Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards 496th Meeting
Docket Number:	(not applicalble)
Location:	Rockville, Maryland
Date:	Thursday, October 10, 2002

Work Order No.: NRC-581 Pages 1-263

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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

496TH MEETING

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THURSDAY

OCTOBER 10, 2002

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ROCKVILLE, MARYLAND

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The Committee met at 8:30 a.m. in Room T2B3, Two White Flint North, Rockville, Maryland, George E. Apostolakis, Chairman, presiding.

ACRS MEMBERS PRESENT:

GEORGE APOSTOLAKIS	Chairman
MARIO V. BONACA	Vice-Chairman
F. PETER FORD	Member
THOMAS S. KRESS	Member-at-Large
GRAHAM M. LEITCH	Member
DANA A. POWERS	Member
VICTOR RANSOM	Member
STEPHEN L. ROSEN	Member
WILLIAM J. SHACK	Member

ACRS MEMBERS PRESENT: (CONT.)

JOHN D. SIEBER Member

GRAHAM B. WALLIS Member

ALSO PRESENT:

JOHN T. LARKINS Executive Director, ACRS

I-N-D-E-X

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

8:44 a.m.

CHAIRMAN APOSTOLAKIS: The meeting will now come to order. This is the first day of the 496th Meeting of the Advisory Committee on Reactor Safeguards. During today's meeting, the Committee will consider the following. The confirmatory research program on high-burn-up fuel, CANDU reactor ACR-700 pre-application review, the Subcommittee report on Catawba and McGuire License renewal applications, policy issues related to advanced reactor licensing and proposed ACR reports. This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act. Dr. John T. Larkins 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 sessions. A transcript of portions of the meeting is being kept, and it is requested that the speakers use one of the microphones, identify themselves and speak with sufficient clarify and volume so that they can be readily heard.

There are a few of items of current

interest. Dr. Gus Kronenberg, ACRS Senior Fellow, will be leaving the ACRS on October 18, 2002. He has provided outstanding technical support to the ACRS on numerous issues, including power uprate review process, reactor fuels, risk-informed and performance-based regulations, genetic safety issues and advanced reactors. The ACRS appreciates the support provided by Gus and wishes him well in his future endeavors. Where is Gus? Stand up.

(Applause.)

CHAIRMAN APOSTOLAKIS: We have two new senior staff engineers who joined our staff on October 7. Mr. Ramin Asa, from the Office of Nuclear Reactor Regulation, joined us. He has been with the NRC since 1991. Before joining the NRC, he worked at Consolidated Edison Company for seven years. He has a Bachelor's degree in nuclear engineering and Master's degrees in mechanical engineering and in international management. And he's a licensed professional engineer. Ramin, welcome.

(Applause.)

CHAIRMAN APOSTOLAKIS: Mr. Michael Snodderling, also from the Office of Nuclear Reactor Regulation, joined us on October 7 as a senior staff engineer. He has been with the NRC since 1989.

Before joining the NRC, he was working for Calvert Cliffs Nuclear Power Plant for three years. He has a Bachelor's degree in nuclear engineering. Mike, welcome.

(Applause.)

CHAIRMAN APOSTOLAKIS: Any other comments? Okay. Hearing none, we'll proceed with the agenda. The first item is Confirmatory Research Program on High Burn-up Fuel. Dr. Powers is the cognizant member.

MEMBER POWERS: I am.

CHAIRMAN APOSTOLAKIS: Dana, would you lead us through this complex issue?

MEMBER POWERS: With pleasure.

CHAIRMAN APOSTOLAKIS: Okay.

MEMBER POWERS: We did have a meeting of the Reactor Fuel Subcommittee yesterday with a focus on the issues of high burn-up fuel. Those of you that were not able to attend missed a real treat. It was like many of our high burn-up fuel meetings, an information-packed, highly technical discussion of this most important issue.

I think most of the members are aware that there is a tremendous economic driving force to take fuels up to higher levels of burn-up. I think they're also aware that there are societal benefits as well. Using fuel for higher burn-up, one has less fuel to store, less fuel to dispose of. So there is a tremendous pressure to use fuel at ever higher burn-ups in the existing fleet in nuclear power reactors. Of course, we've reached the point at which the fuel is being used at levels of burn-up that exceed our empirical database on how that fuel will behave under upset conditions. And the members, I believe, are aware that the first tests in France, and subsequently tests on Japan, on the response of fuel to the reactivity insertion showed that perhaps some of the criteria we use for fuel failure and fuel coolability to reactivity insertion in the licensing process were not adequate to treat these high burn-up fuels. And NRC has limited the burn-up levels that plants can take fuel, pending the available of additional of technical information.

In making the decision, the Agency also put together a research program to confirm the suitability of this limit in preserving the health and safety of the public. That research program is looking not only at the reactivity insertion for high burn-up fuel but also the response fuel of our loss coolant accidents and boiling water ATWS events. The selection of what accidents to consider for research that was done on a risk-informed basis that I thought was an excellent use of risk information to guide a research program.

At the Subcommittee meeting, we covered a tremendous amount of material, and of course at this meeting I'm only going to give you a snapshot focused primarily on the research program and some new activities undertaken at NRR. We did, however, at the meeting have a very delightful presentation from the Electric Power Research Institute on their investigations of the reactivity insertion accident tests that have been done to date. They have been examining this database which is something in excess of 50 tests and have developed a hypothesis on how to explain what fuel rods fail when there is a sudden energy input from a reactivity insertion. This hypothesis is used on the strain energy density and the cladding of the fuel; that is, focusing on the clad ductility rather than just the fuel itself.

They have developed a correlation of strain energy -- the critical strain energy density for failure based on correlating it with the extent of clad oxidation. In reality, probably use hydrogen for cladding, but since you don't have access to the hydrogen concentration in the cladding fuel, they have usually served with only oxidation to describe its underlying physics.

They have also developed a separate explanation for when the fuel becomes sufficiently damaged that capability comes into it. Look at those as two separate issues. They've developed a fairly detailed basis for this correlation and have submitted that as a topical report to the Agency for review, and you're going to hear at the end of today's presentation about the Agency's plans to review that material.

I can't say that there is a complete consensus between the research staff on the details of these explanations. There does seem to be a consensus that the ductility of the clad is an issue for the reactivity insertion event that the criteria for fuel failure is kind of almost like a burn-up. There is an emerging consensus that the original test that started this all, the test in France called REP Na or REP Na-1, may well be an outlier and that it will not fall on all the correlations that are developed.

This EPRI work, as I indicated, is fairly well-developed and being reviewed. We're not choosing to present to the full Committee now; rather we're reserving that for when we have the benefit of our review of the material in comparison with the evaluation report. EPRI also has what they call a Robust Fuels Program, and they have volunteered sometime after the 1st of the year to come to the ACRS and explain that entire program, which should be very pertinent to us in a variety of different areas.

What we want to focus on today is the RES Program itself, which is a program that we've followed closely and have endorsed over the years. This is a fairly comprehensive program that evolved with the collaboration between RES, EPRI and a number of international partners. And they are looking not just at the reactor and insertion accidents but also LOCA accidents, ATWS accidents and even the storage of spent high burn-up fuel.

With that introduction, I'll turn to Ralph Meyer to give us what can only be a synopsis of a fairly elaborate experimental and analytic program.

MR. MEYER: Good morning. I want to tell you about our research work, but I'd like to do it in the context of a document. It's a program plan that we put together in 1998, in which we identified some issues. The research program was then structured around these issues to try and get some resolution on them.

We are in the process of developing an updated program plan which may go beyond the scope of the current program plan. The update is not ready for public display yet, so we've decided to go back to the '98 program plan, look at the issues and tell you what progress we've made, and which issues have been resolved out of those original ones.

I think this will then display much of the research work that is going on at the present time. This is the list of issues that was in the 1998 high-burn-up plan. There were nine of them, cladding, integrity and high-burn-up during normal operation was the first issue.

Incomplete control rod insertion, you may recall was an issue. The matter of the acceptance criteria for the reactivity accidents that Dana mentioned was another one. Then there was the matter of the loss of coolant accident, where we have embrittlement criteria in 10 CFR 50.46 and evaluation models in Appendix K, and whether or not those are effected by burn-up.

When we looked at the risk numbers for the various accidents for the BWR, it looked like the rod drop accident, which is the corollary to the PWR rod ejection that we studied, was of lower consequence than the power spikes that you get during the oscillations associated with anticipated transient without SCRAM.

So we decided to look into fuel behavior during those power oscillations to see if our current understanding was effected by the burn-up that we're now experiencing.

At that time, we were using some fuel rod codes to audit vendor submittals, and our fuel rod codes were not able to handle burn-ups up to the range of 62 to 65 gigawatt days per ton, so it was an issue to improve these codes to handle the higher burn-ups.

MEMBER WALLIS: How do you evaluate this ATWS? No one's actually run a BWR through an ATWS with major power oscillations, and the predictions from the code show all kinds of peaks going on for some time.

It's not clear whether those are realistic or only a factor of the code, so knowing just what's going on in ATWS itself is not something we're very secure about.

MR. MEYER: What I'd like to do is, I will say more about three, four, five, six, seven and eight --

MEMBER WALLIS: Later on?

MR. MEYER: Just after this slide. So let's just finish. We had a source term in NUREG 15.65 where the document itself said it might not be applicable above 40 gigawatt days per ton.

There was the matter of the dry cask for shipment, which had been reviewed only up to 45 gigawatt days per ton, and now we are discharging fuel at around 60, 65.

And then the question of whether we would need enrichments greater than five percent. So in the original document we dealt more or less finally with number one, number two and number nine. I'm not going to say any more about those.

I will now run through the rest of them with a couple of slides, to remind you what the issue was and to tell you what we are doing and have done about these.

So the first one is the one that got the most attention yesterday. The issue in a nutshell is whether the fuel damage criteria in Reg Guide 1.77 works with high-burn-up fuel.

We know rather confidently that it does not, and so the real question is what should we substitute for this 280 calorie per gram number.

I'm going to in some subsequent slides

show you how we're approaching this issue, and let me say now that there's a kind of division of effort here, and this was spelled out rather clearly in the original program plan. At the time we wrote the program plan and realized the existence of these high burn-up issues, the Agency had already approved burn-ups up to 62 gigawatt days per ton. So there was some backward looking to do, and we decided that instead of raising it as a backfit issue, that the NRC Staff itself would accept the burden of confirming the adequacy of the decision to go to 62.

Which meant that RES is going to look at the reactivity-initiated accidents and see if we can, in effect, provide a safety analysis that shows that what the appropriate fuel damage limits should be and that the current operating plants remain below those limits. That's the confirmatory work that the Office of Research is doing.

In addition to that, the industry is interested now in going to even higher burn-ups, above 62 gigawatt days per ton, up to about 75, and the numbers that we quote are average for the peak rod in the core.

And we decided in the original program plan that the industry would have full responsibility for developing the data and presenting for our review criteria to be used in place of this 280 calories per gram limit.

So there are two efforts going on simultaneously. There's an RES effort aimed at confirming things up to 62. There's an industry effort aimed at moving from 62 up to 75. And yesterday both of those were presented to the Subcommittee.

So with regard to the RES effort to confirm the situation at 62 gigawatt days per ton, essentially, for the Zircaloy cladding that was the predominant cladding at the time of the program plan, we have a method of doing this, which I'm going to show you, and we have a schedule for doing this, which ends with a confirmatory assessment in early 2005.

And the reason for this schedule is as follows: We're expecting to be on a nice new plateau of understanding at this time, and this will be a good time to make an assessment because it's going to be a long after that before we learn much more.

Basically, we have two or three tests coming out of the Cabri Program, out of the sodium loop late this year and early next year. We have mechanical properties under the right conditions for analyzing this accident, coming out of Argonne National Laboratory in 2003. And there are some very special tests that we're waiting for in the NSRR test reactor in Japan that will be run in 2004 to look at the effects of test temperature on the results. The NSRR reactor has run a very large number of tests, and it's run them all around room temperature, about 25 degrees centigrade. And the accident that we're interested in is initiated at about 300 degrees. So it's the run test temperature, and we want to get a direct measure of that from tests before we make our best estimate of the failure level and complete our assessment. So that's the schedule.

So this is our infamous paintbrush slide. It's somewhat updated since the last time you saw it.

MEMBER WALLIS: Show me where 280 calories per gram is on there.

MR. MEYER: I'm sorry?

MEMBER WALLIS: Where is 280 calories per gram?

MR. MEYER: Two-eighty calories per gram is up here.

MEMBER WALLIS: It's almost above all the data on the map. It's above everything.

MR. MEYER: It's above everything on the

map.

MEMBER WALLIS: So what's the basis for it? Two hundred and eighty is meaningless.

MR. MEYER: Okay. A little history. First of all, 280 is a mistake; it should have been 230.

MEMBER WALLIS: Why isn't it 50?

MR. MEYER: Why didn't it fix?

MEMBER WALLIS: Why isn't it 50? Why isn't it 50?

MR. MEYER: It should be much lower than that, and you can tell that at a glance from this picture. That's point number one of this picture. It isn't 280 calories per gram, and it's not even 230 calories per gram, except for very low burn-ups.

MEMBER WALLIS: Not even there. It's more like 150.

MR. MEYER: Well, now you have to start being careful about your definitions, because what we have plotted on this graph are points that show whether the cladding has failed or not failed. And initially we used a two-level set of criteria. We used a high number, like 230 or 280 calories per gram, as a limit. Don't go above that limit because bad things happen. We used a lower number based either on

DNB or sometimes 170 calories per gram was used to tell you when the cladding failed, and you used that threshold for those calculations, which are relatively inconsequential for this accident, and we're not too much interested in that subject right now.

What we found was that you could experience a cladding split and not expel any fuel until you got up to an energy at which point you started melting UO2. Now, the enthalpy for incipient of UO2 unirradiated is about 267 calories per gram, and if you do that -- if you have that in the middle and you're doing a radial average, the radial average comes out to about 230 calories per gram. So above 230 calories per gram radial average you started having some molten fuel, and molten fuel expands about 40 percent compared to solid UO2. And it can do two things: It can break the cladding apart and it can throw fuel out into the coolant. And when you get fuel out into the coolant, you can now get a fuel coolant interaction with some mechanical energy released. And so the limit was put at 280, which really should have been 230, to keep the fuel -- to make sure that the fuel didn't get ejected out.

Now, at high burn-up, you have a different mechanism for getting fuel out of a crack in the cladding, because you have all this gassy microstructure. And so when you give it a sudden temperature spike, it can expand. So a number of these tests have experienced fuel loss.

So what we've decided to do for the RES exercise of confirmatory assessment at 62 gigawatt days per ton is to take the conservative assumption that we're going to try and show that if we assume that the limit is the cladding failure threshold, which is conservative, you know, if you don't crack the cladding, you can't get any fuel out, you can't have any flow blockage, you can't have any energetic coolant interactions. And if we can find a line somewhere along here which pretty much lower bounds the failure points and if we can then demonstrate that you can't get that much energy from a currently designed PWR, then that's the bottom line acceptance that we're looking for. And I'm pretty sure that we can do that.

MEMBER WALLIS: But there's no high burn-up indicated on this figure.

MR. MEYER: Ah, you're right.

MEMBER WALLIS: You seem to indicate to have a correlation.

MR. MEYER: Let me make another point.

Now, there's also an awful lot of scatter in the data, and that's because the failure enthalpy depends on more variables than just the oxide thickness. It depends on burn-up, it depends on the shape of the pulse.

MEMBER WALLIS: The subject is burn-up, so you ought to put burn-up on your slides.

MEMBER KRESS: Is oxide thickness the surrogate for burn-up here? It could be --

MR. MEYER: In a way. In a way.

MEMBER KRESS: Maybe it's not one to one.

MR. MEYER: We used to plot these data as a function of burn-up. You get a better correlation if you plot them this way because the most important of all of those variables in determining where failure is going to be is the ductility or brittleness of the cladding, which is largely affected by the hydrogen, which comes from the oxidation process. So oxidation is more important than burn-up per se. Now, yesterday we saw EPRI's relation between burn-up and oxidation, and so you can go back and forth between burn-up and oxidation.

CHAIRMAN APOSTOLAKIS: But that depends very much on the material, right?

MR. MEYER: It does. And then the amount

of oxidation depends on the material.

MEMBER WALLIS: Depends on the chemistry of the fluid in the reactor.

MR. MEYER: All of the above. Let me show you one thing that's interesting here is this set of data right here with the 8's on it and with the 5's on it, which should be right in here, fit the pattern very nicely. They're Russian data on E110 cladding that is very lightly oxidized and does not exhibit a brittle failure at all; it's a nice ductile failure.

And it fits right in with the trend that you get as you go down to zero cladding thickness. If you plotted those data on a plot of enthalpy versus burn-up, they would be way out here at 55 gigawatt days per ton, and they'd be way off the charts.

MEMBER KRESS: Do you have now a theoretical line that goes --

MR. MEYER: No.

MEMBER KRESS: -- does that?

MR. MEYER: No.

MEMBER KRESS: -- that has to do with internal stresses and strains? Or is that what EPRI has?

MR. MEYER: No. That's where we're going.

MEMBER KRESS: Okay.

MEMBER RANSOM: Just one --

MR. MEYER: So here is the --

CHAIRMAN APOSTOLAKIS: There's a question.

MEMBER RANSOM: Quick question, Ralph?

MR. MEYER: Yes, I'm sorry.

MEMBER RANSOM: Is this the criterion that's used to decide whether core damage has occurred or not as far as determining the core damage frequency and accident analysis?

MR. MEYER: It's part of a design basis accident, so there's a requirement that you do -- in the safety analysis that you analyze this accident and demonstrate that you do not exceed this limit.

MEMBER KRESS: It's not the definition of core damage frequency that you generally see. That's a much more severe thing, core damage frequency.

MR. MEYER: Okay. So we're going to try and get a less ambiguous line that includes the right variable effects. And we're going to attempt to do this three different ways, and they're not real clear from this slide, but I'll explain what they are.

One of them is analytical. We have a code called FRAPTRAN which can calculate stress and strain during this rapid transient. It can calculate strain energy density just like EPRI's code with strain energy density just the integral of the stress/strain curve. And we could use EPRI's critical strain energy density curve. We could set up our own limit on strain as a model for failure and do a calculation very similar to what EPRI is doing, in order to calculate the failure enthalpy. We're going to try and do that.

There's another thing that we can do, and that is we can look at these data points that you saw on the last slide, and we can make some corrections to them, because certain data points on that slide, like the whole group of Japanese points from NSRR were taken at temperatures that were too low.

We can estimate the temperature increment at the important time of failure, go to the mechanical properties and make some mapping into enthalpy and so adjust the points on that previous slide so that the data points directly give a more clear demarcation between failure and non-failure.

And the third thing we can do is simply to try and build a multi-parameter correlation and incorporate all of these variables into the correlation and fit it to all of the experiment data.

MEMBER KRESS: Have you got a good explanation for that one bad pretest?

MR. MEYER: Ah, yes, sorry. Here is the bad point. This is REP Na-1, which Dana mentioned. This is the very first test run in the Cabri test reactor on high-burn-up fuel, and it failed at an extremely low energy.

It has received a very large amount of attention in the last couple of years, because as we proceeded with the Cabri program in the technical advisory group, we decided that we wanted to have a better explanation of what happened to REP Na-1 before we continued to run the program.

The bottom line is that we're pretty much convinced this is an outlier and should be disregarded when you consider the whole body of data. Discussion on this started in earnest about two years with a paper by our contractor at Argonne, Hu Chung, at the Park City meeting, and this has been followed up with a task force effort within the Cabri Technical Advisory Group, an effort that is now being finalized by Herman Rosenbaum whom some of you may know. He's an old-timer from GE who is now working as an EPRI consultant to try and bring the opposing views of this together and document the understanding of this test. And that work will be finished about the end of this year, and very early next year we expect to have a full documentation on the RepNa-1 test that concludes that it's an outlier.

MEMBER KRESS: I mean it's obviously an outlier, but what was the cause of it being an outlier?

MR. MEYER: There actually may have been several causes, but among the leading candidates was there was a defect on the rod before it was tested that may have -- and it wasn't a normal kind of defect.

MEMBER KRESS: Now, when you test the rod and you fail it and distort it and do all sorts of things to it, how do you look at it and know there was a defect before you did that?

MR. MEYER: Well, they had a picture of it before they tested it.

MEMBER KRESS: Oh, they have pictures before they test it. Okay.

MR. MEYER: Unfortunately, they don't have quite enough information before they tested to make it easy to figure out if that was -- So initially, that defect was known about and it was ruled out as being the cause. After a great deal of investigation, it's back on as a candidate. There are other possibilities. One of the main concerns about this rod was that it did go through a pre-conditioning period where they -- in the sodium loop they have to wet the instruments and so they have to put it in and run it up to a relatively high temperature. We ran it up to almost 400 degrees and held it there for 14 hours. This is just about high enough for hydrides to redistribute, and there is some evidence that hydride redistribution took place.

MEMBER KRESS: That would make it more brittle.

MR. MEYER: It embrittled the specimen during the specimen preparation. There are rather large uncertainties in the instrument readings and analysis, so this point really may not be at 30 calories per gram; it may be at 50 calories per gram. So it's -- you know, not unlike rudder problems on 737s, it's rather complex to get to the root cause of this thing, but it's pretty sure that this is not a good test point.

MEMBER POWERS: Might just inject here Tom that in the Subcommittee meeting that you didn't intend --

(Laughter.)

MEMBER KRESS: That's why I'm asking.

MEMBER POWERS: We went blow by blow

through all of this. As Ralph indicated, they've had a task force working this poor data point half to death and what not, and I'm terribly disappointed because this was a tremendous rhetorical advantage for beating people over the head on high-burn-up fuels. I'm going to miss this data point.

MR. MEYER: Well, don't forget these two, the HBO1, the very first test. This test was run in November of '93 and by February of '94 the Japanese had run a test called HBO1, their first high burn-up test, and it failed at around 79 or 80 calories per gram.

VICE-CHAIRMAN BONACA: Ralph, one of the things that was presented yesterday was also there was significant spoiling on that sample.

MR. MEYER: Yes.

VICE-CHAIRMAN BONACA: How does it correlate with this oxide thickness?

MR. MEYER: Well, I'm not sure that spalling immediately changes the ductility picture. There certainly is a mechanism for the localization of the hydrides and for further deterioration of the ductility. But these two points have spallation also, and they seem to fit nicely into the trend.

VICE-CHAIRMAN BONACA: Thank you.

MEMBER POWERS: Ralph, it's also true that in high burn-up fuel generally we see incidences of spallation.

MR. MEYER: I wouldn't touch that one. We have seen it in the test program, rods with spalled oxide that have come out of commercial reactors. How prevalent that is I have no idea.

MEMBER FORD: Ralph, you've put forth three physical reasons for why one is where it is.

MR. MEYER: Yes.

MEMBER FORD: Oxide spallation, hydride redistribution and mechanical defect. These are all physical phenomena that can occur on real rods. What's the likelihood that you could have that conjunction of physical aspects occurring in any one rod assembly?

MR. MEYER: Well, I'm really not prepared to discuss that defect, because I haven't spent much time looking at that. But what we have looked at more on our side is this preconditioning matter and the redistribution of the hydrides, and that can't take place during normal operation, because you don't have cladding temperatures up at 400 degrees.

MEMBER FORD: I guess my concern is everyone's trying to get rid of this one ugly fact and put it under the carpet, and until we're absolutely sure that it can't occur to any reasonable extent in the operating reactor, you can't ship it under the carpet.

MEMBER KRESS: You don't have to be absolutely sure, you just have to have a low enough probability.

MEMBER FORD: Yes, that's true.

MR. MEYER: Well, we have a lot of other tests in this database with real high burn-up fuel that has real blemishes on it.

MEMBER KRESS: And that's by probability. You could probably relegate that to low probability.

MR. MEYER: This one just -- okay. So we do have an empirical correlation. I'm not going to go into it in any detail. I'm going to say right off that I don't think this is in good shape. we've got to work on it to improve this correlation. I just want to indicate that the first correlation that has popped up on this subject was developed by Carlo Vitanza who's a well-known guy in our field, and he correlates the failure enthalpy with some measure of cladding ductility, pulse width, oxide thickness and the cladding wall thickness.

MEMBER KRESS: When you say correlation,

that's all empirical?

MR. MEYER: It's all empirical, it's just empirical. But we're going to work with Carlo to try and improve this correlation and use it as one of our several ways of trying to get a failure line.

MEMBER WALLIS: Does he estimate uncertainties with this correlation too?

MR. MEYER: I hope so.

MEMBER WALLIS: You can correlate anything, but you may have tremendous uncertainty.

MR. MEYER: Yes. We'll have to do that. Let me say now that I know that you're going to worry about where this line is and how uncertain this line is, and we're going to worry about those things too, but in the end it's probably not going to be real critical, because I think we're going to have a very big margin between where this line is and where a PWR is able to get a fuel rod up to. So maybe I'll say a little more about that. Pulse width.

MEMBER WALLIS: Some of the boron dilution events that we've thought about can get you up hundreds.

MR. MEYER: Well, we've looked at boron dilution events also. They in fact show up here on the pulse width slide.

MEMBER WALLIS: They're a moving target. The number's been changing in the last month or so.

MR. MEYER: Okay.

MEMBER WALLIS: Some of them we've heard about are up above 100.

MR. MEYER: Hundred calories per gram.

MEMBER WALLIS: Do the worst thing with the slug --

MR. MEYER: Yes. But probably not in the first 100 milliseconds for that first pulse, because that one's a little smaller than the rod ejection accident. And it's going to turn out that if you can't get enough energy in it quickly, you're not going to fail by mechanical means, you're going to end up just heating it up and having the damage done at high temperature at a little higher energy level. And in fact the broad picture that's emerging here is that boron dilution and BWR oscillations look like they fit that pattern. When we examine the power spikes they look small compared to the rod ejection, although they are repeated and there's power left in the core, so you have a mechanism subsequently for getting the cladding temperature to go on up higher; whereas, in the rod ejection accident you don't. It's all over with in a short amount of time. And it turns out the short amount of time matters and has been a big subject of controversy in the last year and a half.

MR. ROSENTHAL: My name is Jack Rosenthal. I'm the Branch Chief of the Safety Margins --

MR. MEYER: My boss.

MR. ROSENTHAL: -- and Systems Analysis Branch. Just to give some time perspectives, like a thermal hydraulic time constant for fuel rod like this is maybe like eight seconds or so. For the rod ejection, we are worried about ten versus 30 millisecond type pulses. For the boron dilution events, which we discussed with the Subcommittee on Thermal Hydraulics, we're talking about events that proceeded over tens of seconds and the ATWS we're thinking of even longer time scales. It is good, though -- so that the underlying thought should be different for these events.

On the other hand, it is good to put boron dilution on this graph, because what we're trying to say is that, you know, a lot of the design basis accidents -- you've chosen an accident and you need it to be a surrogate for a class of accidents so that at the very time that we're thinking in terms of ejected rod, the Chapter 15 analysis, we are mindful that you get more revolution. There are other ways of injecting the -- changing the --

MEMBER WALLIS: Jack, then they would be absent -- off the scale here in the boron dilution event, which takes a long time. The crosswidth is long so it's probably off the scale here; is that right?

MR. MEYER: Actually, the initial pulses are fairly narrow; they're right here. We've looked at a couple of them.

MEMBER WALLIS: But the ones where we get large amounts of delta H --

MR. MEYER: The first one is -- I believe the first one is the biggest one, and that's the narrowest one. And in fact it's the point from this slide -- and if I can try and go on and get to other things, the point of this slide is to show that there is a relation between the energy that you deposit and the width of the pulse. And it's a law of physics.

MR. RAO: But one other point I think needs to be made and that is risk, which is what we're interested in, is the product of frequency and constant. And the frequency of boron dilution and ATWS core oscillations is likely to be quite a bit higher than the frequency of rod ejection. So it's not okay to take boron dilution and rod ejection off the table if we're interested in risk.

MR. MEYER: Well, I'm not trying to. I'm just trying to get through my story on the rod ejection.

MEMBER WALLIS: This is a very funny slide. It has no width at all. It doesn't really happen, if it happens in zero time. You have an infinite amount of energy.

MR. MEYER: This is 20 to 40 milliseconds width on this pulse.

MEMBER WALLIS: But I'm saying the curve is really strange. You say it's the law of physics. The shorter the width, the higher the integrated energy deposition. You're saying it's the law of physics?

MR. MEYER: Yes.

MEMBER WALLIS: So if there's no width at all, it's infinite energy.

MR. MEYER: Well, I don't know about the --

MR. RAO: Clearly, if you have a slow enough pulse, you allow the rod to heat up, you allow the U238 doppler resonance integrals to be affected because you've heated the rods so that the normal fuel feedbacks turn the event off, if you can put the --

pump enough energy in before you shoot the temperature of the pellet up, then you can go to higher energies before it turns itself off. And I think that that's the basis for which you get the analytic correlation that shows that in order to deposit a lot of energy into the system you have to do it fast.

MEMBER KRESS: What is your empirical correlation curve, the one on the previous slide?

MR. MEYER: Up here?

MEMBER KRESS: Yes.

MR. MEYER: Ten milliseconds. And this is the point that I'd like to make from this slide. For the moment, don't look at the BWR points or the boron dilution points. Everything else is for PWR rod ejections. I actually showed a different slide yesterday. Yesterday I had a slide that looked just like this, but it was a bunch of sensitivity data. It's exactly the same. This relation between pulse width and the increase in fuel pellet enthalpy is real. It agrees with an analytic expression, and the Nordheim-Fuchs equation shows the same thing, it's been calculated by many different laboratories in different codes, and this is a relationship that really hasn't been challenged by anybody, yet the controversy comes when you look at what do you expect a power reactor for a pulse size and pulse width, and what should you set up in your test reactor for a pulse size and a pulse width? Power reactors, we're told, have maximum energies on the order of 20 or 30 calories per gram, maybe 40, so pulse widths in the range of 30 or 40 milliseconds or bigger.

For PWRs, you're down here. Very low energy expected if you eject the rod and do a best estimate calculation for the power reactor. What we see from the cladding failure data is that the failure is somewhere out in the range of 80 to 100 calories per gram. Still a rough number, but you can see that wherever it is along here the pulse width that a PWR would produce if you badly designed the core in order to get that much reactivity in would be narrow. And so this has a bearing then on the test conditions when you're out here exploring the failure limits as opposed to running a test back here where you just want to confirm that you don't get any damage for an expected pulse.

And I'll tell you right now that you can see the margin that I think we're dealing with. When we do the plant calculations, we're going to be down here in fuel enthalpies on the order of 20, 30 or 40 calories per gram. And when we get our failure enthalpy -- cladding failure enthalpy curve refined, it's going to be somewhere up here around 80 or 90 calories per gram. So you have a large margin, and if we have some uncertainties on both ends, I think we can accommodate that.

Now, what difference does the pulse width make? This slide's kind of a mixture of two things, but look at the top two curves. These are calculations that we did recently. The black curve up on top is a 30-millisecond pulse, this pink is a 10-millisecond pulse. Both pulses were arranged to have 100 calories per gram. So the mental picture here is that we have a pulse on the order of -- total energy of 100 calories per gram and a fuel rod whose cladding is going to fail at 80 or 90 calories per gram.

So we look at some enthalpy like 80 calories per gram and we see that along this whole range that the 30-millisecond pulse has gotten the cladding up to a significantly higher temperature by the time the failure takes place. Now, this means that the cladding in the 30-millisecond test is going to have more thermal expansions, it's going to try and run away from the pellet which is pushing on it a little more successfully than the 10-millisecond pulse, and the mechanical properties will be different.

MEMBER KRESS: Excuse me, is this cladding picture a result of the fuel thermal time constant or does it have anything with the heat transfer?

MR. MEYER: Heat transfer, it's the heat transfer.

MEMBER KRESS: Does the heat transfer in the coolant matter or is it just the thermal time constant of the fuel?

MR. MEYER: Well, Harold, does the coolant heat transfer -- this is Harold Scott.

MR. SCOTT: That's not really the -- the main thing is that the pellet has heated up because it's energy so it's looking for someplace to put that heat, and the heat is going to go out through the cladding, so the cladding temperature, yes, it's the -- it's just the heat flow through the cladding is going to heat it up.

MEMBER KRESS: So it's a thermal timing, yes It's something like seven or eight seconds, or something?

MR. ROSENTHAL: For the fuel rod as a system, including the coolant, we're working on it in such small time scales that you could heat up the clad, specifically the clad, but you're not going to get very much heat transfer to the coolant.

MEMBER WALLIS: Is it possible for the rod without considering the coolant is a lot shorter than the fuel rate cycle?

MR. MEYER: The coolant is in there, it's in there. This is a calculation with the --

MEMBER KRESS: So the coolant does matter.

MR. MEYER: Yes. I just don't have an answer to your question, which is more important, the coolant heat transfer or the time constant.

MEMBER KRESS: But I was wondering, what sort of coolant conditions the tests were done in because then you worry about the coolant temperature as well as the cladding temperature.

MR. MEYER: Yes. Exactly, exactly. Well, this was done for PWR conditions, this calculation.

MEMBER KRESS: It's just a calculation.

MR. MEYER: Yes. And the tests are done --

MEMBER KRESS: You said some of them were done in sodium.

MR. MEYER: Yes. The 30-millisecond Cabri tests were done in sodium, the 10-millisecond NSRR test were started at a very low temperature, and these were done in stagnant water.

But, anyway, I think we can make at least some first order correction for temperature variation, and this is the source of the temperature variation coming from pulse width and testing.

Just a couple of words about Cabri. This is on an extra slide that's with your handout somewhere, hopefully. The Cabri International Program has 12 tests in the test matrix spread out over a bunch of series called CIP, Cabri International Program. And the two initial tests are in fact the old sodium loop. And I'll show you on the next slide a little more about those two tests. These are coming up later this month and next month. And then the reactor is shut down for some refurbishing and installation of the water hose which is under construction. And that will be brought back up in 2005, with the real test beginning again in 2006 and running for about three more years.

MEMBER KRESS: Is this BR-3 fuel and testing? Is this BR-3 fuel and testing?

MR. MEYER: Oh, no. These test rods are all from commercial reactors.

MEMBER KRESS: At 80 gigawatt days per ton? What's that in the middle of --

MR. MEYER: Well, by that time, yes.

These will come from lead test assemblies.

MEMBER KRESS: Okay.

MR. MEYER: These are all coming from lead test assemblies. The two rods that are being tested now there's one with M5 cladding that's coming from the Grovelins Plant in France, and there's one with Zirlo cladding, which is a Spanish fuel fabricated by and used --

MEMBER WALLIS: Why is there so few tests? I mean there's so much scattered on your graph I'd think you'd need a lot of tests.

(Laughter.)

MR. MEYER: The --

MEMBER WALLIS: Otherwise you've got another outlier and then you have to argue about that.

MR. MEYER: Well, first worried about another outlier, I hope we don't get one. This test program breaks the bank. It costs for these 12 tests, Rosa said 62, I think it's $72 million for 12 tests with the funding split three ways. IRS and the French research --

MEMBER WALLIS: What's the incremental cost of another test?

MR. MEYER: It's about --

MEMBER POWERS: It's about $6 million, $3 million, something like that.

MEMBER WALLIS: Well, you've got the facility.

MR. MEYER: You have the facility but you're handling --

MEMBER POWERS: Ever done a test? It's unbelievable expensive.

MR. MEYER: They can run about three of these a year, and it cost, on average, about $4 million a test. The Japanese and the NSRR reactor run many more tests than that.

MEMBER WALLIS: What's the cost of the risk of now knowing the right answer?

MEMBER POWERS: I think it's my impression that the argument that Ralph's putting together here is that we're coming to a belt-and-suspenders kind of approach here that on the one hand the mechanics of the reactor itself, the way it's loaded and the way the rod can get ejected are such that if you look at his plot of power input versus the time of the input, it's extraordinarily difficult to get to such a short pulse that you put enough power in to fail. And then phenemonologically is coming at it saying that the damage to the cladding occurs at power input levels that are, to be sure, a function of the level of burn-up as reflected in the loss of ductility from the rod. But in fact there's still some residual strength up to the 62 gigawatt days per ton that that protects you for foreseeable power impulses. Is that roughly correct, Ralph?

MR. MEYER: Yes. I think that we'd be able to provide an overall satisfactory answer at 62 gigawatt days per ton for Zircaloy and a short period of time, in spite of the uncertainties and the lack of repeated tests, to accurately demonstrate what the error is and things like that. It's imperfect but I mean it's not going to -- in the end, it's not going to be bad, because I believe there is ample margin and there is also low enough risk with this event that I believe the result will be satisfying.

MEMBER KRESS: Ralph, since these are confirmatory tests, then I presume you don't have a user need list?

MR. MEYER: We don't have -- yes, Jack, you want to handle this? I don't want to answer this question.

MR. ROSENTHAL: Yes. Let me just say that RES and NRR are revisiting the user need process. The program plan that's being described to you was an Agency plan. We're working out -- in the process right now of working out a new Agency plan and that use needs, if necessary, will stem from that revised plan. And so that, pragmatically, in terms of doing the fuel work, generating the experiments, getting the results, coming to conclusions, it's really not a problem. And I would recommend that the way NRR and RES work is a broader issue and it might be best to hear about that in some other context.

MEMBER KRESS: I guess I'll ask my question another way: Is the funding for all of this coming out of the research budget and not from other sources?

MR. ROSENTHAL: It's coming out of the NRC's budget for sure. It's money that's allocated to research to do the work through the PPBM process, which is an Agency-wide process. We've got the money to do the work at Argonne. We've got some money to participate in Cabri. So it's worked.

MR. MEYER: This one is pretty highly leveraged in terms of cost. But when I explained it, it was a high cost. The funding is split roughly three ways: RRSN, the Research Institute in France is carrying a third of the cost; EDF, Energy De France is carrying a third of the cost; and the international community is carrying the other third of the cost.

And in that international community we're about one out of six.

MR. ROSENTHAL: Yes. And in fairness to EPRI let me say that EPRI and we pay I think the same amount or just about the same amount.

MEMBER WALLIS: The result will be another black or white point on your broad-brushed item?

MR. MEYER: Yes, that's true, but perhaps more importantly is these will be the first tests with new cladding alloys. And since cladding ductility was central to the survival or failure of the cladding we have now in these tests for the first time some demonstration for M5 and Zirloy cladding.

MEMBER KRESS: Do you have good database on the M5 and Zirloy ductility versus temperature and oxidation level and --

MR. MEYER: Well, we don't at this time, but in fact the rod and sibling rods within the CAPRI Program are having mechanical properties measured. So in addition to getting this failure or non-failure point out of it, we get mechanical properties, we get strain measurements from the tests, quite a lot of data that does come out of the test program in addition to just the black or white dot on the curve.

I would like to move on because actually the situation with the loss of coolant accidents are probably more important. I'm not sure it's more interesting yet, but it's getting more interesting day by day, because we actually have done some integral tests with high burn-up rods. We've done two of them, and I want to move on fast enough to be able to show you that. I mentioned before that the issue has to do with whether the embrittlement criteria need any kind of adjustment when you move to high burn-up cladding and also the evaluation models.

Let me talk from this slide for a moment. But I'd like you to keep in mind that there are two parameters in 5046 that we call the embrittlement criteria. It's the 17 percent oxidation limit and the 2200 Fahrenheit decladding temperature limit. In addition to that, there are four models or correlations, either in Appendix K or set up in the regulation, that are related to fuel that might be affected by high burn-up that you have to have in order to do a LOCA ECCS safety analysis. One of them is oxidation kinetics rate, one of them is a correlation for when a burst occurs, because these rods are pressurized and when you depressurize a system as they heat up they're going to balloon and burst. You need a model to tell you how much strain, how much deformation is in the burst area of the rod, and then you need another model to tell you how that parlays into flow area reduction or flow blockage for the long-term cooling calculations.

So we are doing several series of tests to get at all of those issues. I actually should have another diagram over here that looks just like this one with a single piece of tubing in it which we have used for oxidation kinetics measurements. So we have completed measurements of oxidation rate for high burn-up BWR cladding that has low corrosion on it, and we haven't done the high corroded PWR rods yet.

These two test streams are addressing the embrittlement criteria and the other models in the process. The embrittlement criteria came from ring compression tests done by Hobson in the late '60s and early '70s, and we are trying to stay close to the original intent of the regulation, the hearing in '72 and '73, and so we're going to try and more or less replicate the ring compression tests that were done before in order to see if we come out with some numbers different than 17 percent and 2200.

In the process, however, we have discovered -- well, 20 years ago it was discovered that there was a phenomenon unknown at the time that the regulation was established having to do with enhanced hydrogen absorption in the neck of balloon region of the fuel rod. And so we have to do some tests to confirm that we have encompassed the effects of enhanced hydrogen absorption in whatever criteria we get out of our ring compression tests.

So we are doing a series of what we call integral tests where we have a segment of a fuel rod about 15 inches long which has the fuel left in it. This is a high burn-up fuel rod with fuel intact. We pressurize it, we run it through a temperature transient that's similar to that in a loss of coolant accident, and the rod heats up, it deforms, it bursts, it continues to heat, it oxidizes, it's cooled a while, it's quenched, it's brought back down, and then after carefully measuring the temperature and pressure at rupture, which is one of those correlations, and measuring the strain on the burst, which is another one of those correlations, then we take that specimen down and do a four point bend to try and gauge the impact of this enhanced hydrogen absorption and fold that back in with the results from the ring compression tests and try and wrap it all up.

It's a fairly complicated scheme. I think we know what we're doing, and we have some test results from the beginning of this. It will take us a couple more years to finish this for Zircaloy, but this is a curve -- a slide that shows some of the oxidation measurements. The LOI, this is an in-cell test. This is high burn-up Zircaloy-2. The others are unirradiated companion material, or Zircaloy-4, and this is the Cathcart-Pavel correlation, and these are data points from the Cathcart and Pavel's work. So we plotted them all together here, and they fit very nicely with the Cathcart-Pavel correlation.

In fact, we've reviewed a large number of other oxidation studies, and the result that seems to be emerging is that all zirconian-based alloys, whether burn-up or not burn-up, seem to fit the Cathcart-Pavel correlation in the vicinity of 1200 degrees centigrade or 2200 Fahrenheit. It's a very convenient handy result, but I have to caution you that we have not yet made the measurements on the Robinson rods that we have, the PWR rods that we have at the lab, which have a heavy layer of corrosion on them, so this could change.

So sort of the simplistic picture of this is that the oxidation rate, the controlling process is the movement of oxygen through the building up oxide layer, and it's ZRO2 on all of them, and this is a process that's more or less taking place from the surface outwards and it's the same. I can't guarantee that that is going to be the scientific picture that holds up forever and forever, but it looks like it about now. You have to keep in mind that you should not expect the same result for the mechanical behavior of the cladding itself, because the ductility or embrittlement of the cladding is not going to depend on what's sitting up on the surface but what's down in the material, and that is going to be affected by the alloy composition and other factors.

MEMBER KRESS: You certainly could use this as measure of the remaining thickness.

MR. MEYER: I'm sorry?

MEMBER KRESS: You certainly could use this as a measure of the remaining thickness of clad you have in terms of the strength of the clad.

MR. MEYER: Well, certainly there is this prior beta layer, and the thickness of the prior beta layer was one measure of the ductility of the material. There are other measures, and we will look at always of characterizing this, but we'll actually do the tests so that we have the data.

MEMBER WALLIS: This assumes the oxide layer stays on. If we heat it up and cool it down cyclically, would this lead to flaking off of the oxide layer?

MR. MEYER: Well, the bulk of the oxide seems to be full of microcracks, so it really only looks like it's depending on the adherent layer that's right down close to the material, which is why all of these seem to be fitting this -- why the high burn-up rods seem to fit the same correlation as if it didn't have that corrosion on it. But the real test is going to be when we get the heavily corroded Robinson rods and get the data on that. Anyway, so far so good. The measurements are really precise and the agreement with the correlation is quite good.

Now, for the integral test, this is where we take a 15-inch long piece of fuel rod, heat it up, coolant it and rupture it. This is the temperature sequence that we have chosen for the test. Initially, we're doing three tests, A, B and C. The first test comes up to the point of rupture. We make sure that it's ruptured, and then when it's ruptured we turn the furnace off. So that's the first test. That was done in September, September the 15th -- no, August the 15th. Second test --

MEMBER KRESS: How long were these specimens?

MR. MEYER: About 15 inches.

MEMBER KRESS: And you repressurize them --

MR. MEYER: Yes.

MEMBER KRESS: -- and the seal the --

MR. MEYER: Yes.

MEMBER KRESS: Okay.

MR. MEYER: The pressure part of it is important, of course, because you want ballooning deformation. The magnitude of the pressure, how much gas you put is important, how big the plenum size is.

MEMBER KRESS: And 15 inches is long enough to get rid of end defects?

MR. MEYER: I think so. You'll see pictures in just a minute. The second test -- the first test was done in argon, it wasn't even done in steam. The second test was done in steam. It was taken through the whole transient down to the point of quenching and we didn't quench it. We just let it continue to cool, just made sure that we didn't bang up the specimen a lot, and also we didn't have the point system installed yet. So a little bit of reality --

MEMBER POWERS: The truth comes out.

MR. MEYER: And the third test, which has not been done, will have the quenching, and it's going to be done next month.

One of the things that we expected was that axial gas flow would be substantially restricted, because you're at high burn-up, you essentially have -- cladding has crept down, you've lost all of the gap, you don't have much open space, actually. And tests in the Halden reactor under operating conditions show very poor gas communication in their fission gas, sweep gas grid. And so we fully expected that the plenum in our test apparatus, which consists of a gauge and some lines up to a valve going to the gas bottle, that that would depressurize slowly and it did not happen. It depressurized very rapidly. And so what you have here in red is the pressure trace for the out-of-cell test and in blue or black the trace for the in-cell test. And you can see that the pressure drop down to a rather low pressure was very rapid. So there's no discernible flow restriction here in the pressures that are affecting the ballooning deformation. It's only when you get down to very low pressure differential that you start seeing some effect of this flow area restriction.

This is a picture of the high burn-up rod that was burst and a picture of the same type of cladding in the unirradiated state that was burst basically in the same apparatus, under the same conditions. In effect, the size, shape and opening are the same. I mean we've already looked at the prophelometry traces and compared them quantitatively, and there's not that much difference.

Little interesting stuff you can see in the opening of the real fuel rod. Here is the second test. This is the one now that was exposed to steam, and this is just two views of the same rod now, the second one, and you can see that the deformation, the opening, is about the same as in the other test that was run in argon and in the out-of-cell tests that were run in steam.

MEMBER KRESS: Is your temperature gradient along this rod or --

MR. MEYER: I'm sorry?

MEMBER KRESS: Was your temperature gradient along this rod or was it the same temperature?

MR. MEYER: It was uniform. There was no intentional temperature gradient. Obviously, you have ends where the temperature falls off, but it's quite uniform.

MEMBER KRESS: And they all failed in the middle like that?

MR. MEYER: Yes. Not exactly in the middle, the position varied a little bit, but, you know, an inch or two.

MEMBER KRESS: Makes me worry a little about an end effect for a short rod when they all fail in the middle.

MR. MEYER: I mean we'll have temperature -- there are four thermal couples on this thing and --

MEMBER KRESS: I was worried more about the physical restraint end of things.

MR. MEYER: Oh, the physical restraint.

MEMBER KRESS: Yes.

MR. MEYER: Well, I haven't told you anything about the test apparatus, but the physical restraint was an important consideration. We spent a lot of time on it. In the end, we have what we call a hanging test train where we put no axial constraints on it other than the weight of the specimen. We have rings on it that constrain its lateral movement, so it is not allowed to move sideways a lot, which would be the case in a fuel assembly where it has grids. And so this is kind of simulating a grid span.

MEMBER KRESS: Yes, it could; you're right.

MEMBER WALLIS: This should have been done 30 years ago.

MR. MEYER: We just make it look simple today.

MEMBER KRESS: Those were done but not at the high burn-up.

MEMBER WALLIS: Not at the high burn-up, yes. So the only difference is the high burn-up?

MR. MEYER: Yes, the high burn-up. The rod burst tests were done 30 years ago.

MEMBER KRESS: They were almost with fresh clad.

MR. MEYER: Yes.

MEMBER POWERS: Ralph, in order to give time for Undine --

MR. MEYER: Yes. You want me to quit?

MEMBER POWERS: Quit being so accommodating to the questions.

MR. MEYER: Okay. I do want to point out that fuel came out of both of these high burn-up rods. We did not expect this. It appeared that some of it came out during the transient because there's a black deposit on the quartz tube that surrounds this in the furnace. We're analyzing all this to see what this deposit is. About a half a pellet's worth of fuel came out, a little bit of that during the transient and some of it after as we were handling it.

MR. ROSENTHAL: Now that we've been asked to go fast, I want to slow it down for just a minute.

MEMBER POWERS: You were asked not to accommodate -- be so accommodating to the questions.

MR. ROSENTHAL: So let's go back to the way -- to put it in a PRA context, let's go back to the way we did analysis in past years. In a boiler ATWS we assumed that you'd lower the cold water level, you trip the recirc pumps and you'd end up at a power between ten and 30 percent of power depending on whose analysis you had one -- whose analysis told you you were using. And then you had the great race between the power that you were putting in the suppression pool and power that you can extract from the suppression pool and what operator reactions would occur, and you drew your event tree accordingly. And you did not ask questions about what's going on inside the core. If the operators could successfully terminate the event before you overheated the suppression pool, then you wrote "okay" on the far right of your event tree. And if not, you wrote "core melt," and that's about what you did.

Now, it's appropriate to go back and think about what's going on inside the core. I think that unlike when we did Limerick PRA or something like that we now recognize post-Liebschtadt and what not. But if you trip the RCP -- or the recirc pumps, you'll go into a region of instability mechanistically so that you'd be at some reduced power but with some oscillations going on. And so it's fair to -- I don't know if I have a safety issue there, but I know that there's enough going on that it's worthwhile to think and analyze and experiment their way through this. So that's the context to saying why we're doing this work, and one can imagine that if in fact one finds that one has fuel damage prior to the suppression pool failing, then you have to rethink what you're doing in PRA space. I'm not there yet, it's some work that we're doing.

MEMBER KRESS: This is the kind of question that comes up in things like power uprates where they do a risk analysis. This kind of stuff doesn't show up in the risk analysis, but it's a potential effect of a power uprate.

MR. MEYER: Yes. Okay. So Jack has presented my first slide, and all I want to say is that we're not making rapid progress on this, but we are making some progress. And I just wanted to indicate to you that we have done something in the past year to try and move this forward. One of the things we -- I say "we" loosely, because this was done by Jerry -- Jerry, as a result of our discussions with him, actually, has run two tests with repeated pulses to look at the PCI component of these power oscillations.

This is brand new work. It will be presented at the Nuclear Safety Research Conference in two weeks by Jerry, and I'm not going to describe this in detail, other than to say they've done two tests. This one had seven pulses in it. In neither test did they see any evidence of what I will call mechanical ratcheting, where the mechanical expansion in the first pulse was somehow amplified into the second pulse and lead to a mechanical failure. They didn't get any mechanical failures.

This is consistent with the conclusion from the experts who decided from looking at this event that the mechanical action of the pellets on the cladding was not going to be the big feature in the power oscillations. These tests were simply run to confirm that that is not the right path to go down looking at the mechanical behavior, as we are doing with the rod ejection accident, but rather looking at this more like a LOCA transient where you have a high temperature excursion.

In that regard, the other thing we have done is to work with Stook and VTT in Finland who have a small thermalhydraulic code called GENFLO, which they have coupled to our FRAPTRAN code. This allows one to actually do some cladding temperature analysis during these oscillations. And we installed this code a few weeks ago up at Battelle, and during the next year we'll be using it along with Stook to try and --

CHAIRMAN APOSTOLAKIS: Is there another presentation after you? Because the whole session is supposed to end at ten o'clock.

MEMBER POWERS: Don't ask so many questions. Okay?

CHAIRMAN APOSTOLAKIS: I didn't ask any.

MEMBER WALLIS: Don't tell us so much stuff that will cause questions.

CHAIRMAN APOSTOLAKIS: Can we wrap it up in 12 minutes or so, do you think?

MR. MEYER: Yes. I can finish up in 90 seconds.

CHAIRMAN APOSTOLAKIS: Okay.

MR. MEYER: Okay. Source term, high burn-up source term. We had a panel of experts and more or less they concluded that NUREG-1465 is okay for the high burn-ups we're talking about, but we'd like to tie up some loose ends and get a little more data. So much for the source term slide.

Transportation and dry storage. The issue here is cladding damage after sitting in a storage cask where in fact it's under -- sometimes it's under higher temperatures and higher pressures than it is during operation. So we ran creep tests. These are long-term tests, they take six, nine months. We ran a full series of creep tests on Surrey rods that have been sitting for 15 years in Idaho, and in July we inserted HB Robinson rods, the highly corroded PWR rods and we're accumulating data on those right now. Thank you very much.

MEMBER POWERS: That covers the research program in a terse fashion that RES has underway for confirming its positions. As I indicated to you at the beginning of the session, the Electric Power Research Institute has submitted a topical that has an extensive analysis of the reactivity insertion accidents. They have asked NRR to review this, and Undine's going to describe their plans.

MS. SHOOP: Okay. Good morning. Thank you, and I'll try and make this very quick. As my slide indicates, what we hope to accomplish by this presentation is similar to the large LOCA codes that we periodically do -- similar to the large LOCA code reviews that we do. We typically come in and present preliminary information to the Committee so that the Committee can come up to speed on the review that we're doing, and that way we can get some feedback and your thoughts on the review so that we can make sure that we accommodate all of your concerns during our review process. So that's basically what this is. This is the pre-meeting. Once we're done with our review, we'll come back to you and share with you everything we found.

Next. As Ralph Meyer has alluded to and Dana's already said, back in 1998, we created an Agency plan for high burn-up fuel. Part of this plan did ask the Office of Research to confirm criteria up to the 62 gigawatt days per metric ton uranium. That reiterated the 1993 that the Office of NRR had asked research to confirm that criteria.

In addition, what we've realized is that in the age of declining budgets we no longer have the resources to be able to do all the research ourselves, and therefore we said that if the industry wanted to go above 62 gigawatt days per metric ton uranium, they would have to provide the criteria, the database, the methodologies and the models to be able to demonstrate the ability to go to higher burn-ups. So that's basically what this is. EPRI developed a program to be able to come up with the criteria, the database, the models and the methodology, and this is their first topical to be one of a series in order to be able to justify the industry going to higher burn-ups.

Our preliminary review plan, we came up with a preliminary plan, we're still working on the final plan, but the focus of the -- the purpose of coming up with a plan is to focus our resources and make sure that we've addressed all the components so that we don't get to the end of the review and then find that there is an issue that surprises us. That's also why we're talking to you today and yesterday.

MEMBER KRESS: EPRI, when you tell EPRI or when you agree with EPRI that they have to provide the database for greater than 62, do you tell them what information you think you'll need like the coolability, fission product release, failure point, energetics of any FCI or do you just leave that up to them and hope they give you the information you think you'll need?

MS. SHOOP: As with any submittal that the industry puts to the Agency, other than putting out reg guides and SRPs, we do not give them additional guidance. Just like with the reg guides and the SRPs, they have the ability to use our information or not as they choose. They have to be able to justify --

MEMBER KRESS: So you assume they know what they -- so you assume they know what you'll need.

MS. SHOOP: They have to provide the data to be able to support what it is that they would like. That's our going-in position.

The elements of our plan include data verification. As you've just heard Ralph say, there are a number of different testing facilities. Each facility has their own unique capabilities or non-capabilities, and so we're going to have to verify that the data is used correctly, it's statistically combined correctly.

SED/CSED theory and model. The EPRI program --

MEMBER POWERS: We might just remind people that this is strain energy density and critical strain energy density.

MS. SHOOP: Yes. They say that they can make an equivalence between that to Rice's J/Jc. That was the revolutionary thing in the '60s that Rice put forward. So they're saying that they have an equivalent way to do this. That way they can use codes and methods to be able to come up with that. We're going to have to look into that, and we're going to code this theory into the FRAPTRAN computer code. That way we have a way to independently assess the industry's proposal.

Their proposal had a fuel failure limit and a fuel coolability limit similar to our current Reg Guide 1.77. EPRI's proposal has these same limits, so we're going to have to look at them. The FALCON code, they used the FALCON code in developing a methodology, and that is a code that we have not reviewed or looked at, so we're going to be looking at that. Fuel dispersal, we're going to have to review the data for applicability to make sure it's all within where they say it is. Uncertainty and conservatism. You know, we always have to make sure that we have the appropriate statistical uncertainty, make sure that they have appropriate conservatisms built in for the areas that we don't know about.

The limitations of the criteria, there, again, the criteria was developed with certain parameters. We have to make sure that it's applicable to other parameters or not as we determine. And of course we're always going to have our safety evaluation conditions. And as part of this whole effort, we're going to revise the appropriate reg guide, and there's actually three SRPs that all reference this limit, and we'll come back to you as part of that effort.

For our future activities, as I mentioned, this is our preliminary plan. The Office of Research and NRR are getting together to be able to develop the final plan, and we hope to have that by December. We'll be keeping you updated of our progress and everything else. Thank you.

MEMBER POWERS: You raise one issue in this plan, which is with the high burn-up fuel we encountered a change in physics that the computer codes didn't predict; we didn't know about it. Is there a hope that we now know the physics well enough that we can use these codes in an extrapolated fashion or do you think that there's going to have to be a fairly extensive database support for these analyses?

MS. SHOOP: As with anything that the industry proposes and we look at, that is part of our analysis procedure. At this time, we're still gathering data to be able to make an intelligent decision. So it would be premature of me to speak to that --

MEMBER POWERS: I know that, for instance, the Trans-Uranium Institute has taken some fuel up to like a 100 gigawatt days per ton. I know that as a point of information. What I don't know is what they found when they went up to those extremely high burn-ups. And I bring it up just to say, well, maybe there is some data out there that would tell us if there's some change in the physics.

MS. SHOOP: Could you please me those references, the ones you were going to provide me yesterday?

MEMBER POWERS: Undine, you make me work so hard.

(Laughter.)

MS. SHOOP: Well, you keep asking me these questions.

MEMBER POWERS: I'll see what I can do for you.

MS. SHOOP: Okay. I appreciate it.

MEMBER POWERS: Now, are there other questions that people have for this -- Undine's very short presentation? I note that she's coming back with us with a schedule for the plan in December, but this is a fairly deliberate undertaking. I don't expect you to come back in February with the results; is that correct?

MS. SHOOP: No.

MEMBER POWERS: February perhaps, but not 2003.

MEMBER KRESS: I did have a point of information. The strain energy density and critical strain energy density hypothesis, is that spelled out pretty well in the EPRI report so that I can just get it and read it?

MEMBER POWERS: Yes. I think it's an extraordinarily simple concept, actually. They write it out in detail in there. They're just taking the integral under the stress/strain curve.

MEMBER KRESS: Yes. I'm primarily interested in the critical strain energy density part of it and whether they factor in the things that would make it critical.

MEMBER POWERS: That's right. That's the empiricism.

MEMBER KRESS: That's an empiricism.

MEMBER POWERS: And they go to elaborate lengths to show you how they derive that empirical quantity.

MEMBER KRESS: Because I think that

relates to your question about physics up to --

MEMBER POWERS: Well, it also relates to what Undine had as the appropriate statistics and things like that, the way you analyze the data derived. It's an intriguing aspect of their topical report.

MEMBER KRESS: Do we have that topical report?

MEMBER POWERS: We do. I have a copy of it, and I think we made Xerox copies of it.

MEMBER KRESS: Okay. Thank you.

MEMBER POWERS: With no other questions, I will give it back to you, Mr. Chairman.

CHAIRMAN APOSTOLAKIS: Thank you, Dana. We'll recess until 10:25.

(Whereupon, the foregoing matter went off the record at 10:09 a.m. and went back on the record at 10:25 a.m.)

CHAIRMAN APOSTOLAKIS: Okay. We are back in session. The next item on the agenda is the overview of the European simplified boiling water reactor, the SWR-1000 and the advanced CANDU reactor, ACR-700, the pre-application reviews. Dr. Kress will Chair the session.

MEMBER KRESS: Thank you, Dr. Apostolakis.

This is the nature of a briefing to get us up to speed a little bit and at least acquainted to some extent with these concepts that are coming in for pre-application certification in the not too distant future. So pay attention and ask questions, and I guess we'll ask Jim Lyons if he wants to make any introductory comments before we get started.

CHAIRMAN APOSTOLAKIS: It's too low level for Jim.

MR. LYONS: I'm passing out this handout. My name is Jim Lyons. I'm the Director of the New Reactor Licensing Project Office. And what I wanted to talk about just briefly to kind of put this in context of where we are on several reviews that we've got coming, projects that we're actually working on the licensing actions. We have three early site permits that I think we all know about coming in June to September of next year. We would see that the Committee would be involved in that as part of the site safety analysis that will be done as part of the early site permit process. The other two portions are the environmental review and the emergency preparedness review, and the Committee hasn't in the past been involved in those.

In the way of design certifications,

AP1000 is in here for design certification, and they'll be -- actually, I think there's a meeting with them. They're coming to the full Committee next month, and we'll talk more about it. I've highlighted for both the AP1000 and the ESBWR when we actually have some more detailed milestones. The items there in red are the points in which the Committee has typically been involved in the review, both the draft safety evaluation report stage and at the final safety evaluation report stage where we actually ask for a letter. So I've tried to highlight those. Those are our due dates. Obviously, the Committee meeting and Subcommittee meetings would be held before that.

And so we have fairly detailed milestones. Obviously, on AP1000 we're well into that review. We've completed issuing our request for additional information on an AP1000 just last week on September 30. We got all our RAIs out, which was our first milestone.

On ESBWR, with General Electric, we've been working with them to develop a schedule for the pre-application and we actually have milestones. For the other designs, the ACR-700, the SWR-1000, which you're going to hear about today, GT-MHR and IRIS and PBMR we've just started talking to some of these organizations about what the pre-application is going to be and what else it will be. But I thought that this would be helpful to the Committee to kind of get an understanding of overall what we're working on. I don't -- I haven't included on here infrastructure type changes that we're working on; rules, regulations, that type of stuff that we may be coming to you and I'll get back to you tomorrow on those. But I wanted to at least lay out this kind of as an overall before you started on your discussion.

MEMBER KRESS: Who will be leading the PBMR aspects?

MR. LYONS: We don't know right now, so we're -- that's just -- we'd had some discussions with PBMR Limited from South Africa, and they talked about maybe coming in for a pre-application review in that 2005/2006 time frame. So we're just kind of waiting to see on that, and we figured we'd put it back on there, because it hasn't completely gone away, and I know that there was a lot of interest in that.

MEMBER KRESS: But the GT-MHR is coming in.

MR. LYONS: We've already started discussions with them, yes. We had meetings in the last couple weeks with both General Atomics and

Westinghouse on IRIS. And Westinghouse will tell us next month also that for IRIS they'd ask us not to defer resources from the AP1000 review, because they see AP1000 as their highest priority.

So with that, I'd turn it over. I just wanted to --

MEMBER LEITCH: One quick question.

MR. LYONS: Sure.

MEMBER LEITCH: The early site permit for Exelon is at Clinton, I believe.

MR. LYONS: Yes.

MEMBER LEITCH: And I think Clinton is for sale, is it not? Do you know if that impacts this schedule at all yet or do they still plan to proceed with the early site permit application?

MR. LYONS: They're proceeding on with that application, and they said that if they did sell the Plant, that there would be a decision about whether or not whoever bought the Plant would pick up that early site permit review. And so that would be as part of that. But they have not backed off.

MEMBER LEITCH: Okay.

MR. LYONS: We're still working with them.

MEMBER LEITCH: Thanks, Jim.

MR. LYONS: Okay?

MEMBER KRESS: Okay.

MR. FLACK: This is John Flack at the NRC Staff, if I could just add a little bit. We will be to the Committee on November 6 with our infrastructure assessment, which had originally included the four plants, PBMR, GT-MHR, AP1000 and IRIS. We have subsequently expanded it to also pick up the ACR-700 and ESBWR at this time, so you'll see some additional information that the Committee has not seen before in looking at an assessment, but we'll be back November 6 to talk to you about that.

MEMBER KRESS: Okay.

MR. RAO: Thank you for giving me this time. I'll just give you a very brief --

CHAIRMAN APOSTOLAKIS: Please identify yourself for the record.

MR. RAO: Sorry. Arturam Rao from General Electric Company. I'm the Project Manager for the ESBWR.

CHAIRMAN APOSTOLAKIS: Thank you.

MR. RAO: I'll be giving you a brief overview of the ESBWR, which is a 4,000 megawatt thermal natural circulation reactor with passive safety systems.

I'll be covering several aspects of the

design and the technology, a little bit about ESBWR evolution, the design philosophy of the safety systems, and what I'll be emphasizing as we go through the design is the basic design philosophy has been to improve the safety margins by putting in design features. It's not reliant on complicated analysis methods and extending and minimizing margins and stretching the limits. What we have ongoing with the NRC is a 12-month pre-application review which is intended to close the technology issues. What we are trying to do in this period is to get the approval o the TRACG code for use in LOCA containment analysis and transient analysis and close the issue on the adequacy of the testing and the qualification of the TRACG computer code.

The ESBWR is actually in some ways an evolutionary design. In a sense, it has evolved, and as you can see in the evolution of the design, the old BWRs used to have steam generators. And almost 30 years ago we gave up the idea of steam generators, decided it was a lot simpler to go with the internal steam separation, external loops, most of the operating plants are there.

The ABWR, an operating plant, would react to internal pumps, and the ESBWR goes a step further in terms of simplification by eliminating all the pumps and relying on natural circulation and passive safety features.

So the basic design approach has always been evolution towards simplicity. We've got a natural circulation reactor which looks like pretty much like any traditional boiling water reactor, just a taller vessel, six meters taller than the ABWR design. You get the feedwater coming in and flowing down by gravity, density difference, the water heats up in the core, you get steam and water and separation in the standard steam separator dryers, and steam goes out to dry the steam turbine.

What we did in this Plant was to enhance the natural circulation compared to standard boiling water reactors, basically by reducing the flow restrictions and a higher driving head. It took three ways we reduced the flow restrictions. We have an improved steam separator with lower resistance. We have a shorter core which reduces the two-phase pressure drop, and we have increased the downcomer area. It's interesting, when you look at the four-circulating plant, what you have in a four-circulating plant is a pump sitting right out here, and this pump actually introduces the resistance to the flow.

And so what the pump actually does it spends half its energy overcoming that resistance, okay, and the other half for providing additional flow. So what we did in this Plant was get rid of that resistance, and what you end up with is much increased natural circulation flow.

MEMBER KRESS: Is the efficiency of your separators related to its fictional resistance? I mean can you maintain the same separating efficiency?

MR. RAO: Yes. The whole philosophy is to make sure that the carry over, carry under are in the exact same range.

MEMBER KRESS: Are still the same.

MR. RAO: And we've done additional testing in the range of application for the ESBWR. So in addition to reducing the flow restrictions, you provide a higher driving head by using what's called a chimney, which basically increases the driving head between the downcomer and the core out here, enhanced natural circulation which makes the operation of the Plant a lot easier, reduces the vibration, flow rates, resistances and all in the vessel.

Evolution of the BWR containment is shown in this chart. Not enough time to go into all the details, but this is the ESBWR shown out here.

MEMBER KRESS: Are those kind of to scale in terms of volume?

MR. RAO: Almost to scale. This is not quite right. We've got to fix this one to get them to the right scale. This is the Mark III containment, which is right circular cylinder which surrounds the traditional suppression systems, the drywell and a wetwell. In the ESBWR, the size of the ESBWR containment, this building, is the same as the Mark III. So that's why it's not quite to scale, okay? But the basic features are shown out here correctly.

What they did is the spent fuel storage has been moved from the refueling floor down to the separate building like the Mark III, and the ESBWR moved to a separate building, inclined fuel transfer system similar to the Mark III except that it's now not part of the containment. Here it's part of the containment, so you can't do refueling operations or movement of fuel during normal operation in the Mark III. Whereas in this one, since it's not part of the containment, the containment boundary is out here. This inclined fuel transfer is outside the containment.

MEMBER KRESS: Do you have an inert gas in your containment?

MR. RAO: Yes. So unlike the Mark III, which is not inerted, it's similar to the ABWR or the earlier Mark Is and Mark IIs. It's an inerted containment, it's smaller containment. Expect no access during normal operation inside the containment.

The containment boundary is basically shown here. This is a drywell head. This is the raised suppression pool shown out here. These are what are called gravity-driven cooling system pools, which provide water makeup following loss of coolant accident. So this is the traditional containment boundary. And what you've got is all the safety systems, as you'll see later on, are inside the containment or just above it, these heat exchanges sitting above it.

MEMBER KRESS: Oh, okay.

MR. RAO: So basically have reduced the size of the safety grade buildings by almost half compared to the ABWR. Just look at the ABWR. The containment of the ABWR and the ESBWR containment look essentially the same, and they're about the same size also. What's different is the reactor building. The ABWR has six floors of safety grade equipment, pumps, heat exchanges, steam generators and other things. Whereas, in the ESBWR, all that's gone because all the safety systems are now in this envelope.

MEMBER KRESS: Since you're operating strictly in the natural convection mode, do you have any enhanced issues with the oscillations?

MR. RAO: No. Because the natural circulation is worth four times that of the traditional four-circulating plant, you --

MEMBER KRESS: I see. You really get a good follow through there.

MR. RAO: Yes.

MEMBER KRESS: Okay.

MEMBER LEITCH: Could you go back to your previous slide for just a moment? The space above the core there, that --

MR. RAO: Chimney.

MEMBER LEITCH: -- chimney or plentum, whatever it is, is there anything in there or is that just an open space to --

MR. RAO: Oh, okay. We do have partitions there. They're one meter by one meter.

MEMBER LEITCH: Just for flow direction?

MR. RAO: Just for flow direction.

MEMBER LEITCH: So it's really just an empty space to give you the differential head that you need?

MR. RAO: Right. Just gives additional driving head through the core.

MEMBER LEITCH: Okay. Thank you.

MEMBER RANSOM: One question one might ask is how do you know one meter by one meter is adequate to prevent the rate of slope transition that could occur in the --

MR. RAO: When we started the initial design we actually had an open chimney, and we went to one meter by one meter because that's where there was data available, so we were sure that at one meter by one meter we could -- we wouldn't have any concerns about flow and bubbly flow.

MEMBER RANSOM: And where is that data from?

MR. RAO: We got -- there was some data from Russia, I don't have the exact reference, okay? That was literature data. And we supplemented it by additional testing at a test facility in Canada. We can provide you the details on that, certainly. But so we've got two pieces of additional data, which provided us confidence that was adequate. In fact, one of the design philosophies was we want to make sure during normal operation we have complete data range. Our expectation is that you probably don't even need that, and if some five or ten years into the operation of a plant you decide you want to take the partitions out, it may be an option. But if there's additional data --

MEMBER RANSOM: One would think that would be an issue, because in pipes the dimension is much smaller than that.

MR. RAO: Yes. No, the -- I don't remember the exact dimension of the Ontario Hydro Test Facility, but I can get you that.

This shows the basic passive safety systems. This is an isometric. You have three pools, I think, which provide the water makeup. It's about 1,000 cubic meters is all you need. The size of the pool -- the size of the safety systems are actually not dependent on the power level. This size is primarily determined by geometrical considerations. It's determined by how much water is needed to fill up the lower drywell. That's all outside. That's why when we scaled up from the SBWR design to the ESBWR it was really easy for us to scale up. In fact, we didn't have to give up any margins. The core always remains covered for any pipe break accident. In fact, it's three meters of water above the core.

In addition to that, we have the standard suppression pool, except that it's raised from the base mat so water can also flow by gravity from this raised suppression pool into the vessel. It provides another backup source of water in case of pipe break.

All pipes and valves are inside the containment, and the decay heat removal heat exchanges, not shown in this picture, are above the drywell head out here. So all the safety systems are basically within the containment envelope. That is where you get the big savings, improvement in economics.

This chart out here shows the comparison of ESBWR parameters to operating BWRs. We've tried to do the comparisons at similar power levels. This is Browns Ferry 3, Grand Gulf, ABWR and the ESBWR. You go from left to right. You'll see small changes in the parameters and basically an evolutionary design in that sense. The active fuel height is 15 percent less, the power density is 15 percent more than the ABWR but still much less than the power density that you're seeing in some of the recent power uprates.

So the Life Star Plant is up at 62, 64 kilowatts per liter. That's a Mark III BWR 6. We eliminated recirculation pumps, the number of control rod drives. This is locking piston for LP; this is fine motion drives. We've got half the number of drives compared to the ABWR similar drives, identical drives to the ABWR.

MEMBER KRESS: Did you have to more Gatalin in to do that?

MR. RAO: I don't know the answer to that question. The safety system pumps basically eliminated them completely. The safety diesel generators also eliminated the vessel pressure. All the parameters -- feedwater temperatures and all of those -- we're keeping them identical for operating plants so that we don't have any of the problems of learning from new designs.

Here is the bottom line: The safety building volume is about half that of the ABWR, less than half that of the ABWR. So that's where you get the big savings in materials. We are basically doing an evolution in the design, which minimizes operations risks. It's a standard direct cycle plant, fairly simple. You pull the control rods and steam comes out of the top, feedwater is pumped in and you drive the turbine. Couldn't find anything simpler than this one.

The design philosophy for core cooling has been basically shown out here to improve the design features. Several of them are new compared -- I listed them as new, because they're new relative to the operating plants. We're using a taller vessel, we increased the amount of subcool water, we've eliminated large pipes below the core. ABWR also did that, but compared to the other plants that are operating, we don't have any large pipes below the core.

The shorter vessel -- the shorter core makes it lower in the vessel, so you've got more water above the core for a pipe break. And in addition to that, because we rely on gravity for water makeup we added the worst depressurization system. All BWRs already have a depressurization system. We've got the worst one on this one. Two very different kinds of valves and going down to two different areas.

And the other thing we're doing is using the TRACG computer code. We're using a code which is based on first principles, not a fixed node code, which has not been fine tuned for the ESBWR. All of the qualification and all the data comparisons we've done we have not done any fine tuning of the code for the application out here. We basically have improved the Plant response by putting in design features, not by improving the analysis, even though we improved the analysis.

I don't have time to go into how the safety systems work, but basically water flows from this upper pool to flood the vessel, as shown on the right-hand side out here.

MEMBER RANSOM: Have you retained the same degree of redundancy in those systems that you did in the SBWR?

MR. RAO: Compared to the SBWR, it's essentially identical. There are one or two very minor differences, which might show -- I'd care to show them out here. In the SBWR, this pool of water, the gravity-driven cooling system pool, was open to the drywell out here, okay? In this Plant, they've closed it off from the driver, it's now part of the wetwell, okay, so there's a connecting pipe out here. And the reason for doing that is not the LOCA response, okay? What it gives us a lower containment pressure. The containment pressure in this plant depends on the relative ratio between the drywell volume and the wetwell aspects. So you want to increase the wetwell aspects.

MEMBER RANSOM: Is there no vacuum breaker valve between the --

MR. RAO: There is a vacuum breaker

between the wetwell and drywell, just like the SBWR. So all that we did was when you get a loss coolant accident and this volume drains down into the vessel or into the lower drywell, this airspace becomes available, and so you effectively increase your wetwell airspace and you keep your containment pressure lower following an accident.

The containment pressure in this plant is not really determined by decay heat directly. It's determined really by where the non-condensable gases are. So it's a question of transferring the gases from the drywell into the wetwell, and that's what determines the containment pressure. So decay heat removal -- the decay heat condensed removal condensers actually had lots of margins in the SBWR. And we doubled the power and we've almost doubled the heat transfer area. Because it's easy for us to increase the heat transfer area in this plant.

MEMBER RANSOM: And condensers, you mean?

MR. RAO: And the condensers. We just added more. And we made them 35 percent bigger, okay? So it's not a major economic penalty to increase the heat transfer area, even though we didn't really need to, because, again, containment pressure is determined not by decay heat removal so much as by the transfer of non-condensable from there to there.

The design philosophy for decay heat removal, of course, is to remove the decay heat from the vessel and if needed remove it from the drywell. You use passive containment cooling heat exchanges, same as the SBWR. We haven't changed the basic philosophy or the basic design; same heat exchanges, just 35 percent bigger. We're relying on the same testing base and the same qualification base using the same computer codes. TRACG was used for the SBWR; we're using it for this plant also.

So we have several diverse means of decay heat removal. We basically followed the same philosophy for our operating plants. The initial steam, blowdown energy, flows to large heat sink suppression pool, basic suppression system. Longer term decay heat flows through the heat exchanges based on the pressure difference. It's not -- because the drywell is at a higher pressure than the wetwell, the steam is pushed through the heat exchanges by the pressure difference.

As I mentioned earlier, the containment pressure is determined by the non-condensables in the wetwell aspects, and that's what controls the containment pressure, not decay heat removal.

MEMBER KRESS: On the longer term decay heat removal, do you have two heat exchanges or just one?

MR. RAO: We actually have four heat exchanges.

MEMBER KRESS: Four? Okay.

MR. RAO: We have four. They have no valves, nothing, they're always open.

MEMBER KRESS: These separate lines go into them?

MR. RAO: Yes. The separate lines go into each of them. And the concept is simple, reliable. There's lot of testing that's been done all over the world at different scales. And the analysis, actually, can be done I still say at the back of an envelope. You just need to do a calculation, transfer the non-condensables from the drywell and wetwell, and you'll know what the containment pressure is within a few Psi. It's not a complicated analysis. Not enough time to go into that.

What I wanted to show on this chart out here was the design features affecting the LOCA response. You know, what we did on the ESBWR, look at the bottom chart out here. Going from the left to right is the ESBWR, ABWR, BWR5, BWR4. What is shown out here is the bottom line. There is just three times as much water in the ESBWR above the top of the active fuel. So the water is in the vessel where you want it. So when you have an accident the water is there. You don't have to bring it in by accumulators or even by gravity; it's all there already. So that's why a loss of coolant accident response is a lot better than that for the operating plants, as shown out here.

This shows the water level above the top of the active fuel after pipe break for different plants. This is a jet pump plant where you get core uncovery and you've got to worry about peak cladding temperature. This is the internal plant, and this is the ESBWR shown in red. You see there's almost three meters of margin to the top of the active fuel, and things don't happen that fast. It takes 600 seconds before it gets down to the minimum water level. And at this stage you only have to make up enough water to account for the boiloff by decay heat. So you don't have to provide much water. That's why gravity actually -- you know, the preferred way for a boiling water reactor, passive boiling water reactor works really well.

The margin to core uncovery is three

meters. We've done all the perk sensitivity analysis and when you measure it compared to the peak cladding temperature it's actually only 0.5 degrees centigrade, so it's much smaller sensitivity to peak cladding temperature. Okay. This is the containment pressure falling at pipe rate. Again, lots of margin to the design pressure.

An extensive technology program has been completed almost over the last 15 or 20 years, and it's a complete program, it's a multi-year program, it involves international partners. Some of the initial testing was reviewed by the NRC, has been observed by the NRC. The NRC's been involved in some of the selection of the matrices, test matrix. And what has been completed? We believe it's very, very comprehensive. Even though the analysis is very simple, we've got -- I don't think any computer course has been qualified as well as TRACG has been qualified for this very simple, unchecked, non-challenging application.

MEMBER KRESS: Did you use the CSAU process to --

MR. RAO: We are using the CSAU process. We are doing the sensitivities, and like I mentioned earlier, the success criteria is the calculation of peak cladding temperature. It's plus/minus 0.5 PCP, okay? So the question is when is enough enough, and we believe we've gone way overboard on this one, as shown out here. The ESBWR is based on the SBWR and the ABWR. We recently submitted over 3,000 pages of new submittals, bring detailed calculations, comparisons. And looking at the PUD, looking at identifying the key parameters, there's overkill, we believe, and we believe that the analysis is fairly elementary and we have to find, I think, collectively, as an industry, a way to move forward. Because every comment we hear from people is that the design is really good, the analysis is not complex, but for some reason the process does now allow rapid closure of some of the issues out here.

Extensive submittals. This shows the interrelationships between the submittals. Again, like I mentioned earlier, some of these calculations can be done on the back of an envelope, but we have extensive submittals. There's the test and analysis plan. What is shown in this chart out here on the right-hand side are reports that the NRC already has, the TRACG model description, TRACG qualification, TRACG application for anticipated operational transient analysis. We will do additional qualification of the TRACG for SBWR and ESBWR. It's almost a two-volume report. It's over 1,000 pages.

There's a summary of all the tests that were done for the SBWR in addition to the detailed test reports, which I believe are 2,000 or 3,000 pages. Then there's additional testing done for the ESBWR, which finally gives us a validated code. Now, this is a computer code that's been used for 20 years, okay, by industry.

And, finally, the application methodology is going to be very simple. As you saw, the uncertainty, the sensitivity of some of the parameters is plus or minus 0.5 degrees. So what we'll do is just combine the parameters in a conservative bounding way. We don't have to do a detailed analysis to get a reasonable answer.

So in summary, what we've done is the passive systems have simplified the plant designs, which in addition to what the calculations show, the gut feel says we've come up with a design which is inherently simpler and is, at least from a gut feel, looks like it's easier for the operator to operate during an accident. The plant evaluations are simpler. You've got less complex analysis, low parameter uncertainty.

Substantial margins exist from the design. They're using a mechanistic code, okay, and we've got a defense-in-depth system. For those who are still uncomfortable with passive safety systems, you've still got the active non-safety systems which are there, which are used for normal operations. You've heard the old story about the boiling water reactor, direct cycle, quiet. Any pump that pumps water can be used to provide water makeup into the vessel, and so in a direct cycle plant, we've still got all the normal pumps needed for the reactor water cleanup or the fuel pool cooling system. We've retained some of those. The PSA told us that it's good to have an LPCI system, Low Pressure Coolant Injection System, so we've made the line connection from the fuel cooling system using the fuel pool cooling system pump, non-safety, which control water makeup. So we've got all of those features in there.

MEMBER KRESS: What do you do about the fuel pool cooling? Do you have to bring a truck in and add water to the fuel pump?

MR. RAO: No. The fuel pool cooling system is a non-safety system. It has enough water for 72 hours. You don't have to --

MEMBER KRESS: Before you uncover the

spent fuel that's in there.

MR. RAO: Yes. Yes.

MEMBER KRESS: Okay.

MR. RAO: And we have provided connections for the outside for --

MEMBER KRESS: Just in case.

MR. RAO: -- 72 hours. It's all there. The basis design is the same as the SBWR. The challenge now is -- there's extensive qualification, the technology issues have been extensively studied. The challenge now is how can we get closure on this and Jim Lyons presented a schedule to you which said it will take 12 months. The last time I made a presentation to the ACRS, Dana said, "Try them and see whether they'll approve it in two weeks." It's 12 months.

(Laughter.)

MR. RAO: Thank you.

MEMBER FORD: I notice that your vessel diameter is the same, your downcomer cap wider, and you've got more fuel rods. Does this not mean, therefore, that the flux on the core internals will be higher?

MR. RAO: The flux on the vessel is about 15, 20 percent higher than ABWR. We're still well below any of the limits that you see on the other operating reactors not --

MEMBER FORD: And the core internal's materials will be welded 316L, presumably?

MR. RAO: I don't know the exact material.

MEMBER FORD: Same as ABWR.

MR. RAO: The same as ABWR.

MEMBER KRESS: But you don't have a beltline weld, as I understand.

MR. RAO: No. You won't have any beltline welds. They are four strings, same as the ABWR. That's one of the reasons why they kept the vessel damage at 7.1 meters. Theoretically, we could go -- you know, we aren't limited technically to the power levels we are at. But what we decided to do was to stay at 7.1 meter vessel, because that's where the industrial capability is right now.

MEMBER FORD: Is the plan to make the internals materials out of noble metal modified alloys?

MR. RAO: I'm sorry, I don't have the answer. Whatever's the latest on the ABWR we'll be using that. Again, we'll be using whatever we'll be learning from the operating plants. The intent on the internals is to make them replaceable. Okay. That's one thing that's a little different than the ABWR here is that they will be replaceable.

MEMBER FORD: And the other question on the materials aspect, what is the experience-based -- one of the new aspects for BWR is this heat exchanger that you have on the top.

MR. RAO: Yes.

MEMBER FORD: Which will only be used hopefully intermittently.

MR. RAO: Yes. On the heat exchanges, there there was extensive testing, which carried them through the life cycle. We actually ran the life cycle -- we tried to simulate the life cycle and the stresses and the behavior during the life of the plant.

MEMBER FORD: There have been studies in terms of the long-term structural integrity of that heat exchange.

MR. RAO: Yes, yes.

MEMBER ROSEN: Is ESBR an acronym for something?

MR. RAO: ESBWR does not stand for European, please, I want to clarify that. ESBWR right now does not stand for anything. The BWR is a boiling water reactor. The ES is still flexible. The highest bidder we leave it up to them.

MEMBER LEITCH: The fuel is rather than 12 feet long is how long?

MR. RAO: Ten feet.

MEMBER LEITCH: Ten feet.

MR. RAO: Yes, three meters.

MEMBER LEITCH: Has there been any experience with fuel of that length?

MR. RAO: Well, there has been fuel of shorter lengths than that in some plants but not -- it will be the same basic design as the GE-1214 or whatever the next evolution of the GE fuel would be. The expectation is that the testing would be done when the plant is built. We always do the CPR testing of that fuel.

MEMBER ROSEN: Have you done a detailed refueling study in terms of the ease of refueling --

MR. RAO: Yes.

MEMBER ROSEN: It's a fairly small containment, so it typically comes up in operation issues as small containments.

MR. RAO: The building is small. The refueling floor is the same size as the Mark III. So in fact we've had utilities involved in this program for the last ten years who are -- in fact, the Finnish utility, which holds the record for refueling outages, made us make several changes to improve the refueling times and outages. For example, even though we've got inclined fuel transfer, the spent fuel is stored in a separate building. We have actually a buffer pool up at the top which can handle 70 percent of the fuel. That was something that the Finnish utility made us put in there.

MEMBER ROSEN: Are there any domestic utilities who are working with you?

MR. RAO: Yes. We've got several domestic utilities working with us. We've got a Utilities Steering Committee, which has worked with us. The domestic utilities joined this program three years ago. EPRI is the official representative, but there are others that come to the meetings. And the old arc utilities are EPRI members.

MEMBER RANSOM: On your containment pressure plot, is that rising mainly due to boil down in the --

MR. RAO: Yes. That's a log plot, so it's extremely exaggerated out there.

MEMBER RANSOM: You have up to 24 hours.

MR. RAO: Yes. What happens is that there is some heating up that's going on, and so that --

like I mentioned earlier, the containment pressure is determined 80 percent by the non-condensables of the air being pushed over from the drywell to the wetwell. So that's why you get that initial rise, okay? Then the long-term is determined by the vapor pressure.

MEMBER RANSOM: Where does this curve go beyond the 24 hours?

MR. RAO: Well, we've carried it out to 72 hours on the ESBWR.

MEMBER RANSOM: It's dry at that point.

MR. RAO: No, no.

MEMBER RANSOM: I thought it was 72 hours you had to refill the --

MR. RAO: Yes. You've got to refill the outside external pools.

MEMBER RANSOM: Right.

MR. RAO: Yes. You've got to refill the external pools.

MEMBER RANSOM: Does the containment pressure then go back down when you refill these?

MR. RAO: No, it stays there. Again, like I said, the pressure is determined by where the air is. So you've got to bring the air back. It's not a decay heat removal issue, it's more where the air distribution is.

MEMBER RANSOM: When does the equilibrium get maximum pressure?

MR. RAO: It actually goes to peak refuel six hours into the transient. This is before the gravity-driven cooling system drains out, okay? So you see that pressure goes down because at that stage the gravity-driven cooling -- there are two dips out there. Let me see if I can -- I have to go back quite a bit. Oh, okay. There are two dips out here. This first decrease is when the gravity-driven cooling system, water, quenches the steam in the vessel.

MEMBER ROSEN: I was looking at the SBWR.

MR. RAO: Yes. The SBWR -- God, it's been so long since I looked at that one. When you look at the blue one, the phenomenon is similar. What happens is -- so this is when the steaming is decreased in the drywell. When that happens the vacuum breakers open and it sucks the air back into the drywell. So the pressure is coming by where the non-condensables are, basically. That's all you're talking about, distribution of the non-condensables.

MEMBER RANSOM: So beyond this 24 hours it continues to decrease then?

MR. RAO: Well, it basically stays steady.

MEMBER RANSOM: It burps back and forth?

MR. RAO: Yes. It goes back and forth a little bit. Out here there's a little bit of decrease, because as the gravity-driven cooling system drains out, you increase the wetwell volume by about 15 percent. That pools adds another 15 percent margin to the fuel cells.

MEMBER ROSEN: You say you eliminate large pipe below the core and minimize other pipes. Are there any pipes below the core?

MR. RAO: Yes, right here. There are four two-inch nozzles at the bottom of the core. That's part of what's called the reactor water cleanup system. That's used during start-up and stratification. There are no pumps. You need to prevent stratification at the bottom of the vessel during the start-up. And so that's what they're used for.

MEMBER ROSEN: Two-inch pipes.

MR. RAO: Two-inch nozzles.

MEMBER ROSEN: Two-inch nozzles. So your total diameter is --

MR. RAO: There's two-inch nozzles.

MEMBER ROSEN: I'm trying to get to the largest size break.

MR. RAO: Two inch.

MEMBER ROSEN: Two inch. But it's not a two-inch --

MR. RAO: Line is greater outside.

MEMBER ROSEN: Oh, okay. So the size of the break is two inches in diameter.

MR. RAO: Yes. There a couple of two-inch lines, there's the reactor water cleanup line. The gravity-driven cooling system lines are also two-inch nozzles. They come in above the core somewhere out here. These are some of the lines that are the -- the big lines are the steam line and the feedwater line. Those are fairly high up in the vessel.

MEMBER ROSEN: Is it correct that if you have a bottom drain line break, you still have enough water in the entire system to maintain the core cover even when you flood that lower compartment?

MR. RAO: Yes. The lower volume there is about 1,000 -- is what's shown out here. This is for a main steam line break, but I had one for a bottom drain line break. What happens is actually the size of the spool is such a size to keep the core covered up to the top of the active fuel.

MEMBER ROSEN: Okay.

MR. RAO: And this is about 700 cubic meters, it's not a very large volume. And this is about 1,000 cubic meters. And so there's a couple of hundred cubic meters to fill up the --

MEMBER LEITCH: But you have control rod drive penetrations coming out the bottom, right?

MR. RAO: Sure.

MEMBER LEITCH: And instrumentation penetrations.

MR. RAO: Yes. Those are the same. And those -- you know, we've also looked at water in the opening areas for some reason during shutdown. What would be the biggest drain at the bottom? You don't -- still again two-inch nozzle is the biggest opening that you'd have during a shutdown in the bottom also. Okay. So we've looked at shutdown PSAs and we've looked at all of these issues.

And, again, like I said, it's a fairly simple elementary design. Everyone seems to like it. And we're still looking for the two-week review that Dana promised us.

MEMBER ROSEN: You're looking for a client and someone to help you name it.

MR. RAO: Well, a client would be helpful too, yes.

MEMBER POWERS: If I get them a two-week review on this, they'll name it after me.

(Laughter.)

MEMBER KRESS: ES stands for extraordinarily simple.

MR. RAO: Yes. Lots of people like this. Even though the ABWR is our current product, U.S. utilities have expressed an interest in this, and they want to know about it.

MEMBER KRESS: Okay. Yes, I guess we better move on to the next. Thank you very much.

MR. RAO: Thank you.

MEMBER KRESS: It was very interesting. Who is up next? Is it the CANDU? Jim, I guess you're coordinating this.

MR. LYONS: I am? Framatome.

MEMBER KRESS: Framatome, okay.

MR. LYONS: SWR1000 will be next.

MR. STOUDT: Good morning, or is it afternoon?

I'm Roger Stoudt, I work for Framatome ANP as an advisor engineer in Lynchburg, Virginia. And I'm here today to present an overview of the SWR 1000, and with some particular focus on the passive safety features of the design.

I would like to say, just before I start, that as I told the NRC staff in August that I'm really happy to be here, because I didn't think that during my career I would ever see interest in the nuclear power plant again in the U.S., and it is kind of refreshing to think that there might be a chance for us.

Just briefly, the SWR 1000 design is an evolution of technology that got started back in the '60s. As you can see, the plants are listed here that have been built and operated, not all are still operating.

But back in '68 there was the Lingen plant with the first fine motion control rod drive. Later on at Brunsbuttel was the first use of internal recirc pump.

And then the latest designs, of course, are at Gundremmingen B and C. And the SWR 1000 uses a number of the same internal components in the reactor vessel from those plants.

The SWR 1000 design was initiated back in the early '90s. Testing programs started about '95, and the design has evolved to where it is viable today.

Just briefly, some of the characteristics of the plant are, thermal power is 3370, normally electric net is 1250. The plant originally started out as a lower power level design, hence the name SWR 1000.

But it probably will settle out at 1250 megawatts in its final form. There are 664 12x12 fuel elements. The active length is about three meters, 157 control rods. We retain the recirculation pumps, there are 8 of those.

The reactor pressure vessel is 75 bar, or close to 1100 PSIA. We have 8 safety relief valves, and some of the passive component ratings are shown there; the emergency condenser -- and I will point out where these things are located in the next slide or two, and discuss those at some, in a bit more depth.

The containment cooling condensers, four of those are rated at 4.8 megawatts, and we have four passive flooding systems, the containment diameter is 32 meters, and its design pressure of 7.9 bar, 115 PSIA.

MEMBER ROSEN: I haven't run the calculation yet but it seems like this is a very efficient plant. Am I correct?

MR. STOUDT: Efficiency is around 35 point something percent.

MEMBER ROSEN: What do you attribute that to, the increase over -- it seems a little, at least ten percent higher.

MR. STOUDT: I think it is about the same as the prior Gundremmingen plants.

MEMBER ROSEN: Is it? Okay.

MR. STOUDT: It depends on the application. Some of the applications in Europe have very cold water available for the condensers at the end of the turbine, that helps a lot. We may not get those kinds of efficiencies in the U.S., depending on the application.

The basic safety approach is that all the active systems have passive safety related backup to perform nuclear safety functions. And, in fact, the passive safety features will keep the plant safe without use of any active systems.

This is a composite slide that illustrates the basic features of the plant. The plant has four containment cooling condensers. And this is the way ultimately all the heat, all the energy inside containment, is removed.

There is a dryer separator storage pool outside containment. And the energy inside containment is transferred by these containment cooling condensers. There is no valves, they simply start to operate if there is a significant temperature gradient inside to outside.

In addition we have the emergency condensers for heat removal from the reactor pressure vessel. There are four of those, there are four core flooding pools. Again, the ECs are passive devices, no valves open, they are simply connected to the reactor vessel. And if the water level drops inside the vessel the condensers begin to operate.

So for a range of design basis events the energy inside the reactor is transferred to the core flooding pool. Eventually, as this pool water gets hot, and begins to generate vapor steam, that is condensed by the containment cooling condensers, and the energy is removed from containment.

So ultimately these are the devices that keep the containment pressure down, or remove the energy that is being dumped inside the containment building.

There are eight safety relief valves, steam relief valves to prevent reactor overpressurization, and also to depressurize the reactor.

In addition, these core flooding pools -- again, there are four of these -- they are connected. But they are separate pools. And each pool has a core flooding line, which is connected to the return line from the emergency condenser.

In addition there are four passive outflow reducers which were installed on the return line from the EC. The reason for those, it is essentially a fluid diode, so that in outflow the resistance is increased drastically to prevent too much water from exiting the reactor vessel, and leading to core uncovery.

MEMBER LEITCH: Roger?

MR. STOUDT: Yes?

MEMBER LEITCH: That dryer separator storage pool, there must be some walls or something there that are not shown. That would appear, how does that work during refueling operations? I don't understand that.

MR. STOUDT: Well, the refueling pool is over here, okay? And the handling equipment is up above it. So the reason for the name is that the internals are stored in here during refueling.

MEMBER LEITCH: There must be some walls that are not shown?

MR. STOUDT: Yes, there are lots of things. This is a very conceptual drawing, there is lots of things that aren't shown here. I do have, if we have time for it, I might be able to find a slide here that is a detailed cross-section of the plant.

A plan view and an elevation view, I think I've got it some place with me.

MEMBER LEITCH: Okay. I see what you are saying, yes.

MEMBER ROSEN: We used to call electrical, water diodes, or whatever you call them, check valves. Is that what you are talking about? Passive --

MR. STOUDT: I've got a picture of it a little bit later, so I think you will see what I'm talking about.

MEMBER ROSEN: Oh, okay.

MR. STOUDT: No, there are no moving parts in it.

MEMBER KRESS: It is like the one they used to have in --

MR. STOUDT: Pardon me, which reactor?

MEMBER KRESS: Are you familiar with the device they had in the PIAS reactor?

MR. STOUDT: No, I'm not. It may be very similar.

MEMBER KRESS: Yes, they called it a diode, no moving parts.

MR. STOUDT: Okay. In addition I would

just like to point out a couple of other items. There is a dry well flooding line shown here, which in the event of a sever accident would flood the dry well and cool the reactor from the outside, to retain melt inside.

There are vent pipes, 16 vent pipes, these vent pipes, in the case of a LOCA, would vent steam into the pressure suppression pool, and condense it in the process. There are overflow lines between the core flooding pool and the pressure suppression pool, which allow any excess water condensed up here to flow into the pressure suppression pool.

And there are also these hydrogen vent lines. So that any hydrogen accumulating near the top of the containment would be directed down into the pressure suppression pool and be removed.

There are two residual heat removal systems shown here. They are not necessary to maintain the safety of the plant. They are available, they can remove water from both the pressure suppression pool, and cool it, return it to the core flooding pool.

The return lines aren't shown, just the suction lines, so there is a connection here. The pressure suppression pool, and also one from the reactor vessel, to be used for decay heat during shutdown.

MEMBER ROSEN: That looks suspiciously like a pump in your graphic. Is that what it is?

MR. STOUDT: Here? Yes, it is. Yes, that certainly is a pump, yes. But it is not, as I said, it is not necessary for mitigating any of the design basis events that might occur.

MEMBER ROSEN: Just for normal shutdown?

MR. STOUDT: It can be used, it is an active system that can be used. It serves the pressure, the function of low pressure coolant injection as well.

You can remove the water from the reactor vessel, send it through coolers down a heat exchanger in this area, and return it by the feedwater lines. So it can be used that way, but it is not necessary.

We can demonstrate adequate accident response without use of the residual heat removal, or LPCI system.

This is an illustration of the emergency condenser. Again, there are no valves in the loop. During normal operation you see, essentially, the water level. Under those conditions there is no circulation through the emergency condenser.

This anti-circulation loop at the bottom prevents hot water from circulating internally within the pipe and returning, similar to the trap on a typical water heater.

When the coolant level drops, and all it needs is about seven tenths of a meter, then circulation begins. Steam flows into the emergency condenser, where it is condensed, and returns to the reactor vessel.

There are four of these things. Each of them is rated at roughly 66 megawatts of energy removal capacity.

MEMBER RANSOM: There must be something missing in that left-hand side.

MR. STOUDT: Yes?

MEMBER RANSOM: You either have it filled with water in the upper part, or something, because it is just a manometer, and it has to balance --

MR. STOUDT: The steam comes down to -- there is a subtle change in colors here. And right about here is the interface between the steam and water. The water here is, of course, at ambient temperature, at core flooding pool temperatures.

And hot water from the reactor vessel stops right about here. So you have some stratification in temperatures.

MEMBER RANSOM: All right.

MR. STOUDT: But that is what balances things, okay?

MEMBER FORD: I take it, you said that is normally stagnant during normal --

MR. STOUDT: Yes.

MEMBER FORD: -- then you've got a steam-water interface?

MR. STOUDT: Yes, right here.

MEMBER FORD: How do you deal with hydrogen/oxygen explosive mixtures?

MR. STOUDT: I'm sorry, hydrogen and oxygen?

MEMBER FORD: I'm thinking of the Brunsbuttel incident recently.

MR. STOUDT: These pipes, this looks horizontal here, but these are designed so that any radiolitic gases, if that is what you are referring to?

MEMBER FORD: That is what I'm referring to.

MR. STOUDT: Will rise and leave the loop. They won't accumulate anyplace because the relative, again, the elevation changes aren't apparent here, but it is designed so that you have continuously increasing --

MEMBER FORD: There will be venting lines?

MR. STOUDT: Yes, right. The other device is the containment cooling condenser. There is one of these located above each core flooding pool. Of course each core flooding pool also contains an emergency condenser.

And these, ultimately, are the devices, as I noted before, that remove the energy from the containment building into the dryer separator storage pool.

Again, these devices, there are valves, there are valves in both lines. But they are there for isolation and closing them off. During operation the valves are always open, so there is nothing that opens or closes to get these devices to function.

If the pressure starts to come up, and the temperature comes up in the containment building, because of the presence of steam, the steam condenses, cold water from the dryer separator storage pool, relatively cold water, I think the design temperature is 100C, begins to circulate through the tubes of the containment cooling condenser, condensing the steam, returning it to the core flooding pool.

MEMBER WALLIS: What comes out is hot water in the --

MR. STOUDT: It could come out as steam. I mean, depending on the temperatures one could get a vapor mixture coming out of this return tube.

But essentially for all events it would require about two or three days before anybody would have to worry about refilling dryer separator storage pool. There is no operator action required.

This does show finned tubing. Actually the current design doesn't use finned tubing, the fins have been eliminated.

This is the thing I alluded to before, my fluidic diode, the passive outflow reducer. This is what is installed in each return line for each emergency condenser. And it functions by changing the rotational component of the flow, depending on which way the coolant is going.

So normal flow direction in this direction, of course, corresponds to a pretty direct path through this component, through the slots in this component. And relatively low flow resistance.

If a pipe should break out here somewhere, and the flow reverses, then there is a significant rotational component imposed, and it essentially is equivalent to imposing a very large flow resistance.

Tests on a device of this type have shown, roughly, a two order of increase in flow resistance, depending if it is in or what.

MEMBER ROSEN: Two to the order of magnitude?

MR. STOUDT: Magnitude, yes. So the K values would go from -- by a factor of 100, and you get about a tenth of a flow in the outflow direction as inflow.

This is a device called a passive pressure pulse transmitter. It is a patented device, and it is there to actuate reactor scram, main steam line isolation valve actuation, and to depressurize the reactor, in case that should be required.

Again, the device itself has no moving parts. Under normal operation, where you see the water level reactor vessel, again, this thing is filled with cold water, and nothing is happening.

It has a primary side, as you can see, and a secondary side. It is sort of a shell and tube heat exchanger of sorts. And the secondary side is also filled with water connected to a pilot valve.

When the water level drops during an accident scenario, the steam begins to flow into this device, and the steam heats the secondary. The secondary pressurizes because of the energy being input, and activates the pilot valve.

Which, in turn can, depending on where these things are located, can initiate reactor scram, can close the main steam isolation valves, and it can open the steam line and relief valves to depressurize the reactor.

This is a very simplified picture. There are actually four levels. There are twelve of these in total, and installed at three different levels. The highest PPPTs scram the reactor, the set below that, if the water level continues to drop, would isolate the main steam lines, and depressurize the reactor.

The very lowest ones activate, or scram the reactor closed main steam isolation valves in the case of water level increase. These devices require no electrical power.

It is true that the subsequent actuation systems downstream do involve valves. But, again, there are no electrical signals, or any kind of electrical power required for these items to work.

And, finally, in the event of a sever accident condition, there is a core flooding line, which ends in -- actually the core flooding line splits, in the most recent configuration, and each exit piping, or each exit line contains two valves.

The valves are actuated by a signal from the safety INC that is measuring water level. There are reactor water level measuring devices, and when the water level gets to, I think it is roughly 13 meters, the top-most valve opens, and if the water level continues to drop, I think the second valve opens at about 6 meters, which is well into core uncovery. And the assumption is, of course, that the severe accident is underway.

There is sufficient water in the core flooding pools to flood the dry well, and still keep the ECs covered, the emergent condensers, which I showed you a couple of slides back.

And then the flooding establishes a flow path between this reactor vessel insulation, and allows the lower head to be cooled sufficiently to retain the melt inside the reactor vessel.

MEMBER WALLIS: Is there another vessel outside the vessel?

MR. STOUDT: This is the insulation package. There is a gap between the two.

MEMBER WALLIS: But there is a container, there is another container -- the insulation between those two cylinders -- insulation is not important, it is --

MR. STOUDT: Yes, it is just creating a path for the flow to be heated and then rise, and the vapor, boiling water, would be cooled by the containment cooling condenser, returned to this core flooding pool.

And, of course, this line is open. So that completes the flow circuit into the dry well.

MEMBER POWERS: What makes you think that the metallic portion of the core melt is less dense than the oxide portion?

MR. STOUDT: Why do I have it shown stratified here? Well, I'm not an expert on this, I'm not going to pretend to be.

MEMBER POWERS: Well, I am.

MR. STOUDT: The analysis has been done by our colleagues in Germany. The person, in particular, I think his name is Nicolai Kolev, who has done a considerable amount of analysis.

I'm quite sure we could very easily get whatever information you would like to have about that. I'm not going to attempt to explain the stratification.

MEMBER POWERS: In 1989 the prediction was made that that would not be the case, that the metallic fraction of the melt would be more dense than the oxidic reaction. That prediction has recently been confirmed by some experiments in St. Petersburg.

MEMBER WALLIS: The Russian work, right?

MEMBER POWERS: That is right. If you have the metallic fraction in contact with the vessel, what prevents a vigorous inter-metallic reaction in the trading vessel?

MR. STOUDT: I don't know the answer. I will certainly record that question and find out.

MEMBER WALLIS: And that color, which is outside the vessel, is the same color as the core melt? What is that?

MEMBER POWERS: That is the metallic, inter-metallic reaction penetrating the vessel.

MEMBER WALLIS: But it stops.

MR. STOUDT: Where is this? You mean like here?

MEMBER WALLIS: Yes.

MR. STOUDT: I don't, no, I don't think that -- I think it must have been the artist's rendition in creating the slide.

MEMBER WALLIS: It shouldn't be there?

MR. STOUDT: No, it shouldn't be there.

MEMBER WALLIS: But Dr. Powers thinks it might be there.

MR. STOUDT: I think he does, yes, clearly.

MEMBER POWERS: I think it is very accurate.

MR. STOUDT: Well, we will have to make sure we fix that, then. It will be easy, right? All I have to do is remove this colored portion and the problem will go away.

I was going to say that I have a brief list of experimental work that has been done, and there has been some investigation, at least the heat transfer of the flow regime, the heat transfer on the outside of the vessel.

But I understand what your question is, and it has nothing to do with the heat transfer on the outside of the vessel.

MEMBER POWERS: It will have a spirited impact on the heat transfer because it changes the material properties of the two fluids, and introduces a chemical compound into the heat generation rate that will get -- capture your attention, especially if the melt is very zirconium rich.

MR. STOUDT: Yes, okay, point noted. Just briefly, some of the testing that has been performed, all in Europe, has been the test of the emergency condenser, the containment cooling condensers, the PPPTs, passive outflow reducer, RP flooding line test, the reactor pressure vessel exterior cooling test is still ongoing.

There was a conga test at the Paul Share Institute that looked at the containment cooling condenser heat transfer in the presence of aerosols. That had broader application than just SWR1000, it also looked at some PWR components, and vapor suppression, pool scrubbing of aerosols, and aerosol effects on hydrogen recombiners.

And then, of course, there is the scram tank test. That is to -- we have a steam driven scram tank, so that rods are driven in by expanding steam space in top of the scram tanks.

That is used instead of nitrogen because we want to be certain that we don't inject any nitrogen into the reactor pressure vessel and scram, and thereby potentially compromise the performance of the emergency condensers.

There are some future tests still upcoming. The fast-acting injection system, spring supported check valve. This is on the RP flooding line, it is a full scale test of the valve function.

And, also, event pipes and quencher tests, looking at the flow dynamics, structural loads. And, finally, this isn't particular safety related, I don't think, but some tests of mechanical drive components, the control rod drives, things that are different from prior applications. That is what has been done so far, and planned so far.

In summary, potentially the SWR1000 has added water inventory inside the reactor pressure vessel, and inside the containment, that increases its ability to ride through accidents without core uncovery.

We have a nitrogen inverted containment atmosphere, and rely on passive equipment for heat removal from both the reactor pressure vessel, and the containment.

The key safety functions are also activated by passive components, the PPPTs. And, finally, we have a system to provide for external coolant and RPV. And possibly RPV in cases -- at any rate --

MEMBER POWERS: It is very much like what we just heard about from GE, except you still got the pumps in there.

MR. STOUDT: We do have the recirculation pumps in there. They are retained largely because we feel that the operational response is better with them, the power maneuvering between 60 and 100 percent which is, often, a value to the customer who is operating the plant, depending on how he is loading it.

So, yes, the pumps are there. They are wet rotor pumps. At any rate, the final point I guess I would make is that in the event of transients, LOCAs, design basis events, utilizing only the passive safety features of this plant, we can mitigate accident consequences for a period of several days, until personnel will have to take action.

And largely the action they would have to take would be to replenish the water in the dryer separator storage pool outside containment.

Thank you, gentlemen, that concludes what I have to say. Any further questions?

MEMBER RANSOM: One question might be the coolers, the finned tube coolers that you have inside the containment, you have non-condensables present there, and you would wonder how much reduction and heat transfer capability does that -- how do you handle that?

MR. STOUDT: Let me go back to the --

MEMBER RANSOM: That is the passive safety systems containment cooling condenser.

MR. STOUDT: It is not finned any more.

MEMBER RANSOM: Well, I guess whether it is finned or not you would wonder, you are still going to have non-condensable build up on the surface, whether it blows away by natural circulation, or --

MR. STOUDT: Yes. What happens, I will refer you to this slide, what happens is that, yes, in the event of some sort of severe accident, where you generate hydrogen, and --

MEMBER RANSOM: Well, you have nitrogen in the containment, normally, right?

MR. STOUDT: You do, as the containment begins to pressurize, you have these hydrogen overflow, these hydrogen vent pipes up here. And the non-condensables will flow, do flow, into the pressure suppression pool, and accumulate in the inner space, or this open space, above the pressure suppression pool.

MEMBER RANSOM: Well, it is assumed that the non-condensables in the steam will separate off of the fins, or off the tubes, or?

MR. STOUDT: Just by virtue of the fact that you are pressuring this whole upper part of the drawing, the dry well. The increased pressure, low pressure here, will cause the flow to go through the hydrogen vent pipes.

MEMBER RANSOM: That won't go on forever, you will eventually pressurize that --

MR. STOUDT: Yes, that is true. This thing, there will be some at the top. But I think the calculations that have been done show that most of the non-condensable remain above the active surface of the CCCs.

There have been tests done at PSI, ate the PANDA facility, where the dry well was simulated, as well as the wet well, with connection of these vent lines between the two, and conditions that were predicted to exist during various design basis events were simulated in that test.

And, yes, the heat transfer can degrade somewhat. But adequate performance was demonstrated in the test. Each one of these, I think there's four of these, and each of them has 50 percent of the required design capacity.

So one could have some degradation, obviously, and heat transfer.

MEMBER RANSOM: Well, you have run tests, then, of that configuration, using nitrogen steam mixtures?

MR. STOUDT: Actually air was used to simulate the nitrogen, and helium was used to simulate the hydrogen. But, yes, those components were put into this test.

MEMBER RANSOM: Well, helium you would worry about the difference in molecular weight, or density, between that of nitrogen --

MR. STOUDT: It doesn't quite, the flow patterns are not quite -- the direction of flow through the condenser tubes is different. But it is in a direction that would give you lower performance in tests rather than higher performance.

MEMBER ROSEN: This plant has eight main recirc pumps?

MR. STOUDT: Yes.

MEMBER ROSEN: Where are they? I know they are not shown in this --

MR. STOUDT: Right here, those are the recirc pumps, right there.

MEMBER ROSEN: Well, they are internal pumps?

MR. STOUDT: Yes, the pumps themselves are internal, and the design is of the wet rotor design, so there are no seals.

MEMBER WALLIS: Why do you have so much water, why is it so deep?

MR. STOUDT: Well, this is a simple schematic to represent the different --

MEMBER WALLIS: To help catch this debris that is falling down?

MR. STOUDT: I don't think so, I don't think there is quite that much space down there. If I had the actual cross section of the design. This is, you know, likewise there seems to be a huge amount of space around the reactor in the dry well, and the core flooding pools seem awfully small, and that is not true.

I mean, this is to illustrate the various components and concepts. But I would not take this as the absolute scale of the various parts.

I would also point out that these condensers were also tested in the aerosol tests I mentioned earlier, where they were subjected to various particles that were, in turn, deposited on these surfaces.

That is one of the reasons, of course, the fins are -- have been removed, is that the fins seem to be a good accumulator of debris. Without them that won't happen.

And I think under those conditions the heat transfer degraded by about 20 or 25 percent, but there was more than enough excess capacity to compensate for that degradation.

MEMBER RANSOM: Just one further question. This is a lot of similarities to the ESPWR. I'm wondering what is the advantage of retaining the pumps?

MR. STOUDT: Well, from my perspective, the advantage is an operational advantage, changing the power relatively rapidly, particularly between 60 and 100 percent, and that is why it was -- that is why they were retained.

I'm quite sure you can get it to work the other way.

MEMBER WALLIS: I think you're, talking about the passive outflow reducer, to show a bit more what is happening, in order to make it clear why it works.

MR. STOUDT: Oh, okay.

MEMBER WALLIS: I'm not asking you to explain it, this is not a very good explanation.

MR. STOUDT: Not a very good -- well, we

will see if we can improve it. I think the central issue is simply the rotation impart to the flow in each direction.

In this direction there is very little rotation, and the flow can --

MEMBER WALLIS: So what does that do? I mean, the maximum loss is still the same, there is nozzles at the top.

MR. STOUDT: Well, it makes it easier, there is a more direct path, and there is less flow change.

MEMBER WALLIS: There must be centrifugal force, there must be focusing of the vortex, as you make the radius smaller. There is a lot of things going on that aren't indicated here at all.

MR. STOUDT: Yes.

MEMBER WALLIS: I don't ask you to explain it.

MR. STOUDT: I do know that I have seen --

MEMBER WALLIS: But they do work, they do work?

MR. STOUDT: Yes. And I've seen the flow, the curves that illustrate form loss as a function of flow. And, yes indeed, they do increase the form loss significantly; two orders of magnitude, in fact.

Test data has been obtained, they do work. It is sort of the pragmatic engineering --

MEMBER WALLIS: They are called vortex valves, aren't they?

MR. STOUDT: Well, I don't know, these are called passive flow reducers. I have seen all sorts of different arrangements, and I've always thought of them as fluidic diodes, but whatever they are called.

If there is no questions I will sit down and concentrate on the core melt issue.

MEMBER KRESS: You are on, please introduce yourself.

MR. SNELL: Good morning. My name is Victor Snell, I'm director of safety and licensing for ACR. I would like to introduce, also, two colleagues sitting towards the back there, Mr. Vince Lyman, who is the manager of licensing for the U.S. application of ACR. And next we have Mr. Cal Reed, who is giving us the specialist licensing expertise up at Bechtel.

MEMBER KRESS: Does Snell mean you are a fast person?

MR. SNELL: Yes, that is the root. In the next short while I'm going to cover seven topics, which I believe is the committee's request to us; what is the ACR, a rather short presentation on the main drive report, which was meeting customer requirements.

Most of the discussion will be on technical summary, including safety improvements, and the technology base. A brief comment on where we are, as a status. An issue which may be of interest to the Committee on what I call licensing opportunities, and then a summary of conclusions.

So what is the ACR? Advanced CANDU reactor, is the acronym, 700 stands for the power level. It is an evolutionary extension of the proven CANDU 6. CANDU 6 is our main single unit design of CANDU.

There is 8 units in operation right now in four continents, two units are currently under constructing. And I'm pleased to report that the first unit in Xinjiang in China went critical last month.

The picture here shows the four CANDU 6 units operating at the Walsing site in South Korea.

MEMBER WALLIS: Which is the fourth continent?

MR. SNELL: South America, North America.

MEMBER WALLIS: Oh, those are two continents?

MR. SNELL: Last time I checked. The main drivers for the evolution has been to meet customer requirements. We are aiming, at a specific overnight capital cost on the fifth design of 1,000 dollars U.S. per kilowatt.

Our construction schedule is 36 months, and you can see 30 dollars per megawatt hour, a capacity factor in excess of 90 percent, and a plant operating life of 60 years.

We are reasonably confident that we can meet things such as the construction schedule, because of the recent experience we have had building in both Walseng and Xinjiang, where -- particularly in Xinjiang both the schedules were met.

However, when you say to achieve low capital costs, you have to make some evolutionary modifications to current operating CANDUs, and that has driven some of the design changes that I will be summarizing.

Current operating CANDUs, as you know, natural uranium fuel, use a heavy-water coolant, and a heavy-water moderator. On ACR major changes to relax the constraint of natural uranium fuel --

MEMBER KRESS: Does that mean you are going to use five percent?

MR. SNELL: Bear with me for a minute.

MEMBER KRESS: Oh, sorry.

MR. SNELL: I mean, the answer is no, it is actually much slower than that. Once you do that, you have a lot of freedom which you don't have on the existing operating CANDUs.

So the first thing you can do is use light-water coolant, and that means you can replace all of the expensive heavy water with light water. You can then reduce the core size, because current CANDUs are somewhat over-moderated, and then reduce the amount of heavy water moderator, as well as reduce the amount of heavy water coolant.

Because you have a few excess neutrons you can increase the pressure tube thickness, which allows you to raise the reactor coolant system pressure, hence the thermal efficiency.

Having said that, we have retained all the other intrinsic proven CANDU features, which is why this is an evolutionary design. So that one change has allowed us to develop a number of benefits in terms of economic optimization.

I'm now going to start building the reactor from a sort of the central part out, just go through, quickly, some of the design features.

The first, the most important part is the fuel, it is a short bundle, it is only about that long. There is a real one behind the very back of the room, full length. You can see the shiny thing near the light switch, and that is a real CANFLEX fuel bundle, full size. It is about 1.6 feet long.

As with other CANDUs, we do on-power refueling. This design, the CANFLEX refers to the geometry. There are 43 fuel rods in this bundle, and to answer your question, the enrichment is relatively modest, it is 2 percent SEU in 42 of them, and natural uranium plus 4 percent dysprosium in the center one, and I will come back to that in a second.

MEMBER KRESS: Now, you stack these?

MR. SNELL: They are stacked 12 in a row, yes, on end, so they make up a string.

Fuel burn-up is very modest compared to light-water reactors. We are not pushing it at this point. We think we can get a lot more out of it than the current targets, 20,500 MW days per metric ton.

It is a little higher than the CANDU average. We have achieved that in some selective cases in Canada, but it is higher than the average. It is quite modest with respect to light-water reactors. We think that as a future product development we can push that higher.

We have managed to get both higher bundle power, and lower rod rating, because of the change in bundle geometry.

MEMBER KRESS: Now, that central rod, is that a burnable poison, is that --

MR. SNELL: Yes. So here is a schematic diagram of current CANDUs versus the ACR. A current CANDU on my left, your left as well, I guess, is a 37-rod natural uranium fuel. You are looking at a cross section, you are looking at it end-on, that is surrounded by a Zr niobium pressure tube, there's a little gas gap, about that much.

And then there is a thin Zr-2 calandria tube. The changes to ACR, the pressure tube diameter is the same, inside diameter is the same. This is the CANFLEX fuel bundle. The different colors actually represent different sizes of pins. There is a slight increase in size in the central ring, compared to the outer pins, that is for balancing the thermohydraulic performance in it, there is and, again dysprosium in the center pin.

The pressure tube is slightly thicker, so you can pump up the coolant pressure a little bit.

MEMBER KRESS: How are they supported on the inner pressure tube?

MR. SNELL: Yes. It is not shown in the diagram, but the bottom elements have little bumps on them called bearing pads, and this lifts them off the pressure tube.

You can see it in the model, actually, afterwards. And, by the way, the model is at NRC if anybody wants to look at it.

The gap, and I will come to this in a minute, why we do this, but the gap is larger between the pressure tube and calandria tube. We had to change the material on the calandria tube to Zr-4. It is also somewhat stronger.

So that is the fuel channel. That is the end of pretty pictures. The pictures I will show you now are actually from the 3D cads design. So we have left the artist's conception, and we are actually pulling material off the plant design.

This is the reactor itself. I will take a little bit --

MEMBER POWERS: Can I ask you a question about the previous slide?

MR. SNELL: Sure.

MEMBER POWERS: You get electrochemical potential between the tin alloy and the niobium alloy on the calandria and the pressure tube?

MR. SNELL: Electrochemical --

MEMBER POWERS: Potential difference?

MR. SNELL: No, not that I'm aware of. I mean, there is a mechanism for high drive migration if you are not careful. But there is no -- I'm not aware of any electrochemical interaction.

MEMBER POWERS: Well, there are two different materials.

MR. SNELL: You mean Zr-4 and niobium?

MEMBER POWERS: Yes.

MR. SNELL: They actually don't touch, they are separated.

MEMBER POWERS: They don't have to.

MR. SNELL: I'm not aware of anything. We've had various types of zirc in CANDUs in the past, and I've not seen anything like that.

MR. LANGDON: My name is Vince Langdon, and as Victor said, I'm the licensing manager, I also happen to be a fuel and fuel channel guy in my previous lives.

We have about a half a million pressure tube years of experience. We've never seen that kind of thing.

MR. SNELL: This is the reactor assembly, it is not a vessel. So we will start, again. These are the fuel channels, the horizontal fuel channels. And you can't see it in this diagram, but if you think of this vessel as a cylinder, this vessel constitutes what is called the calandria.

It is steel, it contains the low pressure, low temperature moderator, moderator runs 60 to 70 degrees centigrade, and supports the fuel channels. The fuel channels are supported at either end.

Surrounding the calandria we have another thin vessel called the shield tank, and it is simply there to provide biological shielding, and it is filled with light water, which provides thermal and biological shield.

The reactivity mechanisms come in two ways, most of them come in from the top, and go from this deck up here, and they go into the moderator, not into the coolant. So they act in the low pressure environment of the moderator.

We do have some detectors, and some units for the second shutdown system, which come in horizontally, through the shield tank, and again into the calandria, into the moderator.

So all the devices act in the moderator itself.

MEMBER KRESS: Is your two percent

enrichment enough to get rid of your positive void coefficient?

MR. SNELL: Yes. Reactor coolant system, if you look at it from this level upwards, it is very similar to a PWR. Basically you have two steam generators and four pumps.

If you look at it from that level downwards, then it becomes like a conventional CANDU. If you, again, each of these little dots is a channel, each channel is connected by a feeder pipe which goes up here to the things in red, which are collectors, or headers.

The headers then connect up, if they are inlet header, it connects from the pump. If it is an outlet header it connects to the steam generator. So they are just large pipes above the core. There are no large pipes at or below core level.

The parallel arrangement of the pumps means you can tolerate pump seizure, single pump seizure. And because of the elevation of the steam generators, with respect to the core, you can -- you do have natural circulation, and even with some void in the system.

MEMBER WALLIS: Your moderator is really cold.

MR. SNELL: Yes.

MEMBER WALLIS: And it is very close to the hot water that is cooling the --

MR. SNELL: Yes. And if you recall that at the cross section of the channel is a gap between the pressure tube and the --

MEMBER WALLIS: That is all that insulates one from the other?

MR. SNELL: Yes. You lose a few megawatts of heat -- the normal heat load to the moderator is in the order of 100 megawatts in thermal. So you do lose some heat.

MEMBER SHACK: Then all the feeder materials, are they still carbon steel, or have you --

MR. SNELL: No, because of some experience that we've had in Canada, and also because of the higher flow velocities, the bottom half of the feeders is all stainless steel in the ACR.

MEMBER SHACK: And what is the top half?

MR. SNELL: The top half, I believe, is still carbon, it is a transition joint.

MEMBER ROSEN: So you have a moderated cooling system in place?

MR. SNELL: Yes. It is not shown in this diagram, but basically there is inlet and outlet pipes near the top part of the vessel here, and they go to heat exchangers and pumps down here.

MEMBER RANSOM: So the flow through the core is countercurrents, some channels go one way, and the others go the other way.

MR. SNELL: Yes. Every channel is -- every adjacent channel goes the opposite direction.

Safety systems, nothing very different here from current CANDU practice. This is a cutaway of the same diagram you saw before. There is actually two independent shut down systems, in addition to the control system.

So there are actually three independent ways of shutting the reactor down, two of which are -- they are all for design basis accidents. We have a number of shutoff rods, which drop in the gravity into the moderator, that is our first shut down system.

MEMBER POWERS: What are those rods made of?

MR. SNELL: I think cadmium, I believe it is cadmium.

MEMBER POWERS: Just cadmium?

MR. SNELL: No, it is clad, cadmium cladded steel, I believe.

MEMBER POWERS: Cadmium cladded steel?

MR. SNELL: Yes, I'm not one hundred percent sure, but I believe that is correct.

The other system consists of perforated tubes. They start perforating once they enter the calandria. These are connected to a pressurized tank filled with gadolinium nitrate. And on a signal the tank, the valves and tank open, and inject the liquid poison into the moderator itself, actually into the reflector, the reflector and the moderator are sharing the same vessel.

In addition we do have four control absorbers, which are part of the control system, which can also shut down the reactor for most accidents.

Emergency core cooling system is, again, nothing very different. We have, I think -- we have initial injection from high pressure water tanks, and in the long term you have pump recovery.

MEMBER WALLIS: So you inject into the reflector, but it mostly goes into the moderator?

MR. SNELL: That is right, yes.

MEMBER KRESS: Do you put boric acid in your coolant?

MR. SNELL: No. We don't need, we don't need any reactivity control in the coolant.

MEMBER KRESS: You use burnable poisons?

MR. SNELL: In the ACR design we have burnable poison in the central fuel pin.

MEMBER SIEBER: Is it a simple, or an expensive process to clean up the moderator after you've injected into it?

MR. SNELL: It takes about 36 hours, you have to circulate the moderator through ion exchange columns. So it is expensive because you lose 36 hours of production time.

MEMBER SIEBER: It is not so expensive that it would become a psychological impediment for an operator to --

MR. SNELL: No. Containment, I'm not going to spend much time on. It is basically a steel lined dry pressure containment. It is very similar to a PWR-type containment. It is nothing unusual about that.

This is a -- in fact I misled you. This is a schematic, just so you can see it. This is a cross-section of the containment. And it shows something we developed initially on CANDU 6, and it evolved through a design we call CANDU 9, and intend to apply here.

This is an evolutionary design, but it has some passive features. One of the passive features is an elevated reserve water tank high in the building. This is an outgrowth of the, what is called the dousing tank in CANDU 6, and it provides water under gravity head, for a number of different sources, namely you do have a direct connection to the reactor coolant system, with more valves than you see in that picture.

We can also add water to the steam generators under gravity, and to the moderator and the shield tank. And the second, but maybe not obvious, why you would want to do that. If you have a reactor that shut down, and there is no water in the channels, you can take away heat to the moderator without melting the UO2. Down to the fuel, but you would not melt the UO2.

And that is fine if the moderator, heat exchanger and pumps are working. If they are not working we can back that up by topping up the moderator for about two days. So we have provided makeup capability to the moderator, so that if we do get into LOCA, plus loss of ECC, plus loss of moderated heat removal, we have a passive backup make up system.

We can also add it to the shield tank. That is somewhat of a last resort, but because that surrounds the moderator, it has the potential for either slowing down or arresting relative slow core damage progression.

MEMBER WALLIS: Graceful isn't the technical term that is used by the NRC.

MR. SNELL: Yes. The reason I'm using it is because the collapse of the core in a CANDU is relative incoherent. You start off as you boil down the water, you will start forming a debris bed, which gradually collapses. It is not like in a Canley. So it takes some time.

This is a highlight of the safety improvements relative to operating CANDU. As one of you already mentioned we have designed it to have a small negative void coefficient. You can place the emphasis where you like.

To me the most important thing is the word small. I'm sure down here the word negative is equally as important. Both give you relatively mild transients on the loss of coolant.

In fact, if you have a loss of coolant you have a slow rundown in power, without depending on the shutdown systems. We do need the shutdown systems for shutdown, but we don't need them as fast.

Once you have a negative void coefficient you do end up, then, increasing the power coefficient in the operating range. Current CANDUs, the power coefficient is about zero, and they operate just fine.

A more negative power coefficient means there is less duty cycle on the control system. CANFLEX fuel is a thermal optimization of our current fuel. So one does get larger thermal margins. So the actual margin to critical channel power in ACR is about ten percent higher.

Current CANDUs, if you have, for some reason, a pressure tube failure, say to an undetected drawing defect, which leaks and you let it go, it may or may not be contained within the surrounding calandria tube.

And the design basis for CANDU is, in fact, failure of both the pressure tube and the calandria tube. But with the stronger calandria tube that is much less likely to happen, and we believe under almost all circumstances, a spontaneous pressure tube failure would actually be contained within the calandria tube.

That is of economic interest to the utility, that is also an aspect of defense-in-depth. Notwithstanding that spontaneous failure, both will be in the design basis.

MEMBER SHACK: Would I have a leak detection system in that space?

MR. SNELL: Yes, exactly correct. It is employing a gas system and you detect moisture in that.

MEMBER KRESS: And I know it by pressure tube that --

MR. SNELL: With some you can narrow it down very quickly to a small group of pressure tubes, and then you can narrow it down further. Once an operator picks up a leak, though, his instructions are to shut down and depressurize, then look for it. You have a lot of time, but that is the instructions.

Improved heat sink reliability, I will cover it very briefly. I won't spend too much time on that. The ACR 700 is being designed as a twin unit plant, and we have, rather carefully, put in inter-unit ties of some of the safety support systems to enhance their reliability.

This has been done in CANDU, actually, with a lot of success on the multi-unit plants, so there has been a fair amount of experience on that.

A single channel failure, because we are using light water, and the heavy water moderator, if you do have a failure of both the pressure tube and the calandria tube, the reactor will tend to shut down, as you displace heavy water with light water.

Containment I've mentioned. We have extended the seismic qualification relative existing CANDUs. So, for example, a main control room does not have, is fully functional for safety reasons after an earthquake.

MEMBER KRESS: How do you displace heavy water with light water? They are just commingled and the light water floats up on top?

MR. SNELL: Well, you've got a channel sitting at about 12 -- I think it is about 1,800 PSI, if I do it quickly in my head. And so if the channel breaks you have a very large pressure differential blowing light water into the heavy water moderator.

MEMBER KRESS: And where does the heavy water go?

MR. SNELL: Well, it mixes like crazy, and then it will rise up. There are rupture disks on top of the --

MEMBER KRESS: Oh, there is rupture disks. Okay, that is what I was looking for, okay.

MR. SNELL: Severe accident prevention mitigation I did cover, with the reserve water tank. We have done, it is called a generic CANDU PRA, it is actually a focus PRA on CANDU6. It was aimed at sort of picking up the high risk areas and saying, identifying them, and is there anything we can do about them in terms of dominant risks.

We have obtained some design insights from that generic PRA, and are using that in ACR. And as the ACR design is progressing we are doing a sort of design assist PRA, along with the design.

Technology base, we've been operating CANDU reactors since the early 1970s. This is an evolutionary version of an operating CANDU. ACL and the CANDU utilities are responsible for developing and maintaining that technology base. Unlike in the U.S. where a lot of the research is done by the NRC, in Canada most of the research, not all, is done by ACL.

We have 2000 people at Chalk River Laboratories involved in various aspects of CANDU technology. The picture here shows one of our main work horses, it is the NRU reactor, which you can't see too well.

This is the top of the reactor there, and the reactor itself is below them. It is a large reactor, physically. It is used for fuel materials and safety tests, will be used for testing the ACR fuel.

The ACR R&D is anticipatory, which means that we expect to focus on a fairly modest extension of the data bases, slightly higher coolant pressure, slightly thicker channel materials, slightly higher temperatures.

Certainly there will be some component testing. We have simplified the fueling machine, we will be testing that quite extensively. The other thing is to confirm the code validity of our existing computer codes under extended ACR conditions.

MEMBER KRESS: Do you have an irradiation embrittlement issue with the pressure tubes, or the calandria tubes?

MR. SNELL: There is a lifetime issue.

MEMBER KRESS: That is what I meant.

MR. SNELL: And that is an early R&D thing, where you take samples and try accelerated radiation, yes.

These are just two examples, I'm not going to go through them in the time remaining. But the R&D is focused on the obvious things, fuel, fuel channel, fuel handling, online refueling.

Certain components we've improved, and safety code qualification. And these are two examples of some of the test results. This is a zero energy reactor called ZED-2, very long history. It is a reactor kit, you can change lattice, do whatever you like with it, and that will be used for fundamental physics measurements on the fuel, and on the ACR lattice array.

This is a moderated test facility. This is set up for a design we call CANDU 9, it will be reconfigured for ACR, which is slightly tighter packing of the channels, and will validate the computer codes which predict moderating temperatures.

It is a fairly sophisticated thing. It doesn't look sophisticated, but it is. You can measure three dimensional velocities through the entire vessel, using laser belt monitoring, and you can also measure three-dimensional temperatures.

So you get pretty good information in terms the way your moderator is --

MEMBER WALLIS: Do you use CFD in the moderator?

MR. SNELL: Yes, it is based on a 3-D water code.

Where we are, we've completed the concept. The ACR 700 is our reference design. We are also looking at ACR 1000. The decision between those two will be driven by our customer needs.

We expect to have the non-site-specific engineering complete in 2005. A company called Hitachi is investing in BOP -- balance-of-plant -- optimization, and plant-wide modularization. We have defined the construction strategy and schedule, and we are working with Canadian, U.S., and U.K. utilities to bring ACR to commercialization.

Which leads to the next point, we have started a pre-application review with US NRC staff. We expect about two years, somewhere between 18 months and two years. And that would be followed either by an application for standard design certification and/or combined license, or both.

And I think it is a bit early to see which direction utilities will want to go at this point. We've also started, in parallel, what we call in Canada pre-licensing review. It is very similar to -- it is a little more than a pre-application review, and a little less than standard design certification.

It has the same objective, which is to assure utility of low licensing risk before they commit to a plant. We have done that before, in Canada. The is a history of it. We've started it again, and that would confirm license ability on the Canadian regulations, the thinking being that it would certainly reassure people outside of Canada, that the plant was licensed within country of origin.

There is a possibility of pre-licensing in the UK. As most of you know there is, what is called, a white paper due in early 2003, which will set a direction for the nuclear power program in the UK.

And I think until that white paper comes out it is not very clear which way the UK will head.

MEMBER WALLIS: I thought they were going out of business?

MR. SNELL: British Energy is, for other reasons, because of the privatization of existing market, is in some difficulty right now. But that won't affect the long term need for nuclear in the UK. That will be done by the white paper.

So I think that is going to have to settle down before we see where that is heading. Certainly British Energy is interested in that as a replacement of the advance gas cooler reactors.

Licensing opportunities, this is a little different from some of the concepts you may have heard. It is a mature technology, and one of the, I think, interesting challenges in licensing it in the U.S. is to what extent, and the method to use to use the extensive Canadian regulatory and R&D and operating experience, in the NRC review.

Very clearly NRC will have to license it in the U.S., by itself. It is the legal entity here to do so. So the issue is not that. The issue is to what extent can they incorporate and use the Canadian experience, but without repeating it.

And what the sub-bullet here says, how can NRC put a program in place for acceptance of equivalence in meeting safety requirements?

MEMBER WALLIS: Do you have a risk-informed regulatory process?

MR. SNELL: The Canadian process has been risk -- it has been very heavily influenced by risk in the early days. It has become a little more prescriptive, actually, as time goes on. But if you look carefully you can see the risk groups, and the way the accent class is set up.

We also were doing PRAs 15 years ago, so there is a heavier risk component in the design, in the way you approach design. It is not quite the same as risk informed here, but the basic idea is the same.

So this is the challenge, I think, can the NRC requirements be made flexible enough to accommodate a technology which is both similar to and different from light water reactors?

Similar to, we use pressurized water as a fluid. A lot of the components look the same. It is different when you get to the core level. Cooperation with parallel regulatory views in Canada, and possibly the UK is, I think, a key aspect of this.

Some of that is about to start. There are -- regulators are starting talking to each other. And we hope that they will focus on the extent to which there is common ground, and the extent to which these reviews can be made consistent.

Conclusions, and I'm sure glad to hear that this is the last slide before, the second to the last slide before lunch. It is an evolution design, building on proving CANDU 6 design operation. It is driven by a meets the market economic, schedule and risk requirements.

A use of CANFLEX fuel geometry with slightly enriched uranium contributes to improvements of both economics and safety. That is our big change. The R&D in our view, is anticipatory, and it is a modest extension of conditions and components.

NRC review requirements and processes could take advantage of prior CANDU licensing experience, along with parallel reviews in Canada, and possibly the UK.

That was the last slide. Thank you very much, gentlemen, I'm very happy to take any questions.

MEMBER RANSOM: Do you still use CANDU for the -- I mean CATHENA for system accident analysis?

MR. SNELL: You didn't miss it, and the answer is yes, we still do. I didn't mention it but, in fact, yes it is our main line for the hydraulics codes.

MEMBER KRESS: Did you say you were going to maybe have ten of these units on a site?

MR. SNELL: The design is for twin units, twin units on the side.

MEMBER KRESS: You said twin?

MR. SNELL: Yes, sorry.

MEMBER KRESS: Any other -- do you want to make some -- well, I certainly want to thank all the speakers. I'm sure this will be highly useful to both the Staff and the ACRS, when they get around to actually reviewing the certification process.

So thank every speaker very much. It has been very enlightening.

CHAIRMAN APOSTOLAKIS: Okay, we will break for lunch until 1:30.

(Whereupon, at 12:28 p.m. the above-entitled matter was recessed for lunch.)

A-F-T-E-R-N-O-O-N S-E-S-S-I-O-N

1:31 p.m.

CHAIRMAN APOSTOLAKIS: We are back in session. The next subject is the license renewal application of Catawba and McGuire. Dr. Bonaca, it is yours.

MEMBER BONACA: There has been quite a bit of time allocated to this but, in reality, all I need is about 20 minutes to give you a briefing on what took place on the subcommittee last Tuesday.

At that subcommittee meeting we reviewed the application, and the SER, and we also came to the conclusion that we did not need an interim letter, and also we do not need a full presentation to the full Committee from the Staff and the Applicant.

So I will give you a brief report on what took place. Again, we met on October 8th, with the Staff and Duke personnel to review the license renewal application, and associated SER for the McGuire 1 and 2 and Catawba 1 and 2 nuclear plants.

These four units are all Westinghouse PWRs in ice condenser containment, and they are pretty much identical, with the exception of some components. For example, vessels are fabricated by two different manufacturers.

The four units are rated at 3,400 and 11 megawatt thermal, for approximately 1,150 megawatt electric. And their current licenses expire between June 12th, 2021 for McGuire 1, and 2026 for the newest of the four plants.

So only McGuire unit 1 qualifies for license renewal consideration because the -- having operated for 20 years already. The NRC had to approve an exemption request.

And the basis for the exemption request was that the other units are similar, and there was a common application being submitted for all four units. The reason why I bring this up is that there have been two intervenors on this application. And one of the issues they raised was this one.

I believe that right now the issue is not any more under consideration by the Commission. The only remaining contention, under consideration by the ESOB, is the severe accident mitigation analysis for station blackout.

And the concern is the loss of igniters during station blackout would lead to a containment challenge. Now, this issue, we felt, is with the current licensing basis of the plant, it doesn't have to do anything with the license renewal, it is being evaluated on a separate track.

We discussed it but, essentially, it is not an obstacle to our review at this stage, nor to granting the license renewal to these four units.

Now, the subcommittee, at the end of the presentations, concluded that the license renewal application is well organized, incorporates Oconee application experience, but also one thing we noted is that it is quite concise, and we've gone, now, from the original two volumes plus we had for the other plants, to just one condensed volume. Well organized.

But together with that we also noted that this application required 273 formal RAIs in order to complete this review. So, you know, the Commission asked us, specifically, to comment on the efficiency and effectiveness of the process.

You know, I think we asked the Staff to let us know what they think about, you know, how far should the application go in being concise, and then when would that become ineffective, or inefficient, given that at some point that requires so much additional information being pulled out of the Licensee. It doesn't speak of the quality, it speaks of the complexity of reviewing the whole application.

The SER came to us with 42 open items

still unresolved. And that is the reason why it was decided to put on the agenda a significant time, because we thought maybe because there would be contentions, we may need to write a letter.

Now, a month later, when we met to review, in the subcommittee meeting, the number of open items was reduced to eleven open items, only. That is apparent that probably the SER came to us too soon.

And so one question was, should we set the criteria for the number of open items addressed on an application before it comes to us? Because we spent a lot of time reviewing the open items, and by the time we came to the subcommittee meeting, there were just a few left.

MEMBER SHACK: Did they resolve the small-bore piping issue?

MEMBER BONACA: That is not resolved yet, and that will be brought up later on. And it is interesting, on that issue, the program that Catawba and McGuire have is one where they have, under the service inspection, the inspection of piping, small bore piping, but only in risk-significant locations.

The Staff is looking more to understand if small-bore piping is, in fact, a concern at all. And from that perspective you want to look at susceptible location. I think they are looking for a one-time inspection of susceptible location. Correct me if I'm wrong.

PARTICIPANT: I don't know that it is a one-time inspection. It may be ongoing inspections. But the Staff is looking to confirm that the risk informed process accounts both for susceptibility, and for consequence.

Once we determine that then we will know that the sample include susceptible locations, and the Staff will be satisfied with that.

MEMBER SHACK: But, I mean, the last license renewal we looked at they got through their small bore piping because they, at least, they had a formal risk informed inspection with respect to the piping.

So this is an informal risk informed --

MEMBER BONACA: Well, they also had identified, if I remember, a number of susceptible limitations in the nuclear --

PARTICIPANT: My understanding is that for McGuire unit 1 they did propose a risk informed process in accordance with the WCAPs. So it should be a fairly formalized process.

What the Staff is looking at its own SER, and perhaps the WCAP as well to make sure that susceptible locations are part of that risk information, not just consequence of a crack failure.

Does that answer your question?

MEMBER SHACK: Well, I guess I have to go back and look and see what the basis for accepting the last small bore inspection piping plan was in the license renewal process. Just an apparent inconsistency, but that may be my memory.

MEMBER BONACA: I thought the susceptibility was always the --

MEMBER SHACK: Well, susceptibility is always part of the risk informed WCAP.

MEMBER ROSEN: Westinghouse approach looks at susceptibility, what are the active mechanisms of degradation, and then do they occur, and in what locations, consequence.

MEMBER BONACA: Yes, looking at the previous application inconsistency you should look at. In fact, you know, just continuing, it is interesting that one of the reasons for these open items is that Duke proposed that fan and damper housing -- and there was an agreement with the industry, because they want to rely on loss of function rather than degraded conditions for that verification. I will discuss that briefly in a little while.

Now, the residual open items that there are, about eleven, don't appear to be an impediment to the projected final SER for January 6, 2003. I think there are, in general, good technical reasons for the difference between them.

Now we, as a subcommittee, felt that the SER was excellent. It was a true improvement over the previous one that we reviewed, and so was the staff presentation to the subcommittee, and we felt that it should be used almost as a template for future presentations to the subcommittee.

Both the application and the Staff evaluation provided adequate technical information this time, and the subcommittee could really form an opinion on the adequacy of programs and monitorings and PRAs.

Now, the subcommittee questioned the presence of some equipment out of scope. The responses could be a little better. A member of the subcommittee questioned the use of PNIB only to bridge the methodology to the list of components that have been identified, and they understand the drawings, they identify lines and piping and so on and so forth and goes down the list of components that belong in the scope.

The point that Steve made was that there are other drawings that identify additional components such as supports. Now, the inspector that came here to give us a presentation on the subject pointed out that they believed that they have all the additional elements are captured by commitments.

Still, I think, Steve has a good suggestion.

Again, as I mentioned, there are five issues on scope that are contested, one is the fan housings, damper housings, and you can see once again that the position of Duke is that failures should be identified by functional failure in the housing or in the component. The Staff feels that the components that could affect pressure boundaries should be in scope, just as items in the statement of consideration that indicates the casings of pumps are in scope.

You cannot wait until you have casing failure to identify the problem. That is really not something that plants like to do.

The other issue was on fire protection. There were a number of issues on fire protection; most of them were closed. Steel jockey pumps and the manual suppression in potential fire exposure areas are not in scope, but they are being debated.

MEMBER ROSEN: I think we had a resolution on the jockey pumps.

MEMBER BONACA: Jockey pumps is already in scope for your performance, I understand. So there is no precedent on that. I believe that we have solved the issue on that. On the issue of surveillance, the issue was raised by a number of members regarding the culture, we got an indication of safety culture, and you know, there was no clear answer provided except, "Yes, and indication is provided by this kind of performance."

On the other hand we also considered the fact that indication of culture or behavior today does not say much about what it will be tomorrow, but it tells about the importance of focusing on the issues.

Just as part of this presentation, we had discussed description of the currently existing programs -- five augmented programs and fifteen new programs, in which eight are one-time inspections. A detailed review of these programs shows that there are a lot of commitments, not hard data. For example, you know, subject criteria are promised, but they are not there yet. You will have commitments over the next twenty years.

So the Committee noted that, you know, by the time you get closer to the license renewals for these plants, there will be a bow wave of work for the NRC.

These are a lot of plants scheduled to go to license renewal about the same time, and it will be an enormous amount of information that will go into those documents; it has to be tracked, it has to be verified by the NRC, inspected probably.

And we may want to point that out, as a comment, we are responsible to the Staff requirement coming back from the Commission, that they have to be answered to by some time, probably, next spring.

Because I believe it's going to be significant load for the NRC.

MEMBER ROSEN: It is almost like, excuse me, Mario. It is almost like the startup test program, you know, where the NRC comes in to verify the startup test program. All those plants will be entering a new licensing environment --

MEMBER BONACA: In a very --

MEMBER ROSEN: -- and the NRC will have a burden trying to -- being required to say, "They are ready, they met all the commitments they made during the licensing."

MEMBER BONACA: Now the Staff is developing a procedure that they will use to track these commitments. They also have developed a new licensing process to help future reviews.

We asked a number of questions about the programs. Concerns were expressed that they have to invest in an internal inspection program, proposed by the applicant, that would only rely on the Occonee I inspections. And there was no basis for McGuire coming over the boundary. To that, we would answer that the Staff have already considered that, and they -- Duke has committed to specific inspections every time at both McGuire and Catawba. So that is an issue that is resolved.

Again, the reactor vessel inspection program should include also susceptible location of small-bore piping, and that issue, actually the in-service and safety inspection, that issue is not closed yet. It will be closed when we hear about it in January.

Residual open items don't seem to be an obstacle to, again, to having this SER delivered to us in January. So they are planning on it in the February meeting.

One last note about time utilization

analysis. We felt that the information provided was quite extensive. We got detailed data regarding embrittlement margins.

PARTICIPANT: Oh, you are planning on doing it? Okay.

MEMBER BONACA: Yes. So there is sufficient margin. Remaining open items include evaluation of pressurizer subcomponents, surge nozzles subjected to outsurge and usage factors being monitored, environmental procedure specs, and underclad, cracking concerns with McGuire 2 due to lack of depth for this plant.

At the end of the meeting the subcommittee members provided the following observations -- these are observations by one or more of the members. First of all, again, an excellent SER, excellent presentation, and we would hope to have this format of information as a template for future presentation. Individual concerns again were fire protection, this issue of the culture, heightening surveillances, the complexity of the whole fire protection issue, the importance of the sites to be addressed in fire issues. There was concern that groundwater is -- essentially, they found groundwater not to be aggressive at this stage; but the feeling that Steve has is that monitoring groundwater is not such a pain, and would be an improvement.

Again, concern with the bow wave of commitments that will come with all these units at the same time; not so much concern with the plants -- I mean, they have their own plants and they can take care of themselves -- but concern about the Staff, handling so many plants in a reasonably short period of time.

And finally a concern, a lot of it expressed by Dana, with the breaking down of systems into active and passive components. We had different opinions on that.

MEMBER ROSEN: We don't have to resolve these things until February, right?

MEMBER BONACA: Well, they will have to come up with the solution.

MEMBER ROSEN: But we don't take any position now?

MEMBER BONACA: One comment that --

MEMBER SHACK: the licensee chooses to include it.

MEMBER BONACA: But there has been some debate on specific generic issues, as they call them, and closure that really were understood to be pretty much acceptable for the industry. Hopefully there will not be -- in this case there were reasons for reopening because in some cases they thought that Oconee, I mean, they filed them in October. They started the preparation of the application before there was general guidance. So you understand why the discrepancy is there.

Anyway, the bottom line is that truly, there was no intent that the report should be written, in particular because since we are not doing it now on Oconee, any time we write a written report we send a message to the staff. And there is no message to be sent right now.

With that, I'll conclude my presentation. I don't' know if any of the members --

MEMBER POWERS: Mario, there was a question that arose during the discussions of the subcommittee about the jockey pumps?

MEMBER BONACA: Yes.

MEMBER POWERS: I would just comment that I checked with some of my fire protection buddies, and asked them a question about prejudice, one way or another. And without even thinking, they said, "Of course there is."

PARTICIPANT: There was a little bit of a disagreement between the Staff and the Applicant on this issue, and Tuesday, when I presented to the subcommittee I indicated that previous Applicants for license renewal had included the jockey pumps, or if there was a tank that maintained pressure on the main fire header that they would include that, even Oconee.

And Mr. Greg Robeson of Duke's staff chimed in and indicated that at Oconee they did not include the jockey pumps. And I remember looking at this, and I remember talking with the Duke folks before Tuesday's meeting, and distinctly remember seeing the PNID that indicated that they were in scope.

So I just wanted to report to the full committee that I've done the research, going back to the Oconee application, and the drawing, and the jockey pumps for Oconee fire protection system were in scope. Thank you.

MEMBER BONACA: Are there comments from members, or questions from members that were not there?

CHAIRMAN APOSTOLAKIS: Thank you, Mario.

(Off the record discussion.)

MR. KING: For the record my name is Tom King, I'm with the Office of Research. I'm called a consultant now, but I'm sort of an NRC special employee, and I report to Mr. Tadani, and I have assignments in the advanced reactor area, some international stuff.

I have been working on policy issues associated with advanced reactors, focusing on non-light water reactors. As Dr. Kress mentioned, there was a SECY paper that went up, back in July, to the Commission, that you have been briefed on, and then sent a letter on, SECY 020139.

We are not quite as far along as your comments suggested, the opening comments suggested, Dr. Kress. I'm here today as a status report. The paper is due to the Commission the end of December.

We are in the process of gathering information right now in terms of what are the options for resolving these issues, what are the pros and cons of the various options.

And what I'm here today to talk to the Committee about is where we stand in terms of identifying options, and pros and cons. We are not asking for a letter at this point, but we would like any verbal feedback we get regarding those options.

We also are conducting a public workshop October 22nd and 23rd, it is going to be at the

Doubletree hotel up the street, to get input from other stakeholders on the options, and the pros and cons.

After that then we will start to formulate recommendations, and maybe at the end of the briefing we can come back and talk about future interactions with this Committee, where we can start talking about recommendations leading up to the paper in December.

We would, probably, request a letter from the committee in December. I would hope we could get on your December full Committee agenda, give you a draft paper in advance of that, where we would talk recommendations, and then get your formal input prior to that paper going to the Commission.

MEMBER KRESS: When did you say your workshop was?

MR. KING: The workshop is October 22nd and 23rd. On the 22nd it begins at 1:00 in the afternoon. There is a Federal Register notice out that gives the agenda.

Advance reactors are still alive and well at NRC. There are, right now, five advance light water reactors in various stages of either review, or planning for review.

There are three non-light water reactor

activities which are really the focus of most of what I'm going to talk about today. Those are the GTMHR, the Pebble Bed is still alive, although we are not actively reviewing it right now, there are discussions taking place regarding the resumption of that review.

And then there is the Department of Energy Generation 4 activity, which is looking at various non-light water reactor concepts. There are also three yearly site permit applications expected next year. Which, to some extent, have a bearing on some of what we are going to talk about today.

Just quickly, by the way of background, you are probably all familiar with this, the current regulations really are a combination of generic and light water reactor oriented regulations.

If you look at non-light water reactors in the past we've done it on a case by case basis. I was involved in the Clinch River review, where we had to go through all the regulations, identify which ones applied, which ones didn't, and what additional requirements, or license conditions had to be added to deal with the fact that it was a sodium reactor.

We also had to comb through all the generic safety issues that had been identified for light water reactors and identify which ones applied, and which ones didn't.

MEMBER KRESS: Did this process also take place for the Clinch River breeder reactor?

MR. KING: Yes. I'm using that as an example, since I was personally involved in that.

MEMBER KRESS: You also did it for the earlier MHTGR.

MR. KING: For MHTGR we did something similar at the pre-application stage, and I imagine Fort St. Vrain, probably, went through a similar process.

And all of that is subject to litigation on a case by case basis. So, you know, there is some element of duplication that you have to go through on a case by case basis. There is some potential for inconsistency in the way things are interpreted as you go through each of those reviews case by case.

Back in '86 the Commission issued a policy statement on advance reactors encouraging these pre-applications --

MEMBER ROSEN: Fort St. Vrain was the only one we actually issued the license to.

MR. KING: Yes, Fort St. Vrain was -- well, there was Peach Bottom 1 before that, that was really early in the game, and that was sort of a demonstration plant. Yes, Fort St. Vrain actually got licensed. Clinch River was probably a few weeks away from getting its CP.

MEMBER SHACK: How about Fermi?

MR. KING: Fermi 1, yes, that was also a demonstration plant too, as I remember. FFTF got a safety review, did not get a license, and it was reviewed only on the design, the site was not looked at, emergency planning was not looked at, only the design.

MEMBER ROSEN: What is the significance of seeing a demonstration plant, is that licensed under 103 instead of 104?

MR. KING: I'm not sure. Those were back in the '60s, and back under the AEC, and I can't really talk to the differences of what was done then, versus what is done now.

MEMBER POWERS: You are absolutely correct about the licenses, the clause in the Atomic Energy Act that you get licensed under, there is a difference, I don't know what else that means.

I know that it's significant, I mean this license by test concept, but I don't understand all the ins and outs of it.

MEMBER RANSOM: Tom, the DOE reactors are not licensed by the NRC, is that right? Like the production reactors?

MR. KING: Production reactors are not licensed, and I mentioned FFTF got a safety review, but it did not get a license.

MEMBER POWERS: That was just because DOE was asking NRC to do it, not doing it because they are required to do it?

MEMBER RANSOM: That is my understanding, they were not required to do it.

MR. KING: Fort St. Vrain had a pre-stressed concrete reactor vessel with a steel liner, which was really treated, in the safety analysis, like a container. I just went through the Staff SER on Fort St. Vrain.

And then they had the confinement building around that, with no pressure-retaining capabilities. So depending on how you look at Fort St. Vrain you can say it had a containment, or it didn't have a containment.

MEMBER POWERS: But there are good things to be said about confinements.

MEMBER RANSOM: I think one of the design basis accidents on Fort St. Vrain was loss of closure, you could flow out the bottom closure and -- I remember I was asked one time to see if it would take off like a rocket.

MR. KING: Yes. I don't recall what the design basis accident -- they had a depressurization as a design basis accident, but I don't remember --

MEMBER RANSOM: They called it the loss of closure, which was the main closure on the bottom of the reactor.

MR. KING: I will go look at the SER again, but I don't remember seeing that in the SER.

Anyway, the Commission had issued a policy statement back in '86 encouraging activities at the pre-application stage to settle some of these major design and policy issues associated with these plants.

And that is really what we are into now with the pebble bed, and the GTMHR, and we've gotten far enough where we felt it was time to go to the Commission with some of these issues, and try and get some feedback, and that was the SECY paper that went up in July.

The scope of the issues really deal with reactor design and operation. We are not dealing with fuel cycle issues at this point, nor security issues. That will be dealt with separately.

I mentioned the schedule already. We will come back after this and talk a little bit about, maybe, future interactions with the committee.

Now what I would like to do is talk about each of the seven issues that were in the SECY paper. The first three are really what we call overarching issues. They have the potential to impact all the other issues, and have a broader scope than the last four issues.

And the first one of those, in the paper, is what we call expectations for enhanced safety. If you recall, in the Commission's advance reactor policy statement, they encouraged -- actually they said they expected advance reactors to have enhanced margins of safety.

They said as a minimum, though, that the plants had to meet the same level of safety as currently operating plants. The severe accident policy statement, which actually preceded the advance reactor policy statement, said that they expected plants to have an enhanced performance severe accident performance.

And then the safety bill policy was issued in '96, and when the Commission issued their SRN in 1990 on the safety bill policy, the Staff had recommended a more stringent core damage frequency goal for future plants. The Commission turned that down in 1990 and said they expected the industry to develop designs with enhanced safety, but they are not going to take those industry goals and turn them into regulations.

MEMBER KRESS: What does CDF mean for a gas cooled reactor?

MR. KING: I think you can define it various ways. You can define it on the basis of fuel temperature, you could define it on the basis of the number of expected particle failures, you could define it on some amount of air that would get in there.

CHAIRMAN APOSTOLAKIS: But can you define core damage in different ways, for different reactors, and still have the same goals?

MR. KING: The same goals?

CHAIRMAN APOSTOLAKIS: Well, the Commission is 10 to the minus 4, I mean, that's what the Commission has at this time?

MR. KING: Yes, I think we can. I don't think defining core damage frequency in gas reactors is a major obstacle, it is just a matter of sitting down and deciding --

CHAIRMAN APOSTOLAKIS: Well, there should be some consistency, I think, for the --

MEMBER KRESS: I think the consistency will come --

CHAIRMAN APOSTOLAKIS: Isn't it 10 percent of the noble gases -- release of 10 percent? I think that is what the definition is.

MR. KING: Yes, I don't recall exactly. But you can come up with some equivalents. I don't think that is a big issue.

MEMBER WALLIS: Wouldn't you have some trouble defining what current level of safety is?

MR. KING: I think you have certain metrics that you can use, core damage frequencies --

MEMBER WALLIS: Are you going to take some average of that, or are you going to take the current level of safety?

MEMBER ROSEN: 103 plants, we add up all the CDFs, and divide by 103.

MEMBER WALLIS: But are you going to take the maximum, or some goal level of safety?

MEMBER KRESS: Well, you have to take the maximum.

MR. KING: I would take the safety goal subsidiary objectives. That is what we are shooting for, for the current fleet of plants.

MEMBER WALLIS: That is not the current

level of safety. It's a goal, but it is not the reality.

MR. KING: It is not reality, but I don't think we have any really good measure of reality. We have the IPs that look at internal events, and external events. We don't have --

MEMBER WALLIS: Isn't that the whole problem? Unless you have a base of current level, you can't really say what's being advanced, what's not being advanced.

MEMBER KRESS: I would, personally, think this would be an opportunity to make 10^{-4} a national requirement. I mean, rather than a goal.

MEMBER ROSEN: Ten to the minus four is not the goal, because if new plants were designed ten to the minus four, that would result in a reduction, a deduction in safety, compared to the last plants that were licensed.

MEMBER KRESS: Yes, but I think that the first sub-bullet is probably a reality. And if you would require ten to the minus four, with expectations that the Applicant would come in with a better number --

MEMBER WALLIS: I don't think you should have any expectations above the requirement, it is not in the requirement, I wouldn't expect anything.

MEMBER KRESS: We don't have any requirements --

CHAIRMAN APOSTOLAKIS: What would you include in the ten to the minus four?

MEMBER ROSEN: If it is only internal events, I might be able to live with that. But if it is -- it really ought to be, whatever number you pick, it ought to include all modes of operation and internal and external.

MEMBER BONACA: If you do that you go beyond whatever we have right here.

CHAIRMAN APOSTOLAKIS: There is another issue here. I don't know what the Commission means by enhanced safety. What exactly does that mean?

MR. KING: In the advanced reactor policy statement they talk about using passive systems, less reliance on operator action, those kinds of things, to achieve enhanced safety. They haven't quantified it, the means for achieving it.

CHAIRMAN APOSTOLAKIS: Because the thing that comes to my mind is, you know, you can talk about an individual reactor being safer than an individual existing reactor. But also you can talk about the fleet, and so far our goals, and subsidiary goals are determined in terms of a reactor review.

And it is different, it seems to me, if you have 103 units. When you have 103 units it is different from having, say, 500. Shouldn't the number of anticipated units play a role some place here?

MR. KING: Leading into my next slide.

CHAIRMAN APOSTOLAKIS: You see, I'm setting it up.

MEMBER KRESS: Yes, but the difference between 300 units and 100 units, and 500 units, is hardly discernible in the PRA space.

CHAIRMAN APOSTOLAKIS: No, because if you have 5 ten to the minus four, three, five, I think it was a spectrum. It is different if you multiply by three or four times. I think you are getting --

MEMBER KRESS: That is beyond the capability of --

CHAIRMAN APOSTOLAKIS: I understand --

MEMBER POWERS: Explain to me, I'm not very bright, I guess. If I am an individual and live 2700 feet --

CHAIRMAN APOSTOLAKIS: For individual risk it doesn't matter, you are right. For societal risk it does.

MEMBER KRESS: For both of the safety

goals you have, it doesn't matter, because they are individuals. But I really think, deep down inside, we have implied some societal goals. I think we worry about total events.

MR. KING: But it does matter. If you have ten plants on a site versus one plant on a site, your individual risk changes.

MEMBER KRESS: It certainly does.

MEMBER POWERS: It seems to me that by the time you got to the 500-plant fleet, you would have some subset of individuals that you exposed several times. And there I can see that you might do some multiplication.

But I don't think you ever do a multiplication by 103, or 500.

CHAIRMAN APOSTOLAKIS: No, no, this is if you want to get the societal issues. Which we don't.

MEMBER POWERS: I would be very careful about driving societal goals which is you get these peculiarities of one gram of plutonium --

CHAIRMAN APOSTOLAKIS: The other thing you have individual risk, I think it depends on how you phrase it, now I'm thinking out loud, which I know is dangerous.

But if the Commission's goal is for a

specific individual, then it doesn't matter. If the Commission's goal is any individual in the United States, then probably it does matter.

MEMBER POWERS: But I don't think you ever multiply by 103 or 500.

CHAIRMAN APOSTOLAKIS: They are relatively mutually exclusive. Not 100 because you have fewer sites. But within the sites you may have the problem Tom mentioned.

MEMBER POWERS: You may multiply by -- there may be a necessity to multiply by 10, or 2, or 3, or something like that, but never 100.

CHAIRMAN APOSTOLAKIS: I think you do. What if you have -- right now we have, what, sixty sites? So if I want to know the individual risk for any individual in the United States will die because of that, I have to multiply by 60, don't I? Because any one can happen, I have 60 opportunities. Any one can die.

If you say a given individual, then you don't multiply. But if you say any individual, you multiply.

MEMBER WALLIS: I think you're way off the point. The point is how are they going to explain whether or not the safety is being enhanced?

MR. KING: We can go back and talk about that. We don't have a good measure, quantitative measure, of the safety level of plants. The best we have is the IP --

CHAIRMAN APOSTOLAKIS: All I'm saying is that if you do that thinking, take into account the possibility of having to multiply by the number of sites, and so on, and see what you get.

I'm not saying that you have to, but this is something that I'm sure will come up.

MR. KING: In fact, if you look at the Commission's strategic plan, and you look at -- they have four performance goals, one is maintain safety, and those are the measures of how they are going to measure whether they are doing that or not.

Most of those are dependent upon the number of plants, total number of plants in the country, not on a site basis, they are on a nationwide basis. And that is an issue that has to be addressed.

MEMBER KRESS: I think in practical reality, though, the chances of us getting so many plants in this country to have to worry about that, is pretty small.

CHAIRMAN APOSTOLAKIS: So now we have a probability goal, based on a probabilistic argument, that the chances of having so many plants is low. Okay.

MR. KING: Maybe that is the answer, that this will happen.

MEMBER KRESS: It could be, it could be.

CHAIRMAN APOSTOLAKIS: By the way --

MR. KING: But I think we are obligated to point out the question.

CHAIRMAN APOSTOLAKIS: By the way, before you go on, did you tell the Committee why you are sitting there?

MR. KING: Yes, I did, you weren't here.

MEMBER KRESS: He hasn't told us why he is qualified.

CHAIRMAN APOSTOLAKIS: That is what I meant. Is he qualified? A consultant.

MR. KING: Automatic qualification, isn't it?

MEMBER WALLIS: Whether he is qualified or not it is his job.

CHAIRMAN APOSTOLAKIS: But you have not been elevated to the exalted level of advisor.

MR. KING: Not yet. Let me go back to your question.

MEMBER WALLIS: I think we will determine if you are qualified after we've reviewed your presentation.

MR. KING: I don't want the answer.

CHAIRMAN APOSTOLAKIS: Would that stop us?

MR. KING: We regulate, we decide on whether we need new regulations based upon certain quantitative measures that are laid out on the reg analysis guidelines that deal with core damage frequency, conditional containment failure probability, there is a cost-benefit test in there for certain things.

If you look at option three, we were looking at risk informing Part 50, there are quantitative measures in there. To me, that sort of represents the current level of safety, that is what we are striving to achieve, that is where we would add regulations if we feel we are not achieving that.

We have the revised reactor oversight process, which is taking quantitative measures with performance indicators to see how well we are achieving that.

When I talk about this first option, required current level of safety, I'm thinking of those types of measures being applied for future plants, as well as today's plants.

When I go to the second option I'm thinking of making those measures a little more stringent, and applying those, and seeing what kind of regulations and inspection program, oversight program --

MEMBER WALLIS: So what you are really saying is your current level of regulations, or enhanced level of regulations?

MR. KING: Yes, you can call it that.

MEMBER POWERS: Dr. Kress, if you are going to look upon this as an opportunity to codify CDF, are you going to look at it as an opportunity to include a requirement on ground contamination?

MEMBER KRESS: Yes, but my CDF, or my LRF whatever I come up with, will include ground contamination. But that is another issue.

MR. KING: The other two options that we're talking about, the third one we call an enhanced level of confidence, and that is keep the same goals, CDF and so forth, as you have today, but you would apply some additional testing requirements, some additional oversight, maybe some additional analysis to really have a much -- try to improve your confidence that those goals are going to be met, given the fact that these designs have less experience, and have probably larger uncertainties associated with them. The fourth one is one where you go to the industry and you say, hey, our policy statement expects you to achieve enhanced safety, tell us how you are going to do it.

Remember the old EPRI ALWR requirements document? Well, they came in and had a ten to the minus fifth CDF, and they had some severe accident features on the plants, and so forth.

That could be, to me, a viable option. Say, okay, we are going to keep CDF ten to the minus four, and so forth, but we expect you to do better, show us how you are going to do that.

CHAIRMAN APOSTOLAKIS: The issue came up when the new production reactor was designed, and there was a number of interpretations. And finally they said, well gee, maybe it is only one reactor, a production reactor. We want it to be safer, they said, than the light water reactors.

So the interpretation was, safer than the safest LWR. And then they realized how much it would cost them. They just said, well, maybe that is not the interpretation of enhanced safety that we should adopt.

So I think, you know, you can interpret

this in a number of ways.

MEMBER KRESS: This Committee, normally, doesn't like these loose, vague things like, we are going to require the current level of safety, but we really expect you to have a higher level of safety, and you tell us how you do that. It just leaves the thing so wishy washy, and vague, that this Committee normally doesn't like that sort of stuff.

If we were to write a letter and say don't do that, pick out some level and say, that is what we are going to require.

MR. KING: The things that could drive this decision, one way or the other, I call them key considerations. The first one is the issue I already talked about, additional reactors.

As I mentioned, if you look at the strategic plan, the performance measures under there are really dependent upon the total number of reactors nationwide.

You also have the issue of reactors per site. It is my understanding that what is being discussed for the early site permit applications that are expected next year are, all three that are expected will be written around existing sites, so they will be written to add new reactors to sites that already have reactors on them.

And they will be written to allow multiple new reactors on those sites, not just one new reactor. So the issue of reactors per site I think is one that has to be addressed.

Go read the safety goal policy and say, what does it apply to, is it written on a per-plant or per-site basis? Depending on what paragraph you read you can go one way or the other.

MEMBER KRESS: But clearly, to me, a LRF criteria, if you had one, like the prompt fatality safety goal surrogate LRF criteria, is a site criteria. It is the site that has to meet that.

MR. KING: If you read the safety goal policy in the paragraphs that talk about the one miles and the ten miles, it talks about people around the site, not people around the plant.

MEMBER KRESS: It is a site criteria. I don't think there is any doubt about it.

MEMBER ROSEN: So in principle, if you have two reactors on site, and they've used up all the work, you are saying that you couldn't put another one?

MEMBER KRESS: That is exactly what you should say.

MEMBER ROSEN: Or you could say maybe we would have to have some additional measures for the other two reactors.

MEMBER KRESS: That is exactly what you would say.

MR. KING: Or you could say the next one has to be much safer, so that it is basically a negligible risk, just like in reg guide 117, where we define some, you know, small and very small.

You know, it could go different ways. But I think to me a fundamental question to go to the Commission is, how do you interpret your safety goal policy, per-site, per-plant?

CHAIRMAN APOSTOLAKIS: Actually I think it was Commissioner Bradford that had that comment, that two can play that game, that this is the goal of the policy reactor, this is the goal of -- maybe we should review it.

MR. KING: He did it on core damage.

CHAIRMAN APOSTOLAKIS: Yes. But anyway, that's the kind of thing we have to revisit.

MR. KING: Another issue is the fourth item down. The fact that these are new plants, we don't have a lot of operating experience. And probably the largest uncertainties are in the area of severe accidents.

Would it make sense to require enhanced accident prevention to help compensate for those uncertainties in severe accident space, so that you have a much lower likelihood of ever getting to severe accidents and, therefore, the uncertainties associated with that don't have the prominence that they might for a --

CHAIRMAN APOSTOLAKIS: Do the designers agree that the terminology "severe accidents" applies?

MR. KING: Do they agree what, excuse me?

CHAIRMAN APOSTOLAKIS: "Severe accidents," the terminology --

MR. KING: Well, in discussions with the pebble bed folks, they do not use the term severe accidents. I use it because it is sort of in our lingo, and it really means something with substantial core damage.

It may not be a core melt in the case of the HTGR.

MEMBER KRESS: Another way to interpret to me, that bullet, maybe we ought to require a really good quantification of the uncertainties and confidence levels on our requirements. Might be another way to do that.

MEMBER POWERS: That's a surprise.

MR. KING: Yes. Confidence level certainly is an issue that, I think, is important in a number of these issues we are going to talk about, it is not just on this issue.

CHAIRMAN APOSTOLAKIS: But already, I mean, if you look at the goals it seems to me that we are saying that accident prevention is a thousand times more important than mitigation, because you are saying 10^{-4} -- can you really do that? That's more of a feasibility issue; you can put even more emphasis on that side. I don't know how high, but it's pretty high, you know? It seems to me it would be easier to do more on the other side to make sure that mitigation is better than 0.1.

MR. KING: I think we can do better. I mean, what is the right ratio--

CHAIRMAN APOSTOLAKIS: Well, in the sense of -- if there is such a thing as a severe accident, then we can contain it, find it, with the probability, the condition probability of better than .1. It is fairly more feasible than working the prevention side.

But this is clearly a defense in depth issue which means a matter of uncertainty.

MEMBER BONACA: Although by designing the passive features, you are enhancing prevention.

CHAIRMAN APOSTOLAKIS: Actually my biggest uncertainties are on that side. I mean, even the AP600 reported something like a few 10 to the minus 7 for core damage frequency. I think there are uncertainties there. I mean, I couldn't find them at the time, but if you put yourself light water reactor history 30 years ago, there are a lot of things have happened since then, that we could not imagine. So the 10 to the minus 7 number is more suspect in my mind --

MR. KING: You are raising an interesting argument in terms of should we consider what is the balance, should we put a ratio to somehow quantify the balance for prevention and mitigation?

MEMBER BONACA: You know, if I could, the safety goal policy I was thinking about, actually, if you think about additional reactors and remember, we talked about four or five hundred, really, you have a viability of the industry objective that goes beyond the safety goal policy. I mean, that is not adequate any more.

CHAIRMAN APOSTOLAKIS: Yes, that is the whole point.

MEMBER BONACA: It would be more of an

industry issue.

CHAIRMAN APOSTOLAKIS: That brings up an interesting point. You know, people have been complaining from the beginning -- you should never have goals in terms of rates, because you run into these issues at some point. Per-year, per-whatever.

It has worked very well for us because we haven't built any more plants, but now maybe it is time to reconsider.

MEMBER KRESS: I hope we don't get tied up on this balance issue, because our real goal is to ensure the risk is not an undue risk. Whether it's achieved by a really good design that stops it from occurring or maybe not so good a design but has an extremely good containment. I don't think we should get tied up on that.

I think we should be interested in the overall number, and you need to worry about the uncertainties.

MEMBER ROSEN: What happened to defense in depth?

MEMBER KRESS: It is coming up.

CHAIRMAN APOSTOLAKIS: There are two or three slides in the presentation --

MEMBER ROSEN: But if you are saying we

should just be worried about the final number --

MEMBER KRESS: You heard me say, yes, I'd worry about the uncertainty in the determination, and that ought to be a consideration in how you do it. But I really think that is the risk that you should be worried about.

MEMBER ROSEN: I agree.

MR. KING: And that is your ultimate measure. But I still, I would give a lot more weight to prevention than mitigation.

CHAIRMAN APOSTOLAKIS: We already do.

MR. KING: And we already do. But is that good enough, or do we want to go further? The only other thing I want to point out --

MEMBER POWERS: My point was, I wouldn't say well, I got ten to the minus 7, but we're going to stick a .01 containment on it, too. That's what I was arguing against, the other direction. I think if you got good enough at the prevention end, you shouldn't get tied up on this balance.

MR. KING: You could carry that to the extreme and say all you need is prevention, you don't need --

MEMBER KRESS: And that is what I'm saying, you very well could get by with that in regulatory specs. As long as the uncertainties are not killers.

MEMBER ROSEN: But we are talking about a new reactors where the uncertainties are going to be large.

MEMBER KRESS: I say we've got to give some estimate.

MR. KING: The other thing I want to point out on this slide is the bottom item, implications for future LWRs. Most of these key considerations, depending on how they -- yes, whatever the outcome is for non-light water reactors, I think is going to have a bearing on the future of light water reactors. So that has to be kept in mind when you go to the Commission with a recommendation.

Defense in depth, that is the second overarching issue. I think the Committee was right in its letter of last July, in saying that is an overarching issue, not a sub-issue under some of these other things.

Right now, we talk about defense in depth in a lot of places, but we really don't have a good definition of what it is. It is not mentioned in the regulations. We have the 1999 white paper on risk-informed performance-based regulation that has a definition, but it reads more like a goal.

MEMBER POWERS: In truth, it is mentioned in the regulations, 50.48, and appendix R, both mention explicitly defense in depth.

MR. KING: Okay; I'm going to look at those. I don't remember seeing that in there.

MEMBER POWERS: Those are all fire protection regulations. The basic principle is prevention, suppression, and mitigation of consequences. And if you are desperate to find a definition of defense in depth, that is not a bad one. If you are looking for this rationalist baloney about compensating for uncertainties that we can't quantify or even articulate, you know, you're in more desperate shape. But I don't want to prejudice you with that point of view. I'm totally open-minded on this subject.

(Laughter.)

CHAIRMAN APOSTOLAKIS: Of course, when they mention fire protection and suppression, you know there was some sort of uncertainty advanced in their minds, because they don't do that for all fires. For some of them, they say that they are so low probability -- you don't do it for every single fire --

MEMBER POWERS: I'm really struggling to remember any of that, in 5048, or --

CHAIRMAN APOSTOLAKIS: It may not say it, whether you do the evaluation.

MEMBER POWERS: But when you are talking about the way the analysis is going back, I mean, yes, it is true that the approach to defense in depth is borne of uncertainty. But they circumvent the need to quantify them because in the end they are saying, "What if I'm wrong about all the analyses, including my analyses for my uncertainties?"

CHAIRMAN APOSTOLAKIS: Now they were wrong in Appendix R, when they demanded that things be near the ceiling, 20 feet above -- but as long you have 20 feet separation horizontally, it was okay. And then there was a search that showed that if you had a fire there was a hot plume that drives the gases up, and then you have a hot gas layer. So whether you have twenty feet or thirty feet, it really doesn't matter; because all of them are immersed in the hot gas layer.

Nobody asked, "What if we're all wrong?" And they were. So you know, there are limitations to that rationalist approach, too. In the scenario approach, it came out. In the scenario approach they identified the hot gas layer, and they said, "Gee, the horizontal distances don't really matter." There are limitations to both approaches.

MEMBER POWERS: I think you see what the prejudice was in setting this up. Yes, they took a conventional wisdom at the time and argued about 20 feet based on the radiation argument and not on hot gas.

But, you see the defense in depth says, first of all, you prevent that fire from ever occurring. Second of all, if that fails, you try to detect and suppress that fire. Now, the 20 feet was in fact and implementation of mitigating consequences.

CHAIRMAN APOSTOLAKIS: Well, but it could be prevented, depends on what you are trying to prevent. But you are saying prevention refers to the fire itself. But if you say "I'm trying to prevent core damage, then failed is a --

MEMBER POWERS: It is preventing damage to safety-related equipment, was the objective in that 24th thing there. But I mean that is compounding a lot of what fails on top of each other before you get there.

CHAIRMAN APOSTOLAKIS: The point I'm making is that just as you can criticize the argument that you should quantify your uncertainties and be rationalist, you can criticize the argument, I will be asking myself, "What if I am wrong?" because we may not ask that question at some crucial point, because you don't know. You don't ask, "what if I'm wrong" every single step of the way.

So maybe theoretically you can quantify the uncertainties like what the press wants, but theoretically, also, you can ask you know, "What if I'm wrong." But in both cases there are holes. That is why it should be risk-informed.

MEMBER KRESS: I think we ought to move on.

MEMBER POWERS: My only point was to say that it's not -- in the regulations, I mean, it is true in the sense that they don't speak of defense in depth for the bulk of the regulations, but there is an explicit mention defense in depth in connection with fire protection. And it is not a half-bad definition of a structuralist view toward defense in depth.

MR. KING: I will go look at that. There have been people that have tried to define defense in depth. IAEA and INSC are two of the most prominent in my mind, where they defined five levels that include design elements, as well as programmatic elements in fairly multi-paged documents that issued, that put those descriptions out.

I saw three options in trying to go to the Commission on this issue. One is we could just continue or previous practice of doing case by case reviews, and making judgements that defense in depth is incorporated into the design before we license it.

You know, that is a potential for some inconsistency, and it certainly has a lack of transparency in how those decisions were made, or has a potential for a lack of transparency.

We can try to develop a description or a policy statement on defense in depth that the Commission could issue that could try and define what those elements are.

We could, maybe, view it as trying to implement the definition that is in the risk informed performance based white paper, which I view more as a goal. And it could have structural elements, rational elements, it could have quantification on it, it could have any level of detail you want.

MEMBER POWERS: It was the case by case process of this committee to conduct a fairly thorough investigation of what it thought about defense in depth, and why the ability to do quantitative risk assessment.

And the problem we saw was that defense in depth was being used to undermine the use of risk information in the regulatory process, because it was always trumped by eliminating defense in depth over the years.

And so I guess I would look, my suggestion to you is don't present that, just that case by case thing, but you might want to consider another option, which says that in those cases where, at a fairly high level in the system, and not in the areas where there is quantitative risk analysis is actually pretty good for evaluating the systems, and what not.

In other words, I think there is more to this case by case than just looking at each subsystem, and what not. Because that is denying that you have this capability to look at a plant in an overall sense.

And I don't think you want to do that at this point.

CHAIRMAN APOSTOLAKIS: This is the so-called pragmatic approach in our paper. And I thought, I'm a little surprised that you don't mention option 3 here, because those guys have done a lot of thinking about it. And they did try to implement, as I recall, this pragmatic approach.

Because, you know, in our paper, between rationalist and structuralist we figure that neither one is really perfect, and you need this combination that Dana just described.

And having defense in depth of the highest level, without even questioning it, is a good thing to do with international mitigation. So I would suggest that we look at the option 3 documents, because they have done thinking about this.

MR. KING: I've looked at the option 3 documents and the discussion in REG guide 1174, I think that philosophy could be imbedded in that second option, if that is the way we decide to go.

CHAIRMAN APOSTOLAKIS: That is right.

MEMBER KRESS: But it raises the question, and I'm not the right person to raise this question, actually the Chairman is the one that should raise the question, but I will encourage him to raise it.

You said defense in depth up here, and not defense in depth philosophy. And maybe that distinction that we tried desperately to draw in 1.174 ultimately failing miserably, but that may be the way to ask the question, rather than casting it as strictly defense in depth.

CHAIRMAN APOSTOLAKIS: I agree.

MEMBER KRESS: I'm the ultra-rationalist in the crowd, and in defense in depth. So let me make a comment.

I think if you go to the option 3 concept you're stuck in this quagmire of prevention and mitigation, along with, perhaps, looking at individual sequence contributions, and not letting any one of them be too much.

But I think that is a problem, and what I think defense in depth ought to be, in the rationalist sense is, let's presume we have good PR risk assessments with uncertainty, and have goals on risk, not goals, you have acceptance criteria on risk, that are appropriate for the whole range of regulatory objectives.

And defense in depth ought to be focused on how these goals, how this thing is met. Is it met by a single element of design, or is it met by redundant systems, and is it met by reliabilities that are highly uncertain, or --

I think you ought to think along those lines for defense in depth. And then, maybe, you can factor into that the uncertainties associated with each element of how it is achieved.

And then say, well, there is too much

uncertainty here, we have to do something.

MEMBER BONACA: But I think, though, you can combine disciplines with what Dana has been saying, by simply calling it defense in depth philosophy.

In other words, you are pointing out that you have to worry about conventional mitigation. At the same time you are saying look at the uncertainties.

MEMBER KRESS: I was arguing against defense in depth philosophy being prevention --

CHAIRMAN APOSTOLAKIS: Well, I think it would be useful to give guidance how to do what --

MEMBER ROSEN: For example, I disagree, I don't like the inside approach, I can tell you that. Because by trying to define what it is, it really weakens the philosophy itself, that has been implemented in so many different forms, so many different judgements and areas, that -- and now if I can implement it with insights from PRA, clearly, then I can have a better defense in depth.

MEMBER KRESS: I really think if you look at the white paper definition, it is pretty good, it doesn't say prevention and mitigation, it says some -- yes, it doesn't say multiple barriers, it is multiple compensation.

And, you know, I think you can build a defense in depth on that.

MR. KING: I don't even think it says multiple. I will read it. It says: Defense in depth is an element of NRC safety philosophy. It employs successive compensatory measures to prevent accidents and mitigate damage if an accident or naturally caused event occurred with a nuclear facility.

Defense in depth philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of the nuclear facility.

The net effect of incorporating defense in depth in the design, construction, maintenance, and operation is that the facility or system in question tends to be more tolerant of failures and external challenges.

That is it.

MEMBER KRESS: That is a pretty good definition. And it doesn't really say anything about the balance between preventive and mitigation.

MR. KING: To me it says that is the goal of defense in depth, I have no quarrel with that. But if I was the designer I'm not sure how that would help me, other than, you know, you want to make sure you don't put --

MEMBER POWERS: Explain to me why it wouldn't help you. I mean, it seems to me that it is pretty explicit, it is not going to be dependent on the single element. So that tells me that I can't be absolutely dependent on passive natural circulation to keep my core cool.

MEMBER KRESS: And does it also tell you you can't be absolutely dependent on the fuel pellet?

MR. KING: Yes.

MEMBER BONACA: Ideally I think the rationalist approach makes sense.

MEMBER KRESS: Frankly I don't think we are well enough in technology, PRA technology and uncertainty to really implement the --

MEMBER BONACA: That is exactly the problem.

MEMBER ROSEN: Well, even though I think PRA is near perfect now I would still say there is still the question of what we don't know, there is this incompleteness uncertainty. Which by its very nature says, if you don't know it, you don't know it.

So you don't know how to quantify it. So because of that, even though of the near perfection in some PRAs, you still have to --

CHAIRMAN APOSTOLAKIS: This committee --

MEMBER ROSEN: -- back those new --

CHAIRMAN APOSTOLAKIS: This committee asks Joe to tell him what she doesn't know.

(Laughter.)

MEMBER ROSEN: I was thinking that Joe would tell us. He would be the only one who could meet on non-negotiable demands.

CHAIRMAN APOSTOLAKIS: Perhaps we have exhausted the --

MR. KING: Let me just talk about this third option. The difference I see between the second option, that is one where you would specify certain structuralist elements in defense in depth.

And you can have some rationalist elements in there, as well. But the third option, to me, is strictly a process that would sort of be a way -- describe a way to treat uncertainties, if that is how you view defense in depth, it would not have any structuralist elements in it.

So that is the difference between the second and the third. The key factors that affect the recommendation on this, certainly the scope of defense in depth, what we've been talking about all along.

Structuralist versus rationalists, should it include things, programmatic type things like QA, and EQ trains.

CHAIRMAN APOSTOLAKIS: But the reason why you make this distinction is the uncertainty, because the uncertainties have got the performance of physical elements, are smaller in general, than the uncertainties regarding the problems.

So this is, really, saying -- I would rather see something physical that I can touch, as a barrier, than have somebody tell me, make sure -- because that is more uncertain.

MR. KING: That is why we make the distinction.

CHAIRMAN APOSTOLAKIS: Because we have faced that before with, you know, reduce the risks. So some people say, okay, we will have better programs to make sure that the transient fuel is not coming to the room. And people saying, gee, we are already supposed to have those, I don't believe that.

Then somebody else says, well, you have these two trains, why don't we erect a barrier between them? And everybody goes, yes. The uncertainty now went down, this is physical.

MR. KING: But the counter argument to

that is you can put a barrier in, but if that barrier is poorly designed, and poorly constructed, and poorly maintained, what good is it?

CHAIRMAN APOSTOLAKIS: But still I think that the main difference between these, where you say versus, I think, is the level of confidence that we have, that one will work versus the other.

MEMBER KRESS: Well, I think there is also a difference, there, and some things can be handled by PRA, and also deterministic analysis, where others can't. Like QA, inspection, passive, all those are not well suited for PRA.

So you maybe just say, well, we are going to require QA, just like we now do, we are going to, for safety systems, we are going to require training, we are going to require inspection, testing, all those things are not quantified, we just require them.

MR. KING: But don't call them defense in depth, you mean?

MEMBER KRESS: Well, I would call them defense in depth. I would tell them, I would --

MR. KING: There is probably a whole set of those things, you call them good engineering practices, or something.

MEMBER KRESS: Yes, maybe do that.

CHAIRMAN APOSTOLAKIS: I think the latest insert to defense in depth -- good engineering practices is part of defense in depth. That is why I think it is important to say philosophy, rather than just defense in depth.

MEMBER KRESS: And I think the issue of redundancy and diversity is definitely defense in depth. And I would say there is some things where you ought to require redundancy.

Like, for instance, I think there is key safety functions that are reactor design independent. Like being able to scram the reactor.

MR. KING: Two independent shut down systems?

MEMBER KRESS: Two independent shut down systems.

MR. KING: I don't care what your PRA says, it --

MEMBER KRESS: -- like being able to have long term decay heat removal. You know, I think there are things like that that you can just say, redundancy and diversity is defense in depth, and we will require it.

Now, that begs the question of how reliable each one should be, and that is another issue.

MR. KING: You are arguing for putting together some sort of high level definition of defense in depth that says, these are the features that future plant has to have?

MEMBER KRESS: Yes, that would be part of my definition.

MEMBER BONACA: By the way, the first bullet on programmatic, it is -- I mean, try to replace an area, talk about the actuary. And that really has a foundation into a lot of operating experience.

MR. KING: If we do go and try and define defense in depth what is the approach we should take? Realize reactor oversight process cornerstones are one structure you could follow, if you want to try and write something down.

That brings in, potentially, things like security, security an element of defense in depth.

MEMBER ROSEN: It should be. Challenges from internal and external threats to the safety systems in the plant.

MR. KING: If you read the definition in the white paper it talks about external threats, that is true.

If you would write, try and develop such a definition of defense in depth, it could form the foundation for future licensing framework, the thing that Mary and her folks are going to be working on, might provide a nice skeleton, a structure from which to step forward and try and write that.

It could also be useful in other areas, like reg analysis guidelines, which don't say much about defense in depth. And you factor that into your reg analysis decisions.

Again, there is implications for future light water reactors, and there is the issue of coordination with non-reactor activities. You know, NMSS struggles with the issue of defense in depth, too, and you have to consider, do we want to write something that is strictly for reactors, or do we want to write something broader for the Agency?

MEMBER KRESS: I don't think we have anything else on the agenda, so we can -- I think this is an important issue, so we shouldn't give it short shrift.

CHAIRMAN APOSTOLAKIS: So you will not complain if we stay here until 7 o'clock? Tom, you have an open house here.

MR. KING: I will stop when you want me to stop.

CHAIRMAN APOSTOLAKIS: Let's take a break now for 11 minutes.

(Whereupon, the above-entitled matter went off the record at 3:45 p.m. and went back on the record at 3:57 p.m.)

CHAIRMAN APOSTOLAKIS: Let's go on, Tom.

MR. KING: We will move on to the third issue, which is called international codes and standards. To me the real issue here is, when you look at the future of design efforts, most of those are international efforts, in terms of consortium of organizations.

And the question is, and they are using international codes and standards in a number of them, in their design work. Should we actively get involved in looking at endorsing and using international codes and standards?

MEMBER KRESS: Things like ISO and --

MR. KING: Yes, those kinds of things.

MEMBER WALLIS: I was thinking if you look at current U.S. policy, --

MR. KING: Current U.S. policy is we should, yes.

CHAIRMAN APOSTOLAKIS: We should --

MEMBER POWERS: There is a lot of pressure to go to ISO2000.

MR. KING: And trust me, if you read NRC management directive 6.5, which is titled: NRC Participation in the Development and Use of Consensus Standards, it says that we should, as a first step, see if there are consensus standards out there were used before we develop our own standard.

And it also says it makes no distinction between domestic and international standards. So to me the management directive is pretty clear, we ought to be doing that.

It takes resources to do that, it takes a commitment --

CHAIRMAN APOSTOLAKIS: There is a difference, though, between what you say now, and what you said in the previous slide. Standards, okay, you can look at them, it is international, maybe carry some weight.

But you say reviewing those existing codes and standards were never practical. And you are going to go now and get the various codes that the European Union has developed, and France, and Germany, separately, and try to, without them coming to you?

Because typically in the United States

that is what happens, right? The Licensee comes to us first proposes something.

MR. KING: That is option one. We could sit back and when an application comes in, or pre-application, we can see, okay, what international standards are they using, and then we get involved in reviewing them, and endorsing them, if it makes sense to do that.

That is one way to do it.

CHAIRMAN APOSTOLAKIS: -- major undertaking to do that? I mean, reviewing the thermohydraulic code is a --

MR. KING: No, I'm not talking about thermohydraulic codes, I'm talking about things like the ASME Board, and pressure vessel code, ISO9000, design codes and safety standards, basically is what I'm talking about, not analytical codes.

CHAIRMAN APOSTOLAKIS: That makes more sense.

MR. KING: Again, the first option is just sit back and wait. Somebody comes in and says, we are using this, we will look at it.

MEMBER RANSOM: I have a question. I never really heard much in nuclear safety with the concept of fail safe, fail operational type design philosophies.

Is there a reason for that, or is that just inherent in what people do? These are concepts that were used in the aerospace program, and they were very successful. It enters into the basic design.

CHAIRMAN APOSTOLAKIS: We are trying to prevent fail dangers, we don't care about fail safe. That is the utility's job.

MEMBER RANSOM: That is an interesting concept.

CHAIRMAN APOSTOLAKIS: Because that creates unnecessary shutdowns.

MEMBER RANSOM: Because, for example, if you put a containment on something, there is nowhere for it to fail safe. It fails -- so maybe a containment isn't good for that.

MR. KING: It could fail open, you know, that is not fail safe. You know, your isolation valves don't close, it doesn't fail like a bomb, it just has a hole in it.

CHAIRMAN APOSTOLAKIS: And we really worry about that.

MEMBER RANSOM: But some of these recent designs, like the gravity driven cooling systems, you know, basically if they fail, they simply dump more water in the core. So that is kind of a fail safe concept.

And it can be carried further. But I was just curious.

CHAIRMAN APOSTOLAKIS: I think it is the terminology that is not being used, but the concept is. But the emphasis is always on dangerous failures, by the nature of the agency. We are not really designed the articles for operation, we make sure they are safe. It is somebody else's job to make sure that there are --

MEMBER ROSEN: The people who do design it can run it in a safe fashion.

MEMBER RANSOM: However, by specifying defense in depth, you know, in effect you are telling people how they have to be designed.

CHAIRMAN APOSTOLAKIS: To be safe.

MEMBER RANSOM: Not specifically, but at least as far as the overarching concepts are concerned, in order to be safe or licensed.

MR. KING: It should have certain features in it, for example. Maybe I can talk about the options.

Like I said, the first one is we sit and wait, we review what we are asked to review. The second one is to go out and try to take a look at what is out there, in terms of existing codes and standards, and anticipate maybe this is something we can use.

Now, when it says whenever practical, I really had non-LWRs in mind, in the sense that you take HTGRs, we don't have reg guides, or design standards for HTGRs. But perhaps maybe the Germans, or the Japanese, or somebody do.

Maybe it would make sense to go target those areas where we don't really have an infrastructure, and go do that. The same thing on the third option, which is more than review what is out there, we would actually participate in the development of what is needed.

Because there are development efforts under way in some of these areas. Should we jump in and participate in those?

And then the fourth one is, going even further, and that is trying to harmonize with other regulatory bodies in terms of what the requirements ought to be, at least the standards that should be used.

So that is sort of the range of options. As I said, the management directive 6.5 is pretty clear that we ought to try and use international, or domestic and international standards wherever we can.

As I said, that does, if we are going to do that, that does require resources, and commitment of some stability. You can't just jump in and out of that kind of thing.

If we did that it might have some public confidence type aspects to it. We could say, hey, we are using international standards, you know, all the other major countries are using the same standards.

To me that might have some influence on public confidence. And I think if we did that it could be useful, an efficient and effective way of beefing up our infrastructure where we don't have it, particularly in these non-LWRs.

So those are the considerations for dealing with that.

MEMBER WALLIS: Well, if you look at our reaction to environmental standards world-wide, or something, we always seem to say we do whatever we like. And I think that is what we do here.

If the standards, internationally, get too strict, we will withdraw.

MR. KING: That is always a possibility. But when I read the management directive it is pretty clear to me that we are to get engaged in that kind of thing.

MEMBER KRESS: Well, I think you are likely to come in to some foreign reactors with designed to certain code and standards, and you will have to know what those are, to see whether they are acceptable to you. So I think it is more --

MEMBER WALLIS: Well, you don't mean something like a CDF or --

MEMBER KRESS: No, that is --

CHAIRMAN APOSTOLAKIS: Well, in fact, this morning, because now from ACL, suggested that maybe since the ACR 700 is being reviewed by the Canadian authorities, and possibly by the UK authorities, that the NRC may want to take advantage of that, and not repeat the work.

So some of the foreign designers are, in fact, urging us to start doing that. So hopefully we will accelerate the process.

MEMBER KRESS: Yes, and it might even go further, for example, if you look at the UK acceptance criteria for things like safety, they are probably different than ours. But you might be able to look at them and say, okay, if they meet these, they very well meet ours also, or something like this.

So it would be kind of -- for that case, it might be well worth your while to check and see what they are doing.

MEMBER ROSEN: Well, as a minimum, if they are licensing, for instance, the ACR700 in UK, one could clearly read the British licensing documents and see whether or not they go to reducing the workload on the Staff, simply by saying, okay, these are reasonable analysis and we will accept them, use them in part for the basis of our work.

MEMBER KRESS: So I think we are supporting some sort of activity.

MR. KING: Again, the paper in December is not going to go to the Commission and say, well, we ought to work on these ten standards, or whatever. It is more to get the direction to then go explore, work out the deals.

Fourth issue, events, what we call event selection.

MEMBER KRESS: Design basis events?

MR. KING: And events for emergency planning purposes. The MHTGR 10, 15 years, came in with a scheme that defined events using some probabilistic criteria, and then depending on the event category there were acceptance criteria.

Those related acceptance criteria that went along with it. The pebble bed folks pretty much picked up that same concept, and I understand that GTMHR is doing the same thing.

It is pretty much a probabilistic-based approach. We had looked at that, back in the MHTGR days, and went to the Commission with the position, a recommendation on how to deal with that.

There was a SECY paper issued back in 1993, '93 or '92, and the Commission issued an SRM. And the Commission basically back then said, let's use a deterministic approach for the MHTGR, but supplement it with PRA insights.

Which, to me, basically said let's pick our design basis accidents deterministically, then look at the PRA and see if there is anything else we want to add in there, because the PRA --

MEMBER POWERS: Why do you have to have a design basis accident?

MR. KING: Why do you have to have one?

MEMBER POWERS: Yes.

MR. KING: What are you going to design the plant for? At some point --

MEMBER POWERS: I'm not going to design a plant, are you?

MR. KING: I'm not going to design the plant, but somebody is going to design the plant.

MEMBER POWERS: That is fine, let them design it. What you are concerned about is what the risk is to the public. You are not concerned, at all, about accidents that, by design, have extraordinarily low probabilities.

You are worried about the accidents that will occur, that have a reasonable probability. You may find those out with a PRA approach.

MR. KING: How do you decide, as a regulator, where you draw the line? I want them to consider these, and I don't want them to consider those? At some points you are going to have to --

MEMBER POWERS: I want them to consider anything that can happen.

MR. KING: Anything that can happen, but?

CHAIRMAN APOSTOLAKIS: Let me phrase it in a different way, because there is a disagreement here.

After I do my PRA, and I do everything Dana wants, then I say, a design that results in this risk to the public health and safety is acceptable. It seems to me the next charge to us is to make sure that the review process of the application is efficient.

So after I do the PRA and identify these things, I'm saying now, now designer, if you do this, and this, and this, and that, then we will review these elements, and then you have met the goals.

In other words, the design basis envelope here will be really a means to facilitate the review, which is what you said, what do you design for? I think it is the same question put in a different way.

But it will not be a deterministic approach where you define the envelope, and then you postulate that anything else that may happen is covered by the envelope, because you are doing your PRA first.

You identify the sequences, and so on, and then after everything is settled, you say, now I need to define a number of events that I will call design basis. So that when they come to me I will tell my people what to look for.

MEMBER ROSEN: What you do is you tell the designer that below a certain frequency we are going to have this kind of treatment for your systems, and above this frequency there will be another kind. Or maybe there will be three, I'm not sure.

And then he goes and designs the plant and does the calculation, I have this design, I have too much stuff in the high frequency category. I want to do something a little different so that I can get less regulatory oversight, so I'm going to put some more barriers here, or some more robustness here or there.

So it is -- the PRA becomes a design tool, it could be used in lots, and lots of different ways. And then the regulator comes, when he is all done, then the regulator comes in and does exactly what he told the designer ahead of time.

He verifies, of course, that the PRA is adequate and correct, and then he applies a regulatory controls to the things that, as Dana said, can happen and have consequences. In other words, have frequency that are reasonably high, and have some consequences.

By the way, that is risk --

MEMBER KRESS: Let's look at this in another point of view. You are allowed to have these reactors come to you, already with a conceptual, pretty good conceptual design. And they all have a good idea of what accidents are likely to happen, events, and how they can go.

And what they are going to say to you is, hey, I want to consider these in my design basis, pick some of them and say, we are going to try to conform to your chapter 15 with these.

And you are going to be faced, then, with saying are those the right ones for the tubes, and how are we going to choose them, and how are we going to decide whether those are the right design basis accidents?

And they might have picked them on some basis of frequency like the PBMR did. And I think your only option here is to start and say, well our purpose is just what Dana said, we want to have a design that has acceptable risk, and has maybe some acceptable depths in terms of whatever that means.

But we would like to have design basis accident because it gives them something to design to, and determines their design licensing basis. And it is like George said, it facilitates the review for any future plant, and things of that nature.

So what I would suggest you have to do is you say, all right, we will, tentatively, we will let you use those that you choose for the design basis events. But after you give me a design that is based on those, you are also going to give me a PRA.

And you are going to show me that you meet my risk acceptance criteria. But you have to have these risk acceptance criteria, and they can't just be CDF and --

MEMBER WALLIS: I don't agree with that.

MEMBER KRESS: And then you tell them, if you don't meet my criteria, you have to include something else in the design basis.

MEMBER WALLIS: I don't agree with that for this reason. It is a perfectly logical way to go until you start saying, now those are your design basis events. To me that says that is basing a whole, something foreign onto this analysis.

You've got an analysis that ranks all the sequences, and all the events. And now to say, well these are design basis doesn't make any sense. It is anachronistic, it is going back to the way that we used to do things, and trying to paste it on a new --

CHAIRMAN APOSTOLAKIS: No, that is not the way we used to do things. We selected the design basis events first, and that makes a big difference, that makes a huge difference.

Let's not forget that there will be a number of reactors, we hope, applications of a particular type. Let's say the ACR700. After you have gone through your PRA, and you have reviewed it exhaustively with the Staff and so on, why is it inconceivable that the licensee and the agency say, in order now to achieve these goals that you and Dana have identified, make sure that the risk is so low, and so on, you have to do A, B, C, D.

And the benefit of that is that you are removing the burden of confirming the PRA and working in uncertainty, from the lower level engineers who run the reviews.

Otherwise you are going to have interminable discussions regarding the validity of the PRA, what do we do here and there. That will be done, once and for all, by senior staff, and the Applicant, and then they agree that this will be the design envelope for this plant.

And if you do these deterministic things you have met the probabilistic goal.

MEMBER BONACA: At some point there will have to be an agreement between the regulator and the designer of which transients, or whatever are going to be considered, and -- because it is very unlikely that all the consequences are -- or whatever.

CHAIRMAN APOSTOLAKIS: It facilitates the review.

MEMBER ROSEN: Well, if you put all these sequences and events down, and --

MEMBER BONACA: I'm not going to call it design basis, so I --

MEMBER ROSEN: -- down and, say, CFR order, or most -- but at some point, I agree, that someone draws a line that says, above these you have to consider them, below these you don't.

But there is an alternative to even that. If someone draws a line and says above this you have to apply all of the standards in 10CFR, whatever, below this line you can do it selectively, or you can do it in some reduced or graded manner.

So at no point in that discussion do you say design basis.

MEMBER KRESS: You guys are presupposing a whole new regulatory system. I think these things are going to have fit into what we have. And what we have is design basis events, we have conservative specifications on how you meet them.

We have figures of merit they have to meet. And I think they are going to have to fit into that.

MEMBER ROSEN: You are right, I'm presupposing a different way of doing business.

MEMBER KRESS: Okay, but I think when we worry about recent certifications that are going to come in, we are going to have to fit them into what we have.

And we are going to have to choose design basis events, and they are going to have to be calculated with thermohydraulic codes, and figures of merit --

MEMBER ROSEN: Now, I ask Tom, is that correct? Is it true that we will have to pick design basis events? Because if so there is no point discussing this.

MR. KING: The options I'm talking about are, do we want to revisit the Commission decision of ten years ago that said for MHTGR pick the events that the plant is to be designed for in a deterministic basis, look at the PRA and see if you missed anything, and fill in the gaps.

What I'm suggesting is, going back to the Commission, and if we agree that doesn't make sense any more, because we are more of a risk informed agency, maybe we want to start with the PRA, and define some probabilistic criteria, somehow we have to figure out how we are going to take that PRA and give guidance to a designer so that he can go do the design.

MEMBER ROSEN: I think what you said is exactly right. You have three options up there. The first one is the way we are doing business now in the

-- we are using deterministic approach supplemented by PRA, that is what south Texas did in the risk informed world.

The third option is what I think I'm arguing for, and I want to speak to Dana, but I think that is what I hear from him, too. Is to use a probabilistic approach, and you supplement it with engineering judgement.

CHAIRMAN APOSTOLAKIS: But at some point you have to define some deterministic criteria that will guarantee that the probabilistic --

MEMBER POWERS: I think we are not -- from a point of view I think we are very consistent. What you are talking about is the next step. It is having done the PRA, and said gee, it looks like you are getting very sensitive station blackout.

So when you build your plant you want to make sure that your diesel generators are in good shape, okay? And whatever it takes to do that. And I don't think I have any objections to that.

CHAIRMAN APOSTOLAKIS: And the form of the design basis accident doesn't have to be the same as it is now, because I think that bothers some people. We can formulate them in a different way.

MEMBER POWERS: The fundamental problem I have is that we spent an entire day yesterday talking about behavior under design basis accident conditions. And those accidents pose very, very little risk.

CHAIRMAN APOSTOLAKIS: I agree.

MEMBER POWERS: And we are spending a huge amount of money on it.

CHAIRMAN APOSTOLAKIS: I want to sensitize the committee to the issue of doing something in relatively large scale. And an analogy is -- the most successful one is, this thing that the Supreme Court has asked police officers to read the rights to a suspect.

The objective is to make sure that the guy knows his rights. And that is all that the Supreme Court says. If you don't read his rights the guy is free, even if he is guilty.

That is a deterministic criteria. Because the police cannot go and say, but he is a lawyer, he knows his rights. The Court says, no, you didn't read them, he walks.

Why do they say that? Because you apply this principle to a country of 260 million. You can't rely on every police officer, everywhere, to make a judgement whether the guy knows his rights.

So they impose a strict deterministic

criteria. And it seems to me that if you are planning to license more than one reactor, you have to have deterministic criteria. You can't expect all these people who get involved in the licensing process to make judgements whether the probabilities are low, and so on.

That judgment has to be made once and for all by a select group of people that says, yes, for this type of reactor if you meet these criteria, then the risks are low.

MEMBER ROSEN: We are not as far apart as we may have seemed. Because I'm arguing exactly for that, using the PRA approach -- use the PRA approach, have a select group of people in the licensing process make that determination, codify it in a way that everybody in the design group, and the maintenance group, and the construction group can understand it.

You don't -- in South Texas they didn't give out the PRA to everybody and say, go out there and get your special treatment. The derivative of the PRA is something that they use every day.

CHAIRMAN APOSTOLAKIS: So I think we are almost in agreement. The more we talk, the more we agree.

MEMBER BONACA: I had noticed, about ten

minutes ago, that you guys were all in agreement.

MEMBER KRESS: We are all in agreement except one of us.

MEMBER ROSEN: And you know who that one is.

CHAIRMAN APOSTOLAKIS: Let's move on.

MR. KING: All right. If you take the probabilistic approach it can apply to more than event selection. It can apply to classification equipment, it can replace the single failure criteria. These are things that are being looked at under risk informing option 3, to various aspects.

And it would seem reasonable to look at them under a risk informed approach to non-light water reactor future plant licensing. So those are caught up in this issue, as well.

Certainly the more you use PRA you get into issues of PRA quality, completeness, document control, perhaps bringing the PRA into the licensing basis. And you have to deal with issues of level of confidence.

MEMBER POWERS: That level of confidence is the one that continues to irk. And I mean maybe diverting us from the main topic here. But we continue to see people come in and present probabilistic assessments for all point values.

And we absolutely cannot judge level of confidence. I have not seen a PRA yet that we can judge level of confidence on.

Now, I'm not even sure how you go about doing it.

MR. KING: Because it is incomplete, you mean?

MEMBER POWERS: No, let's -- if we stipulate that whatever PRA they have for operational events is complete, just for the sake of argument, we don't ever get anything that allows us to judge the level of confidence on that.

People come in and say we've gone through the peer review process and so it is good. I mean, it is a good quality. But they give you a number, and you just have no idea what to do with that number, because you don't know whether it is a mean, a median, or an accident, or what.

Because there is nothing to judge level of confidence from.

MEMBER ROSEN: But you can force that. If you just tell someone to go back home and come back with that, they will. They are getting away with not telling you that number. But if forced they can give you the number, they just don't want to.

Because, typically, it is going to be in order of magnitude around the value they gave you.

MEMBER POWERS: On this pressurized thermal shock we were beating the people over the head over what they meant by their distribution. It turns out computer code calculates out exactly what I was after. All they had to do is write it down.

MEMBER ROSEN: That is right, and George knows that, and I know that. The only question is we haven't forced them to give you that. It is embarrassing, because when you come back and I tell you that the numbers weren't even -- I have to tell you it is really 5 --

(Off the record discussion.)

MEMBER ROSEN: If somebody tells me less than that I would be interest in having a look at how they got --

MEMBER SIEBER: I think your confidence in the answer for an advanced reactor -- so it is going to be hard to apply the principles where you rely on the PRA first, without putting some deterministic overlay on top.

MEMBER ROSEN: You are absolutely right. Which means that once you have that understanding, then you have to say, okay, where does the defense in depth go to help you with --

MEMBER SIEBER: If you don't have a good PRA then you are picking up the deterministic criteria that is pretty arbitrary, too.

MEMBER WALLIS: It is going to be arbitrary --

MEMBER SIEBER: Just because it is a solid naught, because it is a number doesn't mean that it is better. On the other hand, you know, you could come up with a -- because the numbers are really great from a PRA standpoint, and you can conclude you don't need a containment.

So there is an element in defense in depth that disappears. It is not engineering judgement --

MEMBER WALLIS: Not if the structuralists have their way.

MEMBER SIEBER: Of course you put the containment there.

CHAIRMAN APOSTOLAKIS: I even asked that question at the PSA conference this week. A fellow stood up and asked the NRC folks present, on what basis did you decide to force the AP600 design when the PRA results show that we don't need it? And the answer was defense in depth.

MEMBER POWERS: But that was an erroneous answer.

CHAIRMAN APOSTOLAKIS: Why?

MEMBER POWERS: It was a question of confidence.

CHAIRMAN APOSTOLAKIS: -- defense in depth? I asked myself, what if I'm wrong?

MR. KING: Fifth issue, source term. Back when we were looking at the MHTGR Dave proposed using a scenario specific source term, not taking a source term representative of a core melt, or a sever core damage accident, and using that for the purposes of citing under chapter 15 analysis.

The Commission accepted that position back in their SRN of July of '93, basically said, that is okay provided we have sufficient knowledge of the behavior of the plant, and the behavior of the fuel.

Which implied that there had to be a lot of work to make sure we had the confidence to be able to do that. That is different than what Fort St. Veraine did. Fort St. Veraine basically assumed an uncontrolled core heat-up, and had, other than the timing, had releases similar to the TIB source term.

Fort St. Veraine didn't have passive heat removal, and so forth, it needed active systems.

Really we are revisiting this issue with the Commission to see if they still accept that position.

To me the fundamental policy shift is one of -- it really departs from past practice where we've used source term representative of severe core damage accident for licensing, including Fort St. Veraine.

And maybe that is -- should that be considered an element of defense in depth? You will assume severe core damage for licensing purposes, for citing purposes. That is a question, not a conclusion.

Certainly puts more burden on understanding plant behavior. Follow some extensive research to have the confidence, and maybe some extensive monitoring of the plant, and the fuel fabrication process over the life of the plant, to make sure you are getting the quality you need.

So it has some hooks in it, it is not a quick and easy solution to do that.

MEMBER KRESS: I think this question is tied to the previous one about event selection. And in the current system all we do is we select these design basis events, and specify how they are to be dealt with, to some extent.

And one of the ways that they are dealt

with is the source term. You assume there is a pretty severe source term. And the reason we do that, in my mind, is that by doing it you are putting enough conservatism in your calculations, for these design basis events, that you render the plant at acceptable risk level.

And the only way you know that it is rendered an acceptable risk level is you go back and do a PRA with scenario-specific source terms. So we use, we actually should be using both, in my mind.

If you are going to go to the design basis accident concept, I don't care what you use for the source term, as long as what you use renders an acceptable risk level, and acceptable confidence level.

So, you know, you could use a scenario specific ones, or you could use a bounding one, and might treat them differently in terms of how you specify the design basis.

In my mind the way we've just selected design basis events, with the single failure criteria, the specified source terms, and with the figures of merit that they have to meet, like peak clad temperature, and this sort of -- not all those have source terms in them.

In fact the source term only shows up in few of them like that. I guess it shows up in the LOCA, reactivity and source events, and it shows up in containment.

MR. KING: You know what we have now for light water reactor, we have a plant that has ECCS systems to prevent the core from melting, yet we assume the core melts anyway, when we calculate containment performance. So we have conservatism on top of conservatism.

MEMBER KRESS: I think my point is that in order to arrive at bounding source term you have to kind of know what scenario specific source terms are in a given reactor design. And the two are tied together, you can't just say option one is bounding, and option two is scenario specific. You have to have both of them, and you use one -- it is all right to use the bounding one if you use the scenario specific ones to decide what your bounding one is.

And the final result is you have to meet some sort of risk acceptance criteria at a particular confidence level.

MEMBER SIEBER: The TIB source term is not necessarily bound --

MEMBER KRESS: Well, bounding in the sense that if you use it along with a specified design basis accidents, you render the plant to an acceptable confidence.

MEMBER SIEBER: That is right, but it relies on --

MEMBER KRESS: So it is bounding, in essence.

MEMBER SIEBER: -- water and partitioning, and all that.

MEMBER KRESS: That is not all you can get out. It serves the purpose that you want.

MEMBER SIEBER: For light water reactors.

MEMBER KRESS: And I think that is --

MEMBER SIEBER: On the other hand, a different kind of fuel is going to have a different source term, it is usually bigger, right?

MR. KING: This issue will certainly drive the containment issue, depending on which way this goes, it is going to drive the containment issue. That is why the designers are interested in it.

They would like to not have to impose this source term representative of a severe core damage because they say our plant isn't going to have severe core damage, or it is such a low probability, we don't need to worry about it. And they want us to buy into that.

MEMBER KRESS: Well, my basic philosophy would be, if you are going to use a design basis concept, and a source term along with it, choose the one that lets you have an acceptable risk. You have to do both, risk and the -- and, you know, it may very well be that an accident involving air ingression in a PBMR leaves you a huge source term, but it is risk that might still be acceptable if you use a real small source term in your design, and your design accommodates in terms of frequency, for example.

But it doesn't have to use that source term.

MEMBER SIEBER: It doesn't have to. But if you are engineering *** there isn't all that data out there, the correlation --

MR. KING: To me it gets back to it is a fundamental question of defense in depth. Does the Commission want to maintain that policy of saying I don't care what your design --

MEMBER SIEBER: That is where it comes down to.

MEMBER BONACA: And the question is, do you allow the PRA to derive the elements of defense in depth?

CHAIRMAN APOSTOLAKIS: But it's not only whether it's a handicap to design, but I mean what does it do for us?

MEMBER KRESS: No, but I don't think --

MEMBER BONACA: But look at the elements of defense in depth, the cumulative examples --

CHAIRMAN APOSTOLAKIS: Only because it was interpreted as a single hardware --

MEMBER BONACA: The others, if you look at those, still, clearly they suggest that you can have separation, you will have no diversity. So to the degree to which you integrate, you know, some prescription of defense in depth based on the size of your PRA, I think that defense-in-depth ultimately is going to be what you will get.

MR. KING: What you're really arguing about is that considering a large source term is an evolution, and that that is not the right way to look at it.

MEMBER POWERS: I think that, I mean, I don't agree with the Committee at this level, but I think that the structuralist point of view used the analyses that you've done, the flow assessments you've done. I want to know what happens in this -- what is contained in the engineering safety systems that you've got if you put a large source term back in that containment.

MEMBER KRESS: What do you mean by "large"?

MEMBER POWERS: That's a big one, yes. I don't have difficulty with the approach that they've taken in the development of NUREG 1465, which is not different in kind from what they did with TID 1434. They've said, okay, here's the kind of source term that you have to deal with. They use those particular source terms because they're not going to be applicable to all reactors. For instance, a pebble bed modulated reactor, I think, would probably have a little different-looking source term than I would put in the -- I like the idea of having both gaseous and particulate material and debris in there.

I don't know what the exact mix is going to be, but you have something that was never anticipated that dumps a whole lot of reactivity into the containment.

MEMBER KRESS: I don't think I'm disagreeing with you, but my point is, that when we did 1465, what we actually did was we took a set of scenario-specific accidents and calculated releases, and then we kind of took a conservative part of those and said, "Just sit."

I think you could do the same think with the pebble bed modular reactor. If you had enough database for the fuel, and you a description of the accidents it could go through, you could CRA-specific accidents and say, "Here are the source terms I get out of that." Now one of the accident sequences might be an air-ingression accident. But then you've got to use judgment, like we did in 1465. Is that an accident sequence we really ought to have to deal with in terms of the specification of the source term?

MR. KING: But all the accident scenarios that went into making 1465 were core melt scenarios.

MEMBER KRESS: Yes, they were core melt, but they weren't coolant core melt.

MEMBER POWERS: I think he's hinting at the problem I have. You had the advantage for the current generation of reactors and you could get into similar accidents. The people developing these gas-cooled reactors come in and say it's not possible. And they throw up a lot of reasons, none of which do I swallow, for why they can't. And yet, I'm doing this because I'm saying, one of these days, nature will prove these guys wrong.

I'm not sure that I am happy with them going through their accident sequences and doing what we did for 1465 because they'll come up with minuscule source terms and they'll sum them up and take a conservative limit on a minuscule source term, and it's still a minuscule source term. Yet what I'm worried about is that all those analyses are wrong.

I think what we did was just fine for existing reactors, but I don't think that is the prescription I would put on everybody else. I would say give me a decent-size source term that has a mix of particular gaseous materials and show me how you contain it. And I would do that, the guy came back and said, "Here, I've done this mechanistically, I've looked at all my reactor accents. I get a pretty healthy source term on some of them, and it's a mix, and I like using that."

He goes through the analysis much like AP 600 * did; they didn't think their core was going to melt either. They went ahead and came up with a mix. They adjusted their ways from 1465 and went ahead and did the analysis, and I think we were happy with that. We didn't like the numbers they came up with, but clearly you were happy with that.

If the guy did that, I think I would be content. I wouldn't say, "Oh, well, you didn't get 50% of the iodine out; I think you're going to fail." That's not terribly important to me. It's more important to me that the mix is substantial.

CHAIRMAN APOSTOLAKIS: I think the equivalent of what you are saying is, as you said, the mix. At that level, you don't know what your volume is, going to a high temperature. Just to protect myself --

MEMBER POWERS: I give PRA where PRA is due. There's no strong numbers up at this level; I freely admit that someday there will be, but it's not there right now.

CHAIRMAN APOSTOLAKIS: I think that's an important point, and if you put it in that language, you've always talked about confidence language. So what Dana is saying when it comes to the source term, forget about the mean and the median. I don't want you to go with the 90^{th} percentile; some sort of a mix of the very bad case with the standard cases. So you can always play something --

MEMBER WALLIS: You'll be in real conflict with the designers, because they're going to come back and say, "Our source term is minute. That's the whole idea of this wonderful reactor is it has a very small source term. That's why it's so safe and good for the public."

MEMBER POWERS: That's what they're going to say, exactly.

MEMBER WALLIS: They're going to say that.

MEMBER POWERS: And that's just not good enough for me.

CHAIRMAN APOSTOLAKIS: No, but if you can figure out a way to get something that is larger -- Dana is allowing for a mix.

MEMBER WALLIS: But you've got to be realistic. You can't just figure out something that's absurd; you've still got to be --

CHAIRMAN APOSTOLAKIS: Well, that's why it's not an easy problem. But the idea, though, is not bad, that at some point you get away from the mean or the best estimates, and say I want higher confidence now, because this is the end of the line. And the other thing is, of course, Tom mentioned security evaluation; make that part of the whole process. Then maybe the reason why you need the containment is not the source term; to keep things outside, not inside.

MR. KING: Or maybe there is a way or a scenario that PRA isn't amenable to, through the security concerns at least.

CHAIRMAN APOSTOLAKIS: That's right, that's right, so we have to risk-inform the security process.

(Laughter.)

CHAIRMAN APOSTOLAKIS: Well, we gave you extra time, Tom, but come on.

(Laughter.)

MR. KING: You guys are lucky; my wife's out of town, so I don't have to be home at any special time.

Alright, containment, sixth issue, versus confinement. This was an issue raised back on the MHTGR days. What the Staff recommended and what the Commission endorsed was you could have a design, they didn't say it had to have a containment -- they said it must do two things. One, it must meet the release limits, whatever they are in the regulations; and it must for 24 hours have a performance that you can show that its leak rate, whatever leak rate you assumed in the safety analysis, will not be exceeded in the first 24 hours. So if you've got a confinement, and you can show that in the first 24 hours it's going to work the way it's supposed to work for a containment, you could make the case for a confinement.

Again, I think this is a fundamental defense-in-depth issue. It certainly is dependent upon the event selection and source term issues, how they turn out.

Some designers will argue that having the containment on an HTGR makes it less safe because you make the heat removal more complicated. The passive systems have to be more complicated, you have to have active systems.

That's certainly one argument that we will hear. Another one is that you'll retain that hot helium and you'll have a pressurized building and that provides a driving force for any fission products that are in there. That makes it less safe. There have been designs approved in other countries without containment buildings, most notably Germany.

On the flip side, I see that containment is -- can be a way where you don't have to worry so much about fuel performance and heat removal system performance. You don't have to worry so much about air ingress. It can have some positive aspects. So I think looking at the design both with and without the containment might be a reasonable criteria to impose to see what are the safety benefits. Does it really detract from safety or does it really maybe improve safety?

I'm just sort of speaking out loud here, thinking about additional criteria that we might want to think about before going forward to the Commission.

Saying, do they want to stick with their 1993 position, or do they want to embellish the criteria or take a different position?

MEMBER KRESS: This certainly is tied in with everything that's going on.

MR. KING: Yes. If the Commission decides big source term, then I think that settles this one. If they decide scenario-specific, small source term -- there could be other reasons; public confidence is probably something they'll think about.

MEMBER WALLIS: I wonder if that's right. I mean I'm sitting here, you're raising all these questions. You're somehow assuming that the Commission is magically going to be wise enough to make a good choice?

MR. KING: Yes.

MEMBER KRESS: That's their job.

MEMBER WALLIS: No, I don't. I think you've got to lay out the rationale for why they ought to make the various choices.

MEMBER KRESS: I think it's incumbent upon these guys to give them lots of information.

MEMBER WALLIS: And they've got to give a way of thinking as well as just letting them --

CHAIRMAN APOSTOLAKIS: They usually do.

They usually do. They don't just give them all the same arguments.

MR. KING: Our plan is to give a recommendation. Here are the options we considered; here's the pros and cons. Here's what we recommend. Here's why.

CHAIRMAN APOSTOLAKIS: That's why you come here before us.

MR. KING: Yes, that's what I want to talk about, is the steps to do that. Let me just touch on the last issue and then we can talk schedule.

Emergency planning. Again, the HTGR designers are saying we don't need to have off-site emergency planning --

MEMBER POWERS: What's EAB?

MR. KING: Exclusion area boundary; that's the fence around the plant. They say they'll never exceed one rem at the fencepost; therefore, you don't need to evacuate people. This was looked at again ten years ago with the MHTGR. What the Commission said was, they did not agree to making any change to emergency planning at that time. They said what they would need before they would make a change to emergency planning was, get some operating experience on these plants to see if all their safety claims really, in fact, pan out.

They may want to retain that position today, or they may want to reconsider. I don't know. We'll lay out the options and the pros and cons and see where they want to go. To some extent, you could consider this: emergency planning is the last line of defense-in-depth, and if you're going to back off in those other areas, maybe you don't want to back off there until you really do have some operating experience. To me it's a reasonable position.

MEMBER LEITCH: As long the only sites being considered are existing sites, it's kind of a moot point.

MR. KING: For existing sites, it's probably a moot point; I agree. But again, it's also something where, if you do want to change it later, it's not like you have to change the plant design. You could change the emergency planning plans later without -- you know, put a containment on the plant or something.

Schedule. We'll be having this workshop. The next step after the workshop, in a couple of weeks, is to then start formulating recommendations, draft recommendations. I would like to come back to you -- Subcommittee, Full Committee -- certainly, at the latest, at the Full Committee meeting in December. So in closing, think about the schedule, Subcommittee, Full Committee, leading up to the December Full Committee Meeting. Thank you.

(Whereupon, the proceedings went off the record at 4:00 p.m.)