentration of the filters, odometers and optical particle counters.

They looked at the mass collection in addition to concentrations in combination with flow to determine its contaminating factor, and they measured several other parameters.

The major observations from the test program was that there were two forms of aerosol deposition. There's always a fairly uniform layer of fine aerosol on surfaces exposed to aerosol laden flow.

MEMBER BANERJEE: Even with the little spheres?

MR. SALAY: Yes, yes. And in some of the tests there was also clumps of material.

MEMBER BANERJEE: This uses both the titanium dioxide --

MR. SALAY: They had tests with titanium dioxide, silica dioxide and latex. They used silicon --

MEMBER BANERJEE: All of those formed this tenacious list, even the latex.

MEMBER POWERS: Well, the latex is hard to see it because latex is damned expensive they don't run very much through. But, yeah, everything gets a patina on it.

MEMBER BANERJEE: I have a way to generate micron scales from seismic cheaply. They use it for particle imaging velocity.

MR. SALAY: The in-tube retention seems to vary from test to test significantly, and there was also, and I guess I'll show later, that there's high retention immediately upon when the aerosol flow was started, but then the retention dropped off. Resuspension was observed in experiments, indicated that bounce and break-up were important. Break-up in the tubes was noticeable. Large agglomerates didn't survive the transport.

To high flows, particles larger than about one micron would break down to submicron and have a particle smaller than about one micron didn't break up.

Near the tubes there was -- near the rupture there wasn't a significant amount of retention on the tubes.

MEMBER BANERJEE: You mean on the secondary side.

MR. SALAY: On the secondary side, yes, and sort of following the path of the flow.

And far away from the break most deposit mass was on support plate, and the tube's floor plates used broached holes which had a big flow area right there, and so you'd have flow recirculation and a region of low velocity where they could settle out.

And however, for most of the U.S. plants they have drilled holes, and there's a lot less area in between, and it could be filled with crud.

MEMBER CORRADINI: Were any of the test results surprising? I assume people did pre-calculations of what they expected in these tests versus what they measured. So are there any surprises in terms of the physics they saw versus the physics they guesstimated?

MR. SALAY: I think it was pretty much what they expected. The spread of the plume was lighter than expected, but I think basically perhaps the retention -- I wasn't around at the time, but perhaps the initial behavior was complete in the tube. The pressure drop actually dropped. The pressure across the tubes as the flow was going across actually dropped. So you started with some flow across there, and you have some pressure drop. Then when you start injecting the aerosol, the pressure drop increased across the tube. So there was less resistance. It sort of smoothed out.

MEMBER BANERJEE: But they didn't support the industry position or did it support this?

MR. SALAY: No. The bottom line is it supported the expert position.

MEMBER SIEBER: You could almost say we didn't learn much new, but we learned enough to be able to modify.

MR. SALAY: To be more confident about our results, yeah, and that's really --

MEMBER SIEBER: For the answer.

MR. SALAY: There wasn't a lot of retention even with large aerosols in the dryer and separator, and things we're interested in learning more about are bounce, break-up, and the adhesion forces that cause them to hold together or not break up.

Understanding resuspension, thermophoretic deposition, and shapes and sizes of particles coming from the grade in reactor core.

MEMBER BANERJEE: So MELCOR uses a series of is it a 1D model?

MR. SALAY: I think it is classified as 1D.

MEMBER BANERJEE: One D, and it has some empirical correlations for deposition and resuspension.

MR. SALAY: It actually calculates the size distribution. I think, first of all, it emits the fission products as vapors which condense and then agglomerate, and --

MEMBER BANERJEE: And you come to this sort of equilibrium size --

MR. SALAY: Yes. Actually it calculates the individual processes that affect the size distribution.

MEMBER BANERJEE: So that they do the early stage, but now you've got this sort of log-normal distribution coming out.

MR. SALAY: It's sort of how much retention. I mean, MELCOR doesn't model the secondary side in extreme detail. We found that even many of the people in the honors project, they did model it in very much detail with CFD codes and didn't --

MEMBER BANERJEE: CFD codes are not worth the paper they're written on.

(Laughter.)

MEMBER POWERS: And I think we'd agree, right?

MR. SALAY: Yeah. There are lots of colorful plots.

MEMBER ARMIJO: I think we --

MR. SALAY: Should I?

MEMBER ARMIJO: Yes. How did it affect your rating?

MR. SALAY: MELCOR for the secondary side through the lambda factor based on the particle size from the integral test, and we believe there's an insufficient risk incentive to do more work, although we're keeping our eye out on other models that are being developed out there, as well as one developing.

MEMBER CORRADINI: Just that one point. So to kind of follow up Sanjoy's point, when you model with MELCOR on the secondary side of the steam generator, I know the user has flexibility, but historically people kind of just stumble and use the previous model. So what is the typical model of a steam generator with MELCOR relative to this? Are they relatively large lumps in terms of essentially the whole bundleous A node or do they actually break it up?

Because the answer kind of to his question is it's one dimensional if you force MELCOR to look at it one dimensionally versus just a large lump of the bundle.

MR. SALAY: Not your typical model.

MEMBER POWERS: If you pull MELCOR off the shelf right now and say, "Okay. Tell me what the decontamination is. Just run the code on the standard problem," your decontamination is what? Decontamination factor.

MEMBER CORRADINI: What goes in comes out.

MEMBER POWERS: Yeah, because nobody has ever bothered to model it.

MEMBER BANERJEE: And what is it --

MEMBER POWERS: What they are proposing is right now based on these experimental results is just for the lambda factor.

MEMBER CORRADINI: And the lambda factor, assuming you have this lumped model, would say based on some set of conditions it's greater than one.

MEMBER BANERJEE: You can see that it's based on the size distribution.

MR. SALAY: These are the three integral tests that we have results from.

MEMBER CORRADINI: And then that leads me to the C-MAT results. If C-MAT learns something by doing a more sophisticated model, is it open to you guys or C-MAT essentially closed -- their results are closed to the ARTIST community?

MR. SALAY: Well, they're developing a model, and I assume they'd release it.

MEMBER CORRADINI: Well, they may and they may not, I mean.

MEMBER POWERS: We have a very close working relationship.

MEMBER CORRADINI: You do? Okay, fine.

MEMBER BANERJEE: You can participate in that model development or not?

MEMBER POWERS: We do. I mean, there's an active collaboration.

MEMBER BANERJEE: And it's sort of open source code?

MEMBER POWERS: Very. Essentially data, you know, actually.

MEMBER BANERJEE: In which way will it differ from the other, the ARTIST model?

MEMBER POWERS: There is no ARTIST model.

MEMBER BANERJEE: ARTIST data.

MEMBER POWERS: What C-MAT is looking at

is can they take some fairly well known correlations for flows or perpendicular to vertical feet and figure them so that they predict this patina that's observed and things like that, and they make extensive use of fluid calculations and things like that.

MEMBER BANERJEE: That's terrible.

(Laughter.)

MEMBER POWERS: Only to understand what's going on from the flow. They have limited confidence in the ability to use fluid to predict aerosol behavior.

What they would like to do is end up with a correlation based decontamination factor for this near field decontamination, and they're looking at lots of inertial impacts and results that have been obtained in the past and things like that. They've done some interesting experiments in which they were trying to understand the flow -- experimental in nature -- understand the flow distribution around the break, and they quickly found out that particles don't come up to the speed of the gas very closely.

In fact, a surprisingly long time to accelerate the particles, and it was frustrating. What they built was a scaled down version of the ARTIST experimental facility, did quite a lot of flow mapping in this, discovered that the particles weren't coming up to speed, and all of this added intuition on how to apply some of these steady state models of this contamination.

I mean, Mike has characterized it correctly. We're paying attention to this. We're helping them where we can, and if they come up with something, you know, we can see going beyond the lambda factor because it's compatible with the MELCOR coding.

And in fact, they have the MELCOR coding used to provide a model for the experiments.

MEMBER BANERJEE: So the bottom line here that the industry has on contamination are way too high.

MR. SALAY: Their calculations were, yeah.

MEMBER BANERJEE: And you're trying to sort of capture some of the decontamination from these integral tests which lie somewhere between one and, say, 50, whatever, depending on particle size 20.

MR. SALAY: yeah.

MEMBER BANERJEE: And this includes the dryers or everything?

MR. SALAY: Yes, separators, dryers.

MEMBER BANERJEE: And in the SOARCA or

whatever that is, what sort of estimates.

MEMBER CORRADINI: Funny. That's what Dennis was about to ask.

MR. LEE: I think Charlie Tinker and company knows about this results or they're looking at it on the secondary side where retention is, yeah.

MEMBER BANERJEE: At the current time SOARCA is using one.

MR. LEE: I do not know to answer that one.

MEMBER BLEY: Or are they using the industry average?

MR. LEE: I don't think they're using the industry one because the industry one, I believe, was using very large particle size. That's why they have to use 10,000 and so forth. That is understandable. So nothing wrong from the aerosol point of view.

MEMBER ARMIJO: So has this lambda factor officially been incorporated into MELCOR if the staff would do any analyses?

MR. LEE: I think that can be incorporated into MELCOR secondary side very easily. You just put a control function and you can calculate it any time you want.

MEMBER ARMIJO: That's the final --

MR. LEE: That's correct.

MEMBER ARMIJO: -- output of this program as far as NRC is concerned.

MR. LEE: Yes, that's correct.

MEMBER SIEBER: Well, I think the question is the ARTIST program is finished now, right?

MR. LEE: Yes, it's finished. This phase is finished.

MEMBER SIEBER: But there is a continuation beyond that.

MR. LEE: I think that the ARTIST-2 that's being proposed is under the -- they plan to present it to us. They present it to Mike at these meetings many times. So maybe Dana can discuss very briefly.

MEMBER SIEBER: Have you made any decision about whether you're going to participate?

MR. LEE: I can tell you my view is not to participate.

MEMBER SIEBER: Okay.

MR. LEE: I do not speak for our management though.

MEMBER ARMIJO: So it's still under consideration then.

MR. LEE: Yes, correct.

MEMBER SIEBER: What are they going to do in the new part of it?

MR. LEE: I think the new part is, if I'm not wrong, they're going to do some more in tube break size, break stage, some flooded bundle tests and flooded separator. So in other words, all of the tests they propose to do are a further extension of some of the things that we already participate in. Our view is that giving the small particle, I think the particle just stay with the flow. There's no reason for the particles to do more work getting out -- aerosol to get out of the flow stream and pack itself onto something else. So we don't think this DF factor going to change anything even if they do more tests. That's our view.

They have to prove us wrong.

MEMBER SIEBER: Based on the low frequency of occurrence.

MR. LEE: Yes.

MEMBER SIEBER: A decision in this area, beyond that which we've already accomplished, probably doesn't add too much to the picture.

MR. LEE: Correct. This is what is our tentative view at this time, but it is under discussion with us.

MEMBER BANERJEE: I suppose if all the

particles were large, it would have an effect.

MR. LEE: Yes.

MEMBER BANERJEE: You would grow a substantial tail.

MR. LEE: But as you move to every plate, you know, you become smaller. The population becomes smaller and smaller. So you just cannot keep on counting on large particles every stage. Doesn't exist. You can look at the physics itself, right? It makes sense to you. You don't have to do an experiment to find that out.

MEMBER ARMIJO: Well, you might as well get your conclusion chart, Mike.

MR. SALAY: I think conclusions are basically expert panel recommendations for the NUREG-1150 risk analysis were by and large confirmed. MELCOR predicts a contamination factor similar to those that --

MEMBER ABDEL-KHALIK: Does the first bullet include or exclude the factor of ten, decontamination factor, in the tubes that was excluded?

MR. SALAY: Well, it excluded the factor of ten, but there was uncertainty there, and so the uncertainty remains, and so even with these experiments that uncertainty remains.

MEMBER BANERJEE: That is what is reducing that uncertainty. That's a fairly substantial decontamination.

MR. SALAY: We're interested in the studies of break-up and agglomeration, and I don't think their follow-on projects went in that direction.

MEMBER BANERJEE: Now, that, the decontamination factors are what, on the order of ten or something? I've forgotten.

MR. SALAY: Oh, there were ten --

MEMBER BANERJEE: I have to go back and look.

MR. SALAY: The prediction was ten or less for the ARTIST test. I mean there were small periods where it spiked very high, but then came back down, and then some of it --

MEMBER SIEBER: The biggest thing I saw was 1.3 for DF. I might not have seen all of them, but that's the one that --

MR. SALAY: See, it actually went -- for short periods of time it spiked quite high.

MEMBER BANERJEE: But then they were resuspended.

MR. SALAY: And there's also concern that in the reactor you have heat deposition which actually revaporize some of the material, you know.

MEMBER BANERJEE: But these are titanium dioxide?

MR. SALAY: No, no, no. I was referring to a real reactor accident.

MEMBER BANERJEE: What would be the present reactor?

MR. SALAY: Fission products. They carry with them their heat.

MEMBER BANERJEE: Like plutonium or something?

MEMBER POWERS: Cesium iodide.

PARTICIPANT: There's a whole laundry list.

MEMBER BLEY: Is it pretty well established that these surrogates we're using in these tests will behave similarly to the aerosols we'll get out of a core as it degrades?

MEMBER POWERS: I would say the evidence is here. We have never seen any aerosol behavior in a reactor accident that suggests anything different than this more mechanical modeling.

MEMBER BLEY: We don't have a lot of

experience on the reactor accident side.

MEMBER POWERS: That's right.

MEMBER BLEY: However, there used to be people who said there would be chemical effects. You'd get lots of agglomeration and stuff.

MEMBER POWERS: We do get lots of agglomeration. It's not a chemical effect. It's a mechanical effect.

MEMBER BLEY: Okay.

MEMBER POWERS: We've melted down a lot of fuel assemblies now. We see aerosols coming out. The assumption inherent in all of the aerosol codes at anybody's, you know, NRC's, NOAA's, everybody used that an aerosol protocol is an inert beast, and it behaves inertly.

We know that's not true. We know that an aerosol particle has a Vanderwol's attraction (phonetic) to things. We have a very limited database on Hamaker constants to calculate that, but we said it's not important. You can come in and do it by simply saying that it's a mechanical process, and you can treat it as a mechanical process, and consequently it didn't matter what aerosol particle you use as a surrogate because it's an inert thing.

MEMBER BLEY: But it would be nice to have the same kind of shapes and charge distributions.

MEMBER POWERS: You would like to and especially if you knew what those things are. You know, Mike mentioned some recent tests that have been done. One of the issues is what happens when particles are charged. Does that change things at all?

And certainly Hans Jordan and Jim Geesik will be looking at BWR separators and dryers. When they failed to discharge the aerosols, they got different deposition and when they did make sure that the particles were uncharged, and some of the fallout was, "hey, I've looked at these kinds of issues."

MEMBER CORRADINI: Can you repeat that again about what they saw? I'm sorry.

MEMBER POWERS: Hans Jordan did one experiment with a BWR separator and dryer where he simply by omission failed to run things through his electrostatic discharge unit. So the particles coming in had a non-Boltzman charge distribution. He got different deposition patterns in that separator and dryer.

Now, he was using relatively large particles, around five microns so they could carry a charge in case there was no natural drive to a

Boltzman distribution in the experiments.

And so charging has always been an issue, electrostatic charging, one of what we wrestle with because the radiation field -- the reason particles can get charged in a reactor accident is because of the radiation field. It's not because they're radioactive.

And typically what you argue is that in close geometries there's a discharging off the surfaces so that there's not much electrostatic charging, and most of the concern about electrostatic charging has been in the containment where things are not closed.

But there have been some experiments. I don't happen to have the results on electrostatic charging in this ARTIST. I don't know what the results are. That's another issue, but otherwise what one does is in the aerosol codes is assume that the particles are inert beasts. Okay? Any aerosol particle will do for an aerosol experiment.

And my statement to you is we've never found evidence that contradicts that assumption.

MEMBER BLEY: In any experiment?

MEMBER POWERS: In any experiment. Now, there are some arguments that if you can get very hydrophilic aerosols, that they become mushy, you know, solution, sticky. They will behave a little differently, and there will be less tendency to bounce.

Silicon dioxide particles and the latex spheres that have been used should have very high coefficients of restitution. So they'll be bouncy.

Okay. The titanium dioxide particles because of their structure won't be very bouncy, but they'll have a tendency to break up.

Okay. Those are all interesting and arcane issues, and it raises a question on how much detail do you want to go to in your aerosol modeling here because suppose somebody told me that it made an absolute difference what the Hamaker constant of the aerosol particle is. Then I'd be stuck in the problem of, okay, what's the Hamaker constant for aerosols coming from a reactor accident.

Well, that's a hopeless problem.

MEMBER BLEY: Let me ask something a little different. I take it we're not going to go through the Paul Scherrer slides, right?

PARTICIPANTS: No.

MEMBER BLEY: I found an interesting one. If you open it up, page 19, the one at the bottom, what I wanted to ask you guys --

MR. LEE: It's 19 on the other one.

MEMBER BLEY: What I wanted to ask you guys is given this kind of stuff you were talking about and given the experiments we've had, if you had to do what they did in 1150 as experts on a panel, would you live with something closer to the kind of distributions we see in the experiments or pretty much stretch like the experts used back at that time based on these uncertainties that still remain?

MR. SALAY: I didn't quite -- what was the question again?

MEMBER BLEY: If you were asked to be on an expert panel to do the next risk study, the bottom slide, that compares the range of what the experts laid forth against what the experiment saw, and what I was asking is as an expert would you have that broad distribution. Would it be broad on one end or would you look more like what's in the experiments?

MR. SALAY: In my opinion, it's that it would be the broad.

MEMBER BLEY: Too many things that we don't know for sure. I mean, these are nice in that they completely bracketed what the results showed, which isn't always the case.

MR. SALAY: Yeah, what they did with their retention, they sort of did complying with the DFs that we contend is not correct, and so they replaced some of their integral results with this analysis.

MEMBER POWERS: I mean there are some really remarkable things here that I don't presume to quite understand what PSI is doing. Mike put up a slide, and you saw the overall deal was 13. Somehow when Paul Scherrer does the analysis, they analyze it region by region by region, and they end up with 65. Okay?

But we have the experimental result that says it's 13. Okay. Now, how do they get those numbers? I leave you to ask them because I can't explain it. One of the observations we get from the test is we put through a steam generator tube conglomerate aerosol titanium dioxide, which behaves like titanium dioxide. It does not necessarily behave like a reactor accident aerosol. I don't know how a reactor accident aerosol behaved, but okay. It's going to look a little more like titanium dioxide than it is a latex sphere. Okay?

When they put the titanium dioxide aerosols they started with three micro aerosols. What came out was .7. Okay. When Paul Scherrer does their analysis they say, on the secondary side, they say, "Well, we'll start with three micron aerosols."

But by their own experiments they'll never have three micro aerosols, although they all break up inside the tubes. I cannot defend their analysis.

MR. LEE: I think if any of you are attending the Anaheim meeting next week, the PSI will be there. There are two sessions on ARTIST. So they will be presenting the ANS meeting.

MEMBER CORRADINI: Oh, the ANS meeting, yeah.

MEMBER POWERS: I mean, you just have to ask them. This has been the subject of more than a little bit of confusion because like I say, from the separate effects test, we know what kinds of decontamination factors we get at each stage, and we've done the integral test. Okay? And they don't seem to be inconsistent with each other.

But somehow, by some mechanism that we don't even begin to understand, when Paul Scherrer does the analysis, they end up with these numbers, and you can see them here: 65 DF, for a situation in which experimental -- no bigger than 13 and probably less, and some of it --

MEMBER BLEY: Almost worth a trip to

Anaheim.

MEMBER POWERS: -- some of it has to do with saying, well, 1.2 DF is like 1.5, and 1.5 you round off. It becomes two. Well, you start multiplying these things together --

MEMBER BLEY: And you get 65.

MEMBER POWERS: -- you get big numbers very quickly.

MEMBER BANERJEE: Well, they're getting numbers around 65 to 70, right?

MEMBER POWERS: That's right, and I have no idea how because as you saw from the experiments, we never see those kinds of numbers.

MEMBER BANERJEE: which is very different from one and three.

MEMBER POWERS: That's right, which we see in the experiments. Now, do you believe the experiments or do you believe the analysis?

MEMBER BANERJEE: That has to be reconciled, don't you think?

MEMBER POWERS: No, I don't feel we need to reconcile it. They can calculate anything they want to.

MEMBER BANERJEE: Is it taking into account that these are not full height for a lot of it?

MEMBER POWERS: We do that because steam generator, because its alleged tubes with a bunch of tube support plans --

MEMBER BANERJEE: Yeah, right.

MEMBER POWERS: Yeah. So each span between a pair of support plates can be treated pretty much the same, not independently. You have to recognize what's going on beforehand.

MEMBER BANERJEE: Yeah, but if you did a history calculation of the particle sizes.

MEMBER POWERS: Yeah, and they've done enough. When they did three spans for us, that was enough to rest and say, okay, I bet you span number 4, 5, and 65 will be about, as it's say there, one, two and three, and there's not much decontamination there. Decontamination factor, 1.2 in three span, okay? That means 20 percent of the material is being removed.

MEMBER BANERJEE: How do they get to the 65?

CHAIRMAN SHACK: I think we could ask them.

MEMBER POWERS: You're going to have to ask them because I've never understood that. I can't say they're heated. Orthogonal discussions, how did you get to 70? As you can see, the number is 70.

MEMBER CORRADINI: Where are you guys looking? I'm sorry.

MEMBER BANERJEE: Slide 8 on page 19.

PARTICIPANTS: The top one.

MEMBER BLEY: Where they're getting those big numbers.

MEMBER BANERJEE: They are getting a decontamination factors of 65 and 70 in their calculations. That's just the --

MEMBER CORRADINI: That's the calculation above, huh?

MEMBER BANERJEE: And they look at it cumulatively, and yet their experiments don't seem to be in line with that.

MEMBER ARMIJO: I think it's something we don't have to explain. We've got ten o'clock, and we don't have --

MEMBER BANERJEE: We don't have to explain, but if it's actually true, it gives you some factor --

(Simultaneous conversation.)

CHAIRMAN SHACK: All right. Mike are you finished with your presentation?

MR. SALAY: This is it. We're just on the last point --

MR. LEE: We are done, yes.

MR. SALAY: -- to say that we got our data. We want to end the project with the experimental data on the secondary side of steam generators to fulfill the steam generator action plan Item 3.3(a), and we consider that it's complete.

MEMBER ARMIJO: Okay. Any other questions or comments?

MEMBER BANERJEE: How are we going to present this and this in one and a half hours?

MEMBER ARMIJO: Well, it was impossible, and so it may be fortuitous that they missed their plane.

CHAIRMAN SHACK: That's one way to look at it.

MEMBER ARMIJO: Mr. Chairman, the meeting is all yours.

CHAIRMAN SHACK: Thank you very much. Again, an interesting presentation, and we're ready for a break until 10:20.

(Whereupon, the foregoing matter went off the record at 10:03 a.m. and went back on the record at 10:18 a.m.)

CHAIRMAN SHACK: Gentlemen, if we can come back into session. Our next topic is Risk Assessment Standardization Project and George will be leading us through that.

MEMBER APOSTOLAKIS: Thank you, Bill. As the members know, risk information is used routinely by many groups in the Agency and in very important processes such as the significance determination process, the reactor oversight process, the accident sequence precursor program and other areas. And these risk information is produced by of course, some of the groups using various approaches.

So the project we will hear about today, standardization of operation and event risk assessments, RASP, you have RASP, was initiated in response to a user need from the Office of Nuclear Reactor Regulation, which was issued in 2004 and the idea is to standardize these risk assessments so people will be using models that are more or less standard so there will be some uniformity in the information that is being produced and used by the Agency.

This is an information meeting, I understand. We --

MR. STUTZKE: We're not looking for a letter.

MEMBER APOSTOLAKIS: The staff is not looking for a letter and we are not particularly anxious to write one unless the members change their mind after. So we'll start with Mr. Mark Cunningham, an old friend that has disappeared for awhile but showed up today. So, Mark.

MR. CUNNINGHAM: Thank you, it's nice to be back.

MEMBER APOSTOLAKIS: Very good.

MR. CUNNINGHAM: I'm Mark Cunningham, the Director of the Division of Risk Assessment in NRR. Marty is going to talk to you today about work that we requested as Dr. Apostolakis indicated in 2004 with a supplemental request in 2006, and another supplemental request that will come later on this year.

We are -- I guess I'm here to give you a sense as customer or a user of the information that Marty will talk about. In fact, there's really six organizational units of the agency that are the customers for this work, my Division in NRR, the Division of Inspection and Regional Support in NRR, and the four regional offices. So this has an impact on a wide aspect, wide variety of people around the Agency.

We're going to hear a variety of things.

We have been very pleased with the progress. You'll hear about the things -- some things that we're already very happy with. You're going to hear about some things that we're going to see coming in the future that we've requested. Basically, this is a real key piece of our work to improve the consistency of the PRAs that are being used by the staff in the significance determination process and in other areas as well.

Again, we're very pleased with the type of work that's happening here and with that kind of introduction, I'll turn it over to Marty.

MR. STUTZKE: Good morning, I'm Marty Stutzke, the Senior Technical Advisor for PRA Technologies in the Division of Risk Analysis, Office of Nuclear Regulatory Research. I'd also call your attention in the audience is my boss, Christianna Lui, who is the Director of DRA, and her Deputy, John Monninger.

As George and Mark said, we're here to talk about the standardization of operational event risk assessment which is being done through the RASP project. Turning quickly to the presentation outline, you can see the topics we'll discuss. I call your attention, there are some backup slides here that you may find useful. We've tried to summarize previous ACRS meetings we've had on topics related to RASP. There's also a list of RASP contacts for specific aspects of it, things like that.

So briefly, we'll talk about the purpose of RASP, how we got started, the background, a quick introduction to how operational event risk assessment is done. There's several types of assessment. They're done for different purposes. It's not my intention to give you all a tutorial in any great detail about how the assessments are actually done.

Then we'll talk about how we've implemented tasks to help us standardize it, where we are now and where we hope to go.

The origin of this briefing was the draft report that you guys wrote on the review of the safety research program. That's draft NUREG-1635, Volume 8, back in Chapter 10. And it talks about projects to improve the efficiency and accuracy of NRC's significant assessments of findings and events. So we're here to provide some background to tell you in more detail specifically what we're doing in this area.

MEMBER APOSTOLAKIS: What's the date of this ACRS report?

MR. STUTZKE: The version I saw was earlier this year. I think it's still in draft form.

MEMBER APOSTOLAKIS: Okay, I mean, this one that you provided, NUREG-1635.

MR. STUTZKE: Right, that's the current one.

MEMBER APOSTOLAKIS: Okay.

MR. STUTZKE: As far as I know, it hasn't been formally issued yet.

MEMBER APOSTOLAKIS: It has not? It has?

MR. STUTZKE: It was in publication throws last I saw.

MEMBER APOSTOLAKIS: Well, it has been sent to the Commission.

MR. STUTZKE: Right, it was sent to the Commission. But I'll emphasize that RASP is -- it's focused on event assessment. It's not an effort to standardize all PRA within the NRC.

MEMBER APOSTOLAKIS: I'm glad you said that. I'm really glad.

MR. STUTZKE: The implication if we were to standardize everything in PRA would mean that we already know the answers to everything and we obviously, don't. And if we did, I probably wouldn't have a job.

MEMBER APOSTOLAKIS: I don't understand the second bullet. Can you explain it? "To provide background", what does that mean?

MR. STUTZKE: Well, to be honest, I saw an earlier version of this report that went out and it implied that we were not doing sufficient work to standardize our assessments.

MEMBER APOSTOLAKIS: Oh, we hurt your feelings.

MR. STUTZKE: And I took exception to it.

MEMBER APOSTOLAKIS: Okay, very good.

MR. STUTZKE: And as a result, the report got fixed.

MEMBER APOSTOLAKIS: Okay.

MR. STUTZKE: Okay. So RASP was a project started back in 2004. I want to emphasize it's a true collaborative effort. NRR didn't just send us a user need and send us off into a black hole. There's actually something we call the RUG, the RASP Users Group that meets on an almost monthly -- it seems in recent times it's been almost weekly, like this, but the RUG is a composition of somebody at NRR, somebody that works in Mark's Division, the Division of Inspection and Regional Support as well as research. So we have a large cooperative effort that's been helpful. So the idea is to provide some consistent methods for risk analysis of conditions in ASP and SDP Phase 3 as well as the assessment of events and conditions in ASP and MD 8.3 under --

MEMBER APOSTOLAKIS: What is MD 8.3?

MR. STUTZKE: I have a slide. In just the next slide, I'll tell you a little bit.

MEMBER APOSTOLAKIS: Okay, fine, fine.

MR. STUTZKE: But we realize the programs have different purposes and so it's hard to get your arms around it all. We're looking for the common denominator. To give you some -- a little bit of background, as you probably know, SDP was initiated in 2001, okay. So you'll see RASP came along about three years later.

And of course, the people that actually make SDP evaluations are the regional SRAs as well as participation from Mark's group like that. There are 15 SRAs now in the Agency, three per region and three at headquarters like this. And what was observed over time was that sometimes the analyses seemed like they were inconsistent, mutually inconsistent, for the same types of events. There seemed to be a lot of duplicated effort sometimes. And so RASP was an effort to try to get a handle on this, understanding that now we have several years of SDP experience under out belt.

Let me talk a little bit about operational event assessment and what they are. There's basically three here. There's the SDP that's part of the reactor oversight. There's MD 8.3 which is the NRC's incident investigation program and then there's accident sequence precursors. As I said before, SDP got started in about 2001. As I recall ASP was in the late '70s, the recommendation, I think, coming out of the WASH-1400 study. So it's been around for a long time. Tens of thousands of events have been assessed under ASP.

To give you a little flavor of the differences, it's helpful to think about the concept of the best available information. When the staff does an MD 8.3 evaluation, we're talking about hours or days. Okay.

MEMBER BLEY: This is actually done before you'd send out a team to investigate an event?

MR. STUTZKE: Yeah, the idea of doing the MD 8.3 investigations to decide the level of response. You can send out an augmented inspection team, special -- you know, what are you going to do? Okay, so you're trying to -- I'll say it's quick, but it's not dirty. You know, you can do the best you can with the information you have to target the response.

SDP in contrast, is looking at inspection findings, what's the reaction to an inspection finding? Are you going to do more inspections? Are you going to engage the licensee?

MEMBER APOSTOLAKIS: Are they don't still one at a time?

MR. STUTZKE: Separate analysis for each performance deficiency unless it's a common deficiency. Yeah, the other distinction among them is if there are multiple or concurrent events going on, you treat those in ASP and MDA 8.3 as best you can. So you're looking at the totality of the event. SDP is fixated on inspection findings.

MEMBER BLEY: The MD 8.3, two things; how long has it been around and two, does it also, in addition to deciding the kind of response, does it help decide the makeup of a team that would go out if you do an augmented inspection?

MR. STUTZKE: I'm going to kick it to Mark or John?

MR. FRANOVICH: This is Mark Franovich. I'm Chief of the PRA Operations Branch in NRR. The procedure dates back as far as 2001. Actually, earlier versions may exist prior to that. It uses a framework that uses a set of deterministic criteria as kind of entry conditions and it uses probalistic criteria to try to gauge level of response. There is an overlap region that's set up intentionally between the levels of response because there's a great deal of uncertainty.

You're doing this short-term assessment with not a lot of facts. So we try to make some bounding, reasonable bounding assumptions. The composition of the teams will be dictated by the complexity of the event. So for example, if you have an event where there are operator performance issues combined with equipment failures, you'll have both examiners, operating licensing examiners, resident inspector may be involved as well as specialists to look at the component failure, so it depends on the set of circumstances, and that's sort of a management decision between the regional offices and NRR.

MR. STUTZKE: Okay, the concept of event risk assessment is I find pretty straightforward. The idea is to look at what else could have happened in an event, an incident, that didn't actually or necessarily happen and that has implications for core damage or containment failures, these sorts of things like that. So the point is that event risk assessment is future-oriented. That's probably pretty obvious to a PRA engineer. Probability is a description of the future. Once an event happens, we know with certainty whether there was core damage or not to some extent.

So we're trying -- the idea is to extract what lessons we can get out of it, okay, the implications for similar events into the future. And it's done by manipulating the actual logic model. We use two figures of merit, conditional core damage probability for initiating events, so it's given the initiating event and all the other failures, degraded conditions that happen, what's the actual conditional core damage probability.

For events or degraded conditions, inspection findings this sort of thing, the figure of merit is the change in the core damage probability over the duration where the conditions existed, like that. And the idea of something called the failure memory concept, actual failures that were observed in the event or modeled as failures in the PRA, you set them to blue and true.

Okay, successes remain at their nominal failure probability assuming analysis. Okay, so you set up the RPA and basically turn the crank and you regenerate the core damage frequency and from that, you can calculate the other figures of merit that you're interested in. The intention of RASP in one way is to try to make this process consistent among the analysts.

MEMBER APOSTOLAKIS: So essentially, we're calculating how close -- how close --

MR. STUTZKE: Yeah, it gives you a quantitative measure of how close to core damage you were. The one thing that I didn't mention that I probably should have is for the accident sequence precursors. Of course, it's the full-blown analysis. It's used to measure performance against the safety goal and the NRC's strategic plan. There's an annual SECY paper that the staff writes on it and the more important precursor events are actually reported to Congress.

MEMBER BLEY: There's one thing I've always been interested in the ASP program, have we had cases where the event doesn't quite fit the PRA model you have such that we ought to let everybody know that we've learned something that ought to be built into our PRA models and is there a mechanism for doing such a thing if it occurs?

MR. STUTZKE: Yes, there's actually --

that happens actually, quite often and those insights are fed back into the baseline SPAR models that we use.

MEMBER BLEY: Are they published more generally so others who are doing risk assessment might learn from them?

MR. STUTZKE: SPAR models aren't publicly available.

MEMBER APOSTOLAKIS: That issue has come up in the past when -- even when there was the AEOD office.

MR. STUTZKE: Yes.

MEMBER APOSTOLAKIS: And the biggest problem was -- I mean, they were issuing NUREG reports but I don't think that practitioners outside the NRC paid much attention to them.

MEMBER BLEY: They weren't issuing the reports in a way that would have summarized this kind of surprise, we need to -- "Here's something you ought to build into your models". There wasn't a section like that in the reports. You had to read through and find it yourself.

MR. STUTZKE: To be honest, I think we've made a lot of progress in recent years. We'll talk about it a little bit later, but the staff has done detailed and cutset-level reviews of the licensee's PRAs to SPAR models to look at the differences.

MEMBER BLEY: So that a way to feed them?

MR. STUTZKE: That's one way to do it, and it's not all the time we're changing our models. Sometimes they're changing theirs. It also has generated other sorts of research. You know, we're looking at re-evaluating success criteria now as a result of that.

MEMBER APOSTOLAKIS: But there could be some finding that appears in the modified SPAR model that the industry at large is not aware of.

MEMBER BLEY: There could be and if that were a section of the annual ASP summary, those kind of things, that could be a useful bit of information. Sorry, go on.

MR. STUTZKE: Okay, as far as the standardization approach, it breaks down into three large areas; document methods and provide guidelines for the risk analysis and you can look at the sub-bullets and understand that we're talking about all initiating events, all operating modes.

The other major sub-bullet is to improve the fidelity of the SPAR model itself to try to better model the as-built, as-operated plant. Extending SPAR into external events, shut-down events and LERF Level 2 sorts of things.

The last bullet on enhancing analysis methods and providing technical support, this is a reference to the fact that what we do in RASP is to encapsulate other sorts of research that's done, okay. There are research activities that are within RASP such as updating the parameter estimates, the common cause failure methodology and things. But there are other activities that NRR has or that RES has in place that are driven by other types of user needs, okay. And so we're trying to extract the best that we can out of them and feed them into RASP.

MEMBER APOSTOLAKIS: So, Marty, this brings up a related question. How often is this, I don't know, project or report or approach supposed to be updated? I mean, I assume it's a living document. Is it a document?

MR. STUTZKE: Yes, there are handbooks.

MEMBER APOSTOLAKIS: It is a document because all we got was a 10, 11-page summary. Is it NUREG of some sort?

MR. STUTZKE: Well, they're not NUREGs. They're RASP handbooks and they're available on --

MEMBER APOSTOLAKIS: Okay, how often are

these supposed to be updated to accommodated what you just said?

MR. STUTZKE: Well, actually, we've issued the first revision in January and we're already revising them, fleshing them on.

MEMBER APOSTOLAKIS: Is that something on an ad hoc basis?

MR. STUTZKE: It's an ad hoc, continuing basis, like this.

MEMBER STETKAR: Marty, under the bullet that says "Improve SPAR model fidelity", the second sub-bullet for external events, shut-down events, in particular, how are you doing that and in particular, for external events, you typically require a lot of plant-specific information about the location of cables, walk-downs. How do you do that within the SPAR model context and also for shut-down events, you need to know an awful lot about how each facility manages their outages, how they integrate testing, maintenance activities, over the course of plant operating states or whatever jargon you use for breaking up the outage?

It's very, very, very plant specific information and very different from facility to facility. Do you propose to integrate that level of detail into the SPAR models, and if not, how are you going to go about it?

MR. STUTZKE: Well, you've jumped ahead. We have another slide on 11 that talks --

MEMBER STETKAR: Okay, go.

MR. STUTZKE: -- but the quick and dirty answer is we don't have external event models for every plant but we have internal models. Okay, we've built 15 so far. We've got five shutdown event models, two LERF-type models, and we're trying to decide where to go forward now.

MEMBER STETKAR: I'll wait till you get to the more detailed slide, then.

MR. STUTZKE: Okay, so the actual user needs that were specified for Office of Research was to develop the guidelines for internal events, that the guidelines and methods for external events, fire, flood, shut-down low power events, LERF type of analyses, enhancing the SPAR models and that actual GEM/SAPHIRE code, as well as ongoing technical support.

I look at this user need, sort of like a task order vehicle that encompasses a lot of things. It was supplemented in '06 to go after some success criteria work for the SPAR models, some actual thermohydraulic work like that. But it's been a little confusing.

The next couple of slides are summarizing what we've achieved and where we want to go. The handbooks themselves were issued in January of 2008. They are publicly available. The ADAMS accession numbers are there. Volumes 1 and 2, that talk about internal and external events are based on our existing methods that we've used in SDP and ASP analyses. Volume 3 is our guidance on how to review SPAR model revisions. It's following NUREG CR 3485 and as best we can the ASME PRA standard.

Okay, the handbooks are referenced inside the inspection manual chapter 0609, so we've made that link. They've had rather extensive internal review by NRC and the contractors and the actual Volumes 1 and 2 have been in trial use for a couple of years now. We've smoke-tested them pretty well. The other thing that I'll point out is that licensees have, we feel, ample opportunity to feedback on these handbooks.

For example, there are monthly meetings on the reactor oversight process and they can complain and make suggestions there. There's an SDP survey that goes on. I think it's bi-annual like that, and as well, if you read in the introduction, it talks about if you have a comment on the book, here's how to submit it.

MEMBER BLEY: Are you getting many comments?

MR. STUTZKE: Not yet.

MEMBER BLEY: That's after two years.

MR. STUTZKE: Well, you know, the sorts of comments, I mean, to be honest is that when we do an SDP and we say it's yellow and they say it's green, then we get a lot of comments.

MR. FRANOVICH: Marty, this is Mark Franovich again, NRR, DRA. We're expecting some feedback in the more formal structured feedback from NEI. We learned yesterday actually that they're interested in coming in or needing specific feedback on CCF modeling and HRA as well. Lots of perceptions of conservatism in our approach. So that's one view.

MEMBER MAYNARD: I'm just a little confused and I think I'm getting some things mixed up here. You say you're getting, or you have opportunities for the industry feedback on some of these and earlier you said, I think the SPAR models are not publicly available. Am I getting some things mixed up here or how are you getting feedback on --

MR. STUTZKE: Well, the models themselves are not available for security issues like that. The handbooks of how we do the analysis are available. So the idea is, you know, if there's some event going on, the licensee will, of course, make its own analysis.

MEMBER BLEY: Have the licensees looked over your SPAR models? Do they get -- can they see their own?

MR. STUTZKE: Yes.

MEMBER APOSTOLAKIS: Yeah, the individual utilities --

MEMBER BLEY: Have the SPAR models, okay.

MR. STUTZKE: Yeah.

MEMBER APOSTOLAKIS: And there has been a benchmarking, yeah. Harold, would it be worthwhile to look at these volumes for us? I don't know.

MR. VANDER MOLLEN: We could ask to look at them.

MEMBER APOSTOLAKIS: Would you send me a CD with -- these are electronically available, right?

MR. STUTZKE: Yeah, they are electronically available.

MEMBER APOSTOLAKIS: I don't know if anybody else wants them. Do you want them?

MR. STUTZKE: I'll dispatch them.

MEMBER APOSTOLAKIS: Okay.

MR. STUTZKE: I've tested these links at home. They work.

(Laughter)

MEMBER BLEY: You mean, they did.

MR. STUTZKE: You know, you never know. Well, as of a couple of days ago, they did. One never knows.

MEMBER APOSTOLAKIS: You mean I can do it from home?

MEMBER CORRADINI: Sure, if you know CITRIX.

MR. STUTZKE: These are on the public website. You don't need CITRIX for this.

CHAIRMAN SHACK: You can do the ADAMS base public search.

MEMBER APOSTOLAKIS: Okay.

MR. STUTZKE: Okay, so to return a little bit to John's question, we've done the cutset-level reviews for almost all of the licensee's PRAs. I think there's like four that are outstanding like that. There have been updates to the SPAR models for station blackout modeling like this. NUREG CR 6928 was issued that are the updated SPAR parameter models that came out in January of last year.

This is the actual failure rate data okay, that you use to quantify PRA. Did I throw that hard enough? The SPAR model QA plan has been updated and according to the acting branch chief ruthlessly implemented.

MEMBER APOSTOLAKIS: Now SPAR uses, when it comes to human reliability SPAR-H. And I mean, this is a very important activity of the Agency. We use this I would say simple, maybe more than compared to other models and at the same time, the Office of Research has been working on other models like ATHEANA that the industry, using the calculator and all that.

Is there an inconsistency there? I mean, are there any plans to maybe look at SPAR-H and as you said earlier, as more knowledge and models become available, try to adapt it because we are spending a lot of resources on research and yet, we're using SPAR-H for important decisions.

MR. STUTZKE: Well, the way that I would answer you is, and we all know what SPAR-H is and we know what it is not as far as the HRA methodology. To some extent, the staff is, in my mind, between a rock and a hard place. We have to make the assessments now with the imperfect tools that we have available. Part of the last program was to publish the SPAR-H handbook, so at least it was written down.

Right now, we're not doing any development work or additional work on SPAR-H. We're awaiting the results of the international HRA bench marking exercise that the staff was involved in and we'll decide after that's over, where we want to go in this area. I would anticipate changes and then we may scrap SPAR-H all together, we may modify it. We may decide that it's okay for our purposes, anything like that, but we're well aware of the inadequacies of the tool.

The other thing that I would point out is that we have another task that's called RES Technical Support and it's talked about on Slide 13, but let me jump ahead. The idea of the task is that if we need a real HRA analyst in the course of an event assessment. Say NRR, the regional offices, they do an event assessment and they say, "Gee, I'm confused", they have access through this user need directly to our experts. It's not like they need to come and write us a new user need and go through the bureaucracy. They can just call us up and we'll send them to the right people. Not just HRA, Level 2, you know, the full resources of the Agency are available to them through this user need capability.

Now, as I've mentioned before, we have 15 external event models, about five shut-down event models and two preliminary level 2 types of SPAR models. We're not doing any more work on these right now. They're on hold.

MEMBER STETKAR: Those models, Marty, now I can as the veto question because you have the right slide. Those models, the external events models, are they fully detailed models of the exact -- of the actual plants, including cable routing and locations of equipment?

MR. STUTZKE: No, they're simplified.

MEMBER STETKAR: How simplified?

MR. STUTZKE: Well, for example, in the seismic model, there's only three seismic initiators.

MEMBER STETKAR: No, I'm asking about locations of equipment inside the plant for fires and things like that.

MR. STUTZKE: Well, the way the models were constructed was to look at the major results that were coming our of licensee's PRAs and to duplicate them, put them back into the SPAR model, not full-blown bottoms-up types of risk assessments.

MEMBER STETKAR: That's the same for the shut-down events models, they're just hard-wired cutsets?

MR. STUTZKE: They're not as hard-wired.

MEMBER STETKAR: What are they?

MR. STUTZKE: It's something I can -- they model several plant operating --

MEMBER STETKAR: I mean, the term, 15 integrated Rev 3 SPAR models sounds pretty sophisticated to me and from what I'm hearing, it doesn't sound --

MR. STUTZKE: Well, they're integrated in the sense that they're built on the internal events models, so you pick up all the random failures from the operator failures.

MEMBER BLEY: Well, were they based on more detailed models that the plant had?

MR. STUTZKE: Yes, they're based on the more detailed models that came out of the plants.

MEMBER BLEY: They're taking the most important parts of --

MR. STUTZKE: Right.

MR. FRANOVICH: This is Mark Franovich again, NRR DRA. Just a few comments on external events. One thing that we're trying to work with research here in the next few months actually is trying to come up with an approach to capture the PRA insights from NFPA 805 submittals that will be pending here over the next few years and trying to use those to develop some form of models. Don't know how to do that yet. That's still a --

MEMBER STETKAR: The most important insights from anyone who's ever done a shut-down risk assessment, or an internal fire or a flooding assessment is that you have to know what is located inside the plant, where the cables are routed, what is located in what cabinets and where those cabinets are located inside the plant to do a fire analysis or a flooding analysis and for a shut-down analysis you need to know how that utility organizes its refueling outages.

When do they do particular types of maintenance at what stage in the outage as a function of pressure in the vessel, status of isolation and things like that. That's not a philosophical finding about modeling fires it's how the plant is actually configured. So it's not something you'd need to do research. You need to go to the plant.

MR. FRANOVICH: I don't disagree and let me comment on the shut-down piece for a moment. You're right, no two outages are alike. You do need to understand in model development what the operating practices are at the plant, especially for configuration risk management. The models that are the five integrated models that Marty is referring to, the way those were developed was an effort where the analyst, both contractors, NRR, actually and also research, all three, go to the plant and actually conduct interviews with the outage planners, understand what the station operating practices are, to come up with some form of static model.

When you get a specific event to try to model, it's not a matter of simply exercising the static model. If it's a significant event by practice, what we do is we actually send a small team back to the site. Let's look at the specific configuration, let's interview the operators, let's understand if they have any rules of thumb they may be applying that aren't proceduralized.

Those context are very important in doing the assessment. But we have now are just five models and actually, we're looking at doing another user need or a modification of it to develop at least a model for each type of reactor out there as a basic template to start with because trying to develop 71 models is, given our limited PRA resources, it's just not practical. So we need to come up with some sort of stop-gap approach.

MEMBER STETKAR: Okay, you said some things that are encouraging. You said for the five that you have, at least what I heard was that you did go to those plants and interview people in the outage planning departments and recognizing that each outage is slightly different, most plants, especially these days, have a general outage plan. They're getting much better at doing outages.

So that the deviations from outage to outage are much smaller than they used to be. However from plant to plant, there can be significant variations. They're trying to standardize that across a fleet, obviously nowadays. I'm not sure how useful boiling water outages as a generic class versus pressurized water outages as a generic class would be. I'd have to think about that.

MR. FRANOVICH: I think we're looking at more down in the level of BWR-2, 3, 4, 5, not just the simple BWR template, PWR template, but there are some configuration issues in mid-loop operations that have some variability out there.

MEMBER STETKAR: What about, can you tell me a little bit about the external events models, because you have 15 of those. Did you also do a similar type of exercise to go to the plant and determine basic layouts of equipment anyway? You know, where are the switch gear rooms, how many different instrument control cabinet rooms, where are they generally located so you could even make some decent guesstimates of where cables were routed and things like that?

MR. FRANOVICH: Unfortunately, the answer is now, in general. Most of these models were developed largely from the IPEEE submittals. So they have an enormous amount of uncertainty. That's why we're looking at for the population of 805 plants trying to come with some process whereby 805 process itself you have to do those plant walk-downs, the cable routing, you do the circuit analysis and all that. That's a much better set of information to capture, but that's still a lot of work in progress. We're talking years down the road.

MEMBER STETKAR: Okay, thanks.

MR. STUTZKE: Okay, so the third task was actually improving software tools SAPHIRE and GEM. By the way, I'll throw out, we can provide a demo if you're interested in seeing the latest version of the software. In fact, I think we had one scheduled and it got postponed and things like that.

MEMBER STETKAR: That could be useful.

MR. STUTZKE: We're willing to set that up. Let's see how it does it but new user interface for the STP Phase 2 analysis, fixing up user interfaces for Phase 3, ASP, more capabilities from the SPAR model, trying to make the link between the Level 1 and the Level 2, that's the reference to the LERF modeling like this.

And, of course, the calculational methods, the implementation of the common cause failure assessment for operational events, some different mission times. Beta testing is going to start momentarily, within weeks, like that, culminating towards by the end of 2009 to get the tool actually out and up and using it. A nice user fix now, it looks slick.

MEMBER BLEY: As far as -- I'm sorry, is SAPHIRE pretty stable now? There was a time when it was getting changed almost weekly.

MR. STUTZKE: I think it's reached a certain level of maturity. I mean, you know, these software designers always want to mess with things like that.

MEMBER BLEY: Well, to help out their clients.

MR. STUTZKE: And put a few dollars in

your pocket. But the other thing that I'll mention for the beta testing, it's not just the staff. NASA is involved in the beta testing because they use SAPHIRE extensively.

MEMBER STETKAR: Marty, it's been awhile and I don't want to get too far off track here. You have the bullet about common cause modeling. It's been awhile since I've played with SAPHIRE. Is there now an automated generation of the -- you can specify groups and --

MR. STUTZKE: Yes, right.

MEMBER STETKAR: Excellent.

MR. STUTZKE: Yes, you can find the groups and it throws the events in for you.

MEMBER STETKAR: Wonderful.

MR. STUTZKE: I think it even calculates them correctly now.

MEMBER STETKAR: Minor details. Minor details.

MR. STUTZKE: Okay, so the tech support as we had mentioned before, to the various NRR analysts and SRAs as they need to. That includes -- part of the tech support includes training of the SRA counterpart meetings that are held every six months about. In fact we just had one it was just last week or so and all the SRAs were down in Bethesda with us. As I'd said before, any time they need the expertise, they can call us and we'll provide it for the user need.

You can see, here's a list of the more common sorts of areas of tech support that we will provide. We've also summarized a lot of information that's been compiled during the RASP process on something we call the RASP toolbox. This web page is not publicly available. It's only available on the NRC intranet. It's basically a convenient summary, a number of hot links to the various -- for example, NUREG CR's you can pull up the actual handbook, et cetera, like that.

Most of the information on that web page is publicly available in other forms. I mean, you can always get a NUREG. There are some things on there that are proprietary like our link into the EPIX system and things like that. One of the backup pages, I've actually given you the URL, if you want to pull it up and see what's there. I find that personally it's a very useful page. My only problem is the font size is too small. As I get older, I can't read it any more.

But that's a good segue into this what I call the work breakdown structure. There's a great deal more to RASP and this interface than perhaps, just the handbooks. The handbooks are part of it. You can see the tech support, the SAPHIRE/GEM, SPAR model updates and things like that. This kind of breaks down and gives you a big picture sort of thing, but there are other activities ongoing, for example, SPAR model development, that are not under the umbrella of RASP.

For example, we have a user need from the Office of New Reactors to build SPAR models for new plants. Okay, we just received it within the last couple of seeks. It's three now, within the next couple of years.

MEMBER APOSTOLAKIS: For new plants, what does that mean. I mean --

MR. STUTZKE: AP 1000.

MEMBER APOSTOLAKIS: -- the design certification part?

MR. STUTZKE: Right, as best we can.

MEMBER APOSTOLAKIS: I see. What would they do with those, play -- do sensitivity analysis or do --

MR. STUTZKE: Well, I think it's in preparation for when a license is actually granted.

You need to be able to get ready to implement the ROP, Reactor Oversight Process. You need to begin to regulate once the license is issued and --

MEMBER APOSTOLAKIS: So this is the first step because --

MR. STUTZKE: This is the very first step.

MEMBER APOSTOLAKIS: We'd have to do a more detailed --

MR. STUTZKE: Right, I mean, eventually -- I mean, I look at them almost like templates and so an actual licensee that would build an AP-1000 you would make it more plant specific. You know, there are things that are not within the certified design envelop.

MEMBER BLEY: So if you built a SPAR model for one of the new plants, you'd just go to the vendor's fault trees and put them into SPAR, into --

MEMBER STETKAR: Well, we haven't started the work yet, but --

MEMBER BLEY: Is that what you anticipate or something different.

MR. STUTZKE: No.

MEMBER APOSTOLAKIS: That's the only thing that's available, isn't it?

MR. STUTZKE: No. That's the information

that's available.

MEMBER APOSTOLAKIS: Yeah.

MR. STUTZKE: Okay, but, you know, you know enough about the design, you could develop fault tree of entry from scratch.

MEMBER BLEY: Well, you could.

MEMBER CORRADINI: But why would you do that?

MR. STUTZKE: It would be a check and the reconciliation again for awhile. As I say, the user need has just come through us. It's new. It's a balance we're having trouble finding. We have conflict of interest, contractual problems.

MEMBER CORRADINI: Now, wait a minute, I don't understand. I'm sorry.

MR. STUTZKE: Between Idaho.

MEMBER APOSTOLAKIS: Idaho is doing all of his work or most of it?

MR. STUTZKE: Well, Idaho is our contractor for SAPHIRE and GEM and they are the constructors of that. And they're related to Bechtel, okay, and so there are issues like this.

MEMBER CORRADINI: They're related, but they're not related.

MR. STUTZKE: It's an issue, it's an

issue. And to find other contracts is turning out to be a challenge as well. So but the point here of this slide is that there are other activities that go on that are overlapping, RASP and that we're trying to utilize like this.

Okay, so in the future, you know, we're going to complete Volume 1 by adding the new guidance for common cause failure modeling, the new parameter estimates updates, work on sensitivity analysis, HRA, simplified expert elicitation. All of these things are yet to be done, okay.

MEMBER BLEY: What's in your head about simplified expert elicitation?

MR. STUTZKE: Well, there is the report from Idaho Labs that's been issued.

MEMBER BLEY: Current? I mean, it's just come out or has it been out?

MR. STUTZKE: It's relatively current. I haven't read it. I don't know what's in there yet.

MEMBER BLEY: Is it a NUREG or it's an Idaho --

MR. STUTZKE: It's an Idaho Reg.

MEMBER BLEY: Is this publicly available? We could get it.

MR. STUTZKE: Yeah, you could get it.

MEMBER BLEY: We'd like to see it.

MEMBER APOSTOLAKIS: So, Marty, how different is this RASP handbook from the ASME stuff, or to put it differently, why can't --

MR. STUTZKE: I would characterize it -- you know, the ASME standard is here's what you need to do. As RASP handbook is here's how you should do it.

MEMBER APOSTOLAKIS: So it takes off from the ASME standard then.

MR. STUTZKE: Well, it's built on it. It's built in part. In other words, Volume 3 of the QA process is linked to the ASME standard. We went through that to try to capture a process.

MEMBER CORRADINI: So it's a handbook that actually tells you how to do it, a way to do it, not the way.

MR. STUTZKE: Yeah, the specific assumptions.

MEMBER CORRADINI: Yeah.

MR. STUTZKE: Well, it's the way in the -- to the extent we're trying to standardize the staff's operational event risk analysis.

MEMBER CORRADINI: But it's for the staff.

MR. STUTZKE: For the staff and licensees can do what licensees can do and they need to justify it.

MEMBER CORRADINI: But if I were a licensee, this might be used as a path of least resistence.

MR. STUTZKE: Yes.

MEMBER CORRADINI: Thank you.

MEMBER SIEBER: I presume it's focused on the SPAR model and ancillary models, so it's value to a licensee is probably limited.

MR. STUTZKE: Yes. Well, the licensee can gain things out of it. I mean, it will talk about things like mission times and PRAs, what do we assume.

MEMBER SIEBER: And insights about the way you do your business.

MR. STUTZKE: Right.

MEMBER SIEBER: How big is that Volume 1? Is it available to me?

MR. STUTZKE: Yeah, again, that's electronically available in ADAMS. We can give it to you on disk if you want it.

MEMBER SIEBER: That would be good.

MR. STUTZKE: We can make arrangements with Harold and provide some electronic copies. So again, revising Volumes 1, 2 and 3 based on user feedback, we needed to develop new models for shutdown of low power and for LERF. We continue to look at the enhancement of methodologies for common cause failure. We have a draft NUREG CR that came out on that. It's this thing that Dale Rasmussen published, was issued. This is dated April of this year. Here's one on LOCA pipe frequencies, expert elicitation.

MEMBER BLEY: Yeah, that's -- we've reviewed that work. He'd good.

MEMBER APOSTOLAKIS: No, we didn't do the LOCA.

MEMBER BLEY: That's where that came from, right?

MEMBER APOSTOLAKIS: That was not simplified. That was --

MR. STUTZKE: You're talking about the full expert elicitation for --

MEMBER BLEY: Yes.

MR. STUTZKE: This is the reduction of that to come up with initiating event frequencies for SPAR. Okay.

MEMBER BLEY: Oh, great.

MEMBER APOSTOLAKIS: At some point, I remember the ACRS recommended that the Commission or the Staff develop a -- I mean, we recognized that there were several approaches to expert opinion elicitation that various groups within the Agency use and we recommend that maybe one or two should be used agency-wide. So these guys from Idaho now simply find what was done for the LOCA frequencies and at the same time we have the seismic people going back to the Shock (phonetic) methodology and working on it? Is there any effort to create a common methodology? Then I think we have the Materials Office using its own approach.

MR. STUTZKE: Yeah, I think what I would expect, I mean, we haven't started the development of the handbook chapters for the expert elicitation method, okay? So it's in its infancy and what I would envision -- I remembered your comment about a, you know, more broad agency-wide --

MEMBER APOSTOLAKIS: Yeah.

MR. STUTZKE: -- method and I think we ought to revisit it at that time.

MEMBER APOSTOLAKIS: Good.

MR. STUTZKE: One of the things that RASP does, we don't just suck in the information, it also helps us drive the research agenda to some extent, so you know, we really need to look into this. There's give and take in there.

MEMBER BLEY: And all of this stuff is in the RASP website, the toolbox page?

MR. STUTZKE: Yes. If you look at the website, you'll know what we know basically. They keep it up to date. Okay, and then we talked earlier about the HRA and we're waiting to see the benchmarking results to decide where we want to go in the future on that.

Okay, ongoing work, some issues here that you might be interested in, enhancing the internal events SPAR models, two years ago, we got an addendum to our user need about success criteria re-evaluation of thermohydraulic analysis. There were some cases where the SPAR models appeared to be conservative to the licensee's PRA and we wanted to go after them with better thermohydraulic tools, be it MELCOR or TRAC, whatever we have in our arsenal upstairs to do it.

Part of the interesting work that came out of that was a work that Dr. Rick Cherry's been doing on a phenomenological definition of core damage. The idea is when a thermohydraulic analyst makes a computer analysis, how does he know when core damage has occurred? What are the actual parameters that he's looking at? Is it collapsed level, is it temperature? You know, what should it be and Rick's been doing a lot of work in the are. It might be something else you're interested in looked at.

MEMBER APOSTOLAKIS: Yeah, I'd like to see that at some point.

MR. STUTZKE: There are some presentations there.

MEMBER BLEY: We should fire up CITRIX.

MR. STUTZKE: None of that -- that work will be on the RASP toolbox page, under the SRA counterparts meeting. It will be in the handouts to the counterparts meeting. We can show you later how to access the page.

The other thing I would point out is that we have a memorandum of understanding with the Electric Power Research Institute for a variety of research topics. It's one of the backup slides that was the areas we're looking at. We're talking about things like -- let me pull back here, support system, initiating event, fault trees, how to draw those, treatments of loss of offsite power, things like that.

And you inject in the containments and BWRs after they fail?

MEMBER APOSTOLAKIS: So what does it mean, Marty, you're doing it together or what?

MR. STUTZKE: Joint project. There are working groups developed between industry and NRC staff like this. Meetings -- you know, staff has gone to meetings and travel and things like this. The other thing is, I know that we're in the process of at least two addenda to this MOU, one on seismic and another one on HRA Erasmia sent me yesterday. I haven't had a chance to look at it.

It's a good cooperative effort.

CHAIRMAN SHACK: Marty, somehow I would have thought when you're reconciling SPAR model with the PRA licensee model, the success criteria would be almost the first place you'd look.

MR. STUTZKE: That's how a lot of these were identified, in their cutset level review.

CHAIRMAN SHACK: Okay, and --

MR. STUTZKE: The differences.

CHAIRMAN SHACK: Oh, the differences, okay. But I mean, you're not proposing that they re-evaluate with a new core damage criterion for their own success criterion or that may come out of this.

MR. STUTZKE: That may come out of this eventually. I mean, it's real curious, when you look at the ASME/PRA standard, they give you several definitions of core damage, collapse level, temperatures, different temperature limits, 1800, 2200 and it's not surprising, you get a variety of results.

MEMBER APOSTOLAKIS: So if we ask now officially the Agency what is the definition of core damage, when we talk about core damage frequency, what do we mean, is there such a definition?

MR. STUTZKE: I don't think there is right now.

MEMBER APOSTOLAKIS: Oh, my God.

MR. STUTZKE: I think you will find a wide variety and what you tend to find is what the Agency has used as conservative. When we say it's core damage, it may not be.

MEMBER APOSTOLAKIS: I thought the definition had to do with the release of noble gases, five or 10 percent of them, then you have core damage, more than that is core damage, but that's not a valid definition?

MEMBER BLEY: I think somewhere there's that definition but I think operationally doing a PRA, you set other surrogate criteria that may or may not be --

MR. STUTZKE: Remember you're trying to get down to how do you draw the logic structure. You want to know what the success criteria are and I've had the impression for quite some time, you know, the difference between one out of three pumps and two out of three pumps is like night and day in the thermohydraulic analysis. I don't need a very precise definition because I'll draw the fault tree that just says all the pumps failed, end of discussion.

MEMBER APOSTOLAKIS: I think that's --

MEMBER BLEY: And then there's an issue of timing when it happened.

MR. STUTZKE: Timing issue is another thing and I used to be real interested in that because we used time reliability correlations in HRA and you wanted to know, but we don't do that any more.

MEMBER APOSTOLAKIS: So what obviates the need for a precise definition is the discreditization that PRA laments.

MR. STUTZKE: Right.

MEMBER APOSTOLAKIS: And we are never going to say two pumps and one-third of a pump. It's two pumps, three pumps, one pump and then the precise definition is not needed, and especially if your conservative, right?

MR. STUTZKE: Right, but it is of interest. We were handing off this work to another division in research and they wanted to know when to stop calculating. That's basically --

MEMBER APOSTOLAKIS: That's interesting.

MR. STUTZKE: But it's interesting. Okay, so that's probably what it is. Let me -- you know that the handbook is in wide use now by all the risk analysts and the SRAs that do risk analysis of operational events. So in that sense we have achieved some measure of standardization. Something else that needs to be pointed out is in -- I think it was starting in June of 2006, there was a change to the ASP program itself.

Used to be ASP always went off and did its own analysis. Remember that there's a distinction. The ASP analyses are done by the Office of Research. These other ones, SDPs and MD 8.3, that's NRR's responsibility to do that. And sometimes they didn't agree, okay, for different reasons.

Well, and the other problem was, it's a matter of resources, you know. We have limited resources and so back in 2006, ASP was changed to say if there's and SDP inspection and it's been analyzed and we find that it's applicable and appropriate, we can use it. We don't need to make an independent study. You know, it obviously has some time savings for us.

The point is that it also helps standardize things, you know, to some extent because the analyses that we would do under ASP would be done with the same handbook that they're using for SDP. The other thing that I would impress upon you is that there is a large amount of communication now among the analysts. There is a weekly telephone call among the SRAs. That the headquarters participates in. There are -- every six months there are SRA counterparts meetings. I mean, there's a lot of communication going on back and forth between the Office of research and NRR and the Regional Offices like this.

Routinely, SRAs from Region call into Research asking for guidance on how to do their analyses and things like that. There's a lot of give and take back and forth with Idaho Laboratory as well on aspects of using SAPHIRE and GEM like this.

MEMBER BLEY: Do all of the SRAs spend time in headquarters? I know a lot did in the beginning, but I don't know if that's true now.

MR. STUTZKE: Yes. I'll tell you what I know and feel free to jump in, but SRA's are formally qualified. There's a qual card like this. All SRAs are, in fact, used to be inspectors so they have to go through all of that qualification as well. There are required rotations to NRR, so they can go see what's going on. I believe the suggestion was made rotate into RAS for awhile. They have to rotate to another region that's not their home region like this, so there's a lot of cross-pollination going on here.

You know, to be fair, SRAs are not risk experts. They're not the heavy gun PRA experts. They know enough to be able to do their job and hopefully they know enough to call us when they get in trouble. We provide the mechanism for them. And we actually -- you know, SPAR models are getting better. They're more representative of the as-built, as operated plant that was the purpose of the cutset level reviews that we did. So I said, you know, there was give and take there. We modified SPAR models as we needed to. Licensees modified their models as we needed to and we're reaching a better convergence.

MEMBER MAYNARD: It looks to me like it would be a real challenge to keep these up to date. Licensees are always making changes to procedures in their plant and everything. Do you get feedback on those or what attempt is made to keep data in your models current with all the changes that the licensees are making?

MR. STUTZKE: I want to dump that off.

MEMBER APOSTOLAKIS: Please come to the microphone and identify yourself and speak with sufficient clarity and volume.

MR. APPIGNANI: My name is Pete Appignani. I'm the SPAR Model Level 1 Program Manager in research but most I prepare PREP at this time. We're almost done with our initial cutset level reviews. There are four plants that are in the process of changing the software for their model and it's been delayed and so at that point in time we finish them, we'll have all 77 models representing 104 plants.

Going forward, we look to updating about 12 models a year and that's based on the updates that we've done in the past three or four years and we're just going to plan on doing 12 updates per year to keep the SPAR models up to date.

MR. STUTZKE: Good, any questions? Thank you.

MEMBER APOSTOLAKIS: Any questions or comments from the members? This was an information meeting.

MEMBER BLEY: I really appreciate the briefing because I didn't know much about what was going on here and thanks very much. It was very informative and I look forward to looking at your website.

MEMBER APOSTOLAKIS: Okay, thank you very much, Marty.

MR. STUTZKE: Thank you very much.

MEMBER APOSTOLAKIS: Back to you, Mr. Chairman.

CHAIRMAN SHACK: Okay. We have interviews scheduled at 11:45.

MEMBER APOSTOLAKIS: Is it legal to start earlier?

CHAIRMAN SHACK: No. I believe we can if we can find the candidate. We will be holding the interviews in this room and I just noticed the schedule here and I'm a little concerned about the schedule on Friday because I suspect I'm going to be losing people.

MEMBER CORRADINI: That one could be moved up, I would assume.

CHAIRMAN SHACK: Right, and I would like to say that a half an hour would be sufficient.

(A brief recess was taken.)

CHAIR SHACK: We can come back into session now. Our next topic is an Overview of the U.S. Evolutionary Power Reactor, the EPR design. And Dr. Powers is leading us through that discussion.

MEMBER POWERS: Yes, we're going to do a real reactor now instead of these passive, natural convection -

CHAIR SHACK: Well, I notice we got some converts.

(Laughter.)

MEMBER POWERS: Sooner or later the Committee is going to have to plow into the EPR, and it's useful to get an overview of all the things that have to be done on a certification. Is that not right, Mike?

MEMBER CORRADINI: You're going to do it chapter by chapter, right?

MEMBER POWERS: This is a real reactor. I mean, it's actually going to come in with a document and design, a written document that we can look at and printed pages on it, and things like that.

(Simultaneous speech.)

MEMBER POWERS: I mean, this reactor has the advantage that they're actually building one, and maybe even two, maybe even four, so it should be fun, but it's going to take some understanding of the approach and whatnot, and so we ought to get started on that process.

So now on this, you're going to have to forgive me a little bit on the nomenclature here. I'll do my best. Getachew Tesfaye?

MR. TESFAYE: That's correct.

MEMBER POWERS: Who's going to start us off, and then we'll progress on with Sandra Sloan. That was an easy one. Already I like you a lot. And then Marty Parese. Okay. Your floor.

MR. TESFAYE: Thank you, Mr. Chairman. My name is, again, Getachew Tesfaye. I'm the NRC Project Manager for Areva's design certification application. I'm going to give you a very short background of our project at the NRC, and then I'll let Areva present the design.

The EPR project at the NRC is about over three years old. We spent the first two years engaged in pre-application activities. In that time period, Areva made several presentations to familiarize the NRC staff with the design. And also, during that period they submitted several topic reports that were referenced with the application that was submitted last July.

MEMBER APOSTOLAKIS: Is it typical to spend two years?

MR. TESFAYE: Two years, three years.

MEMBER APOSTOLAKIS: Really.

MR. COLACCINO: It's typical. This is Joe Colaccino, the EPR Project Branch Chief. There's nothing atypical about the pre-application period, if you compare it with ESBWR it was probably shorter than that. AP1000 I think -- AP600 is probably comparable.

MEMBER APOSTOLAKIS: And the main idea of pre-application is, as you said, to familiarize the staff.

MR. TESFAYE: Familiarize the staff and submit topics, and have topical report forms so they can approved and be referenced in the application. Areva submitted 15 topical reports that were referenced in the application.

MEMBER POWERS: I do not have a list of those topical reports. I probably ought to.

MR. TESFAYE: I will get -

MS. SLOAN: Getachew, what I have is, I have a list from the FSAR of all the topical reports that are referenced in the FSAR, which includes the ones that we submitted during the pre-application review, as well as others that were already approved. So if you want the whole list, we can give you that. And then I can sort out the ones that were specifically provided during the pre-application review.

MEMBER POWERS: I haven't done anything to you yet. Why do you want to ruin my life with this long list of -- wait until I have harassed you good, and then do those sorts of things.

MS. SLOAN: Okay. I will wait.

MEMBER SIEBER: Another question is, are all these topicals on ADAMS?

MR. TESFAYE: Yes, they are on ADAMS.

MEMBER SIEBER: Okay. So we can get to them.

MR. TESFAYE: They are also incorporated by reference in the FSAR chapters.

MEMBER SIEBER: Yes, we know where to go.

MR. TESFAYE: Yes.

MEMBER SIEBER: Okay.

MEMBER POWERS: I think I need the list. And having them in ADAMS is the same as having them hidden somewhere in Siberia.

MR. TESFAYE: So this pre-application phase ended back in December when Areva submitted the application on December 11, 2007.

MEMBER BANERJEE: What is the difference between a topical and a technical report?

MR. TESFAYE: A topical report is a stand-alone topica report that the staff review and issue a staff evaluation report. A technical report is something that's referenced and reviewed as part of the chapter in the FSAR. A separate SE is not going to be written on the technical report, so that's the difference. All technical reports are stand-alone. They can be referenced with the other applications theoretically.

MEMBER APOSTOLAKIS: From the practical point of view, what difference does it make when you say the pre-application period ended, now you have the application? So what? You are not reviewing -

MEMBER POWERS: They can't be nice to each other any more.

MEMBER APOSTOLAKIS: What?

MEMBER POWERS: They can't be nice to each other any more.

MEMBER APOSTOLAKIS: Does it make any difference?

MR. TESFAYE: Well, it does make a difference. When you officially accept the application, you create a docket, the official review period starts. Before the pre-application period, it was a topic-specific review, general finalization, nothing is in-house for us to start a docket, and also establish a schedule, so there is a big difference.

MEMBER APOSTOLAKIS: So it's a little more formal now?

MR. TESFAYE: More formal, as I can show you in this slide. Not only have separate application, we have also set a schedule for review.

MEMBER APOSTOLAKIS: Okay.

MR. TESFAYE: So there's a big difference. So the application was submitted in December, December 11, 2007, and we accepted it February 25, 2008. We also issued a schedule which are the six-phase milestone schedules on March 26. And the first phase is, of course, the preliminary safety evaluation report with RAI. And phase two is SER with open items, and phase three is we're going to come back to ACRS with SER with open items. In phase four we will show advanced SER with no open items, and phase five we come back to ACRS again with SER with no open items. And the last phase before the rule making for the certification is phase six, which is issuing the final SER with no open items.

MEMBER CORRADINI: So if I might just ask this question now that I see a schedule. So the first time the ACRS will see anything formally, and I'm asking I guess partly Dana and you, is Subcommittee meetings prior to phase three, or in preparation for phase three?

MR. TESFAYE: Well, at the beginning of

phase three.

MEMBER CORRADINI: So not before early '10?

MR. TESFAYE: No, right now we have established -- as soon as we complete phase two, we plan to bring in those portions that we completed to the Subcommittee. That's our plan right now.

MEMBER CORRADINI: Oh, okay. So after Thanksgiving of '09.

MEMBER POWERS: And the first time that you will be put to work on this particular application will be November of this year.

MEMBER CORRADINI: Thank you, Dr. Powers.

MEMBER BANERJEE: Why so early, Dana?

MEMBER POWERS: Because Mike is a little bit slow.

MEMBER BANERJEE: This is specially for Mike.

MR. COLACCINO: If I could add to that; this is Joe Colaccino, again. What we have been -- we've worked with ACRS staff on this. What we thought would be a reasonable approach is to come in as the chapters are completed, and we go through the no open item phase. I see gentlemen giggling because I heard the remark about coming in chapter by chapter before. Coming in a series of waves so that you're not waiting until the latter part of 2009 to see them for the first time. So we've worked out a schedule to do that. I think we looked at three waves of meetings. And if anything changes, we'll make adjustments to that schedule as we go forward in 2009.

MEMBER CORRADINI: Thank you.

MR. TESFAYE: The COL applications referencing EPR, the reference COL application -- they submitted Part One of the application which is the environmental report back July 30, 2007, and was accepted for review January 25th, 2008. It's currently in Phase One of the review.

MEMBER APOSTOLAKIS: What is R-Cola and S-Cola?

MR. TESFAYE: R-Cola is Reference Cola. That's the first combined license application referencing the EPRs.

MEMBER APOSTOLAKIS: S-Cola?

MR. TESFAYE: Subsequent Cola.

(Off the record comments.)

MEMBER APOSTOLAKIS: I understand the R, but the S I didn't. Oh, you mean others have also come in.

MR. TESFAYE: Yes.

MEMBER APOSTOLAKIS: Okay.

MR. TESFAYE: And that review is going to be done concurrently with the design certification review. So, again, it was submitted in two parts. I think that's the first one that's submitted in two parts, first application, first combined license application that was submitted in two parts.

MR. COLACCINO: And, hopefully, the only.

MR. TESFAYE: Part Two was submitted on March 14, and we just docketed it yesterday. We accepted for review yesterday.

MEMBER CORRADINI: So just to help me understand. How does the fact that it's in two parts matter to the staff? You just stop looking until you've got the second part in?

MR. TESFAYE: Well, originally, the plan was to accept the environmental report and start reviewing it, but it had so many problems, we didn't get a chance to start the review. So it took about six months to accept the first part, so there was nothing net-gained by their submitting it in two parts.

MR. COLACCINO: Really, in reality - this is Joe Colaccino, again. This will be the last time you'll hear us speak about two parts. It doesn't matter now, the application is complete for the Calvert R-Cola, so we'll be talking about the R-Cola application, and you'll never see Part One or Part Two again.

MR. TESFAYE: Again, the planned submittals for the subsequent COLAs, combined license applications that will be coming in after the reference quota shown on this slide. And that's all I have for brief background information, so we'll go to Areva and Sandra.

MEMBER POWERS: I appreciate the schedule information as far as the chaptering, we'll discuss that a little bit. You're up. Okay. So now I can start picking on you.

MS. SLOAN: Now is your turn. My name is Sandra Sloan. I work out of Lynchburg, Virginia for Areva NP, and my responsibility is Manager of Regulatory Affairs and New Plants Deployment, which gives me responsibility for EPR licensing in North America.

Our goal for today -

MEMBER POWERS: Are you building a lot of these in Canada and Mexico?

MS. SLOAN: We are talking about that.

MEMBER POWERS: Good luck.

MS. SLOAN: We're exploring possibilities, let's say.

MEMBER BANERJEE: How many of these are being built right now and where?

MS. SLOAN: Two, one in Olkiluoto, Finland, and one in Flamaville in France.

MR. PARESE: And they've just started moving dirt at Tai Shan in China. Tai Shan in China, it's just west of Hong Kong.

MEMBER BANERJEE: How many will be built in China?

MR. PARESE: Well, right now our contract is for two at Tai Shan.

MEMBER POWERS: What it suggests is that a lot of the first-of-the-kind engineering issues that we have on other reactors are hopefully ironed out.

MR. PARESE: We believe so.

MS. SLOAN: The benefit of not being first.

MEMBER POWERS: Please continue.

MS. SLOAN: Okay. Our goal today was to provide simply a broad overview. Again, we have two hours on the agenda. We could talk forever on EPR as long you want, really, but today we have two hours, so it really is a broad high-level overview of the plant design, and basically focused around comparing and contrasting with current generation PWRs. And what we decided to focus on were those features of particular safety-significance, so that's what you'll see us talking about.

And before I turn it over to my colleague, Marty Parese, I did want to acknowledge and be very open about the fact that in the letter providing Areva the schedule for the design certification review, the staff did identify five areas which were classified as areas of potential schedule uncertainty for the design certification review, and they're in the five topic areas that are listed here.

The first one is post-accident containment mixing, and it has to do with the extent of mixing versus thermal stratification within the containment after a LOCA event, and because EPR does not have safety-related sprays or fan coolers. And Marty will talk a little more about containment design in the context of his presentation.

We've already gotten a set of RAIs related to this. We've responded to some of those RAIs, and there are two RAIs, in particular, related to two topic areas that we are going to provide a technical report to the staff to support their evaluations.

MEMBER BANERJEE: Are there any recombiners?

MS. SLOAN: Yes. There are passive auto catalytic recombiners.

MEMBER BANERJEE: Catalytic?

MS. SLOAN: Yes.

MEMBER BANERJEE: But there's no circulation, no forced circulation of any sort.

MS. SLOAN: No circular -

MEMBER BANERJEE: Either by spray -

MEMBER POWERS: We'll have to do a little proselytizing on the virtues of the spray.

MEMBER BANERJEE: I am not in favor of sprays or in favor of sprays.

MEMBER POWERS: So I've got lots of proselytizing to do.

MEMBER BANERJEE: Okay. So you're going to tell us one of the main issues under each of those before we proceed?

MS. SLOAN: Well, these are the big issues. All I'm trying to do - I'm not trying to steal Marty's time, but just to tell you where we stand on responding to or addressing each of the five items identified by the staff. So I don't plan to go into detail right now.

VICE CHAIR BONACA: This is an overview.

MS. SLOAN: Right.

MEMBER ARMIJO: These issues -- now, this plant has gone through regulatory review by the French and also by the Finnish regulators. Have they addressed these issues themselves and put them to bed?

MR. PARESE: Not that we know of.

MR. COLACCINO: This is Joe Colaccino, again. The regulatory review that has been done for LL3 I believe is what would be equivalent to a construction permit in the United States. I'm not familiar with what has been done with Flamaville-3, but I believe it's a similar path, if that helps you.

MEMBER POWERS: Mr. Bonaca, you had a question?

VICE CHAIR BONACA: Yes, I have a question regarding axial growth in M5 guide tubes. This has been experienced for Areva fuel?

MS. SLOAN: Yes. This has been experienced at a U.S. operating plant. And, consequently, because we're using M5 materials and USEPR fuel, it's been raised by the staff as a potential area that can cause schedule delays.

VICE CHAIR BONACA: You have the same fuel.

MS. SLOAN: Right. And I want to make clear that that is M5 guide tube growth. It's not cladding on the fuel rods. This was observed in the guide tubes; which, for the purposes of understanding, guide tubes for Areva reactors are much like thimble tubes in other kinds of reactors. These guide tubes extend throughout the core region and are part of the skeleton of the fuel assembly.

VICE CHAIR BONACA: Now, this is also being called the USEPR. You talked about other plants being built right now, EPRs in France and in Finland. How different are they? Will you tell us at some point?

MR. PARESE: Oh, the difference between the units themselves in the design features -

VICE CHAIR BONACA: I'm talking about the U.S. -

MR. PARESE: -- in particular, or regarding the fuel?

VICE CHAIR BONACA: This is a U.S. EPR.

MR. PARESE: Yes. There are differences between the unit here, and I'll try and touch on some of those as we go through.

VICE CHAIR BONACA: If you could at some point, yes.

MR. PARESE: Based on how we're doing on time, but I'll try and touch on some of those.

MEMBER BANERJEE: And will you also touch on a little bit more than just the topical reports on how you plan to emergency -- give us a little bit of an overview.

MR. PARESE: Sure. We'll go -

MS. SLOAN: Yes, Marty will talk hardware, so he will talk about that. And mitigation, how it's used to mitigate smaller -

MEMBER BANERJEE: Right. Right. Small, and whatever size.

MEMBER ARMIJO: These four topical reports that contribute to schedule uncertainty, are they yet to be written, or yet to be reviewed?

MS. SLOAN: No, they were submitted and under active review. And on some of them, we have seen the RAIs or draft RAIs, so we're in the process of addressing questions right now.

MEMBER ARMIJO: Okay.

MS. SLOAN: And so for the second item, seismic and dynamic qualification of equipment, the concern was that in our FSAR for the USEPR for design certification, we have left open the option for COL applicants to use earthquake or test experience for equipment qualification. And based on feedback from the staff and our own evaluation, we submitted a letter last Friday to close that issue by taking that option out of the USEPR FSAR. So, at this point, it's our understanding that that is no longer on the list of schedule uncertainty items. And that was that one. We just talked about M5 guide tube growth, and Areva does have an active root cause analysis underway to look at that. That's in progress. We have committed to and continue to keep the staff apprized. We're doing post irradiation examinations. We've eliminated a variety of causes that still haven't come up with a single cause yet, but the root cause analysis is ongoing, and we continue to communicate progress to the staff.

MEMBER ABDEL-KHALIK: How did that issue manifest itself? Was it bowing of the bundles?

MS. SLOAN: This was in the actual guide tube growth up into the upper tie plate.

MEMBER ABDEL-KHALIK: But how did that manifest itself?

MEMBER SIEBER: In other words, what's the interference?

MS. SLOAN: Jeff Tucker is a Fuel America --

MR. TUCKER: This is Jeff Tucker. I'm here with Fuel America. We first discovered this issue doing routine post irradiation exam measurements of discharged fuel at TMI-2 after cycle 15.

(Off the record comments.)

MR. TUCKER: During examination we found that growth rates after two cycles were longer than predicted, so we went back and did more examinations on discharged fuel, and it's been predicted that the fuel might grow to solid contact at reactor shutdown, so we made arrangements for contingencies to evaluate the fuel at shutdown, evaluate the internals, and contingencies to modify the fuel if it was too long. So at the shutdown, we did find out that there were additional fuels in there. We've modified the fuel. We're taking similar growth measurements on fuel at other reactors with similar material and designs. And, to-date, the TMI batch 16 fuel is the only fuel that's got this anomalous growth, and that's the root cause that Sandra is speaking about.

We've done hot cell examinations, we've done post irradiated exams at the pool side, we've done manufacturing reviews, design evaluation, so that's the root cause -- it first manifested itself in routine post irradiation exam and discharge flake measurements.

MEMBER POWERS: We have a lot of evidence to suggest Zirconium and Niobium alloys in reactor environments are susceptible to relatively subtle changes. I remind you of the E110 experience, and now we have a single batch of material here which behaves strangely. Is that a subject that perhaps the Reactor Fuel Subcommittee might want to delve into in a little more -- maybe have a little better understanding of why we have this sensitivity, apparently, of Zirconium and Niobium alloys that we've not experienced with Zirconium -

MEMBER ARMIJO: Yes. We'd love to see your root cause analysis results, and also learn a little bit more about these particular materials. But I'm just anticipating that you'll resolve that problem either by design or material change, or something else. But in the interim, we'd like to learn more about it.

MEMBER POWERS: The trouble I'm having is that each one of these things gets resolved, and then the next one comes along.

MEMBER ARMIJO: Oh, yes, there's always -- well, you can always fall back.

MEMBER POWERS: And it seems to me that we have a sensitivity here that I'm unfamiliar with in the tin, Zirconium Tin alloys, that maybe we need a little more understanding. And I, myself, have not gone back and looked at the electronic structures on Niobium alloy and Zirconium, but my perception is that you're closer to changes in the band gap than you are with the tin alloys, and maybe that's where we're getting some sensitivity there. Anyway, I just suggest that maybe the Fuel Subcommittee wants to go into that.

MS. SLOAN: Okay. And as I said, Areva has been committed to sharing information as we go, as we get new information.

The next item on the list, as someone alluded to, are four methodology-related topical reports that have been submitted. And, as I mentioned, we have received RAIs on these, or draft RAIs, are in the process of addressing the questions. The last item on the list was, I think, one familiar to all of us. This was GSI-191 on sump strainer and downstream effects. And with regard to that one, Areva is following what's going on in the industry, and is actively engaged. And, in addition, we have our own global program within Areva to develop our own technical solution for this, so that work is ongoing at Areva.

MEMBER POWERS: And you're buffering your sumps?

MS. SLOAN: Pardon?

MEMBER POWERS: And you buffer your sumps?

MS. SLOAN: Buffering the sumps?

MR. PARESE: It's not a sump, but, yes, we're doing post-LOCA buffering with Trisodium Phosphate. And we've eliminated any use of Calcil. It's actually -- Calcil insulation is precluded in the design of the plant, design guides. I'm sorry?

MEMBER BANERJEE: Also, anything to do with Nucon.

MR. PARESE: Well, say that again.

MEMBER BANERJEE: Nucon.

MEMBER SIEBER: Fiberglass insulation.

MR. PARESE: Yes. No, right now we have reflective metal insulation on the reactor coolant system.

MEMBER BANERJEE: All of it.

MR. PARESE: All of it. And we're looking at the zones of influence for the attached piping to determine whether we want to continue -- what type of insulation we want to use for that. But right now, our goal would be to go to reflective metal for the zones of influence. And one of the advantages of the EPR, you won't see it any of the layout drawings I have to show, but we have concrete walls between the loops and between the hot and cold legs of the loops, so breaks -- the zone of influence is limited to relatively small areas.

MEMBER BANERJEE: Now, your steam generators will be what, insulated by?

MR. PARESE: Reflective metal insulation.

MEMBER BANERJEE: And all the pipes?

MR. PARESE: The entire reactor coolant system and components will be reflective metal.

MEMBER SIEBER: A lot of cool water pipes typically are -

MR. PARESE: So what you have is, you have attached pipes that you have to insulate to a certain length. Okay? Like your let-down lines, those are heat losses, your ECCS line release for a certain distance will have wicking of heat down those lines, and you want to -- all those are fins, and those become places where heat can be released to the containment, so we will have insulation for a certain distance on many of those attached -

MEMBER BANERJEE: And what will that -- because even small amounts -

MEMBER POWERS: Let me suggest that though we'd love to plunge deeply into the details, I can assure you there's going to be more than adequate opportunity to do this. Maybe at this point, we could get the width or the breadth of the material, and then the strategy for plunging deeper into the details.

VICE CHAIR BONACA: In the USEPR, the methodology that you refer to, the four questions of methodology, evidently, it must be Areva methodology that you use in the States.

MS. SLOAN: Yes.

VICE CHAIR BONACA: How different is the licensing package from the one that you have to license in France and in Finland? I mean, is it a different package? Is it different -

MS. SLOAN: Typically, what we've used for the -- not typically, we have used for the EPR codes, like RELAP-5.

VICE CHAIR BONACA: Yes.

MS. SLOAN: And GOTHIC, and NEEM-OK that are already approved for our use, Areva's use in the U.S. to support operating plants.

VICE CHAIR BONACA: Okay.

MS. SLOAN: And, of course, what's being used in the other countries are things their regulators are -- the regulatory regime over there is more familiar with.

MR. PARESE: So, for example, for Flamaville, the LOCA analysis, small LOCA analysis will be done with CATHR, and here we're using COF-5, so we did not rely on work that was done using codes approved in Europe here. We used our own codes.

VICE CHAIR BONACA: Okay.

MR. PARESE: Used US-approved codes. Now, that doesn't mean we didn't learn a lot from everything that had already been done, of course.

MS. SLOAN: And so these are the general topic areas that we had hoped to touch on today. And there's a lot of overlap between these various topic areas. I would encourage you to ask questions as we go along. I know no one is shy to do that.

MEMBER POWERS: Oh, you don't need to do that.

MS. SLOAN: I know.

MEMBER POWERS: That's waving a flag in front of a bull. No. Let's hold your questions and get through this.

MS. SLOAN: And I'll turn it over to Marty Parese. Marty is the Chief Engineer for Areva NP, Inc., and as one of his many responsibilities as Chief

Engineer, he's responsible for technology integration for new plants deployment.

MR. PARESE: Okay. So today, based on your request, we're going to talk about the differences between the EPR and a standard PWR, because I think everyone realizes we're an evolutionary design. But as we go through it, we're going to do some comparisons with existing PWRs, as well. And, generally, a standard four-loop type unit that you'll find in the U.S.

So the important thing about EPR is that the development objectives were clearly to make it evolutionary. And that decision was made at the beginning of the development phase in 1989-1990, and so we built on all of the experience that existed on current PWRs and the plant performance and equipment performance would be predictable. So that was purposely selected.

The French and German regulators were involved in the developed of the EPR design objectives, and the licensing guides that would be used for EPR. So, consequently, increased safety of the unit as measured by increased design margins, increased redundancy, and diversity and physical separation at multiple levels, as measured by a reduced core damage frequency, as a Generation 3-plus unit should have. And accommodate severe accidents and external hazards with no long-term local population effect. And we'll talk about those design features, in particular, and also from an occupational standpoint, to reduced occupational dose to the workers in the plant, and so there are design features aimed specifically at that.

And then, of course, the utilities wanted to get -- obviously, they're going to be buying and using the units. Many utilities in Europe were involved in the original development. They developed a utility requirements document, the EUR. Also, the EPRI URD was also used for guidance, as well as other operator experience with the units. And they wanted to improve the operations by reducing the generation cost by at least 10 percent. And this generally is measured as regulated utilities tended to do that, as a lifetime generation cost.

MEMBER BANERJEE: Is this basically like the German Siemans design?

MR. PARESE: We're going to talk about in just a moment. But, yes, the EPR is an evolutionary design based on the features of the N4 in France, and the Convoy design in Germany. And those designs were based on the previous designs as they moved forward. And those designs were based originally on licensed technology from the United States.

So to meet these economic and safety objectives then, certain design features were developed through the 1990s, and so we'll be talking about many of these features. The nuclear island, we're using a proven four-loop reactor coolant system design; the reason being, the four-loop design can generate large power output, and that large power output when put in the denominator of any O&M cost, of any fuel cost, of any kind of operating cost lowers the dollars per megawatt hour, so you get an economy of scale when you have a larger power output.

MEMBER SIEBER: Gross megawatts?

MR. PARESE: I'm sorry?

MEMBER SIEBER: What the gross megawatt output?

MR. PARESE: Gross megawatt output of the -- the gross output of the units in Europe is over 1750 megawatts electric.

MEMBER SIEBER: That's three LPs and one HP?

MR. PARESE: Yes. In the U.S., we can't use open loop cooling as they do in Europe. And, also, the French have coined the term for us called "tropicalization". That's what we did when we converted the unit to U.S. temperatures. So, whereas, tropicalization - so I'll give you a perfect example. The temperature from the Baltic Sea or even if we look at the English Channel in the summertime, they can pull cooling water in that's 72 degrees Fahrenheit, and so they have in the summertime a back-pressure in the condenser of about 1.8 inches, 1.7 inches of Mercury. And we will have -- we have to use a cooling tower, and we'll expect wet bulb temperatures of 70 some degrees, which will give us a condenser inlet temperature of 84 degrees Fahrenheit, and so we won't produce 1750, we'll produce 1711.

MEMBER SIEBER: Okay. So you can't make it up on the condenser -

MR. PARESE: No. What we did do is we increased the power level, so one of your differences right off the top, the EPR in Europe is generally 4300 megawatts thermal, and here in the U.S. we're 4590.

MEMBER SIEBER: You get the same megawatt --

MR. PARESE: The first heat balance we did on the USEPR in the spring of 2005, we were delivering a net output with house load, so a net output of 1505, and so by increasing the thermal power, optimizing the condenser, and using ultrasonic heat water measurement to reduce the calometric uncertainty, we got the output up to 1711 gross, 1580 net. And that's at 2-1/2 inches of back-pressure. We expect the average output throughout the year to be about 1595 -

MEMBER SIEBER: So your station service is 122 megs?

MR. PARESE: 130 megawatts, approximately, is our house -

MEMBER SIEBER: A lot. Do you have electric feed pumps?

MR. PARESE: We have electric feedwater pumps. We have electric condensate pumps. We have mechanical draft cooling towers.

MEMBER SIEBER: Natural draft -

MR. PARESE: You can use natural draft towers, but it generally takes two 500-foot natural draft towers.

MEMBER SIEBER: Right.

MR. PARESE: Because we're such high power. Whereas, you can use one much smaller mechanical draft tower with 48 cells and produce a little bit better approach temperature, and get a little more megawatts out.

MEMBER SIEBER: Yes, but do you get -

MR. PARESE: Yes.

MEMBER SIEBER: I'm sure you guys have figured that -

MR. PARESE: It turns out to be a wash.

MEMBER SIEBER: If I were buying one, I'd ask that it be -

MEMBER POWERS: Unless you've become a good deal more wealthy than you were last week, you're not buying one.

MEMBER SIEBER: Well, I'd have to change employment anyway.

MR. PARESE: To increase the redundancy of the unit, we use generally four-train safety systems for all the front line safety system. We'll talk about the advantages that that gives us later.

MEMBER POWERS: How about the disadvantages?

MR. PARESE: Well, the disadvantage is, obviously, cost, but you have to offset by putting that big power level in the denominator.

MEMBER ARMIJO: Another thing that goes kind of in your denominator is the design life. You picked 60 years, but is there a fundamental limitation at 60 years, or do you think there's more capability in the system?

MR. PARESE: Oh, there's more capability after that.

MEMBER ARMIJO: You're anticipating there might be a plant life extension.

MR. PARESE: Yes. But right now, 60 is what goes into the design. And there's some equipment that you can't design to 60 years. First of all, there's some suppliers that won't supply equipment with that design life. They just won't do it. And then you have other equipment that has a very short lifetime, anyway, like certain -- and, obviously, all your consumables, like o-rings, and gaskets, and wear parts.

MEMBER SIEBER: But your active equipment is going to be periodically inspected and deficiencies corrected, and parts renewed.

MR. PARESE: That's right.

MEMBER SIEBER: To get this kind of output, what's the size of the core, it's overall dimensions?

MR. PARESE: I knew you were going to ask that. It's 241 fuel -- we're going to get there. It's 241 fuel assemblies. I believe the diameter is 100 -

MS. SLOAN: We'll make it. There is a section on the -

MR. PARESE: Yes, we'll get there.

MEMBER SIEBER: Okay. Good enough. How long?

MR. PARESE: Fourteen foot.

MEMBER SIEBER: Okay.

MEMBER BANERJEE: And you're going to tell us what pressures these safety systems come in as they pass -

MR. PARESE: If we can get to it.

MEMBER POWERS: We are not going to at this rate.

MR. PARESE: To help this out, we're taking suction on the safety injection system from an in-containment refueling water storage tank, and so it's used for refueling operations, as well as for safety, and it's inside containment, so that simplifies a lot of the connections. And it gets rid of the switch over during LOCAs and the operator actions, which we'll talk about later. One of the objectives of this design is to reduce operator action and give long operating times for response, so a minimum design requirement was any action that's required within 30 minutes must be -

MEMBER BANERJEE: These are low pressure injection systems.

MR. PARESE: We have medium head safety injection, and low pressure safety injection. We'll get to that.

MEMBER BANERJEE: There's no high pressure injection.

MR. PARESE: No high pressure safety injection. We'll get to that, too.

MEMBER SIEBER: Containment is a steel shell with concrete?

MR. PARESE: We're going to get to that, too. So we've included severe accident mitigation to meet those requirements we talked about, no long-term effect on the population with separate safeguard buildings to house those four different divisions. And we're using digital I&C and advanced control room.

In electrical, each of those four divisions is supported by its own emergency diesel generator. And to back those up in case of station blackout, we have two smaller diverse station blackout diesels. The emergency diesels are water-cooled. The SPO diesels are air-cooled. And based on their size, it's very likely they'll be by different manufacturers, so that's where we're going to have our diversity. And then we also allow for island-mode operation for the unit, so we can take a full load rejection and transition to delivering our -- disconnecting from the grid if the grid goes down, and delivering our power to the switch yard, and then running the unit off those loads. And that gives us an advantage, at least for some period of time while the grid is down, the reactor can stay operating producing power. And it could provide the ability to black start the power through the units around it, as long as it's not a sustained loss of the grid.

And then site characteristics in regard to we have airplane crash protection, and we also have protection against explosion pressure waves, and we're going to discuss that today.

So quickly, here's generally the layout of the USEPR. I'll point to one of these screens, but the reactor building, obviously, you can see that in the center. That reactor building is a system. It is a post-tension concrete containment building with a steel liner surrounded by reinforced concrete shield. Arranged around the reactor building, we have four safeguard buildings, Safeguard Building One, Two, and Three, and Safeguard Building Four are radially arranged, and I'll talk about the advantages of that.

The main control room is in Safeguard Building Two, and the Safe Shutdown Facility is in Building Three.

A fuel building, we have external storage of fuel in its own fuel building. That includes new fuel acceptance, spent fuel storage, and it also includes simplified methods to take irradiated fuel and put it into casks either for shipment off-site, or for placement in an independent spent fuel storage installation.

And then you can see we have a nuclear auxiliary building which contains all the systems that you would normally expect to keep your reactor coolant water clean, and keep your secondary water clean, and account for changes in volume and boration of the system. And then we have a rad waste building, which is a dual-purpose design right now. If the utility wants to process its radioactive waste in its entirety, we have the equipment and the systems to do that. If they choose to, especially for liquid waste, if they choose to contract with subcontractors like many are now, then we have the ability for the subcontractor to come in and valve up their demineralizers, and process, and then take it off-site, so we basically allow for them to approach.

We have an access building here that

controls all the access to the nuclear island and the radiological controlled area that's set up to handle over 300 people every 30 minutes during an outage. And then you see the turbine island and the switch gear building. Here we have the emergency power generation buildings. Each of these buildings has one EVG in it, and has fuel tanks to support that EVG. And you can see for Safeguard Buildings One and Two, it's on one side of the plant nearest to those buildings, and Three and Four is on the other side of the plant. Again, we'll talk about our separation of these structures for hazards.

What's different about the USEPR and European designs are the ultimate heat sink. These essential service water cooling structures, those are mechanical draft cooling towers with faces, one for each of the divisions. In Europe, they use open-loop cooling, and here it's sometimes impractical to do that with permits with the EPA and whatnot. Also, that means that these structures are inside the protected area.

VICE CHAIR BONACA: Why did you list airplane crash protection as a site characteristic?

MR. PARESE: Because of the way that we approach the protection, which I'll talk about.

VICE CHAIR BONACA: Okay.

MR. PARESE: And that's separation, as well as shield buildings.

VICE CHAIR BONACA: All right.

MR. PARESE: So here, just looking down on it, then what I -- the main point of this slide is simply to point out that everything that's required for protection within the security plan is inside the protected area. And that's about all we'll talk about that today.

So these concepts are shown together, actually, there's three concepts on the slide. There's one in particular I want to talk about, two I want to talk about. The radial design, we have in the four division approach, where we have injection to individual loop, we set it up so that each division, the medium head safety injection, the low head safety injection, the emergency feedwater injects into one loop, and so Division One, Two, Three, Four, each one connects to its own loop. Each takes suction off of the IRWST, what you see here, the In-Containment Refueling Water Storage Tank, takes suction, goes through its heat exchanger and reinjects. The emergency feedwater, obviously, has a tank in the building that it takes suction from to inject. The point is, this radial design then keeps all the links of pipe short, and by minimizing any inter-connections we reduce the number of valves, and complexities. There's no requirement for operators to balance flows during design-basis accidents.

The other thing then, you can see the separation of the buildings. Each of these buildings then, if you have a calamity in one of these buildings, say a fire, then the other buildings aren't affected by the fire due to the separation, the radial design.

Then the N+2 approach allows us for these front line safety systems to have one system in preventative maintenance, so you can do on-line maintenance of a system. We can also then have our single failure criterion on a system. So, for example, you could take loss of off-site power and the failure of an emergency diesel generator, and then all the powered equipment on that division is assumed out. And that leaves us two divisions to mitigate the event. So for those events that could affect the delivery of the cooling water, for example, a loss of coolant accident, one of our active divisions could be in a broken leg, and it could be falling on the floor. That allows one division to deliver water into the vessel and mitigate the event. For those events where that's not possible, we have two divisions out there.

MEMBER SIEBER: Each of the four divisions is full capacity?

MR. PARESE: Essentially, all you need is one.

MEMBER SIEBER: Well, tell me why you use the words -

MR. PARESE: Well, the reason I used essentially is that we took credit for the fact that generally -- well, under these assumptions, two RHR systems would be operated. So even if one is dumping on the floor and running into the IRWST, it's taking suction out of the IRWST and it's running through a heat exchanger, and it's reinjecting it back to the either the floor or the IRWST. So, in reality, during a loss of coolant accident, I have two divisions taking heat out of the building. Okay? That's why I said "essentially". There's some -- and we're going to talk about -- in just a few minutes, we're going to talk about systems that are 2X100, not 4X100.

And then the other thing that shows here, which we'll talk about in a moment. This blue building is the reinforced concrete building that goes around the reactor building. It goes around the fuel building, and safeguard buildings. So on slide 12 then, what this does for us is it allows us to lower the cost of the unit in some ways. We increased the cost because we have four divisions, but we reduced the cost, or at least we improved the economics of the unit because we can do on-line maintenance.

Because you can do on-line maintenance, you take EDG maintenance, MHSI, EFW pumps, heat exchangers, component cooling water, you take surveillances and maintenance out of the outage, and so you can shorten the outage time to 15 days. Current plants are running about 35, the best PWR outage I think is still Byron at 15 days something hours. So if you shorten your outages by 17 to 20 days, you're going to improve the economics, because you're going to produce power during those days. That's one thing.

Second, because we can do the preventative maintenance on line, we can have a higher availability of the equipment. But, also, we can use equipment that's literally the same size or capacities that we're used to now. This is a 4590 megawatt unit. Our MHSI pumps are about 600 gallons per minute, at around 600 psi. What's the size of MHSI pumps now on current units? It's the same. Our LHSI pumps are 2200 gallons per minute at around 200 psi. That's the same, so we're using equipment that we're used to operating. We're not using special or newly developed equipment. And also then we know the Lessons Learned on all the existing fleet and materials of construction, and problems.

MEMBER POWERS: You might actually be able to estimate reliability on these things.

MR. PARESE: That's our expectations. So on slide, I guess it's 13, it's cut off a little bit. For the main safety systems, as we've said, we have four-train ECCS, so we have four medium head safety injection pumps. We have four combined LHSI RHR pumps. They're one per division.

MEMBER ABDEL-KHALIK: What is the shut-off head of your SI pumps?

MR. PARESE: The shut-off head of the SI pumps is around 1380 to 14 psi. And we're going to get into that later in the presentation.

Obviously, we have charging pumps, non-safety charging pumps. And it's pretty interesting how some of the changes that were made even to a subtle system like that; for example, current units vary the charging flow to adjust pressurized flow and account for changes in density of the coolant system.

Right?

MEMBER ABDEL-KHALIK: Right.

MR. PARESE: Well, that causes variations in flow of those nozzles. And because those nozzles are in the stream of the cold leg, that cold leg water goes in and comes out, and causes thermal penetration, and causes cyclic fatigue of a nozzle. Well, we solved that. We control pressurizer level by varying let-down. And by varying the let-down flow, you're just changing the flow of a relatively hot system 570 degrees, and so there's very little fatigue on that nozzle due to variations in flow. So we solved one of those big problems with make-up nozzle cracking, and other problems, and thermal sleeve cracking by just making a simple adjustment to how we run the unit. So that's an example of how lessens were incorporated.

MEMBER SIEBER: By using the let-down flow you charge back in, I take it, your EG trains or arrangement is such that you don't have a big temperature differential in -

MR. PARESE: Right. We're using a combination of regenerative and non-regenerative heat exchangers to warm the charge -

MEMBER SIEBER: Okay. The resulting temperature is usually lower because you're affecting the non-regenerative part.

MR. PARESE: Right. And it is lower, but it continuously injects; and, therefore, we don't get the thermal transients on the nozzle.

VICE CHAIR BONACA: What is there, the shut-off head of your charging pumps?

MR. PARESE: Shut-off head of the charging pump, I believe approaches 2750 psi.

VICE CHAIR BONACA: Okay.

MR. PARESE: So one part of the flow curve we're still getting a flow of 2680 psi.

VICE CHAIR BONACA: Okay.

MEMBER SIEBER: And it's a centrifugal -

MR. PARESE: It's a centrifugal, it's two centrifugal pumps in parallel, one normally operational, the other one is in standby. We do have two positive displacement pumps in that extra borating system, and they deliver about 40 gallons per minute. And we use those with hydro tests on the reactor coolant system, but they have a safety function, as well.

MEMBER SIEBER: And you can put boron in for shutdown insurance. Right?

MR. PARESE: That's right. So our extra borating system is manually actuated, it's not automatic. It's manually actuated. We have two of them, so we can take a single failure, and it injects 7700 ppm boron of enriched B-10, 37 percent enriched B-10.

MEMBER SIEBER: That's safety-related?

MR. PARESE: It's safety-related, and so to meet Branch Technical position, used to 5.1, it's now 5.4, I think. To get to cold shutdown, we can borate to cold shutdown using those pumps.

MEMBER ARMIJO: Your two non-safety-related charging pumps, are they on different power supplies or the same? If you've got two in parallel, normally one running.

MR. PARESE: I don't know the answer to that. I'd have to look if they're on the normal power bus, and I don't know if they're on the same or different buses.

All right. And then for severe accident mitigation, we have a non-safety-related containment spray system that has a dedicated component cooling water and central service water train that goes out to one of those mechanical draft cooling towers. And we'll talk about severe accident mitigation.

MEMBER STETKAR: You're selling -- does it have a containment vent?

MR. PARESE: I'm sorry?

MEMBER STETKAR: Does the USEPR have a containment vent system?

MR. PARESE: Well, the answer is we have it, but it's not part of our normal severe accident mitigation. In other words, it will be in the SAMGs as a last resort, but we've designed -

MEMBER STETKAR: It's part of the design.

MR. PARESE: It's part of the design, but we've designed the plant so you won't need to use it.

On the secondary side, as we said, each steam generator has its own EFW supply for safety assured water, and that tank is in the safeguard building. And there's one pump, and one tank, and it discharges to the steam generator. It has suction valves, and discharge valves so that we can, after the early stages of the event, whatever event you might have, and what single failures you might have, later in the event, the operator can get access to any tank of water to deliver to any steam generator, depending on what's failed and what's not failed, so we have that capability. But when the event begins, each injection line goes to each steam generator.

MEMBER SIEBER: What's the capacity of each steam water tank in terms of hour, decay heat removal hours?

MR. PARESE: We've got decay heat removal capability of at least 24 hours hot.

MEMBER SIEBER: Single tank.

MR. PARESE: A single tank is -- well, it's approximately -- they're not equal in size, but it's approximately one-fourth of that.

MEMBER SIEBER: Six hours or so.

MR. PARESE: The four of them together give us 24 hours hot, or allow us to cool down to cold shutdown, or to get to RHR. I should say to get to RHR actuation, and at 250 degrees Fahrenheit.

MEMBER SIEBER: If you only have one train of emergency feedwater, you have to cross-tie tanks to get to 24 hours.

MR. PARESE: Yes. You would open up -- you would take suction from those other tanks to get there.

MEMBER ABDEL-KHALIK: Can the steam generator inventory itself, how much worth of decay heat can -

MR. PARESE: We've got almost 30 units of decay heat removal in the steam generators post reactor trip. There's 182,000 pounds of water, and we're going to show that in a comparison slide in just a second.

Also on the system, each steam generator has, besides the turbine bypass system, on each steam generator we have two spring-loaded main steam safety valves that are worth 25 percent each, and we have one main steam relief train, which is safety-related, ASME qualified. And it's made up of an isolation valve, and of a control valve, and it's seismically qualified, redundantly powered, and we can use that to depressurize the plant to cold shutdown using those safety-related atmospheric dumps. So this is something a lot of the current units wish they had, so that they could take credit for depressurization of the steam generators. We built it into the design. It's 50 percent total flow at full pressure.

It turns out in our -- it doesn't turn out, the plant was designed so that for the limiting over-pressure event for the secondary side, either the main steam relief train by itself, or the two spring-loaded safety valves by themselves can prevent the system from exceeding 110 percent.

So Slide 15, checking my time, slide 15, this is just an example where you can see in a division, say the safeguard building, let's pick Safeguard Building Four, the residual heat removal system which would take -- would drop off the hot leg, or take suction out of the IRWST if it's an accident, goes through a heat exchanger, the RHR heat exchanger, and reinjects back into the reactor coolant system. That's inside the safeguard building. Also, a component cooling water pump is inside the safeguard building. And with a heat exchanger there, the component cooling water heat exchanger, all of that is self-contained in the safeguard building. And then the essential service water system connects, and so one division has its own RHR component cooling water, and essential service water, and alternate heat sink. And that's consistent in the design. And in that safeguard building, we have everything to control that system, so we have the mechanicals in there, we have the electrical power supplies, we have the I&C control, and we have the HVAC in that building to keep that building cool from all the heat loads that could be deposited in the building.

MEMBER ABDEL-KHALIK: Are there structural differences between the Safeguard Buildings One, Four, versus Two and Three?

MR. PARESE: Yes. Well, partially. The actual building itself, no. They're all seismically qualified safety-related buildings, but One and Four do not have a shield building for external hazards from airplane crash. The reason for that is, they are separated by the reactor building, which does have a shield building. Consequently, if there's a calamity on one side of the plant, it can only affect one safeguard building, and can't affect both. So even if we had an aircraft hazard or an external explosion that damages some of the equipment in the safeguard building, you still have three divisions available to perform functions and get the cold shutdown.

And so this just shows exactly what we were talking about, where everything is self-contained in one building. You can see the mechanicals are the low level in case of line breaks or flooding. Here's our pool that's inside the building, so the tank is inside the seismic structure. Then we've got our cable spray for -- we've got some cable spraying force, and our electrical floor that has our I&C cabinets inside this shear wall, and our electrical switch gear in the outside of the shear wall. And here you can see the main control room. And above that, our HVAC equipment, so it's all logically aligned inside a building.

Now one of the differences between the USEPR and the European version is that these electrical cabinets, when you go to IEEE cabinets, they're much bigger. We needed much more space, and so we had to make room outside the shear wall. So all the safeguard buildings are three meters longer in the radial direction, 9.9 feet in a radial direction, which costs money to do that, but it also gave us the advantage of having some room for some of this other equipment, because in our tropicalization discussion, we had to improve the heat transfer and the component cooling water to help us jump to a higher heat sink in the cooling towers. So that gave us the possibility to increase the sizes.

MEMBER SIEBER: Where did you say the control room was?

MR. PARESE: The main control room is right there. So, as we said, our front line safety systems, the protection system, which includes reactor protection and ESF functions, so the protection system, the emergency power supplies, emergency core cooling, component cooling water, essential service water, EFW, those are 4X100, but not all systems are 4X100, so we wanted to point that out so that there wasn't confusion. And you can see, much of our iodine filtration, annulus ventilation, safeguards and fuel building filtration, control room iodine filtration is

2X100, largely because those systems can't be affected by an accident. And maintenance on those systems are pretty straightforward. All we have are fans and filters, so maintenance can often be done on line on the systems, but they can easily be done during an outage. It's not a critical path item.

Containment isolation by its nature, there's a valve on each side of containment, and you power one off division one, and one off division four. Well, then it's a two division system, whether you like it or not, unless you put in extra valves, and that didn't seem appropriate with a single-failure criterion.

Our extra borating system is two divisions. It's actuated manually, so we felt two met our single-failure criterion, and that was appropriate. And then spent fuel pool cooling is 2X100. Again, it is not affected by an accident.

MEMBER APOSTOLAKIS: You said earlier that the ECCS essentially, you used the word essentially, what -

MR. PARESE: Right. The ECCS, if you have a small or large loss of coolant accident, the ECCS, one division will function to mitigate the event. But because the divisions are actually running, we take credit for the functions that they perform that might not be injection functions.

Another way of putting it is if I have two RHR systems operating, which I always will under the N+2 assumption, I can always cool the unit down in a relatively short time. I think our target is 34 hours or something like that. If I only have one, it takes much longer. Can I get there? Yes, but it takes longer with one, but I always have two. So we credited the fact that I always have two. But for the injection into the vessel for flooding the core, we take credit for the one -

MEMBER BANERJEE: So without an HPIS you have to do something else to bring the pressure down, I mean in a SB LOCA.

MR. PARESE: Well, you're jumping ahead in the homework. We'll get there. You're right. You're exactly right, and we're going to talk about this.

MEMBER STETKAR: You're not going to talk about -- I looked ahead. The extra borating system, does that ATWS, direct ATWS mitigation capability, or is just a cold shutdown?

MR. PARESE: It has that ability, but we handled ATWS completely different.

MEMBER STETKAR: Okay. That's fine.

MR. PARESE: I'll tell you.

MEMBER STETKAR: No, that's okay. I just want to make sure -

MR. PARESE: We handled ATWS through diverse actuation, but you can use it for that. No doubt about it.

MEMBER ABDEL-KHALIK: If your component cooling and service water are 4X100, wouldn't that imply that you can do the cooling with one set of heat exchangers? Why would you need to take credit for the cooling provided by the affected loop?

MR. PARESE: Well, we take credit for it because it's there, simply because it's available. Whether that leg is broken or not, I'm cooling the water in the IRWST -

MEMBER ABDEL-KHALIK: I understand, but if you're implying that your component cooling and service water are four times one hundred, that means you can do it with one set of heat exchangers.

MR. PARESE: It could. It could. That's not how we applied it in our safety case. For injection into the vessel, it's one division. Okay? It's one division, and for your large and small break analysis to show peak clad temperature and cladding oxidation and whatnot, that analysis is a certain length when you hit your stable condition and you quench the core and cover the heat. You show the continued cool down of the unit to cold shutdown, we take credit for the equipment, for the safety-related equipment that's available. Would it take longer if we only had one division? Yes, it would take longer, but we credit two because we have two. That's all.

Protection against external hazards, as we said, we use two basic philosophies to protect structures from external hazards. One is with shielding, a shield building, a concrete shield building, and the other is with physical separation. So as you see, for example, our emergency power supply buildings that have our emergency diesel generators are on opposite sides of the building so a calamity on one side of the plant can't affect both. The same as for the essential service water, they're protected by separation. Building One and Four, the ultimate safety response of the unit is protected by separation. These other items, access building, rad waste building, turbine island, they're not protected. That's simply a commercial risk depending on what calamity you might postulate. So that's the general philosophy of the approach, and that's why some of the buildings don't have the shield buildings.

MEMBER SIEBER: Your main unit transformers and auxiliaries are in a building?

MR. PARESE: No, they're up here.

MEMBER SIEBER: They're outside then.

MR. PARESE: Just outside the turbine island up here. The switch yard is usually up here.

MEMBER SIEBER: Far enough away that when they explode and burn, they aren't going to burn the turbine building down. Right?

MR. PARESE: Yes.

MEMBER SIEBER: Okay.

MR. PARESE: And, also, for further separation, our two station blackout diesel generators are in the switch gear building. And that's also close to where they connect to those buses and give us power, separation there. So a calamity to the turbine building isn't -- and it could affect the switch yard, isn't likely to affect our power generation. A calamity that could affect our emergency power generation is unlikely to affect the switch yard, and so on.

MEMBER SIEBER: On your main unit transformer, is it a single three-phase transformer, or three one-phase transformers?

MR. PARESE: We're using three normal

auxiliary transformers.

MEMBER SIEBER: Main unit transformers, three.

MR. PARESE: Three.

MEMBER SIEBER: One per phase.

MR. PARESE: And, also, we have two emergency power supply transformers. It meets the emergency –

MEMBER SIEBER: About 100 megawatts apiece?

MR. PARESE: I don't know.

MR. FRANKANESE: Excuse me?

MR. PARESE: He asked if they're 100 megawatts apiece?

MR. FRANKANESE: The GS used, generation up transformers?

COURT REPORTER: You need to identify yourself.

MR. FRANKANESE: I'm sorry. I'm Dick Frankanese, Electrical I&C Manager.

MEMBER BANERJEE: Microphone, you have to use the microphone.

MR. FRANKANESE: Okay. The question was?

MEMBER SIEBER: How many auxiliary or station transformers do you have? What's their capacity?

MR. FRANKANESE: I don't know the capacities. I believe we have, we consider we have five, three plus two, and there's three single-phase generators to up transformers, 500 kV. They're at the end of the turbine building, and the rest of the electrical equipment is in the switch gear building, which is to the left of the turbine building.

MEMBER SIEBER: So you probably have two station service transformer chains with probably 120 megawatts apiece?

MR. PARESE: I couldn't tell you the size.

MEMBER SIEBER: I'll find out later, I'll bet.

MR. PARESE: Here you can see on the reactor building, you can see the reinforced concrete in these buildings, and it's decompartmented from the containment building. In other words, they don't touch in their design in case of an aircraft hazard, aircraft impact that they don't touch, the deflection won't cause the outer building to touch the inner building, so that any affect of the impact is driven through vibrations down to the basement and back up, but no direct contact.

MEMBER SIEBER: Is that a negative

pressure in-between?

MR. PARESE: That is a negative pressure. It's small and large in short filtration.

MEMBER SIEBER: Okay.

MR. PARESE: So that's one of the design features of the containment, is that this annular region is filtered so that any leakage during the design basis event that could get into that annulus is filtered before release. And that's done by a safety-related system, 2X100.

The free volume is about 2.8 million cubic feet, and the design pressure is 62 pounds, and the in-containment refueling water storage tank is about 500,000 gallons per minute, so we've also included severe accident features.

Now, as we said before, the containment does not have safety-related spray, and it doesn't have safety-related fan cooler units. Normal cooling of the containment is done with standard HVAC equipment which is in these equipment spaces. Well, on this one it's C, are in these equipment spaces. And it was designed so that you can access these equipment spaces and any of these spaces above the bio-shield during power operation, and the atmosphere is maintained at less than 86 degrees Fahrenheit. So during a loss of coolant accident, the discharge from the vessel or from the break goes up through these cubicles, both the pump, or the pump and the steam generator cubicles, and exhausts into the building where then steam begins to condense on all the concrete and steel structures. And that's typical of a current containment, for example.

MEMBER SIEBER: HVAC.

MR. PARESE: Yes.

MEMBER SIEBER: To avoid overload on all the fixtures.

MR. PARESE: Yes. And it's all non-safety anyway, so -

MEMBER SIEBER: Yes, but you -

MR. PARESE: Yes. You don't want to ruin it.

(Off the record comments.)

MR. PARESE: So generally then during a loss of coolant accident, circulation patterns are up through these compartments into the main containment where we condense on all these surfaces. We have a little over 700,000 square feet of sealant and concrete surface area in this unit.

MS. SLOAN: We should mention that these are not -- this is a backup slide that Marty jumped to, so you won't find it in our slide packet.

MR. PARESE: And I jumped here because you asked if we were going to discuss it, so we'll discuss it. So that condensation path then allows the water to run down to the lower levels of the containment, all of these floors are lined with drains so all water drowns down, and then goes into the IRWST where it can be reused for injection and cooling by the ECCS, so the ECCS system will take suction out of the IRWST, it'll cool it in a heat exchanger. It'll inject some of it back into the vessel, most of it, and it'll recirc some of it to the IRWST to cool the IRWST. Also, some of the fluid is injected across the sump screens or the IRWST screens, we'll call them sump screens for now, to provide flushing of the screen.

MEMBER MAYNARD: Are each of the four steam generators enclosed individually?

MR. PARESE: Yes.

MEMBER MAYNARD: Okay.

MR. PARESE: It's like current D-rings but with a wall between.

MEMBER ABDEL-KHALIK: Could you just mark the boundary of the area that's accessible during operation?

MR. PARESE: I had a better slide. I

didn't provide it. Accessible -- let's do the unaccessible area. That's easy. The unaccessible area is inside this shield wall right here, basically this area right here, what we call the equipment center. Outside the shield wall we have rooms and other compartments of equipment that you might want to access during operation or getting ready for an outage. The design for OL-3 is that even on the operating deck -

MEMBER STETKAR: Marty, come back to the -

MR. PARESE: I'm sorry, the microphone.

MEMBER STETKAR: Yes.

MR. PARESE: Even at the operating deck at OL-3, the design is to maintain the dose rate to less than 2 MR per hour. Clearly, it wouldn't be a requirement in the United States to be 2 MR per hour, but we do have shielding in place to protect workers who have to enter containment, or we might want to enter containment. It also allows us to do certain calibration of the refueling equipment, the heavy crane, maintenance on the stud tensioner if we leave it inside containment. All that can be done while the power plant is down-powering for the outage.

MEMBER ABDEL-KHALIK: So even though these areas are accessible during operation, there is -- they're physically separated, and yet during an accident you allow steam to escape into the accessible area?

MR. PARESE: Yes. And so now you've hit on one of the design features. This steam generator cubicle is covered with a metal foil. That metal foil helps us keep the air separated between the two compartments during operation and controlling to different temperatures. Obviously, all this zone out here is 86 degrees Fahrenheit, and here our limitation is concrete temperature, so it's 140 Fahrenheit.

During an event, the over pressure for the loss of coolant accident ruptures the foils and just opens up. Also, on top of the pump we have dampers, metal dampers that due to the pressure open up. Also, down here to allow water to drain to the IRWST, we have radial dampers around the IRWST that open and let the water flow in, and so what happens is it becomes one large containment. So the heat source here act like chimneys and cause the steam to rise. It causes a lower pressure, the condensation is going down around the outside, so we've got liquid going up, or vapor going up, and liquid coming down. But it also allows us to pull an air vapor mixture through the

IRWST through these holes, and back up through the chimney, so you get a circulation effect that is one of the features of severe accident mitigation, and allowing our hydrogen -- our passive auto recombiners to reduce the hydrogen content.

MEMBER BLEY: What opens the dampers at the bottom?

MR. PARESE: The dampers at the bottom, I believe they are opened by -- those are held shut I think by springs, and they are opened by an actuation of the protection system.

MEMBER BLEY: Like releasing a catch or something like that?

MR. PARESE: And so then they'll open, the failsafe has to open.

MEMBER SIEBER: I take it it's an atmospheric containment?

MR. PARESE: Yes.

MEMBER SIEBER: Maximum temperature occurring, the number --

MR. PARESE: I think we did -- I don't know the exact number. I thought we did our analysis at 86 Fahrenheit plus. I'm uncertain -

MEMBER SIEBER: Well, that's the outside area, inside containment is usually well over -

MR. PARESE: No. What I'm saying is, we control this air inside containment, but outside the equipment space at less than 86 degrees Fahrenheit. And then inside has to be less than 140. But in our containment analysis, we applied some uncertainty on the initial condition. I don't remember what that is.

MEMBER BLEY: Up in the upper corner of the inside shell compared to the outer one, kind of nubbins where they come together.

MR. PARESE: Right here?

MEMBER BLEY: Yes. How close is that? In a bad seismic event, maybe beyond the design basis, can they bump? Have you done a seismic PRA or anything like that?

MR. PARESE: No, I can say we haven't. What we've looked at is our design aircraft impact, and they don't touch. I don't know the answer to that.

MEMBER ABDEL-KHALIK: What's the gap -

MR. PARESE: We haven't done any calculations, I think. And I don't remember what that space is. The space of this annulus here is approximately 6 feet.

MEMBER BLEY: I knew that was -- it's hard to tell how close -

MR. PARESE: This wall is 5.8 feet, and 5.8 feet, but -- 4.3, so right there.

MEMBER MAYNARD: For the U.S. regulations, is it the inside one that you're taking credit for for containment?

MR. PARESE: Yes. That's exactly right. So the inside one keeps what's inside in, and the shield building keeps what's outside out.

VICE CHAIR BONACA: Could you show me what is the ground elevation?

MR. PARESE: Ground elevation is like right in here.

VICE CHAIR BONACA: Okay. So it's mostly out above ground.

(Off mic comment.)

MR. PARESE: Yes, right near that equipment hatch.

All right. So here's a place where we can save time. The reactor coolant system is a conventional four-loop PWR, and we built in a lot of Lessons Learned, or experience gained, as our marketing people expect us to say. And we've increased the grace period for a lot of transients by increasing the capacities of sizes of a lot of the equipment.

MEMBER SIEBER: What materials are the welds made from?

MR. PARESE: Well, that's a good question. All of the materials, the hot legs and cold legs are all forged stainless steel, and the -

MEMBER SIEBER: Forged, not cast.

MR. PARESE: Forged, not cast. And the service line is made of stainless steel, as well.

MEMBER SIEBER: Joining welds, are they nickle welds?

MR. PARESE: I don't know the answer to that right now. I'm sure we said something in the SAR, but there's debate between using an I-52 type weld, or using stainless material to weld them together, so that's a good question. I don't think I know the answer to that.

The use of forgings does reduce the number of welds that we have to inspect, obviously, and that's pretty standard. And the heavy components are SA-508, and we use stainless 308 and 309. That's all pretty standard use.

MEMBER SIEBER: Okay. Thank you.

MR. PARESE: Slide 21 just shows a comparison of some of the data to an existing four unit. And the main thing to point out is an increase in reactor coolant volume will increase power level, and there's a significant increase in steam generator secondary -- even on a per megawatt basis, so we extended how much the heating level -- and significantly larger pressure at volume; 2650 cubic feet. Again, on a per megawatt or on a volume basis, it's significantly larger, and that slows down the transient response. And then the operating pressure to this unit is 1109 psi at the exit of the steam generator nozzle, and that raises the efficiency of the unit.

MEMBER SIEBER: 33 percent.

MR. PARESE: From 33 up to 35.

MEMBER SIEBER: Oh, it does?

MR. PARESE: Yes.

MEMBER SIEBER: Somewhere in your list it's 33.

MR. PARESE: This unit has a efficiency of 35 percent.

MEMBER SIEBER: Okay. Got it.

MR. PARESE: And what allows us to do that is we've raised the design pressure of the steam generator to 1450 psi. So what that means is from 1150 or 1250, and that allows us to for certain transients absorb a lot more energy as you get closer to the design pressure, and that then energy is a large thermal inertia, so -

MEMBER SIEBER: And you do that by raising T-h to 618 or more.

MR. PARESE: To get the 1109 psi, we have T-hot of 624, and we can do that because we've gotten it out of the unit. And also, the steam generators use an economizer design which is another extra 40 pounds.

(Off the record comment.)

MR. PARESE: And the advantage of our component designs and our steam generator designs is that these steam generators are very similar to the N4 steam generators already operating.

MEMBER ARMIJO: Same temperatures and pressures, though?

MR. PARESE: The N4 runs at 622-1/2, we're running at 624, and they operate at right around 1090 psi. The N4 units have a pretty good output. They're 4250 thermal, and I think they're 1490 or 1480 electric. And they also use 14 foot cords.

MEMBER SIEBER: So 628 that precludes nickel-based alloys -

MEMBER POWERS: Mr. Parese, you're lagging seriously here, so -

MR. PARESE: Okay. We're going to move.

MEMBER POWERS: Keep trucking.

MR. PARESE: The core design, you can see the evolution of the design from typical four-loop unit to the N4s in France and the USEPR. We use 241 fuel assemblies, 17X17, and our active link is 13.78 feet. The reason it's 13.78 instead of 14 is that gives us a little more annulus area to handle it. And we have 265 pins per assembly -

MEMBER BLEY: I'm sorry. Would you say that last one again? I didn't get that. The reason you're at 13.78 -

MR. PARESE: A standard design -- well, I should have prefaced that, the N4s and the P4s in France are 14 foot active stacks, and we're 13.78, so that .22 gives us more area in the annulus above the active stack to absorb --

MEMBER ARMIJO: It's kind of the other way around, isn't it?

MR. PARESE: I'm sorry?

MEMBER ARMIJO: You have plenum volume if you have a shorter fuel -

MR. PARESE: The total overall height, I'm talking about the active fuel stack.

MEMBER ARMIJO: Oh, this is a fuel -

(Simultaneous speech.)

MEMBER ARMIJO: Okay.

MEMBER BLEY: And that difference does what for you?

MR. PARESE: Well, it allows us to build a higher -

(Simultaneous speech.)

MR. PARESE: It's substantial enough to give us the margin we want, which also one of the margin improvements that was in the average linear heat rate for this design. We went up to 4590 megawatts, but if I have 241 assemblies, we've decreased the average rate, and we've increased the cubic feet so that gives us some additional margin.

MEMBER ARMIJO: Just a quick question. Is your vessel diameter pretty much standard, or you've got more fuel in there, larger diameter vessel?

MR. PARESE: This is a larger diameter vessel.

MEMBER ARMIJO: More than the N4s?

MR. PARESE: Yes, I believe it is.

MEMBER ARMIJO: So that is a step.

MR. PARESE: It's a step, but we don't think that's a dramatic step.

MEMBER SIEBER: Sixteen, 18 inches wide.

VICE CHAIR BONACA: The 205 and 241 are really standard designs, are they?

MR. PARESE: I guess you would know.

VICE CHAIR BONACA: Yes, I used to work for them, and 241 was -

MR. PARESE: I remember seeing your name on a lot of stuff. So yes, those are pretty -

VICE CHAIR BONACA: 241, I mean, was there in 1973.

MEMBER POWERS: This is what you'd call proven technology.

MEMBER ARMIJO: You have a big reflector.

MR. PARESE: A heavy reflector.

MEMBER ARMIJO: In-between the core and the vessel.

MR. PARESE: It basically replaces the baffle and former plates on current designs, and we get rid of all those bolts that can crack from radiation, and it prevent baffle jetting because there's no way water can get through there. And it reduces the fluence on the vessel.

For the EPR, we're going to capitalize on the digital I&C operating experience in Europe, the N4s that have digital controls.

(Off the record comments.)

MR. PARESE: And so the digital I&C architecture that we're using, we're using two systems. We're using the Teleperm XS system for the safety I&C protection system, and ESF functions, and we're suing the Teleperm T200 system for the distributed control system. So, generally, we would have the operators operate the plant from the process information and control system, what we call the PICS, and that would be his main interface. But, if for some reason, that interface isn't available, he can go to his qualified display system and actuate safety functions from the other system, safety information control system.

And the one thing I wanted to say about that slide is that our safety functions, like protection system and ESI are 4X100, so each division is processing the protection system signals. All right. So the safety system are 4X100. That also means that if each division is comparing for pressure signal say from the pressurizer and doing two by four, each division is doing two by four, so that's an increase in redundancies.

The distributed control system is 2X100, so we get our redundancy there and better diversity, so we get better reliability that way. Except for what we talked about, diverse automatic system, it's really not a system. It's a collection of functions to mitigate ATWS. We put certain reactor trip functions and other ESF functions on the T3000, so that if we take an ATWS failure, we have a diverse method of actuating it on a diverse platform. So that is our mitigation for ATWS. And we've increased protection and automation on the unit, so we have a hot channel DNVR trip. We have a high linear power density trip. Those trips are -

MEMBER BANERJEE: Well, how do you -- DNVR?

MR. PARESE: I'm sorry?

MEMBER BANERJEE: What do you trip on, power?

MR. PARESE: On DNVR, we actually measure the power in the floor and the flow rate, and the pressures and temperatures, and we calculate the DNVR, and we approach the trip set point, we trip the reactor.

MEMBER BANERJEE: Is the reactor DNVR, or large break LOCA limits within power? Appendix K?

MR. PARESE: I don't think it's -- our realistic LOCA output right now is predicting a temperature of 1425 for the peak UO2 pin, and 1513 for the peak gad pin, so right now I don't think we're LOCA-limited. I think we've been DNVR limited.

MEMBER SIEBER: Have you done an Appendix K-type calculation?

MR. PARESE: No.

MEMBER BANERJEE: So this is what, a best estimate?

MEMBER SIEBER: Yes, that's about right for -- that would be about 2,000 degrees on Appendix K. There's some margin there.

MEMBER BANERJEE: Oh, it just tripped itself. Okay.

MR. PARESE: So we've implemented those trip functions using self-powered neutron detectors in the floor and protection system. We put in a high steam generator pressure trip, so if we get an upset that exceeds certain pressure and we trip the reactor, that helps us with pressure mitigation. And we've included other systems, like computer-controlled heat-up and cool-down.

MEMBER BANERJEE: So this protection, I mean, the -- since we've lost that, what you call it, the protection system SG depressurization, this is how you get your low pressure, I mean, your medium pressure in.

MR. PARESE: Yes. And we need to get there.

MS. SLOAN: Yes. There's a section that talks about getting to that, and how that's applied.

MEMBER BANERJEE: What is -- you're going to tell us what this protection system SG depressurization is?

MR. PARESE: Yes. Core monitoring, we use fixed and movable system.

MEMBER: We have books, so maybe you can -

(Off the record comments.)

MEMBER POWERS: There is not a requirement that we have a transcriber, so would you please go ahead.

MR. PARESE: Okay. We're using self-powered neutron detectors to continuously monitor the core. They're cobalt-based so that makes them fast responding, but we calibrate those SP&Ds every 15 days approximately by using a moveable system called Aeroball Measurement System. It's extensively used in Germany. It's very reliable, and it gives us 3-D power map. It does each quadrant in 15 minutes, and so it gives us a full-core quadrant map, a full-core map in an hour, about an hour. And you do that every

15 days.

MEMBER SIEBER: You do not use any external NIs. Right?

MR. PARESE: No, we have external NIs.

MEMBER SIEBER: That's your high power trip?

MR. PARESE: Well, coupled with we also have a power trip on primary heat, calometric.

MEMBER SIEBER: I'm surprised you don't use the self-powered neutron detectors.

MR. PARESE: Well, they're used for high linear power density and for --

CHAIR SHACK: Better let him go on. He's got a number of important features to get to.

MR. PARESE: The reason we wanted to point it out is that it's not new. It's used a lot in Germany for decades, but it's new to people in the United States.

Slide 28 shows the locations where those Aeroball probes go into the fuel assembly into one of the thimble tubes, and we have about 40 locations. And that just shows how they work. Vanadium balls get irradiated and then they're sent by high helium gas off to a counting table, and then it counts them.

For severe accident mitigation features, to prevent high pressure melt-through scenarios, we've installed primary depressurization valves on top of the pressurizer. There's two of them. They're in parallel, not in series. And each one is about 1.9 million pounds per hour, so it can depressurize the plant from full pressure to less than 200 psi in about 20 minutes. Okay? So if core exit temperatures exceed 1200 degrees Fahrenheit, the EOPs will have them open those valves and drive them below pressure.

MEMBER SIEBER: Are they squib valves?

MR. PARESE: No. These are power operated valves.

MEMBER SIEBER: Okay.

MEMBER ARMIJO: Are they qualified for steam, water, and two-phase flow?

MR. PARESE: Yes, but they are not safety-related valves, so they're not seismically qualified. They're qualified to two over one. In other words, if I have a seismic event, I can't have these valves affect my safety valves.

MEMBER SIEBER: Right.

MEMBER MAYNARD: They have block valves in it?

MR. PARESE: Yes.

MEMBER MAYNARD: Are the block valves

safety valves, or safety-related?

MR. PARESE: No. These valves are normally left closed, and there's no automatic function that opens them. They're manual. And it's based solely on core-exit thermal couple temperatures.

MEMBER ABDEL-KHALIK: Isn't 1200 a bit too far? It's way beyond the thermodynamic critical temperature.

MR. PARESE: I don't know how to answer that. We don't think it's too far, because we think if we actuate by the time we get 1200 degrees, then we prevent any other downstream failures, for example, temperatures on the tubing that could cause a failure of the tubing, or failure of the pressure boundary.

MEMBER SIEBER: You're in severe accident space anyway. Right?

MR. PARESE: Right. But the way we do it is, you would enter -- you would open the valves and depressurize, and you could have accumulators or LHSI quench the core. Then you don't enter your SAMGs. But if you continue with high temperatures, then you'd enter SAMGs at that point, and then we would preclude safety injection to avoid a vapor export.

Then the method we used for stabilizing the melt and cooling is ex-vessel stabilization, so we have a reactor pit that will collect any melt from the vessel, and then allow it to distribute to a spreading area where we will cool it passively for at least 12 hours or longer, and then actively after that point. And then we control the hydrogen concentration inside containment by using passive autocatalytic recombiners. We have 47 of those distributed around the containment.

So most notably, this reactor pit area is always kept dry. We haven't talked about the heavy reflector, but the heavy reflector will control how the material collects, and it will have to melt first. And then it will collect in the lower head, and then as your oxidic and your metallic melt separate out, you get different heat transfer capacities, and you could get different melt scenarios, like through the side of the vessel in a partial core, or you could then get heating from above and below, and get a catastrophic failure of the head. Those uncertainties are handled by having a special concrete inside here that ablates and mixes with the material while holding it, and lowers the viscosity of the material.

MEMBER CORRADINI: Are you allowed to say what that is?

MR. PARESE: It's concrete, and I don't

know if -

MEMBER POWERS: I'm going to be fascinated to find a concrete that will lower the viscosity of core debris.

MEMBER CORRADINI: Of what?

MEMBER POWERS: Core debris.

MEMBER CORRADINI: I think you meant to say lower the solid's temperature, I assume you meant to say.

MR. PARESE: Yes.

MEMBER CORRADINI: Okay.

MEMBER POWERS: But it's not going -- all that's going to do is raise the viscosity.

MR. PARESE: Yes, it will separate the liquidous and solidus temperature.

MEMBER POWERS: That is -

MR. PARESE: Also, this is lined with -- the plutonium elements are behind the concrete, except for this melt plug here which has concrete, and then it has a steel and aluminum, so this is the failure point of the system.

MEMBER CORRADINI: So it's designed to basically cook -- a special cooking mechanism which then releases in force?

MR. PARESE: That's exactly it. And we

make this the weak link so that this will fail first, and then we'll get a pour and a spread into the spreading area. The large spreading area, then once it spreads into the spreading area, we activate passive valves. That's thermally actuated valves, that's another way of saying it, spring-loaded valves with chains, the chains melt and the spring-loaded valve will -- so there's nothing fancy about that. And what it does is, it allows water from the IRWST to flow underneath the spreading area, which cools it from the bottom. And then up over the top of the weir and on top, and cools it on top. The flow rate is restricted, so that we don't generate too much steam -

MEMBER BANERJEE: How does the water flow underneath there?

MR. PARESE: Well -

MEMBER BANERJEE: It's not clear to me.

MEMBER CORRADINI: It's just a European -

MEMBER POWERS: We will have an opportunity to explore this in enormous detail.

MR. PARESE: There's a line - those valves a line that allows water to go under the cooling channel, and these have cast iron plates with cooling channels, and the water runs underneath. And the

IRWST level is above the spillover on the WIR, so that promotes the flow.

MEMBER CORRADINI: This is, I guess, a small question, if the Chair will allow me. What is the elevation of the bottom of the IRWST? It shows here that it's below the vessel bottom. Is that correct?

MR. PARESE: Yes.

MEMBER CORRADINI: Okay. All right.

MEMBER POWERS: We've got not passage stuff in there, we've got pumps.

MEMBER CORRADINI: I just wanted to know the elevation. I was just curious.

MR. PARESE: We can passively cool, the steam will go up in the containment, condense in the methods we talked about for the loss of coolant accident. The condensation will go back into the IRWST, and at least for 12 hours, we can do that without exceeding the containment design pressure.

MEMBER BANERJEE: What does IRWST stand for?

MR. PARESE: In Containment Refueling Water Storage -

MEMBER POWERS: We should tell him in French what it stands for.

MR. PARESE: All right. So at 12 hours, the operator is credited to turn on the severe accident heat removal system, which can then use those non-safety-related sprays to depressurize the containment system. And at any time after that, he can also switch to active cooling of the melt, and that active cooling then will pump up that cabin and fill up the vessel, and to chimney up to the top, so now you have active flow and cool.

MEMBER BANERJEE: Now we don't have to look at -

MEMBER SIEBER: Not today.

MEMBER CORRADINI: Dr. Powers will.

MEMBER BANERJEE: Dr. Powers, are you going to have to look at this in detail in the future?

MEMBER POWERS: Exhaustive.

MEMBER ABDEL-KHALIK: You indicated that you are using 35 percent enriched boron.

MR. PARESE: Yes.

MEMBER ABDEL-KHALIK: What sets the isotopic enrichment that you need?

MEMBER POWERS: Water solubility.

MEMBER ABDEL-KHALIK: Isotopic.

MEMBER POWERS: Yes, water solubility, more than anything else.

MR. PARESE: Yes. The solubility limit is important for our post-LOCA mitigation, and that affects the time at which you must turn on hot leg injection. I don't know if you noticed in the pictures that showed the injections, we can open valves to inject into the hot leg, and that's our primary method to prevent boron played out in the vessel, and exceeding the solubility limit.

The other issue is if you saw -- this unit operates at 624 degrees, even at a lower kilowatt-per-foot,. we have to always be wary of crud-induced power shift, and so having the enriched boron allows us to have a critical boron concentration of only 1400 ppm for an 18-month cycle.

MEMBER POWERS: And again, this is an area that you want to pay very close attention to because boron shifting in these kinds of high power reactors are going to be an issue.

MR. PARESE: And so the other thing is once you decide you're going to go to enriched boron, you make sure that's what you use everywhere, and you don't allow anything else on the site.

MEMBER ABDEL-KHALIK: But are you sure enriched boron is going to help you with axial offset anomaly?

MR. PARESE: Critical boron concentration is extremely important, as well as the -

MEMBER BANERJEE: What was the reason for making the core longer?

MR. PARESE: Fourteen feet, to get the power out of it.

MEMBER SIEBER: I guess you made it bigger, so you had to make it longer.

MR. PARESE: We made it longer -- if you want to get -- the original design of the EPR was to handle 4900 megawatts, and if you're going to do that, you either have to have a much wider -- a bigger diameter core, or a taller core, or -

MEMBER BANERJEE: So it's a foot and something longer than the current full rate operated design.

MR. PARESE: Yes. But it's the same basic fuel that's operating in the French units in the P4s and the N4s for decades. Areva has a lot of 14-foot experience. We need to get on to your main topic, is SGTR mitigation and small break LOCA mitigation. This is your depressurization.

For SGTR mitigation, medium head injection pumps were purposely selected. The view from the utilities that were helping design the unit and from

Areva at the time was that the event that is most likely to cause radiation release to the environment and to the public was a steam generator tube rupture. And even though we've improved the materials, we use Alloy 690 in our steam generators, the German generators have Alloy 800. You could have a loose card or something else. You can't say what could cause damage in a steam generator. It's not just stress corrosion cracking.

Consequently, the way to keep the iodine in the plant is not to vent liquid that contains that iodine outside the plant. So the medium head safety injection pumps were perfectly selected so that even if they went to their dead head, a shutoff head is below the main steam safety valve set point on the steam generators.

MEMBER BANERJEE: But this is the German -- the Siemans, from what I -

MR. PARESE: I would agree that that was originally the driving philosophy, but I think it was embraced entirely by the whole design team. Now you're getting into other issues between French and Germans, and French and German regulators, and we don't need to talk about that today. But the point is, consensus was reached, to keep from venting liquid that contains iodine outside the steam generators. And regardless of your regulatory assumptions on Partition factors, the reality is from a physical point of view, most of the iodine is going to stay in the liquid phase, so keep that liquid phase inside the plant. So that insures there's no challenges to your safety valve in the affected steam -- there's no operator action required.

(Announcements.)

(Off the record comments.)

MEMBER SIEBER: Do you want us to continue?

MEMBER POWERS: Please.

MR. PARESE: All right. So we meet our dose consequences from a regulatory standpoint -

MEMBER POWERS: Can we please close the door?

MR. PARESE: And also from a design standpoint, we meet those goals by minimizing bypass. So now that gives you the interesting problem that you jumped on right away at the beginning, was for very small loss of coolant accidents, the energy discharged through the break isn't sufficient to remove all the energy. You have to dump some of the energy to the generators, so for small breaks you're coupled to the steam generator pressure. Primary pressure couples to the steam generator pressure. If those steam generator pressures are above the dead head on the MHSI, then for those smaller breaks you will not get any significant MHSI flow until you can completely drain the loops and open the loops seals, and get steam to the break. And now you're in a race for depressurization versus water coming in.

MEMBER ABDEL-KHALIK: What's your T-ave?

MR. PARESE: Our T-ave is 594 degrees Fahrenheit.

MEMBER ABDEL-KHALIK: And what is the saturation pressure at T-ave?

MR. PARESE: I don't have my steam table with me.

MEMBER ABDEL-KHALIK: Is it greater than or lower than the shutoff head of your SI pumps?

MR. PARESE: It's greater than -- let's see. The shutoff head is 1400 - I don't have my steam table with me. Anybody have a steam table? I don't know.

MEMBER ABDEL-KHALIK: That's okay. Continue, please.

MEMBER BANERJEE: So you use the can-do method, basically. That's what they've been doing for years.

MR. PARESE: And it turns out that every EOP ever written for mitigating small break tells the operator to depressurize the steam generators and keep them below the saturation temperature of the primary. In other words, keep them at heat sink

MEMBER BANERJEE: The only thing that it requires then is that you have sufficient flow area that you don't get flooding during reflux condensation. Because if you do, then you don't get any steam in.

MR. PARESE: That would be true.

MEMBER CORRADINI: Where is the reflux condensation coming from, Sanjoy? I don't think I understand.

MEMBER BANERJEE: Because they have to pull the heat out of the steam generators. Therefore, if you get water condensing, it runs back counter-current to the steam flow going. And, therefore, there's a chance of flooding at this tube sheet.

MEMBER CORRADINI: Oh, in the tube sheet.

MEMBER BANERJEE: Just at the entrance.

MEMBER ABDEL-KHALIK: Did you say your primary TM at full power is 590?

MR. PARESE: 594.

MEMBER ABDEL-KHALIK: 594. Saturation pressure is roughly 1475, so if you have a small break on the high end of it, and the primary saturates, your SI pumps will be dead headed.

MR. PARESE: If it happened that way, but it doesn't happen that way, because you get a core shutdown which reduces the heat production in the core. Zero power temperature is more indicative of where you'd go once you've dumped the sensible heat to the steam generators, and that's 577 Fahrenheit.

So we're down to five minutes, so let's punch through this. So the plant has a safety-related function that's driven by the protection system that depressurizes the steam generators, and that signal is a low-low pressurizer pressure signal which starts the safety injection system. So we start this depressurizer when there's still water in the steam generators. We depressurize the steam generators at 180 F per hour, 100 C per hour, to about 870 psi, where then the valves control to that set point.

MEMBER BANERJEE: Just blowing steam.

MR. PARESE: Blowing steam. We're blowing down the steam generators, we're feeding with emergency feedwater, and we're just depressurizing the steam generators. And then at 870 psi, we hold the pressure constant. So what we do is we lower that pressure so that for those breaks that couple to the steam generator, they couple to a lower pressure, and we insure then that we have adequate MHSI flow.

MEMBER ABDEL-KHALIK: What are you blowing down the steam generators with, atmospheric dumps?

MR. PARESE: The main steam relief train that we discussed, which is safety-related, seismically qualified. It's 50 percent steam flow. We're using that system, and it's got redundant power supplies. It's actuated by the protection system. So we've developed a safety-related depressurization system. We're looking at some power uprates for some units in the U.S., and putting the same kind of safety-related system on to get this credit.

MEMBER BANERJEE: Is this plant peak-clad temperature? Is this occurring for the largest LB-LOCA or is it shifted to a smaller break?

MR. PARESE: It's shifted to a small break. If you looked in our FSAR, our peak clad temperature is for a 6-1/2 inch break, and it's -- it's in the FSAR.

MEMBER BANERJEE: But your FSAR is in now. Right?

MR. PARESE: Yes.

MEMBER BANERJEE: So we can take a look at it.

MR. PARESE: You can see it in Chapter 15. There's a table, and the 6-1/2 inch break. There's actually a plot of PCT versus break size for the small breaks.

MEMBER BANERJEE: Okay. Thank you.

MR. PARESE: So, in fact, what this does is for one and two inch breaks, there's no core uncovery, and for three and four inch breaks, and five inch breaks that require loop seal clearing, anyway, this helps a little, but -

MEMBER BANERJEE: Are your steam generators fairly large, is there a large flow area?

MR. PARESE: Yes.

MEMBER BANERJEE: Then I'm much less worried.

MEMBER POWERS: In exhaustive detail.

MEMBER BANERJEE: No, it's a question of whether that has enough flow area during the condensation part.

MR. PARESE: Well, and a big part of the depressurization -- for the breaks that matter, which are the smaller breaks like two inches, and three inches, the depressurization is occurring early in the vent before you even get much boiling, so they're ready to go when you produce some vapor.

MEMBER BANERJEE: Okay.

MR. PARESE: All right. I'm going to zoom, I know that people are interested in the PRA. It's in Chapter 19. I'm going to zoom through this, because we're almost out of time. What I will say is our design target was to a core melt frequency from all plant states and initiators to be less than 10 to the minus 5. We wanted the at-power states to be less than 10 to the minus 6, and the shutdown states to be less than the power states, and so when we went through the PRA, our core damage frequency from at-power and shutdown events is less than 5.8 times 10 to the minus 7, so that's well below our design goal.

MEMBER STETKAR: Does that include any contribution from seismic events?

MR. PARESE: No.

MEMBER STETKAR: Okay. Thank you.

MR. PARESE: So, Todd, we're back to seismic margins again. I'm going to leave the slides on operating experience for you to take with you, because we're really out of time, so you can see that the built-in -- the operating experience on the existing units to help with materials, event materials, to reduce the degradation from materials, to ease the outages, make the outages faster and easier to do. I think everyone is going to tell a story.

MEMBER APOSTOLAKIS: Did you say you could do a margins analysis -

MR. PARESE: No, I don't believe we have. We did?

MEMBER APOSTOLAKIS: What did you say?

MR. PARESE: All right. I need to ask Todd Oswald to step up to the microphone then.

MEMBER APOSTOLAKIS: Oh.

MR. OSWALD: Yes. This is Todd Oswald, the Manager of the Civil Structural Group. Actually, we did do seismic margins assessment to demonstrate the 1.67 heat capacities.

MEMBER STETKAR: What's the SSE for this plant?

MR. OSWALD: 0.3g is the -

MEMBER STETKAR: .3.

MR. OSWALD: Is the PGA.

MEMBER STETKAR: .3g.

MR. OSWALD: That's correct.

MEMBER APOSTOLAKIS: Then it's very hard to demonstrate that you met your target, isn't it?

The targets on slide 30 something, 35. Anyway, we'll come to this at some other time. And the fire is also a margins kind of analysis, like the EPRI fire thing? Although, in your case it's probably very low because of the preventive separation.

MR. PARESE: Yes, but the number is so low that fire still has a contribution. It's like an operating unit, instead of fire events being 30 some percent of 5E to the minus 5, or maybe a similar fraction of 6E to minus 7, so we drastically reduced the -

MEMBER APOSTOLAKIS: It has been submitted already?

MR. PARESE: Yes. Chapter 19 is there. In fact, this slide -

MEMBER POWERS: It's the orange on his segment there. Fire is the orange.

MR. PARESE: And that's a whole range of different fire events, fire in the control room, fire in the switch gear, fire in the different safety buildings, so there's -- it all in Chapter 19. And one of our safety goals is to reduce the occupational dose, and our design goal is to put features in the plant to reduce the dose to less than 50 person-rem per year. And we've had 50 utilities estimate that number based on their activities, and they think they'll be seeing a number average including refueling outages, a two-year running average of around 38. But that proof isn't in the pudding, it's in the -- so in the time we had, we didn't get to answer all the questions, but -

MEMBER POWERS: Oh, you'll get the opportunity.

MR. PARESE: I'm sure we will. But EPR is an evolutionary design. The features that you saw are very much like features you've seen. We took the maximum benefit from the operating experience, and R&D of the existing units, and so most of the features are typical PWRs. And, as we've discussed, we included features to improve safety, enhance reliability, and protect critical systems from external events, which were some of the major design goals of the unit at the very beginning. And with that, you've gotten the rapid fire overview of the EPR.

MEMBER POWERS: That's what we asked for.

MEMBER ABDEL-KHALIK: The 1400 ppm boron that you mentioned earlier, what is that value exactly? Is that the -

MR. PARESE: The number I mentioned, that's the range of the initial critical boron concentration in the reactor coolant at the beginning of a cycle.

MEMBER ABDEL-KHALIK: So this is the critical boron concentration at the beginning of cycle.

MR. PARESE: Right. And since the boron concentration goes down with burn-up, it's that initial critical boron that can lead to boron plate-out if you have a high --

MEMBER ABDEL-KHALIK: So you need 1400 ppm with 35 percent enrichment in boron-10 to do this job.

MR. PARESE: Correct. Otherwise, your critical boron concentration will be over 2000.

MEMBER POWERS: The portion of the material you did not ever suggest, your materials from metallurgy, the Subcommittee will have to contribute, as well, here. So you're going to carry a big load again.

What can I say except thank you. That was good. We asked for a whirlwind, we got a whirlwind. We asked for a schedule, we got a schedule. You're putting all together too much on us, we'll be all very grouchy next time, and probably interrogate you must more closely on all these things, but I appreciate it very much. If the members have any other questions on this, now I have taken notes on where the questions were asked, and I will be assigning each one of you to report on what you asked about here in detail for the August Subcommittee meeting we'll schedule.

(Off the record comments.)

MEMBER POWERS: Other than that, thank you very much. We'll turn it back to you, Mr. Chairman.

CHAIR SHACK: Okay. We will recess for 15 minutes.

(Whereupon, the proceedings went off the record at 3:48 p.m., and went back on the record at 4:06 p.m.)

CHAIR SHACK: Time to come back into session. Our next topic is essentially a briefing in the safeguard and security area, and Mario will lead us through that.

BRIEFING ON SAFEGUARDS AND SECURITY

VICE CHAIR BONACA: Yes, good afternoon. And thank you for coming.

For the information of the committee, there are many activities or developments of the rules and regulations under the security rulemaking. And so they are all coming together pretty much in the month of July.

There are four rules as far as I

understand, the security interface, cyber security, the contingency mitigative measures, large fires, and explosions rule, and the aircraft impact rulemaking.

In addition to those there are a list of, I could see two reg guides, one cyber security, and the other one the safety-security interface.

And of course then there is NEI 404, then also is the reference, I believe that's 5.2, cyber security.

Now what is happening is that I believe the Commission is expecting all these rules to be completed by the month of July. And we are in a squeeze because, if I understand it, all these rules will not be ready in final form until the end of the month, and they are supposed to write a letter in July.

So we are in a squeeze that -

VOICE: Are we supposed to be here on the 4th of July?

VICE CHAIR BONACA: No, what happened is that I invited this gentleman, Andrew Pahlevi, to come and tell us about their plan, and when we can expect to see material to review and see how we can work around it and see if we can support them.

MEMBER APOSTOLAKIS: We have a subcommittee meeting.

VICE CHAIR BONACA: The committee meeting will take place in the July meeting. So it will be probably the first day of the meeting in July. We have no materials to go there.

MEMBER CORRADINI: But you're not going to have anything that earlier part of the week on the 7th or the 8th.

MEMBER APOSTOLAKIS: On the 8th is already another -

(Simultaneous voices)

VICE CHAIR BONACA: We are going to hear now when they believe that they can deliver to us some information so we can review, clearly, we are looking typically for finalized documents, because we don't want to comment on documents which are still in flux.

So we will hear about that. And I wanted to make this introduction, because at the end of this presentation we should spend a few minutes to do some planning.

First of all, determine what can be done, and second, within that, see how we can do it.

So with that I'll turn it over to you, and we'll have the presentation.

MS. BANERJEE: This is Maitri Banerjee.

Can I add something to answer George's question on the subcommittee meeting. We did plan a subcommittee meeting with the committee, but because of the compressed schedule that the staff was under it was very difficult for them to support it.

VICE CHAIR BONACA: But furthermore on security and safeguards, we don't have a subcommittee; we have a full committee. The whole committee is being - because we never - we'd do well to redouble the efforts.

Anyway that's where we are.

MS. SCHNETZLER: Good afternoon. My name is Bonnie Schnetzler. I work for the office or NSIR, and I'm the project manager for the security rulemaking for nuclear power plants.

Today I'd like to talk to you a little bit about the status of the security rulemaking. We came here last year about this time and kind of gave you a brief of what we were doing, and the complexity and large pieces of rulemaking that we had, and then focus on the parts of this rulemaking that will need ACRS review.

And then give you a status of the regulatory guidance that supports the regulation that we have in the proposal and now in final draft.

Next, please.

MEMBER APOSTOLAKIS: Well, just to go back.

MS. SCHNETZLER: Go back.

MEMBER APOSTOLAKIS: Which one are you using now?

MS. SCHNETZLER: Right, the - in actuality the part that was in appendix charley which you spoke of, sir, was rolled into - and moved to 50.54(hh), which is the imminent attack and mitigative measures.

MS. HOLOHAN: He did mention the aircraft impact rules, which is separate, that's a separate track. It's not part of this.

MEMBER APOSTOLAKIS: Yes, but the aircraft rule is going to come to us in July too.

MS. HOLOHAN: Yes, but it's not going to be part of this.

MEMBER APOSTOLAKIS: I understand. It's got to be on our table for review anyway.

MS. SCHNETZLER: This is - following is a list of the rulemaking that we are currently engaged in, 50.54(hh), mitigative strategies and response procedures for potential or actual aircraft attacks.

73.54, protection of digital computer communication systems and networks.

55, which is physical security for power reactors.

56, which is personnel access authorization requirements.

MEMBER APOSTOLAKIS: Is the fact that one of them is under five-fifty and the other is five-seventy-two, does it make any difference in real life?

MS. HOLOHAN: No, it was originally part of Part 73, and one of the comments we got was, it would be better served to be in Part 50, so we moved it into Part 50. But it's going to be part of the final rulemaking.

MEMBER APOSTOLAKIS: Why is it better served?

MR. MORRIS: Because, if I could take it, Part 73 is what you have to do to respond to, within design basis, threat attacks. And everything in 50.54(h) is outside of design basis threat.

That's the short answer. The long answer is a lot more complicated.

MR. REED: Dr. Apostolakis, in addition to that, 50.54 also would place it in as a license condition on the licensee. So it goes over to Part 50. These are broad actions. They are operator actions. Emergency preparedness and fire protection, okay. They are much broader than security force; they involve security. So in that context is makes more sense to go to Part 50 and it works in licensing space better two.

MR. MORRIS: Most of the things that need to be done in response to aircraft attacks and mitigation strategies are all - they are not generally done by the security organization. They are done by operators, emergency responders, things like that. That's the other big reason on this.

Thanks.

MEMBER CORRADINI: A question just for my edification. I understand what you said. So that separates us, so that's in the 50.54 side.

MR. MORRIS: It's analogous to how it's been treated with the operating reactors right now. The mitigation measures piece is really interim compensatory measure b-5-b actions, which have all been handled as a condition of the license, the operating license.

So we are just mimicking that in the rule.

VICE CHAIR BONACA: Now the question I have is, if I go back to the previous slides I see 50.54(hh), I see 73, for cyber security. Now there are two more actors here. Could you go through the next slide?

One is 73.55, physical security - we have not reviewed that, have we?

MS. HOLOHAN: That is correct. I just want to give you the pack that we're working on.

VICE CHAIR BONACA: And then 356 also we don't review?

MS. HOLOHAN: That's correct.

MEMBER APOSTOLAKIS: How was this decided?

VICE CHAIR BONACA: That was decided a long time ago because really each of those persons et cetera from which review were excluded from participation. So I wanted to keep track as we move through.

MS. HOLOHAN: Right, and I'll narrow it down as we go along.

MS. BANERJEE: This is Maitri again. I believe there is a commission SRM that sort of directs ACRS to stay outside of physical security.

VICE CHAIR BONACA: Yes, so those areas -

MS. SCHNETZLER: And this follows that SRM.

The next parts of the rulemaking, 75.38, safety-security interface requirements, Appendix B, which is training and qualifications for security personnel, in Appendix C, which is safeguards contingency plans.

VICE CHAIR BONACA: But now again, here, this involves most of the first part, which is set the security interface.

MS. SCHNETZLER: Correct.

So where we're at right now as of today is that we are in the stages of the development of the FRN. That is being put together, being reviewed by OGC and other offices before we place it into formal concurrence which we plan to do on 6/16 of this month.

Our goal is to have it to the EDO on 6/30, so we're moving along very quickly.

VICE CHAIR BONACA: What is FRN?

MS. SCHNETZLER: Federal Register Notice.

VICE CHAIR BONACA: Federal Register Notice. So it would not be however complete or approved until 6/30?

MS. SCHNETZLER: That's correct.

MR. MORRIS: The plan is to deliver it to the executive director by the end of this month by which time the EDO's office will have an opportunity to provide their input. Ultimately the commission and the OMB and - so we're projecting that probably if all goes well probably the early part of 2009 the rule would be effective.

MS. HOLOHAN: EDO has told us they want to move it to the commission as quickly as possible on June 30th.

MEMBER APOSTOLAKIS: What is it that would be introduced?

VICE CHAIR BONACA: Well, they will deliver to the EDO the part of the rule package.

MEMBER APOSTOLAKIS: And then what happens?

VICE CHAIR BONACA: Aircraft impact rule.

MS. HOLOHAN: No, we don't have anything to do with that.

MR. MORRIS: NRO has - and I think NRO has the lead on that. There she is.

MS. GILLES: This is Nanette Gilles from the Office of New Reactors. The aircraft impact rule is on a separate schedule from the security rule. The aircraft impact rule has been provided to the ACRS, and we will be discussing that in the July full committee meeting.

And our schedule is to deliver that rule to the commission in September.

MEMBER APOSTOLAKIS: But again, the question is, the final rule you say will be submitted to the Commission in July? And then what happens? Because you said it's going to be in fact a year - so what happens during that year?

MS. HOLOHAN: No.

MEMBER APOSTOLAKIS: So what happens during the year?

MS. HOLOHAN: It won't be a full year later. When we get a Commission SRM, and Tim may deal with it, then we'll have to go through OMB clearance with the final rule package, and that takes 60 days.

MEMBER APOSTOLAKIS: Is there a period of public comment here at some point?

MS. HOLOHAN: No, we have already had public comment.

MR. REED: George, it's pretty much the standard rulemaking process at this point. In other words, the Commission has to deliberate. They are going to take some time. Then they issue a staff requirements memorandum. I'll give you an idea, the proposed rule had 300 items in it. It was substantial. It took many months for us to address that down. We have to address that; make those fixes; go back to SECY, okay, then start the OMB clock for 60 days. So what it is, it runs you all the way through the end of the year into the very beginning of next year if you start running the calendar time. And that's 30 days effective when you get into the Federal Register. It adds up; it's pretty amazing.

So we build in roughly about two months for the Commission in there, which is pretty aggressive. This thing is going to be a very very large package.

VICE CHAIR BONACA: The question I have is, what time does the ACRS have to comment on these rules?

MEMBER APOSTOLAKIS: July, right? That's what you are saying.

MR. MORRIS: Essentially.

MEMBER APOSTOLAKIS: Essentially means what.

VICE CHAIR BONACA: If I understand your comment in July, on giving us a presentation on this on the final documents, and you expect to have us turn around the letter immediately. We will have to discuss whether or not ACRS can do this.

MR. MORRIS: Our intent was to deliver the package to the EDO's office, and then nearly simultaneously provide that to the ACRS for their review, and knowing how big this package is, and how long it's likely to take the Commission to deliberate on it, it was our expectation and hope that the ACRS could complete whatever review they work that you all decided to conduct in parallel but preferably early on before the Commission gets too far down - this thing is on an incredibly fast track as you have sensed. And there are a number of reasons for that, and we can go into that if you'd like.

But the net result is that the staff was provided very little time to conduct the business that we would ordinarily conduct, particularly for a project of this scope.

MR. REED: I would also say, Dr. Apostolakis, is that in July certainly you can inform the Commission, and I think this committee can provide good input with regard to the requirements themselves, the new language requirements themselves, the implementation guidance will still be in draft form, and I think the committee can get involved with that through some period of time, because that has to be finalized, that's going to take much longer. And I don't know if we have any detailed schedules for that.

MR. MORRIS: Well, what I can say about that is, with respect to the cyber-security piece, we will be - in fact it just came back from publication today, the draft reg guide that supports the 73.54 rule, so we are going to put that out for a 45-day public comment period, have a meeting. That won't obviously be finalized for some time, and we'll have plenty of time to discuss some of the implications of that regulatory guidance.

Similarly, the 50.54(h) guidance will likely not be issued for stakeholder comment until the July timeframe.

And then what's the third piece? The third piece was the safety-security interface which has been out. We actually put a draft of that out for comment, and had a public meeting on it last August. Since that time the industry has indicated a desire to provide their own guidance, and let us comment on that.

That guidance from industry has not yet been forthcoming. So we are kind of at the point right now where we are almost ready to go back to what we started with.

MEMBER APOSTOLAKIS: So all these things, we have time to get involved with later. So what Tim is saying that we are going to review only the requirements of the rule in July?

MR. REED: Obviously I'm not going to direct the committee. I mean it's up to the committee. I'm just making a suggestion that in July you certainly will have sufficient information to make a judgment whether you think requirements in these three areas are adequate, and we'll probably be able to give you as much as we can in terms of implementation guidance at that time to start that review.

MR. MORRIS: And additionally where we are in terms of the language, and Bonnie is going to get to this, but the language of the rule that we are asking ACRS to take a look at is available right now, and in fact we are going to provide that to you.

VICE CHAIR BONACA: But it's available in not-Internet file form. Only documents in hands for the past few days, okay. And on a rule there is one page. On other rules, there are two pages. There is no support to information.

I spent a lot of time on cyber security guidance, 404, NEI-0404 in thinking that that would be the actual reg guide, and now I come here and I discover there is a reg guide that supercedes the NEIA guidance.

So everything is so in flux and ACRS does not typically review and comment on a document which is still in flux, because we may make a recommendation that is inappropriate, because the rule changes or the guidelines change.

So what I'm trying to do including for the aircraft attack rule, is to understand when we can expect to have something finalized enough so that even if we jump on it, we can at least start to review it.

The reason why I also mention the aircraft impact rule is because the version I have I think is articulate, et cetera. However it's not titled. The pages are out of order. The members are not there, et cetera. That's not final, what I've seen.

And so anyway, we can proceed now. But I wanted to make sure before we proceed that we first of all understand the pieces that are going to be presented to us, and the challenge we are having in providing you with any comments.

With that proceed.

MS. BANERJEE: This is Maitri again. I was wondering if the members may want to see the draft guides in whatever form they are together with reviewing the rulemaking, because otherwise reviewing the rulemaking under rule language is going to be kind of in a vacuum.

The regulatory guides might provide a little bit better perspective.

VICE CHAIR BONACA: Well, I'll tell you, in receiving the pieces that are being received for the record, it takes a long time to review. And then at the end, you compress that because things are changing.

So before anything else let's understand when can we expect to have some documents in a finalized form. Then we can talk about reviewing them.

MEMBER APOSTOLAKIS: But can we review these documents at home?

MS. HOLOHAN: Guidance are OUO-SRI, so -

MR. MORRIS: Safety-security interface is public, and the other two are OUO.

(Simultaneous voices)

MS. BANERJEE: As long as there is no SGI.

MR. MORRIS: No.

VICE CHAIR BONACA: But you want to receive, George, something that is final. Again, otherwise, you know you say that is the rule. So you are searching for the rule, and you find there's a page with four bullets, that's a rule. That's not a rule. It's a space that would be contained in the rule. And you don't want to spend your time on that.

So okay.

MS. SCHNETZLER: So this focuses us down to the pieces that we need ACRS review in our rulemaking package. And as we have discussed, it's 50.54(hh), mitigative strategies, and this is basically what everybody calls Brave Five Bravo. And it also includes imminent threat.

And the draft guidance for that has not been finalized. It is in production, and we're anticipating completing it in about a month.

MEMBER APOSTOLAKIS: So that's an interesting fact. Potential or actual?

VICE CHAIR BONACA: We could say that about everything we do.

MEMBER APOSTOLAKIS: Why did you think you can say that?

VICE CHAIR BONACA: I mean everything we do here is with potential.

Potentially it's really pre-warning the communication. Actually is - yes.

MS. SCHNETZLER: And we have technical people here that are ready to jump on this.

CHAIR SHACK: Lou.

(Whereupon at 4:28 p.m. the proceeding entered a Closed Session to return to open session at 4:32 p.m)

MEMBER ABDEL-KHALIK: The determination as to what is appropriate and what is inappropriate to answer or question will be determined by the staff?

MR. GILLESPIE: NSIR is the security experts for the industry, yes. If we have to go into closed session we can arrange that.

(Remarks off the record)

MR. GILLESPIE: By and large we keep everything open unless it needs to be closed.

MEMBER APOSTOLAKIS: Didn't we write a letter on the digital stuff? We reviewed something?

MR. MORRIS: What you reviewed was part of the digital INC steering committee effort in which they were - and still are - a number of subgroups looking at a variety of issues. And cyber security was one of them. But you all took a look at the interim staff guidance associated with cyber security for safety related system. This rulemaking goes beyond safety related.

MS. SCHNETZLER: So the second part of the rulemaking that we need ACRS review for is 73.54, protection of digital computer and communications systems.

We do have a draft guidance, and actually it's just being published today. The draft guide is OUO-safety related, and we can provide you copies of this. There are some control measures that need to be taken with it, but we can provide those for you so you can look at those and read it.

MEMBER APOSTOLAKIS: Are we going to have this in a binder?

MS. BANERJEE: I can put in a CD.

MEMBER APOSTOLAKIS: Maybe you can do that before we leave.

MS. BANERJEE: I will do that.

(Simultaneous voices)

MR. MORRIS: Somebody had mentioned NEI-404, which as you know is the industry's program to implement cyber security and nuclear power reactor sites, and that came up in the context of the digital IMC steering committee as well.

This draft reg guide recognizes and draws on a lot of what is already in NEI-404, but it takes it a step or two past that.

MEMBER APOSTOLAKIS: I guess I missed that. What you are saying is that we are reviewing both the rule and the corresponding guide.

MR. MORRIS: Correct.

MEMBER APOSTOLAKIS: I was wondering whether we would have too little to do.

(Laughter)

MS. SCHNETZLER: The last piece of regulation is 73.58, safety security interface. And there is guidance with this. It has been published, and it is public. It's not classified in any methodology. Not controlled.

Next, please.

VICE CHAIR BONACA: Going back to the NEI-0404.

MR. MORRIS: I'm sorry?

VICE CHAIR BONACA: Going back to NEI-0404, there was an extra tension it seemed to me when I read the NEI-0404, extra tension, and that would be really the reg guide in a way. Or the reg guide would be a very brief reference in the NEI-0404.

MR. MORRIS: Industry has, NEI in particular has indicated a desire for the NRC to formally endorse NEI-0404, the latest revision of NEI-0404, in our regulatory guidance document.

What I have said to them was, we will publish our own guide - because NEI-0404 is specifically for power reactors. 73.54 and this reg guide are not. It could be adopted by - what I said was when we open this up for public comment, which will be in the very near term obviously that we would be willing to accept that comment or request in that comment period, and we'll take it on then.

VICE CHAIR BONACA: Okay, I thought there was some conflict there. You are telling me there is complementarity.

MR. MORRIS: They are very complementary. It's just that our guidance document is generic, it doesn't specifically focus on power reactors. And it goes into not just the what but the how, how to. Not just what you have to do, but how to do it.

MEMBER ARMIJO: Would it apply to a fuel facility?

MR. MORRIS: It could. Whatever we would ultimately allow. 73.54 is silent on the type of facility.

MS. HOLOHAN: But right now it's only upon the tower reactors to probably do a separate rulemaking. To do anything with the facility.

MS. SCHNETZLER: Right, we need a conforming change to make it applicable to other facilities right now.

MS. HOLOHAN: But the guide applies to everything.

MS. SCHNETZLER: And basically I kind of moved us, as long as we're talking about digital security, I moved us to this slide just to let you know that it does lay out the programmatic requirements for cyber security. It treats cyber security much like physical security in that you need a cyber security plan. It needs to be reviewed and approved by us. It's a condition of license.

So those things are being applied. It is also tied to the piece in the DBT, the Design Basis Threat, 73.1, that was issued earlier this year - last year, sorry - that specifically lays out the cyber threat.

Like I said we have just issued 5022, cyber security program and that is being distributed to, as it is OUO, it is being distributed to the licensees.

MEMBER APOSTOLAKIS: So a condition of licensing -

MR. MORRIS: Yes, essentially what we are saying, we are intending to treat cyber security programs in the same fashion that we treat physical security, treating security officer training plans. They are formally reviewed, submitted and reviewed and approved, safety evaluation written, and an operating license condition established for those plants. And we're doing the same thing for this.

MEMBER APOSTOLAKIS: But my question is, when this becomes the rule, the existing plants will have to comply with it.

MR. MORRIS: Exactly. Yes, no we are still talking about implementation period. We are still talking about the licensing mechanism to make that happen. But yes.

MEMBER APOSTOLAKIS: why is this outside 1.09?

MR. MORRIS: 51.09?

MR. REED: This is a back fit, you are correct. And it's a back fit that we are justifying as a safety enhancement under 51.09(a)(3). We're saying this is substantial additional protection of public health and safety, and the costs are justified.

MEMBER APOSTOLAKIS: It's an added protection kind of thing.

MR. REED: No, added protection would be the top exception. If you got 51.09(3), this is the classic rule where you have to go and see, okay, what in fact does this do for the good side. How much enhancement does this make? And then look at the cost?

This is the classic back-fit analysis.

MR. MORRIS: There is more to the story though. We issued cyber security requirements under - by order under adequate protection after 9/11. We also did a formal notice and comment rulemaking on the design basis threat, which we completed early last year in which we added as a specific adversary characteristic external cyber attack.

Current licensees are currently required as of last April, and in fact before that when we issued the DBT order back in 2003, they are required today to be able to defend against an external cyber attack with high assurance. That is an added protection requirement.

What we are talking about in these rules are specific programmatic elements that we believe are necessary, prudent and necessary to be able to demonstrate consistently that you can provide that high assurance of added protection.

So if you look at the elements of the rule, it's a very high level programmatic elements. You have to do a complete digital systems inventory of all the systems on your site, and determine which ones are critical and which ones are not. You have to have a training program. You have to have a number of different programmatic elements to be able to meet the design basis threat requirement, and the order requirement.

So Tim's right, there are some specific things in here that I think would arguably would fit the 1.09(a)(3) kinds of things. And we put that in the Federal Register notice that advanced the proposed rule back in 2006. And we got comment on it, and we have addressed the comments. And they are reflected in the final comments language that we are about to send to media.

MEMBER APOSTOLAKIS: But the final word is we are not subjecting these to 51.09?

MEMBER MAYNARD: They are saying they did a 51.09 evaluation.

MEMBER APOSTOLAKIS: They do get one? They are evaluating -

MR. REED: Yes, we are. Scott is right, it's adequate protection in the order which is in place, and this goes beyond the order. So we are costing this thing out. And it's substantial, substantial cost on reg analysis, and we are making a judgment that this one, as well as a bunch of others in this entire package -

MEMBER APOSTOLAKIS: I understand.

MEMBER MAYNARD: Do these rules, other than programmatic elements of changing some programs, do they impose substantial additional requirements over and above what came out in the orders and stuff after 9/11? There have been 50.54(f) orders and stuff come out. I'm trying without having seen some of this stuff -

MR. MORRIS: The answer is, in short, yes. Because the order requirements that we issued were very, very, very high level, and frankly, nebulous in terms of the details of how to, and even the what in some cases.

So what we are trying to do is narrow in what we really meant when we issued those requirements by order. To reflect what we learned over the years, and what -

MS. SCHNETZLER: And provide a regulatory framework so that you would have a document that would be in place for every site, explaining how that site is addressing cyber security, that is a document that is a licensing document that we would review so everybody has a good understanding of where we're at.

MR. MORRIS: Right now we - I won't go any further. That's accurate. I don't need to say more.

MS. SCHNETZLER: Well, let's go back a little bit, if you could go back to the last slide, I just wanted to give ACRS an opportunity to talk a little bit abou8t 50.54(hh) which we did. It was originally contained in Appendix Charley of the proposed rule. We moved it to 50.54 conditions of license which we have explained.

I just wanted to cue you that there was a supplemental rule published in the Federal Register on 4-10-2008, some of the expansion from the proposed rule was the imminent threat requirements as we've discussed a little bit.

So we've received comments back on that, and those have been incorporated into this Federal Register notice that we are pulling together now.

So I just wanted to make you aware of that. We do have guidance that is being developed, and is a little further along than I expected actually, and I have good news today that it should be ready early next month.

VICE CHAIR BONACA: This is piece by piece?

MS. SCHNETZLER: Yes.

MR. MORRIS: It's two pieces, it's B-5-A and B-5-B. B-5-A was an imminent attack; B-5-B is now that you've been attacked what are you going to do about it?

MS. SCHNETZLER: Right, and the guidance has the required guidance for imminent threat. But it also takes and puts into one guide the documentation and the advisories that we had issued before on how to meet Bravo-Five-Bravo and put that into a guide so that we have a formal document from our agency that puts it altogether.

Questions on that? Okay, if we could skip then the next one and go to safety-security interface.

The safety-security interface 70.358, is a requirement for coordination of security and operations and other plant groups to make sure that there are no adverse interactions; that something security does on a regular or irregular basis doesn't adversely affect operations and vice versa.

This also addresses in part a petition for rulemaking that we received and specifically on this topic we have issued guidance on this, draft guidance, 50.21. It was published in the Federal Register, July 24th, 2007. We had a public meeting in September of last year. We received several comments on it, and the comments are under consideration for incorporation into the title and guide.

VICE CHAIR BONACA: A comment from NEI seems to me, if I remember, is the concern that by putting those check lists of questions, a la 50.59, you are expanding or you are going beyond really what the plants already have implemented, which seems to be a problem to them at least.

MS. SCHNETZLER: And it's not mentioned,

the NEI said they were going to submit their version of the safety-security guidance, and we haven't received that yet.

MR. REED: And they, NEI actually took that, I think that is what NEI did. They did not actually provide a document. I don't think they are going to. They, from what I could tell, they translated that document into another of comments that we just got here recently. And we are looking at those comments in addition to the original comments.

But you are correct, Dr. Bonaca, that I think the original concern was a concern that we were imposing broad programmatic - a new broad programmatic change control system to the whole facility. Clearly we want them to rely on using what's there to the maximum extent possible.

MEMBER MAYNARD: What I got out of the NEI comments was, all that was being said in some of the public meetings and discussions with them was different from the way they were reading the draft guidance documents coming out, and as to whether the current programs are or are not there I think is what I read -

MS. SCHNETZLER: And that is our intent in the final guidance to clarify that and make sure that the licensee can take credit for the programs that they already have in place.

VICE CHAIR BONACA: A question I have of you, do you expect one letter from the ACRS at some point, or do you expect multiple letters? Because I mean some of these issues, I noted this earlier, they are separate. Each one of them would deserve a review.

MEMBER APOSTOLAKIS: Well, that is up to us.

VICE CHAIR BONACA: Yes, I understand.

MR. MORRIS: I guess what I would say about that, and I'll let Tim comment as well, is that because of the unfortunate but real time crunch that we are under, I would prefer to get comments as they are available as opposed to waiting until all at the end when you get all your comments.

I don't know how that works out in a practical sense, but the longer we wait unfortunately the more untenable it gets.

MEMBER APOSTOLAKIS: I am confused now. I thought the last time we were going to see these things is July.

MR. MORRIS: That's the intent. The guidance document.

MEMBER APOSTOLAKIS: So how can we give you -

MR. MORRIS: A week right now seems like -

MEMBER APOSTOLAKIS: Oh, you mean before -

MR. MORRIS: A week to me right now seems like forever.

MEMBER ABDEL-KHALIK: How about comments on the reg guides? I thought those can be delayed much further than July?

MR. MORRIS: They can, absolutely.

VICE CHAIR BONACA: So what you need by July means comments on the rulings mostly?

MR. REED: Yes, I think I would - if I could - my preference would be if it's possible for the Committee to make a decision on the requirements, based on everything we can provide you in July. Basically we can provide you all the pieces of the roll-up package that go with those requirements, the draft guides, everything that can help you to make a decision why you think the requirements are adequate. And then the guidance, I think that can continue on, on a longer timeframe.

But we are trying, and the Commission is obviously pushing hard, we are trying to get these requirements in place in the Code of Federal

Regulations, okay. So that is the major goal right now.

Implementation guidance obviously is important for that in reality, so that has to be done too.

So that's how I would - I'd love for the Committee to deal with it that way, but obviously it's up to the Committee to decide.

MEMBER MAYNARD: I understand that, and may be able to do that after we see the documents. But sometimes it's difficult to understand what the real requirements are until you see how it's really going to be implemented and what the guidance documents say.

MS. HOLOHAN: But we are providing you with the guidance documents as they stand now, the draft guidance.

MR. MORRIS: This has been a particularly challenging exercise, not necessarily because of the time pressure, but because in many cases we are trying to translate what we issued by order under safeguards into publicly available notice and comment language.

And what happens as a result is, a lot of the guidance then as to move into OUO and safeguard space.

So you are absolutely right. In many

cases it's very helpful to have the guidance documents, although we have been very careful to explain to the public and the stakeholders that having the guidance document was not essential nor required or necessary in order to provide meaningful comment on the publicly available language.

So it's an interesting -

MEMBER CORRADINI: That is a very interested description you just gave. So you'll need it to understand it, but we made sure we wrote it so you don't really need it.. That's kind of what you just said.

MR. MORRIS: Well, what I'm trying to say is, what I'm trying to indicate to you is that in response to the comment is that the publicly available rule language should and does stand on its own. What we need in order to conduct sufficient licensing work and ultimately write a safety evaluation that gives us high assurance that they are actually able to meet the language - there is a different level of information that we need, and that is not information we can necessarily put in the public domain. That is what we are trying to say.

MEMBER ABDEL-KHALIK: It is quite possible that you get your most insightful comments without the detailed guidance.

MS. SCHNETZLER: And I will say in the whole rule package we received over 600 pages of comments, and they were very detailed and very insightful, and we spent a lot of staff time going through those, and trying to ensure that the final rule really explains to a licensee what is expected, and at a level that they can understand what we need to do.

MR. MORRIS: This is clearly not the ideal way to do business, by issuing a draft guidance of the final rulemaking phase. I would have much preferred to issue draft guidance with the proposed rule, but it didn't work out that way.

MS. SCHNETZLER: So that leave us with, we're a rulemaking proceeding. The guidance for 50.54(hh) is not developed, and by that I mean really not published. It is in development, and we expect to have it the 1st of July.

The guidance for 73.58 is publicly available. The guidance for 73.54 is developed and is being distributed today, so we'll be able to provide the Committee with copies of that as they need.

Then the last thing I have attached on your program is the rule text itself. And this is the text that has not been through final concurrence yet, which will happen the week of the 16th, but it's what we have as of today. And that is a draft final ruling.

VICE CHAIR BONACA: I understand. Again, I saw this before. And was - now it's changed, and it's not final. When do you think we will have a final language of the rule?

MR. MORRIS: I guess when the Commission SRM comes out.

MS. HOLOHAN: Yes, when the Commission SRM comes out. But we'll have a final when it's concurred on by the EDO to go to the Commission. But it won't be final language until the Commission votes on it.

VICE CHAIR BONACA: All we can do is distribute it to the members, and to the members to review it. I certainly would dedicate my time to that, try to see if I could also develop some thoughts on how to do it definitely will help.

And then when we come to the July meeting we will decide whether or not we have sufficient basis to write a letter. With the realization again that if things are still in flux they are not going to make a determination, because things are changing.

So one item that I still need to bring up is the aircraft impact rule, and the rule that goes - I'd like to know on what kind of track - is it a separate track that we are working on? And what's the timing for our review? What are the expectations? I thought that was coming in July too?

MS. GILLES: We have provided the ACRS with a version of the rule that has been concurred in by the first level of our management.

VICE CHAIR BONACA: Well, they have actually - it's not complete. Actually paging - pages that I remember directly and things of that kind.

MS. GILLES: I'll get with the staff, because I don't believe that should be what you have. So I will see if perhaps you don't have what we thought you had. But yes, you should have a complete Federal Register notice for the draft final rule for the aircraft impact rule.

VICE CHAIR BONACA: So we'll need to get that.

MS. BANERJEE: This is Maitri Banerjee again. The Federal Register notice if I remember right does not have the exact words of the rule. It talks about - is this a supplemental notice? Oh, I'm sorry.

MS. GILLES: No.

MS. BANERJEE: Okay. This is in the CD that you - I think may be relevant to the members. This is what it looks like.

MEMBER SIEBER: Could you state your name for the record, please?

MS. GILLES: I'm sorry, Nanette Gilles, Office of New Reactors.

MS. BANERJEE: So I will check and see what you have, and bear with any error we make in copies, and I'll correct it.

VICE CHAIR BONACA: Now and that comes to the meeting of July 2?

MS. HOLOHAN: It is I believe scheduled for the meeting in July, yes.

VICE CHAIR BONACA: So we would have a separate letter?

MS. GILLES: Yes, it would be a separate letter.

MS. HOLOHAN: But our rule is one rule, three pieces of one rule. So the aircraft impact rule is a separate rule. So you are really seeing two rules total.

VICE CHAIR BONACA: Right.

MEMBER ABDEL-KHALIK: Well, the July meeting will be closed.

MR. MORRIS: Yes, the aircraft impact rule - I mean I'm now asking a question. But it's my understanding, that's more aimed at the design certification applicants, whereas our rules, whereas the rules we're talking about are really aimed at the Combined Operating License and existing operating reactor licenses.

So it's a little, slightly different audience.

VICE CHAIR BONACA: But there are three rules.

MS. SCHNETZLER: Well, three parts or pieces of the one rule. So there are really two separate rules.

MS. BANERJEE: This is Maitri again. Can I ask you about the status of the comments resolution package? Are they available yet?

MS. SCHNETZLER: For these pieces?

MS. BANERJEE: For these pieces.

(Comments off the record)

MR. REED: We have comment responses for 73.58, 50.54(hh). I don't - I'm not sure on cyber.

MS. SCHNETZLER: Cyber is not final yet.

MR. REED: Yes, and all of them, they are all drafts final. I mean all those have been only at the very lowest technical level. So I would be a little reluctant at this point until we get some more review - fortunately, the way this is going to work, or unfortunately depending on your point of view, since we are going to give this thing to the EDO on June 30th, what we give to the Committee will have been through an awful lot of review, and it won't be in flux anymore, because we will have handed it off.

So you will have the same version that is with the EDO in a sense. I know what you're going - Meredith, address your concerns about things changing. It will be out of our hands and with the EDO at least, and maybe even with the Commission, by the time we meet with you.

VICE CHAIR BONACA: Well, I mean as I said already, we will see what comes in July. We will spend the time in June to look at whatever we get.

MR. REED: Yes.

VICE CHAIR BONACA: You have to realize it is very unusual. We don't normally review documents unless they are finalized. And we give ourselves time to review it, to have the Committee talk about it.

Here when you present us with this information, and at the meeting we have to make a decision on whether or not to write.

MR. REED: I understand.

MS. HOLOHAN: But I would like to say again it's not really a final rule until the Commission puts it -

VICE CHAIR BONACA: I don't want to belabor it, but you keep calling it all one rule. One of them is called 50.54(hh); another one is 73.54. And then there is 73 Federal Register --

MS. HOLOHAN: It's all part of the same Federal Register notice. It's all one piece of the same Federal Register notice.

MEMBER CORRADINI: Now what is that?

MR. REED: This is the current FRN.

MEMBER CORRADINI: So what we have is the rule?

MR. REED: You just have three small pieces of a very very big rule.

CHAIR SHACK: We have the rule. He has the rulemaking package.

MR. REED: This has got section by section analysis in it; substantive changes; the significant comments portion of it. It's got a lot more to it, but all of this stuff is, I didn't want to give that to the committee at this point. Again the flux issue, and reg analysis, and a lot of other things in here, okay.

The fact is this is going to be very substantial, and what Bonnie was going through a little earlier was trying to identify all those things in the package, the biggest pieces being physical security, 73.55, access authorization, 73.56, and appendix B, I think those are the three biggest ones. The rest of them get a little bit smaller, but they are all pretty substantial when you add them together.

Again, back to this Committee, though, this Committee only being involved with the safety-security interface, 50.54(hh) and cyber.

So I mean we wanted to give you the whole context, and that's why we call it one rule, because we call it the power regs Security requirements rule. So that's what Curtis is talking about. It's one rulemaking.

VICE CHAIR BONACA: All right. So I guess we will get these packages from you over the next couple of weeks. And we will communicate and see how we can transmit them and send them.

MR. GILLESPIE: I have to say, the problem the staff has is the same one you have, Mario, is if you change a piece of this you can go back and it can affect actually your comment answers on all comments.

So getting this too piecemeal to the committee I think is just going to frustrate everyone further. So I think it really is going to be far more efficient to just get it from the staff when they say it is in fact final. Because there is going to be tier domino changes through the whole package.

So one small sentence change in one section actually could change 30-40 pages in the rule, I mean just little pieces here and there. But it'll change page numbering. It'll do everything that you said.

VICE CHAIR BONACA: You suggest that we wait?

MR. GILLESPIE: I'd suggest that you wait. Because otherwise you are going to be re-reading the same material again multiple times.

MS. HOLOHAN: We will get it to you, I think what Frank is saying, after we go through concurrence before we send it back to EDO.

MR. GILLESPIE: Yes, we're looking at 220 - 250 pages of information. And just the version control by sending it to each one of the members. Because each time you go through a major revision. I'm going to guess, Tim, once a week you probably end up printing it out and rezeroing yourselves.

MR. REED: Yes, it's pretty bad. There are a lot of people doing a lot of things right now. It's pretty crazy.

MR. GILLESPIE: So I think it's really going to be more efficient for the committee to just bite the bullet, get the final package that is final, and really not frustrate yourself. Because it's going to be, I've been through this before. There are numerous little changes through it.

MS. SCHNETZLER: Would you like us to supply the guidance that is available now, now, or as a package at the end of the month?

MR. GILLESPIE: I think that is a good questions for the committee. Is the committee willing to look at the rule as a stand-alone rule much as the public was asked to do, and write a letter on that, and then deal with the guidance in a more orderly way through the fall, because there is time to deal with the guidance.

CHAIR SHACK: WE can get the guidance now and deal with it.

MS. SCHNETZLER: I'm saying it is available. I can make it available.

CHAIR SHACK: The draft guidance isn't going to change between now and then. The draft reg guides. Not the aircraft impact one; not the one that's in preparation.

MR. GILLESPIE: The ones that are ready to go out for public comment, or comment to the approved audience.

MS. SCHNETZLER: So I have two guides that are currently available for comment or have been commented on. So I'll provide those for ACRS so you can look at those. They won't change. And then the very first part of next month we'll have the other guide.

MEMBER CORRADINI: Just - I have been trying to listen and not ask questions. So what you waved is public, the thing you held up?

MR. REED: Oh, yes, this is a rulemaking document, so everything here will obviously be public. It will be public, of course.

MEMBER CORRADINI: All right, and the reg guides will be in draft form still OUO, whatever you call it.

MS. HOLOHAN: OUO, only two of them.

MEMBER CORRADINI: That's fine. Don't try to explain it to me. I'll forget it. There's no point. But in particular the rule itself is what we have in front of us, and all the - I'll call it justification. You used other language. All the associated stuff is there. Thank you.

MR. MORRIS: So let me just get clarification on something. So is the committee interested in getting the complete package that we send to the EDO, or the complete package that the EDO sends to the Commission, because they could be two different things?

MR. GILLESPIE: Traditionally on a rulemaking the committee would get a complete package that goes to the EDO. Normally they would get it when the office director signs it out. Normally the EDO does not significantly change it, and any editor changes that do get made are easy to deal with.

MS. SCHNETZLER: We have one other person here.

MR. RACKLEY: Bill Rackley, Office of New Reactors. I did just want to clarify for the aircraft impact rule. Maitri is going to give you the CD; it has the draft final rule.

Also accompanying it will be a draft NEI guidance document, NE 0713. We plan ultimately to endorse, assuming we can work out the last details, that in a reg guide, and we'll be coming back to the ACRS for the reg guide.

However, just to provide additional information such that you could see in that particular rule how it was implemented or likely to be implemented, we gave you the current version of that guidance. It's a work in progress; we are continuing to work within NEI to do some details. But it will give you a good impression of how the industry plans to implement that particular assessment.

MS. BANERJEE: I distributed this aircraft impact rule and draft guide at the last meeting. The draft guide is the January version. I will give you another copy with a May version if you want me to so that you can destroy that one, and this is OUO also.

MR. GILLESPIE: A one-for-one replacement; that way there is no confusion.

MR. ZIMMERMAN: Dr. Bonaca, if I could just add one last comment. I'm Jake Zimmerman. I'm from the office of NRR.

From a process standpoint this is clearly not the way that we would like to continue doing business with the ACRS as far as rulemaking, or even with our external stakeholders. We would like to have the proposed language and the regulatory guides available simultaneously.

In this case we weren't able to accomplish that just due to the knowledge and skills available also needed to be working on these same documents. And so the resources had to be carefully scheduled.

It is something that we have discussed with the Commission on how we can do better on rulemaking, and in our streamlining initiative. But clearly going forward we intend to try to do a better job, and I think the aircraft rule is a model now that we want to continue to follow, which is to give you those documents at the same time so that when there is cases of high level language you will have the regulatory guidance available that would show you how we intend it to be implemented.

MS. SCHNETZLER: So I have that. I'm going to, when the EDO package is final, and ready to go to the EDO, we will provide that to you. But in lieu of that ahead of time I will put the draft guidance on disk for Maitri to distribute to everybody.

MEMBER ABDEL-KHALIK: Can we get that before we leave?

MS. BANERJEE: It would be possible, if you give it to me tomorrow.

MR. GILLESPIE: Yes, Bonnie, give her until - I'm going to guess the hearing until about 11:30 to 12:00 on Friday. So if we can get it by Friday morning, we will be all set to distribute it.

MS. SCHNETZLER: That would be perfect. I was thinking tonight. So that's okay, thank you very much.

(Laughter)

MEMBER APOSTOLAKIS: Are we done, Mr. Chairman?

CHAIR SHACK: No, but we will go off the record.

(Whereupon at 5:11 p.m. the proceeding in the above-entitled matter was adjourned)

CERTIFICATE

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

Name of Proceeding: Advisory Committee on Reactor Safeguards

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.

Charles Morrison
Official Reporter
Neal R. Gross & Co., Inc.

Security Rulemaking for Nuclear Power Plants

ACRS Presentation
June 4, 2008

Discussion Topics

- Status of Power Reactor Security Rulemaking
- Staff Draft Final Rule Text needing ACRS review
    - 50.54(hh) Imminent Attack/Mitigative Measures
    - 73.54 Cyber Security
    - 73.58 Safety/Security Interface
- Status of Regulatory Guidance

Security Rulemaking

- Part 73 Power Reactor Security Rulemaking (proposed rule published 10/06)
  - 50.54 (hh) Mitigative Strategies and Response Procedures for Potential or Actual Aircraft Attacks
  - 73.54 Protection of Digital Computer and Communication Systems and Networks
  - 73.55 Physical Security for Power Reactors
  - 73.56 Personnel Access Authorization Requirements for Nuclear Power Plants

Security Rulemaking (cont.)

- Part 73 Power Reactor Security Rulemaking (proposed rule published 10/06)
  - 73.58 Safety/Security Interface Requirements for Nuclear Power Plants
  - Appendix B to Part 73 – Section VI, Nuclear Power Reactor Training and Qualification for Personnel Performing Security Program Duties
  - Appendix C to Part 73 – Licensee Safeguards Contingency Plans

Security Rulemaking for Nuclear Power Plants

ACRS Presentation
June 4, 2008

Discussion Topics

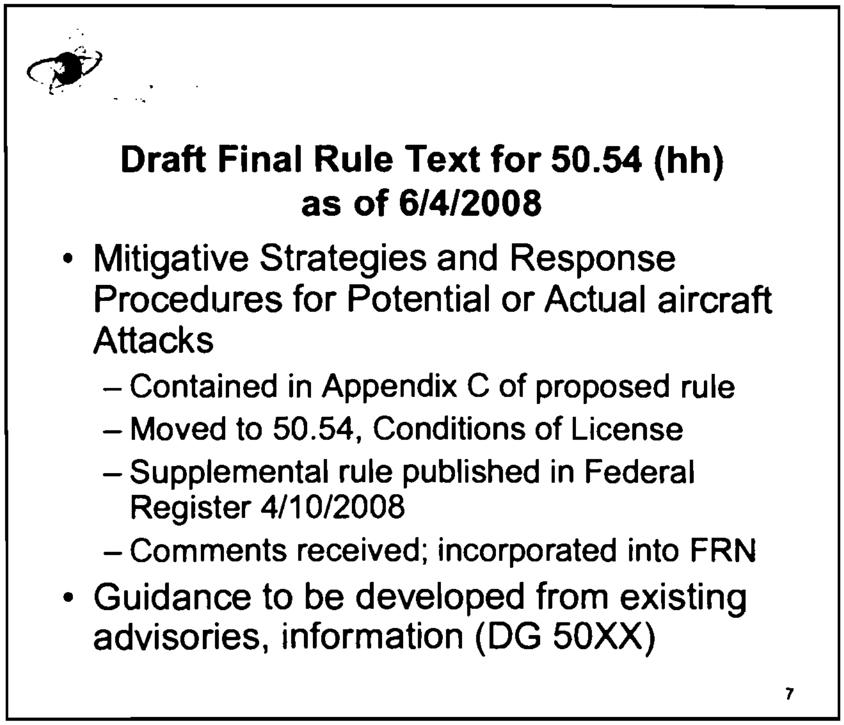
- Status of Power Reactor Security Rulemaking
- Staff Draft Final Rule Text needing ACRS review
  - 50.54(hh) Imminent Attack/Mitigative Measures
  - 73.54 Cyber Security
  - 73.58 Safety/Security Interface
- Status of Regulatory Guidance

Status of Rulemaking

- FRN developed
- Begin formal concurrence on 6/16/2008
- Provide to EDO on 6/30/2008

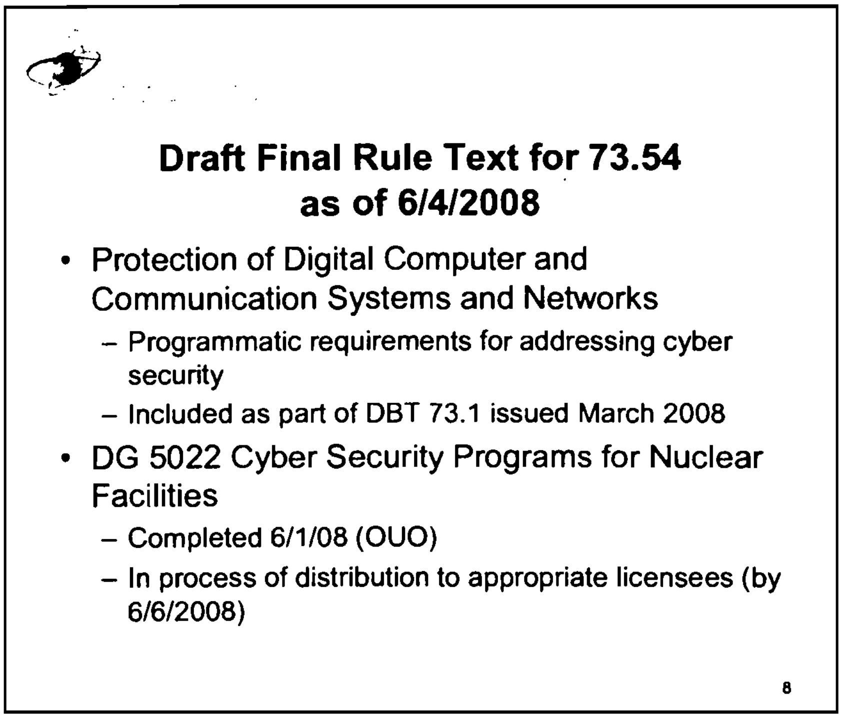
ACRS Review for Rulemaking

- 50.54 (hh) Mitigative Strategies and Response Procedures for Potential or Actual aircraft Attacks
  - DG-50XX (July 2008)
- 73.54 Protection of Digital Computer and Communication Systems and Networks
  - DG 5022
- 73.58 Safety/Security Interface Requirements for Nuclear Power Plants
  - DG 5021 Safety/Security Interface



Draft Final Rule Text for 50.54 (hh) as of 6/4/2008

- Mitigative Strategies and Response Procedures for Potential or Actual aircraft Attacks
  - Contained in Appendix C of proposed rule
  - Moved to 50.54, Conditions of License
  - Supplemental rule published in Federal Register 4/10/2008
  - Comments received; incorporated into FRN
- Guidance to be developed from existing advisories, information (DG 50XX)



Draft Final Rule Text for 73.54 as of 6/4/2008

- Protection of Digital Computer and Communication Systems and Networks
  - Programmatic requirements for addressing cyber security
  - Included as part of DBT 73.1 issued March 2008
- DG 5022 Cyber Security Programs for Nuclear Facilities
  - Completed 6/1/08 (OUO)
  - In process of distribution to appropriate licensees (by 6/6/2008)

Draft Final Rule Text for 73.58 as of 6/4/2008

- Safety/Security Interface Requirements for Nuclear Power Plants
  - Requires coordination of potential adverse interactions between security activities and other plant activities
  - Addresses PRM 50-80, in part
- DG 5021 Safety/Security Interface
  - Published in Federal Register July 24, 2007
  - Public Meeting held; comments received & under consideration

Summary

- Security Rulemaking proceeding
- Supporting Regulatory Guidance for 50.54(hh) not developed
- Supporting Regulatory Guidance for 73.58 and 73.54 developed and drafts published or distributed

SECURITY RULEMAKING
STAFF DRAFT FINAL RULE LANGUAGE
As of 6/4/2008

§ 50.54(hh) Mitigative Strategies and Response Procedures for Potential or Actual Aircraft Attacks.

(1) Each licensee shall develop, implement and maintain procedures that describe how the licensee will address the following areas if the licensee is notified of a potential aircraft threat:
(i) Verification of the authenticity of threat notifications;
(ii) Maintenance of continuous communication with threat notification sources;
(iii) Contacting all onsite personnel and applicable offsite response organizations;
(iv) Onsite actions to enhance the capability of the facility to mitigate the consequences of an aircraft impact;
(v) Measures to reduce visual discrimination of the site relative to its surroundings or individual buildings within the protected area;
(vi) Dispersal of equipment and personnel, as well as rapid entry into site protected areas for essential onsite personnel and offsite responders who are necessary to mitigate the event; and
(vii) Recall of site personnel.
(2) Each licensee shall develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire, to include strategies in the following areas:
(i) Fire fighting;
(ii) Operations to mitigate fuel damage; and
(iii) Actions to minimize radiological release.
(3) This section does not apply to a nuclear power plant for which the certifications required under § 50.82(a) or § 52.100(a)(1) of this chapter have been submitted.

§73.54 "Protection of digital computer and communication systems and networks"

(a) Each licensee subject to the requirements of this section shall provide high assurance that digital computer and communication systems and networks are adequately protected against cyber attacks, up to and including the design basis threat as described in Title 10 of the Code of Federal Regulations (10 CFR) Part 73, Section 73.1.
(a)(1) The licensee shall protect digital computer and communication systems and networks associated with:
(a)(1)(i) safety-related and important-to-safety functions,
(a)(1)(ii) security functions,
(a)(1)(iii) emergency preparedness functions, including offsite communications,
(a)(1)(iv) support systems and equipment which, if compromised, would adversely impact safety, security or emergency preparedness functions.
(a)(2) The licensee shall protect the systems and networks identified in paragraph (a)(1) of this section from cyber attacks that would:
(a)(2)(i) adversely impact the integrity or confidentiality of data and/or software;
(a)(2)(ii) deny access to systems, services, and/or data, and;
(a)(2)(iii) adversely impact the operation of systems, networks, and associated equipment.
(b) To accomplish this, the licensee shall:
(b)(1) analyze digital computer and communication systems and networks and identify those assets that must be protected against cyber attacks to satisfy paragraph (a) of this section,
(b)(2) establish, implement, and maintain a cyber security program for the protection of the assets identified in (b)(1) of this section, and;

(b)(3) incorporate the cyber security program as a component of the physical protection program.
(c) The cyber security program must be designed to:
(c)(1) implement security controls to protect the assets identified by paragraph (b)(1) of this section from cyber attacks,
(c)(2) apply and maintain defense-in-depth protective strategies to ensure the capability to detect and respond to cyber attacks,
(c)(3) mitigate the adverse affects of cyber attacks, and;
(c)(4) ensure that the functions of protected assets identified by paragraph (b)(1) of this section are not adversely impacted due to cyber attacks.
(d) As part of the cyber security program, the licensee shall:
(d)(1) ensure that appropriate facility personnel, including contractors, are aware of cyber security requirements and receive the training necessary to perform their assigned duties and responsibilities effectively.
(d)(2) evaluate and manage cyber risks.
(d)(3) ensure that modifications to assests identified by paragraph (b)(1) of this section, are evaluated prior to implementation to ensure that the cyber security performance objectives identified in (a)(1) are maintained.
(e) The licensee shall establish, implement, and maintain a cyber security plan that implements the cyber security program requirements of this section.
(e)(1) The cyber security plan must describe how the requirements of this section will be implemented and must account for the site-specific conditions that affect implementation.
(e)(2) The cyber security plan must include measures for incident response and recovery for cyber attacks. The cyber security plan must describe how the licensee will:
(e)(2)(i) maintain the capability for timely detection and response to cyber attacks,
(e)(2)(ii) mitigate the consequences of cyber attacks,
(e)(2)(iii) correct exploited vulnerabilities, and;
(e)(2)(iv) restore affected systems, networks, and/or equipment affected by cyber attacks.
(f) The licensee shall develop and maintain written policies and implementing procedures to implement the cyber security plan.
(f)(1) Policies, implementing procedures, site-specific analysis, and other supporting technical information used by the licensee need not be submitted for Commission review and approval as part of the cyber security plan; but are subject to inspection by NRC staff on a periodic basis.
(g) The cyber security program shall be audited as a component of the physical security program and will be subject to the same requirements and controls.
(h) The licensee shall retain records and supporting technical documentation required to satisfy the requirements of this section until the Commission terminates the license for which the records were developed, and shall maintain superseded portions of these records for at least three (3) years after the record is superseded, unless otherwise specified by the Commission.

SECURITY RULEMAKING
STAFF DRAFT FINAL RULE LANGUAGE
As of 6/4/2008

§ 73.58 Safety/Security Interface Requirements for Nuclear Power Reactors

(a) Each operating nuclear power reactor licensee with a license issued under part 50 or 52 of this chapter shall comply with the requirements of this section.
(a)(1) The licensee shall assess and manage the potential for adverse affects on safety and security, including the site emergency plan, before implementing changes to plant configurations, facility conditions, or security.
(a)(2) The scope of changes to be assessed and managed must include planned and emergent activities (such as, but not limited to, physical modifications, procedural changes, changes to operator actions or security assignments, maintenance activities, system reconfiguration, access modification or restrictions, and changes to the security plan and its implementation).
(b) Where potential adverse interactions are identified, the licensee shall communicate them to appropriate licensee personnel and take compensatory and/or mitigative actions to maintain safety and security under applicable Commission regulations, requirements, and license conditions.

Synthesis on the findings from the ARTIST tests on aerosol retention in the secondary side of steam generators

Presented to the ACRS
June 4, 2008

M. Salay
U.S. Nuclear Regulatory Commission
Washington, D.C., USA

Overview

- Steam Generator Tube Ruptures (SGTR) background and NRC interest-SGAP
- ARTIST test program pertaining to SGAP
- Major Observations
- MELCOR modifications
- Conclusions

Steam generator tube rupture accidents

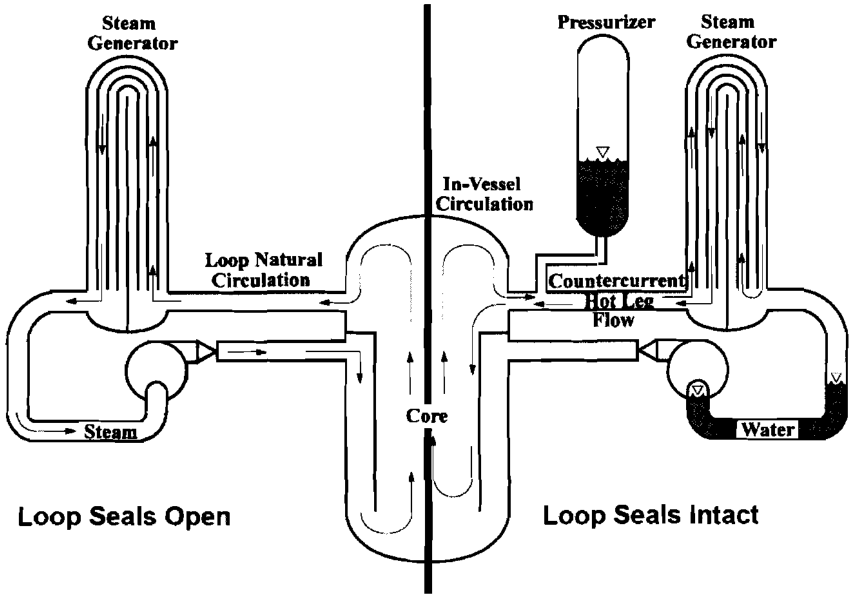
- Design basis event
  - Plants designed to cope
  - Have for all events to date
- Progresses to severe accident only if something else happens
  - Operator error

Induced steam generator tube rupture

- Induced rupture greater concern
  - Plants operate with detectable flaws in tubes
  - Limit on flaw size
  - Stress corrosion cracking is the cause of most flaws
  - Crevice corrosion at tube support plates of concern

Induced steam generator tube rupture

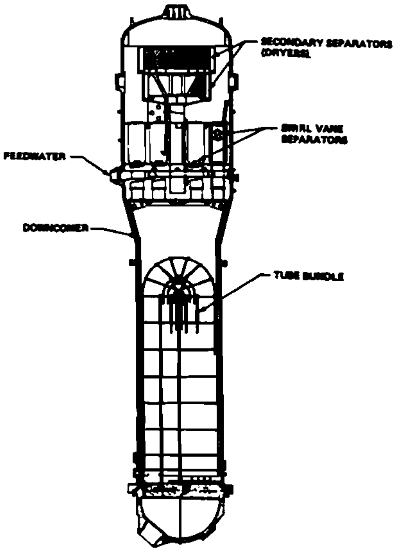
- Heat transfer from core to primary pressure boundary weakens structures
- Vulnerable locations
  - Hot leg nozzle
  - Surge line to pressurizer
  - Steam generator tubes
- Codes do not reliably predict failure location and depressurization timing



Severe accident natural circulation flows

Aerosol retention in SGTR SA

- at tube inlet from steam generator plenum (inlet efficiency)
- in the steam generator tube prior to reaching the tube rupture
- in the immediate vicinity of the break where particles could impact on adjacent tubes
- in tubes between one tube support plate and another
- on top of tube support plates
- on envelope by thermophoretic deposition
- in the steam separators and steam dryers at the top of the steam generator.
- at steam generator safety relief valve (inlet efficiency)



Westinghouse Model F Steam Generator

Aerosol retention processes

- Removal mechanisms particle size dependent
  - Laminar
    - large – impaction, settling, interception
    - small – diffusion
  - Turbulent
    - turbulent deposition
      - bounce
      - flow resuspension
      - saltation
- Removal of particles alters particle size distribution
  - maximum penetration size
  - retention of individual sections can not be simply combined to obtain overall retention
    - integral tests
  - SETs obtain individual section retention as function of size

Aerosol size

- A recommendation of prototypic aerosol size based on an IRSN survey of AECL, PBF-SFD and PHÉBUS experiments:
  - "size distribution at SG: near-lognormal, AMMD ~1μm or less, $\sigma \sim 2$; larger particles comprise agglomerates of small (~0.1 μm) highly coordinated clusters"
    - Sizes in two of the facilities were in the maximum penetration size range
    - Larger size range in third facility

Consequences of tube rupture

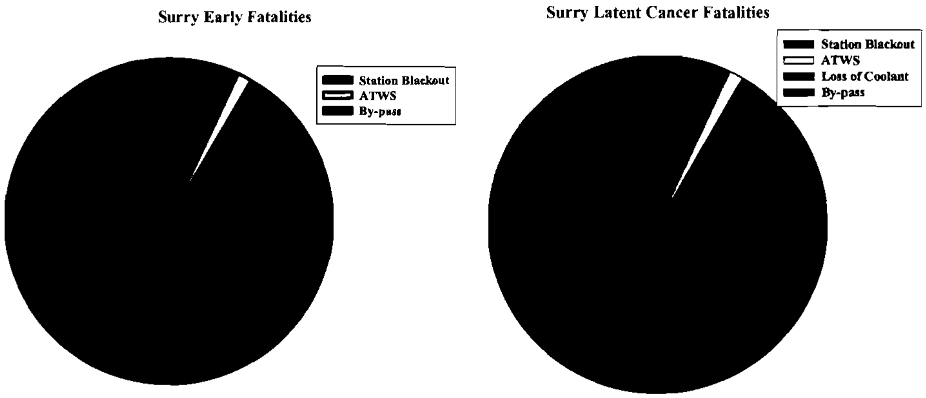
- Radionuclides vent directly to environment or to auxiliary building without any attenuation from engineered safety features in containment
- Accidents have sufficiently high consequences that they are risk dominant despite low probability

NUREG-1150

- Risk analysis of five US plants
  - Two PWRs had significant probabilities of steam generator tube rupture
  - All three PWRs could suffer induced steam generator tube rupture
- Limited modeling of aerosol behavior on secondary side of steam generators
  - None in the Source Term Code Package
  - Data unavailable

NUREG-1150 expert opinion elicitation

- Inlet efficiency from steam generator plenum to ruptured tubes – DF (mass in/mass out) ~2
- Retention in tubes - DF <~10 - no credit given
  - resuspension
  - revaporization
  - agglomerate breakup
- Retention in secondary side - DF ~4 to 6
  - deposition on outside of tubes resisted by thermophoresis
- No credit for steam dryer/separators
  - proprietary design information
- Large uncertainty in estimates



Alternate retention analysis

- Industry analyses provided far different estimates of retention in the secondary side of steam generator
  - Calculated steam generator DF on the order of 10,000
    - >100 in tube, depending on break location
    - 10s secondary near break
    - 2-3 far from break

Focus on SGTR bypass accident

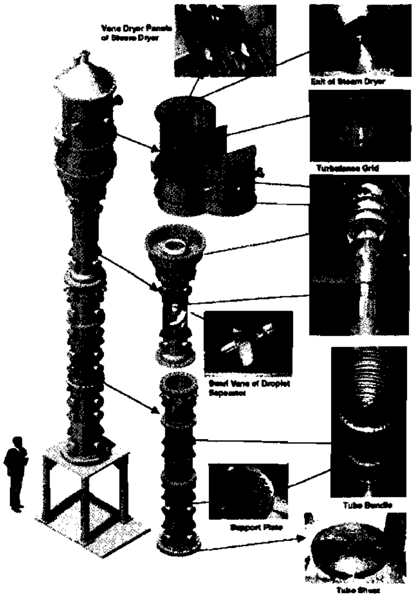
- attention to SGTR bypass accidents justified by risk
- Direct connection between risk and source term attenuation
- "are safety resources being misdirected to an unneeded attention on containment bypass accidents because we underestimate attenuation"

SGAP ITEM 3.3a

- STEAM GENERATOR ACTION PLAN (SGAP) ITEM 3.3a – DEVELOP EXPERIMENTAL INFORMATION ON SOURCE TERM ATTENUATION ON THE SECONDARY SIDE OF STEAM GENERATORS

ARTIST Project

- AeRosol Trapping In a STeam generator
  - International project conducted by the Paul Scherrer Institut (PSI)
  - seven phase project (NRC participated in 5)
  - separate and integral tests (38)
- retention measured:
  - in the steam generator tube prior to reaching the tube rupture (15)
  - in the immediate vicinity of the break where particles could impact on adjacent tubes (9)
  - in tubes between one tube support plate and another and on top of tube support plates (6) (1 stage,2 stage)
  - in the steam separators and steam dryers at the top of the steam generator. (5)
  - overall with all steam generator components (3)
- Other phases (not NRC)
  - retention in flooded bundle
  - droplets in dryers and separators



ARTIST facilities

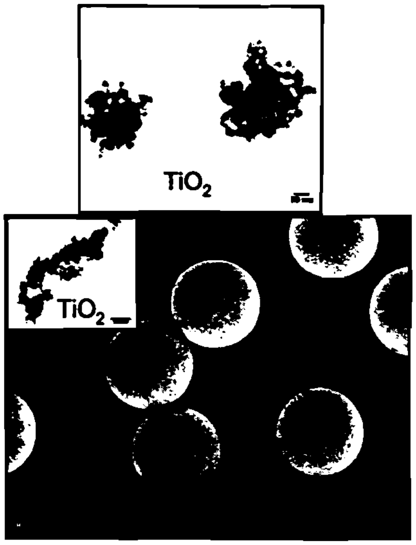
- ARTIST
  - based on Beznau plant: 365 MWe Westinghouse 2 loop PWR (1969,1972)
  - scaled for SGTR
  - 19.08 mm tube diameter
  - approx 1:20 flow area and number of tubes
- Main facility
  - shortened and narrowed bundle with U-bend tube section
  - a tube sheet
  - 3 support plates
  - full scale separator and dryer
- SET facilities
  - in tube
  - at break
  - rods far from break and support plates
  - separator and dryer

	Beznau	ARTIST
Number of tubes	3238	270 (89)*
Dryers	12	1
Separators	12	1
Bundle dia. (m)	2.68	0.57
Max tube height (m)	9	3.8 (9)**
Flow area (m^2)	3.79	0.185
Sup. plate flow area (m^2)	1.288	0.052
Bundle D_h (cm)	3.1	3.1
Total height (m)	17	10.5

*separate test section for assessing retention far from break
**in tube retention tests

Test Parameters

- Guillotine break
- Aerosol particles (composition/size)
  - TiO_2 agglomerates (AMMD 1-5 µm)
    - Degussa
    - Nanophase
  - SiO_2 spheres, D_{ae} = 0.7, 1.4, 3.7 µm
  - Latex spheres, D_{ae} = 0.4 µm
- Concentrations
  - 0.01 to 100s of mg/m^3
- Flow rate:
  - nitrogen (steam)
  - few 10s – several 100s kg/h
- scoping tests to determine suitable parameters precede experiments
- tests to determine experimental uncertainty



TEM micrographs: Dr. Jerry Egeland / PSI

SEM micrograph: Dr. Unto Tapper / VTT

Primary Measurement Methods

- Size distribution, concentration, retained mass, and DF
  - sampling at inlet, outlet, and other locations
- Size distribution:
  - Berner Impactor
  - Electrical Low Pressure Impactor
  - Optical Particle Counter
- Concentration:
  - Filter
  - Photometer
  - Optical Particle Counter
- Mass collection, concentrations with flow used to determine DF
- Flow rates at inlet and outlet and at all sampling devices, gauge pressures at inlet and outlet, gas T

Major observations

- Two forms of aerosol deposition:
  - Always a fairly uniform layer of fine aerosol on surfaces exposed to the aerosol-laden flow. "tenacious"
  - A second form of deposit noticed in some tests consists of 'clumps' of deposited material.
- Widely varying retention in tubes
  - from test to test
  - high retention over short periods of time
- Resuspension can occur for deposits in tubes
  - bounce and break-up of aerosol important
- Large agglomerates did not survive transport at high flows
  - uniform size distribution leaving tube
  - particles smaller than ~1 μm don't break up but larger particles do
- No major retention at rupture site
  - Expected based on studies of rupture propagation

Major observations

- Away from break, most of deposited mass on support plate
  - May be flow recirculation at broached holes for steam generator tubes
  - May not occur for US plants with drilled tube support plates
    - Flow occurs through larger holes; jets
    - Gaps around tubes usually filled with "crud"
- Dryer/Separator not a major source of aerosol retention even for relatively coarse aerosols
  - Fin spacing large and little aerosol diffusion

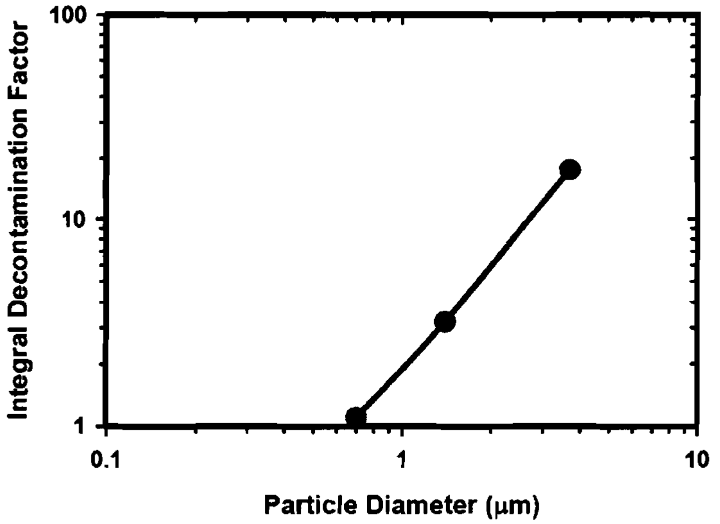
Outstanding issues

- Understanding "bounce"
- Understanding breakup
  - specific to test aerosol?
- Understanding resuspension
  - effect of vibrations
- Features of steam generator
  - Thermophoretic deposition on envelope
- Shapes and sizes of particles coming from the degrading reactor core reaching SG

Changes to MELCOR

- include a "lambda" factor based directly on the ARTIST results
  - based on particle size
  - insufficient risk change incentive to do more in the face of other pressing work
- monitoring 1D model being developed at Ciemat in Spain

ARTIST integral test results



Conclusions

- Expert panel recommendations made for NUREG 1150 risk analyses by and large confirmed
- MELCOR predicts decontamination factors similar to those obtained from ARTIST data.
- Modifications made to MELCOR based on ARTIST data
- ARTIST provides experimental data on source term attenuation on the secondary side of steam generators
  - Steam Generator Action Plan (SGAP) item 3.3a complete

ARTIST

Aerosol Trapping in Steam Generator ARTIST: Findings and Potential Effects on SGTR Risk Profile

ARTIST team:

Salih Güntay, Abdel Dehbi, Steffen Danner, Ralf Kapulla,
Terttaliisa Lind, Hauke Schütt, Detlef Suckow
Paul Scherrer Institut, Switzerland

Outline

- History
- Aims of ARTIST
- ARTIST International Consortium Project
- Facility and scaling
- Model aerosol particles
- Experimental Program and results
- Conclusions
- A new SGTR risk assessment methodology and use of ARTIST data
- Final remarks

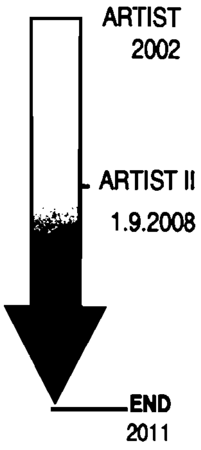
History

- **Motivation and support from Utility: Large contribution of SGTR in CDF and Risk in NPP-Beznau due to excessive tube problems in 1997**
- **Design and Procurement: 1998-2000**
- **EU 5. Framework Project SGTR: 2000-2002: PSI (Vertical SG without Dryer/separator), VTT (Exp: horizontal SG), NRG , Rez, CIEMAT**
- **ARTIST International Consortium Project**

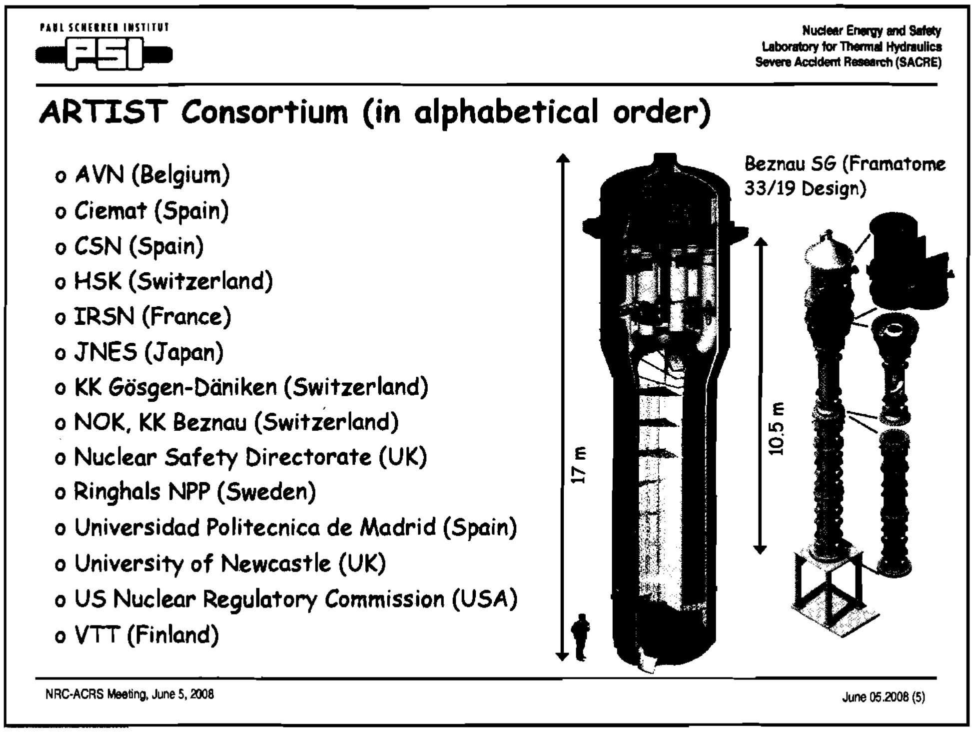
  Phase I: 2002-2007

  Phase II: 2008-2011
- **Potential continuation >2011: in form of Fundamental Studies (PhD), model development efforts at PSI**

Aims of the ARTIST International Consortium project



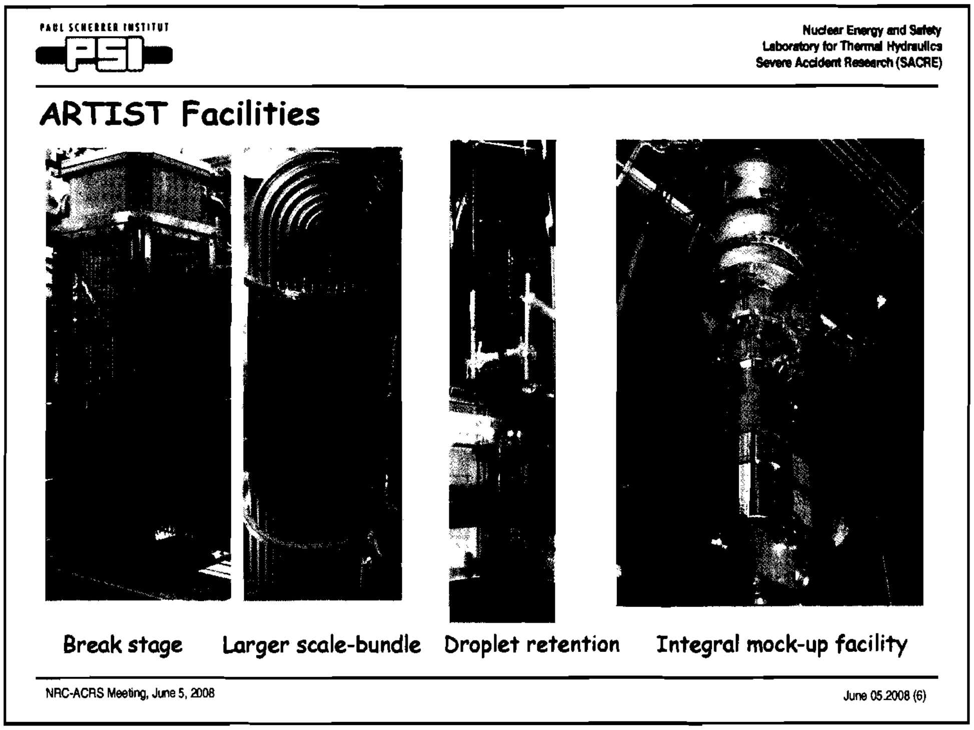
- **Provide an international forum to develop new information and share among partners**
- **Produce high quality data for:**
  - Development of fundamental and detailed to simplified and application oriented models
  - Facilitate evaluation of effectiveness of SAMG
- **Develop methodology for SGTR Risk Assessment**
  - Re-assessment of SGTR induced environmental risk
  - Provoke international consensus about the risk significance of SGTR events during DBA and SA
- **Initiate fundamental investigations in form of PhDs/Masters**



ARTIST Consortium (in alphabetical order)

- o AVN (Belgium)
- o Ciemat (Spain)
- o CSN (Spain)
- o HSK (Switzerland)
- o IRSN (France)
- o JNES (Japan)
- o KK Gösgen-Däniken (Switzerland)
- o NOK, KK Beznau (Switzerland)
- o Nuclear Safety Directorate (UK)
- o Ringhals NPP (Sweden)
- o Universidad Politecnica de Madrid (Spain)
- o University of Newcastle (UK)
- o US Nuclear Regulatory Commission (USA)
- o VTT (Finland)

Beznau SG (Framatome 33/19 Design)



ARTIST Facilities

Break stage Larger scale-bundle Droplet retention Integral mock-up facility

Scaling

Design basis: Framatome 33/19 Design

- Separator: 1:1 (steal or mostly transparent)
- Dryer: 1:1 (with actual Chevron panels) (all steel or inlet transparent)
- Bundle: 264 straight tubes, height: 1:0.42, with 1:1 layout
  - Broached support plates with 1:1 layout
- Single tube length: 1:1 with smallest and medium curvatures
- Tube dimensions: 1:1

Flow rates: 40 kg/h to 800 kg/h (fully representative)

Pressure: < 5 bar in primary, ~ 1 bar secondary

Dry conditions (except 1 in-tube test with slight steam condensation)

Model Aerosol Particles

- Evaporation and Condensation generated single/multi component Particles (SnO/CsI/CsOH, etc) (not used for ARTIST due to high costs)
- Fluidization of mono/polydisperse powders (TiO_2 (two types), SiO_2)
- Dispersion of suspended material (Latex, SiO2 in solution) and drying droplets
  . Monodisperse particles (SiO_2/Latex): well known size
  . Polydisperse particles (TiO_2): lots of problems due to

  unknown surface finish characteristics affecting deposition and no size control due to de agglomeration at high velocity/sonic front

Particle Morphology and Size in PWR Hot leg

- Working group: M. Kissane (IRSN), D. Powers (SNL), M. Reeks (NC)
- Very complicated and not resolved issue since many parameters (pressure, core degradation, etc) influence
- Hot leg conditions based on Phébus and other tests
- Phébus:
  - 15-40 % control rod metals, similar amount of oxides, and rest FPs
  - implies an "onion-skin" type of structure where the kernel rich in highly refractory materials and on top condensed species of more volatile species containing cesium and rubidium and perhaps migrated into and interact chemically with the substrate
  - **For practical purpose AMMD at SG inlet or in SG based on impactor data**
    - **3 µm (gsd 2) at 150 °C, 1.7 µm (gsd 2) at 730 °C, 0.1 µm at 930 °C following an exponential increase along inverse temperature**

ARTIST experimental program

BDBA source term quantification		ARTIST
Phase I:	In tube	15
Phase II:	Break stage	9(+2)
Phase III:	Far field	8(+2)
Phase IV:	Separator&dryer	5
Phase V:	Flooded bundle	2(+3)
Phase VII:	Integral mock-up	3
	Total	**42(+7)**

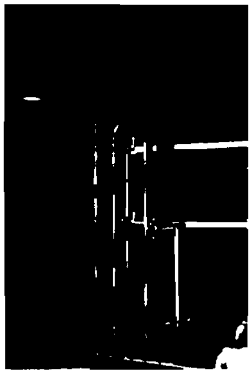
DBA source term quantification

Phase VI:	Droplets (in separator & dryer)	yes

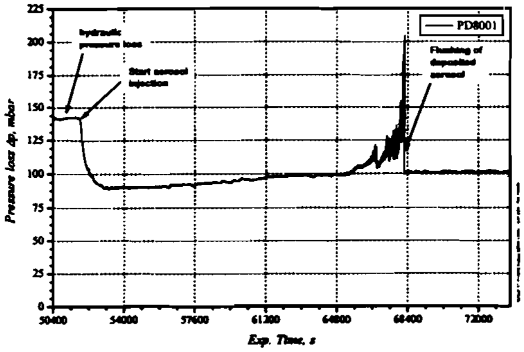
(x): EU-SGTR

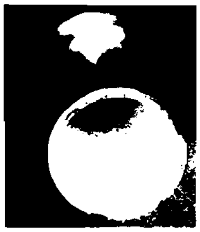
Phase I, In-tube retention (1:3)

- 15 tests
  - 225 - 364 kg/h, with pressure ratio of 3.5:1
  - Straight tube and
  - U-tube with two bend diameters (83 and 384 mm)
  - Dry conditions, except 1 test with slight steam condensation
  - Mono/Polydisperse particles
  - Very low to modest concentrations



Phase I, In-tube retention (2:3)



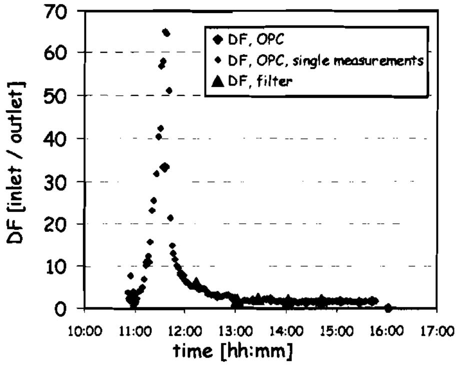


- 2*9 m with 83.2 mm curvature
- 70 -240 m/s velocity in Tube
- Dry TiO2 (2-3 µm inlet/<1 µm outlet)

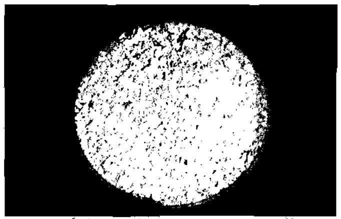
- Very dynamic aerosol processes (turbulent deposition/resuspension, de-agglomeration of TiO_2)
- Challenge for modeling (PhD Pamela Longmire/SNL)
- Effect on flow re-distribution among intact tubes in inlet plenum

Phase I, In-tube retention (2:3)

DF	Conc.	Particles
< 65	medium	SiO_2
1.0 - 2.2	medium	TiO_2
8.2	Slight steam cond.	TiO_2
< 100	very low	SiO_2, latex

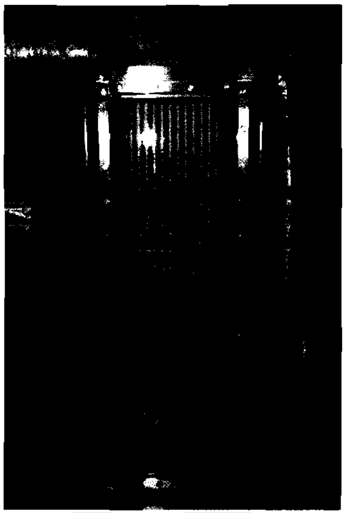


1.4 μm SiO_2, high concentration



Aerosol (SiO_2) fragments collected in the outlet plenum

Phase II: Break-Stage Retention: Dry conditions (1:6)



- **Chocked flow at the break**
- **Guillotine Break**
- **Dry conditions**

9 tests

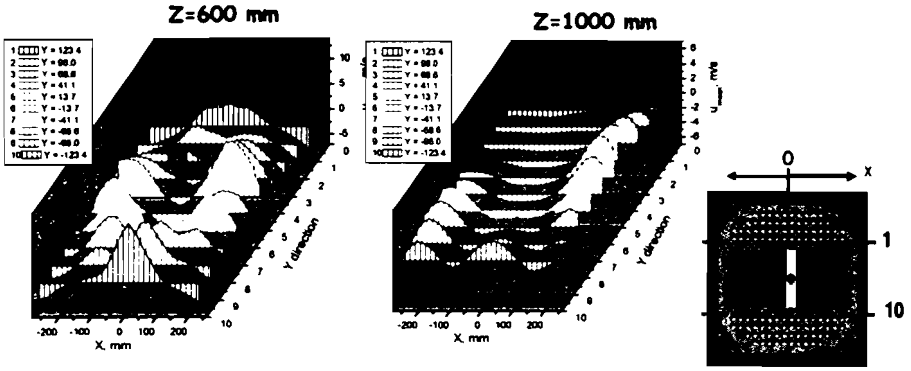
- 360 kg/h,
- Monodisperse SiO2 particles
- AMMD: 1.4 to 3.8 µm

2 tests with full bundle

- 600 kg/h
- **Polydisperse TiO2 particles**
- AMMD: 2.3 µm before break

Phase II: Break-Stage Retention: Velocity profiles (2:6)

Measured velocity profile: **Guillotine Break,** 360 kg/h

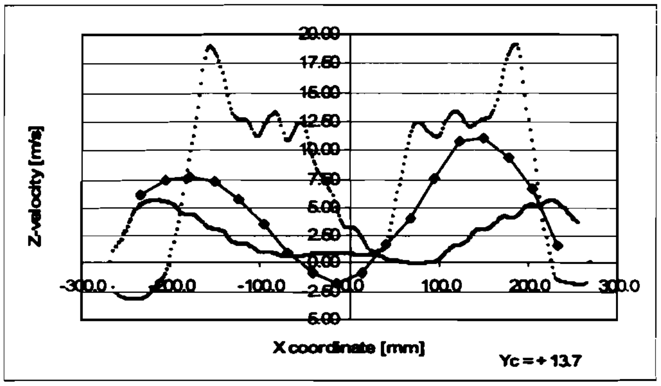


Z=600 mm

Z=1000 mm

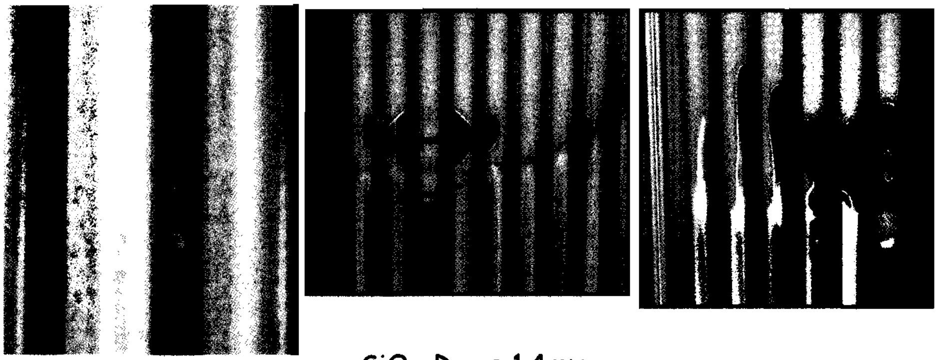
- ➢ Very 3D flow

Phase II: Break-Stage Retention: Velocity profiles (3:6)



- Measured velocity profile
  FLUENT Simulations by Ringhals/EPSILON
- with k-ε
- with Reynolds Stress Model (RSM)

Phase II, Break stage (4:6): Aerosol material type dependent local deposition pattern



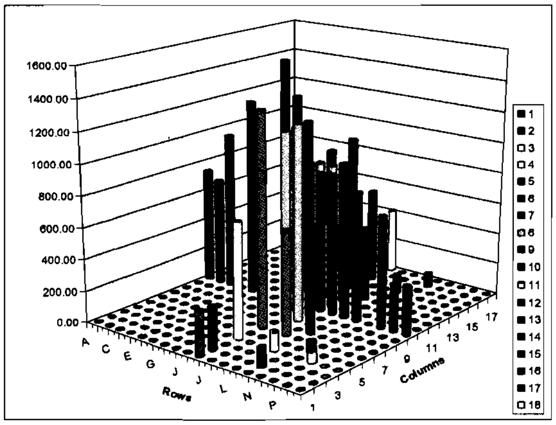
TiO_2, Dae = 2.3 μm

SiO_2, Dae = 1.4 μm

SiO_2, Dae = 3.7 μm

- ➢ Flow rate: 600 kg/h for TiO_2, 360 kg/h for SiO_2 tests

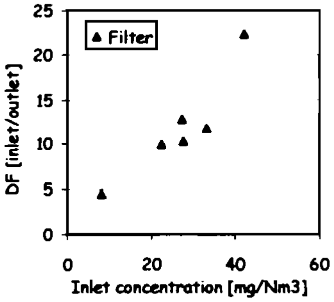
Phase II, Break stage (5:6): Deposition pattern

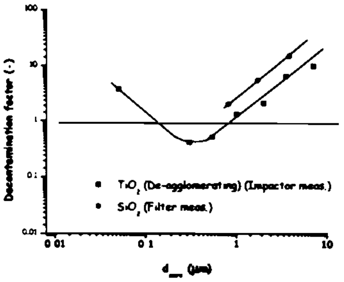


Tube to tube aerosol deposition profile (SiO2, 3.8 μm)

Phase II, Break stage (6:6): Retention

- o **Highest retention potential among other retention stages**
- o **Decontamination Factor =**
  - increases with increasing inlet concentration
  - increases with increasing D_p



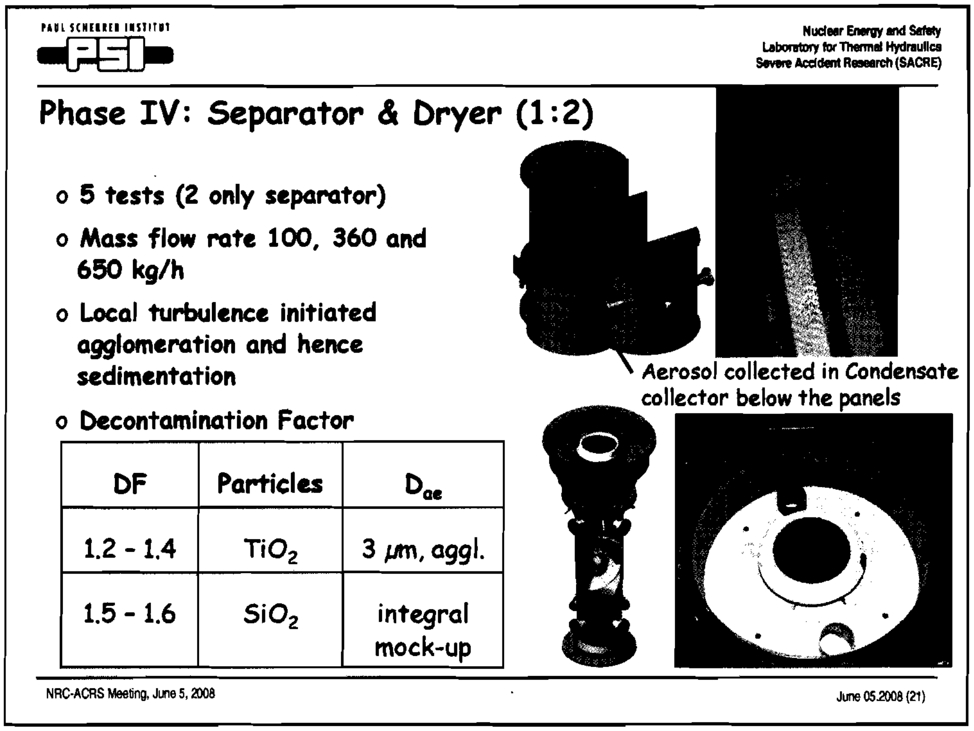


Phase III, Far field stage (1:1)

- o **8 (+2 EU-SGTR) tests**
- o **Mass flow rate 33 & 105 kg/h**
- o **TiO2: deposition everywhere**
- o **Collected mass on certain tubes indicates roughly constant DF per stage**
- o **SiO2: mostly on support plates**
- o **SiO2 (d_{mm}, 3.7 µm) DF: ~1.07**
- o **DF might be higher at higher inlet concentration**


TiO_2 Bundle test


SiO_2 Far field stage test

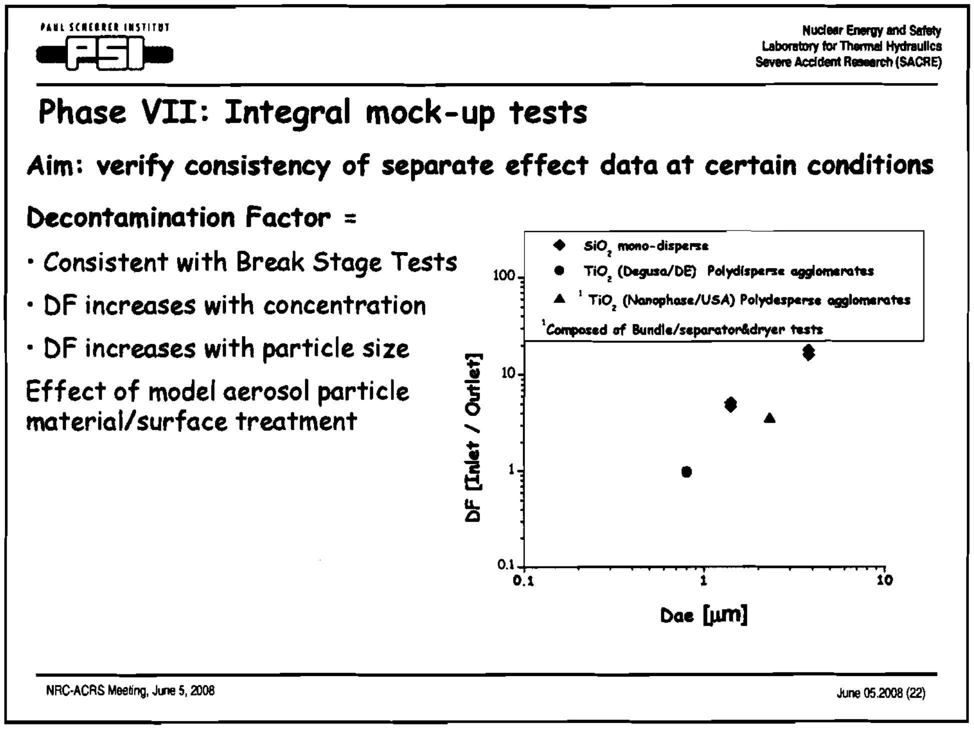


Phase IV: Separator & Dryer (1:2)

- 5 tests (2 only separator)
- Mass flow rate 100, 360 and 650 kg/h
- Local turbulence initiated agglomeration and hence sedimentation
- Decontamination Factor

DF	Particles	D_{ae}
1.2 - 1.4	TiO_2	3 µm, aggl.
1.5 - 1.6	SiO_2	integral mock-up

Aerosol collected in Condensate collector below the panels



Phase VII: Integral mock-up tests

Aim: verify consistency of separate effect data at certain conditions

Decontamination Factor =

- Consistent with Break Stage Tests
- DF increases with concentration
- DF increases with particle size

Effect of model aerosol particle material/surface treatment

Conclusions #1, aerosol tests

- **In-tube retention**
  - Dynamic, depends on particle size and concentration
  - Steam condensation increases DF significantly
  - => **the effect of particle concentration?**
  - => **the effect of bounce/resuspension?**
- **Retention largest in the break stage**
  - Depends on particle size and concentration
  - => **the effect of particle concentration?**
  - => **fish-mouth break leading to higher gas/particle momentum and deeper penetration in Bundle?**
  - => **data with minimized bounce/resuspension needed for modeling**

Conclusions #2, aerosol tests

- **Retention in the far field**
  - => **the effect of particle concentration?**
  - => **Effect of aerosol composition?**
- **Retention in the flooded bundle**
  - => High DF (50 - 2000) with submersion 1.2 - 3.8 m
  - => **retention close to the break (?) with smaller submersion**
- **Retention in Separator & Dryer**
  - => ~ 30-40 % of incoming mass retained independent of Flow Rate
- **Retention in the integral mock-up facility**
  - Dominated by retention in the break stage
  - Consistency of separate effect data demonstrated

Transport/Removal of Activity in Steam Generator

- SGTR concurrent with core damage involves:
  - Major activity in vapour form at SG inlet
  - Rest of activity and inactive material in aerosol form
- Transformation of activity in vapour form by vapour condensation dependent on local temperature
- Removal of some fraction of vapour by condensation on structure surface
- Transport/removal of Rest of vapour of condensed on particles or form new particles dependent on aerosol removal/transport process

ARTIST addresses only aerosol removal/transport process in SG

Motivation for a new SGTR risk assessment methodology

- MELCOR contains models for vapor/aerosol behavior but lacks specific aerosol transport/removal in SG complex structures at relevant thermal-hydraulic conditions
- For risk assessment with many hundred variations to consider uncertainties: MELCOR is too expensive
- A fast running lump parameter model including Monte-Carlo sampling for uncertainties under development
- Preliminary sample analysis demonstrates the strength and provides feasibility of SGTR risk reduction

A new SGTR risk assessment methodology

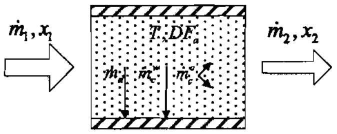
- Lump Parameter Model tracking vapor/aerosol phases in each release path in SG secondary side with:
  - T/H and Vapor/aerosol boundary conditions and uncertainties from SA code predictions
  - Temperature dependent ultimate particle size based on Phébus tests
  - Temperature dependent vapor fractions of released classes including all species from SOPHAEROS code (IRSN/FR) analysis
  - Release path dependent ARTIST DFs (d_p, c)
- Monte-Carlo sampling for all uncertainties
- APET for all SGTR sequences
- Running Model for each APET branches for determination of risk

---

Lump Parameter Model: Key Aspects

- Accounts for aerosol behavior in complex structures of SG at hydrodynamic conditions by use of ARTIST data for each SG retention stage
- Accounts for vapor conversation using temperature dependent vapor fraction data base generated from SOPHAEROS code runs
- Accounts for vapor fraction condensed on structure and converted to particles by user input including its uncertainty
- Accounts for temperature dependent aerosol size determined by measured sizes in hot leg in all Phébus tests with AgInCd
- Neglects other processes playing a secondary role: thermophoresis, diffusiophoresis,...

Lump Parameter Model Description



$$\dot{m}_2 = \frac{1 - x_1(1-\alpha)}{DF_a - x_2(DF_a - \alpha)}\dot{m}_1$$

α: Vapour split fraction on walls/ particles = 0.5 (0.1-0.9)

DFa: ARTIST DF

m: mass flow of release class (I, Cs, ..)

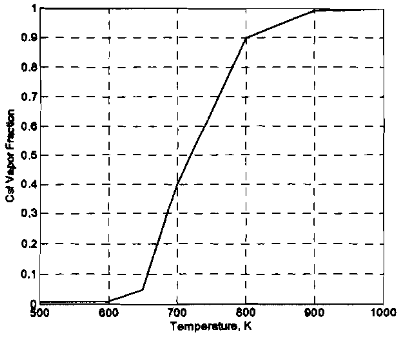
X: vapor fraction of the mass flow

T: Gas temperature

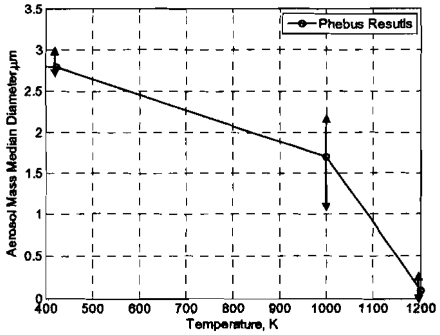
1: donor volume

2: current volume

Lump Parameter Model Data Base (1:3)

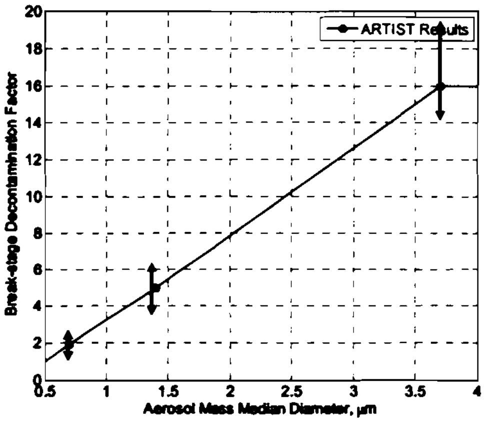


Vapor fraction data base generated from SOPHAEROS code runs



Particle size as measured in all Phébus tests with AgInCd

Lump Parameter Model Data Base (2:3)



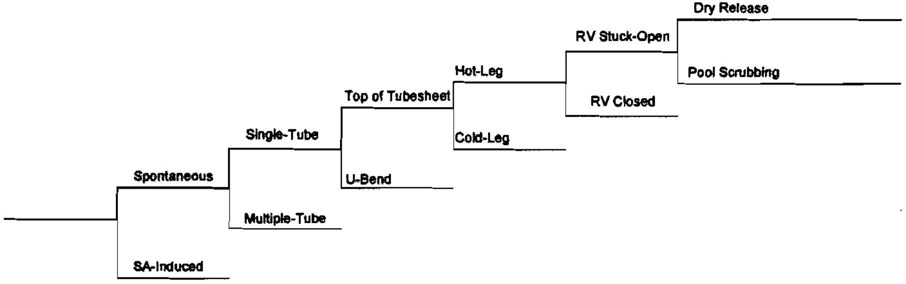
ARTIST Break Stage Particle Size Dependent DF

Lump Parameter Model Data Base (3:3)

Retention Stage	DF	Error Factor	Source
Reactor vessel	1.2 (I), 1.8 (Cs)	1.06 (I), 1.04 (Cs)	Phébus
Primary circuit	1.1 (I), 1.2 (Cs)	1.09 (I), 1.2 (Cs)	Expert judgment
In-tube retention	Time variant	1.5	ARTIST
Break stage	Aerosol-size variant	1.5	ARTIST
Far-field stage I-VII	1.05	1.21	ARTIST
Top of shroud	1.20	1.09	Expert judgment
Separator	1.20	1.06	ARTIST
Recirculation	Model	Model	MELCOR, SR5
Downcomer	1.10	1.05	Expert judgment
Intra-volume	1.10	1.07	Expert judgment
Dryer	1.20	1.09	ARTIST
Dome	1.10	1.05	ARTIST

Multiple SA Code Analyses for Model Uncertainties for the same APET Branch

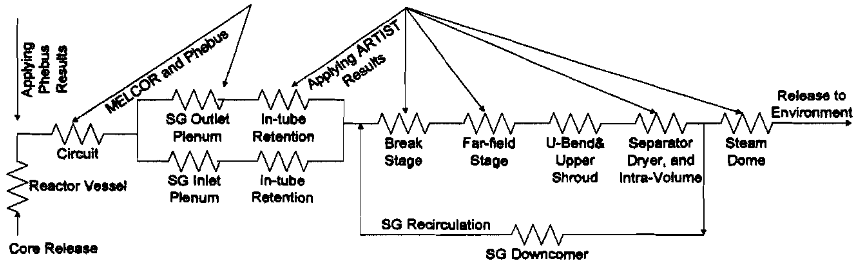
SGTR Frequency	Category	Rupture Size	Rupture Location		Accident Mitigation		
	No SA-Induced SGTR	No Multiple-Tubes	No U-Bend Rupture	No Cold-Leg Rupture	No Reclosed RV	No Refilled SG	[illegible]
Node A	B	C	D	E	F	G	H



SGTR Accident Progression Event Tree

Retention Stages from Core to SG Steam Outlet

- *For each APET sequence, consider a series of retention stages in the fission product release path from the core to the environment*
- *For retention stages of the SG, the lumped parameter model is used*



Multiple SA Code Results: An example

Temperature predictions from MELCOR and SCDAP/RELAP5

Running multiple cases to estimate the temperature distribution

- SGTR sequence from NPP - Beznau PSA L2
  - SRV stuck-open at the affected SG
  - SRV opened manually at the intact SG at core exit temperature>923K
- Calculation stops at lower head failure

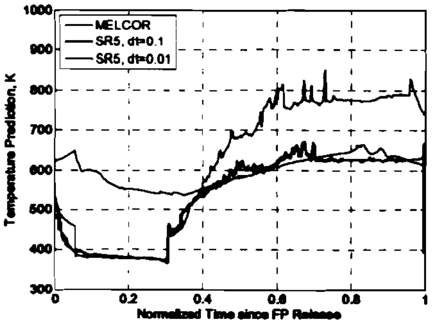
(a) MELCOR

(b) SR5, dt=0.1

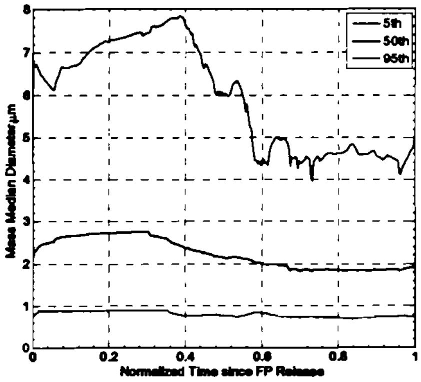
SCDAP/RELAP5, max. time step=0.1s

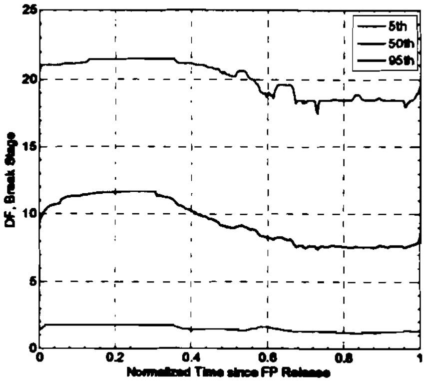
(c) SR5, dt=0.01

SCDAP/RELAP5, max. time step=0.01s

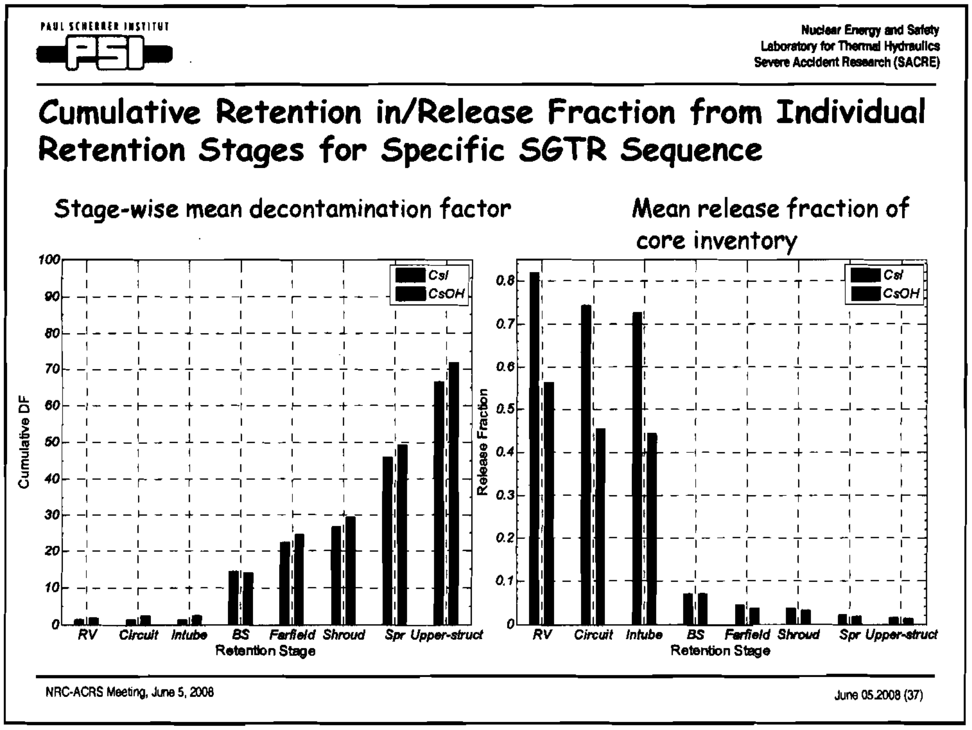


Monte-Carlo Simulation: Examples of 90% confidence interval of Particle Diameter and Decontamination Factor in Break Stage





Presentation not given.



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Cumulative Retention in/Release Fraction from Individual Retention Stages for Specific SGTR Sequence

Stage-wise mean decontamination factor

Mean release fraction of core inventory

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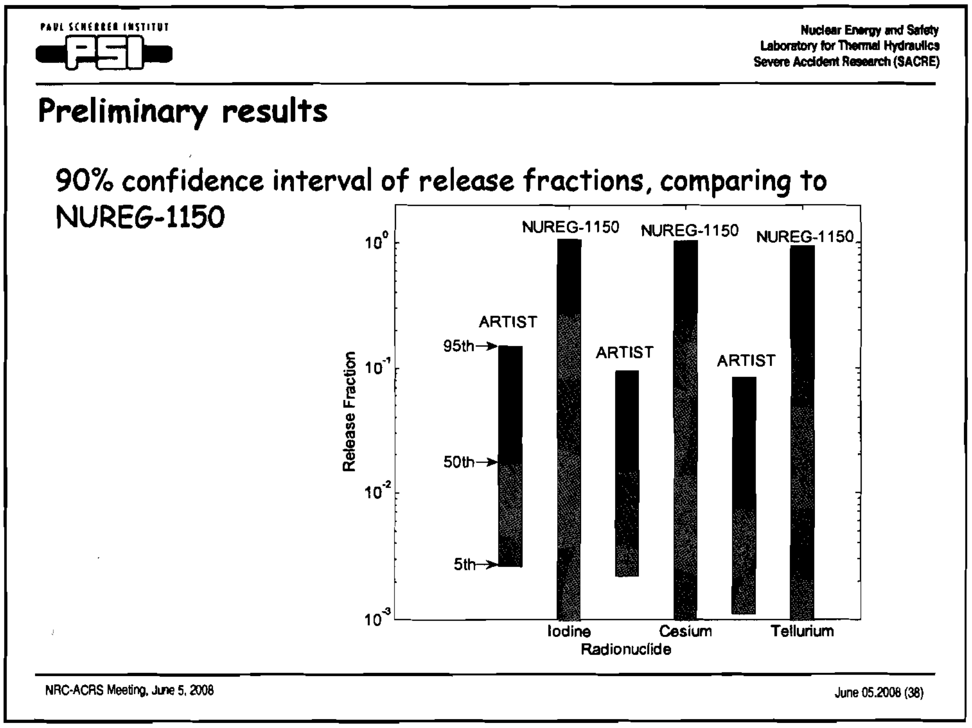
Only slide referred to →



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Preliminary results

90% confidence interval of release fractions, comparing to NUREG-1150

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Assessment of Methodology (1:2)

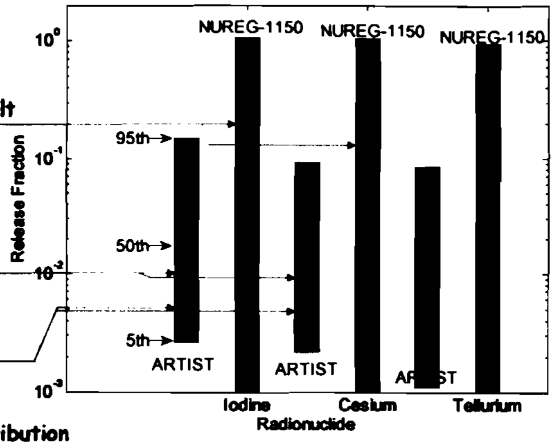
- MELCOR 1.8.6 runs for point estimates of source term
  - use of ARTIST data through „filter function"
  - Superimposing user input „aerosol size" to overwrite MAEROS

- Three MELCOR runs
  - Standard MELCOR 1.8.6 for the same SGTR sequence
  - MELCOR 1.8.6 with ARTIST DFs
  - MELCOR 1.8.6 with ARTIST DFs + PHÉBUS inferred temperature dependent particle size

With MELCOR default vapor and aerosol physics

Assessment of Methodology (1:2)

Comparison of PSI-Risk Model Results to MELCOR Point Value Estimates



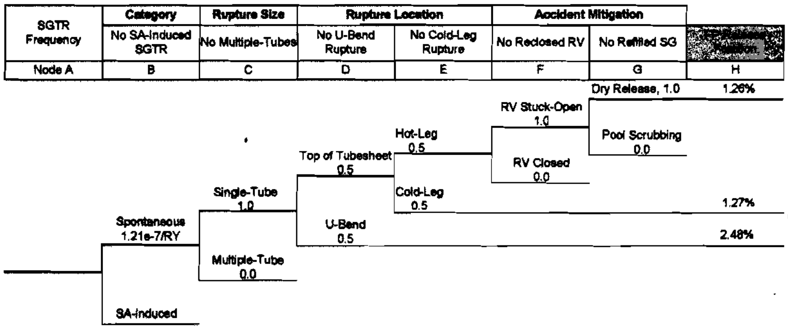
Point estimate of MELCOR default

Point estimate of MELCOR using MAEROS with incorporation of ARTIST DFs

Point estimate of MELCOR using PHEBUS[1] with incorporation of ARTIST DFs

[1] superimposing particle size distribution

APET: branching fractions



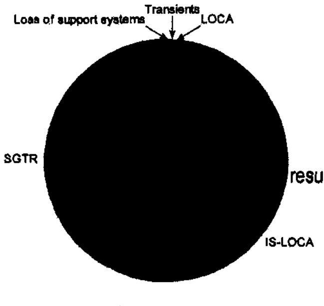
Case presented

Other cases not presented

SGTR Accident Progression Event Tree

Preliminary Risk Profile of NPP-Beznau Spontaneous SGTR

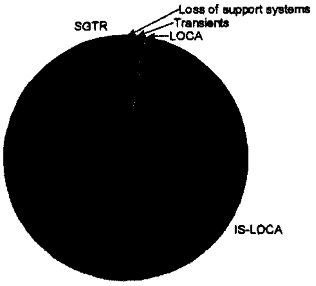
Comparison of the SGTR (without SG Reflooding) Risk significance to other internal initiating events for the Beznau NPP



NPP Beznau: PSA L2 BERA: 2002

SGTR risk reduction

resulting from using the ARTIST data



Conclusions

- **Methodology consistent with Point values from MELCOR**
- **Further development for inclusion of other dependencies and their uncertainties (e.g., DF (dp, C)**
- **Generic model requires user to input from plant specific SA analysis**
- **APET to be revised with plant specific information (frequencies, split fractions)**

Final Remarks

- PSI data supported by additional data from CIEMAT (Spain) for break stage retention and from VTT (Finland) for in-tube deposition/resuspension, both at low flows
- CFD Simulations of flow^1 and particles^2 by CFD (FLUENT) by Ringhals, AVN^1, CIEMAT^1, JNES ^{1,2} and NRC^{1,2} (Sandia)
- Model development for aerosol removal in flooded bundle (IRSN) and in break stage (CIEMAT)
- 4 PhDs (de-agglomeration, aerosol motion through DNS+LES, bubble hydrodynamics in bundle) at PSI
- 3 PhDs (removal in far field, break stage hydrodynamics, aerosols) at UPM and CIEMAT
- 1 PhD (particle motion in SG pipe) at Sandia
- 1 masters (flow fields by CFD in Separator) at AVN
    - ➢with involvement of 7 Universities

PSI thanks for all supporting and participating organizations in ARTIST

Phases V and VI: Flooded Bundle and Droplet Retention in Separator & Dryer

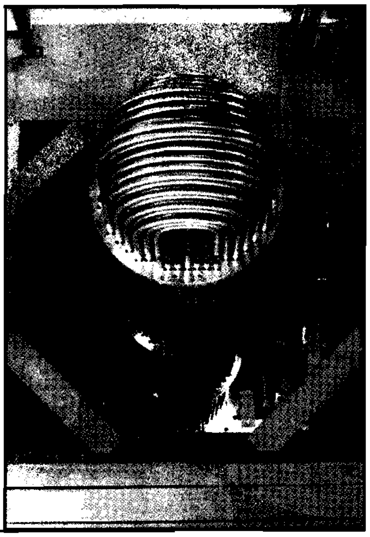
NRC does not participate in ARTIST Project Phases V and VI, however, the following information is introduced for those in ACRS who have interest in the Aerosol Scrubbing in Bundle Environment from High Jet Flows and Dissolved Activity (Iodine, mostly) Retention/Release by Droplets during the initiation of aSGTR event

---

Phase V: retention in the flooded bundle (1:2)

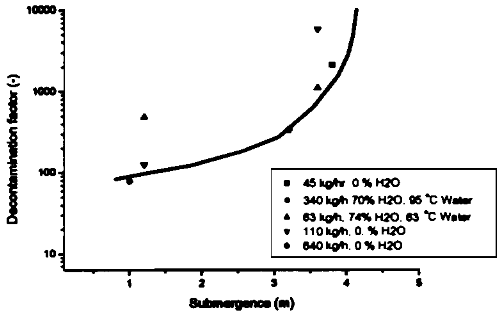
- o **2 tests (+3 EU-SGTR)**
- o **Decontamination Factor**
- o Determined for relatively large submersion

DF	flow rate	submersion
2 100	45 kg/h	3.8 m
335	640 kg/h	3.2 m



Phase V: retention in the flooded bundle (2:2)

- Very high DF due to bundle-hydrodynamic interactions, especially at the break; models not able to reproduce DF
- Aerosol removal in hot pools without bundle: ~ DF 20 (PSI - POSEIDON, 1991- 1996)



tests	Main features	Submergence m	Experimental DF	IRSN Model DF
A62	Steam, hot, medium flow rate	1.3	80-100	362
A63	NC, cold, low flow rate	1.2 2.3 3.6	124 1251 6730	37 64 66
E04	NC, cold, low flow rate	3.60	2887	46
E06	NC, cold, high flow rate	3.20	271-485	67

Iodine Source Term during Steam Generator Tube Rupture Initiated Design Basis Accidents: Introduction

- **Spontaneous or initiated Steam Generator Tube Rupture**

  => activity release until the operators can reduce the RCS pressure to the secondary side level

  => activity release at least 30-40 minutes (so-called "grace period")
- **SGTR event is a design basis event**
- **The amount of activity release controlled by:**

  a) amount of dissolved activity in the primary system (leaking rods, iodine spiking (reactor trip) and pressure change)

  b) the submergence of the leak; single or multiple tube ruptures; total break flow

  c) pH and iodine chemistry in the secondary side

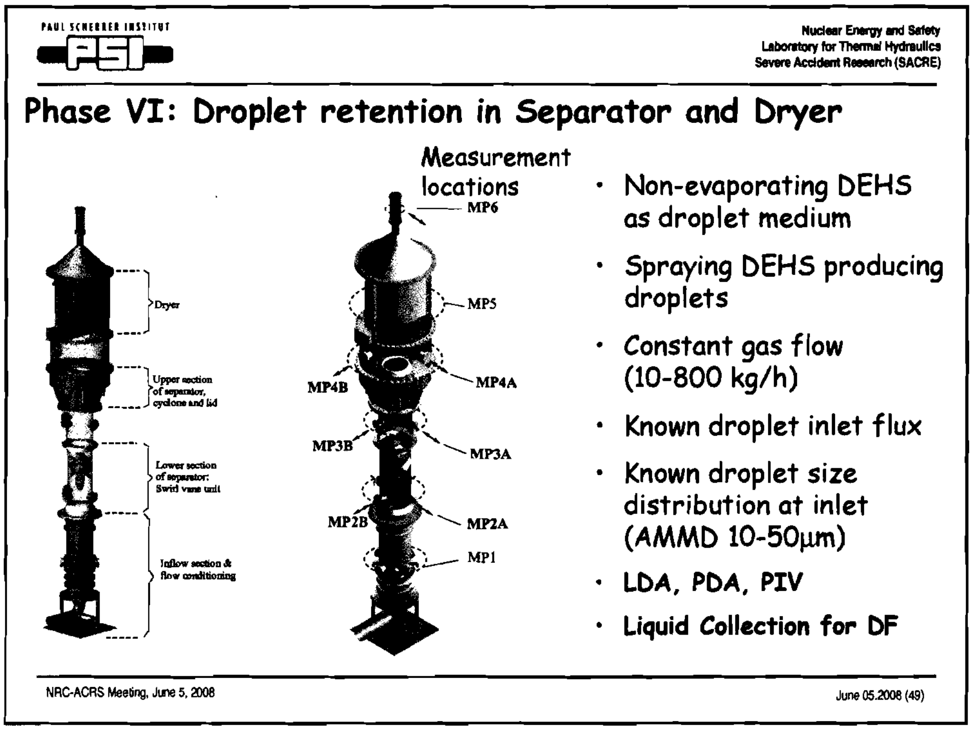
  d) iodine mass transfer from the boiling pool

  e) **The break at the tube bend**

  <= 80-85 % of primary water in droplet form as a result of flashing

  => efficiency of separator and dryer to retain droplets

  ➔ ARTIST - Phase VI



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Phase VI: Droplet retention in Separator and Dryer

- Non-evaporating DEHS as droplet medium
- Spraying DEHS producing droplets
- Constant gas flow (10-800 kg/h)
- Known droplet inlet flux
- Known droplet size distribution at inlet (AMMD 10-50μm)
- **LDA, PDA, PIV**
- **Liquid Collection for DF**

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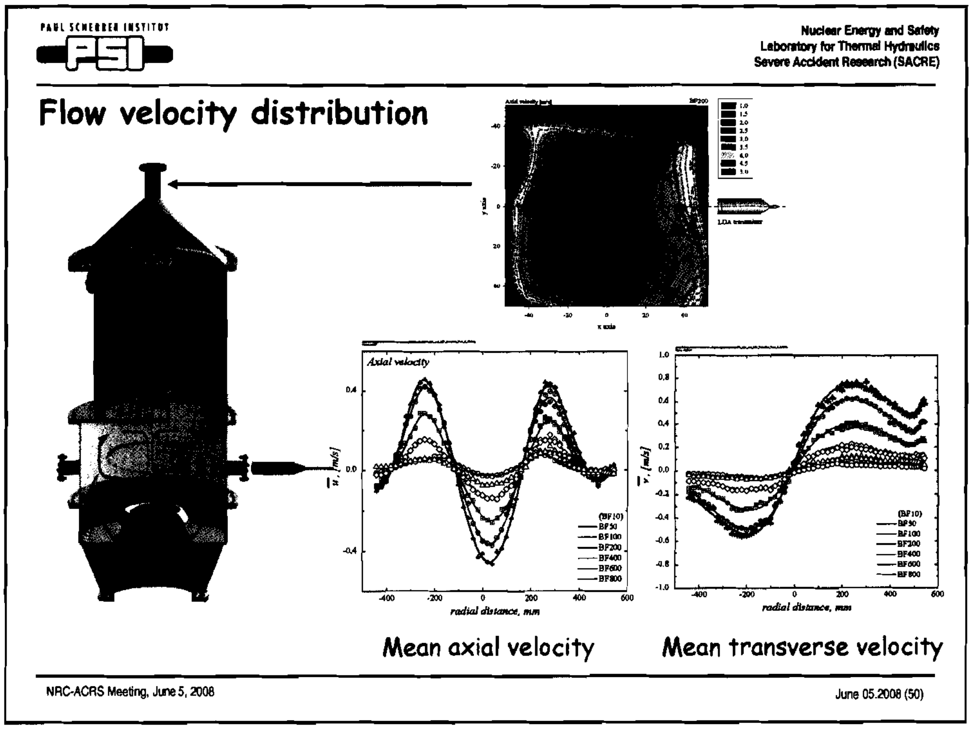

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Flow velocity distribution

Mean axial velocity

Mean transverse velocity

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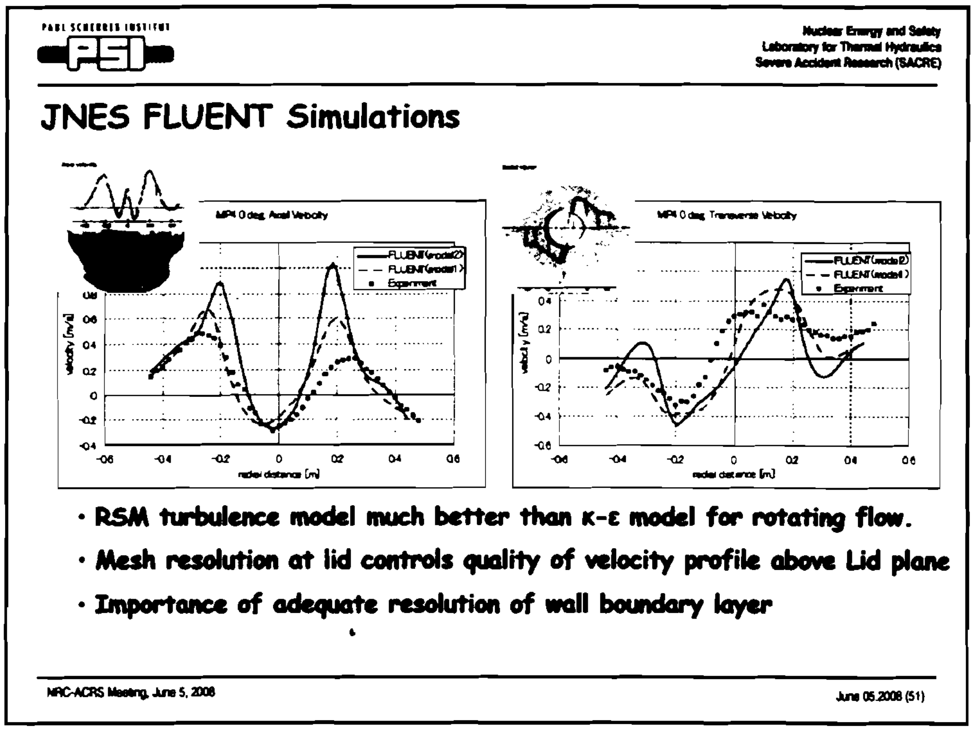

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JNES FLUENT Simulations

- **RSM turbulence model much better than κ-ε model for rotating flow.**
- **Mesh resolution at lid controls quality of velocity profile above Lid plane**
- **Importance of adequate resolution of wall boundary layer**

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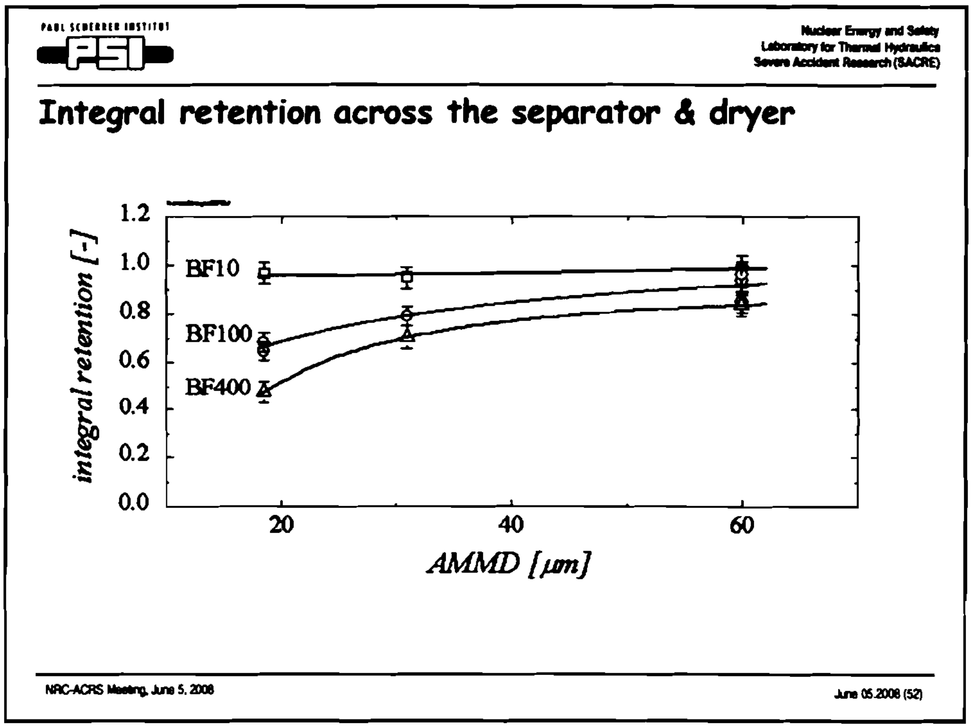

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Integral retention across the separator & dryer

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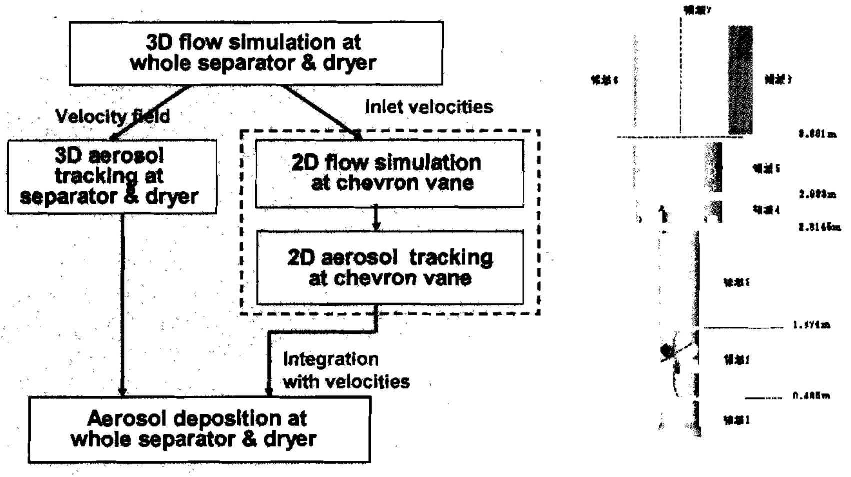
Particle Decontamination by FLUENT with PSI discrete-particle tracking model (JNES)



3D flow simulation at whole separator & dryer

Velocity field

Inlet velocities

3D aerosol tracking at separator & dryer

2D flow simulation at chevron vane

2D aerosol tracking at chevron vane

Integration with velocities

Aerosol deposition at whole separator & dryer

Particle Decontamination by FLUENT with PSI discrete-particle tracking model (JNES)

DF (300kg/h)

	1 μm	3 μm	10 μm
Separator	1.25	1.32	1.35
Dryer	1.09	1.14	1.25
Total	1.36	1.51	1.68

- Capturing hydrodynamic behavior is crucial prerequisite for aerosol behavior
- PSI discrete-particle tracing considers particle turbulence based on DNS simulations
- JNES predicted Overall retention is in agreement with Phase IV test results

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Standardization of Operational Event Risk Assessments

Marty Stutzke
Senior Technical Advisor for PRA Technologies
Division of Risk Analysis
Office of Nuclear Regulatory Research

June 4, 2008

Presentation Outline

- Purpose
- Background
- Concepts of operational event risk assessment
- Implementation of standardization tasks
- Ongoing and future work
- Conclusions

Purpose

- To describe the activities undertaken by RES and NRR to standardize the risk assessment of operational events.
- To provide background to findings in draft NUREG-1635, Vol. 8, “Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program,” Chapter 10, “Operational Experience.”
- To summarize the status of completed and ongoing RES activities in support of the standardization of operational event risk assessments.

Background

- In 2004, the staff initiated the Risk Assessment Standardization Project (RASP) as a collaborative effort between NRR, RES, and regional Senior Reactor Analysts (SRAs).
- The purpose of RASP is to provide consistent methods for risk analysis of conditions in the ASP and SDP Phase 3 programs and the risk analysis of events/conditions in the ASP and MD 8.3 programs, while recognizing differences in purpose among the programs.

Risk Assessment of Operational Events at NRC

- ***Significance Determination Process (SDP):*** Risk analysis of inspection findings (e.g., conditions with performance deficiencies) to determine the safety significance of inspection findings. (Regions, NRR)
- ***NRC Incident Investigation Program (MD 8.3):*** Risk analysis of initiating events and conditions to determine the appropriate level of reactive inspection in response to a significant event. (Regions, NRR)
- ***Accident Sequence Precursor (ASP) Program:*** Risk analysis of initiating events and conditions to identify significant precursors, adverse trends, and insights. (RES)

Event Risk Assessment – Introduction

- The aim of event risk assessment is to identify what else could have happened in an incident, which did not necessarily happen during the incident, and that would lead to core damage.
- The event risk assessment is *future-oriented*
  - What is probability that a similar event, occurring in the future, would lead to core damage?

Event Risk Assessment – Basic Concepts

- The figures of merit are conditional core damage probability (CCDP) for initiating events and change in core damage probability (Δ CDP) for degraded conditions.
  - The CCDP given the event and the nominal or adjusted failure probabilities of the components and operator actions that did not fail, yields a measure of how close we came to core damage.
- The "failure memory concept"
  - All *failures* observed in the event are modeled as failures in the risk analysis:
    - Basic events representing failed components and operator actions are modeled as failed (e.g., with TRUE house events).
  - System and operator action *successes* receive a different treatment:
    - Basic events representing successes are ignored (i.e., successes are not set to FALSE house events).
    - These basic events remain at their nominal failure probability, or adjusted to represent complications observed during the event.

Standardization Approach

- Document methods and guides for event risk analysis
  - Internal event analysis
  - External event analysis, including internal fire and flood events
  - Low-power/shutdown (LP/SD) event analysis
  - Large early release frequency (LERF) calculation
- Improve SPAR model fidelity
  - Enhance Rev. 3 internal events SPAR models to better reflect the risk of the as-built, as-operated plant
  - SPAR models for external events, shutdown events, and LERF/Level 2
- Enhance analysis methods; provide technical support

User Need Tasks for RES

- Task 1: Develop guides for the analysis of internal events during power operations.
- Task 2: Develop new methods and guides for the analysis of the following events:
  - External events, including internal fire and flood
  - Internal events during low-power and shutdown (LP/SD) operations
  - Calculation of large early release frequency (LERF) for containment-related events
- Task 3: Make enhancements to SPAR models and SAPHIRE/GEM code
- Task 4: Provide ongoing technical support.

Tasks 1 & 2 – Guides for Event Risk Analysis

- RASP handbook (Rev. 1) issued January 2008 (publically available):
  - Volume 1, Internal Events (ML080070303)
  - Volume 2, External Events (ML080300179)
  - Volume 3, SPAR Model Reviews (ML080300182)
- Volumes 1 and 2 based on existing methods used in previous SDP and ASP analyses; Vol. 3 based in part on PRA Review Guide (NUREG/CR-3485) and PRA Standard (ASME RA-Sb-2005).
- Inspection Manual Chapter (IMC) 0609, “Significance Determination Process,” references use of handbook.
- Internal reviews by NRC and contractor staffs; Rev. 0 of Vols. 1 and 2 been in trial use for 2 to 3 years.

Task 3 – SPAR Model Development

- Internal events models:
  - Detailed cut-set-level reviews against most licensee's PRAs
  - Updates to station blackout/loss of offsite power models
  - Updates to SPAR model parameters based on NUREG/CR-6928[1]
  - Updates to SPAR model QA plan for Rev. 3 SPAR models
  - Other enhancements based on staff and licensee feedback
- External events models: 15 integrated Rev. 3 SPAR models
- Shutdown events models: 5 integrated Rev. 3 SPAR models
- LERF/Level II models: 2 preliminary Level II SPAR models

[1] NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," February 2007 (http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6928/)

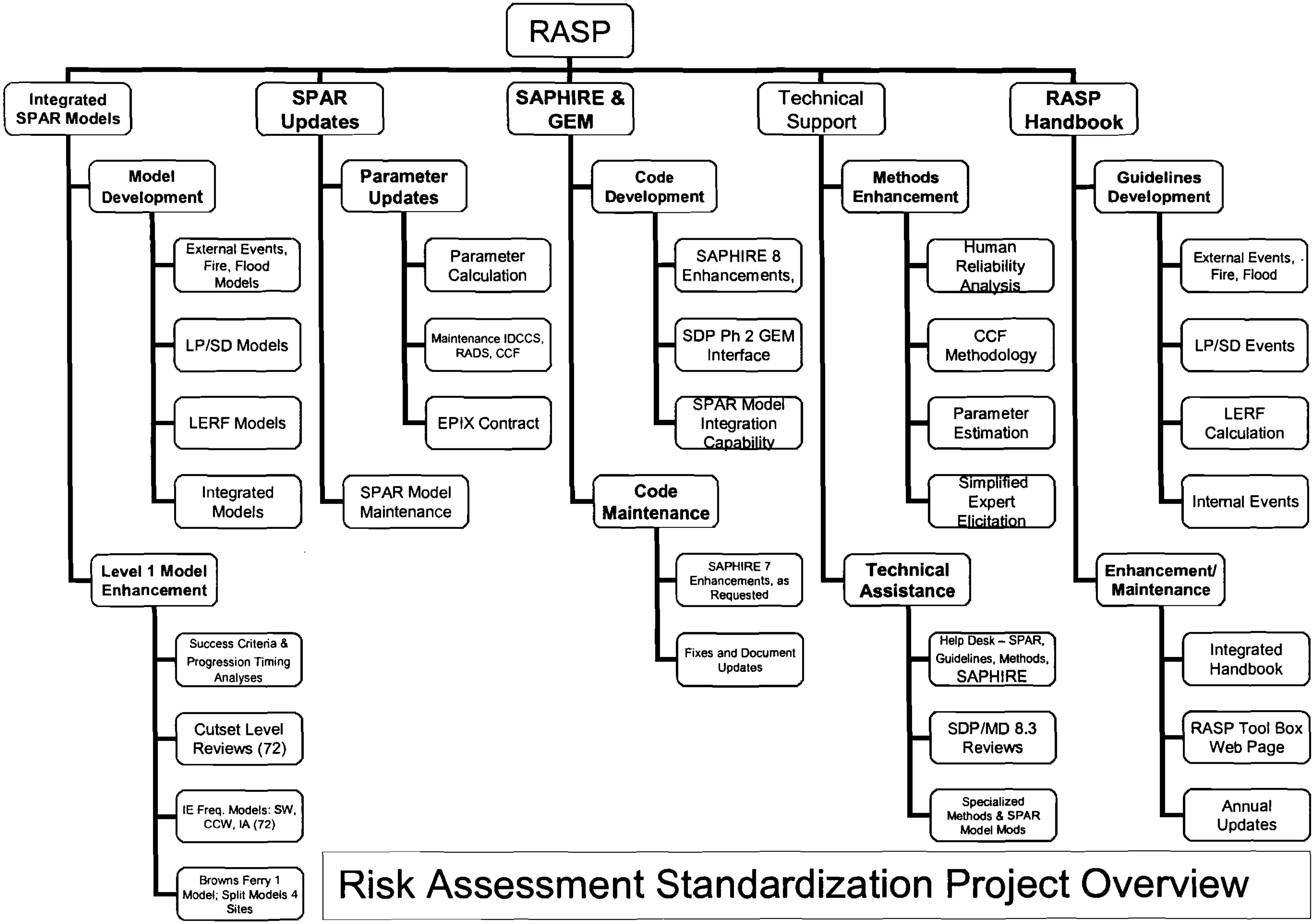
Task 3 – SAPHIRE and GEM

- A new version of SAPHIRE code being developed to meet requirements for:
  - New user interface for conducting SDP Phase 2 assessments
  - Improved user interface for conducting SDP Phase 3 and ASP analyses
  - Improved features and capabilities for SPAR model development and use (e.g., LERF modeling approach, support integrated models)
  - New modeling and calculation methods (e.g., common-cause failure analysis, phase mission time analysis)
- Beta testing and peer review to be performed during 2008 and 2009 to support release of SAPHIRE Version 8 by end of 2009.

Task 4 – RES Technical Support

- Technical support provided to NRR analysts and Senior Reactor Analysts on methods, models, and analysis.
- Training provided at SRA counterpart meetings.
- Areas of support for event risk analysis include:
  - Common-cause failure modeling, parameter estimation
  - HRA and simplified expert elicitation applications
  - Uncertainty and sensitivity analyses
  - Internal event analysis guidance and SPAR model application
  - External event analysis guidance and SPAR model application
  - LP/SD event analysis guidance and SPAR model application
  - LERF calculation guidance
  - SAPHIRE/GEM code
- RASP Tool Box Web page developed for analysts.

Risk Assessment Standardization Project Overview



- RASP
  - Integrated SPAR Models
    - Model Development
      - External Events, Fire, Flood Models
      - LP/SD Models
      - LERF Models
      - Integrated Models
    - Level 1 Model Enhancement
      - Success Criteria & Progression Timing Analyses
      - Cutset Level Reviews (72)
      - IE Freq. Models: SW, CCW, IA (72)
      - Browns Ferry 1 Model; Split Models 4 Sites
  - SPAR Updates
    - Parameter Updates
      - Parameter Calculation
      - Maintenance IDCCS, RADS, CCF
      - EPIX Contract
    - SPAR Model Maintenance
  - SAPHIRE & GEM
    - Code Development
      - SAPHIRE 8 Enhancements,
      - SDP Ph 2 GEM Interface
      - SPAR Model Integration Capability
    - Code Maintenance
      - SAPHIRE 7 Enhancements, as Requested
      - Fixes and Document Updates
  - Technical Support
    - Methods Enhancement
      - Human Reliability Analysis
      - CCF Methodology
      - Parameter Estimation
      - Simplified Expert Elicitation
    - Technical Assistance
      - Help Desk – SPAR, Guidelines, Methods, SAPHIRE
      - SDP/MD 8.3 Reviews
      - Specialized Methods & SPAR Model Mods
  - RASP Handbook
    - Guidelines Development
      - External Events, Fire, Flood
      - LP/SD Events
      - LERF Calculation
      - Internal Events
    - Enhancement/ Maintenance
      - Integrated Handbook
      - RASP Tool Box Web Page
      - Annual Updates

Ongoing and Future Work – Methods and Guides

- RASP Handbook
  - Complete Volume 1: Guides for CCF modeling, parameter estimation and updates, uncertainty/sensitivity analysis, HRA, simplified expert elicitation, convolution analysis).
  - Revise Volumes 1, 2, and 3 based on user feedback.
  - Develop new volume for analysis of LP/SD events.
  - Develop new volume for LERF analysis of containment events.
- Technical support
  - Enhance methods
    - CCF methodology for event assessment (draft NUREG/CR)
    - HRA (based on results of international HRA benchmarking project)
    - Update pipe break LOCA frequencies (draft NUREG/CR)
  - Provide training support.
  - Provide on-call SDP analysis assistance.

Ongoing and Future Work – SPAR Models

- Internal events SPAR model enhancements
  - Success criteria re-evaluation of key sequences based on thermal hydraulic analyses.
  - Work with industry to resolve key technical issues affecting SPAR and licensee PRA models (through NRC/EPRI Memorandum of Understanding).
  - Complete detailed cut-set-level reviews for 4 remaining models.
- Shutdown SPAR model development
  - Continue model development for shutdown events.
- SAPHIRE/GEM Version 8 development
  - Complete beta testing.

Conclusions

- RASP handbook widely in use by risk analysts and SRAs in the risk analysis of operational events in NRC programs:
  - Conditions in the ASP and SDP Phase 3 programs
  - Initiating events and conditions in the ASP and MD 8.3 programs
- ASP Program changed to eliminate duplicative analysis of SDP inspection findings.
- Communications and documented guidance improved consistency among analysts and enhanced knowledge transfer.
- Enhanced SPAR models better reflect the risk of the as-built, as-operated plant.

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Backup Slides

Past Briefings to the ACRS (Full and Subcommittees) on RES Risk Activities

- SPAR model development (10/10/2003)
  - Internal events (9/9/2005, 9/15/2005, 11/17/2005)
  - External events, including internal fire and flooding (11/18/2005)
  - shutdown event (11/11/2002, 10/10/2003, 11/18/2005)
  - Large early release frequency (LERF) (11/18/2005)
- SAPHIRE development (1/25/2002, 10/10/2003)
- Risk methods and databases
  - SPAR-H human reliability analysis method (10/09/2003, 12/15/2005, 3/22/2007)
  - Common-cause failure method, RADS/EPIX (12/15/1999, 04/6/2000)
  - Uncertainty (10/10/2003, 11/16/2004, 12/19/2007)
- Accident Sequence Precursor (ASP) Program (12/15/1999, 3/10/2006)

NRR User Need Requests

- "User Need Request for Support in the Development of Standard Procedures and Methods for Risk Assessments of Inspection Findings and Reactor Incidents," J. Dyer Memo to A. Thadani, February 17, 2004 (NRR-2004-005)
  - Task 1: Guides for risk analysis of internal events
  - Task 2: Guides for risk analysis of external events, LP/SD, and LERF
  - Task 3: SPAR model and SAPHIRE/GEM enhancements
  - Task 4: Technical support (methods, models, SDP analyses, handbook updates)
- "Supplement to User Need Request for Support in the Development of Standard Procedures and Methods for Risk Assessments of Inspection Findings and Reactor Incidents," Dyer Memo to B. Sheron, June 22, 2006 (NRR-2004-005)
  - Initiating event fault trees for cooling water systems (e.g. service water)
  - Revised models of success criteria for specific sequences using thermal hydraulic analyses

NRC/EPRI MOU

- **SPAR model/industry PRA key technical issues:**
  - Support system initiating event analysis
  - Treatment of loss of offsite power
  - Standard guidance for event tree development
  - Treatment of injection following containment failure (BWRs)
  - Treatment of containment sump recirculation during small and very small loss of coolant accident
  - Human reliability analysis dependencies and recovery modeling issues
- **Other NRC/industry technical issues:**
  - Treatment of uncertainty in risk analyses
  - Aggregation of risk metrics
  - Human reliability analysis
  - Digital instrumentation & control risk methods
  - Advanced reactor PRA methods

RASP Tool Box Web Page

- http://www.internal.nrc.gov/RES/RASP/index.html *(Internal to NRC)*
- Provide web links to tools and access to references for Senior Reactor Analysts and risk analysts, e.g.,
  - RASP handbook volumes
  - Handbook references
  - SPAR models
  - SAPHIRE/GEM codes and manuals
  - Parameter estimation references (NUREG/CRs)
  - Databases and calculators (ASP, CCF, EPIX, LERs, RADS)
  - Plant information
  - PRA training manuals
  - PRA related references (NUREG/CRs)
- RASP Handbook kept current in the Tool Box.

Point-of-Contacts

- Accident Sequence Precursor Program: Chris Hunter (RES/DRA)
- RASP Handbooks
  - Vol. 1, Internal Event Analysis: See-Meng Wong (NRR/DRA), Don Marksberry (RES/DRA), Paul Bonnett (NRR/DIRS)
  - Vol. 2, External Event Analysis: Selim Sancaktar (RES/DRA)
  - Vol. 3, SPAR Model Reviews: Pete Appignani (RES/DRA)
- Risk Analysis Methods for Event Risk Analysis
  - CCF, parameter estimation, and RADS and CCF calculators: Jack Foster (RES/DRA)
  - SPAR-H HRA enhancements: Pete Appignani (RES/DRA)
  - Uncertainty/sensitivity analysis, simplified expert elicitation: Gary DeMoss (RES/DRA)
- Risk Databases (EPIX, LER, RADS, CCF): Bennett Brady (RES/DRA)
- SAPHIRE/SDP User Interface: Dan O'Neal (RES/DRA)
- Significant Determination Process: Paul Bonnett (NRR/DIRS)
- SPAR Models: Pete Appignani (RES/DRA)
- SPAR Model Success Criteria Re-Evaluation: Rick Sherry (RES/DRA)

6/03/2008 3:00 p

Abbreviations

ASP	accident sequence precursor
CCDP	conditional core damage probability
CCF	common-cause failure
EPIX	Equipment Performance and Information Exchange System
EPRI	Electric Power Research Institute
GEM	Graphical Evaluation Module
HRA	human reliability analysis
LER	Licensee Event Report
LERF	large early release frequency
LP/SD	Low-power/shutdown
MD	Management Directive
NRR	Office of Nuclear Reactor Regulation
NRR/DIRS	Division of Inspection and Regional Support, NRR
NRR/DRA	Division of Risk Assessment, NRR
PRA	probabilistic risk assessment
QA	quality assurance
RADS	Reliability and Availability Data System
RASP	Risk Assessment Standardization Project
RES	Office of Nuclear Regulatory Research
RES/DRA	Division of Risk Analysis, RES
SAPHIRE	System Analysis Programs for Hands-on Integrated Reliability Evaluations
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk (model)
SRA	Senior Reactor Analyst

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

U.S. EPR OVERVIEW PRESENTATION
553rd ACRS MEETING

JUNE 4, 2008

Getachew Tesfaye

EPR Project Background

- Three years of pre-application activities: December 2, 2004 to December 11, 2007
  - Several public meetings were held to familiarize the NRC staff with the EPR design
  - 15 topical reports and 4 technical reports were submitted in preparation for the design certification application

AREVA EPR Design Certification

- Application submitted: December 11, 2007
- Accepted for review: February 25, 2008
- Review schedule issued: March 26, 2008
- Currently in Phase 1 review
- Review Milestones:
  - Phase 1, PSER and RAI
    - Target date for completion 1/29/2009
  - Phase 2, SER with open items
    - Target date for completion 11/20/2009
  - Phase 3, ACRS review of SER with open items
    - Target date for completion 3/05/2010
  - Phase 4, Advanced SER with no open items
    - Target date for completion 11/2010
  - Phase 5, ACRS review of advanced SER with no open items
    - Target date for completion 03/2011
  - Phase 6, Final SER with no open items
    - Target date for completion 05/2011

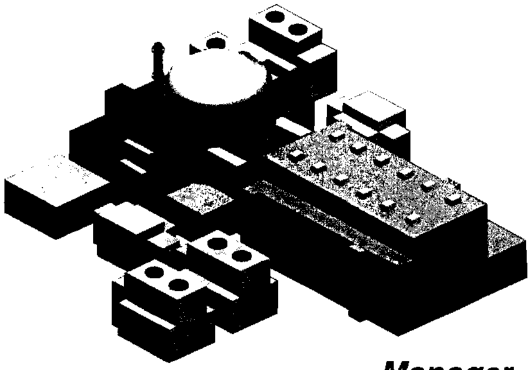
COL Applications Referencing EPR

- R-COLA
  - Calvert Cliffs COL Application
  - Part I – Environmental review
    - Submitted July 13, 2007
    - Docketed January 25, 2008
    - Currently in Phase 1 review
  - Part II – Balance of the COL Application
    - Submitted March 14, 2008
    - Docketed June 3, 2008
    - Currently review schedule is being developed
- S-COLA planned submittals
  - AmerenUE, Callaway Plant Unit 2: August 4, 2008
  - PPL, Bell Bend: September 2008
  - UniStar/Constellation, Nine Mile Point: September 2008
  - UniStar/Amarillo Power, site TBD: 4Q 2009
  - Alternate Energy Holdings, Bruneau, ID: TBD

ACRS Meeting:

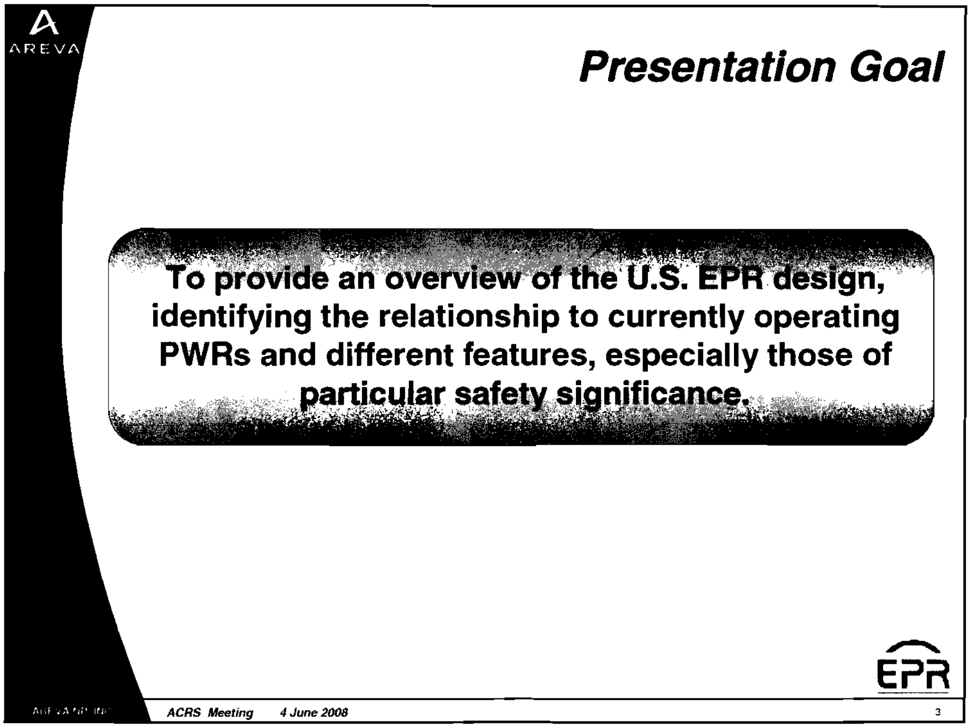
U.S. EPR Design Overview

June 4, 2008



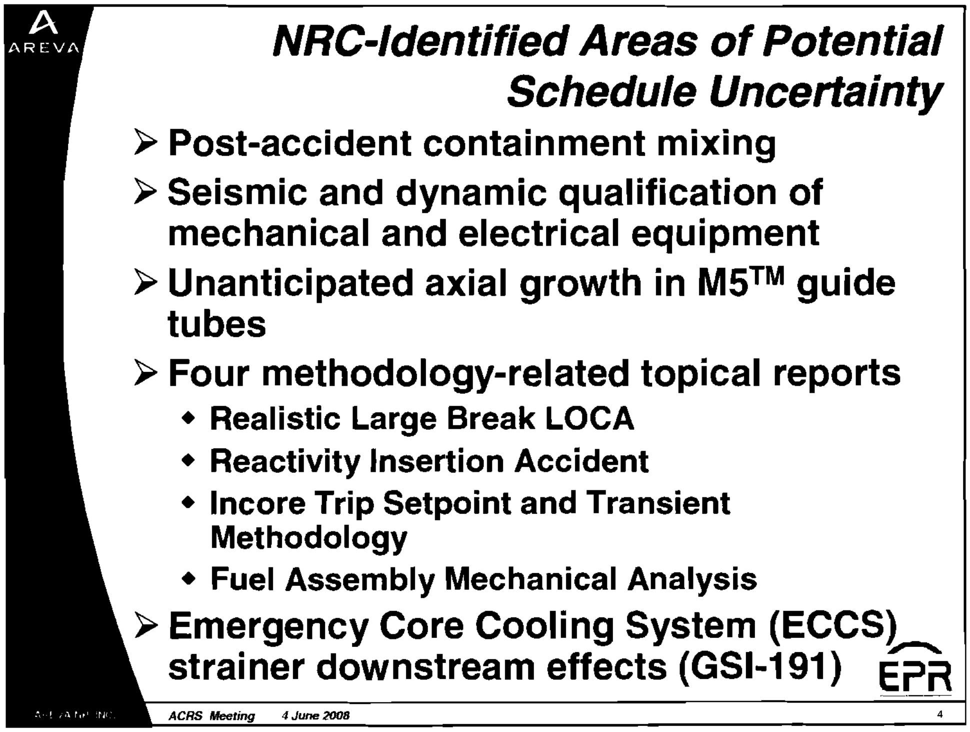
Introduction

Sandra M. Sloan
Manager, Regulatory Affairs
New Plants Deployment



Presentation Goal

To provide an overview of the U.S. EPR design, identifying the relationship to currently operating PWRs and different features, especially those of particular safety significance.



NRC-Identified Areas of Potential Schedule Uncertainty

- Post-accident containment mixing
- Seismic and dynamic qualification of mechanical and electrical equipment
- Unanticipated axial growth in M5™ guide tubes
- Four methodology-related topical reports
  - Realistic Large Break LOCA
  - Reactivity Insertion Accident
  - Incore Trip Setpoint and Transient Methodology
  - Fuel Assembly Mechanical Analysis
- Emergency Core Cooling System (ECCS) strainer downstream effects (GSI-191)

AREVA

Presentation Topic Areas

- ➢ General design objectives
- ➢ Plant layout
- ➢ Safety systems
- ➢ Core design
- ➢ Instrumentation and controls
- ➢ Severe accident mitigation
- ➢ SGTR and SBLOCA mitigation
- ➢ Probabilistic risk assessment
- ➢ Operating experience feedback

EPR

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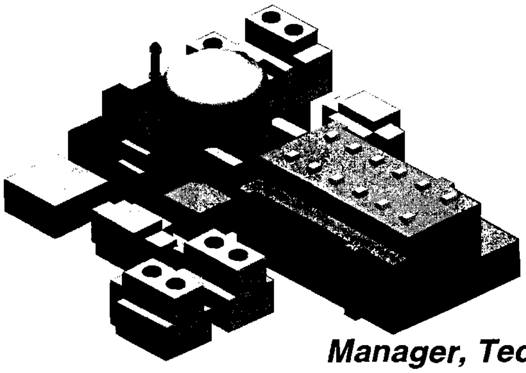
U.S. EPR Design Overview



Marty Parece
Chief Engineer
Manager, Technology Integration
New Plants Deployment

EPR

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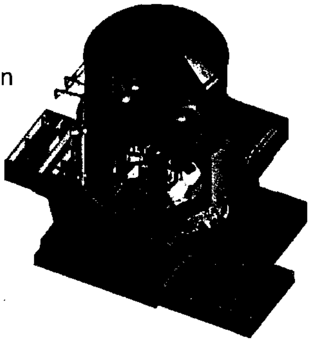
EPR Development Objectives

- **Evolutionary design** based on existing PWR construction experience, R&D, operating experience and "lessons learned"
- **Safer**
  - Reduce occupational exposure and LLW
  - Increase design margins
  - Increase redundancy & physical separation of safety trains
  - Reduce core damage frequency (CDF)
  - Accommodate severe accidents and external hazards with no long-term local population effect
- **Improved Operations**
  - Reduce generation cost by at least 10%
  - Simplify operations and maintenance
  - 60-year design life



EPR

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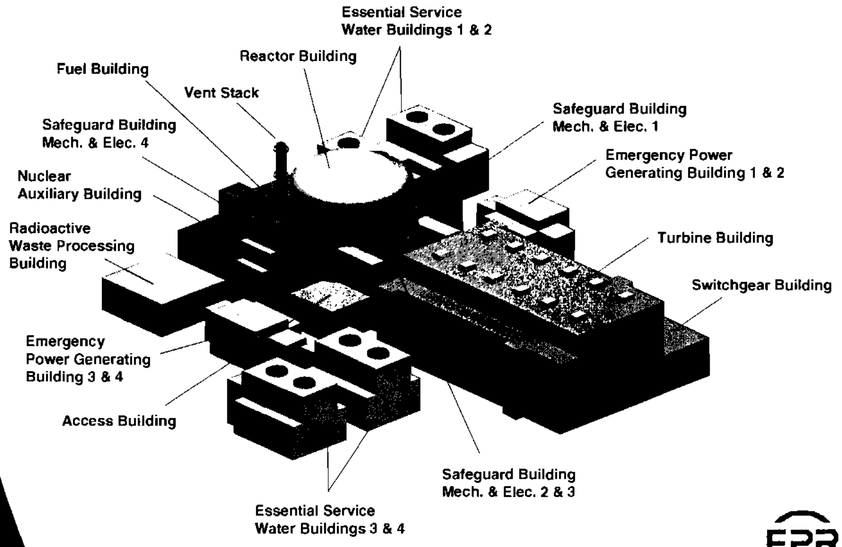
Major Design Features

- Nuclear Island
  - *Proven Four-Loop RCS Design*
  - *Four-Train Safety Systems*
  - *Containment & Shield Bldg*
  - *In-Containment Borated Water Storage*
  - *Severe Accident Mitigation*
  - *Separate Safety Buildings*
  - *Advanced Control Room*
- Electrical
  - *Island Mode Operation*
  - *Four Emergency D/Gs*
  - *Two Smaller, Diverse SBO D/Gs*
- Site Characteristics
  - *Airplane Crash Protection (military and commercial)*
  - *Explosion Pressure Wave*

Reflects full benefit of operating experience and 21st century requirements.

EPR

The U.S. EPR



Essential Service Water Buildings 1 & 2

Reactor Building

Fuel Building

Vent Stack

Safeguard Building Mech. & Elec. 4

Safeguard Building Mech. & Elec. 1

Emergency Power Generating Building 1 & 2

Nuclear Auxiliary Building

Radioactive Waste Processing Building

Turbine Building

Switchgear Building

Emergency Power Generating Building 3 & 4

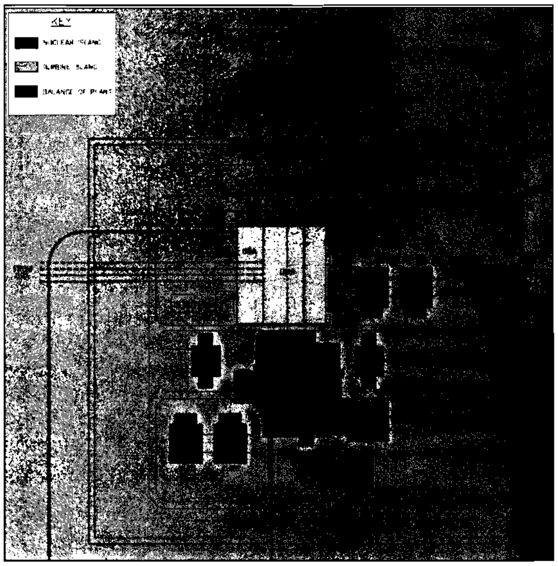
Access Building

Safeguard Building Mech. & Elec. 2 & 3

Essential Service Water Buildings 3 & 4

U.S. EPR General Plant Layout

UAA	Switchyard
UBA	Switchgear Building
UBP	Emergency Power Generating Building
UFA	Fuel Building
UGC	Demineralized Water Storage Area
UJA	Reactor Building
UJH	Safeguard Building Mechanical
UJK	Safeguard Building Electrical
UKA	Nuclear Auxiliary Building
UKE	Access Building
UKH	Vent Stack
UKS	Radioactive Waste Processing Building
UMA	Turbine Building
URA	Cooling Tower Structure
URB	Essential Service Water Cooling Tower Structure
USG	Fire Protection Storage Tanks and Building
UST	Workshop & Warehouse Building
UTG	Central Gas Supply Building
UYF	Security Access Facility



Not To Scale

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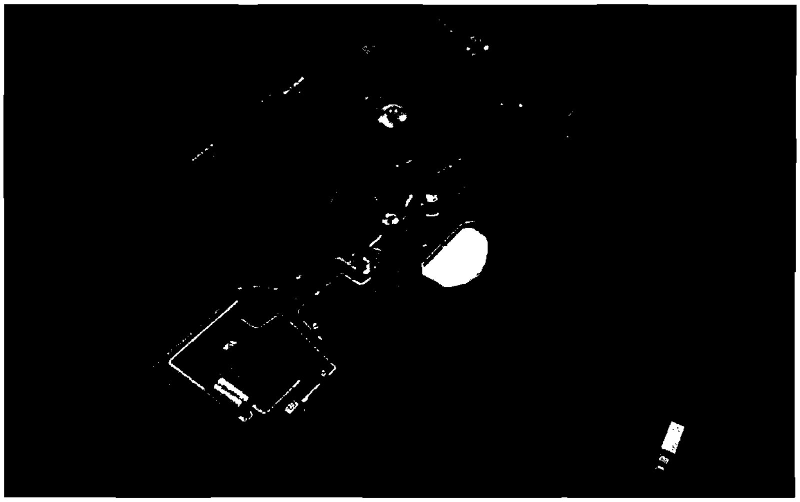
Radial Design
N+2 Approach



EPR

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The Four Train Concept

4 Independent Safety Trains

Arranged into 4 divisions

- ⇨ Preventive maintenance during power operation
- ⇨ Shorter outage time

Higher Availability

- ⇨ Smaller components
- ⇨ Fewer valves per train

Easier Maintainability

- ⇨ Efficient hazard protection
- ⇨ Reduced piping and components
- ⇨ Optimized plant layout

Lower Unit Cost

EPR

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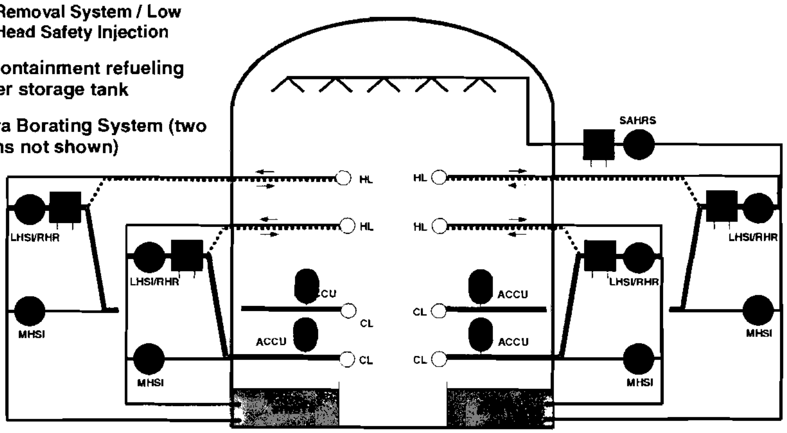
Main Safety Systems

- Four train Safety Injection System (SIS)
  - Medium head SI pumps
  - Combined Residual Heat Removal System / Low Head Safety Injection
- In-Containment refueling water storage tank
- Extra Borating System (two trains not shown)

- Non-safety containment spray for severe accident



Division 1 Division 2 Division 3 Division 4

EPR

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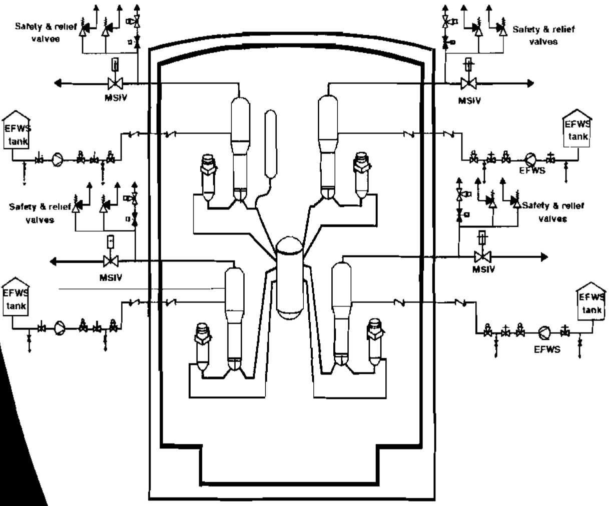
Main Safety Systems
Secondary Side

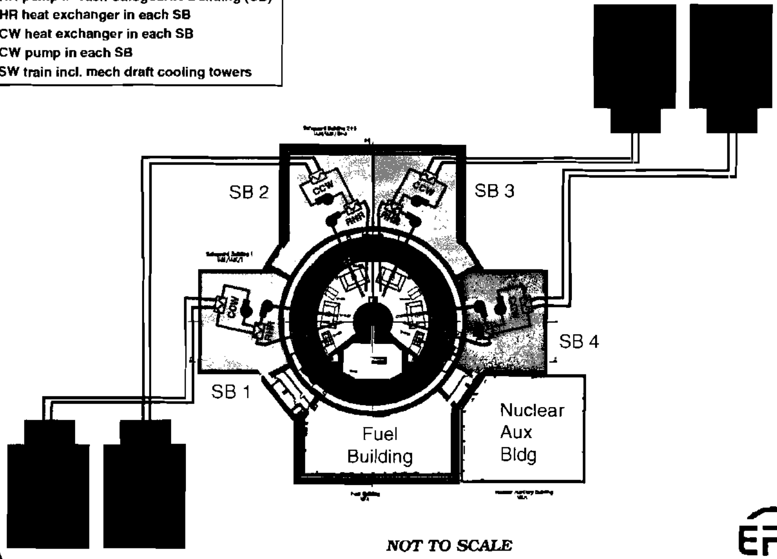


- ➢ Safety-related main steam relief train
- ➢ Four separate Emergency Feed Water Systems (EFWS)
- ➢ Separate power supply for each
- ➢ 2/4 EFWS also powered by Station Black Out (SBO) diesels
- ➢ Interconnecting headers at EFWS pump suction & discharge

EPR

Example: RHR Systems

Each Train Connects to Different RCS Loop
- **1 RHR pump in each Safeguards Building (SB)**
- **1 RHR heat exchanger in each SB**
- **1 CCW heat exchanger in each SB**
- **1 CCW pump in each SB**
- **1 ESW train incl. mech draft cooling towers**



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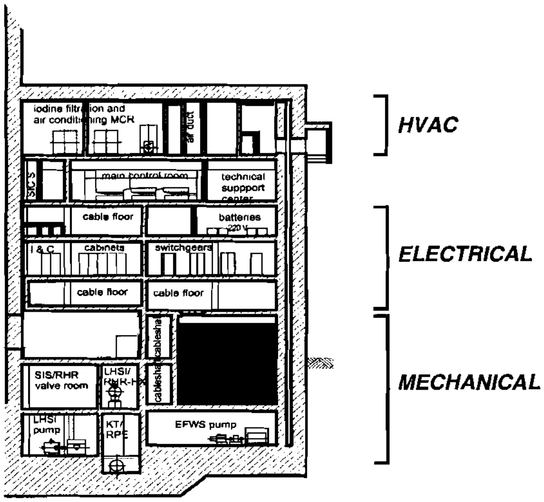
Safeguard Building Layout

Safeguard Building Division 2

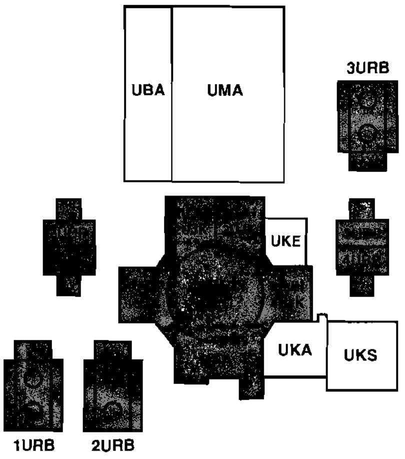


Divisional Approach

Four Versus Two

- Front-line safety systems 4 x 100%
  - Protection System
  - Emergency Power Supply System
  - ECCS
  - CCWS
  - ESWS
  - EFWS
- Many 2 x 100%
  - Annulus Ventilation
  - Safeguards & Fuel Building Iodine Filtration
  - Control Room Iodine Filtration
  - Containment Isolation
  - Extra Borating System
  - Spent Fuel Pool Cooling

Protection From External Hazards

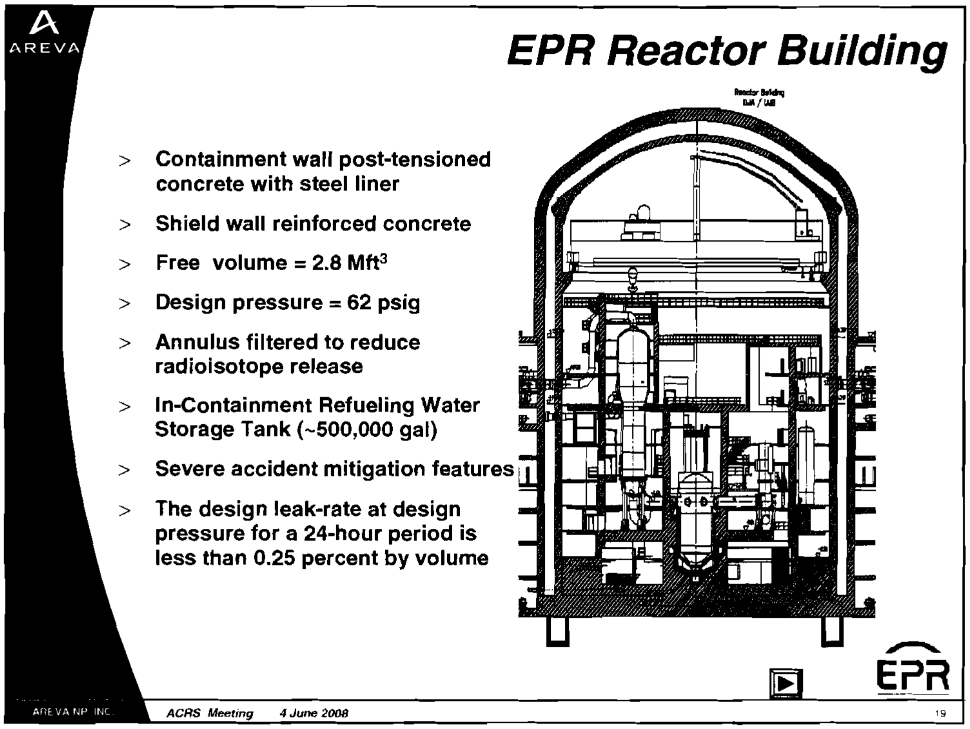


UBA	Switchgear Building
UBP	Emergency Power Generating Building
UFA	Fuel Building
UJA	Reactor Building
UJH	Safeguard Building Mechanical
UJK	Safeguard Building Electrical
UKA	Nuclear Auxiliary Building
UKE	Access Building
UKH	Vent Stack
UKS	Radioactive Waste Processing Building
UMA	Turbine Building
URB	Essential Service Water Cooling Tower Structure

▩ PROTECTED BY SHIELD BUILDING

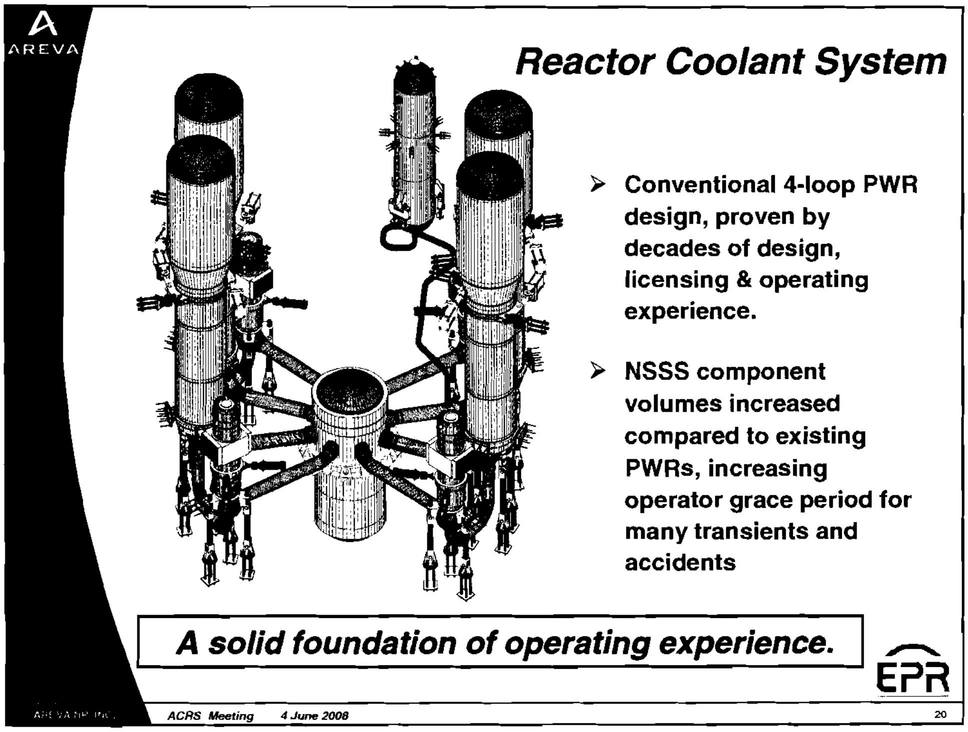
▨ PROTECTED BY PHYSICAL SEPARATION

☐ NOT PROTECTED



EPR Reactor Building

- Containment wall post-tensioned concrete with steel liner
- Shield wall reinforced concrete
- Free volume = 2.8 Mft^3
- Design pressure = 62 psig
- Annulus filtered to reduce radioisotope release
- In-Containment Refueling Water Storage Tank (~500,000 gal)
- Severe accident mitigation features
- The design leak-rate at design pressure for a 24-hour period is less than 0.25 percent by volume



Reactor Coolant System

- Conventional 4-loop PWR design, proven by decades of design, licensing & operating experience.
- NSSS component volumes increased compared to existing PWRs, increasing operator grace period for many transients and accidents

A solid foundation of operating experience.

AREVA

U.S. EPR Plant Parameter Comparison

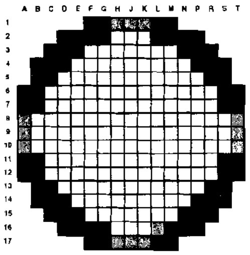
Parameter	4-Loop (Uprated)	U.S. EPR
Design Life	40	60
Thermal Power, MW	3565	4590
Electrical Power (Net), MW	1170	1595
Plant Efficiency, Percent	33	35
Hot Leg Temperature, F	618	624
Cold Leg Temperature, F	558	564
Reactor Coolant Flow Per Loop, gpm	100,500	124,700
Primary System Design Pressure, psia	2500	2550
Secondary System Design Pressure, psia	1200	1450
Primary System Operating Pressure, psia	2250	2250
Steam Pressure, psia	1000	1109
Steam Flow Per Loop, Mlb/hr	4.1	5.2
Total RCS Volume, cu.ft.	12,265	16,245
Pressurizer Volume, cu.ft.	1800	2650
SG Secondary Inventory at Full Power, lbm	101,000	182,000

EPR

AREVA

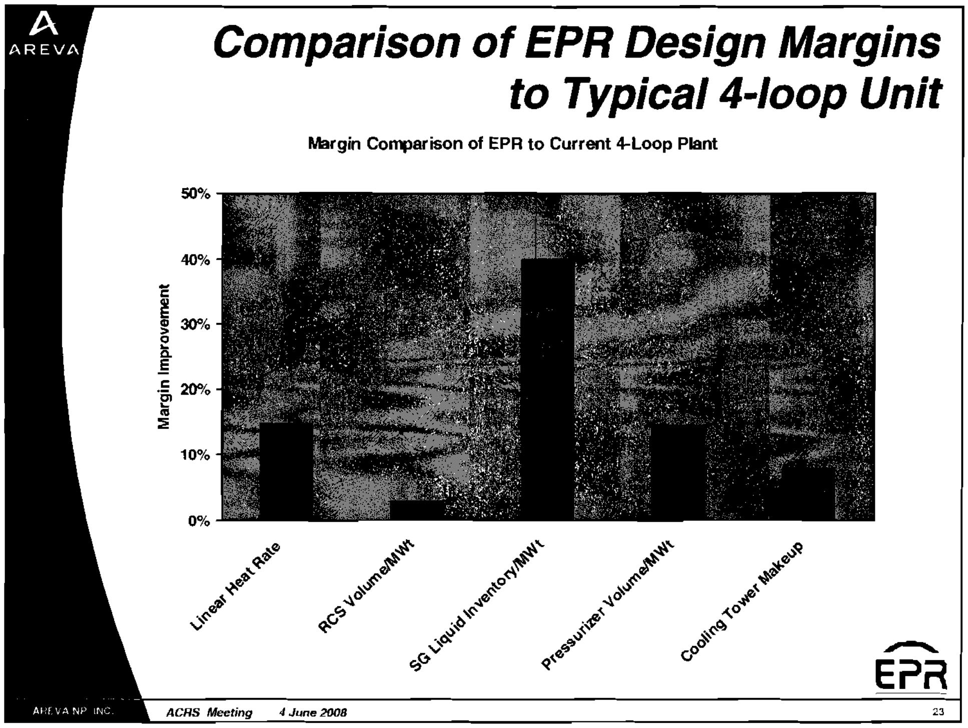
EPR Core Design Parameters

Parameter	Current 4-Loop (Uprated)	EPR
Core Thermal Power, MW	3565	4590
Number of Fuel Assemblies	193	241
Fuel Lattice	17x17	17x17
Active Fuel Length, ft	12	13.78
Rods Per Assembly	264	265
Average Linear Heat Rate, kw/ft	5.8	5.2
Peak Linear Heat Rate, kW/ft	14.6	13.8
Number of Control Rods	53	89



Type of Plant	No of Fuel Assy	
4-loop 1300 MWe	193	■
4-loop N4	205	▨
U.S. EPR	241	■

EPR

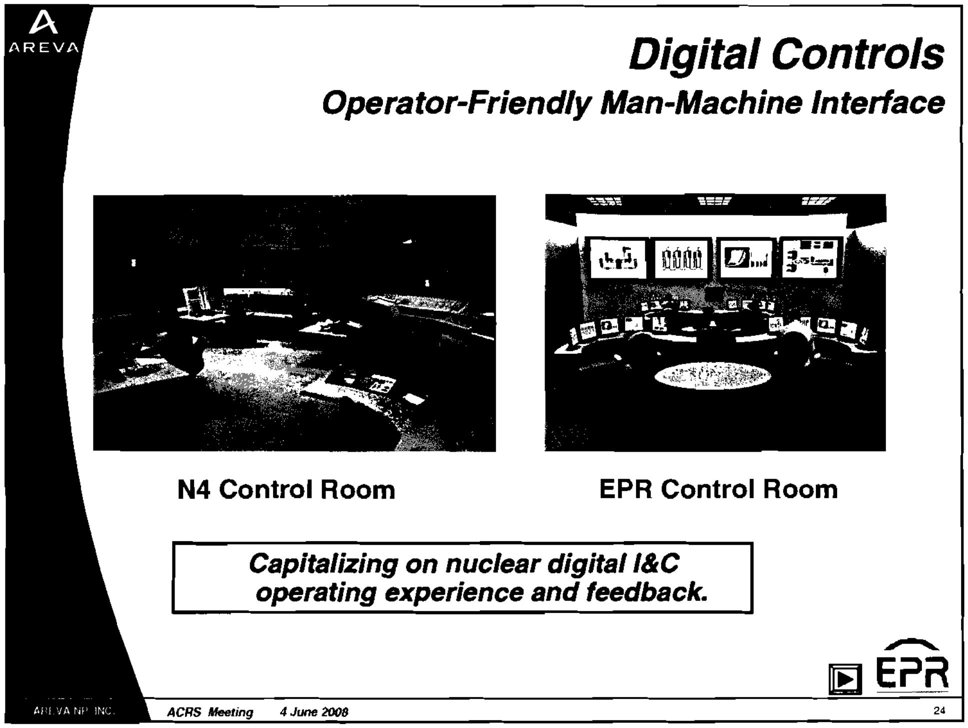


AREVA

Comparison of EPR Design Margins to Typical 4-loop Unit

Margin Comparison of EPR to Current 4-Loop Plant

EPR



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Digital Controls

Operator-Friendly Man-Machine Interface

N4 Control Room

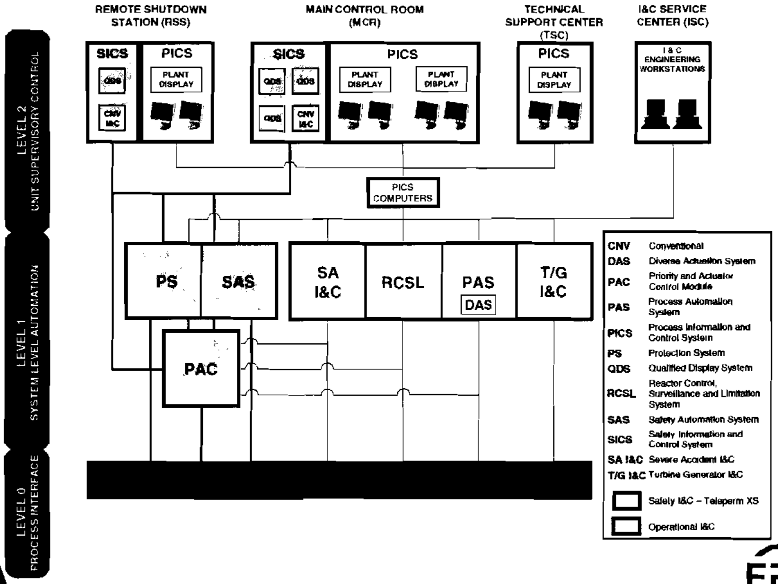
EPR Control Room

Capitalizing on nuclear digital I&C operating experience and feedback.

EPR

AREVA

Digital I & C Architecture



CNV	Conventional
DAS	Diverse Actuation System
PAC	Priority and Actuator Control Module
PAS	Process Automation System
PICS	Process Information and Control System
PS	Protection System
QDS	Qualified Display System
RCSL	Reactor Control, Surveillance and Limitation System
SAS	Safety Automation System
SICS	Safety Information and Control System
SA I&C	Severe Accident I&C
T/G I&C	Turbine Generator I&C
▭	Safety I&C – Teleperm XS
▭	Operational I&C

EPR

AREVA

Increased Protection & Automation

- Hot-channel DNBR trip
- High linear power density trip
- High SG pressure trip
- Protection System SG depressurization
- Automatic boron dilution detection
- Computer-controlled heat-up & cooldown
- On-line procedures
- Electronic tagging
- Self-checking

EPR

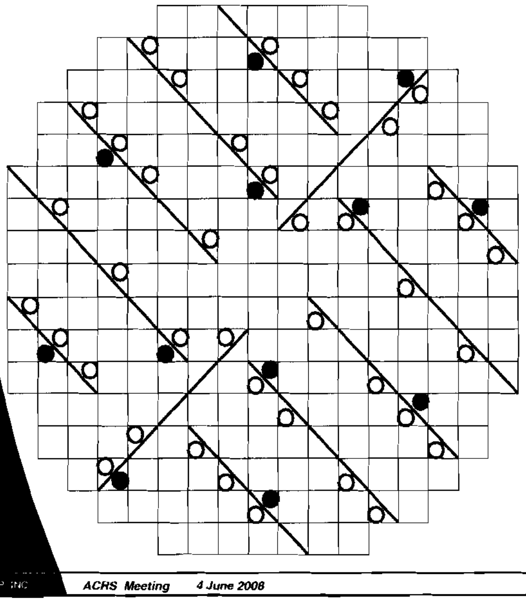
In-Core Monitoring

- Fixed and moveable core monitoring systems
- Self-Powered Neutron Detectors continuously monitor core power
  - ☐ Provide input signals to POWERTRAX/E* software
  - ☐ Safety and non-safety functions are generated by SPNDs
  - ☐ SPND signal drift with burnup compensated by calibration
- The Aeroball Measurement System is used to calibrate SPNDs
  - ☐ About every 15 EFPD, the SPNDs are calibrated to the AMS reference signal
  - ☐ AMS is a moveable system that provides accurate 3-D core power maps
  - ☐ The AMS provides no signals to any protection or monitoring functions

* POWERTRAX/E provides a comprehensive system for on-line 3-D power distribution monitoring and for reactor operation support calculations

EPR

Nuclear Instrumentation

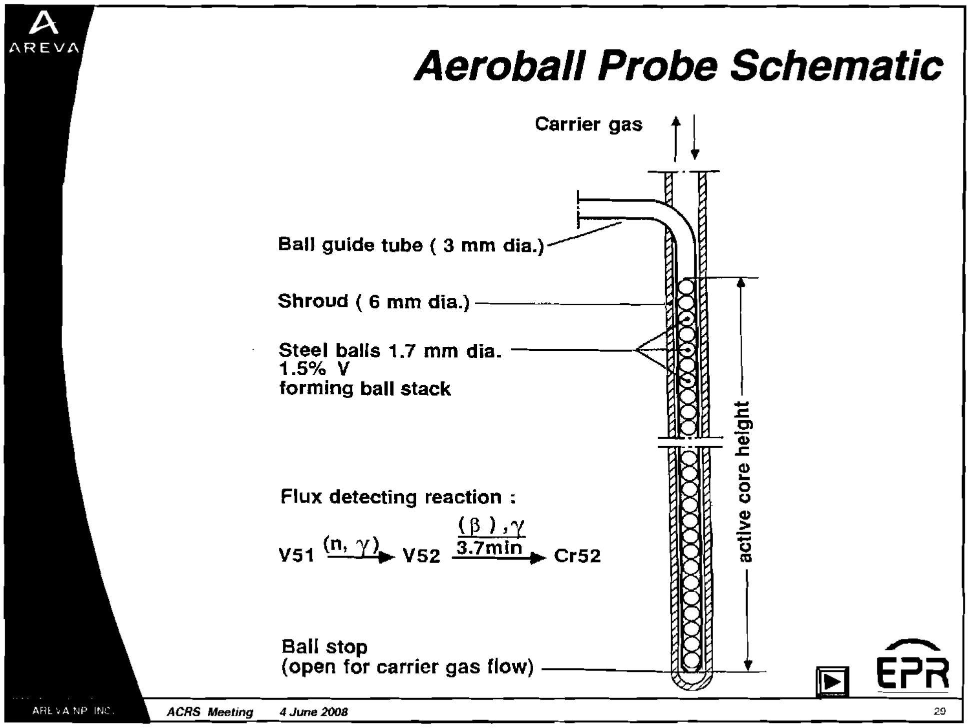


☐ 241 ASSEMBLIES

● 12 SPND FINGERS

○ 40 AEROBALL PROBES

EPR



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Aeroball Probe Schematic

Carrier gas

Ball guide tube (3 mm dia.)

Shroud (6 mm dia.)

Steel balls 1.7 mm dia.
1.5% V
forming ball stack

Flux detecting reaction :

$V51 \xrightarrow{(n,\gamma)} V52 \xrightarrow[3.7min]{(\beta),\gamma} Cr52$

Ball stop
(open for carrier gas flow)

active core height

EPR

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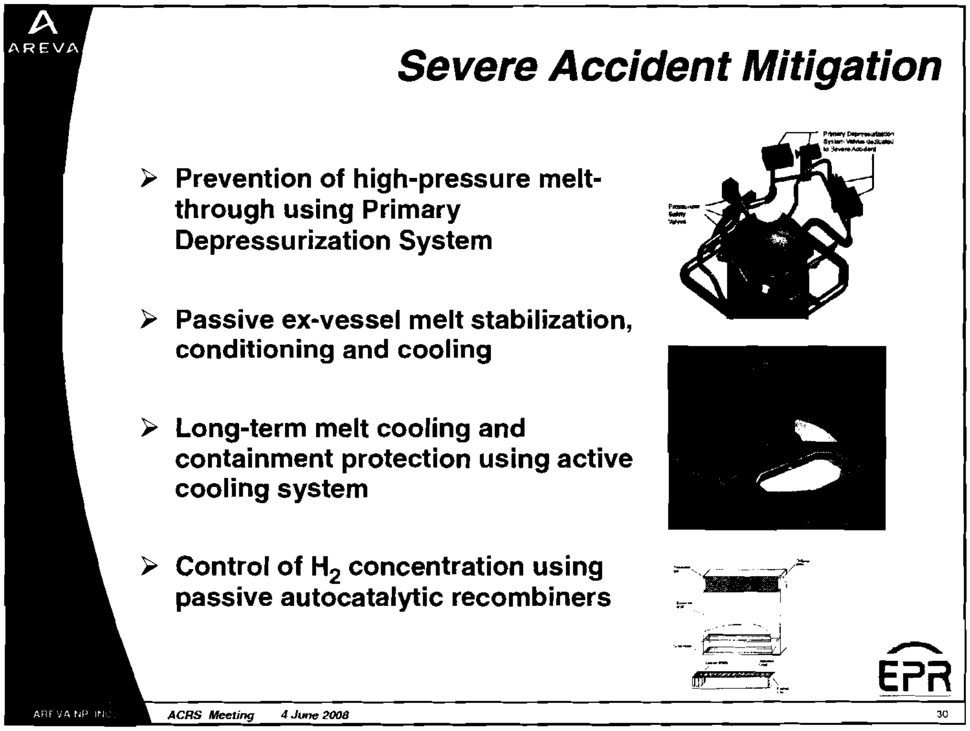

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Severe Accident Mitigation

- Prevention of high-pressure melt-through using Primary Depressurization System
- Passive ex-vessel melt stabilization, conditioning and cooling
- Long-term melt cooling and containment protection using active cooling system
- Control of H_2 concentration using passive autocatalytic recombiners

EPR

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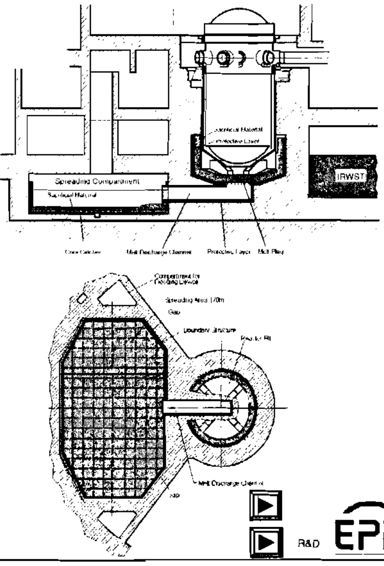
Severe Accident Mitigation

Melt Conditioning and Stabilization

- **Reactor cavity temporarily retains molten core debris prior to spreading and stabilization processes**
  - Limits uncertainties associated with RPV release states
  - Corium/concrete interaction within reactor cavity lowers melting temperature of corium and promotes spreading
- **Melt spreading and relocation**
  - After melt plug failure, conditioned melt will relocate into spreading area (shallow crucible)
  - Large spreading area promotes cooling
  - Spreading area is dry at time of melt relocation to preclude ex-vessel steam explosion
- **Stablization**
  - Water from IRWST passively cools melt for up to 12 hours
  - Thereafter, severe accident heat removal system actively cools the melt and depressurizes containment

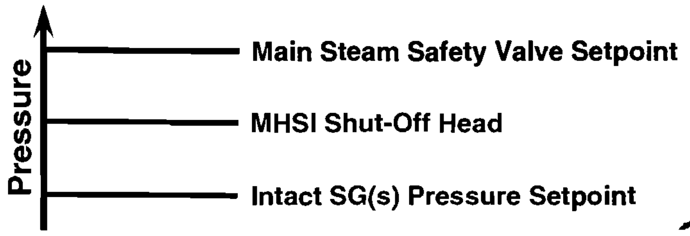


R&D

SGTR Mitigation

Safety Injection System

- **Medium Head Injection selected for SGTR mitigation:**
  - **Shutoff head below MSSV setpoint**
  - **Ensures no challenge to MSSVs during SGTR (no operator action required to throttle safety injection)**
  - **SGTR dose consequences meet safety goal by minimizing containment bypass (eliminate possibility of discharging reactor coolant)**



Main Steam Safety Valve Setpoint

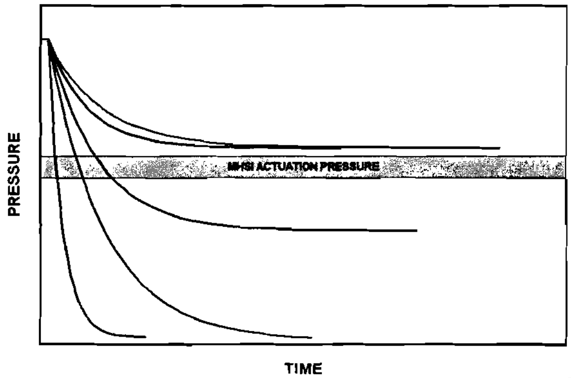
MHSI Shut-Off Head

Intact SG(s) Pressure Setpoint

SGTR & SBLOCA Mitigation

SBLOCA Spectrum Studies

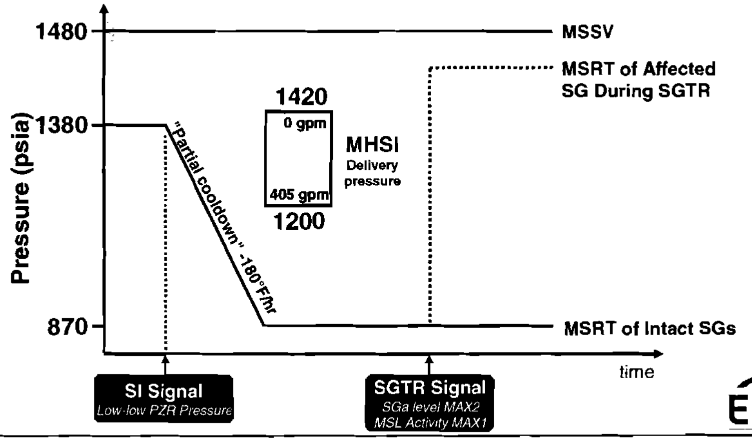
- For very small LOCAs, RCS pressure "couples" to SG pressure because SG heat removal is maintained
- SI flow begins when RCS/SG pressure falls below the MHSI shut-off head



SBLOCA Mitigation

Partial Cooldown

- Safety-related function (Protection System)
- Depressurizes SGs to reduce T_{sat} at 180 F/hr
- Ensures adequate MHSI flow for SBLOCA



AREVA

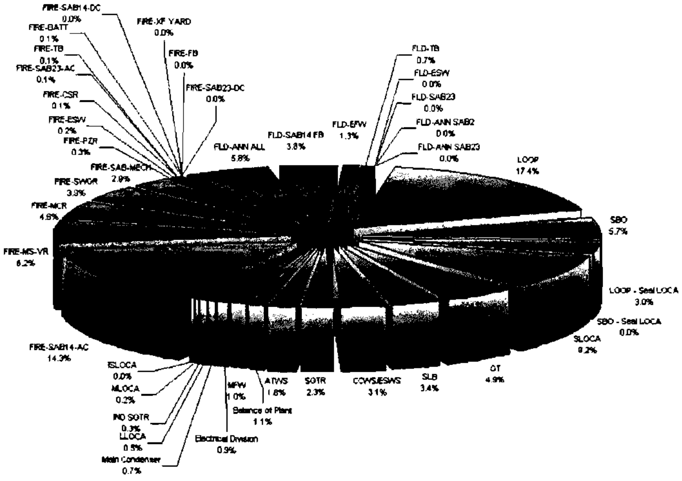
Probabilistic Objectives And Targets

- Safety objective for integral core melt frequency (all plant states, all types of initiators): $< 10^{-5}$ per year
- Design target for core damage frequency for internal events
  - from power states: $< 10^{-6}$ per year
  - from shutdown states: less than power states
- Design target for core damage with large and early releases from containment: $< 10^{-7}$ /year

EPR

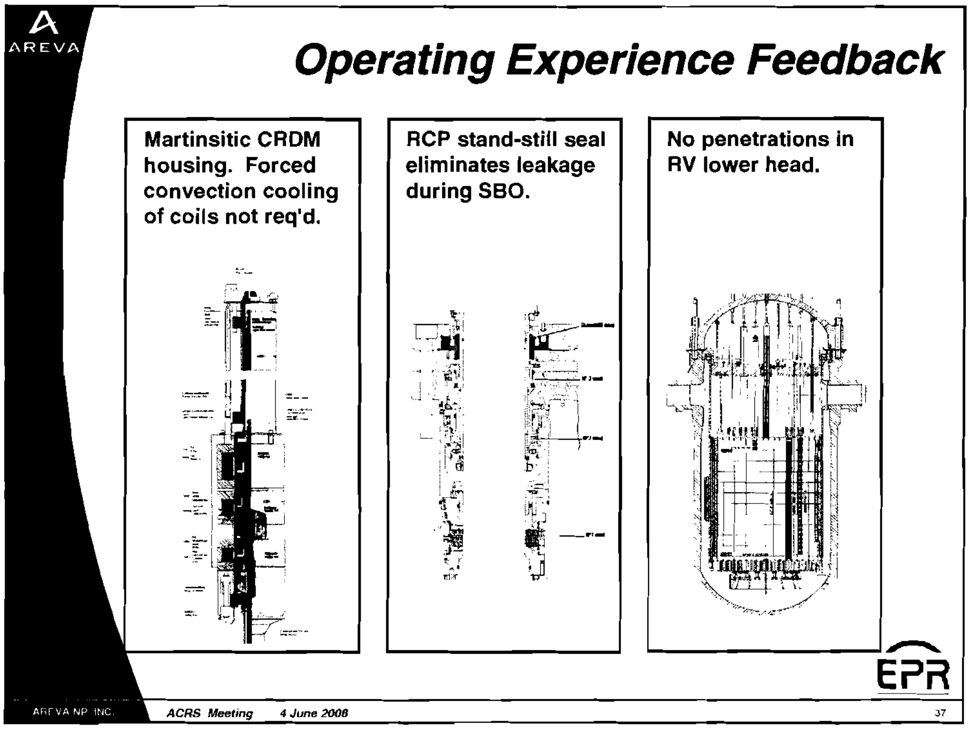
AREVA

U.S. EPR CDF (At-Power Events)



Level 1 At-Power, Internal Events CDF = 5.3×10^{-7}/yr
CDF For All Events < 5.8×10^{-7}/yr

EPR



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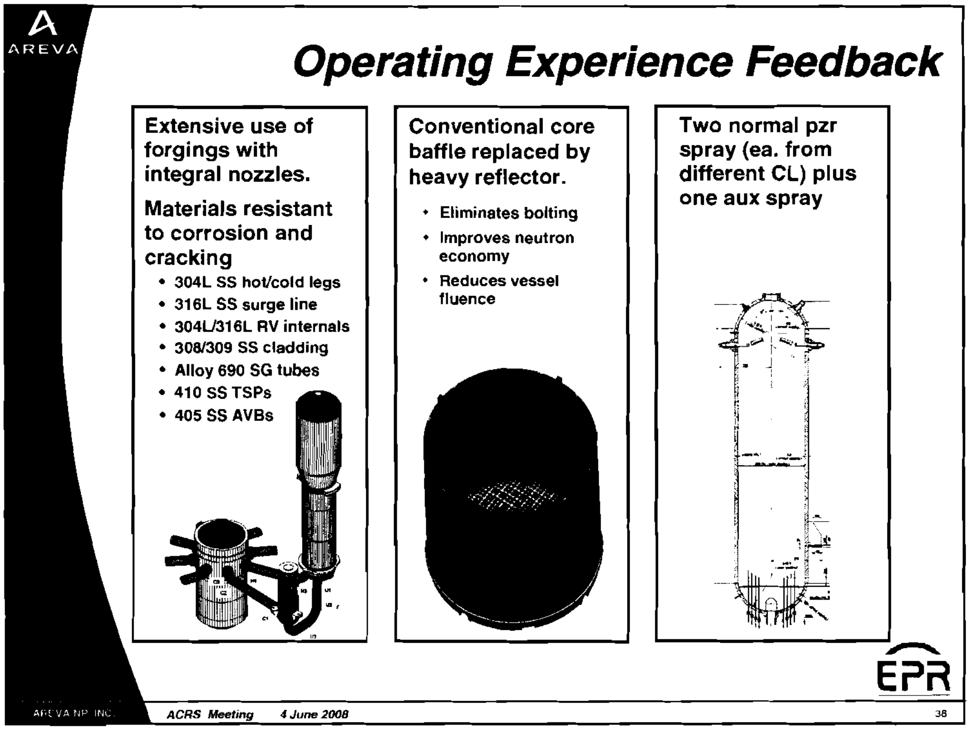
Operating Experience Feedback

Martinsitic CRDM housing. Forced convection cooling of coils not req'd.

RCP stand-still seal eliminates leakage during SBO.

No penetrations in RV lower head.

EPR



AREVA

Operating Experience Feedback

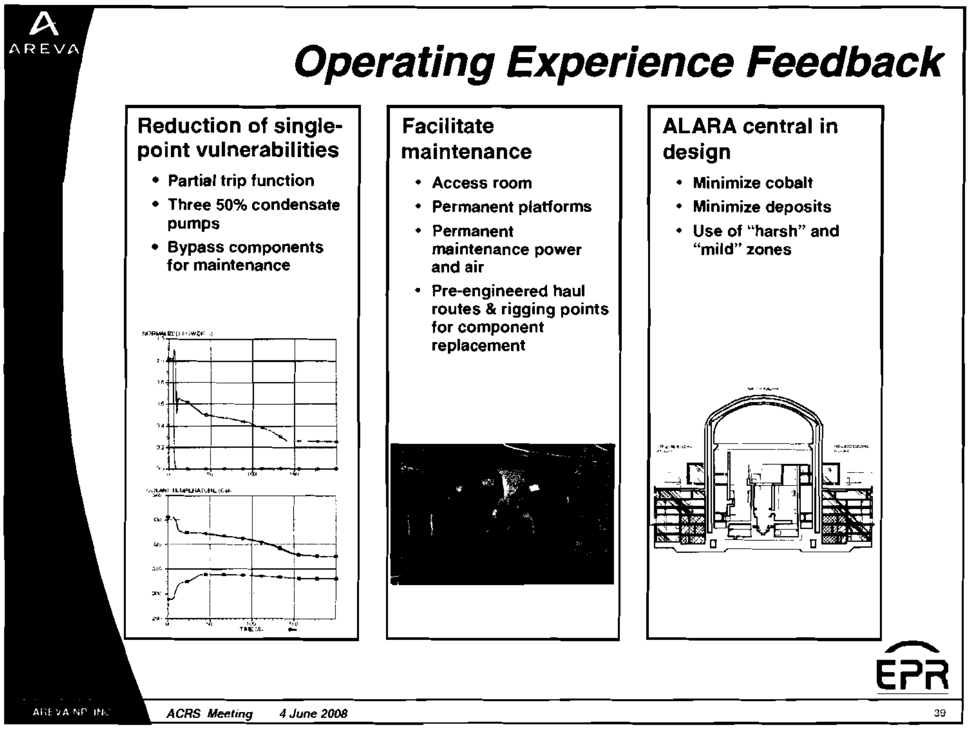
Extensive use of forgings with integral nozzles.

Materials resistant to corrosion and cracking
- 304L SS hot/cold legs
- 316L SS surge line
- 304L/316L RV internals
- 308/309 SS cladding
- Alloy 690 SG tubes
- 410 SS TSPs
- 405 SS AVBs

Conventional core baffle replaced by heavy reflector.
- Eliminates bolting
- Improves neutron economy
- Reduces vessel fluence

Two normal pzr spray (ea. from different CL) plus one aux spray

EPR



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Operating Experience Feedback

Reduction of single-point vulnerabilities

- Partial trip function
- Three 50% condensate pumps
- Bypass components for maintenance

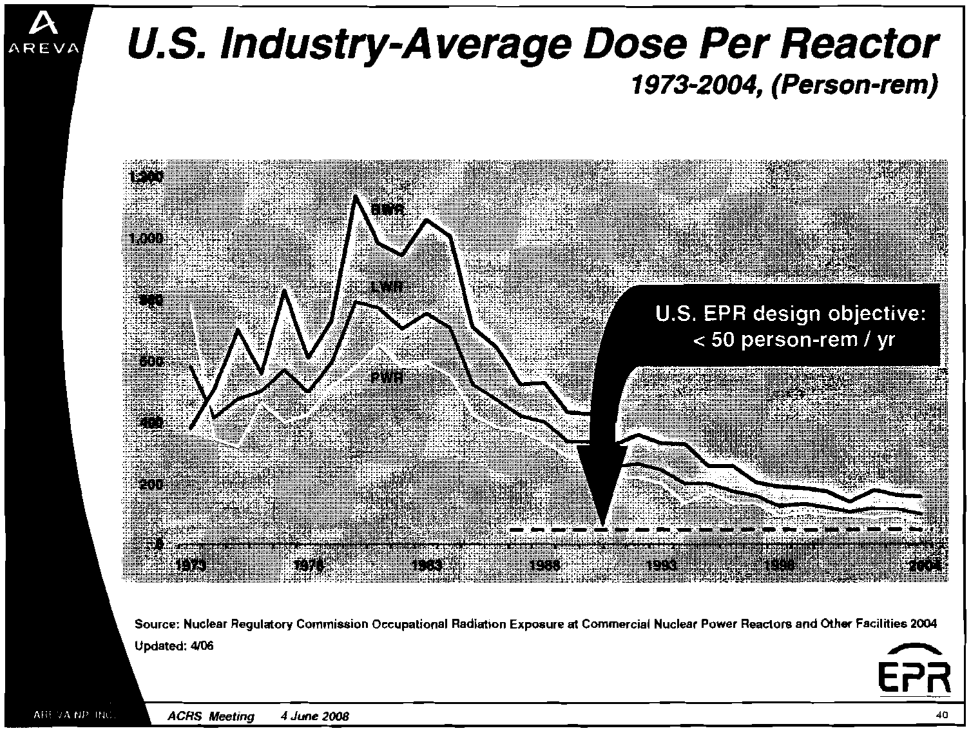
Facilitate maintenance

- Access room
- Permanent platforms
- Permanent maintenance power and air
- Pre-engineered haul routes & rigging points for component replacement

ALARA central in design

- Minimize cobalt
- Minimize deposits
- Use of "harsh" and "mild" zones

EPR



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U.S. Industry-Average Dose Per Reactor

1973-2004, (Person-rem)

U.S. EPR design objective: < 50 person-rem / yr

Source: Nuclear Regulatory Commission Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities 2004

Updated: 4/06

EPR

Design Summary

- U.S. EPR is evolutionary
- Most features are typical of operating PWRs
- Features included to
  - Improve safety
  - Protect critical systems from external events
  - Improve human factors
  - Enhance reliability

Backup Slides

Security Rulemaking for Nuclear Power Plants

ACRS Presentation
June 4, 2008

Discussion Topics

- Status of Power Reactor Security Rulemaking
- Staff Draft Final Rule Text needing ACRS review
  - 50.54(hh) Imminent Attack/Mitigative Measures
  - 73.54 Cyber Security
  - 73.58 Safety/Security Interface
- Status of Regulatory Guidance

Security Rulemaking

- Part 73 Power Reactor Security Rulemaking (proposed rule published 10/06)
  - 50.54 (hh) Mitigative Strategies and Response Procedures for Potential or Actual Aircraft Attacks
  - 73.54 Protection of Digital Computer and Communication Systems and Networks
  - 73.55 Physical Security for Power Reactors
  - 73.56 Personnel Access Authorization Requirements for Nuclear Power Plants

Security Rulemaking (cont.)

- Part 73 Power Reactor Security Rulemaking (proposed rule published 10/06)
  - 73.58 Safety/Security Interface Requirements for Nuclear Power Plants
  - Appendix B to Part 73 – Section VI, Nuclear Power Reactor Training and Qualification for Personnel Performing Security Program Duties
  - Appendix C to Part 73 – Licensee Safeguards Contingency Plans

Security Rulemaking for Nuclear Power Plants

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- Status of Power Reactor Security Rulemaking
- Staff Draft Final Rule Text needing ACRS review
    - 50.54(hh) Imminent Attack/Mitigative Measures
    - 73.54 Cyber Security
    - 73.58 Safety/Security Interface
- Status of Regulatory Guidance

Status of Rulemaking

- FRN developed
- Begin formal concurrence on 6/16/2008
- Provide to EDO on 6/30/2008

ACRS Review for Rulemaking

- 50.54 (hh) Mitigative Strategies and Response Procedures for Potential or Actual aircraft Attacks
  - DG-50XX (July 2008)
- 73.54 Protection of Digital Computer and Communication Systems and Networks
  - DG 5022
- 73.58 Safety/Security Interface Requirements for Nuclear Power Plants
  - DG 5021 Safety/Security Interface

Draft Final Rule Text for 50.54 (hh) as of 6/4/2008

- Mitigative Strategies and Response Procedures for Potential or Actual aircraft Attacks
  - Contained in Appendix C of proposed rule
  - Moved to 50.54, Conditions of License
  - Supplemental rule published in Federal Register 4/10/2008
  - Comments received; incorporated into FRN
- Guidance to be developed from existing advisories, information (DG 50XX)

Draft Final Rule Text for 73.54 as of 6/4/2008

- Protection of Digital Computer and Communication Systems and Networks
  - Programmatic requirements for addressing cyber security
  - Included as part of DBT 73.1 issued March 2008
- DG 5022 Cyber Security Programs for Nuclear Facilities
  - Completed 6/1/08 (OUO)
  - In process of distribution to appropriate licensees (by 6/6/2008)

Draft Final Rule Text for 73.58 as of 6/4/2008

- Safety/Security Interface Requirements for Nuclear Power Plants
  - Requires coordination of potential adverse interactions between security activities and other plant activities
  - Addresses PRM 50-80, in part
- DG 5021 Safety/Security Interface
  - Published in Federal Register July 24, 2007
  - Public Meeting held; comments received & under consideration

Summary

- Security Rulemaking proceeding
- Supporting Regulatory Guidance for 50.54(hh) not developed
- Supporting Regulatory Guidance for 73.58 and 73.54 developed and drafts published or distributed

SECURITY RULEMAKING
STAFF DRAFT FINAL RULE LANGUAGE
As of 6/4/2008

§ 50.54(hh) Mitigative Strategies and Response Procedures for Potential or Actual Aircraft Attacks.
(1) Each licensee shall develop, implement and maintain procedures that describe how the licensee will address the following areas if the licensee is notified of a potential aircraft threat:
(i) Verification of the authenticity of threat notifications;
(ii) Maintenance of continuous communication with threat notification sources;
(iii) Contacting all onsite personnel and applicable offsite response organizations;
(iv) Onsite actions to enhance the capability of the facility to mitigate the consequences of an aircraft impact;
(v) Measures to reduce visual discrimination of the site relative to its surroundings or individual buildings within the protected area;
(vi) Dispersal of equipment and personnel, as well as rapid entry into site protected areas for essential onsite personnel and offsite responders who are necessary to mitigate the event; and
(vii) Recall of site personnel.
(2) Each licensee shall develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire, to include strategies in the following areas:
(i) Fire fighting;
(ii) Operations to mitigate fuel damage; and
(iii) Actions to minimize radiological release.
(3) This section does not apply to a nuclear power plant for which the certifications required under § 50.82(a) or § 52.100(a)(1) of this chapter have been submitted.

§73.54 "Protection of digital computer and communication systems and networks"
(a) Each licensee subject to the requirements of this section shall provide high assurance that digital computer and communication systems and networks are adequately protected against cyber attacks, up to and including the design basis threat as described in Title 10 of the Code of Federal Regulations (10 CFR) Part 73, Section 73.1.
(a)(1) The licensee shall protect digital computer and communication systems and networks associated with:
(a)(1)(i) safety-related and important-to-safety functions,
(a)(1)(ii) security functions,
(a)(1)(iii) emergency preparedness functions, including offsite communications,
(a)(1)(iv) support systems and equipment which, if compromised, would adversely impact safety, security or emergency preparedness functions.
(a)(2) The licensee shall protect the systems and networks identified in paragraph (a)(1) of this section from cyber attacks that would:
(a)(2)(i) adversely impact the integrity or confidentiality of data and/or software;
(a)(2)(ii) deny access to systems, services, and/or data, and;
(a)(2)(iii) adversely impact the operation of systems, networks, and associated equipment.
(b) To accomplish this, the licensee shall:
(b)(1) analyze digital computer and communication systems and networks and identify those assets that must be protected against cyber attacks to satisfy paragraph (a) of this section,
(b)(2) establish, implement, and maintain a cyber security program for the protection of the assets identified in (b)(1) of this section, and;

(b)(3) incorporate the cyber security program as a component of the physical protection program.
(c) The cyber security program must be designed to:
(c)(1) implement security controls to protect the assets identified by paragraph (b)(1) of this section from cyber attacks,
(c)(2) apply and maintain defense-in-depth protective strategies to ensure the capability to detect and respond to cyber attacks,
(c)(3) mitigate the adverse affects of cyber attacks, and;
(c)(4) ensure that the functions of protected assets identified by paragraph (b)(1) of this section are not adversely impacted due to cyber attacks.
(d) As part of the cyber security program, the licensee shall:
(d)(1) ensure that appropriate facility personnel, including contractors, are aware of cyber security requirements and receive the training necessary to perform their assigned duties and responsibilities effectively.
(d)(2) evaluate and manage cyber risks.
(d)(3) ensure that modifications to assests identified by paragraph (b)(1) of this section, are evaluated prior to implementation to ensure that the cyber security performance objectives identified in (a)(1) are maintained.
(e) The licensee shall establish, implement, and maintain a cyber security plan that implements the cyber security program requirements of this section.
(e)(1) The cyber security plan must describe how the requirements of this section will be implemented and must account for the site-specific conditions that affect implementation.
(e)(2) The cyber security plan must include measures for incident response and recovery for cyber attacks. The cyber security plan must describe how the licensee will:
(e)(2)(i) maintain the capability for timely detection and response to cyber attacks,
(e)(2)(ii) mitigate the consequences of cyber attacks,
(e)(2)(iii) correct exploited vulnerabilities, and;
(e)(2)(iv) restore affected systems, networks, and/or equipment affected by cyber attacks.
(f) The licensee shall develop and maintain written policies and implementing procedures to implement the cyber security plan.
(f)(1) Policies, implementing procedures, site-specific analysis, and other supporting technical information used by the licensee need not be submitted for Commission review and approval as part of the cyber security plan; but are subject to inspection by NRC staff on a periodic basis.
(g) The cyber security program shall be audited as a component of the physical security program and will be subject to the same requirements and controls.
(h) The licensee shall retain records and supporting technical documentation required to satisfy the requirements of this section until the Commission terminates the license for which the records were developed, and shall maintain superseded portions of these records for at least three (3) years after the record is superseded, unless otherwise specified by the Commission.

§ 73.58 Safety/Security Interface Requirements for Nuclear Power Reactors

(a) Each operating nuclear power reactor licensee with a license issued under part 50 or 52 of this chapter shall comply with the requirements of this section.
(a)(1) The licensee shall assess and manage the potential for adverse affects on safety and security, including the site emergency plan, before implementing changes to plant configurations, facility conditions, or security.
(a)(2) The scope of changes to be assessed and managed must include planned and emergent activities (such as, but not limited to, physical modifications, procedural changes, changes to operator actions or security assignments, maintenance activities, system reconfiguration, access modification or restrictions, and changes to the security plan and its implementation).
(b) Where potential adverse interactions are identified, the licensee shall communicate them to appropriate licensee personnel and take compensatory and/or mitigative actions to maintain safety and security under applicable Commission regulations, requirements, and license conditions.

Synthesis on the findings from the ARTIST tests on aerosol retention in the secondary side of steam generators

Presented to the ACRS
June 4, 2008

M. Salay
U.S. Nuclear Regulatory Commission
Washington, D.C., USA

Overview

- Steam Generator Tube Ruptures (SGTR) background and NRC interest-SGAP
- ARTIST test program pertaining to SGAP
- Major Observations
- MELCOR modifications
- Conclusions

Steam generator tube rupture accidents

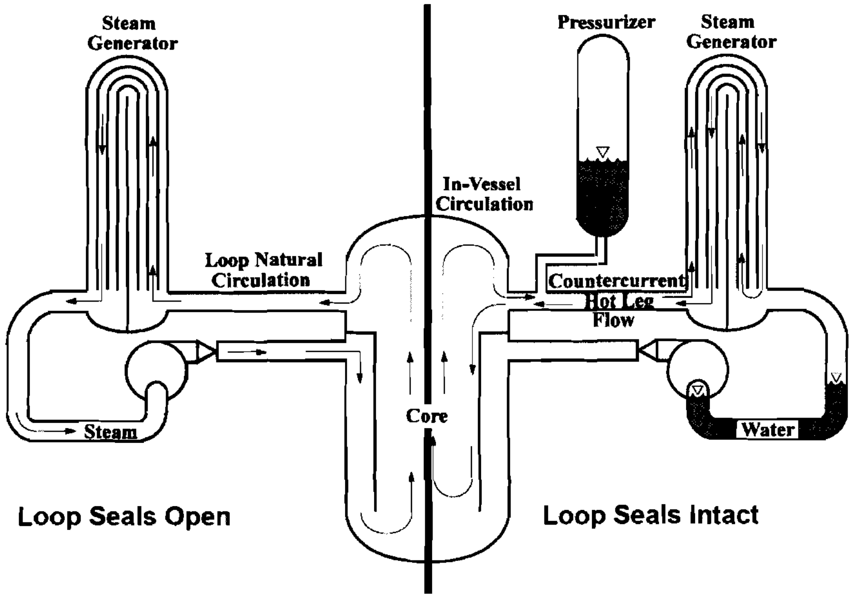
- Design basis event
  - Plants designed to cope
  - Have for all events to date
- Progresses to severe accident only if something else happens
  - Operator error

Induced steam generator tube rupture

- Induced rupture greater concern
  - Plants operate with detectable flaws in tubes
  - Limit on flaw size
  - Stress corrosion cracking is the cause of most flaws
  - Crevice corrosion at tube support plates of concern

Induced steam generator tube rupture

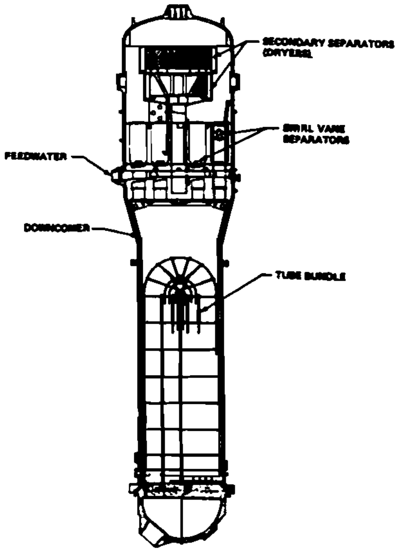
- Heat transfer from core to primary pressure boundary weakens structures
- Vulnerable locations
  - Hot leg nozzle
  - Surge line to pressurizer
  - Steam generator tubes
- Codes do not reliably predict failure location and depressurization timing



Severe accident natural circulation flows

Aerosol retention in SGTR SA

- at tube inlet from steam generator plenum (inlet efficiency)
- in the steam generator tube prior to reaching the tube rupture
- in the immediate vicinity of the break where particles could impact on adjacent tubes
- in tubes between one tube support plate and another
- on top of tube support plates
- on envelope by thermophoretic deposition
- in the steam separators and steam dryers at the top of the steam generator.
- at steam generator safety relief valve (inlet efficiency)



Westinghouse Model F Steam Generator

Aerosol retention processes

- **Removal mechanisms particle size dependent**
  - Laminar
    - large – impaction, settling, interception
    - small – diffusion
  - Turbulent
    - turbulent deposition
      - bounce
      - flow resuspension
      - saltation
- **Removal of particles alters particle size distribution**
  - maximum penetration size
  - retention of individual sections can not be simply combined to obtain overall retention
    - integral tests
  - SETs obtain individual section retention as function of size

Aerosol size

- A recommendation of prototypic aerosol size based on an IRSN survey of AECL, PBF-SFD and PHÉBUS experiments:
  - "size distribution at SG: near-lognormal, AMMD ~1µm or less, $\sigma \sim 2$; larger particles comprise agglomerates of small (~0.1 µm) highly coordinated clusters"
    - Sizes in two of the facilities were in the maximum penetration size range
    - Larger size range in third facility

Consequences of tube rupture

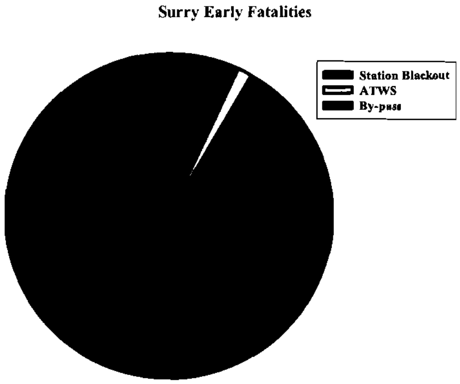
- Radionuclides vent directly to environment or to auxiliary building without any attenuation from engineered safety features in containment
- Accidents have sufficiently high consequences that they are risk dominant despite low probability

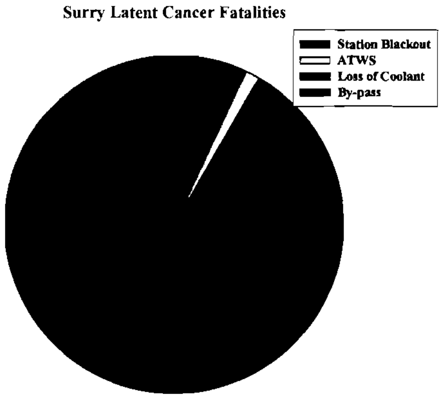
NUREG-1150

- Risk analysis of five US plants
  - Two PWRs had significant probabilities of steam generator tube rupture
  - All three PWRs could suffer induced steam generator tube rupture
- Limited modeling of aerosol behavior on secondary side of steam generators
  - None in the Source Term Code Package
  - Data unavailable

NUREG-1150 expert opinion elicitation

- Inlet efficiency from steam generator plenum to ruptured tubes – DF (mass in/mass out) ~2
- Retention in tubes - DF <~10 - no credit given
  - resuspension
  - revaporization
  - agglomerate breakup
- Retention in secondary side - DF ~4 to 6
  - deposition on outside of tubes resisted by thermophoresis
- No credit for steam dryer/separators
  - proprietary design information
- Large uncertainty in estimates





Alternate retention analysis

- Industry analyses provided far different estimates of retention in the secondary side of steam generator
  - Calculated steam generator DF on the order of 10,000
    - >100 in tube, depending on break location
    - 10s secondary near break
    - 2-3 far from break

Focus on SGTR bypass accident

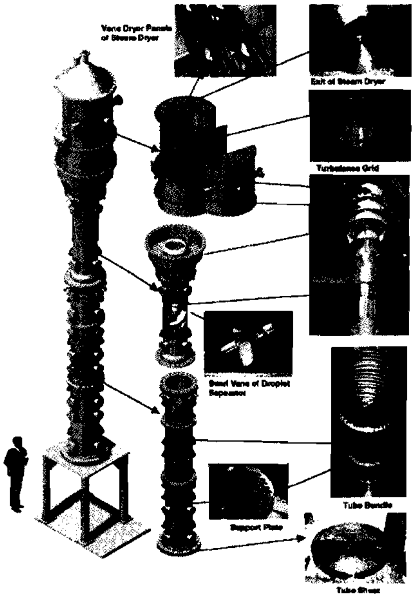
- attention to SGTR bypass accidents justified by risk
- Direct connection between risk and source term attenuation
- "are safety resources being misdirected to an unneeded attention on containment bypass accidents because we underestimate attenuation"

SGAP ITEM 3.3a

- STEAM GENERATOR ACTION PLAN (SGAP) ITEM 3.3a – DEVELOP EXPERIMENTAL INFORMATION ON SOURCE TERM ATTENUATION ON THE SECONDARY SIDE OF STEAM GENERATORS

ARTIST Project

- AeRosol Trapping In a STeam generator
  - International project conducted by the Paul Scherrer Institut (PSI)
  - seven phase project (NRC participated in 5)
  - separate and integral tests (38)
- retention measured:
  - in the steam generator tube prior to reaching the tube rupture (15)
  - in the immediate vicinity of the break where particles could impact on adjacent tubes (9)
  - in tubes between one tube support plate and another and on top of tube support plates (6) (1 stage,2 stage)
  - in the steam separators and steam dryers at the top of the steam generator. (5)
  - overall with all steam generator components (3)
- Other phases (not NRC)
  - retention in flooded bundle
  - droplets in dryers and separators



ARTIST facilities

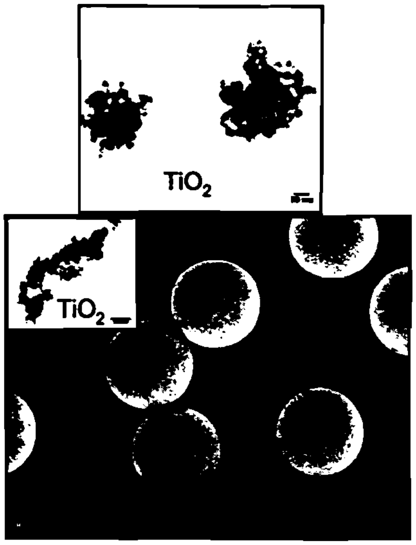
- ARTIST
  - based on Beznau plant: 365 MWe Westinghouse 2 loop PWR (1969,1972)
  - scaled for SGTR
  - 19.08 mm tube diameter
  - approx 1:20 flow area and number of tubes
- Main facility
  - shortened and narrowed bundle with U-bend tube section
  - a tube sheet
  - 3 support plates
  - full scale separator and dryer
- SET facilities
  - in tube
  - at break
  - rods far from break and support plates
  - separator and dryer

	Beznau	ARTIST
Number of tubes	3238	270 (89)*
Dryers	12	1
Separators	12	1
Bundle dia. (m)	2.68	0.57
Max tube height (m)	9	3.8 (9)**
Flow area (m^2)	3.79	0.185
Sup. plate flow area (m^2)	1.288	0.052
Bundle D_h (cm)	3.1	3.1
Total height (m)	17	10.5

*separate test section for assessing retention far from break
**in tube retention tests

Test Parameters

- Guillotine break
- Aerosol particles (composition/size)
  - – TiO_2 agglomerates (AMMD 1-5 µm)
    - Degussa
    - Nanophase
  - – SiO_2 spheres, D_{ae} = 0.7, 1.4, 3.7 µm
  - – Latex spheres, D_{ae} = 0.4 µm
- Concentrations
  - – 0.01 to 100s of mg/m³
- Flow rate:
  - – nitrogen (steam)
  - – few 10s – several 100s kg/h
- scoping tests to determine suitable parameters precede experiments
- tests to determine experimental uncertainty



TEM micrographs: Dr. Jerry Egeland / PSI
SEM micrograph: Dr. Unto Tapper / VTT

Primary Measurement Methods

- Size distribution, concentration, retained mass, and DF
  - – sampling at inlet, outlet, and other locations
- Size distribution:
  - – Berner Impactor
  - – Electrical Low Pressure Impactor
  - – Optical Particle Counter
- Concentration:
  - – Filter
  - – Photometer
  - – Optical Particle Counter
- Mass collection, concentrations with flow used to determine DF
- Flow rates at inlet and outlet and at all sampling devices, gauge pressures at inlet and outlet, gas T

Major observations

- Two forms of aerosol deposition:
  - Always a fairly uniform layer of fine aerosol on surfaces exposed to the aerosol-laden flow. "tenacious"
  - A second form of deposit noticed in some tests consists of 'clumps' of deposited material.
- Widely varying retention in tubes
  - from test to test
  - high retention over short periods of time
- Resuspension can occur for deposits in tubes
  - bounce and break-up of aerosol important
- Large agglomerates did not survive transport at high flows
  - uniform size distribution leaving tube
  - particles smaller than ~1 μm don't break up but larger particles do
- No major retention at rupture site
  - Expected based on studies of rupture propagation

Major observations

- Away from break, most of deposited mass on support plate
  - May be flow recirculation at broached holes for steam generator tubes
  - May not occur for US plants with drilled tube support plates
    - Flow occurs through larger holes; jets
    - Gaps around tubes usually filled with "crud"
- Dryer/Separator not a major source of aerosol retention even for relatively coarse aerosols
  - Fin spacing large and little aerosol diffusion

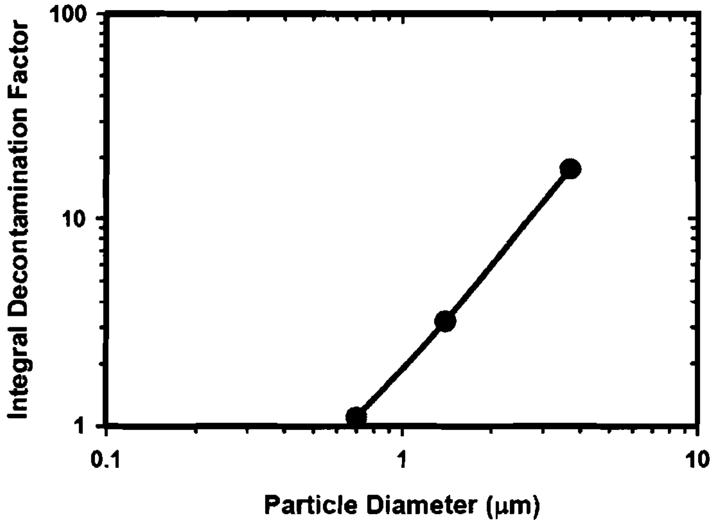
Outstanding issues

- Understanding “bounce”
- Understanding breakup
  - specific to test aerosol?
- Understanding resuspension
  - effect of vibrations
- Features of steam generator
  - Thermophoretic deposition on envelope
- Shapes and sizes of particles coming from the degrading reactor core reaching SG

Changes to MELCOR

- include a “lambda” factor based directly on the ARTIST results
  - based on particle size
  - insufficient risk change incentive to do more in the face of other pressing work
- monitoring 1D model being developed at Ciemat in Spain

ARTIST integral test results



Conclusions

- Expert panel recommendations made for NUREG 1150 risk analyses by and large confirmed
- MELCOR predicts decontamination factors similar to those obtained from ARTIST data.
- Modifications made to MELCOR based on ARTIST data
- ARTIST provides experimental data on source term attenuation on the secondary side of steam generators
  - Steam Generator Action Plan (SGAP) item 3.3a complete

ARTIST

Aerosol Trapping in Steam Generator ARTIST: Findings and Potential Effects on SGTR Risk Profile

ARTIST team:
Salih Güntay, Abdel Dehbi, Steffen Danner, Ralf Kapulla,
Terttaliisa Lind, Hauke Schütt, Detlef Suckow
Paul Scherrer Institut, Switzerland

Outline

- History
- Aims of ARTIST
- ARTIST International Consortium Project
- Facility and scaling
- Model aerosol particles
- Experimental Program and results
- Conclusions
- A new SGTR risk assessment methodology and use of ARTIST data
- Final remarks

History

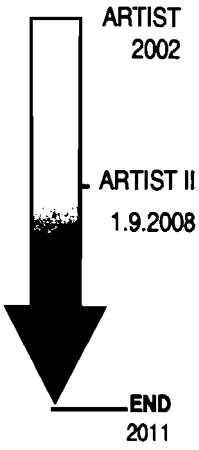
- **Motivation and support from Utility: Large contribution of SGTR in CDF and Risk in NPP-Beznau due to excessive tube problems in 1997**
- **Design and Procurement: 1998-2000**
- **EU 5. Framework Project SGTR: 2000-2002: PSI (Vertical SG without Dryer/separator), VTT (Exp: horizontal SG), NRG , Rez, CIEMAT**
- **ARTIST International Consortium Project**

  Phase I: 2002-2007

  Phase II: 2008-2011
- **Potential continuation >2011: in form of Fundamental Studies (PhD), model development efforts at PSI**

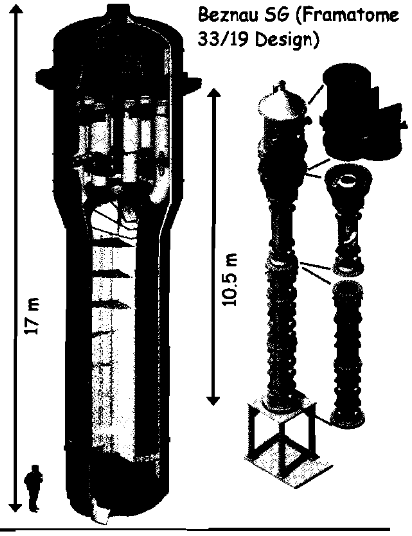
Aims of the ARTIST International Consortium project

- **Provide an international forum to develop new information and share among partners**
- **Produce high quality data for:**
  - Development of fundamental and detailed to simplified and application oriented models
  - Facilitate evaluation of effectiveness of SAMG
- **Develop methodology for SGTR Risk Assessment**
  - Re-assessment of SGTR induced environmental risk
  - Provoke international consensus about the risk significance of SGTR events during DBA and SA
- **Initiate fundamental investigations in form of PhDs/Masters**

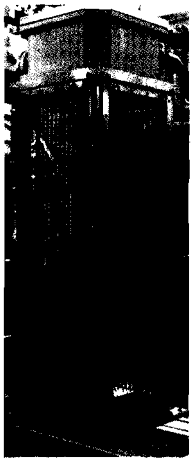


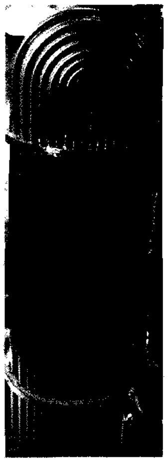
ARTIST Consortium (in alphabetical order)

- o AVN (Belgium)
- o Ciemat (Spain)
- o CSN (Spain)
- o HSK (Switzerland)
- o IRSN (France)
- o JNES (Japan)
- o KK Gösgen-Däniken (Switzerland)
- o NOK, KK Beznau (Switzerland)
- o Nuclear Safety Directorate (UK)
- o Ringhals NPP (Sweden)
- o Universidad Politecnica de Madrid (Spain)
- o University of Newcastle (UK)
- o US Nuclear Regulatory Commission (USA)
- o VTT (Finland)

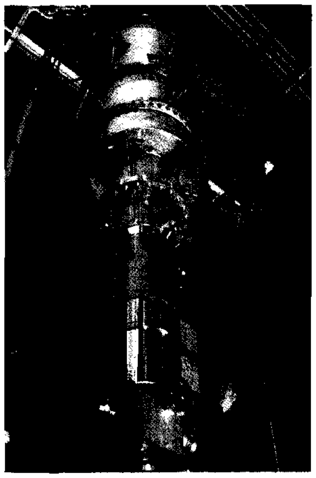

Beznau SG (Framatome 33/19 Design)

ARTIST Facilities


Break stage


Larger scale-bundle


Droplet retention


Integral mock-up facility

Scaling

Design basis: Framatome 33/19 Design

- Separator: 1:1 (steal or mostly transparent)
- Dryer: 1:1 (with actual Chevron panels) (all steel or inlet transparent)
- Bundle: 264 straight tubes, height: 1:0.42, with 1:1 layout
    - Broached support plates with 1:1 layout
- Single tube length: 1:1 with smallest and medium curvatures
- Tube dimensions: 1:1

Flow rates: 40 kg/h to 800 kg/h (fully representative)

Pressure: < 5 bar in primary, ~ 1 bar secondary

Dry conditions (except 1 in-tube test with slight steam condensation)

Model Aerosol Particles

- Evaporation and Condensation generated single/multi component Particles (SnO/CsI/CsOH, etc) (not used for ARTIST due to high costs)
- Fluidization of mono/polydisperse powders (TiO_2 (two types), SiO_2)
- Dispersion of suspended material (Latex, SiO2 in solution) and drying droplets

. Monodisperse particles (SiO_2/Latex): well known size

. Polydisperse particles (TiO_2): lots of problems due to

unknown surface finish characteristics affecting deposition and no size control due to de agglomeration at high velocity/sonic front

Particle Morphology and Size in PWR Hot leg

- Working group: M. Kissane (IRSN), D. Powers (SNL), M. Reeks (NC)
- Very complicated and not resolved issue since many parameters (pressure, core degradation, etc) influence
- Hot leg conditions based on Phébus and other tests
- Phébus:
    - 15-40 % control rod metals, similar amount of oxides, and rest FPs
    - implies an "onion-skin" type of structure where the kernel rich in highly refractory materials and on top condensed species of more volatile species containing cesium and rubidium and perhaps migrated into and interact chemically with the substrate
    - **For practical purpose AMMD at SG inlet or in SG based on impactor data**
        - **3 μm (gsd 2) at 150 °C, 1.7 μm (gsd 2) at 730 °C, 0.1 μm at 930 °C following an exponential increase along inverse temperature**

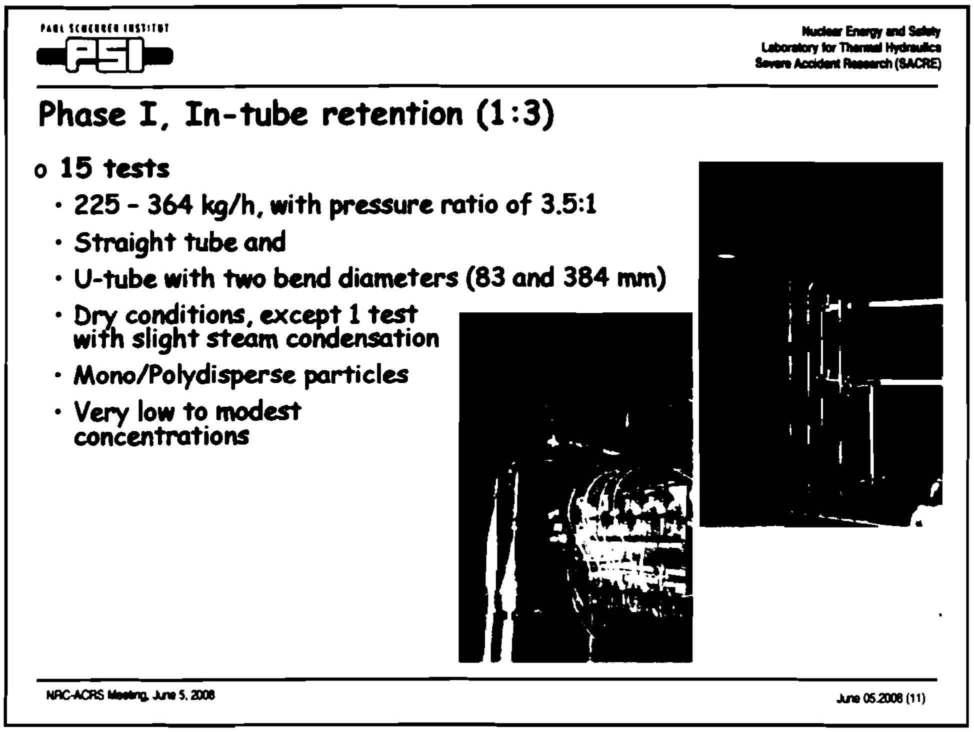
ARTIST experimental program

BDBA source term quantification		ARTIST
Phase I:	In tube	15
Phase II:	Break stage	9(+2)
Phase III:	Far field	8(+2)
Phase IV:	Separator&dryer	5
Phase V:	Flooded bundle	2(+3)
Phase VII:	Integral mock-up	3
	Total	**42(+7)**

DBA source term quantification

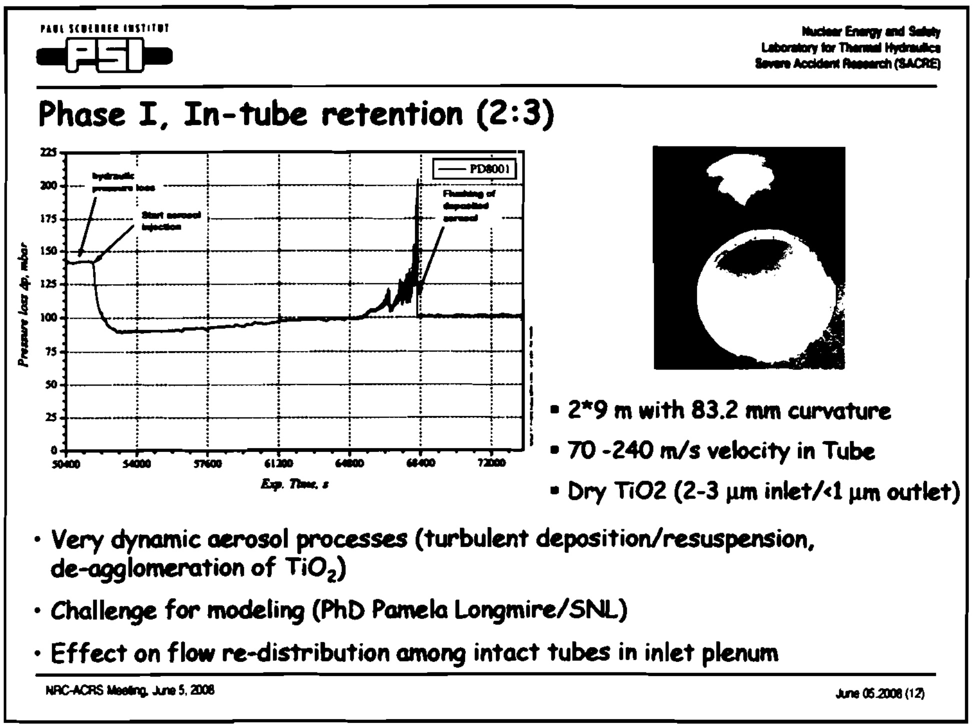
Phase VI:	Droplets (in separator & dryer)	yes

(x): **EU-SGTR**



Phase I, In-tube retention (1:3)

- 15 tests
  - 225 - 364 kg/h, with pressure ratio of 3.5:1
  - Straight tube and
  - U-tube with two bend diameters (83 and 384 mm)
  - Dry conditions, except 1 test with slight steam condensation
  - Mono/Polydisperse particles
  - Very low to modest concentrations



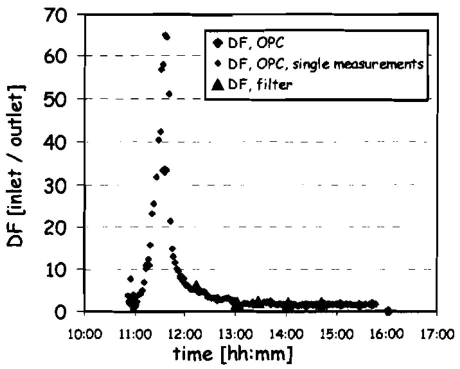
Phase I, In-tube retention (2:3)

- 2*9 m with 83.2 mm curvature
- 70 -240 m/s velocity in Tube
- Dry TiO2 (2-3 µm inlet/<1 µm outlet)

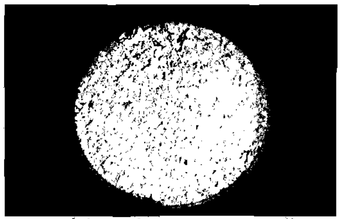
- Very dynamic aerosol processes (turbulent deposition/resuspension, de-agglomeration of TiO_2)
- Challenge for modeling (PhD Pamela Longmire/SNL)
- Effect on flow re-distribution among intact tubes in inlet plenum

Phase I, In-tube retention (2:3)

DF	Conc.	Particles
< 65	medium	SiO_2
1.0 - 2.2	medium	TiO_2
8.2	Slight steam cond.	TiO_2
< 100	very low	SiO_2, latex

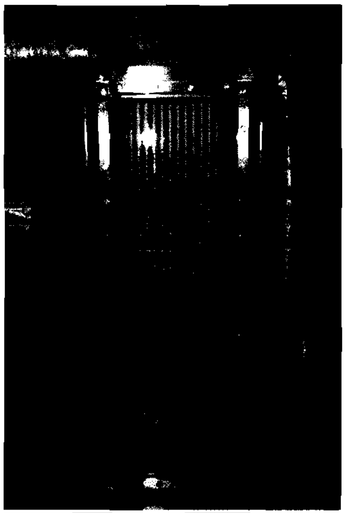


1.4 µm SiO_2, high concentration



Aerosol (SiO_2) fragments collected in the outlet plenum

Phase II: Break-Stage Retention: Dry conditions (1:6)



- **Chocked flow at the break**
- **Guillotine Break**
- **Dry conditions**

9 tests

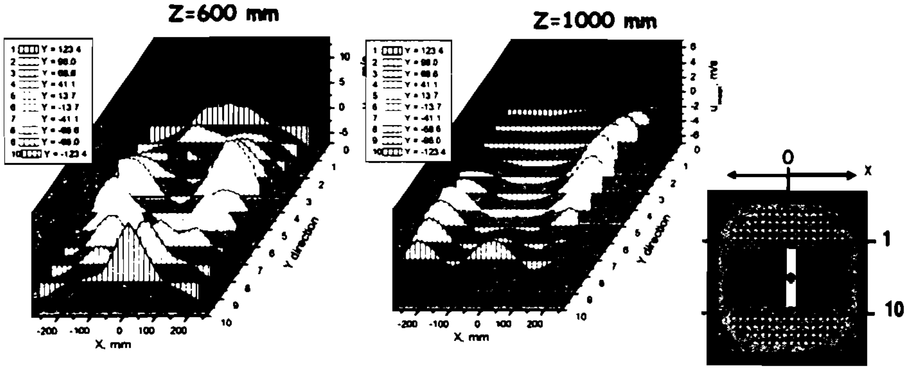
- 360 kg/h,
- Monodisperse SiO2 particles
- AMMD: 1.4 to 3.8 µm

2 tests with full bundle

- 600 kg/h
- **Polydisperse TiO2 particles**
- AMMD: 2.3 µm before break

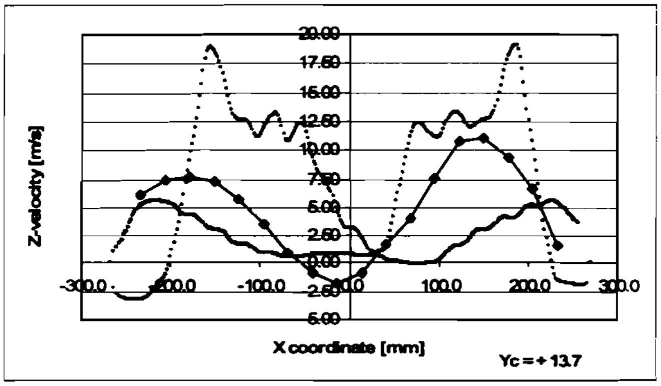
Phase II: Break-Stage Retention: Velocity profiles (2:6)

Measured velocity profile: **Guillotine Break**, 360 kg/h



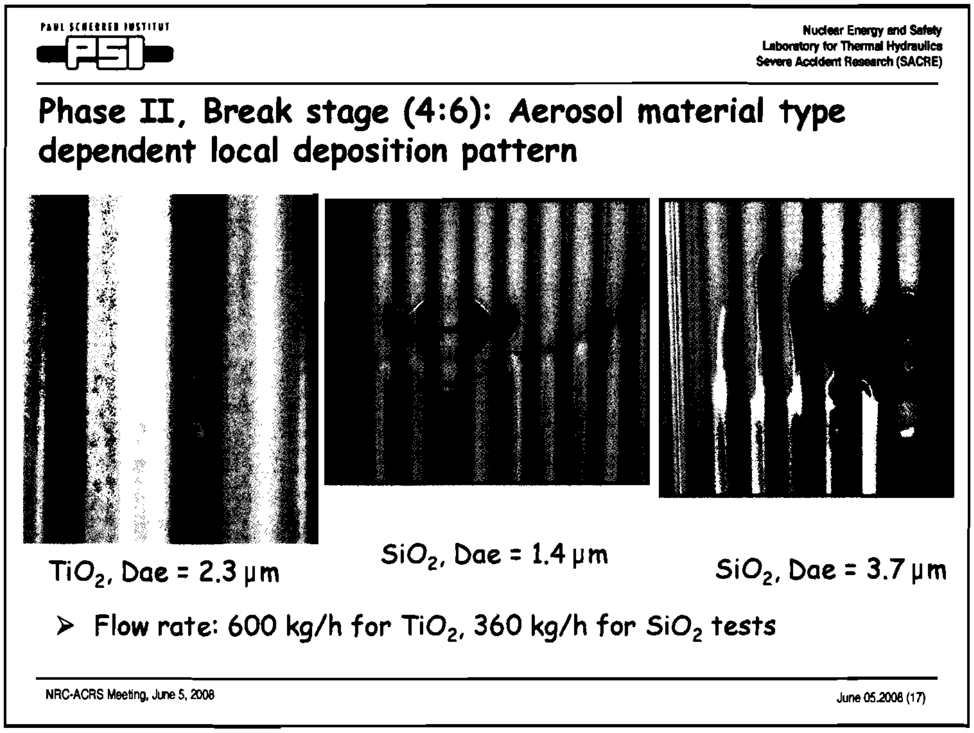
- ➢ Very 3D flow

Phase II: Break-Stage Retention: Velocity profiles (3:6)



- Measured velocity profile

  FLUENT Simulations by Ringhals/EPSILON
- with k-ε
- with Reynolds Stress Model (RSM)



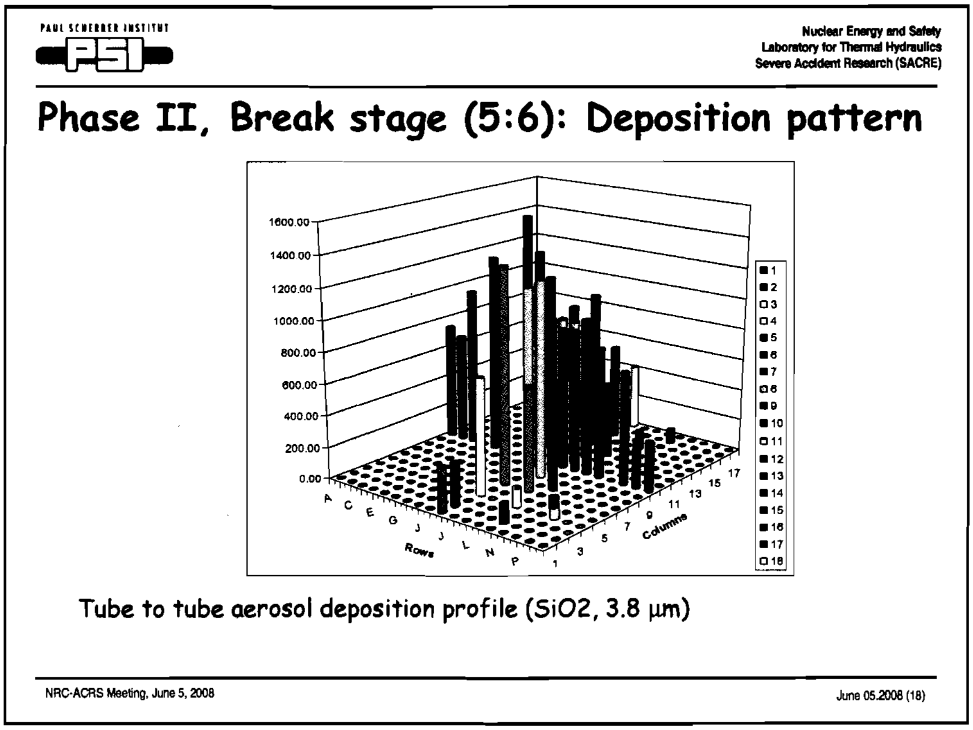
Phase II, Break stage (4:6): Aerosol material type dependent local deposition pattern

TiO_2, Dae = 2.3 µm

SiO_2, Dae = 1.4 µm

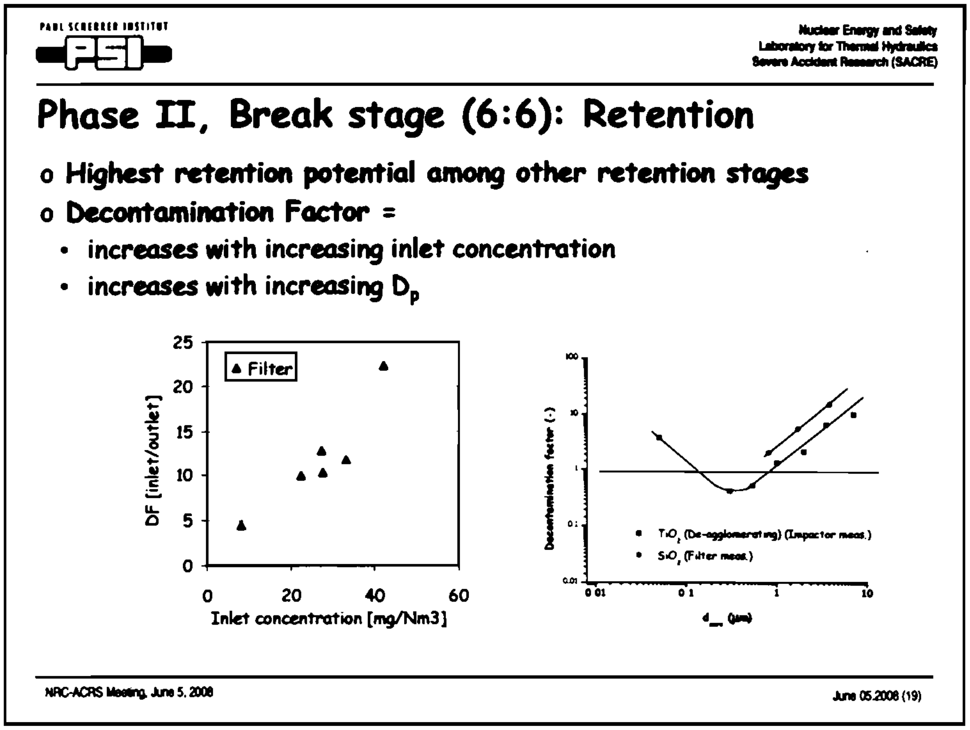
SiO_2, Dae = 3.7 µm

- Flow rate: 600 kg/h for TiO_2, 360 kg/h for SiO_2 tests



Phase II, Break stage (5:6): Deposition pattern

Tube to tube aerosol deposition profile (SiO2, 3.8 µm)



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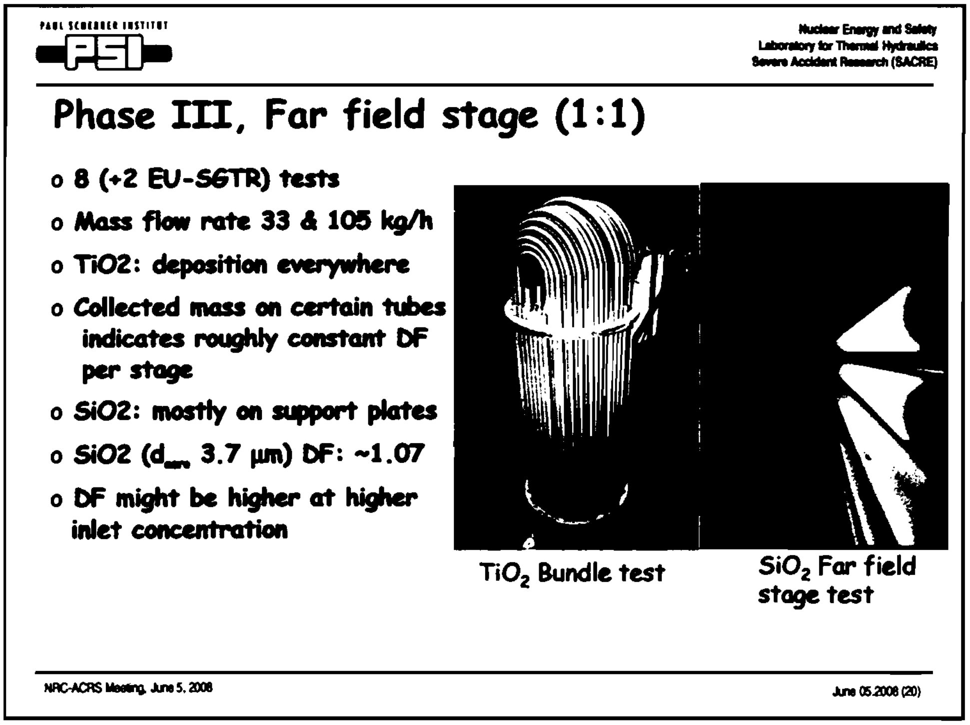
Laboratory for Thermal Hydraulics
Severe Accident Research (SACRE)

Phase II, Break stage (6:6): Retention

- o **Highest retention potential among other retention stages**
- o **Decontamination Factor =**
  - increases with increasing inlet concentration
  - increases with increasing D_p

NRC-ACRS Meeting, June 5, 2008

June 05.2008 (19)



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Phase III, Far field stage (1:1)

- o **8 (+2 EU-SGTR) tests**
- o **Mass flow rate 33 & 105 kg/h**
- o **TiO2: deposition everywhere**
- o **Collected mass on certain tubes indicates roughly constant DF per stage**
- o **SiO2: mostly on support plates**
- o **SiO2 ($d_{[illegible]}$, 3.7 µm) DF: ~1.07**
- o **DF might be higher at higher inlet concentration**

TiO_2 Bundle test

SiO_2 Far field stage test

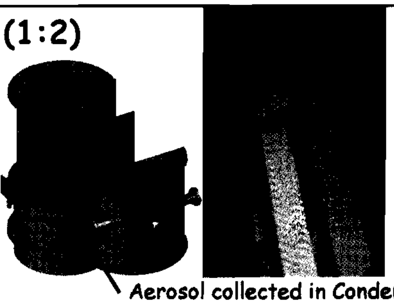
NRC-ACRS Meeting, June 5, 2008

June 05.2008 (20)

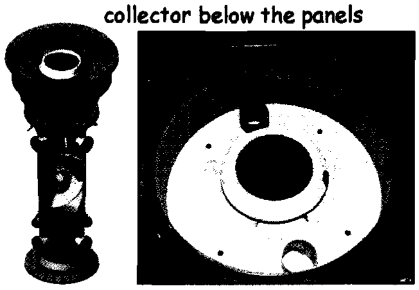
Phase IV: Separator & Dryer (1:2)

- 5 tests (2 only separator)
- Mass flow rate 100, 360 and 650 kg/h
- Local turbulence initiated agglomeration and hence sedimentation
- Decontamination Factor

DF	Particles	D_{ae}
1.2 - 1.4	TiO_2	3 μm, aggl.
1.5 - 1.6	SiO_2	integral mock-up



Aerosol collected in Condensate collector below the panels



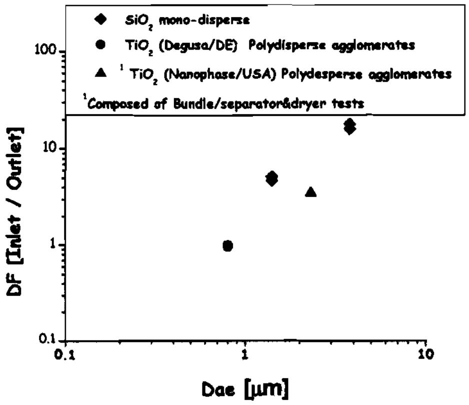
Phase VII: Integral mock-up tests

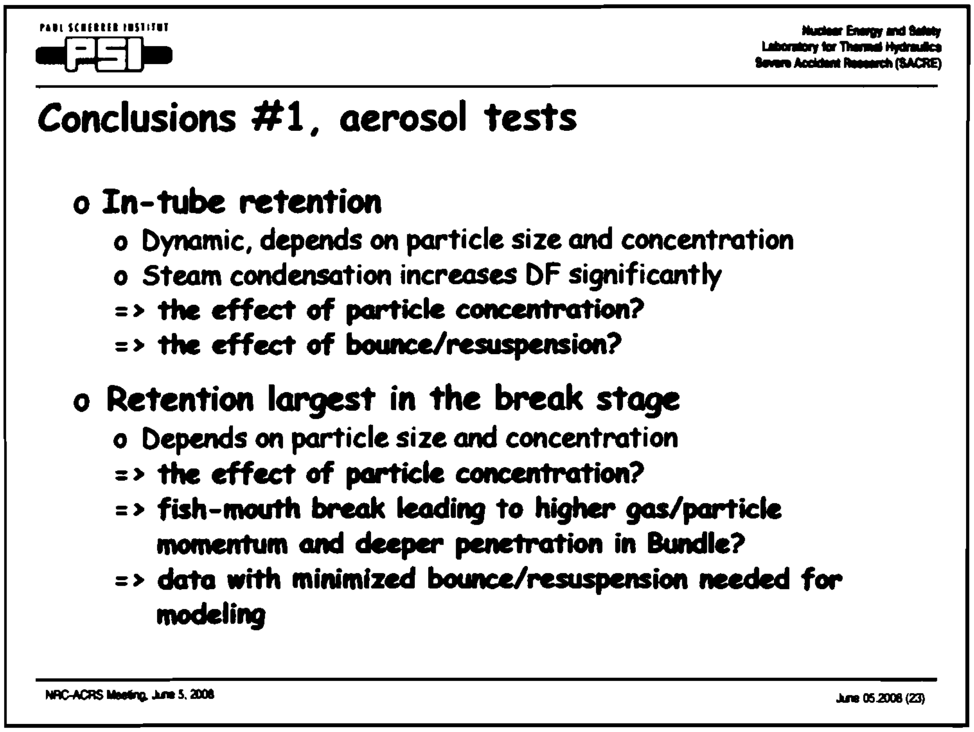
Aim: verify consistency of separate effect data at certain conditions

Decontamination Factor =

- Consistent with Break Stage Tests
- DF increases with concentration
- DF increases with particle size

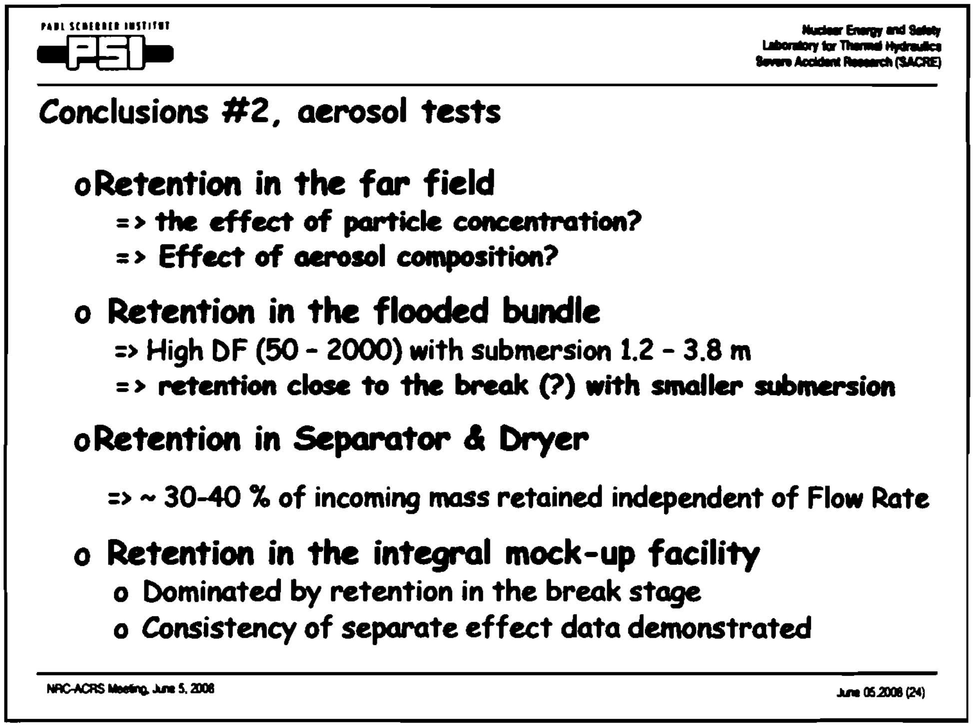
Effect of model aerosol particle material/surface treatment





Conclusions #1, aerosol tests

- o **In-tube retention**
  - o Dynamic, depends on particle size and concentration
  - o Steam condensation increases DF significantly
  - => **the effect of particle concentration?**
  - => **the effect of bounce/resuspension?**
- o **Retention largest in the break stage**
  - o Depends on particle size and concentration
  - => **the effect of particle concentration?**
  - => **fish-mouth break leading to higher gas/particle momentum and deeper penetration in Bundle?**
  - => **data with minimized bounce/resuspension needed for modeling**



Conclusions #2, aerosol tests

- o**Retention in the far field**
  - => **the effect of particle concentration?**
  - => **Effect of aerosol composition?**
- o **Retention in the flooded bundle**
  - => High DF (50 - 2000) with submersion 1.2 - 3.8 m
  - => **retention close to the break (?) with smaller submersion**
- o**Retention in Separator & Dryer**
  - => ~ 30-40 % of incoming mass retained independent of Flow Rate
- o **Retention in the integral mock-up facility**
  - o Dominated by retention in the break stage
  - o Consistency of separate effect data demonstrated

Transport/Removal of Activity in Steam Generator

- SGTR concurrent with core damage involves:
  - Major activity in vapour form at SG inlet
  - Rest of activity and inactive material in aerosol form
- Transformation of activity in vapour form by vapour condensation dependent on local temperature
- Removal of some fraction of vapour by condensation on structure surface
- Transport/removal of Rest of vapour of condensed on particles or form new particles dependent on aerosol removal/transport process

ARTIST addresses only aerosol removal/transport process in SG

Motivation for a new SGTR risk assessment methodology

- MELCOR contains models for vapor/aerosol behavior but lacks specific aerosol transport/removal in SG complex structures at relevant thermal-hydraulic conditions
- For risk assessment with many hundred variations to consider uncertainties: MELCOR is too expensive
- A fast running lump parameter model including Monte-Carlo sampling for uncertainties under development
- Preliminary sample analysis demonstrates the strength and provides feasibility of SGTR risk reduction

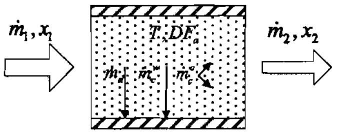
A new SGTR risk assessment methodology

- Lump Parameter Model tracking vapor/aerosol phases in each release path in SG secondary side with:
  - T/H and Vapor/aerosol boundary conditions and uncertainties from SA code predictions
  - Temperature dependent ultimate particle size based on Phébus tests
  - Temperature dependent vapor fractions of released classes including all species from SOPHAEROS code (IRSN/FR) analysis
  - Release path dependent ARTIST DFs (d_p, c)
- Monte-Carlo sampling for all uncertainties
- APET for all SGTR sequences
- Running Model for each APET branches for determination of risk

Lump Parameter Model: Key Aspects

- Accounts for aerosol behavior in complex structures of SG at hydrodynamic conditions by use of ARTIST data for each SG retention stage
- Accounts for vapor conversation using temperature dependent vapor fraction data base generated from SOPHAEROS code runs
- Accounts for vapor fraction condensed on structure and converted to particles by user input including its uncertainty
- Accounts for temperature dependent aerosol size determined by measured sizes in hot leg in all Phébus tests with AgInCd
- Neglects other processes playing a secondary role: thermophoresis, diffusiophoresis,...

Lump Parameter Model Description



$$\dot{m}_2 = \frac{1 - x_1(1-\alpha)}{DF_a - x_2(DF_a - \alpha)} \dot{m}_1$$

α: Vapour split fraction on walls/ particles = 0.5 (0.1-0.9)

DFa: ARTIST DF

m: mass flow of release class (I, Cs, ..)

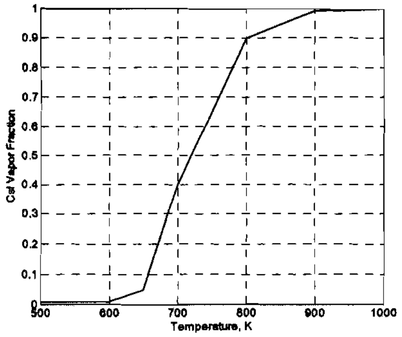
X: vapor fraction of the mass flow

T: Gas temperature

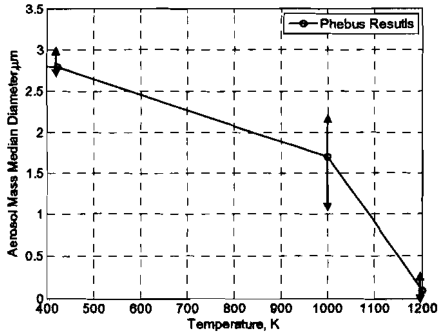
1: donor volume

2: current volume

Lump Parameter Model Data Base (1:3)

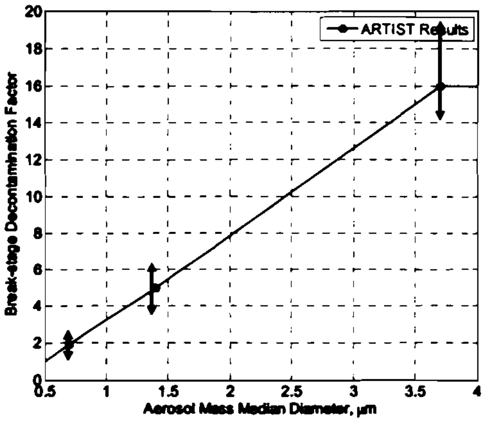


Vapor fraction data base generated from SOPHAEROS code runs



Particle size as measured in all Phébus tests with AgInCd

Lump Parameter Model Data Base (2:3)



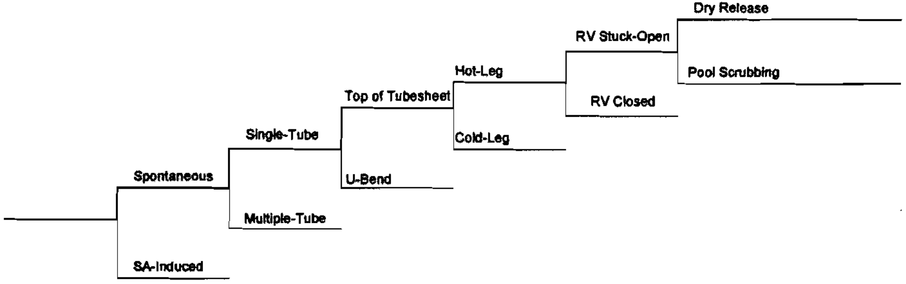
ARTIST Break Stage Particle Size Dependent DF

Lump Parameter Model Data Base (3:3)

Retention Stage	DF	Error Factor	Source
Reactor vessel	1.2 (I), 1.8 (*Cs*)	1.06 (I), 1.04 (*Cs*)	Phébus
Primary circuit	1.1 (I), 1.2 (*Cs*)	1.09 (I), 1.2 (*Cs*)	Expert judgment
In-tube retention	Time variant	1.5	ARTIST
Break stage	Aerosol-size variant	1.5	ARTIST
Far-field stage I-VII	1.05	1.21	ARTIST
Top of shroud	1.20	1.09	Expert judgment
Separator	1.20	1.06	ARTIST
Recirculation	Model	Model	MELCOR, SR5
Downcomer	1.10	1.05	Expert judgment
Intra-volume	1.10	1.07	Expert judgment
Dryer	1.20	1.09	ARTIST
Dome	1.10	1.05	ARTIST

Multiple SA Code Analyses for Model Uncertainties for the same APET Branch

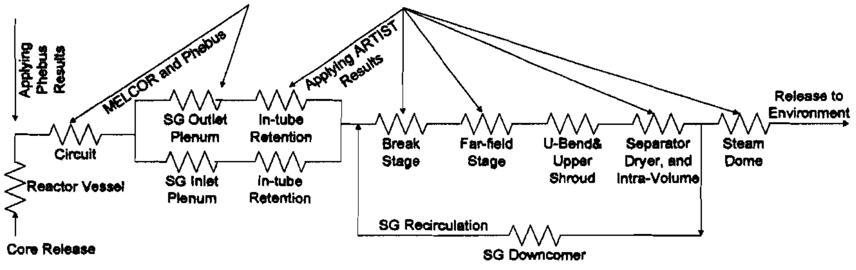
SGTR Frequency	Category	Rupture Size	Rupture Location		Accident Mitigation		
	No SA-Induced SGTR	No Multiple-Tubes	No U-Bend Rupture	No Cold-Leg Rupture	No Reclosed RV	No Refilled SG	[illegible]
Node A	B	C	D	E	F	G	H



SGTR Accident Progression Event Tree

Retention Stages from Core to SG Steam Outlet

- *For each APET sequence, consider a series of retention stages in the fission product release path from the core to the environment*
- *For retention stages of the SG, the lumped parameter model is used*



Multiple SA Code Results: An example

Temperature predictions from MELCOR and SCDAP/RELAP5

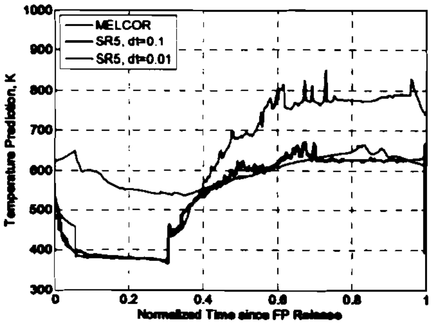
Running multiple cases to estimate the temperature distribution

- SGTR sequence from NPP - Beznau PSA L2
  - SRV stuck-open at the affected SG
  - SRV opened manually at the intact SG at core exit temperature>923K
- Calculation stops at lower head failure

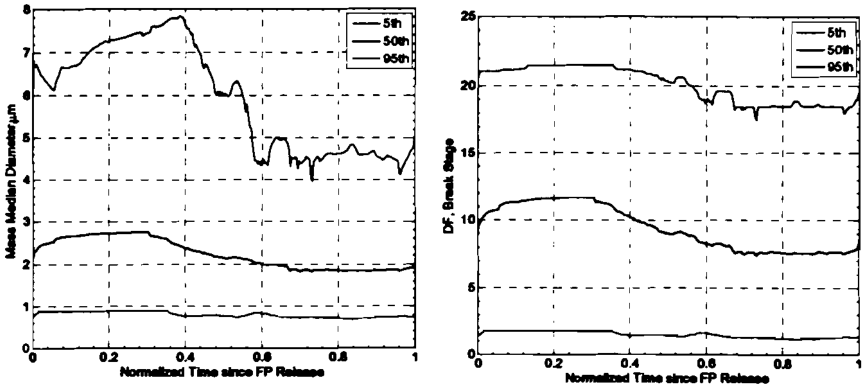
(a) MELCOR

(b) SR5, dt=0.1
SCDAP/RELAP5, max. time step=0.1s

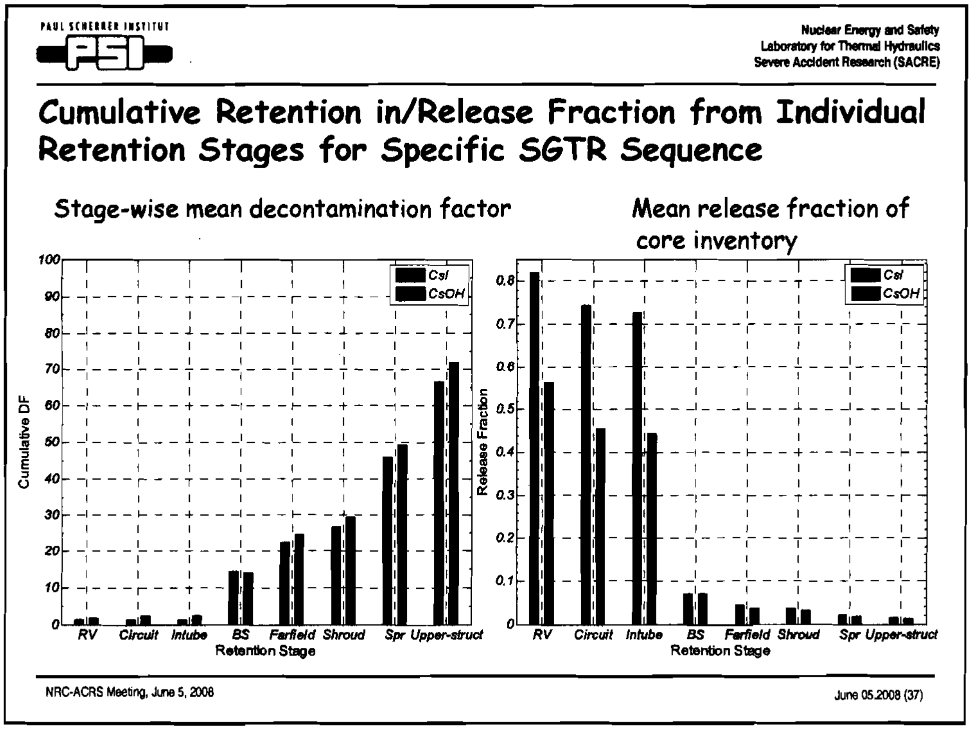
(c) SR5, dt=0.01
SCDAP/RELAP5, max. time step=0.01s



Monte-Carlo Simulation: Examples of 90% confidence interval of Particle Diameter and Decontamination Factor in Break Stage



Presentation not given.



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Cumulative Retention in/Release Fraction from Individual Retention Stages for Specific SGTR Sequence

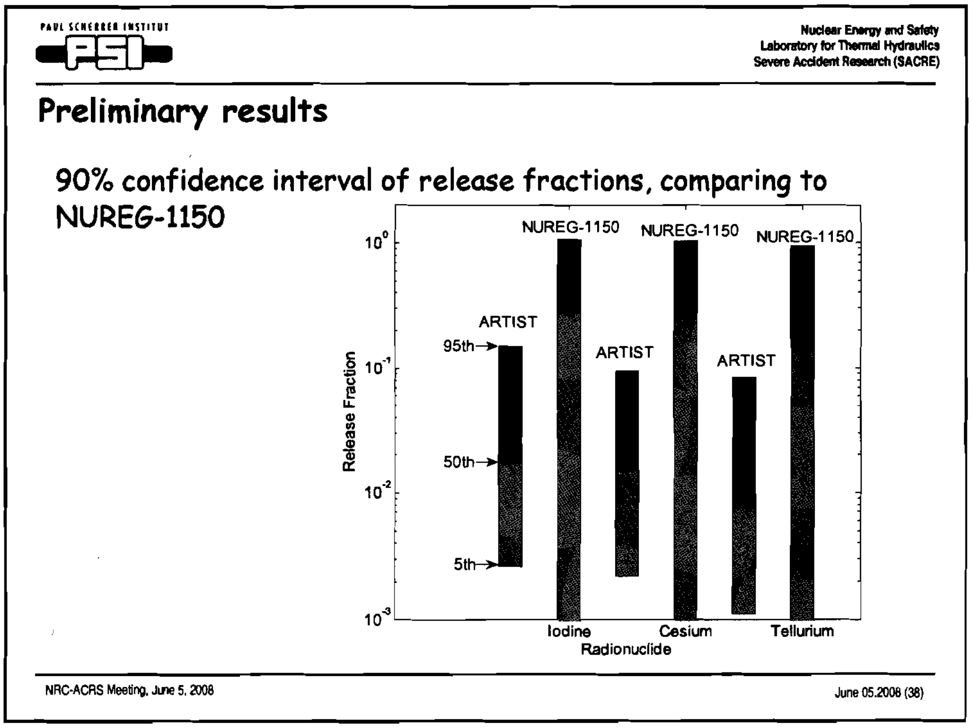
Stage-wise mean decontamination factor

Mean release fraction of core inventory

NRC-ACRS Meeting, June 5, 2008

June 05.2008 (37)

Only slide referred to →



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Preliminary results

90% confidence interval of release fractions, comparing to NUREG-1150

NRC-ACRS Meeting, June 5, 2008

June 05.2008 (38)

Assessment of Methodology (1:2)

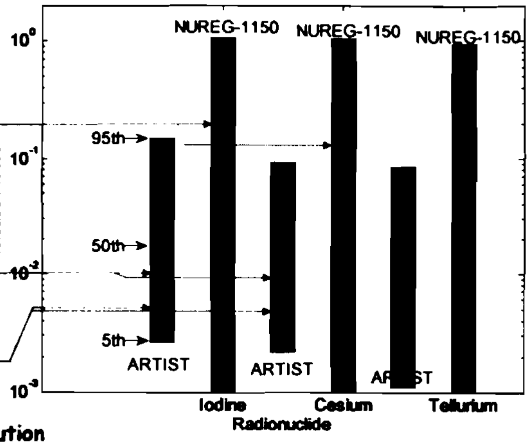
- MELCOR 1.8.6 runs for point estimates of source term
  - use of ARTIST data through „filter function"
  - Superimposing user input „aerosol size" to overwrite MAEROS

- Three MELCOR runs
  - Standard MELCOR 1.8.6 for the same SGTR sequence
  - MELCOR 1.8.6 with ARTIST DFs
  - MELCOR 1.8.6 with ARTIST DFs + PHÉBUS inferred temperature dependent particle size

With MELCOR default vapor and aerosol physics

Assessment of Methodology (1:2)

Comparison of PSI-Risk Model Results to MELCOR Point Value Estimates



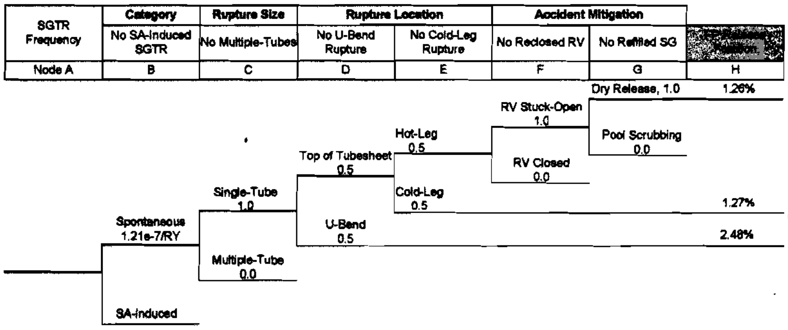
Point estimate of MELCOR default

Point estimate of MELCOR using MAEROS with incorporation of ARTIST DFs

Point estimate of MELCOR using PHEBUS[1] with incorporation of ARTIST DFs

[1] superimposing particle size distribution

APET: branching fractions



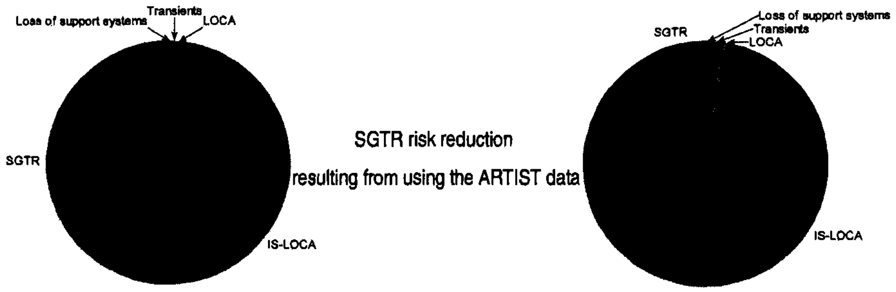
Case presented

Other cases not presented

SGTR Accident Progression Event Tree

Preliminary Risk Profile of NPP-Beznau Spontaneous SGTR

Comparison of the SGTR (without SG Reflooding) Risk significance to other internal initiating events for the Beznau NPP



SGTR risk reduction
resulting from using the ARTIST data

NPP Beznau: PSA L2 BERA: 2002

Conclusions

- Methodology consistent with Point values from MELCOR
- Further development for inclusion of other dependencies and their uncertainties (e.g., DF (dp, C)
- Generic model requires user to input from plant specific SA analysis
- APET to be revised with plant specific information (frequencies, split fractions)

Final Remarks

- PSI data supported by additional data from CIEMAT (Spain) for break stage retention and from VTT (Finland) for in-tube deposition/resuspension, both at low flows
- CFD Simulations of flow^1 and particles^2 by CFD (FLUENT) by Ringhals, AVN^1, CIEMAT^1, JNES ^{1,2} and NRC^{1,2} (Sandia)
- Model development for aerosol removal in flooded bundle (IRSN) and in break stage (CIEMAT)
- 4 PhDs (de-agglomeration, aerosol motion through DNS+LES, bubble hydrodynamics in bundle) at PSI
- 3 PhDs (removal in far field, break stage hydrodynamics, aerosols) at UPM and CIEMAT
- 1 PhD (particle motion in SG pipe) at Sandia
- 1 masters (flow fields by CFD in Separator) at AVN
    - ➤with involvement of 7 Universities

PSI thanks for all supporting and participating organizations in ARTIST

Phases V and VI: Flooded Bundle and Droplet Retention in Separator & Dryer

NRC does not participate in ARTIST Project Phases V and VI, however, the following information is introduced for those in ACRS who have interest in the Aerosol Scrubbing in Bundle Environment from High Jet Flows and Dissolved Activity (Iodine, mostly) Retention/Release by Droplets during the initiation of aSGTR event

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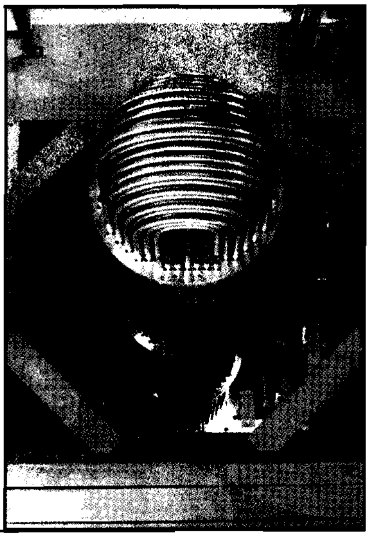
Phase V: retention in the flooded bundle (1:2)

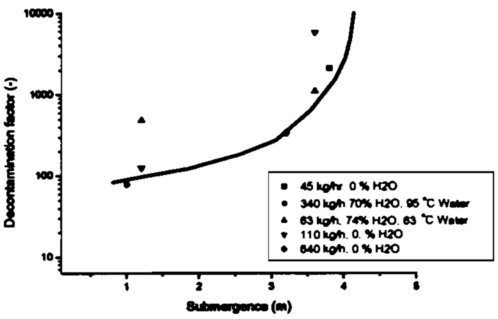
- o **2 tests (+3 EU-SGTR)**
- o **Decontamination Factor**
- o Determined for relatively large submersion

DF	flow rate	submersion
2 100	45 kg/h	3.8 m
335	640 kg/h	3.2 m



Phase V: retention in the flooded bundle (2:2)

- Very high DF due to bundle-hydrodynamic interactions, especially at the break; models not able to reproduce DF
- Aerosol removal in hot pools without bundle: ~ **DF 20** (PSI - POSEIDON, 1991- 1996)



tests	Main features	Submergence m	Experimental DF	IRSN Model DF
A82	Steam, hot, medium flow rate	1.3	60-100	362
A83	NC, cold, low flow rate	1.2 2.3 3.6	124 1251 6730	37 64 68
E84	NC, cold, low flow rate	3.60	2887	46
E86	NC, cold, high flow rate	3.20	271-405	67

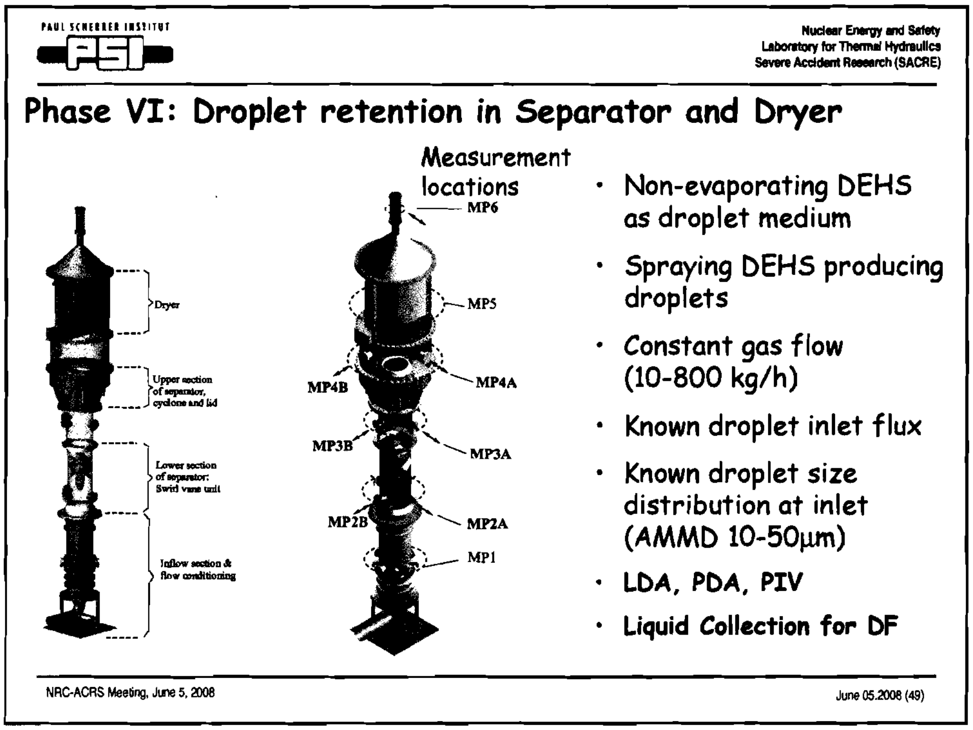
Iodine Source Term during Steam Generator Tube Rupture Initiated Design Basis Accidents: Introduction

- **Spontaneous or initiated Steam Generator Tube Rupture**

  => activity release until the operators can reduce the RCS pressure to the secondary side level

  => activity release at least 30-40 minutes (so-called "grace period")
- **SGTR event is a design basis event**
- **The amount of activity release controlled by:**

  a) amount of dissolved activity in the primary system (leaking rods, iodine spiking (reactor trip) and pressure change)
  b) the submergence of the leak; single or multiple tube ruptures; total break flow
  c) pH and iodine chemistry in the secondary side
  d) iodine mass transfer from the boiling pool
  e) **The break at the tube bend**
  <= 80-85 % of primary water in droplet form as a result of flashing
  => efficiency of separator and dryer to retain droplets
  ➔ ARTIST - Phase VI



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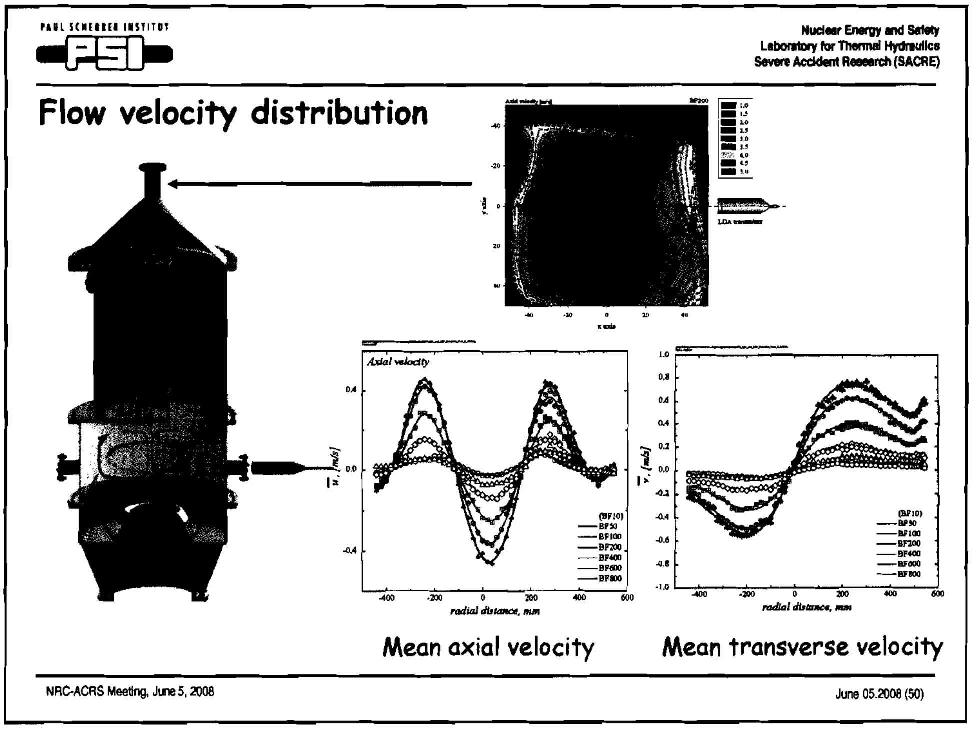
Phase VI: Droplet retention in Separator and Dryer

- Non-evaporating DEHS as droplet medium
- Spraying DEHS producing droplets
- Constant gas flow (10-800 kg/h)
- Known droplet inlet flux
- Known droplet size distribution at inlet (AMMD 10-50μm)
- **LDA, PDA, PIV**
- **Liquid Collection for DF**



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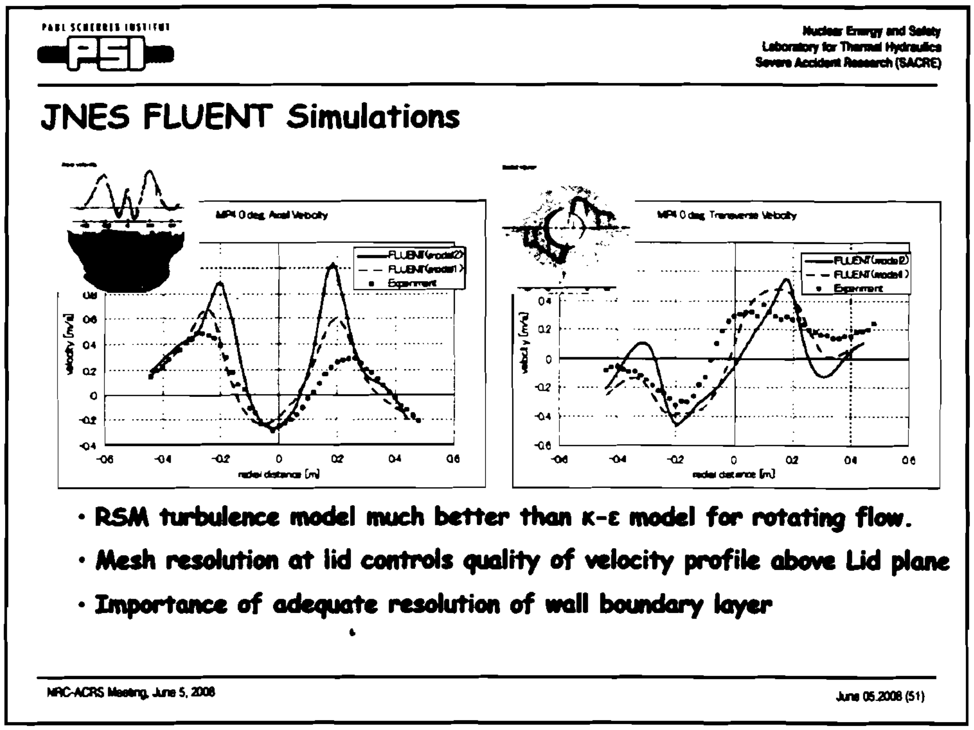
Flow velocity distribution

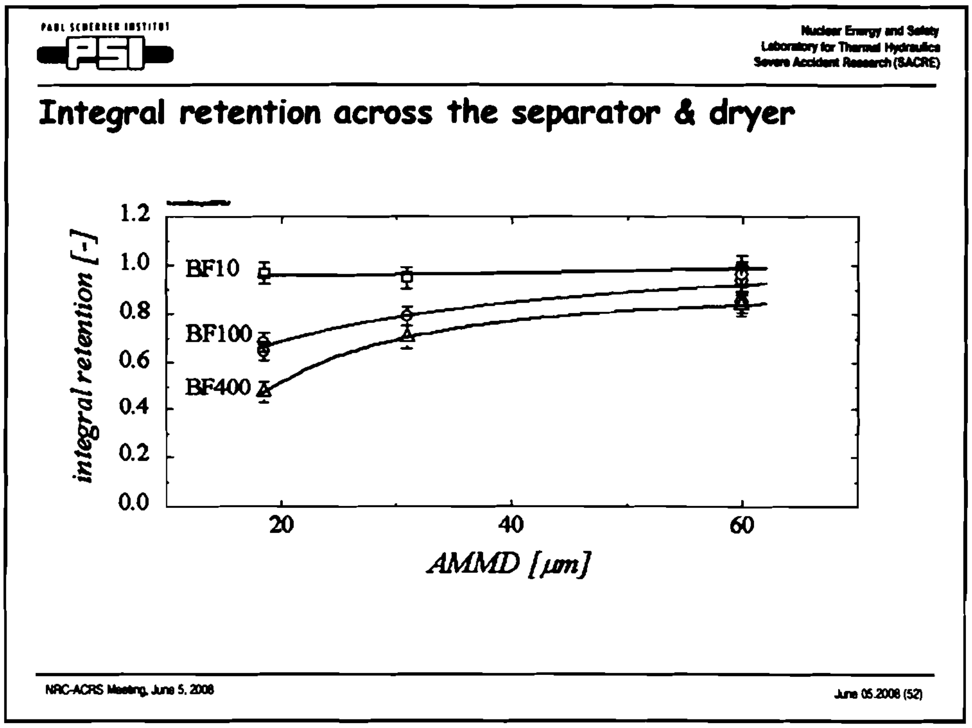
Mean axial velocity

Mean transverse velocity



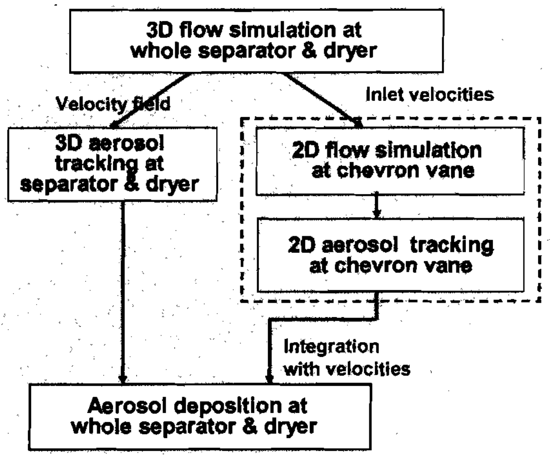
JNES FLUENT Simulations

- **RSM turbulence model much better than κ-ε model for rotating flow.**
- **Mesh resolution at lid controls quality of velocity profile above Lid plane**
- **Importance of adequate resolution of wall boundary layer**



Integral retention across the separator & dryer

Particle Decontamination by FLUENT with PSI discrete-particle tracking model (JNES)


3D flow simulation at whole separator & dryer

Velocity field

Inlet velocities

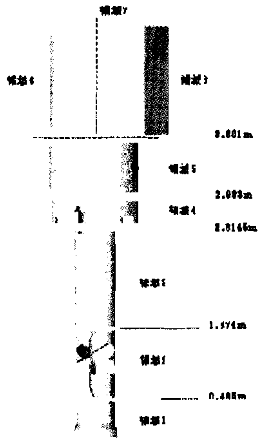
3D aerosol tracking at separator & dryer

2D flow simulation at chevron vane

2D aerosol tracking at chevron vane

Integration with velocities

Aerosol deposition at whole separator & dryer



Particle Decontamination by FLUENT with PSI discrete-particle tracking model (JNES)

DF (300kg/h)

	1 μm	3 μm	10 μm
Separator	1.25	1.32	1.35
Dryer	1.09	1.14	1.25
Total	1.36	1.51	1.68

- Capturing hydrodynamic behavior is crucial prerequisite for aerosol behavior
- PSI discrete-particle tracing considers particle turbulence based on DNS simulations
- JNES predicted Overall retention is in agreement with Phase IV test results