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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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563rd MEETING

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WEDNESDAY,

JUNE 3, 2009

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ROCKVILLE, MARYLAND

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The Committee convened in Room T2B3 in the Headquarters of the Nuclear Regulatory Commission, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, at 8:30 a.m., Dr Mario Bonaca, Chair, presiding.

COMMITTEE MEMBERS PRESENT:

MARIO V. BONACA, Chair

SAID I. ABDEL-KHALIK, Vice Chair

J. SAM ARMIJO, Member-At-Large

JOHN D. SIEBER

DENNIS C. BLEY

JOHN W. STETKAR

DANA A. POWERS

COMMITTEE MEMBERS PRESENT: (cont.)

MICHAEL T. RYAN

OTTO L. MAYNARD

CHARLES H. BROWN, JR.

HAROLD B. RAY

MICHAEL CORRADINI

GEORGE E. APOSTOLAKIS

WILLIAM SHACK

NRC STAFF PRESENT:

TOM BLOUNT

WILLIAM KENNEDY

FRED BROWN

RICHARD CONATSER

STEVE GARRY

STEVE SCHAFFER

PAUL CLIFFORD

ZENA ABDULLAHI

BILL RULAND

ROYCE BEACOM

JEFF CIOCCO

Kathryn Brock

ALSO PRESENT:

WADE RICHARDS

ROBERT WILLIAMS

MIKE ROWE

GEORGE OLIVER

TOM EICHENBERG

ROBERT MONTGOMERY

DOUGLAS W. PRUITT

CHRIS HOFFMAN

KEN SCAROLA

In attendance from NIST:

Dr. Dimeo

Tom Myers

Dave Brown

T-A-B-L-E O-F C-O-N-T-E-N-T-S

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

8:29 a.m.

CHAIRMAN BONACA: Good morning. The meeting will now come to order. This is the first day of the 563rd Meeting of the Advisory Committee on Reactor Safeguards. In today's meeting, the Committee will consider the following: license renewal application and the revised Final Safety Evaluation Report for the National Institute of Standards and Technology Reactor; draft final Regulatory 1.21, DG-1186, Measuring, Evaluating and Reporting Radioactive Materials in Liquid and Gaseous Effluence and Solid Wastes; draft final Regulatory Guide 4.1, DG-4013, Radiological Environmental Monitoring for Nuclear Power Plants; pellet-clad interaction failures under extended power uprate conditions; diversity and defense in-depth topical report associated with the US-APWR design; subcommittee report; and preparation of ACRS reports.

Portion of the session dealing with pellet-clad interaction failures under EPU condition and diversity in-depth topical report associated with the US-APWR design will be closed to discuss information that is proprietary to Global Nuclear Fuel and/or Westinghouse and to Mitsubishi Heavy

Industries, respectively.

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

We have received no written comments or requests for time to make oral statements from members of the public regarding today's sessions. We have some TVA personnel on the phone bridge line to listen to the open portion of the session dealing with pellet-clad interaction failures under EPU conditions. To preclude interruption of the meeting, the phone line will be placed on listening-in mode.

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

With that, I think we can move to the first item on the agenda. That is the License Renewal Application for the NIST reactor and Mr. Sieber will lead us through that presentation.

MEMBER SIEBER: Thank you, Mr. Chairman and good morning to all that are present today.

This is actually the third meeting that we've had concerning the NIST reactor and we refer to it officially as a license renewal, but I think of it mentally and conceptually as a relicensing of the NIST reactor and that occurs because of the differences in the licensing requirements that are set forth in Title 10 for Type 104 reactors.

Research and test reactors are not required to do an annual update of their FSAR and so when the license is renewed or when the plant is relicensed that means that FSAR, the technical specifications, the supporting analysis including the safety analysis for the plant has to be brought up to date and all of that has been completed.

We had our subcommittee meeting on February 4, 2009. We've met with the full committee on April 2, 2009 at which time there was identified by the NIST personnel an open item and we will deal with that open item today.

I would mention that NIST stands for National Institute of Standards and Technology. In my younger days, it was known as the National Bureau of Standards. It's an agency in the Department of Commerce. The reactor is named for the National Bureau of Standards and it's called NBSR and it is one of the highest power research and test reactors licensed in the United States at 20 megawatts-thermal. I think the other one is at Brookhaven. And so it presents interesting opportunities for tests and experiments and since because of its higher power output it is called a test reactor as opposed to a research reactor.

This reactor has been in existence for many years. The design was started in 1961 I believe and I was a relatively young man at the time and it has a outstanding operating history.

What I would like to do is to introduce to you Mr. Tom Blount who is Deputy Director of the Division of Policy and Rulemaking, NRR, and that is the organization within the staff that's responsible for the relicensing effort of this reactor.

Tom.

MR. BLOUNT: Thank you, Mr. Sieber.

Good morning, Mr. Chairman and Members of the Committee. My name is Tom Blount and I am the Deputy Director for the Division of Policy and Rulemaking in NRR. NRR or my division is the one that's responsible for reviewing the license application or license renewal applications before you today.

Here with me I have Kathryn Brock sitting right behind me as the chief of the Research and Test Reactor Branch A, the branch which has primarily responsibly for the license renewal review and to my left is Bill Kennedy, William Kennedy, the project manager for the renewal review who will be leading the staff presentation.

Today we will begin with the presentation by the Licensee that will include a discussion of the open item identified by the Licensee prior to full committee meeting this past April.

On March 30th, the Licensee self-identified an error in a dataset used to benchmark the model of loss of offsite power accident. The Licensee reported the error in a timely manner and has since revised the dataset and submitted an updated analysis of the accident scenario.

Mr. Kennedy has reviewed the dataset and submitted an updated analysis of the accident scenario. Mr. Kennedy reviewed the new analysis and found that there is still an adequate safety margin and the staff's principal safety conclusions remain unchanged.

After the Licensee's presentation, Mr. Kennedy will discuss the staff's review of the open item and explain the update to the staff's Final Safety Evaluation Report. The open item has been successfully resolved and the staff's Final Safety Evaluation Report is currently in the concurrence process. I expect the Final Safety Evaluation will be published as a NUREG series document early this summer.

With that, I'll turn the presentation over to the Licensee. Thank you.

MR. RICHARDS: Good morning, Mr. Chairman, Members of the ACRS. My name is Wade Richards. I'm the Chief of Reactor Operations and Engineering and today we will be presenting the analysis for this single open item that had as a result of the last meeting.

With me today is Dr. Dimeo, the Director of the NIST Center for Neutron Research, Mr. Tom Myers, the Chief of Reactor Operations, Mr. Dave Brown, Chief of Health Physics and up here to present that analysis actually is Dr. Rowe and Dr. Williams.

As was stated, while we were responding to questions in the earlier meetings we found that we identified an issue with the pump coast-down and the bottom line is that data that was in the present FSAR, the analysis for that data was done under different circumstances than what we actually had and Dr. Rowe and Dr. Williams have done the analysis to bring it up to date and in line with what we actually have today. So I'll let Dr. Rowe go ahead and go through this analysis with you.

DR. ROWE: Yes. What we identified was the model which had been used in the RELAP calculation did not -- have been compared to a dataset which was not in fact measured under the same conditions as we have in the reactor and we wanted to ensure that the analysis was conservative with respect to the actual measure. So we remeasured that curve immediately following the meeting that we had with you here. We did remeasure the pump coast-down curve and compared --

MEMBER SIEBER: You actually conducted a coast-down test.

DR. ROWE: Yes, we did. What we did is we did an exact duplication of what would happen in a power failure. We established a system that cut power to all of the pumps. We were running at full flow. We cut power to the pumps and we tracked the flow. We tracked when the scram occurred. And we tracked all of the activities that would go on in a power failure and we did that for the most conservative case in which the shutdown pump does not turn on. So we only had coast-down.

We did it for both, but we analyzed both and we analyzed the limiting characteristics when neither shutdown pump -- Two shutdown pumps are available. They both have both AC power and DC power. But we analyzed the case when neither one turned on.

MEMBER SIEBER: Okay. Just for the members' information, the NIST reactor differs from power reactors that we ordinarily deal with in that if you lose power to the coolant pump of a power reactor you get a instant reactor trip from the loss of power to the coolant pumps.

And in this reactor that does not occur. It measures the flow and the reactor has a number of -- The instrument and control system has a number of setpoints at various levels of -- that actuate various equipment in the plant, one of which is to trip the reactor. So there is a delay that occurs where the reactor is running but the pump flow is decreasing.

DR. ROWE: Thank you.

MEMBER CORRADINI: Just so maybe you said this last month or maybe two months ago when you guys were here. How long is that delta time?

DR. ROWE: It's one and a half seconds.

If you would step forward two of the graphs. If I can go up.

MEMBER CORRADINI: I don't think they let you, Mike.

MEMBER SIEBER: You got to speak into the microphone. We can pick it out on the --

DR. ROWE: Okay. Let me try and bring it up. It's hard without being up there and I don't have a pointer with me.

MEMBER APOSTOLAKIS: Can you use a cursor here on the computer?

DR. ROWE: Yes.

MEMBER APOSTOLAKIS: Sit down here and do it.

DR. ROWE: Okay.

MEMBER CORRADINI: They'll pick you up.

DR. ROWE: That should be okay now.

MEMBER APOSTOLAKIS: Yes.

DR. ROWE: Yes. Michael, where we scram -- First let me explain this curve. This is what we discovered was incorrect. If you look here at the RELAP, the green curve that is labeled RELAP, that was the curve that was used in the analysis. It was compared to the magenta curve which is a coast-down curve that was measured under different conditions which is what we discovered and we wanted to be sure that we were being adequately conservative because we had compared to data that were done under different conditions.

We then measured the yellow curve. The yellow curve is the one that had been measured recently in which we exactly duplicated everything. Now in this diagram we actually for the measured curve which is where the cursor is, 1.5 seconds into the event.

MEMBER CORRADINI: Okay. Thank you.

DR. ROWE: So now coming back to this curve where again we have the curve used in the prior analysis, this was a curve that was derived from RELAP and then we have the magenta curve which is again the newly measured data. This is the data that we just completed the measurements on.

And you can see that those two curves are essentially the same curve out to two seconds and beyond. So they are essentially identical out to beyond the point of risk to reactor scram. The point at which we get the minimum critical heat flux is just before the reactor scram. So they are essentially identical past the point of minimum critical heat flux.

Then two curves continue on out and we have measured them out further than we show there and what we see is summarized here. The minimum critical heat flux ratio occurs at approximately 1.5 seconds. The two curves are essentially the same. The one we used originally, the one we have now measured. I add that we actually have reanalyzed all of the data using the measured curves.

MEMBER POWERS: So there is nothing conservative about your analysis at all. It's just realistic.

DR. ROWE: I'm sorry, sir.

MEMBER POWERS: There's nothing -- It's not conservative. It's just realistic, right?

DR. ROWE: It's now realistic at this point, but the curve that I showed you there that was used in another set is a conservative representation of the data.

MEMBER POWERS: It's identical to the data.

DR. ROWE: It is conservative. It is a functional form --

MEMBER POWERS: Speak to me of why it's conservative.

DR. ROWE: It's a functional form which

goes below all of the datapoints, through or below.

MEMBER POWERS: Show me the curve again because it doesn't look conservative at least.

DR. ROWE: It goes below. It never goes above a single datapoint.

MEMBER APOSTOLAKIS: Are you showing the data here?

DR. ROWE: The data is the magenta curve. But I will take your point that it is not extraordinarily conservative. It is not -- What I would say is it is conservative in the fact that it is not only square fit. It is a fit which was chosen to be below all datapoints.

MEMBER APOSTOLAKIS: But the actual datapoints, are they shown there?

DR. ROWE: Yes, they are, sir.

MEMBER APOSTOLAKIS: This is the curve.

DR. ROWE: The datapoints --

MEMBER APOSTOLAKIS: But where are the datapoints?

DR. ROWE: They're the diamonds.

MEMBER APOSTOLAKIS: Oh.

MEMBER SIEBER: Right.

DR. ROWE: Yes, and the dome is actually correct. They go -- On this scale, they go through the data. What I'm saying is if I look at the data seriously they are below the curve which is slightly below every point. I didn't do a least-square fit to use for the analysis. This is the curve. The curve is what is used in the analysis.

MEMBER APOSTOLAKIS: So you just did the curve going through each point. Is that right?

(Several simultaneous comments.)

MEMBER POWERS: So if I wait five years and we measure the flow, does the flow still go through all the data through this curve?

DR. ROWE: I would only say that they did for three curves that we measured this time.

MEMBER POWERS: You measured three on three different days. How about five years from now?

DR. ROWE: I believe that they will and I believe that we had adequate margin.

MEMBER POWERS: You must have a reason for believing that. Faith.

DR. ROWE: Faith.

MEMBER POWERS: I mean why do you believe this? Why are there no degradation in this pump?

DR. ROWE: Simply because I don't know of a physical mechanism with which to get it. We're not -- They don't depend on the performance of a mechanical system. I mean it is a passive system at that point. The pumps are shut off.

MEMBER POWERS: You don't have data taken five years ago that you can prepare against.

DR. ROWE: I do have data that was taken which I haven't shown you yet which was taken -- Do you remember a year, Bob?

DR. WILLIAMS: In the early '80s.

DR. ROWE: In the early '80s which is essentially the same as this data.

MEMBER POWERS: Okay.

DR. ROWE: As it is then. So, yes, we know.

MEMBER POWERS: That's good enough for me. It would be nice to see the comparison, but I'll take you on faith.

MEMBER SIEBER: Actually this system started operating in 1967.

DR. ROWE: The first criticality was '67. Power of operation was '69.

MEMBER SIEBER: Okay. So there is a history on these pumps.

DR. ROWE: There is, but I mean in direct response to the question we do have one other measurement which is consistent with these data.

MEMBER SIEBER: And this is a mild environment that's 110 degrees and very low pressure. It's not a major issue and you could see the flywheel effect in the first few tenths of a second up there as the pump is (Clearing of throat) full scale plant that initial curve due to the inertia of the flywheel is much more pronounced. But you can actually see it on these curves.

DR. ROWE: Yes. In this case, all that we have is the inertia of the pump itself at the beginning.

MEMBER SIEBER: Right. The motor and the pump and the shaft.

DR. ROWE: So we redid the analysis using the curve that I showed you that we were using the measured curve which as you saw does deviate at higher longer time. We did the analysis. The minimum critical heat flux ratio which we did with the original curve changed to 2.17 from 2.19 which I say is the same number. I don't claim any difference between those numbers.

We did a detailed analysis out to 30 seconds to ensure that there was a smooth transition to a long-term stable state and the fuel temperature remained below 137 degrees C which is substantially below a safety limit.

MEMBER BROWN: How much is substantially?

DR. ROWE: It just says it remains -- Four hundred and fifty.

MEMBER BROWN: That's substantially. I'll take your word.

DR. ROWE: Yes.

MEMBER BROWN: That works.

MEMBER SIEBER: There is a lot of margin.

DR. ROWE: There is a significant margin.

MEMBER CORRADINI: So just out of curiosity, what is the applicable CHF correlation that you used to compare to here to get your ratio?

DR. ROWE: This one was done originally with the Mirshak. The original data were taken with the Mirshak. We then used and my memory's slipping --

DR. WILLIAMS: I'm Bob Williams. This is also with Mirshak.

MEMBER CORRADINI: Which is essentially tied to your fuel geometry.

DR. ROWE: It is the same. It is done with our fuel geometry. We have also used a separate correlation.

MEMBER CORRADINI: That's all right. If you have one that's empirically tight in your fuel geometry, that answers my question.

DR. ROWE: I mean empirically in the sense that it did the right geometry.

MEMBER CORRADINI: Right. Well, I just wanted --

(Simultaneous conversation.)

DR. ROWE: -- the geometry.

MEMBER CORRADINI: Yes, I understand, but you actually have CHF data for this fuel geometry, for this fuel lattice geometry.

DR. ROWE: I'm trying to make sure I don't mislead you. They've been done four square channels heated from both sides.

MEMBER CORRADINI: Okay.

DR. ROWE: And if that's the question which of approximately -- but it's not identical. It's not that we have mocked up the exact fuel.

MEMBER CORRADINI: Yes. I'm with you.

DR. ROWE: So from that we conclude that we have looked at the limiting loss of flow accident with no shutdown pump cited.

Now again as I said we do have two shutdown pumps and those shutdown pumps have DC motors. So in the event of a power failure we actually expect them to come on. But we did not assume that for this accident. So this is a limiting accident when neither of those shutdown pumps are cited.

MEMBER BROWN: They have a DC power supply.

DR. ROWE: Yes.

MEMBER BROWN: Separate.

DR. ROWE: They have --

MEMBER BROWN: Independent power supplies, one for each one, or they both come off the same power supply?

DR. ROWE: They're both off the same one. But there is also diesel generator system. There is another power source. But in order to be conservative we did not analyze them with that system coming up. We analyzed it without that system.

MEMBER SIEBER: Well, the fact is that you don't need the DC pumps to maintain core limits.

DR. ROWE: That's why we did it that way. It is the conservative.

MEMBER SIEBER: Right.

DR. ROWE: The conservative calculation.

MEMBER SIEBER: That's why it's not safety related.

MEMBER STETKAR: Just to remind the rest

of the Committee. The genesis of this whole reanalysis was closure of a single valve that indeed would block all flow. Is that correct?

DR. ROWE: That is correct.

MEMBER STETKAR: And this analysis was done because the flow coast-down on this particular transient occurs over a shorter period of time than the closure rate for that valve. Is that correct? I'm trying to remember my --

DR. ROWE: That is exactly correct. You raised the issue of what would happen if the DWV-19 valve were closed which is the one on the out -- We analyzed that in response to your question. In looking through that analysis, we became concerned because we discovered not that there was anything wrong with the RELAP curve, but that it had been compared to data which were not taken under the same circumstances. So we wanted to be sure that the curves that were actually used were representative of the real system. So, yes, that is correct.

And in fact that accident is limiting. In this case, we have redone the analysis. We followed it all the way into a stable equilibrium. Minimum critical heat flux of greater than two. The fuel temperature is well below the safety limit and we will change the SAR to reflect this analysis as we have presented it.

MEMBER SIEBER: What is the safety limit for critical heat flux?

DR. ROWE: I'm sorry, sir.

MEMBER SIEBER: How low can you go on critical heat flux before you exceed a safety limit? Do you know?

DR. ROWE: This is always a difficult question. Statistically we have looked at that analysis and we can go down to about 1.4 and maintain satisfactory limits.

MEMBER CORRADINI: For the fuel you mean.

DR. ROWE: Yes.

MEMBER CORRADINI: Okay. So you actually get to concern about fuel temperature before you get a CHF ratio of one.

MEMBER SIEBER: Right.

DR. ROWE: Well, no. That's not what I said, Michael.

MEMBER CORRADINI: That surprises me.

DR. ROWE: What I said is that they're always uncertain in the correlation and in the calculation.

MEMBER CORRADINI: Okay. Fine.

DR. ROWE: And when I put those uncertainties in when I get to about 1.4 I'm not going to say I'm definitely not going to hit one. Different state.

MEMBER CORRADINI: Okay.

DR. ROWE: It has to do with the uncertainties in the correlation and the uncertainties in the calculation.

MEMBER CORRADINI: That's fine.

DR. ROWE: The uncertainties in the fuel. All of the things that go in and that statistical analysis is in the SAR.

VICE CHAIR ABDEL-KHALIK: And that's in the 95 confidence level.

DR. ROWE: Yes, that would be about correct. Yes.

MEMBER SIEBER: So the margin that you have is between 1.4 and 2.17 that would allow for things like degradation of pump performance and so forth which is the subject of Dr. Powers' previous question.

DR. ROWE: Yes.

MEMBER SIEBER: Okay. Are there any further questions?

(No verbal response.)

If not, I think I'm impressed by the way in the inquisitiveness of the NIST personnel and their initiative to pursue this matter and I think that they have done the right thing and so we thank you very much for your presentation and your excellent work and maybe it's time for NRR.

DR. ROWE: Thank you. Thank you all.

(Off the record comments.)

MR. KENNEDY: Good morning, Mr. Chairman and Distinguished Members of the Committee. My name is William Kennedy. I'm the Project Manager for the NIST relicensing effort before you today. I'd like to thank the Committee members for taking the time to be here and my goal again today is to provide a common understanding of the staff's review and closure of this open item.

As we've already heard, this open item was identified by the Licensee in response to an additional analysis they performed on a closure of DV-19 loss of coolant flow accidents. They discovered that an outdated dataset had been used to benchmark the RELAP model used to analyze the loss of offsite power accident and in this case benchmark means that they were comparing their RELAP model against measured data to make sure that it was conservative. They weren't using outdated measured data as input to the model.

And this item was promptly reported by the Licensee to me by telephone on March 30th which was just four days, three or four days, prior to our last meeting here.

So upon being identified of this issue, my initial response was to perform a preliminary independent review and calculation to assess the safety significance of the error that may be introduced by benchmarking against this outdated dataset. I looked particularly at the rate of flow coast-down during the time after the reactor coolant flow has reached the trip set point, but before the shim safety arms actually begin to move into the reactor core because that time delay is where the rate of the flow coast-down directly impacts how the safety margin will decrease.

So I performed sort of a sensitivity analysis to see how quickly the flow could coast down and still not reach a point where the minimum critical heat flux ratio went below that 95 percent confidence level. And from that initial analysis, I was satisfied that this probably would not be any type of an immediate safety issue, but that it did warrant taking new data and performing a new analysis and me reviewing that analysis.

MEMBER BLEY: What kind of time did you come up with?

MR. KENNEDY: I found that even if it went to zero flow in two seconds which was the stroke time of the valve that I initially found that there would still be no fuel damage, still be up in the 1.7 to 1.8 ratio for minimum critical heat flux.

MEMBER SIEBER: Yes, you don't mean zero flow. You mean natural circulation.

MR. KENNEDY: I went to zero flow.

MEMBER SIEBER: Okay.

MEMBER STETKAR: It would be zero flow if that valve goes --

MEMBER SIEBER: Yes. Okay.

MEMBER STETKAR: The stroke time, NIST didn't remind me of this, the stroke time on that valve is 21 seconds. Is that correct?

DR. ROWE: That's correct.

MEMBER STETKAR: Thanks.

MR. KENNEDY: I should clarify. I was going to zero flow through the bypass valves that bypass the primary coolant pumps. So once they've stopped and the pumps have isolation valves to prevent backflow through the pump in the case of one pump failure. So I was looking at the three isolation valves, the four isolation valves, on the primary coolant pumps which have the stroke time of something like I believe two to three seconds. During the closure of those valves, the bypass, other valves that should open to allow the flow to coast down through the shutdown cooling pumps even if those don't start.

MEMBER STETKAR: And you allowed for that flow in your analysis.

MR. KENNEDY: Yes.

MEMBER STETKAR: Okay. That's a little different.

MR. KENNEDY: So I looked at linear coast-down from full flow to zero flow in two seconds and I found that there would still be a safety margin. I didn't think that that would actually happen, but that was the conservative case that I choose to look at for this preliminary analysis to try to assess the safety significance.

In addition, I reviewed all of the material in the application regarding the loss of offsite power accident and the primary coolant design of components that would directly impact this accident and the flow coast-down. I didn't find any indication in any of that material that this should have been an error that was caught by the staff or the contractors or even the Brookhaven National Lab that performed this analysis for NIST. So it appeared to me to be an isolated error and NIST confirmed that they were reviewing their other accident analyses to ensure that they didn't have any other unknown errors.

I also spoke to NIST about the results of my preliminary calculation and asked if they had done anything similar.

MEMBER BLEY: I'm sorry.

MR. KENNEDY: Yes.

MEMBER BLEY: I don't completely understand that statement. What does it mean that nobody should have caught that error? Does that mean that -- What errors should people catch?

MR. KENNEDY: What I mean by that statement is that in performing the analysis this wasn't an error in a calculation methodology or an error that would have come out during the analysis. This was the case where they had benchmarked against an old dataset and that was not caught. So you can say that the error, that this dataset, should never have been used in the first place.

MEMBER BLEY: So pedigree checking on

input data is not part of review. That's out of bounds.

MR. KENNEDY: That was not something that we looked at as part of our standard review plan to check all of the input data that is measured data, to actually go out to the site and verify that data.

MEMBER BLEY: Go ahead. Okay.

MEMBER CORRADINI: Can I say that back differently just so I make sure? So you're saying within your standard review plan if the licensee says, "We've tuned this to data, to experimental data, for the particular conditions," you take it on faith and move on. You don't go and do some sort of spot check audit of that. That's what I heard you just said.

MR. KENNEDY: Correct. However, we would look and compare that measured data to system design to make sure that it's reasonable for that type of system design.

MEMBER CORRADINI: But it doesn't look crazy. But on the other hand, it might be in this case off and you would not have gone to audit and check for that directly.

MR. KENNEDY: Correct.

MEMBER CORRADINI: Okay.

MEMBER SIEBER: But the dataset that they used is actual data. There is not an error in that.

MEMBER BLEY: It's just not representative.

MR. KENNEDY: That's correct.

MEMBER BLEY: It just came from the wrong application.

MEMBER SIEBER: Right.

MR. KENNEDY: It was actual data that was measured that seemed reasonable for the system design and for the accident being handled in this.

MEMBER MAYNARD: I think the thing that some of us are struggling with a little bit is I can understand that it may not be something that the NRC would normally look at or would catch 100 percent. But your statement earlier was kind of a broad one that basically, did to me, that nobody expected to find this and people do make mistakes. But somebody should have verified that in the beginning if they were using the right data and stuff. It's not a big deal other than just the fact that the broad statement that nobody should have caught this is a little bit troubling.

(Off the record comments.)

MEMBER CORRADINI: I understand what you're getting at.

MEMBER MAYNARD: I understand this isn't a blatant gross error or whatever or across the whole thing, but at some point in the beginning somebody should have identified this and caught it.

MEMBER BLEY: But if we were not at least spot-checking pedigrees on things, that leaves a lot of question about if we're doing reviews that are leveled at going to ensure plants are designed and built to the standards that we think they are designed and built to.

MEMBER CORRADINI: I guess I think I understand what you said. The way I view it is the licensee should have caught it or saw it and they did. There was some question and they found it. But it's not -- The only thing that I'm checking is it's not in your standard review plan to do an audit of these sorts of things necessarily. That's what I got out of your discussions.

MR. KENNEDY: That's correct.

MEMBER CORRADINI: Okay. Thank you.

MEMBER SIEBER: Well, in the broad scheme of things, the best situation is when the applicant catches it. A little worse is when the staff catches it. And really bad is when we catch it. And so here we had of an unfortunate situation this is the best way to get it.

MEMBER BROWN: But a question was asked in order to trigger the thought process in the first place and it was -- The red curve, I mean I'm not a real thermal hydraulic guy, but how many times do you see a flow coast-down go up after you turn the pumps off before it goes down and stays stable for about two seconds. I mean that is just -- I've never seen a flow coast-down curve do that. They all go down.

So if you look at it and say, "Is the data consistent with the flow coast-down" I'm not sure I would have come to that and again I'm not mechanical guy for pumps. But I've just never seen one where the pump went up initially and stayed stable for about two seconds before it starts going down.

MEMBER SIEBER: It has a hump in it.

MEMBER BROWN: It did.

MR. KENNEDY: On that issue, that flow coast-down curve that was presented today is not what we had to review in the initial application. What we had in the initial application, and I don't have it in my slides, but it is in Appendix A to their accident analysis, shows a curve.

I could pass it over to you, but it shows a curve that's fairly similar to the RELAP model on the scale that it's shown on that curve. It's in the bottom there. You can see the solid line being the RELAP model and the dotted line being the measured data that we were shown that it was benchmarked against.

MEMBER CORRADINI: So I don't want to --

MEMBER BROWN: Why is the curve shown with the flow goes up?

MEMBER ARMIJO: It's using a Kirfitee (phonetic) technique.

MEMBER SIEBER: I don't remember --

MEMBER BLEY: They're different, Sam.

MEMBER CORRADINI: I think we need some clarification from you.

MEMBER BLEY: Yeah. I don't understand --

MEMBER SIEBER: Would you come to a microphone please, Dr. Rowe?

DR. ROWE: Mike Rowe. That's my fault that I did not specify when I was speaking to you. That line is just to help you see the points. The points were very broadly spaced because I had condensed them. That line does not have significance.

MEMBER BROWN: The red.

DR. ROWE: That does not have -- The points have significance.

MEMBER CORRADINI: The dots are the only thing that are real.

DR. ROWE: The points I measured.

MEMBER BROWN: Okay. But the points still stay fairly stable. I mean a significant difference between the RELAP model and the --

DR. ROWE: Yes. There is a significance.

MEMBER BROWN: Out to two seconds. I mean three seconds. Four seconds. It's a massive difference relative to flow. There is three times the flow that's shown. It's just unusual to see something that disparate.

DR. ROWE: First, the people who should have found this are us. We did. I think we should have found it earlier, but just the point is that the flow did not go up in that earlier measurement. It went down. It was measured under different circumstances than the current ones. The calculation was done for the circumstances under which we did the new measurements. So in fact the calculations did a great job of reproducing the actual geometry in the system.

MEMBER BROWN: I'm not questioning that. I'm only questioning the comment that nobody would have thought about looking at the initial data when you have that much of a disparity in a flow coast-down curve that looks significantly different than what you would normally experience.

MEMBER BLEY: I guess back to my original question there's a lot of data. You can check everything completely. But what bothered me was nobody could have caught this. An explanation of what you did do instead of that is much more satisfying and that leaves me kind of questioning do we check pedigree on anything. So go ahead. It was the way it was presented.

MR. KENNEDY: If I could clarify what I meant by no one should have caught it, what I meant was that this was not a glaring error in the analysis. This was fairly well embedded in the analysis and that the licensee has ultimate responsibility for the error in this analysis and it wasn't something that would normally be caught during the NRC review.

As I had mentioned, my calculation and speaking with the Licensee it was in close agreement with their initial assessment and I believe there are others who performed similar calculations and found that a decrease in the safety margin of 10 percent could be possible given the faster flow coast-down.

I discussed the significance of this error with the Licensee and we formed a plan to address the outdated data and to perform new analysis and to perform an evaluation of that new analysis.

VICE CHAIR ABDEL-KHALIK: Just out of curiosity, your hand calculation where you assume a linear coast-down to zero at two seconds, you also need the trip time to calculate the minimum critical heat flux ratio.

MR. KENNEDY: Correct.

VICE CHAIR ABDEL-KHALIK: Where did you get the trip time?

MR. KENNEDY: The trip time is specified in their technical specifications, at least, the delay between when the value reaches the flow coast-down at the low flow setpoint.

VICE CHAIR ABDEL-KHALIK: So you got it from the tech specs.

MR. KENNEDY: Correct. There's a 400 millisecond delay for the instrumentation and motion of the scram arms into the core. The actual time when the flow reaches the setpoint I used from their specified value of I believe it was 4700 gallons per minute flow on the outer plenum.

NIST submitted to me a revised loss of offsite power accident analysis on April 22nd. They notified me immediately after the finished performing the new flow coast-down data taking. So I had an idea of what their new flow coast-down data was. Very shortly after they identified this error, they provided the official analysis on April 22nd.

I compared the updated flow coast-down dataset against the old dataset and I found them to be nearly identical. I would qualify the statement "nearly identical" as meaning during that area of interest where the flow has reached the flow setpoint but before the shim arms have started to move into the core, the 400 millisecond delay, and if you look again at the Licensee's viewgraph you can see that the slope of the two lines for the flow coast-down is nearly identical during that period, the magenta line and the green line and the yellow lines. So the slope there is what's important during that 400 millisecond delay.

MEMBER SIEBER: And how long does it take the shim arm to drop? About a second?

MR. KENNEDY: The reactor begins to shut down immediately once they've moved into the core and so the power begins to decrease.

MEMBER SIEBER: Yes. It has to be a certain amount there before it begins to shut down.

MR. KENNEDY: Yes. That's at the end of

the 400 milliseconds you start to see negative reactivity inserted into the reactor and you see the power begin to decrease.

(Off the record comments.)

I also reviewed the assumptions used in the updated accident analysis and found them to be as conservative as those used in the original analysis. There was the question about the new measured flow is not as conservative as the old measured flow when compared to the model. But the other conservatisms built into the model remained.

And my updated safety evaluation, again I included the new numbers for the minimum critical heat flux ratio which is the safety margin and the new numbers for the maximum fuel temperature which is less than a degree higher than the old number and as a result I concluded that there is reasonable assurance that the loss of offsite power accident will not lead to fuel damage and that its consequences are bounded by the maximum hypothetical accident.

Any additional questions?

(No verbal response.)

CHAIRMAN BONACA: Any questions?

(No verbal response.

MR. KENNEDY: Thank you.

MEMBER SIEBER: If there are no questions, thank you very much and thanks again to the NIST people and it's yours, Mr. Chairman.

CHAIRMAN BONACA: Thank you very much and this gives us a long break. I guess we will start again at 10:00 a.m. Off the record.

(Whereupon, a recess was taken.)

CHAIRMAN BONACA: On the record. Okay. Let's get back into session.

A Draft Final Regulatory Guides 1.21 and 4.1 and Dr. Ryan will take us through this presentation.

MEMBER RYAN: Thank you, Mr. Chairman. I'd like to have the staff today present the work that they've done on updating Reg. Guide 1.21 and Reg. Guide 4.1 and we had a very successful and productive subcommittee meeting last month where we offered insights and opportunities to improve the reg. guides and I think the staff's responded to those suggestions quite well.

So without further ado, let me introduce Fred Brown who is the Director of the Division of Inspection and Regulatory Support. So, Fred, without further ado, let me ask you to introduce yourself and your team and help us out.

MR. BROWN: Great. Thank you very much. I am Fred Brown, Division Director for Division of Inspection and Regional Support in NRR. We do have the two reg. guides. We appreciate the opportunity to present them to you and we're looking for your support and a recommendation to approve both reg. guides.

The presentation will be given by two of our staff members dealing with radiological effluence and environmental monitoring, Richard Conatser and Steve Garry.

MR. CONATSER: Thank you, Fred. My name is Richard Conatser and thanks to the Committee for inviting Steve and I here today to discuss these two reg. guides, two proposed reg. guides.

The first one is Rev 2 to Reg. Guide 1.21. This is for radiological effluent reporting and monitoring and the second one is Rev 2 to Reg. Guide 4.1 which is for environmental monitoring. And like Fred said, we're looking for any comments that you might have. We had a lot of good comments from the subcommittee meeting a month ago. We were able to incorporate those comments. If you guys would have additional comments, we would encourage that and we're looking for recommendation for approval.

Okay. A brief outline of what we're going to cover this morning. There will be an introduction here. I'll talk about the people who are involved in the project, what it is, and the project itself. We'll go over some history and it's not really history. It's more like drivers for change. It's why I'm speaking before you today and bringing these two reg. guides to you.

Then we'll go into the documents themselves. We'll spend about ten minutes, maybe 15 minutes, on the documents, covering them ever so briefly from a very high level. Then we'll talk about how these documents fit into the Regulatory Guide Update Initiative that the NRC has.

And then we'll list the reasons for revising the reg. guides and then discuss some things that may be of interest to the ACRS such as backfit considerations, consistency between documents and whether or not -- there was some suggestions of maybe delaying these publications and we'll discuss that as well. I know those are questions you guys would probably have anyway. And we'll close it up with some questions and that's basically what I have for you.

The presentation goes about 40 minutes. It's designed for asking questions as we go. So I know I was sitting in on the previous presentation and

I noticed you guys weren't shy. So chime in. And there will be, of course, questions at the end.

Okay. The people on the project. We had a team formed in 2006. It was formed of members from Headquarters and that's NRR, Nuclear Reactor Regulation, NRO, New Reactors, FSME, Federal and State Materials and Environmental Protection, And Research. And we also had input from the regions.

All four regions participated in this team and some of those are there today. We have Steve Schaffer, Dr. Schaffer back here. Raise your hand, Steve. He's in New Reactors and provided input on that. And Jim Shepherd from FSME is here as well. So thank you for coming today.

MEMBER ARMIJO: I noticed in the handout NMSS was in the handout, but not on your slide. Was that an error or?

MR. CONATSER: That was yes. I actually covered this project here. I've been with the agency now for a year and the team was formed before I got here. I had NMSS on that slide initially that you probably have there before you.

MEMBER ARMIJO: Yes.

MR. CONATSER: We did get comments from NMSS but in conversations with the group members I did find out just a few days ago that actually NMSS did not participate in that group. So, yes, that's one -- That's the only thing I've changed on these slides from what you have before you. Good catch.

MR. GARRY: Their role was to review the ISFSI language with regard to environmental protection. There are some requirements for environmental protection around ISFSI and effluent monitoring.

MR. CONATSER: Okay. And the progress on where the project is today, what we have is a *Federal Register* notice go out October and November of last year. We held a public meeting in January.

We've got a lot of public and industry comments and thanks, George, and NEI for putting all those together from the industry. There was a number of those and we were able to incorporate those and I think that strengthen the document considerably. We did present this through our office concurrence process and got comments frm the different offices. We've incorporated those comments as well. And we went to the ACRS subcommittee last month and they had additional comments and we incorporated those. So it's been a long process.

Right now it's in OGC. That's the lawyer for you. And now I'm here before this committee and hopefully we're toward the end of this, although it's been fun.

Okay. History, why are we here? What are the drivers for change? Well, you guys have probably heard the tritium in the groundwater issues. We had the Salem spent fuel pool leak in 2003, the Braidwood leak in 2005 or the recognition of tritium in well in 2005, Indian Point in September 2005 with a crack in the spent fuel pool and as we go through these events, there's been more events since that time obviously.

But as a result of primarily these events and some others, there was a task force formed, an NRC task force. And that task force had a report that was published in September of 2006. That report had a total of 26 recommendations. Ten of those recommendations would be incorporated into Reg. Guide 1.21 on radiological effluent monitoring and four of the recommendations would be incorporated Reg. Guide 4.1. And I'm not going to -- Question?

MEMBER SIEBER: What happened to the other 12?

MR. CONATSER: There were other documents that those got incorporated into external to Reg. Guides 1.21 and 4.1 and I can't recall the scope of those right now.

MR. GARRY: Some of those were directed at Research to look into the research of things such as whether boric acid could be used as an indicator. Pipe integrity went to the engineering group. So they were recommendations outside of our scope.

MEMBER SIEBER: Thank you.

MR. CONATSER: So we incorporated those recommendations into this document. I don't want to go over all these recommendations here. But I will list some language that was listed in some of these just to give you an idea of some things we incorporated into these documents.

The task force asked us to include guidance for detecting leaks and spills before they migrate offsite at the power plants. They asked us to include guidance in these reg. guides to include spills and leaks in annual reports. And they asked us to include guidance for making our guidance consistent with industry practices.

MEMBER BLEY: Let me interrupt and go back to what Steve said about some of these other recommendations that got sent out to more approach places. But if your guidance is telling people how to identify leakage before it becomes a problem and some of those other areas deal with that, should that not get incorporated into this even though it's not your area so that it's another indicator of leakage that you could use to prevent the radiological problems later on?

MR. CONATSER: That's a good comment and let me chime in before you have something to say there, Steve. What we tried to do on these documents, a lot of this stuff, a lot of the recommendations related to different things, things not related to like Reg. Guide 1.21, the scope of this document is for measuring and evaluating and reporting effluence from -- plants. So we could have put and actually it as discussed whether we want to include additional things in this document.

But what we found that may occur when you do that aside, a thing that happens when you do that, is you get duplication in different documents. So there may be other documents that cover those other recommendations. What we don't want to do is have different documents covering the same thing and what would happen eventually is you get inconsistencies between the documents.

MEMBER BLEY: Even though it might help you with your problem, you think it's covered well enough in the other documents.

MR. CONATSER: That's correct.

MEMBER BLEY: Okay.

MR. CONATSER: Yes, if they were unrelated to this document, if they were not in this scope, we did not include them. Good question.

And then the last one, the last comment I was going to read here, the last recommendation was to include guidance to use historical information in the decommissioning files for doing surveys and monitoring around the sites. So we includes those types of things in this document and these other recommendations as well.

Okay. Now we'll take a look at the documents themselves and I think you guys, I do have some copies over here at the table if you want to take a look at copies. I think you may have gotten those already. I'm not sure.

But Reg. Guide 1.21, that reg. guide is 71 pages long. It's on measuring, evaluating and recording radioactive materials in liquid and gaseous effluence and solid waste. It's got like a title page. It's got a couple of pages of table of contents. It's got like a four or eight page glossary, references. If you take away all that stuff, there's like 43 pages of meat in this document. So a lot of stuff there.

The original document I think had like 25 pages, something along that line. So it's a little bit heavier than the old document and includes additional guidance.

The Rev 1 of this document was issued in 1974, 35 years old. At that time, no one in here had a personal computer on their desk. So a lot of things have happened in the last 35 years. We thought it was about time to update these reg. guides.

That's the reg. guide. Let's see what it covers here. It covers things like reporting the normal effluence from power plants and that's what you'd expect to have in here. It has things like reporting abnormal releases from nuclear power plants.

Now Rev 1 of this reg. guide the way it defined abnormal releases excluded spills and leaks onsite. It only talked about leaks that went offsite and that was really I think an oversight on the part of the original authors of the document, the NRC at that time. But due to industry experience and lessons learned, we wanted to make sure that any spills and leaks onsite got put in the report. So we included that guidance about abnormal releases.

Let see what else it covers. It covers Carbon-14. The original Rev 1 talked about different radionuclides, but it was silent on Carbon-14. Well, this document mentions explicitly Carbon-14 and that a licensee should report that if it's principal at their site and that's right in line with the regulations. That's a little bit different from the Rev 1 version of the document.

The other things that are listed up here, the sampling, the surveys, the principal radionuclides and what we call the LLD or lower limit of detection, that's the sensitivity level for analysis, I'm not going to cover that, only to say with those concepts as they're covered in this document, this kind of includes a risk-informed context to those, meaning that when it talks about this and how the licensees can implement this at their sites it allows some room there for licensees to take a look at what are the major contributors at their site and those major contributors they should be spending more time on. They have to report everything obviously and look at everything, but it says the major ones you need to spend more time on those. It's kind of a risk-informed concept. So we tried to incorporate that into these documents as well.

Now what it doesn't cover and I think this alludes to the question we had earlier. It doesn't cover funding for decommissioning planning. That's going to be covered in a different reg. guide. It doesn't cover design for new reactors, that type of stuff. That's going to be covered in a different reg. guide.

So we tried to narrow the scope, the NRC staff that worked on this, narrow the scope to make sure that we had no duplication, that we tried to eliminate that to maintain consistency in all the documents. We thought that was important.

MEMBER RAY: What's the status of public input to these changes?

MR. CONATSER: The status of public input, we have incorporated the public comments and we had numerous of those comments and --

MEMBER RAY: There is nothing outstanding on all these changes and reporting requirements.

MEMBER RYAN: We're going to have a presentation from NEI shortly to at least summarize their involvement and their input.

MEMBER RAY: Okay. That's fine, Mike. Thank you.

MR. CONATSER: Yes. We covered everything of significance that needed to be included. There were some comments that would have marginal value I guess I should say.

MEMBER RAY: Right. I just didn't see anything about any controversies here. We'll get that later on.

MR. CONATSER: That will come at the end. We'll get there.

Okay. That's Reg. Guide 1.21. Any questions on Reg. Guide 1.21 before we leave that topic? That's pretty much all I was going to go into from a high level of detail.

MEMBER BROWN: Just a general comment to follow up on Harold's. When you get public comments and you incorporate them, do you then get the public's buy-in on how you incorporate the comments or do you just -- They just take what you give them?

(Laughter.)

MEMBER POWERS: You get their buy-in of course.

MR. CONATSER: The way this works is these documents are not disclosed to the public yet.

MEMBER BROWN: How did they comment on them then?

MR. CONATSER: Well, the original draft

versions that hit the *Federal Register* those are public.

MEMBER BROWN: Okay.

MR. CONATSER: Now how we've incorporated that.

MEMBER BROWN: They don't know.

MR. CONATSER: They don't know that yet. Now I've talked to NEI and told them we've incorporated --

MEMBER BROWN: All right. You answered my question.

MR. GARRY: Those get published with the final reg. guide though.

MEMBER BROWN: I understand that.

MR. GARRY: There's a common resolution.

MEMBER SIEBER: Dispositioned.

MEMBER RYAN: That's not the case of these reg. guides.

MEMBER BROWN: I understand. As I said, that was a general comment. I had not asked that in previous --

MEMBER BLEY: But, Charlie, that comment resolution section, it's often an appendix, list each comment or each class of comments and says what they did.

MEMBER BROWN: Yes, I've seen that, but my question was really on did the commentor see how it was done and I hadn't asked that before. So I took this opportunity in this area to ask that and you've answered my question.

MR. GARRY: It's the same process as you mentioned for all the reg. guides.

MEMBER BROWN: That's fine. Thank you.

MR. CONATSER: But that is an oddity.

MEMBER BROWN: Yes.

MR. CONATSER: Okay. So Reg. Guide 4.1 we'll cover the briefly here. Reg. Guide 4.1, a much smaller document, 20 pages total. It's got a cover page again, a table of contents, a bibliography, references. You take away all that stuff it's like 10 pages, 13 pages. A real neat document. The original document three and a half pages. So it's about six times larger. A lot more guidance in here for environmental monitoring. The original guidance was very thin at three and a half pages.

Some of the things that we have put into this and this covers environmental monitoring around the site, we included in a lot of detail on exposure pathways. Exposure pathways meaning how did the radionuclides get from the environment into man.

There are three exposure pathways, ingestion, inhalation and direct radiation. It's on everybody's -- EPA or DOE or whoever does radiation type stuff and you'll see those are the three common exposure pathways. That wasn't really gone into a lot of detail in the original Rev 1 version.

We also discussed routes of exposure. That wasn't addressed in Rev 1. The routes of exposure, let me give you an example like the ingestion pathway which is one of the three I just mentioned. So the ingestion pathway, that means you would eat it or drink it. There are different ways you can get radionuclides into your body and it could be like from drinking milk and drinking water, eating meat. Those are three routes of exposure for getting an uptake of radionuclides into the body, potential routes. So we addressed that in here, that terminology and that's a new terminology for a reg. guide, although it's been used in the industry for a while, for a long while. It just wasn't in the guidance.

And of course the good part of that is once you break it down that way then you can see why people need the samples. You say the rapid exposure is through milk. Well, then you need to sample milk.

The rapid exposure is through meat. Well, then you need to sample meat. And that's the way it's broken down in this document, just a very simple type approach.

This reg. guide also covers how to address spills and leaks with respect to the radiological environmental monitoring programs. It says if you have a spill or a leak, take a look at that. See what impact that has on your program. And if you need to expand your program you need to expand your program. That's basically how it's in here.

And again, that's all I was going to go over for this document. Any questions on Reg. Guide 4.1?

MEMBER BLEY: Well, on the combination, you began telling us about the three tritium episodes. If these reg. guides had been in place and followed, do you think those would have been picked up earlier and not become the problems they became? It seemed that was one of the main driving forces.

MR. CONATSER: I guess the question is could these documents have prevented a leak in the spent fuel pool. No. Probably would not have prevented a leak. The leak probably would have still occurred, right? Indian Point had that crack in their spent fuel pool wall. Would this have prevented that? No.

Now if it was there would they have had additional monitoring capability earlier on? They probably would have.

MEMBER BLEY: I mean some of these went on for awhile before they were identified.

MR. CONATSER: That's correct.

MEMBER BLEY: So you think they would have picked up these, all three of them, earlier on than they did.

MR. CONATSER: The design of this is that if they would have incorporated this guidance they would have had a much better chance of picking up any type of leak in the power plant. The way this guidance is addressed is the plant should look at their site, look at the highest potential leak points and then look at their hydrogeology at the site, look down-gradient it from that and locate monitoring wells in that location and with that type of setup they should be able to catch that. Yes.

MR. GARRY: I think the other thing I would like to add is that the industry has quickly responded to the groundwater leaks and has done the NEI initiative to install onsite groundwater monitoring and the results of that would pick up on the leaks a lot earlier. Our guidance supplements that and kind of helps manage the reporting of those effluence that have leaked onsite so that we have good public disclosure and freedom of information.

MEMBER RYAN: I think in the subcommittee we went into a little more detail on that, Dennis, about the tritium task force results and as you know many power plants around the country have done more detailed geohydrologic study of those issues and I think the reg. guide reflects the need to be a little bit more sophisticated in those kinds of analyses up front rather than waiting for a monitoring result to tell you to do that.

MEMBER SIEBER: It seemed to me that the Braidwood leak was discovered by the state.

MR. GARRY: I think there's some history on the Braidwood that really isn't well-known. From what I've heard in industry meetings, the Braidwood issue was fairly well-known among the technical people at Braidwood and reported to the local media and to the NRC early. But it was like those overnight discoveries where all of a sudden people really became aware of it and when it became aware was when the state reported that they had an offsite well with

1,000 picocuries per liter of tritium. That's when it became a sensation. But it was actually known technically and reported properly prior to that.

MEMBER SIEBER: Will your revised reg. guide enhance that?

MR. GARRY: Yes.

MEMBER SIEBER: It is far better for a licensee to discover his problems than to have an outside agency come in and point them out.

MR. GARRY: Right. And that's where the NEI initiative has really taken the lead and got out a couple years ahead of us on the reg. guides to monitor onsite, establish relationships with the local authorities and work them on that.

MEMBER SIEBER: Okay. Thank you.

MR. GARRY: I think the other thing for perspective purposes, you know, this is radioactive effluence. You know, we release effluence in the air, we the industry, and in the discharge, liquid discharges to the rivers, lakes or to the roof and Appendix I controls that and minimizes that to within what's considered as low as reasonably achievable.

You know I started in this industry in 1973 and groundwater was known as potential leak path. It just doesn't have much for dose significance. It gets diluted in the groundwater and a lot of it gets discharged to the rivers, never causing any public dose to speak of. So groundwater is kind of a new recent issue of the year or issue of the decade, but from a radiation exposure, the Lesson Learned Task Force identified that there's been no public health or safety impact.

MEMBER RYAN: I think the important difference in what's been captured at least from my perspective is it's not driven necessarily by a dose-significance kind of endpoint, but on a detect-and-understand endpoint of what's happening and where are things going before you get to that ultimate question of a dose significance so you can be predictive rather than reactive to those kinds of monitoring results.

MR. GARRY: Yes, we're taking it from an unknown situation to a well-known, quantified and reported situation.

MEMBER RYAN: Exactly.

MR. GARRY: With no dose impact.

MEMBER RYAN: Okay.

MR. CONATSER: And in response to your question there in the Braidwood case if we would have had this guidance out there they would have put this into their annual report which would have been a public document and it would have been out there and then if anybody would have said, "Hey, what have you guys done about that leak" they would say, "Well, didn't you take a look at a public report that was submitted in 1900 whatever it was" and they would have listed that in there and been tracking. So it would have been better communicated. I think that was the big issue at Braidwood.

MEMBER SIEBER: Yes, the issue is a public perception issue in Braidwood and in a lot of cases similar to that because the public really doesn't appreciate --

MEMBER RYAN: And, Jack, I think the interesting point about tritium is that 1,000 picocuries per liter that was identified and 20,000 picocuries per liter is the EPA drinking water standard.

MEMBER SIEBER: Right.

MEMBER RYAN: So the health impact is negligible. I mean it's four millirem per year is the basis for the EPA standard. So 1/20th of four millirem would be the annual impact if that was your only source of water. So it's not necessarily the dose impact that drives concern. It's the identification of something that's previously not recognized.

MEMBER SIEBER: Yes, I just keep thinking about public relations.

MEMBER RYAN: Sure. As we do we all.

MR. GARRY: It's fear of the unknown and we put that out there by having open disclosure, measuring, monitoring, reporting and it's now on the webpage. It's all openly reported about the industry. Voluntarily put that into the effluent reports starting a couple of years ago so that we have better disclosure now.

MEMBER RYAN: All right, Richard. Why don't you press on a little bit?

MR. CONATSER: Okay. Sorry.

How do these documents now fit into the NRC Reg. Guide Update Program? Well, NRC had identified a number of regulatory guides, 400 and some odd, that needed to be revised and they recognized this one and started looking at getting to build new reactors and what guidance needed to be updated and they said, "Okay. Of these, which ones need to be done quickly?"

And they made six different phases. They divided all these into six different phases, Phases 1 through 6. Phase 1 needs to get done pretty quickly just for new reactors. These two reg. guides here go in Phase 3. All Phase 3 reg. guides by schedule are to be done by December of this year. We are right in line with that initiative there to update these reg. guides.

Remember these reg. Guides are 35 years old. So it is time to update the reg. guides we think. And that's all I have to say about that.

Some benefits to revising the reg. guides. Of course, we want to support the NRC's reg. guide update program. We want to meet those dates.

We want to support the Lessons Learned Task Force recommendations, those 26 recommendations and the 14 or so that dealt with these two reg. guides. And we're going to do that.

The two reg. guides are dated. We wanted to update the guidance to make sure it conforms to what licensees are currently doing and the like.

And we wanted to incorporate a lot of the operating experience and lessons learned gained over the last 35 years. And I listed just a few here and I'll discuss these ever so briefly.

The first one that I have mentioned there is the total effect of dose equivalent. That came out with the change in 10 CFR 20 back in 1992-ish time frame, early '90s. I was talking to somebody the other day and they said, "Yeah, New 10 CFR 20 says this." I was thinking "New 10 CFR 20. That's 15 years old." But you know what? These reg. guides are 35 years old. With respect to what was in the existing guidance this was new. 10 CFR 20 was new. We did not incorporate how to meet this TEDE dose limits to the public.

MEMBER RYAN: Richard, we had briefings and understand the update plan that Don Cool and his team are working on to update dosimetry systems. So I think we're pretty well versed on the history of that, what sits in what reg. guide.

MR. CONATSER: Good. I won't belabor that. Only to say that what we wanted to do was update the reg. guide to make sure it did conform with the new 10 CFR 20.

MEMBER RYAN: Right. That's fine.

MEMBER BLEY: And now as those get updated you'll have to --

MEMBER RYAN: You'll have to cycle again, but that's the way it is. At least it's a big step toward what the new one will look like.

MR. CONATSER: And direct radiation, that's another thing that we needed to take a look at from operating experience and lessons learned. Initially when plants were built and they did their designs and they saw what the dose would be for like the shine dose of BWRs that was pretty much what it was when they built the plants. Now we're going through power uprates. That affects the N-16 production will take the sky shine and can potentially affect the direct radiation doses.

Plants are going to like noble metals injection and hydrogen water chemistry and different type of chemistry regimes that may impact the direct radiation off the reactors. They're storing fuel onsite in an independent spent fuel storage installations. All these things are more common now and they can affect direct radiation. The old guidance didn't really have how to address that. The new guidance now does in Rev 2.

MR. GARRY: Let me just radwaste storage, too, storage of heads and generators.

MR. CONATSER: And lots of people are replacing their heads.

MEMBER RYAN: You mentioned in the subcommittee meeting something else that's kind of outside the fence, but land uses around power plants have changed and are updated and may be different. So they're not necessarily as remote as they've been in the place and that's kind of an outside the fence aspect.

MR. CONATSER: Right. People are closer to power plants.

MEMBER RYAN: Right.

MEMBER POWERS: What is a typical average of effluent release from a plant nowadays?

MR. CONATSER: If you look at and break down all the plants into like a quartile or ranking system top quarter, you guys are plenty aware of that type ranking system, probably a top quartile now, there are some plants that have zero release. So zero is the best for liquid effluence now.

For liquid effluence, the first quartile might be like, I don't know, 100 millicuries or less and then --

MEMBER BLEY: Per year?

MR. CONATSER: Per year. One hundred millicuries in a year and that would include everything except for noble gases. It doesn't include noble gases. It doesn't include tritium that don't have a lot of doses.

MEMBER POWERS: I wanted to know both and the quartile is exactly what I'm looking for. I wanted to know it for both airborne and liquid effluent where would I look.

MR. CONATSER: Now for airborne, typically the first quartile is like a curie being discharged in a year, a curie, maybe two curies, ball park range.

MEMBER CORRADINI: This is because of the off gas systems just delaying it and then eventually releasing it when it's within 10 CFR 20.

MR. CONATSER: That's correct. If your off gas containment purges, waste gas to K tanks (phonetic), those types of thing.

MEMBER CORRADINI: Right.

MR. CONATSER: That's absolute tritium.

MEMBER CORRADINI: And again that's excluding tritium. That's noble gasses and gaseous effluence.

MEMBER POWERS: And if I wanted to know the worst one.

MR. CONATSER: I don't know offhand. I think China. Typically there's a trailer. I don't know who it is.

MEMBER CORRADINI: We don't want to know who it is. I think he wants to know --

MEMBER BLEY: Where do you find it?

MR. CONATSER: Oh. The different

rankings.

MEMBER POWERS: Yes.

MR. CONATSER: For gaseous releases, noble gases and gaseous releases probably on the order of I want to 1,000 curies maybe.

MR. BROWN: Yes, Richard. There's an annual report on that that's publicly available on dose.

MR. CONATSER: It is publicly available.

MR. BROWN: Yes, there's an annual report on the website.

MR. CONATSER: From the NRC's website.

MR. BROWN: You can get it to the Committee.

MEMBER CORRADINI: That would be good.

MR. GARRY: Each of the plants submit this annual effluent report and they're all on the NRC website for each of the plants. So anybody can go look at that and take a look at a detailed breakdown of the liquid and gaseous and by category, noble gases, tritium and particulates and iodines and so forth.

MEMBER CORRADINI: That's what I was trying to get on the record.

MR. GARRY: Okay. It's all on the

webpage. You click on -- Go to the NRC public webpage. Click on radiation protection. Go to tritium and you can follow the links to the detailed reports for each one.

MEMBER RYAN: Thank you.

MR. CONATSER: The numbers I gave you were just general numbers. If you want the specifics, go to the website.

Okay. Any other questions before we trudge on?

(No verbal response.)

Good. Okay. To revising the reg. guide, that's where we were at. We were covering this top list right here. We were talking about direct radiation. We also mentioned Carbon-14, that the new Rev 2 discusses Carbon-14. It talks about lower limits of detection. What's the appropriate analytical sensitivity level that licensees should use when they're looking for radioactive effluence?

And what's another benefit of revising these reg. guides? Well, a lot of other groups here like NEI, Nuclear Energy Institute, EPRI, Electric Power Research Institute, and the American Nuclear Insurers, all those groups have issued guidance relative to this groundwater-tritium issue. The NRC had really not issued guidance in the form of a reg. guide necessarily. So there is some benefit there for the NRC to get their position out there so that if a licensee would have an issue they could go to that reg. guide and say, "What would be an acceptable method that the NRC would allow?"

MR. GARRY: I would just like to add something on the LLD to show how we've risk-informed these reg. guides, risk-informed meaning take a look at the most important things and don't spend a lot of time looking at the less important things.

In the old days the guidance was very prescriptive and it said that you had to count your chemistry samples for a long enough time to reach some very low levels for radionuclides that are very hard to detect. By definition, hard to detect radionuclides do not deliver much radiation dose. So why would we force licensees to spend a lot of time looking for something real hard that's hardly there? So we've risk-informed this telling them that they can evaluate and take a look at their principal radionuclides and spend less time on the less important ones, reducing some of their counting times so that they don't have to waste, use, resources. That's how we've risk-informed this.

MEMBER BLEY: I would assume, but I'm not sure that when you do that to the ones that are low if there were any of those that had a very, very long biological half-life that maybe that would elevate the risk importance even if their levels are kind of low. Is that true in how you see that?

MR. GARRY: That would be factored into their dose assessment. If you have a radionuclide and you release it and disperse it in the environment, the modeling will pick that up saying, "All right. If it goes into a person's body and then it's there for a long time then that committed dose over those 50 years gets brought back into the year that the radionuclide was released."

MEMBER RYAN: I think Dr. Bley is asking a little different question and let me jump in. I think that one of the things we covered in the subcommittee was this idea that if it's a biologically -- Let's say it's got a high dose conversion factor.

MEMBER BLEY: Yes.

MEMBER RYAN: That is -- per curie inhaled. You kind of weigh that and at the same time you weigh the detection capability that you have. So you ask yourself, "If I'm having an intake at or below my detection limit, what would be potential dose consequence? If it's very low, I don't need to lower my detection limit. If it's not acceptable, then I do need to lower my detection limit."

MEMBER BLEY: That's what I was asking. Thank you.

MEMBER RYAN: So the detection limits that I -- for is factored in based on the potential dose consequences based on what your detection limit might miss.

MEMBER BLEY: Thanks.

MR. CONATSER: Now all that -- all those factors that you were talking about -- the dose conversion factors aren't located in these documents. That's a supporting document.

MEMBER BLEY: No. I was just asking how you decide how far you want to go and the -- included that.

MR. CONATSER: Okay. I guess the bottom line here is we feel that updated NRC guidance is needed and that's why we're here today to make sure that we would get a good recommendation from ACRS to go ahead and publish these reg. guides.

Some public comments here. I think I have three of these and then we'll open it up for questions. These are some things that go over why we wanted to go ahead and issue the reg. guides, some good reasons for that. We'll cover now some things that we need to make sure that we cover well and we dot all of our i's and cross all of our t's.

One of the issues that came up as a public comment was a backfit. Is this a backfit? They were saying now we needed more analysis of this before we proceed and maybe not issue the reg. guides. But when we looked at this, the reg. guides are really not regulations. They're just staff guidance. They're acceptable methods. One method that may be used to meet the regulation.

With these two reg. guides that we're issuing here, the licensees are still free to use the Rev 1 of the reg. guides if they would so choose. If they want to choose their own method, they could choose their own method. These are not regulations. This is just a suggested method that if the licensee would use, the NRC would have very little questions about the use of this method. We thought there was no backfit so that we could proceed.

There was a public comment and you may have heard this.

MEMBER MAYNARD: I understand that. I wanted to get into that just a little bit because if a licensee chooses to stay with Rev 1 is it going to change the inspection process or are they still going to be dealing with the questions from the newer Rev. I'm trying to figure out how this is going to actually be used out in the field from an inspection and enforcement standpoint.

MR. CONATSER: Yes. We actually haven't changed our inspection procedures yet to incorporate this guidance. We have inspectors look for these types of things.

But keep in mind. This document, this guidance, here even though the NRC is issuing this guidance there is a lot of guidance out there. I said that NEI has issued guidance. EPRI has issued guidance. And the American Nuclear Insurers has also issued guidance. So the industry really has already adopted a lot of this stuff.

MEMBER MAYNARD: Yes, and I understand. This will get into a little bit more of a legal question as well in what's the meaning of this Rev if they don't have to commit to it, if they don't update it. They're still going to have inspectors out there inspecting. Are they going to be looking for the stuff that's in the new Rev or if the licensee just says, "No, we're sticking with Rev 1" are they going to inspect basically to the Rev 1 reg. guide?

MR. BROWN: If I -- That's really an inspection program. So let me take that, Richard. But the answer is the inspectors are directed to look at the licensing basis for the plant and if the plant is committed to Rev 1 that's what they'll inspect against from that perspective.

There are other inspection components which include if the licensee is implementing the NEI voluntary program. Then do a sampling of that to ensure that it's being implemented.

MEMBER MAYNARD: Okay.

MR. GARRY: And the other thing to add to that is that we also inspect to the regulations.

MEMBER MAYNARD: Right.

MR. GARRY: We don't inspect to a reg. guide.

MEMBER SHACK: But just out of curiosity, how many, for example have adopted the NEI guidance rather than the Rev. 1 guidance?

MR. BROWN: I think all of the plants are implementing the NEI voluntary initiative to the best of my knowledge in parallel. They are committed to Rev. 1 of the reg. guide.

MR. GARRY: You have to realize, too, that the groundwater NEI initiative is a subset of this reg. guide. The bulk of this reg. guide is dealing with the gaseous and liquid effluents through normal releases, quantifying, measuring and reporting those.

MEMBER BLEY: I get a little confused between wanting to not have leaks and laws and regulations. So the more I commit to do the harder we're going to look at them, it doesn't seem like those two together generate the long-term least likely chance of thinking. It would seen the less they commit to, somehow the more we would want to inspect. That's why I'm handing a little funny there.

MR. GARRY: Well, maybe if I describe a little bit of the NEI initiative or maybe we can ask NEI to describe the initiative, but it involves a risk assessment where they go through and they take a look at their plant systems, the components.

MEMBER RYAN: Steve, if you don't mind, let's let George do that when he comes up in a few minutes and we can let you guys finish up, but it is a question we can certainly address when we have everybody at the table.

MR. CONATSER: So backfit, we thought there was a lot of backfit. In consistencies though, that's the next big public comment. Public comment is actually mentioned at the ACRS Subcommittee meeting last month as well. It came up. I put it in quotes here because as we go through, I brought four examples of things that were discussed as potential inconsistencies. I wanted to review those real quick just to show you what types of things we would be looking at there.

There was an inconsistency between NUREG-1301, which is a NUREG is just information documents that the NRC puts out. It contains information. Licensees don't have to do anything with it. It's just information that's out there basically, but there was an inconsistency between NUREG-1301 and 10 CFR 50.

NUREG-1301 that was issued in 1991, at that time licensees were still doing semi-annual reports. Both 10 CFR 50 was changed in the mid-'90s to say licensees could go to an annual report. So there was an inconsistency between those two documents.

As it turns out, Reg. Guide 121, being issued in 1974, talked about semi-annual reports in the old revision. The new revision says annual reports. So they brought it up and said, well, now there's going to be an inconsistency between Reg. Guide 121 and NUREG-1301. One is going to say annual reports. One is going to say semi-annual, and that's correct., but the Reg. Guide 121 was revised to meet the regulations 10 CFR 50. We wanted to make it consistent with the regulations.

It does point out that we need to take a look at some of the older documents out there, some of the older NUREGs, but remember those are just information, but we are taking a look to see which of those we do need to update as well.

So that was an example of an inconsistency. Another example was 10 CFR 20 and 10 CFR 50, and as Mike said, you guys have already discussed that, the TEDE, the total effective dose equivalent concept versus whole body dose concepts, and 10 CFR 20 talks about TEDE. Ten CFR 50 includes a concept of whole body doses. They are different concepts. They are both measures for protection of the public, and they both can work well that way.

One is an older concept; one is a somewhat newer concept. ICRP-103 will be the newest concept once they get that flushed out, but what we did with this issuance of the Rev. 2 of this Reg. Guide 121, we wanted to make sure that we had guidance in there that talked about how licensees can comply with the 10 CFR 20 TEDE and how they can comply with the 10 CFR 50 whole body dose.

So we wanted to update this to make it consistent with everything, even though there may be some internal difference between 10 CFR 20 and 10 CFR 50. As Mike said, that's being addressed by a SECY 08-197, I believe.

MEMBER RYAN: Correct.

MR. CONATSER: Okay. Another example of an inconsistency, NUREG-1301 and Reg. Guide 121. I think we already covered this. I'll be quick on this one.

The NUREG-1301 was silent of Carbon-14, didn't mention it. Reg. Guide 121 explicitly mentions Carbon-14 and says the licensee should report that, and they said, well, that's an inconsistency. It's different.

And actually that is different. That's something we've learned over the last 35 years, and now we have the ICRP publication, the International Council on Radiation Protection. Their Publication 81 talks about Carbon-14 and the environment. Much more is known about that now. That came out -- I forget what year that was now, but anyway, it's been out for many years now, and also IAEA has come out with their Publication 421 that talks about Carbon-14, ways you can analyze for it, sensitivity levels and those types of things.

So now plants should be taking a look at Carbon-14, and we've included that in this guidance, and we think that's appropriate.

And the last item is NUREG-0543 and Reg. Guide 121. They said there was an inconsistency there. The NUREG-0543 talks about calculating the Environmental Protection Agency's 40 CFR 190 dose. That's what they call a total dose, and it's total dose meaning all of your effluents, all of your direct radiation from outside storage tanks, et cetera, et cetera, a total dose.

Well, NUREG-0543 when it was issued years ago, it talked to how to address that. Reg. Guide 121 talks additionally how to address that. They both say as long as there's no significant direct radiation component, then if you comply with 10 CFR 50, Appendix I, you're good to go on the ETA dose on it.

But both documents do say if there's significant direct radiation dose at your site, you need to take a look at that to make sure you're in compliance with the EPA 40 CFR 190.

So although people had said now there's some inconsistencies there, actually they both said exactly the same thing. So I guess the point of this slide here is just to say that there are inconsistencies between other documents perhaps, some other older documents, but what we've tried to do and what we have done in this reg. guide is to issue guidance that is up to date and consistent with the regulations.

MEMBER SHACK: And just coming back to this backfit again and the Carbon-14, I mean, as I understand now you would expect them to monitor Carbon-14. Because it's not in Rev. 1, your argument is that it's really covered by the regulations.

MR. CONATSER: That's correct.

MEMBER SHACK: They just didn't identify it. So you, in fact, do expect them to monitor to Carbon-14 even if their licensing basis is Rev. 1.

MR. CONATSER: That's correct., and some plants are doing this. Indian Point, there are some plants that are monitoring Carbon-14 and reporting in annual reports.

MR. GARRY: Carbon-14 did not used to be a significant component back in the old days. The plants have done a good job with improving or reducing failed fuel to where the effluents have gone down and Carbon-14 is now a bigger contributor than what it used to be on a percentage basis.

MEMBER CORRADINI: So what you're saying is as everything else falls, other things rise up to be watched.

MR. CONATSER: That's right. Principles.

PARTICIPANT: It's called "principles" in this document.

MR. CONATSER: Not that they're dose significant, but that they are the bigger contributors.

MEMBER RYAN: And I think that's a good point in this case. Carbon-14, you know, you can deal with as an indicator of the performance of the plant from an effluent standpoint because you reduce the other effluents to where you don't have those leading indicators as crisp and clear as you once did. So it's not that it's dose significant, but as a larger fraction of the released radionuclide inventory it's perhaps a useful leading indicator.

MEMBER SHACK: The only thing that there is to get all effluents to the same level.

MEMBER CORRADINI: From a dose standpoint.

MEMBER POWER: The Carbon-14 just really bothers me because that means all of the trash contributes to the effluent, you know, the paper, stuff like that.

MR. CONATSER: There is a plant-related, nuclear produced type Carbon-14 that's --

MEMBER POWER: In the way it functions, the Carbon-14 in scrap paper and then coming out of the plant are identical.

MEMBER CORRADINI: But just to clarify, is Dana correct then? Has it gotten to a level that anything that enters the plant which is unmonitored but leaves the plant that is monitored is reported?

MR. CONATSER: No.

MR. GARRY: Plant related. It has got to be plant produced and related.

MEMBER CORRADINI: Plant produced.

MR. GARRY: Carbon-14, there's like 100 times more of it out there from natural --

MEMBER CORRADINI: Yeah, I'm with you.

MR. GARRY: Okay.

MEMBER CORRADINI: I just wanted to make sure I didn't misunderstand.

MEMBER MAYNARD: I'm concerned a little bit as we keep driving these things down something else may become a percentage contributor that is still not significant, and are we going to keep adding things as we keep reducing others?

At some point we have to say enough is enough on what new things we want to monitor, unless they provide for significant contributions.

MR. GARRY: Appendix I addresses that to a certain extent in that it defines an ALARA level, saying if your rad waste systems are operating sufficient to keep the doses below these numbers, then you by definition have met the design objectives, and the rad waste systems are operating properly.

But from a reporting perspective, we require that the effluents, the main effluents that are released be reported.

MEMBER MAYNARD: Okay, but to report them, you're going to have to monitor and measure them.

MR. CONATSER: Remember on the risk-informed concept for their analytical sensitivities, the lower limits of detection, we're allowing the licensees to say, "Hey, you know, we've got this nuclide out there. It's a very low contributor, low dose contributor, not a significant nuclide."

Well, you know what? We recognized that the dose significance there is not great. So they can pick a sensitivity level that they can analyze to that covers the safety aspects, but still allows them to report it appropriately if it is present in quantities that need to be reported.

Does that make sense?

MEMBER MAYNARD: But it still sounds like it's still another item that they have to track and measure.

MR. CONATSER: Yeah, the bottom line is that may not be any significant contributor at all. We should be interested in making sure that we are protecting health and safety of the public.

MEMBER RYAN: I took the meaning of Rich's statement to mean that you can set a detection limit of the analysis at such a level that you know you're protecting public health and safety, and you don't have to chase it down to the last ten atoms of Carbon-14, but you can set a detection level that's straightforward and easy to measure and meets the objective of demonstrated safety.

MR. CONATSER: As the regulation is that the licensees are not likely to underestimate the doses, and that's what we're striving for in this document. We want to say, okay, at this level you're not significantly underestimating the doses, and therefore, you're okay at that point.

MEMBER MAYNARD: And I understand that, but it still seemed like it's another column in the database that you've got it tracked. I mean, I didn't hear where you can justify not even measuring it and monitoring it. So it is another essential detectable limits different maybe, but it's still an additional item to track and monitor that may not be giving you any real meaningful information.

MR. CONATSER: We do say if we like to see it detected in their effluents, that they need to report it, and that's just for disclosure to the public. The whole thing with the tritium in the environment, et cetera, we want to make sure there's good disclosure there. If you detect it, we want you to report it. If there's not a safety significant issue and you detect it, then what's the down side to reporting it when you've already detected it, right?

MEMBER MAYNARD: I'm just trying to figure out where it stops. As we keep lowering other things, what additional things may pop into there that becomes a percentage contributor that's really not any meaningful contribution to health.

We can't monitor every possible item.

MEMBER POWER: Otto, I think since many of the plants are cutting holes in the concrete that we ought to require them to monitor the Potassium-39 here as well.

MR. SCHAFFER: Steve Schaffer from NRO.

From our perspective, when we do our predictive dose calculations for new plants, Carbon-14 turns out to be the major contributor to the gaseous doses. Now, over 50 percent of the gaseous pathway doses come from Carbon-14. So from a new plant perspective, it is an important risk radionuclide.

MEMBER POWER: Thank you.

MR. CONATSER: Okay. Any other comments on that?

The last public comment we got, we got input that maybe NRC wants to delay publication of these reg. guides. This possibility was discussed at the HP Subcommittee meeting, and a couple of items that were brought up in this regard. They said, "Well, you've got this International Council on Radiation Protection, the 013 dose methodology pending with this SECY. Shouldn't you just wait for this to come out to revise these regulations and just hold off on it?"

Well, this actually maybe years out to get this guidance implemented. We, the NRC staff, thought it important to go ahead and issue guidance so that the licensees could have guidance out there should they come across issues in between now and ICRP-103 gets addressed.

So we think it's important to get the guidance out there so that we don't think this is a big issue for delaying the publication of the reg. guide.

Additionally, some input was that, you know what, plants aren't required to commit to this. So why even issue this reg. guide?

Well,t hat's applicable to all reg. guides. All reg. guides are just guidance. Plants don't need to commit to them. This is just one acceptable method that the NRC finds acceptable. So we don't think that's a good reason to hold up and delay publication of the reg. guides.

So the staff recommendation is that we issue the reg. guides consistent with the NRC's reg. guide update initiative.

MEMBER SIEBER: The agency has to endorse ICRP-103 anyway, right?

PARTICIPANT: Right.

MEMBER RYAN: If I may just on the point back up, please, Richard, the ICRP updated that is in the reg. guide -- no, no, no, just that way; yeah, right there -- is to ICRP-26 and 30. That's a major improvement that looks like 90 percent of where 103 will take it as opposed to zero percent.

So I think there is credit that needs to be recognized for improving the dose system that's consistent with what they have to do for internal exposures for workers and everything they've got to do now, you know, and ask to do now. I mean, licensees now have the ability to write a letter and say, "You ought to use ICRP-26 and 30 dose methodologies. May I?" and the answer back is, "Yes, please do."

So it's like a 70, 80 percent step toward where 103 ultimately will be.

MEMBER SIEBER: But 103 or its predecessors is not a regulation until the agency says it is.

MEMBER RYAN: Right, but 26 and 30 are.

MEMBER SIEBER: yes. And so what we're doing is consistent with what the agency is officially --

MEMBER RYAN: Right, as opposed to being our date with stuff that came out in 1956 or '59.

MEMBER SIEBER: Well, it was good in '59.

MR. CONATSER: It still provides adequate protection.

MEMBER RYAN: Okay. Thank you.

MR. CONATSER: And then I was going to

open it up for questions. That's all I had for the text. Any questions?

MEMBER RYAN: Before we get into it, if we may, maybe we could hear from NEI and hear their presentation and then have one round of questions at the end.

Mr. Oliver, please.

Mr. Oliver from the Nuclear Energy Institute was kind enough to participate in the subcommittee meeting. So we're happy to have him back here for the full Committee briefing.

Thanks for meeting with us.

MR. OLIVER: Many of the issues and discussion points I'll bring up were presented at the earlier subcommittee, and we very much appreciate being here today. Steve and Richard have already mentioned that the comments that we're providing are on an informed basis, go back to the public comment period. We do not have access to the documents that you have access to, the Commission staff and the Advisory Committee have access to. So a number of the issues that I'll be discussing relate to those earlier versions and have likely been addressed by the staff.

We've had a rather robust effort inside the industry where some 30-plus individuals who we've taken into account. There are only 57 operating sites. So we had a very high percentage of the industry represented in the comments, and they were very detailed comments that we provided, and I think many of them well received by Commission staff.

But actually the comments were longer than the regulatory guides that we're commenting on. So they were that extensive.

We spent numerous hours with the staff sharing comments with the staff that we had received from the industry, many of them in raw form, prior to them being distilled. I also would note that the workshop on January 15th, staff was very open and accommodating to the discussion that was being offered. I mean, they did their job as staff and so forth, but they were very open and professional about their approach to things, not only in that but to the entire process.

With the emergence of SECY 08-0197, the process on revising these reg. guides started well before the SECY document that we've been talking about came forward, and that's really the process to overhaul or assess the overhaul of the NRC's regulations for protection against radiation and advise. That's going to be a multi-year process.

What we're finding as we start looking at these things is we find we've got one process that we're going to be going through with the review of the SECY document, and then we have the regulatory guide revisions that we're dealing with here. It's not so much that I don't believe the current position is that we want to absolutely stop this reg. guide, but what we would like to do is to see consistency in the process, that some integration in the approach being taken by Commission staff would be well advised at this point.

And supporting points to that, what I would say that would provide for efficient utilization of Commission resources as well as the industry from an implementation standpoint. We don't want to be going through significant revisions to our programs. The inspector and enforcement staff, headquarters staff, and the industry people, it makes sense to do this in a coordinated fashion.

And I think Steve and Richard did an excellent job in identifying some of the inconsistencies. I would not be overly put off with the fact that there are duplication of inconsistencies. Change is needed, but you have to start somewhere in doing this. What we've got is a system of interlocking guidance documents, informational documents, et cetera, and they're all linked together to form a fabric that the licensees use to meet the regulations of the Commission.

But you do have to start somewhere, and inconsistency can't necessarily be avoided at this stage. But I would recognize, too, that at SECY 08-0197 it really offers an opportunity to provide consistency as we go forward. And I think Steve and Richard did a good job of, you know, showing you where some of the inconsistencies lie between 10 CFR 20, 10 CFR 50.

It would be wonderful to have all of this stuff come into alignment at some point, but all of the regulations are based on the same ICRP, and that we have a consistent set of guidance across the board, but I think this may be probably a once in a lifetime opportunity to accomplish some of that.

There's a lot of guidance out there if you would go back and -- thank you. The correct slide -- there's a lot of guidance out there on groundwater as we discussed. You know, we've talked about ANI guidance. In the industry that's all linked together by one common point. NEI is the focal point for all of that guidance, ANI as well as EPRI. So the industry has managed to gain some alignment in terms of the groundwater guidance, but what we're looking at at the Commission is we have guidance in decommissioning planning. We have guidance of new plants which we're just in the process of having OGC and New Reactors review the template, NEI-0808, which is basically groundwater monitoring and minimization of contamination to meet Reg. Guide 4.21, right?

What we have is a series of groundwater related efforts inside the Commission. It would be nice to have some of that come into similar alignment for us because there is multiplicity of guidance there. And I'm not saying that we have inconsistencies, but we need to monitor for these inconsistencies as we go through the processes at the Commission.

Steve has already mentioned that the existing guidance should remain applicable, and I think that is an important point. It does get past a number of thorny issues that Steve outlined.

The licensing basis for the plants, what the plants can actually be held to, the licensing basis is largely not impacted through the issuance of the reg. guides as long as Revision 1 still remains applicable.

The reg. guides as part of the licensing basis would probably only be applicable to a few plant that have yet to file applications for COL. I think some of the plants will probably start using some of the additional flexibility that's in this new reg. guide because of some of the flexibility that's discussed. I think that's probably an inevitable process.

Additional issues. Clarification of solid radioactive waste. That was a part of the presentation you heard from NRC. This has not been a very precise point in terms of the existing reg. guides for reporting. It's when you flip something off site. Is it for disposal or is it equipment being sent out for refurbishment or is it being sent out for waste treatment compaction or sorting or some other process?

And Steve tells me that the reg. guides have been revised to make this much more clear because what we had out there in reality is inconsistent reporting around the industry, and this would solve some of that problem.

The elimination of the on-site radiological programs from DG-4014. What we have is during implementation in power plants, what we have is generally the reg. guides are written to apply to program areas. The RMP program would largely be the impacted program by this, and if you had the on-site radiological monitoring, then you start getting overlaps developed with the plant health physics program protection of the occupational radiation, et cetera.

So I think this is a well informed and appropriate thing to do in the reg. guides.

I found the discussions on C-14 somewhat similar to some of those that we've had inside the industry earlier, but there is a piece of flexibility in the reg. guides I find useful. Monitoring the C-14 is a somewhat difficult process. The reg. guide allows for the computation of C-14 based on effective power of the reactor over an extended period of time; allows us to basically ratio it because it is pretty much a constant function of reactor power, and I think this is a piece of flexibility I would commend and I understand that remains in the guidance document.

There is flexibility in there such as the LLDs that Steve mentioned as well, and we appreciate those as well.

VICE CHAIRMAN ABDEL-KHALIK: Is that relationship between reactor power and C-14 releases empirically based?

MR. OLIVER: Yes, yes.

VICE CHAIRMAN ABDEL-KHALIK: So you would have to collect data for a certain period of time.

MR. OLIVER: -- of data out there that supports that, a number of papers out there that --

VICE CHAIRMAN ABDEL-KHALIK: What I meant was is this a totally a priori calculation? Someone with a clean sheet of paper can start off and tell you this or do they have to rely on prior data to come up with that relationship>

MR. OLIVER: It's strictly a function of reactor power, and it's observation based.

MEMBER MAYNARD: But does each plant have to do a baseline or is it you don't even have to do anything? You can just calculate from your --

MR. GARRY: Use the NCRP, National Council of Radiation Protection, report on Carbon-14 and scale to that.

PARTICIPANT: And there is a handbook out for it.

MR. OLIVER: And there are a few plants who -- the ones that do report C-14, this is largely the way that --

MEMBER CORRADINI: But just answer Said's question. The way you're answering it it's empirical.

MR. OLIVER: Yes.

MEMBER RAY: And not plant specific in that plant.

MR. OLIVER: Right. Not plant specific. That is correct.

MR. CONATSER: Now, we do say in the document that if one would choose to sample and analyze for it, we can they can do that as well.

MR. OLIVER: And my comments are largely based on my understanding of what the document says. I have not actually seen the document.

VICE CHAIRMAN ABDEL-KHALIK: This actually assumes that the only source of Carbon-14 is activation of Carbon-13 in the containment atmosphere.

MR. OLIVER: Oxygen-17, too. Oxygen-17 is significant.

MR. GARRY: Most of it actually starts in the liquid, in the RCS.

VICE CHAIRMAN ABDEL-KHALIK: Oh, I see.

MR. GARRY: And then permeates out as carbon dioxide.

VICE CHAIRMAN ABDEL-KHALIK: Okay.

MR. GARRY: So most of it is -- so the method that we've provided is simply to use the NCRP report that says that basically PWRs in general at power levels we have in the United States are going to generate something like seven curies or is it 11 Curies for PWRs and seven Curies for BWRs. I can't remember which is which.

And all they have to do is scale to power production. So you've got so many megawatt hours scaled to it and reported. Now we've got public reporting. We're open.

MR. OLIVER: I would point out one thing on C-14. Power releases of C-14 are virtually impossible to see in the environment because it's such a minuscule fraction of the C-14 that's already out there, as you've already noted.

MR. GARRY: We have a specific exclusion for monitoring for Carbon-14 in the environmental monitoring program.

VICE CHAIRMAN ABDEL-KHALIK: Okay.

MEMBER SIEBER: You can find C-14. It's just you can't pick out which is yours.

MEMBER RYAN: Is that the end of your comments?

MEMBER RAY: Is this question time now?

MEMBER RYAN: Question time.

MEMBER RAY: Is there anywhere in the

guidance that indicates what pathways should be monitored? I noted there was a reference to exposure pathways here on Slide --

MR. CONATSER: In the environmental monitoring program?

MEMBER RAY: Right.

MR. CONATSER: And this would be monitoring environment around the power plants.

MEMBER RAY: Correct.

MR. CONATSER: In the unrestricted areas. We tell them yes. You should be taking a look. There's a requirement to do what they call a land use census each year. You go out and see how the land is being used. You see if there's cattle in the area. You see if there's milk animals in the area.

MEMBER RAY: I'm familiar with that. I'm thinking more you could argue that, well, no, I didn't anticipate a spent fuel pool liner would leak, but you clearly should anticipate that a valve is going to leak. That happens to create a pathway to the environment.

And the question is is there any guidance about monitoring such leakage.

MR. CONATSER: Now, that would be leakage from a plant system now, and that would be leakage you're to detect on site.

MEMBER RAY: No, not necessarily. One went two miles out to the river. They had valves all along the way. I'm just asking is there any guidance to monitor for leakage from valves -- I'm choosing valves here -- anticipated pathways on a -- misjudgment.

MR. CONATSER: The way this would work is -- and this is, remember, in conjunction with the NEI voluntary initiative -- licensees are implementing other things in addition to this guidance that the NRC has.

MEMBER RAY: The answer is going to be no, is it?

MR. CONATSER: What we said here is to monitor the groundwater wells. If you suspect you have a leak, then you need to be monitoring your groundwater wells to see if there's any transport of radionuclides through the ground in your area.

MEMBER RAY: I don't mean to be arbitrary. I just was going to ask you a simple question, which was if you have an anticipated pathway like a valve, and I use break away as an example, there's no guidance. This is in terms of domain of requirements. There's no guidance here to monitor.

MR. CONATSER: Well, yes, there is guidance. If you find like a leak on a vacuum breaker, like at Brightwood, they would directed by this guidance to go and sample and analyze.

MEMBER RAY: But only after you find the leak.

MR. CONATSER: That's correct.

MEMBER RAY: That seems kind of odd to me, but okay.

MR. GARRY: Well, the guidance identifies, I mean, specifies that you should look at your expected or principled release points and that you should monitor those, and you should be aware of the other ones and be ready to monitor in case you need to.

MEMBER RAY: Okay. I think that's going to the direction that I'm asking about.

MS. GARRY: So specifically to Braitwood Knoll (phonetic), there would be no requirement to go out and monitor for those vacuum breakers, okay, but there would be an expectation that they would have evaluated that they have a two mile line running out to the river and they need to know how many valves are on the path on the way out, and they need to be aware of that.

And if they expect or have a release, that they're ready to go monitor. But proactively, no, they don't have to set up monitoring at each of those valve locations unless they say, "Hey, they leak all the time and we need to go monitor those." Then they would need to.

MEMBER RAY: Somebody might not unreasonably say it if that's the case, but okay.

MR. OLIVER: Mike, is this the right time to discuss the elements of 0707? I think some of them relate to --

MEMBER RYAN: You should mention that to help answer Hal's question, I think, yeah.

MR. OLIVER: Oh, seven, oh, seven, or the groundwater protection, 0707 is our position paper for the groundwater initiative. There are elements of risk assessment that includes design, material condition of the systems, analysis of pathway to the environment. It also includes work practices, et cetera, particularly if you're carrying on a radiological process outside where there's high risk involved, and a lot of this centers around what's the defense in depth, you know. Can a single failure cause a leak or a spill, single pipe rupture, et cetera, and that's part of the risk assessment.

How much risk are you taking? What's the concentration of the fluids that would leak should you have it?

So you weigh all of that, and there was a whole section on hydrogeology going back and reconstructing the hydrogeology for the site and their ongoing requirements for both the leaks; that you look at the risk assessment, the hydrogeology. The wells are to be located based on the hydrogeology so as to form basically a sentinel well where the well is strategically located down gradient from the systems that might leak.

So if you have a leak that you're not aware of, then you catch it through the monitoring, but the idea is to prevent first and then if you have a failure.

So then there's other portions of it that deal with public disclosure, transparency, how we do public communications, and there is also oversight with your self-assessments, independent peer assessments, not done by the utility. So it's an independent --

MEMBER RAY: I think that goes to the question. It just seemed very odd to me that anybody would assume a vacuum breaker on a discharge line like this wasn't going to leak and, therefore, it didn't need to be monitored. It sounds like you would pick that up in the process you just described.

MR. OLIVER: Yes, yes, we should.

MEMBER RYAN: And I think you probably know as much or more about this, Hal, that there's been a pretty substantial effort that is derived for the tritium questions, detailed hydrogeologic modeling and starting with the '70s more global recommendations on these issues.

MEMBER RAY: Yeah, I am. I just like to have simple solutions which would in this case would be that's a release point. I don't care what you say about it, and you have it monitored.

MEMBER RYAN: The rule of geohydrology is let's drill one more well.

MEMBER RAY: No, I'm talking about just treat it as a release point even if you hope it doesn't discharge a lot.

MR. GARRY: We do not require that every potential release point be monitored. That would be very burdensome on the utility and the industry to try to monitor every potential release point.

So what we require is that they evaluate all of their potential release points, determine which ones are likely significant, the problem ones, and monitor those. They're the ones they need to be aware of. They need to have it in their off-site, those calculation manuals, and be ready to go monitor basically.

MEMBER RAY: All right. Okay. Well, that's better than it was anyway.

MEMBER RYAN: Any other questions?

Again, I'd like to thank the NRC staff, thank NEI, and we appreciate your presentation. And with that, Mr. Chairman, I'll turn it back to you.

CHAIRMAN BONACA: Thank you very much for your presentation. Very informative.

We have 40 minutes before the break, and so I think we're going to -- there are two options. One is go off the record and look at the letter on NIST.

MEMBER SIEBER: Yeah, Sam was looking at it. I don't know if he's finished or not.

CHAIRMAN BONACA: It doesn't matter. He said to me that --

MEMBER SIEBER: Let me see what's on the G drive.

CHAIRMAN BONACA: We can go off the record now.

(Whereupon, at 11:18 a.m., the meeting was recessed for lunch, to reconvene at 1:00 p.m.)

AFTERNOON SESSION

(12:58 p.m.)

CHAIRMAN BONACA: Let's get back into session.

The next item on the agenda is pellet-clad interaction failures under extended power uprate conditions. It is a concern that has been raised by a member of the ACRS, and Dr. Armijo is going to lead us with a presentation.

MEMBER ARMIJO: Okay. Thank you, Mr. Chairman.

One of the things I'd like to let every one, I believe we have TVA personnel on the phone, a bridge line to listen in on open portions of this session. So to preclude interruption of the meeting, the phone line will be placed on the listen in mode. When we get to the question and answer period, we would, of course, be happy to take nay questions from people on the phone.

We have a very tight agenda, and in order to try and make the meeting move along, I decided to open with some detailed remarks to get everybody on the same page. I'll be talking about CLI. Many of you are experts in this area. Many of the members are not. So I want to get everybody on the same page. So if I'm covering material you already know, please accept my apologies, but I think we need to do this.

This meeting, the objective of this meeting is really to assess the issue of PCI, how much of a problem it is or can be in AOOs, particularly for cores operating at higher powers.

In December of 2007, the ACRS full Committee made a recommendation to the staff as shown in the yellow on that slide as part of the Susquehanna extended power uprate review. Susquehanna was the first plant that was using what I call conventional fuel or non-PCI resistent fuel in its cores for the power uprate, and we recommended, the committee recommended that the staff should develop capability to perform a thorough review and assessment of this risk and as shown on this slide.

Our Subcommittee on Materials, Metallurgy and Reactor Fuels met in March of this year. We discussed that with industry and the staff, and so this is a follow-up for the full Committee.

Next chart.

We're going to cover a number of items. The first thing is background and why are we concerned so that everybody understands that.

The PCI basics because we will be using

some terminology that may not be familiar to everybody, I'd like to get those out of the way pretty quick.

Some standard parameters that people who have worked in this area use related to PCI failure powers, failure strains, and times to failures. The latter two are very important because some key regulatory assumptions are made that the appropriate control or limitation on failure strain would provide protection for conventional fuel.

The other assumption is that there is sufficient time or adequate time to terminate a transient that might cause PCI, but to terminate it by operator action, and so there would be no need for a PCI resistent fuel design in that case.

Then I'll wrap up with some conclusions and recommendations, which at this stage are my conclusions and recommendations and not necessarily those of any other member.

Next chart.

This background chart is as condensed as I could possibly make it, and it reflects how we got to where we are today. During the late '70s and early '80s, both the NRC and the nuclear industry fuel manufacturers in particular were working on the problem of pellet cladding or action fuel failures, to understand it, to control it, and obviously with the long-term goal of preventing it.

NRC at that point, I circulated to the Committee a report by Michael Tokar, a 1979 report that was prepared for the ACRS on the subject of PCI and what was known at that time, and basically what Tokar concluded, that it was really time to introduce specific regulatory analysis requirement for PCI into the plant safety analyses.

Now, that was his thoughts in 1979. Now, time went on, and vendors were notified it was coming. I happened to be a member of a vendor organization, and I was working on this problem. So we were well aware that the NRC was interested in regulating PCI and not necessarily so much for normal operation, but for the case of transience where the risk was perceived to be significant.

Well, what happened was during the '70s and '80s the industry did basically solve the problem. Operating recommendations were introduced which controlled the risk of PCI fuel failures. They were known as preconditioning interim operating management recommendation. They were very effective.

Fuel manufacturers decided to quit using or actually started seven-by-seven fuel assemblies operating at peak powers of 18 kilowatts a foot. They systematically reduced them to lower powers, 13.4, by increasing the number of fuel rods to nine-by-nine rods and ultimately ten-by-ten, and the LHGRs just kept getting lower and lower, and since PCI is a power driven mechanism to lower the peak power, to lower the risk.

In addition, a PCI resistent fuel design was licensed and placed into commercial service. That first design was a zirconium liner design introduced in reload quantities in 1982-83, and demonstrated resistance in '84 and '85 by in reactor testing.

So the PCI was really brought under control. In addition, the NRC and pretty much the whole industry had this assumption that existing thermal mechanical licensing criteria that were designed to protect the fuel from departure for failure during DNB or fuel center line melting, that those licensing limits or criteria would also protect you against PCI during these abnormal operating occurrences.

The other assumption was that the transients were really way too fast to cause any real problem since PCI would probably take more time and an operator could terminate the transient before that time had elapsed.

So that was the situation in the mid-'80s, by the mid-'80s, and really the problem had been solved. There really was no incentive as far as PCI specific regulatory changes. That was the view of the NRC, and it has persisted to this day.

But things have changed, and that's the next slide. Why is there a concern?

Well, first of all, margins that were introduced in the 1980s are disappearing. The people in your heat generations of today's modern ten-by-ten fuel, for example, are back up to the 13.4 kilowatts per foot of the old eight-by-eights. So that conservatism has disappeared.

We're putting more fuel into service at high power at EPU. At EPU we don't change the peak LHGRs, but we put more fuel in the core to operating at those peak LHGRs.

Butt hose things are tolerable during normal operation with conventional fuel because these preconditioning rules are still very, very effective. they're difficult. They're complicated, and in some cases pretty expensive, but they work.

In the AOO regime, which I'll talk about later, you cannot protect yourself with these remedies. The protection afforded by the PCI resistent fuel design, however, is only there if you use that fuel, and there's growing use of this non-PCI resistant fuel in these courts, and that's a concern.

And the other two things, and I'll show data during this presentation that the PCI failure strain is much, much lower than the one percent strain acceptance criteria in the regulations. So while the regulations would sort of assume that fuel won't fail unless it gets strained up to one percent, well, the data show that's not correct.

The other data that I myself and staff and our consultant found is that the PCI failure times at high powers typical of AOOs are very, very short indeed, and os can you rely on prompt operator action to terminate the problem and keep fuel from failing in large quantities.

This chart is just an introductory to get some terms and concepts in your mind. The actual vertical scale is power in kilowatts per foot. The horizontal is the burn-up, and the yellow and the orange colors represent safe operating regimes of power and burn-up for BWR fuel and PCI risk areas. As you can see, at low powers, you really can do anything you want with the fuel.

You have no operating restrictions and no risk of PCI, but there is a threshold power and it's defined by that diffused band between the yellow and the orange. There's a threshold above which you can fail fuel if you change power very quickly to a power that you hadn't operated at before.

The green or yellowish-green line shows the licensed operating limit for the fuel typically for BWR fuel. That's 13.4 kilowatts per foot up to a certain burn-up and then decreasing with burn-up.

Now, above that I've put a dotted line to represent the AOO range. This is the power levels that you can achieve during some of these operating transients, and it's of particular concern because the higher you go in power, the greater the risk of PCI, and I'll show you quantitative data of how that works.

So the issue is: how can we protect the fuel during AOOs? And are we doing it?

So what happens when you change the power in a fuel rod? And I'll go through this quickly. What happens is the pellets are cracked. They expand an outward fashion. They fan axially as well as radially. So you get a biaxial stress at the pellet-pellet interfaces pressing on the zirconium alloy cladding.

At the same time, from the high inner part of the fuel, fission products are being released and deposited on the inner cladding, and you have aggressive chemistry and a very severe stress state.

So what happens is shown here. This is a picture from a hot cell. This is some work done at G.E. years ago, and it's one of the best resolution pictures of typical PCI cracking. If you don't have good vision, you couldn't see it. There it is. It's that fine, little, tight, tight axial crack. It happens to be right over pellet-pellet interfaces, and you can see the rod is not deformed. You can't be quantitative, but this rod was measured, and the plastic strain after the test was much, much less than one percent. It was probably closer to .1 percent, but just to put that in perspective.

And so that's what it looks like. It's tight axial cracks of very short length. In cross-section, this is a picture at high magnification. The lower part, the dark gray is the fuel pellet with a crack in it, and the light area is the zirconium alloy cladding with a branching crack penetrating about a third of the way through the wall.

If there were stainless steel people in

here or nickel base alloy people that worked on stress growers and cracking, this is the fingerprint or the poster child of a stress corrosion crack, and the industry is totally in agreement that PSI is a stress corrosion mechanism.

So this doesn't look like much strain, but can you measure it, and the answer is yes, and I'd like to show you what the strains look like.

Next slide.

These are data from two independent power ramp programs, one run by G.E., one run by an international program which I'll get into a little bit later. But what was measured in this chart is the failure strain. That's the change in diameter after the ramp test, and the fuel was taken from a lower tower, typically let's say around eight kilowatts a foot up to a failure power, and in this case the peak powers are about 16 kilowatts a foot, and these tests were done, as I say, totally independent. G.E. rods were tested under a G.E., government, Commonwealth Edison program, but the fact is all of these rods failed, but if you look at the strains, the maximum strain at failure was about .2 percent, and that's measured with very high quality prophylometry.

the other test program was done with KWU rods, and for those who don't know who KWU is, it's Kraftwerk Union, which later became Siemens. Siemens bought Exxon Nuclear in the United States. Exxon Nuclear got somehow absorbed into the Framatome. So it now really represents Areva. Okay? So it is relevant.

There was differences in the way the cladding was manufactured and heat treated, but what was remarkable is that the strains were very consistent between these two test programs. So they were tested in the same test reactor. All of these rods had been incubated in power reactors. So these weren't laboratory curiosities. These were real BWR fuel rods.

So these data, this is the first time I've ever -- you know, I plotted this data up from these various reports just to see what we could find,a nd it's clear that the failure strains are much lower than the one percent acceptance criteria.

VICE CHAIRMAN ABDEL-KHALIK: Now, were the ramp rates in all of these tests the same?

MEMBER ARMIJO: Yes. Yes, they were all aggressive ramp rates. The G.E.s tended to be a little more aggressive, but yes, they were all ramped in the same way, in the same test reactor in very similar test programs.

The only difference in the KWU test is that they were interested in time to failure, and so all the blue triangles, all the triangles, the tests were interrupted before the rod failed. You deliberately terminated. In the case of G.E., the rods were kept at power until they failed. Every one of those blue triangles has an incipient crack, partial crack through the cladding as much as 60 percent of the way through.

So even though they didn't fail, that was not their primary objective. Their primary objective said how long did it take before this damage occurs, and I'll show you the data shortly.

Okay. So what is PCI? Well, it's stress corrosion cracking. All you need is a sufficient stress, in this case driven by the thermal expansion of the fuel, the heating of the fuel. You have a susceptible material which is irradiated zircoloy cladding. You have an aggressive chemistry coming from the fission products coming out of the field.

And when those three, all three requirements are met, you have PCI. The point of this chart, and I'll get off of it, is this is a classic PCI description or the Venn diagram, and you'll notice there's not one mention of strain. It's not a strain corrosion. It's a stress corrosion cracking problem. I just want to make that point.

So let's get quantitative, and here's data from a very large G.E. test program that was done to develop a reference base to compare PCI resistant fuel to PCI susceptible fuel. All of the red data points are power ramp tests of fuel rods that were operated first in the power plant, then shipped to a test reactor for testing.

The open circles are rods that were tested in the same way but didn't fail, and then the light blue lines represent the percentiles of the statistical analysis of those data showing much higher risk as you go higher and higher in power level.

So below the green line, which is your fuel duty limit, you have a pretty high percentage probability of failing the fuel unless you do something to protect it. Above that range in the AOO region, you have an extremely high probability of failing the fuel unless you do something to protect it.

So the question is what can you do in the AOO regime, and right now the only options are prompt action by the operators or use a PCI resistant fuel limit, and I'll get to that shortly.

Yes, George.

MEMBER APOSTOLAKIS: Sam, I'm trying to understand this figure. Let's take the exposure of the 20.

MEMBER ARMIJO: Right.

MEMBER APOSTOLAKIS: So vertically up.

MEMBER ARMIJO: Yes.

MEMBER APOSTOLAKIS: I hit the blue lines for one percent out of 90 feet.

MEMBER ARMIJO: Right.

MEMBER APOSTOLAKIS: What do these percentiles mean? Percentile of what?

MEMBER ARMIJO: I'm not a statistician. If you tested 100 rods, one would be likely to fail even at a power as low as ten kilowatts a foot.

MEMBER BLEY: And the blue is percent of pins failing.

MEMBER ARMIJO: Correct.

MEMBER APOSTOLAKIS: What confuses me is that in addition to the red dots, you have three yellow.

MEMBER ARMIJO: All the yellows are sound. They didn't fail.

MEMBER APOSTOLAKIS: Sound. So they

didn't fail.

MEMBER ARMIJO: That's right.

MEMBER APOSTOLAKIS: How can they still be between the one-th (phonetic) and 95th?

MEMBER ARMIJO: Well, because it's statistical. You know, you have a range of --

MEMBER APOSTOLAKIS: How do you start out?

MEMBER ARMIJO: We started out -- this particular database includes probably 70 rods, and so the confidence in the percentiles actually gets better as you get to where everything is failing, and as you get down towards the PCI threshold, you know, it'd debatable whether it's one percent, but when you get up into the 50, it's getting pretty confident that that's not a good regime to be in.

MEMBER MAYNARD: Well, the percentiles are really only applicable to the failed ones, right?

MEMBER ARMIJO: That's correct.

MEMBER MAYNARD: It really has no correlation for the yellow. So the yellows are out.

MEMBER ARMIJO: Right.

MEMBER MAYNARD: But the --

MEMBER ARMIJO: No, the yellows are in. Both count. You know, you add up all of the rods and say what percentage failed.

MEMBER MAYNARD: Right, but the percentile doesn't. The one percentile, 50 and 95 percentile, that's for the failures. That's for the dead ones.

MEMBER ARMIJO: Correct.

MEMBER APOSTOLAKIS: If I start with 70 rods, what they're saying is that the power of about 50-plus rods per meter, 95 percent of them have failed. Is that the interpretation?

MEMBER ARMIJO: Yes, yes. So that means --

MEMBER APOSTOLAKIS: If I have 70 rods, what's 95 percent?

MEMBER ARMIJO: It's a lot of rods.

MEMBER BLEY: It's about 63, 64.

MEMBER APOSTOLAKIS: Rods to fail.

MEMBER ARMIJO: And every once in a while, you know, remember these are fuel pellets in a fuel rod. The pellets aren't exactly all the same dimensions. The cladding diameter isn't exactly all the same dimension. So there is a variability. There's an uncertainty of whether it will fail or not, but once you get up to a high enough power, you have high enough stress that, you know, that uncertainty gets to be pretty small, pretty much all failed.

MEMBER BLEY: I don't know quite how to

ask this, Sam, but the blue lines must come from some predictive model because from the data most all the rods --

MEMBER ARMIJO: No, quite a few of them succeed at t he lower powers.

MEMBER APOSTOLAKIS: What you don't see here then is the successes. There's one bit of drugs in the test.

MEMBER BLEY: The blue lines came from the data?

MEMBER ARMIJO: The blue lines came from the data, and I believe there was also additional data that's not shown here that was earlier data, but these blue lines are just -- just use them for now as a reference because later we're going to compare them, exactly the same test data, but with a different fuel design just for reference.

MEMBER APOSTOLAKIS: You had 70 rods, and all of them had the same exposure?

MEMBER ARMIJO: No, they had exposure that varied in that's shown on the curve.

MEMBER APOSTOLAKIS: Oh, okay. So at a line of 20, I'm not looking at 70 of them.

MEMBER ARMIJO: No, you're just looking at a small percentage there, and so the actual percentile curves were developed by statisticians which I cannot explain or defend. That took into account all of these data and came up with a curve.

MEMBER APOSTOLAKIS: I see two, four, maybe five reds and three what, yellows?

MEMBER ARMIJO: Well, in that range, what would you say? You'd say about half of them are going to fail. If you took that very limited population at 20 and you stayed below the green line and above the one percentile line --

MEMBER APOSTOLAKIS: Percent below the green.

MEMBER ARMIJO: Yeah, yeah.

MEMBER APOSTOLAKIS: Additional.

MEMBER ARMIJO: Yeah. The key point I want to get at is when you get up to these very high powered like 16 kilowatts per foot, you're going to fail a lot of fuel unless there's something done to protect it, and that's the main point.

MEMBER APOSTOLAKIS: How many kilowatts you say?

MEMBER ARMIJO: Sixteen.

MEMBER SHACK: Go over to the kilowatts per foot.

MEMBER ARMIJO: On the right-hand side,

the kilowatt per foot. The other side, Europeans and many other people --

MEMBER APOSTOLAKIS: Those were heaters.

MEMBER ARMIJO: Kilowatts per meter, yeah, and so I just put it that way.

MEMBER APOSTOLAKIS: Okay.

MEMBER ARMIJO: So now we're calibrated. The key point is high power region is pretty dangerous.

VICE CHAIRMAN ABDEL-KHALIK: All of these experiments were conducted with the same ramp rate.

MEMBER ARMIJO: Yes, yes. Fast ramp rates.

VICE CHAIRMAN ABDEL-KHALIK: Okay.

MEMBER ARMIJO: Some were slightly faster than others, but remember --

VICE CHAIRMAN ABDEL-KHALIK: I mean fast compared to what one would expect during a transient, for example?

MEMBER ARMIJO: No, no.

VICE CHAIRMAN ABDEL-KHALIK: Faster?

MEMBER ARMIJO: Slower.

VICE CHAIRMAN ABDEL-KHALIK: Slower.

MEMBER ARMIJO: Slower than one would expect it, in fact.

MEMBER APOSTOLAKIS: So what is faster? I mean what --

MEMBER ARMIJO: Well, in PCI, you can fail fuel in the operating range with ramp rates of something like .2 kilowatts per foot per hour, very, very slow. PCI does not require super fast ramp rates. It just requires enough for ramp rate to create stress.

So ramp rate is not a variable here, but it was evaluated in these programs. Now, this looks like, okay, interesting laboratory data, but what happens if you did this in a power plant?

Okay. The next chart is a description of the Oskarshamm 1 event. In this reactor in Sweden, they did a deliberate experiment in the core. At that time the engineers there believed that their conventional Swedish eight-by-eight Zircaloy-2 was not susceptible to this PCI problem that was affecting the U.S. fuel. They decided to do a demonstration of their fuel resistance, and what they did at the end of cycle, they pulled a control blade on one blade out of 112 in the core. Okay? They only reach powers as high as 11.3 kilowatts per foot. That was as high as they went. So not particularly high.

They failed 45 rods in 14 bundles by PCI.

That's a lot of fuel. Now, that's one percent of the core. If they had moved all of those rods, just multiply by 100. So this is a very potent failure mechanism, and the concern is obviously they wouldn't have done that but the concern is the only thing that can move the power of fuel rods up in a core is a whole core transient, and that's where we get to the concern about the loss of feedwater heater.

VICE CHAIRMAN ABDEL-KHALIK: Now, these were calculated peak kilowatt per foot during that rod pull.

MEMBER ARMIJO: Well, it's measured in the sense they know when they pull the rods where the power is in the rods. This is an operating reactor. So they've already modeled the fuel and they know what the power is actually. They know what the power is rod by rod in each assembly.

VICE CHAIRMAN ABDEL-KHALIK: In other words, did they do a gamma scan, for example, afterwards to find out what the power history in those particular rods that failed?

MEMBER ARMIJO: I don't know if they did a gamma scan, but I suspect they did because they all went to hot cells, but even without it Said, they know what the power is in those fuel rods from just their whole operational history.

But they did go into hot cells and verified that they failed by PCI. So it's a very potent failure mechanism, but it has been under control for a long time, but what happens in a loss of feedwater heater transient in a BWR?

Well, first of all, it's something that you can't terminate automatically. There is no system involved that can do that. You get a lot of cold water coming in. It's most severe if all of the feedwater -- there's a missing word there -- if all of the feedwater heaters are bypassed. Then you can get up to a maximum of about 100 degrees subcooling. What that does, it raises the core power at the bottom of the core -- all of this cold water comes in -- by as much as 100 percent of the rated power.

So your peak powers in this type of event could be 16 kilowatts a foot, much, much higher than what happened at the Oskarshamm event, and every rod is to one degree or another increased in power, and it takes about a minute to get up to peak power, and then it's maintained at that peak power until somebody turns it off.

So we're going to get to say, you know, how much time do you really have to terminate that situation.

Well, here's the PCI mitigation options. Well, in normal operation I'm not worried about it. People know how to handle it. They can use PCI resistent fuel. They can precondition the fuel. Some people do both, but in the AOO regime, you can't precondition. You're never allowed to operate up there deliberately.

So the only mitigation you have is either fundamentally in the design of the fuel demonstrated resistance or prompt operator action.

Okay. So I'm going to show you the differences of what these options look like.

Next slide.

MEMBER STETKAR: Sam, just a quick education. Prompt operator action is scram the reactor?

MEMBER ARMIJO: Terminate it, but bring your flow back. I don't know all the options they have, but scramming is one. They normally would reduce flow because they're getting too much cold water in there. So they'd back up on flow.

Now, this is the same type of test, exactly the same size of fuel rods, test reactors, incubated and power reactors. The only difference here is this is a PCI resistent fuel design, and they've been tested in the same severe way. So you can see up in the AOO regime this fuel is resistent by design. It can operate with a very low failure probability even in this very, very aggressive power regime.

Now, there are a couple of failures there. This is not labeled PCI immune field. I don't think there is such a thing, but it is highly resistent compared to the conventional stuff that you saw in the previous chart, and those percentile numbers are just for reference. They're the same ones that were in the previous chart.

MEMBER APOSTOLAKIS: What is the meaning of those blue bars now?

MEMBER ARMIJO: The blue bars is the start of the test. That means in a few seconds we raise the power from that level all the way to the very high levels. The intent there was to demonstrate the different really severe power ramp tests. So the BWR has that capability. When you pull a control blade, the power changes very fast.

Okay. So that's what you can do. One option of mitigation is use a fuel that's resistent. The other option is prompt operator action, and here's something that I got from these two different test programs, one the G.E. program and one the Demo-Ramp II program.

This charts shows the peak power in the vertical axis and failure time in the horizontal axis, and you notice the failure time starts at one minute. Okay. The crack nucleates and propagates through the cladding as early as one minute, and sometimes it takes a lot longer.

And in the power regime of interest to AOOs, let's say, 14, 16 kilowatts per foot, you can have quite a few failures in three minutes.

You'll notice on that chart most of the fuel rods in that high power regime fail during this test, but they take time, but a lot of them don't take much time at all, and five out of the 25 failure on that chart failed in one to three minutes. Okay?

So that means the threshold, if you want to avoid, let's say, 19 percent of the fuel that's exposed, if you want to keep that from failing, you have to terminate that transient before one minute. That's what that G.E. data set says.

Now, the next chart show --

MEMBER MAYNARD: Sam, I want to ask you.

MEMBER ARMIJO: Yeah.

MEMBER MAYNARD: Was that fuel preconditioned to 100 percent power or was it not preconditioned?

MEMBER ARMIJO: This fuel was not preconditioned at all. It was operated in power reactors around four to six kilowatts per foot. Then it was what we call kind of like in a sense preconditioned. Then in the test reactor it was operated around eight kilowatts a foot for a certain period of time so that everybody started from the same power level.

So you could say it was preconditioned up to eight kilowatts a foot, but eight kilowatts a foot is usually as safe as you -- so it was not at all preconditioned. If you had preconditioned the fuel let's say up to 13.4, you probably would have had better performance, but you would not have solved the problem because a delta kilowatt per foot of half a kilowatt a foot or one is sufficient to cause PCI even if you start with preconditioned fuel. So that's a good point

MEMBER MAYNARD: Would you expect it to maybe shift the time though?

MEMBER ARMIJO: You might. You might, but you know, this is the data we have.

Go ahead.

VICE CHAIRMAN ABDEL-KHALIK: What is the main independent variable that causes that large change in failure time?

MEMBER ARMIJO: I really don't know. I think it's just the variability of the mechanism of when does the stress build up. Are the fission products that are causing this in the right concentrations? You know some people believe it's the fresh fission products coming out during the ramp test that are causing the problem, not the accumulated fission products over the life of the fuel rod.

So all we can say is that it is variable, and you know, it can take a long time or a short time, and you can't predict in advance, but you can count the number of rods that fail in a short time, and they're a significant fraction of the total number, and just to verify that this was really representative, I dug up the data from the Demo-Ramp II program, and this was done with the NRC participation deliberately to find out what is the threshold time for failure, and this is same scale, peak power in kilowatts per foot and time to failure in -- this time we started at tenth of a minute, minute, ten minutes, and so on.

This was a smaller database, and in the case in this program, the objective was to terminate the test and then examine the fuel to see if it was damaged at all, and all of the and I'll use the word magenta because I heard that this morning; all of the magenta diamonds are rods. The test was terminated at those times, and there was no failure, but every one of them had PCI cracks part way through the cladding up to 60 percent of the way through.

Now, the one red data point failed after a little over an hour. Okay? So you have damage in a lot of the rods. The one diamond rod at less than a minute, that was terminated in less than a minute, and there was no damage, but statistically, you know, if you looked at that area you'd say there could be damage or failure in shorter times.

So if you combine the two data sets, you say, look, if you have to rely on prompt operator action and you look at all of these failures or damage to the fuel, you've got 22 percent of the rods that were tested or damaged or failed within three minutes. So that means your operator action has to be effective within three minutes, and can you do that and should you rely on that?

MEMBER APOSTOLAKIS: Scram is one

possibility.

MEMBER ARMIJO: It doesn't scram, George.

MEMBER APOSTOLAKIS: It doesn't scram?

MEMBER ARMIJO: That's one of the problems with this particular transient. There is no scram function.

VICE CHAIRMAN ABDEL-KHALIK: But isn't there an over power trend, scramble?

PARTICIPANT: Can you scramble 110 percent power?

MEMBER BLEY: It's the skew in power that keeps it from --

MEMBER ARMIJO: Well, I can tell you from the FSARs I'm not a systems guy on this thing, but this transient, and you know, there's some proprietary stuff I can show you from G.E.. There is no scram termination of this transient. There are a lot of other automatic functions that will terminate other transients, but not this one.

MEMBER POWER: What you're saying is there's no automatic scrambling.

MEMBER ARMIJO: That's correct.

MEMBER POWER: It's a manual scramble.

MEMBER ARMIJO: Correct. An operator has got to --

MEMBER POWER: The other question is that suppose I do fail 22 percent of my fuel and it is strictly a fuel failure and I violate my tech specs, to be sure, but am I posing any risk to the public?

MEMBER ARMIJO: I don't -- I think I really don't know. It's thousands of fuel rods. Twenty-two percent would be thousands of fuel rods. Obviously, you don't want to get into that.

The answer is I think there may be.

MEMBER POWER: Twenty-two percent of the gap inventory released, which, okay, take it reasonably conservative and say five percent of the inventory is in the gap. So what? I'm releasing one percent of my fission gas?

MEMBER ARMIJO: Right.

MEMBER POWER: I don't get any of the water soluble radionuclides out. All I'm getting is fission. How significant is this?

MEMBER ARMIJO: Well, I think looking at it in that term, you know, I would say that the off gas systems are designed to handle fuel failures, but I'm not sure they're designed to handle huge numbers of fuel failures. You know, that's beyond my expertise.

MEMBER POWER: Yeah, an off gas system, I mean, it's going to wreck it up through the stack. That's all you're going to get because it's just fission gas. You're not going to get any iodine, cesium, tellurium. All you're going to get is fission gas.

MEMBER ARMIJO: Right. What you're going to get is those fission gases, and you're going to have a lot of failed field.

MEMBER POWER: Well, you're going to have a problem. I mean, as an owner-operator you're going to be kind of irritated, but --

MEMBER ARMIJO: Well, we can get into that. I think as a regulator you're going to be highly embarrassed.

MEMBER POWER: And if i'm a reporter for the Village Dispatch I'm going to be happy as a clam, but I'm not sure anybody else is at risk here.

MEMBER ARMIJO: Well, that's one of the nice things about defense in depth.

(Laughter.)

MEMBER ARMIJO: These things, if they happen, you do have back-up systems to protect the health and safety of the public. The point I'm trying to make is you can fail a lot of fuel unless you move very, very fast, and is it realistic to expect that this fuel -- that an operator can detect, analyze.

Well, he doesn't know before it happens. Should we count on an operator to terminate this thing before a minute, and the answer is I don't think it's very likely.

MEMBER POWER: Well, we never give them credit for something within a minute.

MEMBER ARMIJO: Well, there's the data. That's all I can tell you, and like I said, these are two independent data sets with very, very similar --

MEMBER APOSTOLAKIS: If I understand Dana's point, Dana, you're arguing that the obvious may be harmful to the owner, but it's not risk, sure. Is that what you're saying?

MEMBER POWER: Yeah, well, that's what I'm asking. I'm asking what the threat is here. I mean, it's one of degree always. It is that do I cut my veins over this? No, I don't think so.

MEMBER ARMIJO: Do you ignore it?

MEMBER POWER: Well, probably not. I'm trying to put it in a context that, yeah, your fuel failed, but it is not like fuel failure when we've had an overpowered transient that's going to core melt.

MEMBER ARMIJO: This is not in that category.

MEMBER APOSTOLAKIS: Damage to the first defense in depth barrier.

MEMBER POWER: Well, if you take the main thinking of the first, it's the second.

MEMBER BLEY: But if you go back 30 years to when we had a lot of PCI problems and bloomers, there was a potential significant problem for workers from this because they were reconstituting fuel bundles and the fuel pool pulling things up near the surface of the fuel pool, jury rigs and things, but for public risk there's not anything.

MEMBER ARMIJO: I think that the beauty of defense in depth is we do have all of these protections, but I don't -- I just want to make the point this is -- look. Do you want to design fuel and operate fuel in a way that you expect huge numbers of fuel rod failures?

And the answer should be no, and the way I look at it is, you know, you have two options. You can wear a bullet proof vest when somebody is shooting at you or you dodge. Which do you prefer? I prefer both. I would put on the vest and I would dodge the bullet.

(Laughter.)

MEMBER ARMIJO: I've way overdone my time, and I'm apologizing in advance to everybody else, but I think this was important to get these data out.

Jack.

MEMBER SIEBER: I think that Dana's point is worthy of additional consideration. The question, it seems that it is fairly clear that this is a regulatory issue because G.E. tech says you have to have design fuel to handle, anticipate transients. On the other hand, is it a safety issue?

And there are two things that come into play. One of them is does it exceed Part 20 or Part 100 requirements for both the public and/or workers, and it's not clear to me that we've answered that question.

Now, whether we get involved or not, and it depends on whether it's a safety issue; otherwise we can tell the staff it's a regulatory issue. We can deal with it in Criterion 10 space.

And I think it's a key question to ask or at least think about as we go through these presentations.

CHAIRMAN BONACA: Let's complete your presentation.

MEMBER RAY: I know we're out of time. If you could just make one comment, why are we not seeing

PCI resistent fuel?

MEMBER ARMIJO: Well, I don't know. I don't know the answer.

MEMBER RAY: That's an answer.

MEMBER ARMIJO: Most of the BWRs, most of the BWRs have PCI resistent fuel. Okay? So I was kind of surprised that people would take risk with their reactors. You may not be a safety problem, but it's going to be a huge problem to whatever has a core that has this problem.

MEMBER SIEBER: It's a financial risk, and it's a big risk.

MEMBER ARMIJO: And why people are using conventional fuel in this environment I don't -- you'll have to talk to them.

MEMBER RAY: I don't make the distinction of regulatory versus safety. To me the requirements are the requirements.

VICE CHAIRMAN ABDEL-KHALIK: I guess the point I'd like to ask, you stated that there are no automatic actions --

MEMBER ARMIJO: Correct.

VICE CHAIRMAN ABDEL-KHALIK: -- in response to this, that there are no run-backs or trips that would reduce the peak powers in transients of this sort.

MEMBER ARMIJO: That's correct.

MEMBER SIEBER: I'm not sure that's the case.

MEMBER ARMIJO: Well, I'll just show you -- well, it's a proprietary analysis, and we'll have to get to that.

MEMBER SIEBER: Well, there's no doubt that you take the feedwater heaters out and the efficiency of the cycle goes down. The throttle valves stay at the same position. So the core power has to go up perhaps by as much 20 percent.

Now, when that happens you get a redistribution, the rapidity in the floor which makes the bottom skewed; we know you do have a high trip, and the question is does it go high enough in order to trip before the--

MEMBER MAYNARD: I think to say if there's no automatic trip is wrong. I think to say that it's possible to have a transient that could get into this regime without an automatic trip is a true statement. You know, loss of feedwater heating and stuff, that can occur, and you can get power levels down low in the core.

Eventually you're going to hit an

automatic drip if something is not done.

MEMBER ARMIJO: Eventually you will, but, you know, that -- and when we get to the GEH, I'll show a slide if they have time. But the main point is here is a problem. Here is data. All of this data has been around, but I don't think it has ever been plotted this way. And so that is what we did.

I would just like to quickly go through the conclusions. Some are pretty obvious: Failure driven by chemistry stress, not by strain. Strain required to cause PCI failures is much lower than one percent strain criterion. So that doesn't protect you at all. And so those aren't really the right criteria to use for a stress corrosion cracking problem.

The crack nucleation propagation rates are very fast. And so you have got very limited time to do anything. And the number of fuel rods you put at risk just increases with EPU. And so there are 20 percent more rods at risk than there were before.

Recommendations. If I were going to advise anyone, that's the job I've got. The next one is set your PCI failure criteria based on the mechanism at hand. And that's measured. You measure the failure powers or the failure times and don't set your failure criteria on calculated strains.

Use empirical, measured data. And do the testing in a way that the fuel is tested in a prototypic way, representative of the powers, delta powers, burnups that you would expect during an AOO unless you can solve the problem some other way.

You know, if there's a smarter system solution or something else, I have no problem with that, but you have to address that.

And with that, I'll close. And, again, apologies for running over. I thought this would go faster, but it obviously didn't.

MEMBER POWERS: That is terrific, Sam.

MEMBER ARMIJO: Thank you.

Well, at this point, Paul, I would like to ask you to -- you have one minute, Paul.

(Laughter.)

4.2) BRIEFING BY AND DISCUSSIONS WITH REPRESENTATIVES OF THE NRC STAFF AND NUCLEAR INDUSTRY

MR. CLIFFORD: Hello. My name is Paul Clifford in NRR, Division of Safety Systems. And we are responsible for the review of fuel designs and power uprates and all licensing actions associated with Chapter 15 or Chapter 6 of the FSAR.

Today I am going to be talking about some staff concerns with the previous ACRS direction. I will also be outlining three proposed strategies for dealing with PCI, including pros and cons of each. And we will discuss in general backfit versus forward-fit.

Sam already went over here. This slide provides the information, the text from the ACRS letter and staff response.

Some of the concerns the staff has with the direction that was provided in the ACRS letter are: first of all, it asks the staff to develop an analytical method for evaluating PCI.

Back in the 1980s, the staff spent considerable time and effort trying to develop an analytical model that could be calibrated and then verified against ramp data to actually predict crack nucleation, propagation, and eventually cladding failure. A lot of time was spent. And ultimately it was decided that it couldn't be done at that time. So there is some resistance with the staff to go down that road again because it is a very complex phenomenon to model analytically.

Second is all fuel designs have some susceptibility to PCI stress corrosion cracking. And to develop a regulatory approach for dealing with it, you really have to understand the varying degrees of resistance offered.

For instance, doped pellets, which are being developed and will be introduced by certain inherent PCI resistance, we will need to understand those if we are going to regulate PCI stress corrosion cracking. And sponge or natural zirconium barrier would have a different PCI resistance than a low alloy zirc barrier, which we are starting to see introduced also.

And one reason. to digress a little, there is a fear based on past experience that if you get a primary failure due to debris fretting or a bad manufacturer, manufacturing issues. And if you get a primary failure and the natural sponge zirconium barrier can lead to a severe secondary failure of the fuel rod, you can see a couple of feet of actual split in the fuel rod and loss of fuel due to the sponge zirconium.

So, to counteract that, there is a move to use low alloy to put some tin or put some iron in the barrier. And when you do that, you provide protection against the secondary failure mode. But at the same time, you reduce the inherent PCI resistance. So there is a balance that has to be done.

Another issue, which was talked about earlier, was the concept of crediting very prompt operator action during a Chapter 15 event. This would be kind of a departure from where we are normally in Chapter 15.

Chapter 15 is very conservative, use conservative assumptions, inputs, and modeling techniques to try to maximize to come up with the worst of the worst. And coming up with three or five-minute operator action would be a departure from what we are trying to do.

Now, moving forward, I will be going through some potential strategies, but it is important to remember as we consider moving forward with PCI that we need to define performance requirements. We can't require a specific design feature. We can only require a performance of a given feature, of all field designs.

And the regulations have to apply universally, not just to a particular design or particular cladding type, and that also stress corrosion cracking, although it's primary a BWR issue, is not inherently a BWR issue. And it is also not inherently an EPU issue. So it would have to apply across the board. That is all I am saying.

I am trying to move fast, but you guys ask questions.

Now, this isn't an excuse, by any means. This is just a reality. All tasks that come into NRR get assigned a work prioritization based upon the need for the product and the availability of resources.

And the staff has essentially assigned a low safety significance to PCI. And we can go into that some more. And because of its low safety significance, it kind of falls down on the work assignment list.

And I say that because, as Sam showed in his picture of the hot cell, stress corrosion cracking can lead to a very tight hairline crack in a fuel rod, and you can get a release of some of the gas that's present in the plenum. But there's no challenge to cool well geometry. There's no fuel dispersal. There's no fuel coolant interaction. There's no challenge to the pressure vessel integrity. There's no challenge to the containment building. And there's no challenge to any of the systems which are designed and available to mitigate the consequences of the transient in order to minimize off-site doses.

MEMBER BLEY: Paul, can I ask you a question about the prioritization system? When something gets a low priority, is there an aging factor? You know, the longer you stay at low priority, do you --

MR. CLIFFORD: That's an excellent question, but in the next bullet -- you know, you only devote so much time to regulatory improvement because you have --

MEMBER BLEY: But is there some kind of an aging thing to things that get put in low priority that it starts to escalate when you design computer systems, you do that and --

MR. CLIFFORD: I would say, unfortunately, not if you go back in history.

MEMBER BLEY: If you are low, you may never get out of there.

MR. CLIFFORD: Right now there are three major -- there are more than this, but there are three major regulatory improvements that are underway. And that is the revision of 50.46(b), ECCS acceptance criteria, a revision of reg guide 1.183, which the staff will be seeing next month, I believe, and a revision to reg guide 1.177.

So we are already working on a lot of regulatory improvements. And we feel that this issue is just a little bit below that. So we will get to it. It's just this is a prioritization that it has been given.

One thing that is important to know is there are no regulations specific to PCI and there is no reg guide specific to PCI and that currently each plan has its specified licensing basis that is documented as part of their tech spec and then their FSAR. And that will include citations to approve topical reports, which will provide the cladding failure mechanisms and the SAFDLs, which is the specified acceptable fuel design limit, that make up each plant's licensing basis.

So each of the fuel SAFDLs that we have, like during an AOO for BWR, we would have essentially three fuel SAFDLs. We would have fuel temperature. We would have cladding strength. And we would have critical heat flux. And each of those is inherent in each of the plant's licensing basis.

So any change in the treatment of PCI, as we are talking about today, would constitute a change in a regulatory staff position. And I put that in red because --

MEMBER ARMIJO: You probably want to challenge that because if the people who identify the SAFDLs are the fuel manufacturers or the vendors, the authors of the topical reports, --

MR. CLIFFORD: That is correct.

MEMBER ARMIJO: -- they identify them and they identify them until they protect you in these transients against all known failure mechanisms. Now, if they identify SAFDLs that don't protect you against a known failure mechanism, that's a flaw in their work. So when you find that problem and you ask them to correct it, that's not a backfit.

MR. CLIFFORD: I will get into that. We have specific --

MEMBER ARMIJO: Okay. I just want to get that point across.

MR. CLIFFORD: It is a change because you have a plant's licensing basis, which is documented in their FSAR in the tech specs. And that basically says that as long as you preserve these three or four SAFDLs, you are good.

MEMBER ARMIJO: But if the SAFDLs are proven to be inadequate, how can you just ignore it?

MR. CLIFFORD: No, I'm not saying ignore it.

MEMBER ARMIJO: Oh, okay.

MR. CLIFFORD: I'm just walking through the process.

MEMBER ARMIJO: All right.

MEMBER RAY: You know, you said something about a GDC10. And I just wanted to have you repeat it because I didn't catch it.

MR. CLIFFORD: Well, if we approve a fuel design, they have to show compliance for GDC10. So we have gone through, and we have evaluated the failure mechanisms that have been proposed for the fuel design and the SAFDLs that have been proposed for the fuel design. And we have accepted that fuel design, which has then become inherent in the plant's licensing basis.

MEMBER RAY: I understand, but there is an omission, well, it can sound the same. That would seem to simply be an omission and --

MR. CLIFFORD: Right. And there is a process to follow. And I will get into that. Hold your breath for five minutes, please.

MEMBER RAY: Okay.

MR. CLIFFORD: Okay. So I believe there would be a change in the regulatory staff position. And so there is some consideration for 50.109, which is backfitting requirements. And also there is a process called a regulatory analysis that the staff follows as a kind of guide to go down to make decisions on how to change staff positions or how to change regulations. And there is cost-benefit. And there are other things that are in there.

A regulatory analysis is an in-house process to help the staff. It's not a regulatory requirement. Just keep that in mind.

Okay. There are three strategies that were identified. And I'll go through the pros and cons of each one of these. The first one is to maintain the current approach.

MEMBER APOSTOLAKIS: I didn't get a chance to read 03.

MR. CLIFFORD: I am trying to move fast.

MEMBER APOSTOLAKIS: All right. Sorry.

MR. CLIFFORD: Okay. So the first strategy would be to maintain the current approach. And the benefits of this would be that the current approach provides a reasonable level of protection during core-wide AOOs. And we have talked about that. And also the staff resources would be devoted to more, we believe, substantial regulatory improvements at this time.

The problem is we recognize that there is a potential that you would get fuel cladding breach under certain conditions in a BWR but under certain conditions. It's not every AOO. It has to be a certain AOO.

And also if you don't do anything, then there is a lack of specific guidance and regulatory criteria for future designs. I mean, you can prioritize this based upon knowledge today, how plants operate today, how fuel is designed today.

But if you don't do anything to capture for the future, then today is just a snapshot. Ten years from now, you know, in theory they could have moved to a corner of operating space which is worse. So that is never a good thing.

Strategy number two would be to revise NUREG-0800, which is standard review plan SRP-4.2, to provide specific guidance on how to address this new fuel failure mechanism or specifically just this in addition to the other fuel failure mechanisms and provide guidance on what level of qualification is needed to demonstrate that you don't get fuel failure during AOOs.

In this proposed strategy, you would need to quantify the PCI resistance of each and every fuel designs. Whether it has a sponge barrier or a low-tin barrier or whether it has no barrier or whether it has doped pellets or whatever it has, you have to quantify that level of resistance.

And then you would rely upon power uprate data to come up with an empirically derived fuel rod failure threshold based on change in rod power and the duration at the elevated power.

In the analysis world, you would then use calculated rod powers for all of your Chapter 15 AOOs and compare them against this empirical threshold.

Now, the benefits of this would be more strict compliance with GDC10, which I believe is what this Committee is -- what Sam is looking for.

And also this is the same approach we used for rod ejection. For rod ejection, I mean, it is a very difficult thing to model. So for rod ejection, we have an empirically based -- rod ejection would be a reactivity-initiated accident or a drop accident for BWR. There you have an empirically based PCMI failure threshold, which is based on in-reactor tests. And then you use fuel enthalpy as a measure to determine whether you fail.

So this approach is very similar to how we handle that accident. And a good way of doing this or what is good about this approach is that we have tools available today that could calculate rod power history with high confidence. You know, you may not be able to calculate crack propagation with high confidence, but you can calculate rod power history with high confidence.

The cons of this approach would be that you would require probably additional empirical data from power ramp testing. And there are just limited facilities. With the closure of the Studsvik reactor, there are just not a lot of reactors out there to run these tests.

MEMBER ARMIJO: Paul, I would call that a pro. I don't think anybody should be fielding fuel that they haven't tested rigorously. And, you know, that is the business these guys are in.

MR. CLIFFORD: Right.

MEMBER ARMIJO: And if it takes some time, takes money, and if they haven't already done it, they should do it because their risk is really there.

MR. CLIFFORD: Right, right. And I would give you a kind of a better feel for compliance with GDC10, that you wouldn't get a feel for that. But it is a con because it is difficult to find in these facilities. And it would be expensive.

So another con would be it would be expensive. And it would have to be analyzed. And when you do your analysis, it may require that you go out and you change, say, trip setpoints. I mean, the event we talked about earlier, the loss of feedwater, you could go up to 118-120 percent trip. That's because in the analysis world, they find the worst decrease in temperature sot hat they won't trip the cooler.

I mean, in real life, you can get a partial loss of feedwater heater. You can get a whole rainbow of events, some of which are going to give you a trip and some of which may not give you a trip.

MS. ABDULLAHI: May I comment? Excuse me. This is Zena Abdullahi. I want to comment on the loss of feedwater heater.

MR. CLIFFORD: Okay.

MS. ABDULLAHI: Loss of feedwater heater is analyzed for 118. Neutron flux, high neutron flux, is 120 to 118. It is analyzed, but it is analyzed for CPR. It is not analyzed for other staff violation --

MEMBER ARMIJO: Or any other mechanism.

MS. ABDULLAHI: SSC.

MR. CLIFFORD: Right. But the point is if you get the trip --

MS. ABDULLAHI: It is analyzed for --

MR. CLIFFORD: If the scenario is so significant that you get the trip, you're not going to have PCI.

MEMBER ARMIJO: Right.

MR. CLIFFORD: If you get a trip, you are not going to get PCI.

MEMBER ARMIJO: So you could argue that if you could change a trip setpoint to guarantee a trip, no problem. But then there are issues you don't want to lower the trip setpoint so low that you could get spurious trips.

MEMBER SIEBER: In a feedwater action, the profile changes quite a bit. The bottom of the core naturally, in effect, has an effect on the core instrumentation. So it really doesn't see the floor that reaches the flux. That's not the issue.

MR. CLIFFORD: Sure.

MEMBER ARMIJO: Paul, maybe we shouldn't solve the problem now.

MR. CLIFFORD: Right, right. The next --

MEMBER SIEBER: We want to see if there is a solution.

MR. CLIFFORD: The next strategy would be to once again revise the guidance to the SRP to provide guidance on stress corrosion cracking as a failure mechanism and a level of qualification needed for no fuel failures.

Here you would rely on analytical models. You would need to verify and validate an analytical model capable of calculating with high confidence -- and I emphasize that, "with high confidence" -- crack propagation and cladding failure under this combined mechanical loading and chemical attack.

Here in the analytical world, once you have developed the model to determine what the failure thresholds are, you would during your Chapter 15 AOOs calculate local cladding stresses. And you would show that you remain below the stresses that your other model determined was failed.

MEMBER SIEBER: Which we don't know yet.

MR. CLIFFORD: Right. Now, the pros, here are, once again, strict compliance with GDC10. The cons are you still need a lot of empirical data. You still need power ramp tests because you need to validate your models.

MEMBER ARMIJO: Absolutely.

MR. CLIFFORD: And that takes a lot of data. Also, stress corrosion cracking is a very difficult phenomenon to model, as I mentioned. The staff tried, spent a lot of effort, and at the end of the day didn't have success. And also there are certain assumptions on the chemical effects and our initial crack depth that would go into the modeling and that would need to be justified.

As far as we know, there is no well-verified analytical model that exists today. And also when you develop a model to today's standards and you try to get a 95/95 confidence on the prediction of this crack propagation and failure, once you get there and you are seeing Sam's data, there is a lot of spread in that data.

If you want to get a 99 prediction of those failures, now you're going to be calculating failure all the time. Even for rods that wouldn't fail, you are going to be predicting that they would fail just to be conservative because that is how we do things in Chapter 15.

So this approach would probably necessitate --

MEMBER SHACK: There is nothing that says we have to do this as a Chapter 15 kind of analysis. I mean, there are lots of analyses we don't do to 95/95. You don't calculate crack growth for a pre-crack pipe to a 95/95.

MR. CLIFFORD: But as far as showing compliance with GDC10, we generally take that standard. I'm just saying if we follow the standard practice.

And so this approach I believe would cost a lot more than approach number two. I believe it would result in potentially overly burdensome requirements and changes necessitated because of the analytical approach.

VICE CHAIRMAN ABDEL-KHALIK: Wouldn't the second approach have the same uncertainty in terms of -- I mean, after all, the uncertainty comes from the spread in the data.

MR. CLIFFORD: But you also have to predict it. Future designs are predicted for different conditions. And you are not only putting an uncertainty on your prediction of the failure, but you are also putting an uncertainty on your prediction of cladding stress.

You have uncertainties on fuel swelling, cladding stress, and then you have uncertainties over here on your failure threshold based upon your model prediction of the empirical data, as opposed to you kind of get rid of that one layer of uncertainty by just saying here is my measured ramp test data, here is the failure based upon measured ramp test data, and here is my calculation of raw power, which is very accurate.

MR. RULAND: Just a comment about Paul said strict compliance with GDC10. I think everybody here on the Committee knows that GDC10 is not a regulation. The staff has used GDC10. And we determine, the staff determines, that yes,, in fact, they meet the design criteria, so just a minor point.

MR. CLIFFORD: Okay. Backfitting, 10 CFR 50.109 provides -- really, it is a regulatory hurdle for implementing new requirements on existing licensees using approved methods and design.

The words that are in 50.109 essentially are that it requires a substantial increase in the overall protection of the health and safety of the public and costs have to be justified.

Now, the rule allows exceptions, but I don't believe that this PCI issue would qualify as an exception to the rule under compliance or adequate protection.

MEMBER POWERS: Now, why do you believe that?

MR. CLIFFORD: It is because of the failure mode and the fact that, as we believe, it's a low safety significance as far as dose to the public when you --

MEMBER ARMIJO: Now, Paul, every one of

these fuel manufacturers has a PCI-resistant design, which they offer commercially and which they have test data for, ramp test data. So what is the big problem of asking them to show it to you and see if it is good enough?

MR. CLIFFORD: I can get to that next slide.

MEMBER ARMIJO: Okay.

MR. CLIFFORD: The next slide is what I would call forward-fitting. I am going to read this right off of here because I went through this. And I will get yelled at if I deviate from it.

No regulatory expectation that requirements or staff positions remain stable for future requests for agency approval or action. That being said, if you expanded the fuel design requirements to include specific accounting for PCI for future designs, it would not be a backfit.

And realistically due to the continuous evolution of fuel designs we have seen, if you were to implement a forward-fit requirement, you would probably most likely encompass a majority of the fleet in a reasonable time frame. Most plants aren't using the same fuel they did seven or eight years ago.

MR. RULAND: Could you elaborate a little bit on what do you mean by "requirement" here, Paul?

MR. CLIFFORD: Yes. It is not a requirement. That's good because I get carried away. It would be a change to the standard review plan, which would be guidance to the staff on how to approve a new fuel design when one comes across our desk.

MEMBER ARMIJO: But wouldn't that be a disincentive to innovation?

MR. CLIFFORD: You can ask the industry. They will be here next.

MR. RULAND: And that is something we would like to think about.

MEMBER ARMIJO: But you want innovation that addresses the failure mechanisms. You can't say, "Gee, we are not going to require you to demonstrate that your fuel can survive this kind of expected transient" without demanding that they show you data.

I mean, why would you want them to innovate without addressing failure mechanisms? I don't understand the logic here.

MR. CLIFFORD: Well, look --

MEMBER ARMIJO: Go ahead. We probably have to wrap up, Paul.

MR. CLIFFORD: Okay. I just want to say, you know, as I mentioned earlier, we have what we call our regulatory analysis process. When we put more meat on the bones of these three potential solutions, then we are going to have to follow that process. And that process has a lot of the same features as backfit as far as classification and improvements to health and safety of the public.

And we are going to have to go through those and follow the process. Right now I couldn't tell you how that shakes out, but that's what we intend to do.

MEMBER ARMIJO: Okay. Well, look, thanks Paul. And I am the cause of the delay, but we are going to have to really move fast. I apologize to the industry reps because we are running --

MEMBER SHACK: Let me just make one last comment. You are planning to do a regulatory analysis of the forward-fitting option, then? Is that what your last statement said?

MR. RULAND: As Paul had alluded to, at this moment, no. But our intention is to early next year revisit that decision about exactly how we need to, when we need to do this.

At some point I believe the staff will, in fact, examine this issue and decide, do a regulatory analysis, and kind of move forward. But we are not prepared to make that decision today. So that is kind of where we are at.

You know, we are not looking for a letter, by the way, from the Committee.

(Laughter.)

MR. RULAND: That is clearly your decision. It is clearly your decision.

MR. RULAND: Thank you, Paul.

MEMBER ARMIJO: Okay. EPRI is first.

MR. EICHENBERG: My name is Tom Eichenberg I am the Senior Specialist Reactor Safety, Tennessee Valley Authority. And I am here today presenting industry's perspectives and positions on this particular issue.

A lot of things have been covered already today. So I don't want to get too bogged down in some of this. I want to talk specifically about the industry's view of the safety significance and some perspectives on this entire issue.

To start off with, the industry's perspective here is that the GDC10 does not specifically preclude fuel failure. And part of our reasoning for our understanding of this is reflected in the SRP 4.2, which specifically recognizes that PCI is a special category of things. And there is no

SAFDL which says PCIs are not allowed to happen.

And so, consequently, the industry's view is that current safety analyses are in compliance with all the existing SAFDLs and that our on and off-site dose requirements are met.

One thing which is important to think about here in terms of safety is that the postulated PCI failures during an AOO are not in terms of concerns. As has been mentioned before, if there is a PCI failure, the geometry remains coolable.

For the AOO of interest, which we have discussed in some previous slides here, we were talking about loss of feedwater heating, the primary coolant boundary and containment remain intact.

I think something important to remember about this is that an AOO is typically a single failure event that we analyze. In order to have a complete failure that impacts the health and safety of the public, we have got a multi-failure scenario to get there.

We would like to briefly go over some of our perspectives on ACRS concerns. One concern which has previously been mentioned is what has been called the growing use of non-PCI-resistant fuel. And what we would like to sort of get out there for people is that the licensees have only moved away from barrier cladding in the context of a fuel vendor change. And there are some plants out there which have never used barrier cladding.

One of the key reasons -- and it was alluded to in some earlier discussion -- why people were hesitant to adopt barrier cladding had to do with the secondary failure performance.

Some plants may, in fact, be very susceptible to debris failures. And if your plant is susceptible to debris failures in fuel, their normal operation, you don't want to risk the consequences of secondary failures in your fuel if you don't have to. That is, maybe your particular plant hasn't seen a significant PCI problem. So each utility is looking at it from their perspective of what they have experienced.

MEMBER MAYNARD: I'm sorry. I'm not sure I quite understand what your --

MR. EICHENBERG: Well, what I am trying to get here is that the industry does not believe that there is a growing use of non-PCI fuel or PCI-resistant fuel.

MEMBER MAYNARD: I am just trying to see, is there any correlation between the debris-susceptible plants and barrier fuel? Is there an argument that --

MR. EICHENBERG: Barrier fuel is independent of whether a plant is susceptible to debris.

MEMBER MAYNARD: Okay. All right.

MR. EICHENBERG: And we would also like to get our perspective that barrier cladding by itself is not the only option to achieving PCI-resistant fuel. I can --

MEMBER ARMIJO: I want to make it clear. Is there any other design in commercial service in the U.S. that makes a claim of being PCI-resistant and has the data to back it up? And the answer is no. There is a lot of stuff in development. And I would be happy to --

MR. EICHENBERG: I guess at least with TVA's experience with our transition from General Electric to AREVA at Browns Ferry, in that particular case, we were going from a GE-14 design, which was a barrier clad design, and we were moving to an ATRIUM-10.

And at that time, we were not comfortable with the secondary degradation performance of the barrier design for that fuel product. And so we chose to stick with what some might call the classical design.

MEMBER ARMIJO: But that is not what I asked. You said it's not the only option to achieve PCI-resistant fuel. I'm just saying commercially are you using PCI-resistant fuel or not? And the answer you're --

MR. EICHENBERG: I guess it depends if you --

MEMBER ARMIJO: You're using conventional fuel to classic design.

MR. EICHENBERG: We are using --

MEMBER ARMIJO: And you are not claiming it is PCI-resistant.

MR. EICHENBERG: We are using AREVA's product. And test data for AREVA's product indicates that it has reasonably good PCI resistance compared to what may have been seen 25 or 30 years ago.

MEMBER ARMIJO: If you have that quantitative data, then that should be --

MR. EICHENBERG: That's something that the vendor can speak to.

MEMBER ARMIJO: Yes. Well, we would like to see that.

CHAIRMAN BONACA: Would you explain to me what do you mean specifically by secondary degradation? This is not hydriding.

MR. EICHENBERG: It could by hydriding. It could be additional failures.

CHAIRMAN BONACA: Why? Could you explain to me why the barrier cladding would be susceptible to --

MR. EICHENBERG: This has to do with some of the history in the early barrier designs, which had a zirconium or a pure zirconium liner. And there were severe secondary degradation issues associated with that.

Over time the vendors have dealt with different alloying effects to try and compensate for that. Some vendors use --

MEMBER SHACK: Is this something that happens after you damage the cladding by the fretting?

MR. EICHENBERG: There could be a fretting failure, which allows --

CHAIRMAN BONACA: That is the connection susceptible to failure. Okay. I understand now.

MEMBER ARMIJO: You know, just to set the record straight, the secondary degradation was initially observed in the barrier fuel after years of operation, including operation with failed fuel. Only a very small fraction of that fuel failed, and it split.

The industry addressed that by alloying it, in some cases adding tin to the liner, in some cases adding iron. The idea was to make the ideas corrosion-resistant and hydriding-resistant as the cladding ODE.

What has happened, unfortunately, is that there has been splitting in alloyed iron liner cladding, 3 30-inch-long splits. So that means that the mechanism that was attributed -- now, if the solution doesn't work, you probably have the wrong understanding of the mechanism. So that's clouding the issue.

The issue really is, do you have protection during AOOs with your fuel or not? And if you have data to show that your conventional fuel has that, then you should show that and march ahead. I don't think you do, but I don't know everything you have in your hip pocket.

MR. EICHENBERG: That is something which the vendor specifically will have to deal with because it is proprietary.

MEMBER ARMIJO: Okay.

MR. EICHENBERG: As far as TVA's

experience with this issue of changing from barrier back to what I'll call non-barrier, TVA has recently based on available information from the vendor recently actually gone back to a barrier product, which in the current unit 2 Browns Ferry outage will be the first introduction of AREVA's barrier product into the cores. So all of the AREVA fuel lines are being transitioned to that barrier product.

The reason that that was chosen as a mechanism was because we felt that the incremental improvement in PCI protection from the barrier was now essentially a wash relative to potential problems with secondary degradation.

And so as part of the 0 by 2010 initiative, which I think the numbers you are familiar with, TVA made the choice to go ahead and put the barrier fuel in.

MEMBER BROWN: So, on the one hand, you are arguing against it but saying you are doing it anyway? Did I get that wrong?

MR. EICHENBERG: What we are saying is that there isn't a growing use of non-PCI resistance. People are very, very aware of the PCI performance of the particular products. And there are more issues involved in deciding whether to go to barrier or not than simply the PCI performance. There are other considerations that utilities are taking into account.

Now, my understanding here in the United States is that there are very few utilities who are using the non-barrier designs.

MEMBER ARMIJO: Well, look, I am responsible for the words "growing use of non-PCI-resistant fuel" because I was informed that TVA was going to use that in their EPUs --

MR. EICHENBERG: Right.

MEMBER ARMIJO: -- for the next units 1, 2, and 3.

MR. EICHENBERG: Right.

MEMBER ARMIJO: So that was misinformed or, else, it changed. I don't know. But that is where that came from.

MEMBER BROWN: They are not using it, then? They are going to use barrier-type fuel in 1, 2, and 3?

MR. EICHENBERG: Right. Currently unit 2 and unit 3 use AREVA fuel. And that fuel since the original contract change has been a non-liner clad product. And starting this spring with our most recent unit 2 reload of fuel, we are actually introducing AREVA's barrier clad product.

MEMBER BROWN: Okay. I just wanted to confirm. Thank you.

MR. EICHENBERG: Yes. PCI failure times are short at AOO power levels. We have previously heard discussion about the loss of feedwater heating event as being the only identified as a credible event with the potential for large-scale PCI failures. And we have also heard some previous discussion about control rod withdrawal error, but that is not a core-wide-type event. It is very localized.

One thing that we wanted to make a point about is that the failure conditions in operating experience have systematically been below ramp test data. And this comes right out of the EPRI PCI guideline report.

And the key thing we want to get across here is that test reactor data do not automatically translate into power reactor experience. And test reactor data for one particular product, that metal heating process, does not directly apply to --

VICE CHAIRMAN ABDEL-KHALIK: Loss of feedwater heating is an anticipated transient. So I would imagine over the past several years some of these events have occurred around the world. And they may have occurred at plants which operate at EPU conditions.

Do you have data on what actually happened at these plants during events of this type?

MR. EICHENBERG: My understanding is that there was a recent loss of feedwater heating event at Susquehanna, which seems to be the unit of interest. My understanding also is that that event did not lead to any appreciable significant power increase. So it was not a large --

VICE CHAIRMAN ABDEL-KHALIK: So it's not a loss of --

MR. EICHENBERG: -- full-blown loss of feedwater heating. It was a smaller --

MEMBER ARMIJO: Some of these events are very mild. There was one in a utility I won't name in 1999. And it occurred. They were operating at 70 percent power when it happened. So when they got up to a little bit over 100 percent power, nothing happened.

They all had a barrier fuel. But it took them an hour before they realized that they had a loss of feedwater heater event. And they only knew it because they were making a lot more power than they thought they should be making. So that's not a good --

MEMBER MAYNARD: It was my understanding --

VICE CHAIRMAN ABDEL-KHALIK: I will get to the licensee event report, but I am trying to get to the sort of the basis for this statement, that failure conditions in operational experience are systematically below ramp testing data.

MR. EICHENBERG: Yes.

VICE CHAIRMAN ABDEL-KHALIK: Where does this statement come from?

MR. EICHENBERG: This statement comes from the EPRI PCI guideline report, which was recently issued. It's based on a review of operational and ramp test data. I think EPRI would probably be in a better position to address that specific aspect.

MR. MONTGOMERY: Good afternoon. My name is Robert Montgomery. I am a consultant to EPRI. And I was the author of the EPRI PCI guideline report.

The statement up there that is in question relates to the operational experience of known failures in BWRs that occur during normal operation and comparing the conditions that those rods saw in terms of power, delta powers, burnup with the database of ramp test rods.

And the statement there basically says it, that what you see is that there are differences between the conditions that a fuel rod in an operating power plant sees. And that led to failure versus the conditions that would lead to failure in a ramp test.

And below in this statement means that the power levels that were observed in the operational experience base were generally lower by 10-20 percent maybe than the ramp test data power levels.

MEMBER ARMIJO: Let me make sure I understand, Robert. You are saying that in operation, the fuel failed at lower powers than the ramp test data or at higher powers?

MR. MONTGOMERY: Generally lower powers. And that's because there are other contributing factors than just power and burnup that come into play. Things like pellet quality, as you may be aware, can lead to failure conditions.

What it really translates to is stress, as you alluded to, Dr. Armijo, in your presentation, that stress is the contributing factor. And how you get there will affect the stress.

VICE CHAIRMAN ABDEL-KHALIK: So this sort of pushes the problem in the --

MR. MONTGOMERY: I think the point of the statement that Mr. Eichenberg is pointing out here is that you just can't take the data as is out of the ramp test data and apply it directly to power reactors.

MEMBER ARMIJO: The intention of the ramp test data was to make them very severe so that they would apply because you didn't have large statistics. So you made them very severe.

And you are telling me that, in fact, when the ramp test data predict a one percent probability of failure in operating plants, you would get a higher --

MR. MONTGOMERY: It may be a little higher percentage, but then you have other contributing factors. You've got to factor in things like pellet quality, clad quality, and other variables like that that are not considered in the ramp test data.

VICE CHAIRMAN ABDEL-KHALIK: So these data come from essentially during control rod movements?

MR. MONTGOMERY: Correct, generally during control rod movements, sequence exchanges, as we call them sometimes.

MEMBER ARMIJO: The ramp test data are not necessarily conservative.

VICE CHAIRMAN ABDEL-KHALIK: You are not sort of helping your case.

MR. MONTGOMERY: I understand that.

MEMBER ARMIJO: Okay. Let's keep moving.

MR. EICHENBERG: I just want to quickly address a couple of other issues. Early 10 by 10 mitigation benefit gradually are being lost. The industry doesn't feel that a larger pin array by itself automatically mitigates PCI. And, regardless of whether you are at a 9 by 9 array or a 10 by 10 array, your thermal limits still have to meet your SAFDL requirements.

VICE CHAIRMAN ABDEL-KHALIK: But that is really not the point. By going to 10 by 10, you are reducing your heat generation rate.

MR. EICHENBERG: You are reducing your average, right.

MEMBER SIEBER: Your pin power.

MR. EICHENBERG: You are reducing your average pin power, but there are other things that people do in fuel design space.

MEMBER ARMIJO: But you don't want to reduce it. You want to take advantage of all that fuel. And in modern fuel, you really want to take advantage of it. So you have to solve the PCI problem some other way. That's really what people do. But you could.

When 10 by 10 was first introduced, the claim was, hey, we're going from 13.4 peak power down to 11. And so the difference between the PCI failure threshold and the peak LHGR is so small that preconditioning isn't going to be much of a burden to the industry. It will be inexpensive, and it will be very low-risk.

Well, that was all true, but with time, the economics of putting more -- it's back up to 13.4. So that is the argument, not the actual matrix itself.

MR. EICHENBERG: Another concern that the ACRS expressed was the inadequate analytical capability to quantify risk of failure. And from the industry's perspective, we believe that we have effectively managed the frequency of PCI failures and that we are capable of assessing failure potential and margins.

I believe this concern is the one which gets right to the heart of the matter. The number of fuel elements at risk during AOOs increases proportional to the magnitude of an EPU.

And the industry agrees that the average rod LHGR is increasing, but we don't believe that the risk of failure is increasing proportionately. We believe that the risk of failure is similarly low.

The potential for a rod failure is limited both in time and space. We have space considerations in the sense that the rods, which are going to be in a high-stress state at the start of this proposed loss of feedwater event are those that just occurred after a sequence exchange, where they have been running at deconditioned power for a long time.

And so once you have had your sequence exchange, you also have a time condition because it is going to take you a couple of days to recognition those rods at their new power.

So it is really just this small window of space and time which is going to contribute to any potential failures if you were to have this hypothetical loss of feedwater event.

Another thing which is going on in the --

VICE CHAIRMAN ABDEL-KHALIK: I am not sure about the truth in this statement because in EPU what you try to do is flatten the power distribution in the core.

MR. EICHENBERG: Right.

VICE CHAIRMAN ABDEL-KHALIK: So, even though the average goes up, obviously, because you increase power, you may keep the heat the same. But there are a lot more clogs in the core that are now closer to the feed.

MR. EICHENBERG: They are closer to the feed so that they are at a condition, they are conditioned. And so the delta stress --

MEMBER ARMIJO: Sorry. They are conditioned up to the peak power, --

MR. EICHENBERG: Yes.

MEMBER ARMIJO: -- your LHGR peak power. Let's say your bottom peak beginning of cycle up to 13.4. I know you have margin, but let's just say you're at 13.4.

You're conditioned at 13.4. I don't dispute that. You get a slug of cold water coming into the bottom of the core. You throw in a ton of reactivity. That 13.4 peak is going to move up to a regime of 14, maybe 16. And you're not preconditioned for that. You have never --

MR. EICHENBERG: And we are not saying we are preconditioned for the final power. What we are saying is we are preconditioned for the power at the start of the event.

And depending on whether you are having this event immediately after a sequence exchange or whether you are having this event two weeks after a sequence exchange, the number of pins which are highly susceptible to the failure threshold is very different.

MEMBER ARMIJO: No, no. Just think of the bottom peak core. You are going from 13.4 to 16. You are not preconditioned beyond 13.4. So you are risk for that 13.4 to 16. And that is a very aggressive state of stress, chemistry and stress. So that is what I am trying to get at, is that is where the problem is.

I have no doubt that you are doing a great job during normal operation and preconditioning and so on by a number of techniques. But you can't precondition fuel for operation in the AOO regime. You can minimize. You can do the best you can, but you can't precondition over your licensing limit.

MR. EICHENBERG: Right. And we are not saying that we are preconditioned over our license limit.

MEMBER SHACK: But you are arguing you are getting considerable benefit from that. And the only thing that is at risk is these high-stress ones.

MR. EICHENBERG: What we are saying is that, for example, let's say in a worst-case sequence exchange, you have got 24 control cells which are being removed and you have got some other ones that are going in. So now you have got 24 control cells where you have what is essentially you're operating at the high power but you are not conditioned at the start of the event. So you are not --

MEMBER SHACK: Dr. Armijo will agree you are worse off for those but --

MEMBER ARMIJO: One is not safe from the other one. That is what I am trying to say.

MR. EICHENBERG: And I believe that the analysis which I am going to be discussing here --

MEMBER ARMIJO: Okay.

MR. EICHENBERG: -- actually indicates that you are better off. The particular analysis that I am alluding to here is -- I'll skip one here in a minute.

Some other things that are going on in EPU design are that in order to achieve the kind of cycle energies that an EPU requires, it requires a big increase in your fresh fuel loading. And so you have got a lot more rods in the core which are not in a gap closure condition or at a lower exposure. So, again, you've got another thing affecting you in space and time.

MEMBER SIEBER: And burnups.

MR. EICHENBERG: And burnups. And it contributes to this concept that you can't really change the ultimate failure risk just by EPU by itself because there are other things working in an opposing direction.

Now, because Susquehanna was sort of the impetus for this, it was we'll say fortuitous that the Susquehanna analysis was actually presented at the water reactor fuel performance meeting in Seoul, Korea. And that information is now in the public domain.

What that particular paper discusses is that when the detailed Susquehanna analysis was looked at in the case of the pre-EPU evaluation, it was predicted that you might have something on the order of four to five pins that would meet the stress thresholds for failure and that when they looked at the EPU conditions, that for that specific analysis, the number of pins that were predicted to hit that stress threshold actually went down. Part of the reason it went down is because you have these competing effects.

MEMBER ARMIJO: Well, we would have to see the paper. We will take a look at that.

MR. EICHENBERG: My understanding is that the paper is very similar to the analysis which was shown as part of the EPU submittal.

MEMBER ARMIJO: Okay. Tom, you are really going to have to wrap it up.

MR. EICHENBERG: The general conclusion from this is that industry does not believe that fuel potential failures do not necessarily scale with EPU power uprates.

From an operational standpoint, we just wanted to identify some important aspects. One of them is that the fleet operating history is that with modern fuel and cladding designs, we have not had any AOO-induced PCI failures. And this includes the use of modern non-liner fuel.

CHAIRMAN BONACA: Now, that "extremely unlikely" is highly speculative.

MEMBER SIEBER: You don't have any AOOs.

MEMBER ARMIJO: Well, you know, you have got to see numbers. But, anyway, look, I think that's a good --

MR. EICHENBERG: Just, in summary, the industry does not believe that PCI under AOO was a public health and safety issue. And we believe the standard review plan currently provides adequate guidance. And industry at this time does not see the need for additional regulation in the absence of a public health and safety issue.

MEMBER ARMIJO: Okay. Thank you, Tom.

Look, we have got very limited time. And I apologize again. The AREVA and GE Hitachi want to make presentations. The AREVA presentation is nonproprietary. I had some questions that were proprietary. I won't ask them because we'll be emptying the room, filling the room, and it is unfair to the presenters.

So, AREVA, try not to cover what has already been covered. And same with GE Hitachi. Appreciate that.

MR. PRUITT: This is Doug Pruitt. I am the Manager of Thermal Hydraulics, Richland. I am here as a substitute for the Thermal Mechanical Materials Manager, but I did participate in the PPNL EPU. And I think that is where a lot of this discussion started.

Just as a background, going back to December 2007, the question of PCI and EPUs came up. And at that point, we did offer technical positions with respect to both a fuel mechanical analysis, which was a full-core stress survey for both pre-EPU and post-EPU cycle designs.

And, in fairness to the staff at that point, the tool was a developmental tool. And so you probably discounted it with respect to necessarily believing any absolute numbers in that analysis. That has been 18 months. And that tool is now internally qualified against ramp data and has been internally certified.

Basically two points were made. One was the stress survey that showed a very limited number of rods that might be susceptible to PCI failures. And the other is the rapid response of the PPNL operators in response to the actual event they had as well as a single failure occurred.

A PPNL EPU submittal was approved by the NRC. We have had additional analysis provided to the NRC as well in support of the TVA Browns Ferry plants.

What we see in that analysis -- and, really, basically all the analysis we performed to date says it's not a significant difference between EPU and non-EPU with respect to the number of rods that are in the range of stresses associated with these PCI failures.

Looking at AOOs, this has been covered: loss of feedwater heating, control rod withdrawal error, and flow runups. Basically I will skip down for a time here just looking at the loss of feedwater heating and flow runup events that are core-wide. I think the dominant one here is loss of feedwater heating events.

Because it is slower in nature, there are opportunities for decreasing clad stresses. One is the operator intervention to mitigation the power transient. It is possible as evidenced by plant operators. Now, that's not necessarily a licensing basis, but they will operate within that amount of time. In actuality, it is likely that the design basis analysis and analysis we assumed in the stress survey was 100-degree Fahrenheit reduction in feedwater temperatures.

MEMBER STETKAR: Doug, is the success of the operator performance observed, success of the operator performance, due to the fact that we have not seen what I would call dramatic loss of feedwater heating and instantaneous bypass of the feedwater heaters and that they have been more gradual-type events or is --

MR. PRUITT: You would have to have some of the industry people to tell us what their experiences are with it, but --

MR. HOFFMAN: This is Chris Hoffman from

PPNL Susquehanna.

The general feedwater heating events that we have seen have been partial loss of feedwater heaters. Generally we would lose one heater. We would have temperature changes that were on the order of roughly 60 degrees or less.

For ACRS, we did a demonstration in the simulator of operator response. I don't remember the details, but it was a blind scenario. The operators did not know the scenario that was coming at them. And they did adequately respond to that scenario for ACRS, that subcommittee that was visiting the facility.

MEMBER SIEBER: What's the cause of the loss of feedwater heater: valves or tube rupture or what?

MR. HOFFMAN: I do not know. I am not prepared to discuss that.

MEMBER SIEBER: Okay.

MR. PRUITT: Basically what that analysis showed, what it did is it went through the entire cycle design step-through. And it looked at both prior to the sequence exchanges and just after the sequence exchanges.

MEMBER SHACK: Wait. You said you did it

for a 100-degree temperature change?

MR. PRUITT: Right.

MEMBER SHACK: So that's more severe than their observed 60.

MR. PRUITT: Right. And within that, basically what you saw was that the -- well, the only rods that were challenging to the stress limit were those that had changed the control rods.

And so, just coming back to a previous point on maneuvering criteria, the point that EPRI is seeing is that operating experience and failures occur at a lower LHGR than in the ramp test. Okay? And that is just because of variations. A lot of it had been associated with missing pellet surface and other such things.

So maneuvering criteria are just to try to protect that operational experience.

VICE CHAIRMAN ABDEL-KHALIK: But, again, the point I tried to make earlier that these are lower calculated peak local linear heat rates --

MR. PRUITT: Right. Let me just get to my point first. Then we'll come back.

VICE CHAIRMAN ABDEL-KHALIK: Okay.

MR. PRUITT: So if your maneuvering criteria address this larger margin compared to the ramp database to cover anomalies, to cover variations in clad, variations in pellets and everything else, then there is a margin between that and, shall we say, a systematic failure.

You are actually building more margin because you are protecting the anomalies, the variations compared to what you would experience for a, shall we say, sound rod.

VICE CHAIRMAN ABDEL-KHALIK: I'm sorry. You lost me.

MR. PRUITT: Okay. Just basically, maneuvering criteria set the initial conditions prior to the AOO. They're going to do a sequence exchange. An they're going to abide by the conditioning criteria. So essentially that sets the maximum stress that can exist anywhere in the core prior to the AOO. Okay?

And if the failure threshold is based on the ramp testing, that is a much higher threshold than what we would base off commercial operation, which has been shown to be lower. Okay?

MEMBER ARMIJO: Doug, I really have to ask you to pick your best slides because we are going to try and wrap up at 3:00 o'clock. I will probably ask my colleagues to give me five minutes at the break.

So to give GE ten minutes?

MR. PRUITT: Okay. Let me just say there's a combination of what is the ramp failure criteria and what is your maneuvering criteria? And both of those work together. Fuel mechanical analysis indicates very few rods, if any, exceed the stress limit established from the ramp test for AOO.

PCI failures can occur when high clad stresses develop in a corrosive environment. There are a number of design features that have been developed and successfully employed for many years to reduce clad stress, including the pellet design; optimization of the cladding materials, heat treatments, optimization of the liner cladding; and elimination of missing pellet surfaces. All those things make fuel failures less probable. And we are now developing and implementing doped fuel pellets as well.

So there is a combination of design elements as well as maneuvering criteria and operator actions that all work together to minimize the probability PCI.

Use of optimized cladding material with or without liner provides protection against PCI by assuring the cladding can withstand anticipated stresses. And within the existing ramp test data, there are some suggestions that some non-liner cladding performs similarly to liner cladding.

MEMBER ARMIJO: Have you shared that data with the staff? I mean, that is revolutionary. If you can get conventional cladding to perform --

MR. PRUITT: Okay.

MEMBER ARMIJO: If you've got data to show that, that is a huge accomplishment. I would like to see it because that is not typical.

MR. PRUITT: I think both the analysis as well as evaluation of this type of material is something for a different venue.

Okay. So we produce both SRA, stress relieved annealed, and recrystallized cladding. We offer them both within liner with iron-enhanced zirconium. And both can be optimized for effective fuel performance because certainly what we are analyzing with 100-degree Fahrenheit change is the stress relieved cladding.

Back to the time-to-failure issue. We have looked at it from the perspective of what is the change in LHGR from the ramp database versus time-to-failure. We have also looked at the change in LHGR associated with an AOO event.

And so basically you are looking there from a conditioned state because assuming that the ramp data is run from a conditioned state, then you would see a loss of feedwater heating, the kinds of kilowatt-per-foot increments would give time for operators to take action.

MEMBER ARMIJO: What is that time that you are talking is adequate time? Have you got a number?

MR. PRUITT: Twenty, 30 minutes. Okay? What we see is the two or three-minute type -- those are one-minute to three-minute-type failures -- are associated with the large changes in LHGR, more like the rods that have come from a controlled state to an uncontrolled according to maneuvering criteria and then undergo a loss of feedwater heating. That is going to be the five, six kilowatt-per-foot-type changes, not the two pill of these types changes.

MEMBER ARMIJO: So you are saying with, say --

MR. PRUITT: I think it would look --

MEMBER ARMIJO: -- 13.4 to 16 more or less, you've got 20 minutes based on your data.

MR. PRUITT: Based on the ramp, you know, the international ramp database, right. I think if you recast your data in that fashion, it would look a little bit different.

MEMBER ARMIJO: Okay. We'll look. I would love to see the data.

MR. PRUITT: Okay. Yes. And data exchange is a good idea. So ramp test data short times may be due to overly conservative test conditions.

The conclusions. We don't believe AOOs present a potential for a large number of PCI failures and are, therefore, not a safety issue. There are numerous ways to prevent PCI failures from product design features, material characteristics, and operator control. Industry is actively pursuing these features independent of regulation due to desire to further decrease the potential for fuel failures. And regulation of specific product features in order to prevent PCI is not appropriate.

MEMBER ARMIJO: Thank you.

And then we have got to get GE on. I am going to ask the indulgence of the Committee for five minutes on the break. So, GE, you have got until 5 after. So take your best charge.

This is proprietary. So we will have to ask AREVA and the other folks --

(Whereupon, the proceedings in the foregoing matter went off the record at 3:17 p.m. and went back on the record at 3:31 p.m.)

CHAIRMAN BONACA: Back into session.

The next item on the agenda is Diversity and Defense-in-Depth Topical Report, and that is -- Mr. Maynard will take us through that presentation.

MEMBER MAYNARD: Okay. Thank you, Mr. Chairman. Our subcommittee met on this a couple of weeks ago and aired this item out in detail. This is to review an SER that the staff proposes to issue relative to a topical report submitted by Mitsubishi.

MHI submitted the topical report in 2007 for their defense-in-depth and diversity approach using their digital I&C platform, MELTAC. And they were proposing that for use with -- in the US-APWR as well as being able to modify the existing operating fleet. Recognize that this was submitted during a time, actually before a lot of the guidance was developed and reviewed during a time when some of the guidance as to what is acceptable was being developed.

Their desire, of course, was to get approval of their methodology on a generic basis, so they could then apply it on a plant-specific basis. After some of the review, the staff determined that they would not be able to review this relative to the existing operating fleet on a generic basis and really limited their review to review for the US-APWR. And it is the methodology used. It is not all of the detail for the entire diversity and defense-in-depth, but it is the generic methodology for the US-APWR.

In so doing, this topical report, it doesn't have all of the detail to be able to make final decisions on all of the aspects of diversity and defense-in-depth. Therefore, the staff has documented within the SER what's called application-specific action items, and those are the areas where there is more detail needed, either in the design, or more information to address the specifics for the US-APWR to finalize this. They are documented in the SER, and those will be finalized during the DCD review, those -- where there is additional detail.

Now, it is possible some of those may then be carried over as an action item for the COL stage, but at this time it is to be determined at the DCD stage.

I think the staff may go back over a little bit of this history, and I don't think that is bad, because sometimes you need to kind of get this twice to understand kind of what the topical report is doing. So --

MR. BEACOM: I think you might have to. Yes.

MEMBER MAYNARD: With that, I will turn it over to Royce and let him lead us through the presentation.

MEMBER ARMIJO: Just one quick question.

MEMBER MAYNARD: Sure.

MEMBER ARMIJO: Initially, the topical report was submitted to have a broader application?

MEMBER MAYNARD: Right. What they wanted to do was to use their MELTAC platform and be able to get the methodology and the generic aspects of that approved in the diversity and defense-in-depth arena for not only US-APWR but also if they were able to sell it to other powerplants for modifying the existing fleet. If somebody was upgrading to a digital I&C platform --

MEMBER ARMIJO: They would use as a --

MEMBER MAYNARD: -- they would have the approved topical report for that.

MEMBER ARMIJO: Got it.

MEMBER MAYNARD: The staff determined that that was too much to be able to be accomplished with this, and they limited it to the US-APWR, the methodologies for that. So let's turn it over to --

MR. BEACOM: Okay. Thank you, Mr. Maynard.

My name is Royce Beacom. I'm in the Office of New Reactors, Instrumentation and Control Branch. I've been in the Office of New Reactors I&C since the inception of NRO, before that a couple of years in operating reactors, and before that outside the agency at Westinghouse for 20-some years. So I have had some digital I&C experience.

And we have had some very good comments and questions by the subcommittee at the previous two meetings, and I look forward to the full -- a full committee meeting and your comments.

So as far as the agenda goes, we are going to look at the diversity and defense-in-depth scope for both the topical report and the SER, what the SER found as far as findings and conclusions were. We are going to look at the complete listing, as far as what the staff took away of the subcommittee points of discussion from the previous two meetings about two weeks ago. And then, we are going to try to address each point of discussion. No guarantees if we are going to complete it, but we will take away as much as we can and try to get an answer for you.

Now, as far as the diversity and defense-in-depth, D3 as we like to call it, scope, the intent here is to identify diversity with the safety and non-safety systems, that it being the protection and safety monitoring system is a safety system, a PSMS. The plant control monitoring system is a non-safety, the PCMS.

Both of these systems, which is a considerable portion of the plant, are using the MELTAC platform. Now, the premise here is that a common cause failure disables both the safety and non-safety system due to the commonality of the MELTAC platform. It is the most conservative and best way to go.

Now, we are going to discuss, in the functionality of the diverse actuation system, the DAS as we commonly refer to it. It's an analog, conventional -- sorry.

MEMBER MAYNARD: The MELTAC platform, there is another topical report that goes into the full details of the MELTAC system, and so that is really not part of this topical report. Now, see, although you have to understand parts of it in order to move forward here, so --

MEMBER APOSTOLAKIS: Sorry for the question. But has the staff approved the topical report?

MR. BEACOM: No, we have not. It is in the middle of review, as is part of the other topical reports, and --

MEMBER APOSTOLAKIS: Why are we here today, then?

MEMBER MAYNARD: Okay. That was for the MELTAC system. I'm saying this report they are finalized. I mean, this is basically a draft final SER that we are reviewing today, just for the diversity and defense-in-depth.

MEMBER APOSTOLAKIS: And we'll write a letter on this?

MR. BEACOM: Yes.

MEMBER MAYNARD: That's what -- that's what --

MEMBER APOSTOLAKIS: Tomorrow, whatever.

MR. BEACOM: That's the intent, yes.

MEMBER APOSTOLAKIS: Okay.

MEMBER MAYNARD: All I was saying is that part of the system that we are going to be talking about, the MELTAC system, that system is part of another topical report that -- the design of that will be reviewed separately.

MR. BEACOM: That's correct.

MEMBER SHACK: Just now we have wiped it out with a common cause failure, though.

(Laughter.)

MEMBER APOSTOLAKIS: Did we ever have such a topical report for other designs?

MR. BEACOM: For D3, no, we have not.

MEMBER APOSTOLAKIS: So why is this unique?

MR. BEACOM: Well, this is unique -- as far as addressing all of the requirements, guidelines, and criteria in one topical report, and applying it to, as much as you can, a topical report being without a specific application versus a technical report with a specific application in mind, but a topical report, we try to apply all possible criteria and guidelines and requirements to as many applications as possible.

MEMBER CORRADINI: Can I try it a different way? Is it my interpretation that this, and then the DCDC, they will be no need for a DAC for digital I&C. The anticipated thing is there will be no DAC for a digital I&C.

MR. BEACOM: Well, okay -- well, in the scope of diversity and defense-in-depth.

MEMBER CORRADINI: Ah.

MEMBER MAYNARD: This is just addressing

diversity and defense-in-depth. This is not addressing the overall acceptability of the digital I&C system.

MEMBER CORRADINI: No, I understand that.

MEMBER MAYNARD: Just the diversity and defense-in-depth issues. And, again, I mentioned earlier this was submitted before some of the current guidance was in place. I believe that MHI was trying to get the methodology approved. Again, they would like to have been able to apply this on a more generic basis.

The staff review was limited to just the US-APWR, and if it was all starting over from scratch again, this would probably end up being a technical report for part of this review, since it is applying to this specifically.

MEMBER APOSTOLAKIS: Right.

MEMBER MAYNARD: But it was submitted back in 2007, and it is referenced heavily in the DCD. So --

MR. BEACOM: And to their credit, yes, advanced submittals of topical reports provides the staff with an advantage of reviewing as much as possible before the actual design certification came in for review.

MEMBER APOSTOLAKIS: So there will be a similar session and discussion for the ESBWR?

MEMBER MAYNARD: I don't know how they are going to -- I doubt that. I think they are -- they plan to have a DAC, I believe.

MEMBER CORRADINI: Well, that's why I asked the question the way I did. Maybe I misphrased it, but --

MEMBER MAYNARD: I don't believe it's -- MHI's desire was to not have any DACs. Now, whether that occurs or not --

MEMBER CORRADINI: Okay.

MEMBER MAYNARD: -- but they were trying to do it without --

MEMBER CORRADINI: But with that as its goal, there is much more detail coming up front here that you guys have a chance to look at and review, this being one of the key functions within the digital I&C.

MR. BEACOM: Yes, it is.

MEMBER CORRADINI: Thank you. That helps.

MEMBER BROWN: Yes, there is much more detail in this than there is in the ESBWR.

MR. BEACOM: Very good, Mr. Brown, that's right.

MEMBER BROWN: Pardon?

MR. BEACOM: That's very -- that's true.

MEMBER BROWN: Well, having sat in with Michael on those sessions, and I made the observation then it was detail, non-specific.

MR. BEACOM: And to their credit, they tried as much as they could to get as much information in here and to get away -- steer away from the DAC issue and see how far they have come by this description.

So one of the key issues in diversity and defense-in-depth, this is diverse actuation system as we mentioned. It is analog, we will say, conventional, no software involved, non-safety system. It provides a defensive measure to cope with anticipated operational occurrences and postulated accidents concurrent with a common cause failure in the PSMS, of course, that we know is beyond design basis.

The diverse actuation system also provides the ATWS mitigation function, ATWS being anticipated transient without scram. In current operating plants, there is a single system that does that. In the DAS, it provides the ATWS function and the common cause failure mitigation.

So it also provides automatic actuations, the DAS does, where time is insufficient for manual operator action.

Now, MHI is proposing less than 10 minutes from a given event, that there should be automatic actuation versus manual actuation. There is also delayed -- there is a delay function within the DAS that from a given set of anticipated conditions where PMS should trip there is a delay timer, allowing the PMS to trip, and, therefore, the DAS does not.

The DAS is blocked from actuation, if there is proper PSMS actuation. This is done by feedback from the actuator components back to the DAS saying, "We have already actuated from the PSMS initiation. Therefore, DAS, you don't have to tell us to initiate."

Now, the DAS also provides manual actuation as well. That is done on a separate HSI panel with conventional controls and indications. MHI is proposing in less than 30 minutes, but greater than 10 minutes for the -- where automatic actuations are done, so that between 10 and 30 minutes they are proposing manual actuation from the prompting alarms.

There are isolated signals from sensors, which are normally shared with the PMS -- PSMS and provided to the DAS. Now, the DAS output is -- goes to the discrete person or power interface module. The power interface module has an input from the DAS and also from the PSMS. At -- I'm sorry.

MEMBER APOSTOLAKIS: DAS manual actuation proposed for less than 30 minutes.

MR. BEACOM: Yes. From a given event, we are proposing that the operators --

MEMBER APOSTOLAKIS: But greater than 10.

MR. BEACOM: But greater than 10.

MEMBER APOSTOLAKIS: Why can't they do it at 40 minutes? I don't understand the intent of this.

MEMBER MAYNARD: Well, the 30 minutes is because of the current -- of the guidance on the 30 minutes. It is only to count on manual actions after 30 minutes, but the current guidance says less than 30. So that's what -- they are just addressing the less than 30 minutes.

MEMBER BROWN: But ISG-02, which would do -- that came out after it, I think that is the point. That came out after you all had already -- correct me if I'm wrong -- developed their system. So now they are --

MEMBER APOSTOLAKIS: Okay. So the system --

MEMBER BROWN: So it was -- so they are addressing the fact that they actually are doing between 10 and 30 minutes, but --

MEMBER STETKAR: It is less confusing if you just say they are trying to take credit for manual action when the time is greater than 10 minutes. Okay?

MEMBER BROWN: Yes.

MEMBER APOSTOLAKIS: Then, I wouldn't have asked the question.

MEMBER BROWN: Which is inconsistent with ISG-02, which says 30 for the manual actions.

MEMBER APOSTOLAKIS: Well, that's a guidance of --

MR. BEACOM: That's correct. That's correct. And as we see further on here, we'll see if they bring in sufficient analysis and information. The staff can't accept it for less than 30 minutes.

So in the findings and conclusions in the SER, the one finding we found was with regards to the large break LOCA coping strategy. And that -- MHI's position was essentially three points. Their coping strategy dealt with the MELTAC platform being a high quality, high reliable platform, which has over 20 years of history in Japanese operating plants.

There is a very low -- very low, it should say -- frequency of AAO and PA events, and this is all supplemented with a diverse actuation leak protection.

Now, the staff found that strategy unacceptable for several reasons. The frequency for AAO and PAs are still a finite possibility. A software latent defect is still a possibility, and that must be considered concurrent with a common cause failure with an AAO or PA.

Leak before break doesn't apply here. Leak before break has been authorized for a very narrow application. That is, particularly for consideration of dynamic effects for pipe ruptures. This is discussed in the NRC inspection manual, where the definition of "leak before break analysis" is done.

The staff has revised the standard review plan in Chapter 7, the latest revision done a couple of years ago that is consistent with the very narrow application of leak before break.

Now, we go more into the discussion of the protective action, manual versus automatic. So, again, MHI was targeting the less than 10 minutes for automatic. Greater than 10 minutes, they want to say manual action is assumed.

This differs from the digital I&C Interim Staff Guidance 02 where the guideline, the recommendation, not a requirement, is less than 30 minutes where from a given event there should be automatic actuation.

Now, the licensee or the applicant can bring in sufficient analysis and information to provide why they believe less than 30 minutes for manual is feasible. But in this particular case, there wasn't sufficient information to assess the manual action between 10 and 30 following an event. That particular justification will be done in the US-APWR HSI certification.

Now, in its conclusion, the staff found that the D3 approach and the D3 analysis provided per NUREG-6303, which is the method for performing D3 analysis of reactor protection systems, had met the acceptable basis for conforming the requirements and supporting industry standards. And to their credit, MHI did a much --

VICE CHAIRMAN ABDEL-KHALIK: But on the issue operator actions between 10 and 30 minutes, they are based on prompting alarms, and my recollection is that these alarms are prioritized. How are the alarms prioritized? Is there software involved?

MR. BEACOM: No. They are doing it from -- the DAS is strictly -- and the HSI panel is strictly a -- yes.

VICE CHAIRMAN ABDEL-KHALIK: An analog system. So are these prompting alarms prioritized in any way?

MR. BEACOM: That I can't directly -- I cannot directly answer that.

MEMBER MAYNARD: I think MHI can.

MR. SCAROLA: Yes, thank you. Ken Scarola from MHI. The alarms that we are referring to here are only processed by the diverse actuation system logic, which is fully analog. They are inherently prioritized because they come from a different panel inside the main control room. We have a diverse HSI panel, which is kind of tucked, you know, in a special corner of the main control room. Those of you who have been to the simulator saw that.

Those alarms, both the visual and the audible, are distinctly from that panel, and the operator training is exclusively, when you get that alarm it is a knee-jerk reaction to go to that panel and execute the required EOP actions in a very timely manner. So in that sense, they are inherently prioritized, because they are from a different location in the control room, a different visual, a different audible.

MEMBER BROWN: "Prioritize" sometimes can have a different meaning. You've got a bunch of alarms coming into a microcomputer, and it decides which ones will be told to the operator. This is not a circumstance. If all the alarms and all the red lights needed to come on at one time, they will all come on at one time.

MR. SCAROLA: Right.

MEMBER BROWN: Is that what --

MR. SCAROLA: What makes these a higher priority is simply the inherent distinction inside the control room and the operator training.

MEMBER BROWN: Yes. The operator will respond. He has a specific, immediate action to be taken. At least that is based on the explanation in the subcommittee meeting as well.

MEMBER MAYNARD: John?

MEMBER STETKAR: To try to perhaps cut off some of this discussion, I think a lot of this concern seems to be addressed toward what MHI describes as the coping analysis for the actual ability of the operators to respond within some period of time. Is that right? If I can ask you that.

MR. SCAROLA: Yes.

MEMBER STETKAR: Okay.

MR. SCAROLA: But that coping analysis is dependent upon the prompting alarms.

MEMBER STETKAR: That's right.

MR. SCAROLA: Prompting alarms are --

MEMBER STETKAR: It is part of -- the coping analysis must demonstrate that the operator can perform the action within the available timeframe, given the availability of cues to perform that action.

Now, the coping analysis -- I'll get to the point here. The coping analysis is part of what Royce has in his last bullet there, that it is not part of what is being reviewed under this topical report. Is that correct?

MR. BEACOM: Okay. Coping analysis, right, it's not part of this topical report. It is part of the DCD SER review. However, as we are looking at here, the HSI certification is a Chapter 18 issue. Okay? That's something different again.

MEMBER STETKAR: Well, I just wanted to get it on the record that a coping analysis or the HSI is part of this -- a good review of this topical report.

MEMBER MAYNARD: A couple of things.

We're getting some things mixed up here. You know, the human factors engineering review and stuff is going to get lost here. This is not being approved by the staff at this time. This is one of the application-specific action items, and so all of these questions and issues will be dealt with during the DCD review stage. That's when that will be addressed.

So the SER really doesn't take any position on the acceptability of this, of the operator actions point. That is to be done at the DCD review.

George, did you have something?

MEMBER APOSTOLAKIS: I must say, this is not clear to me, but maybe it's not my fault. But what I also didn't understand is I --

MEMBER MAYNARD: That's clearly my fault.

(Laughter.)

I don't know whether to feel good or bad about that. I don't know.

MEMBER APOSTOLAKIS: I feel a little bad, but maybe it's because I have --

MR. BEACOM: Well, stick with this now. We are going to bring up this issue several other times. We are going to bring up how the technical reports and the topical reports fit into the DCD review and certification. So stick with it.

VICE CHAIRMAN ABDEL-KHALIK: If later on, after going through the analysis of this certification, this later certification, the staff just finds that operator actions between 10 and 30 minutes to be unacceptable, does this negate what we are doing here?

MR. BEACOM: Well, at that point, yes. At that point, then, MHI would have to come up with an alternative.

MEMBER MAYNARD: It doesn't negate what is being approved or looked at here.

MR. BEACOM: Right.

MEMBER MAYNARD: Because this isn't approving or disapproving the acceptability of the manual operator actions less than 30 minutes. It may mean very well that they would have to do some modifications, or they would make changes, or whatever. But that is really no different than any other design would have to do at that stage if it is found to be unacceptable during the DCD review.

MR. BEACOM: That's correct. Most designs are still at that stage where they are determining times for manual action.

MEMBER MAYNARD: So the SER is just approving those generic and methodology -- the things that can be pinned down in --

MEMBER APOSTOLAKIS: So this is saying, for example -- I mean, I don't know, but is it saying, "Yes, we're going to look at what you propose for actions between 10 minutes and 30 minutes"?

MR. BEACOM: Yes.

MEMBER APOSTOLAKIS: And we're going to make a decision later.

MR. BEACOM: Yes, that's what it's saying.

MEMBER APOSTOLAKIS: Oh.

MEMBER MAYNARD: That's what the action -- that's what the application's specific action item says.

MEMBER APOSTOLAKIS: Why do I need this in order to say that? I mean, you could do that without this report.

MR. BEACOM: Yes. That's a fair question, because, yes, it is in the HSI certification where they will review extensively --

MEMBER MAYNARD: I think the reason that it's here is because the topical report took credit for this.

MEMBER APOSTOLAKIS: Yes, yes.

MEMBER MAYNARD: The staff SER has to take a position on that, and so they have to address --

they can't just ignore it, or it becomes approved by default there.

MEMBER APOSTOLAKIS: But what you said earlier -- I want to make sure I understand that approving this does not commit the agency to anything.

MEMBER MAYNARD: It does not commit the agency to approve manual operation in less than 30 minutes.

MEMBER APOSTOLAKIS: Does not.

MEMBER MAYNARD: It does not. But it does --

MEMBER APOSTOLAKIS: Does it permit the agency anywhere else?

MEMBER BROWN: Does it what?

MEMBER APOSTOLAKIS: Does it commit the agency anywhere else?

MEMBER BROWN: Well, I think conceptually.

PARTICIPANT: No.

MEMBER BROWN: Well, yes, they presented a concept for what they want to do here.

MEMBER APOSTOLAKIS: Can they come back and say, "You guys approved this in June. When you do the design certification thing, now you must approve what we are saying here"? Can they do that? Is this such a serious document, or is it just talk about it and prepare the ground? I don't understand.

MR. SCAROLA: Excuse me. This is Ken Scarola. I just want to point out, because I think there is some misunderstanding here.

MEMBER APOSTOLAKIS: I'm sure there is.

MR. SCAROLA: The ISG-02, which was the first one to be issued on this particular topic, establish the 30-minute guideline, the industry had real difficulty with this, because it seemed to be an arbitrary number. As a result of that, TWG-5, which is the human factors TWG, produced ISG-05, which says there is no line. Here is a methodology for demonstrating the acceptable manual actions.

MEMBER APOSTOLAKIS: Sure.

MR. SCAROLA: So it was the intent all along of MHI to follow the ISG-05 methodology before ISG-05 was written. Again, they submitted this topical report the early part of '07. 05 was not written until '08, so there was a timeline here. The intent was to get a methodology in place that would allow these actions.

MEMBER APOSTOLAKIS: I understand that in the context of this manual action, and then I had a broader question. Is there anything else in this report that will create a commitment on the part of the agency?

MR. SCAROLA: Well, that's my concern.

MEMBER APOSTOLAKIS: Well, Bill, you are shaking your head. Is there?

MEMBER SHACK: Yes.

MEMBER MAYNARD: Yes.

MEMBER SHACK: Except for these application-specific action items, they are saying that they have satisfied the D3 requirements.

MEMBER STETKAR: The key that you are looking for, George, is, is the SER written with all of the appropriate caveats in it related to application-specific --

MEMBER APOSTOLAKIS: Yes.

MEMBER STETKAR: -- deferrals?

MEMBER APOSTOLAKIS: And what's the answer?

MEMBER POWERS: I'm not going to tell you.

(Laughter.)

MEMBER BROWN: George, it's hard to figure that out in some circumstances. Okay?

MEMBER SHACK: That's what we have come to a conclusion on. We have to write a letter. That's -- we have -- that's the whole point of our letter, George, is whether they've got enough caveats.

MEMBER CORRADINI: I mean, just from an understanding standpoint, since I -- I was trying to understand this, and I thought I almost had it there for a minute. This is a small piece of the complete design that we have yet to see officially, and this -- what I -- at least how I understand it is you are saying the one attribute that digital I&C must have of diverse and defense-in-depth is -- might be acceptable here. And then, after that point, we must wait and see the next set of steps beyond that, correct?

MR. BEACOM: Right.

MEMBER BROWN: Hold it. Stop, guys. Just a minute. If you look at the SER, just to make this clear, there is a table at the end which the staff has identified. What is it? Is it 11 application-specific items which says -- and they address those by SER, you know, section, which says, "These items are covered in the report, but yet more action is required at some other stages of the evaluation in order to say they are okay."

The other stuff that is in the topical report will be accepted if we write a letter saying it's okay. That's my understanding.

MEMBER BLEY: Although we had a bit of a discussion, and they pointed out -- and this is the caveat -- anywhere you use this you have to meet the underlying assumptions that are here. If the particular place you are using it is somewhat a different situation, then this isn't official. You have to look at it again.

VICE CHAIRMAN ABDEL-KHALIK: I guess I want to put George's question in different words. The question is, in my mind, does a system that provides automatic actuations for actions that would require -- for actions requiring actuations -- automatic actuations in less than 10 minutes, does that meet the D3 requirements? Are we committed to this 10-minute constraint?

The concern I have is that if we come back and say the justification that they will provide for manual operator actions between 10 and 30 minutes turns out to be unacceptable to the staff, does that negate what we approve today?

MR. BEACOM: Yes.

MEMBER CORRADINI: That's why I asked the question, that all I hear now is the qualitative attribute looks okay until we see the details. And if we see the details, we don't like the details, everything essentially is off the table. That's the way I interpret --

MEMBER MAYNARD: With regards to specific items that they identify. Let me give you an example.

MEMBER CORRADINI: But, I mean, I think -- I'm just trying to rephrase what Said said, which is if they -- if right now the attribute of less than 10 minutes is automatic, 10 to 30 is manual with some sort of thing, beyond 30 is status quo, and right now, given these set of attributes, we feel comfortable. Later, if we see the details and we are not comfortable, that doesn't mean we are committed. That's what I think he just asked.

MEMBER BROWN: And that's the -- there's about seven or eight items right now that take credit for manual action between zero -- between 10 and 30 minutes. And if the analysis that they provide doesn't satisfy the staff relative to whatever the criteria are, say, "Hey, we don't think you guys can respond that fast," then those items -- something has to be done with those. They have to come automatic or some other --

MEMBER STETKAR: The DAS for the US-APWR DCD would require --

MEMBER BROWN: A modification.

MEMBER STETKAR: -- a modification.

MEMBER BROWN: Or do some other action to make it appropriate.

MEMBER APOSTOLAKIS: Can we have this gentleman say his piece?

MR. SCAROLA: Yes, I would like to make a point, because I think it's very important here. In this report we describe the automated systems, for everything that is fully automated in less than 10 minutes.

In the event we have to conclude that we need more of this same type of automation, because we are not able to justify manual actions, then this topical report establishes the design methodology, the design description, for all of the supplementary automated functions, the method of ensuring that we don't have spurious actuations, the method that assures that we can sustain fires, the method that ensures that we can do main control room evacuation.

So there is -- the intent of defining 10 minutes in this report was to establish the basis of the automation that we do describe. If more automation is needed, it follows the same concepts.

MEMBER BROWN: And that's why I said earlier this has concept-type stuff in it as well, because it is --

MEMBER MAYNARD: And I will point out that this was a topic of discussion at the subcommittee meeting that we dove into pretty good as to, are we approving something that could then cause us problems later? You know, are the right caveats, are the right action items specified here? And are we getting tied down into something that -- so --

MR. SCAROLA: Can I address one more comment, which is, how can we approve this without seeing all of the digital I&C? I think it's important to understand that there are four MHI topical reports. Three describe the digital I&C. This one starts with the assumption that everything that is in those other three fails, so you don't have to know about those other three. We start with the basis that everything that is digital fails, and then this is how we cope with that failure.

So there is no need to understand the rest of the digital I&C system, only to understand that it doesn't work.

MEMBER CORRADINI: So if I can just say it back to you. That's very helpful, at least for me. All right? Which is, then, what you're really saying is that if staff -- just say the "if." If staff is okay with the 10-minute demarcation, then the details here are specific enough that they should know what the design is. If they are not comfortable, and 10 becomes more than 10, you have put in concepts here that would essentially take the detail design and essentially require additional modifications, to go out to something greater than 10 in detail. Have I got it right?

MR. SCAROLA: Absolutely.

MEMBER CORRADINI: Okay.

VICE CHAIRMAN ABDEL-KHALIK: I'm not sure if the assumption that everything digital will fail represents -- that we can look at this with that assumption, because partial failures can actually lead to much more complex scenarios.

MEMBER MAYNARD: And there is application-specific action items for them to come back at the DCD stage and address partial failures within the digital system. That's another thing documented in the SER.

MEMBER POWERS: It kind of begs the question, doesn't it?

MEMBER MAYNARD: That's correct, and that's --

MEMBER BROWN: Item 5-9 talks about addressing partial failures. That's the way I read that one anyway. It says, "Shall address partial failures in the PCMS and PSMS."

MR. BEACOM: That is one we took away with, and that is something that both the industry and the staff are going to be -- are going to have to deal with, because it has not been dealt with, particularly as far as coming up with a method or methods to bound that. We will get to that as one of the items that we wanted to look again at.

MEMBER APOSTOLAKIS: Dana, what question does this beg?

MEMBER MAYNARD:

MEMBER POWERS: It begs the question of partial failures. I mean, the -- I mean, it's the same question that we have in fires. Things work, things don't work, things work, but they work badly. And it's that badly part that is just very, very difficult to handle.

MEMBER BLEY: Especially if we're talking about manual actions going on under that condition.

MEMBER POWERS: That's right. Now suddenly your errors of commission become paramount here.

MEMBER BLEY: It's a big deal.

MEMBER MAYNARD: Again, that is an application-specific action item that MHI will have to address during the DCD review, and the staff will review it. The approval of this SER does not finish that subject off. That is something that is still to be reviewed.

MEMBER POWERS: But the contention was that I didn't have to understand the other -- the rest of the digital system. It is not clear that that's absolutely true. It may be mostly true. It's not obviously false, but it may be somewhere in between.

MEMBER MAYNARD: You're right, because you do have to understand it to the point to be able to determine whether the DAS system truly is diverse, and is it truly --

MEMBER BROWN: I don't work for MHI, okay, but --

MEMBER APOSTOLAKIS: I hope not.

(Laughter.)

MEMBER BROWN: My point being is that while the partial failures issue is one that has to be addressed, if you maintain -- this is a philosophical thought process. If you ensure your total independence between your diverse actuating system and all of the other stuff that is in there, and they are not completely diverse, they happen to have one point of commonality, and that's in the detectors and sensors. Correct me if I'm wrong, at least that's one of them. I may remember the other one if I can -- pardon? Oh, they all -- pardon?

PARTICIPANT: PIF at the bottom.

MEMBER BROWN: Oh, yes, the PIF, right. I forgot about that one. Thank you. But that is where maybe some of this, you know, partial failure stuff -- if you look at all of the other stuff in between, they'll just have to go walk through it piece by piece and show why those partial failures are, you know, truly independent and won't affect the diverse actuation system. And if they can do that, then you'll be okay.

I don't think -- I just don't think the way the system is laid out it is going to be as hard to do the partial failures. It is just it hasn't been done. So I'm not trying to say we've got to accept it; that's not the point. It's just that I don't -- you asked me that about the ESBWR system. I, you know, just throw it up the air. There is absolutely no idea.

Pardon? John?

MEMBER MAYNARD: Let's go ahead and move on, and see if we need to come back to some of these after we've gone through more of the findings and conclusions.

MR. BEACOM: So I think we're at the -- yes, we're at the subject -- yes, the findings and conclusions concluded the D3 approach was good. They had met the acceptable basis for conforming to requirements and standards subject to the satisfactory completion of the application-specific action items.

Now, one of the takeaways the staff had from the subcommittee meetings was to go further into how the technical and topical reports fit into the overall US-APWR design certification and the ASAIs. So in a couple of slides here we will get further into application-specific action items.

So listing the subcommittee points of discussion that we had a couple of weeks ago, and these are what the staff felt we came away with -- these questions. Now, how D3 fits in the overall US-APWR. Should there be a separate approval of D3 from the US-APWR?

Bypassing the DAS, how that is done with the PSMS actuation and during startup and shutdown. Concept of just two DAS subsystems. The ASAI, the application-specific action item on partial output failure from the common cause failure, what we were just talking about.

MEMBER APOSTOLAKIS: So D3 refers to this report. Is that what you mean?

MR. BEACOM: Yes, sir. D3 -- diversity and defense-in-depth refers to this report.

MEMBER APOSTOLAKIS: This report.

MR. BEACOM: Okay.

MEMBER APOSTOLAKIS: Not a general D3, correct?

MR. BEACOM: That's correct.

MEMBER APOSTOLAKIS: Yes.

MR. BEACOM: That is a good point.

MEMBER APOSTOLAKIS: All right.

MR. BEACOM: And then, the three DAS inputs to reactor trip and turbine trip.

Now we will go into discussions on each one of those points. How D3 fits into the overall US-APWR. Now, as was mentioned before -- yes, as Mr. Maynard mentioned before -- this was originally intended for not only new reactor design, but the operating fleet also. That was MHI's intent.

Now, during the initial acceptance review, the staff made the determination that the diversity attributes -- and I think we mentioned them -- design, equipment, function, human, software, and signal -- I think that's six -- are unique for each operating plant.

And since there was no licensee or applicant having an upgrade to use an MHI/MELTAC, you know, platform, they could not apply this particular -- or approve this particular topical report on a generic basis.

Now, topical reports are stand-alone, and they will have their separate SERs. These are the ones within the I&C world. Topical reports -- I mentioned they are non-application-specific versus a technical report which is. These are within digital I&C.

As MHI mentioned, the Chapter 7, I&C-related ones, are the safety I&C system description, design process, the D3 -- that is this topical report -- the MELTAC topical report, and the HSI system design and process. So those four topical reports relating to digital I&C are going to have their own separate safety evaluation reports.

Now, the safety evaluation for these technical reports -- that is, the defense-in-depth coping analysis, which is specific to the US-APWR, the software program manual application software for the MELTAC platform -- those will be technical reports, they will be included in the DCD safety evaluation report. No separate SERs.

Now, the application-specific action items, there are 11 -- as Mr. Brown mentioned, there is 11 separate application-specific action items in the D3 SER. Two will be addressed by the MELTAC topical report when that review is completed. Two will be completed by the HFE process when that is completed. Two will be part of the coping analysis, the defense-in-depth coping analysis, that technical report, when that is completed as part of the DCD SER.

And then, there is five -- well, setpoint, the quality of the DAS, which is a GL -- Generic Letter 85-06 requirement. Then, there is the partial failure, large break LOCA, and then there is future submittals. But those five at MHI's option, and the licensee's option, will be addressed either directly in the DCD revision, as an ITAAC, or as a COL action item.

So there is still plenty of possibilities on how all of these application-specific action items will be addressed for those five. There's about six that will be addressed in other topical or technical reports.

MEMBER BROWN: So we may still have to deal with ITAAC.

MR. BEACOM: Yes, sir. ITAAC, but not

necessarily DAC ITAAC.

MEMBER BROWN: That doesn't make me feel any better. Oh, that's really good. Yes, okay.

MR. SCAROLA: To clarify --

MEMBER MAYNARD: Is there a question, or do we need --

MR. SCAROLA: Just simply to clarify, ITAAC, you know, is an option. But it is not an option that MHI is intending to take.

MEMBER MAYNARD: Oh, I wasn't --

MR. SCAROLA: We intend to close these out during the DCD --

MEMBER MAYNARD: That's fine. I understand that you all are trying to avoid that at all costs. Trying to avoid DACs. There will be ITAACs.

MEMBER BROWN: But he also said ITAAC for this. He said he is -- they also want to avoid ITAAC. Whether they do or not is --

MR. BEACOM: The construction ITAACs, right. There is not --

MEMBER MAYNARD: Yes, I understand that.

MR. BEACOM: Okay. Design acceptance criteria, okay.

Yes. Overall in the -- well, as far as

topical and technical reports there is something like -- in the US-APWR design certification there is 27 referenced topical reports that will have separate SERs on them. And there is something -- and there is also like 13 technical reports that will be rolled into the DCD SER.

If you go to the EPM -- enterprise project manager's website, I don't know if you guys have it, but there is quite a few -- or both of these type of reports will be included in the DCD SER, and those will have separate SERs.

Now, separate approval of the D3 from the US-APWR -- essentially, the -- I didn't put an answer here, but --

MEMBER BROWN: Yes.

MR. BEACOM: -- there should be separate approval, the staff feels, from the US-APWR of this D3 topical report. Why? Well, we believe the attributes as approved provide a level of detail that is sufficient. That is, we will not see this level of detail or additional detail in the DCD. It provides an excellent reference to and covers a lot of information and scope that the DCD does not.

The staff is confident that the ASAIs are sufficient, and that they address additional D3 information needed, particularly the coping analysis.

Now, we have briefly touched on that, too. If the applicant or licensee cannot meet the ASAIs, then is at their risk to proceed with the design, or immediately take an exception to the topical report and provide an alternate path for staff approval.

Now we get to the bypassing of the staff during PSMS actuation and during startup and shutdown. Well, the DAS, the diverse actuation system, is bypassed when the PSMS actuates, if proper feedback from the actuated components is received. That is, during the PSMS actuation, the actuated components will send signals back to the DAS, whereupon the DAS will not actuate because a PMS has actuated those components. This prevents unexpected competition between the systems.

Now, also during startup and shutdown concerns -- DAS has bypassed, at the same time, most of the PMS that is required during modes 1, 2, and 3, when pressurizer pressure is greater than the P-11 interlock. However, they are enabled by different means. The PSMS is automatically interlocked, whereas the DAS is enabled by a manual switch.

MEMBER STETKAR: Yes. Just for the benefit of the other members who weren't at the subcommittee meeting, the first slide regarding bypass from success of the actuated equipment, that is an automatic signal. Is that correct?

MR. BEACOM: Yes.

MEMBER STETKAR: Whereas the second bypass is an operator manual bypass. Is that right?

MR. BEACOM: That's correct. Do you mean this one? Yes.

MEMBER STETKAR: Yes.

MR. BEACOM: That's an operator --

MEMBER STETKAR: This topic.

MR. BEACOM: That's the operator --

MEMBER STETKAR: Okay.

MR. BEACOM: -- on the stand.

MEMBER BROWN: Say that again. Go back to that. What are you --

MR. BEACOM: That's the automatic bypass.

MEMBER STETKAR: Yes. That's the signal, an automatic signal.

MEMBER BROWN: From PSMS.

MEMBER STETKAR: Right.

MR. BEACOM: Whereas that has to be a manual, that's an operator manual switch enabling the DAS.

MEMBER BROWN: Well, that just means you -

- there is a manual switch that if you bypass it you can still put it in play by operating the manual switch. I mean, you can bypass the bypass.

MEMBER STETKAR: I think of it in the reverse. I think of it as an off switch, but you could think of it as an on switch also.

MEMBER BROWN: Hold it just a minute. Doesn't that -- if you bypass -- I look at that as a bypass of the bypass. If you -- I am reenabling the DAS, either that or I missed the picture when I was reading this stuff. I mean, if the PSMS bypasses the DAS because it supposedly sent in a trip signal, but something doesn't happen, you still have the manual capability to go to the DAS and manually operate stuff.

MR. BEACOM: Yes.

MEMBER BROWN: Is that what that means?

MEMBER MAYNARD: Whoa. There's two different parts here. We're talking about the automatic part, which is what was on the first slide.

MEMBER BROWN: Yes.

MEMBER MAYNARD: This is the intentional bypassing during startup and shutdown where you basically -- similar to what the requirements are for --

MEMBER BROWN: Oh, okay. I got it. Yes, yes, yes. Okay.

MEMBER MAYNARD: Yes, this is --

MEMBER BROWN: I got the differentiation now. I remember now. Thank you.

MEMBER MAYNARD: Okay.

MEMBER BROWN: I lost the bubble. Not hard.

MR. BEACOM: The concept of the DAS subsystems. So two subsystems, I believe, balances the two issues. That is reliability and spurious actuation of the DAS. Remember, the requirements for the DAS, the diverse actuation system, there is no 50.55(a)(h) applicability, where we have 603 safety requirements, since this is a non-1(e) system. Therefore, you don't have all the criterion for single failure independence, equipment qualification, quality, and so on.

In fact, the only applicable -- QA applicable to this particular is the ATWS QA, and that's identified in Generic Letter 85-06. It does not require an Appendix B program for the DAS system.

Now, it brings us to the --

MEMBER BROWN: Did you all get what that means? I mean, both of the DAS systems has to actuate in order to trip.

MR. BEACOM: Yes.

MEMBER BROWN: If that didn't come out --

MR. BEACOM: That's a good point, yes. Both subsystems of the DAS have to actuate for the DAS to have an actuation itself.

MEMBER BROWN: Not single failure proof, other than for spurious actuation.

MR. BEACOM: Now, with regards to the partial output failure, this -- it seems to collect a lot of comments, but the concept was captured by the D3 task working group or the digital I&C steering committee.

It states that the staff position is that a simple failure of the total system may not be the worst case, particularly when analyzing the time required for identifying responding to the condition. For example, a failure to trip may not be as limiting as a partial actuation of the emergency core cooling system with indication of a successful actuation.

Now, from this one staff member, this brings up two requirements. This is not just a guideline or something that is tangentially --

MEMBER MAYNARD: We need to go back to that previous one. Dana had asked a question on the partial that is -- is that -- he'll address your question that you had, Dana.

MEMBER POWERS: No, it did. It's a partial restatement. Yes, it --

MR. BEACOM: Yes. The analogy to fire protection is a good analogy, and that was just discussed a few days ago with me. So, to my understanding, it is a very good analogy.

And from the staff -- there is two requirements that this has to -- that we have to consider with regard to partial failures, and that is always -- wait a minute. We are talking -- again, we are talking partial output failure from the common cause failure. Now, we are talking about the PSMS here. We're not talking about the DAS. We are talking about the primary system itself. So that has to be brought up.

Now, here we do have 50.59 requirements. I'm sorry, 50.55, excuse me.

(Laughter.)

Oops. 50.55(a)(h), 603 requirement, says, "Completion of protection actions. Safety system shall be designed so that once initiated, automatic or manually, the intended sequence of protective actions of an execute feature shall continue to completion."

So we have to keep that in mind when we are discussing partial failures.

We also have to consider systems status indications where display instrumentation shall provide adequate and complete timely information for the safety system status. So just those two items come to mind when you are talking about partial failures that we can't get away from.

And again, no, the staff has not proposed a method for addressing partial output failures from common cause failure. And I have addressed this with several people and they say nor should we at this time.

We are waiting to hear from all design certifications on this. No one has really addressed this.

I say all digital upgrades. When I put that in there, all digital upgrades eventually will have to address this issue. I'm thinking digital systems start with the smallest upgrade, and then they are expanded into different applications, and so on. But this has to be an issue that the industry has to look at, or has looked at, at least to some extent, when they start with the smallest upgrades and go from there.

MEMBER BROWN: Why isn't a partial failure applicable to an -- you know, the older analog systems as well as the digital? I mean, common cause failures can be in anything.

MR. BEACOM: Yes, correct.

MEMBER BROWN: And I -- and we didn't force this on -- just to be devil's advocate here, we -- I haven't heard anything in the last year and several months where that concept has been forced back to the analog-type plants on partial failures.

MR. BEACOM: It did address different types of single failures, such as inadvertent rod insertion or --

MEMBER BROWN: I'm just taking it from a general concept of partial failure, doing something which doesn't do something else, which causes in -- with the analogy that was posed. Now we are applying a new set of rules to the digital I&C upgrades that you didn't apply to the original I&C designs in the plant. Is that right or wrong? Not right or wrong. Is that correct or incorrect? It sounds like nobody is arguing with me, so I guess I'm right.

MR. BEACOM: I think we're getting smarter. How about that? I think this was brought up in the part of the task working groups, and it was a situation where staff is getting smarter.

MEMBER BLEY: And some operating experience has made us smarter.

MEMBER BROWN: Well, I won't argue with operating experience.

MEMBER POWERS: And it also has to be recognized that in analog systems we have a great deal more intuition on what things happen, whereas your digital systems seem to be mysterious and convoluted.

MEMBER BROWN: Well, you know, okay. Let's -- I want to put that in context, okay? I started doing these in 1979, so that's only 28 years or 29 or almost, you know, 30 years. And the common cause digital what I call microcomputer failures were non-existent in failures that we had to deal with. And there are -- those are installed in 120 reactor plants, something like that, almost all of them now, in multiple versions.

So I don't -- I understand your point. I agree the analog stuff philosophically is more tangible in terms of dealing with it.

MEMBER BLEY: They are not invisible in --

MEMBER BROWN: Bits and bytes are terrible.

MEMBER BLEY: I don't disagree. But I'm

saying you've got to be a little careful about it.

MEMBER BROWN: I just -- I don't want to put -- I don't want the brush swept so broadly without thinking about, you know, that aspect of it. There has been a very -- there has been a fairly large time in which digital systems have been applied with very, very good success, without, you know, what I would call many, if -- you know, I'm not -- maybe any, okay, common cause failures that propagated from train to train to train or occurred at the same time or caused a partial failure that caused something else. That's all I'm saying is the data is sparse.

MR. BEACOM: Well, when we have something as plant-wide as this scope, it is something we can't get away from, I don't think. Somehow --

MEMBER BROWN: I'm not arguing with you. I'm just saying, you've got to be thoughtful about it. That's all I'm saying -- be thoughtful.

MEMBER POWERS: In all things.

MEMBER BROWN: Absolutely. Okay. I quit right now. Let's go on.

MEMBER MAYNARD: Let's go ahead.

MEMBER BROWN: I'm sorry.

MEMBER MAYNARD: That's all right.

MR. BEACOM: Now we get to a discussion of the DAS inputs for the automatic reactor trip and turbine trip. These are the automatic actuations within less than 10 minutes, and that is -- there are three inputs to the DAS for -- that will trip the reactor and do a turbine trip, and that is high pressurizer pressure, low pressurizer pressure, or low steam generator level.

VICE CHAIRMAN ABDEL-KHALIK: Now, if I look at these trip signals, you know, in terms of classes of accidents, high pressurizer pressure can probably trip the plant, and, you know, decreased heat removal events in general eventually will get you to a high pressurizer pressure trip.

Low pressurizer pressure will probably protect against loss of inventory events. Low steam generator level would probably protect the plant against high -- increased heat removal events, like a steam line break or something like that. So how about other accident categories? Why aren't there any reactor trips that would address that? Like reactivity and power distribution anomalies?

MEMBER BROWN: Do you mean like reactivity addition transients or cold ones? Well, all right. I thought I asked that question in the subcommittee meeting. Now I don't remember the answer.

VICE CHAIRMAN ABDEL-KHALIK: You know, looking at it broadly from the accident categories that these particular trips would protect against, albeit it might be at a delayed point than what you would expect from the normal reactor trip system, it seems like there is something missing here.

MR. SCAROLA: Excuse me. Ken Scarola. I think it's important that we not forget the acceptance criteria that is defined by BTP-19, which is only prevention of 10 CFR 100 offsite boundary problems. We are not trying to protect the fuel in the same way we are for Chapter 15 events.

So there is -- this significant difference in the acceptance criteria leads to relaxed requirements on the protective actions and the ability to take manual actions for many of the events that you're talking about.

VICE CHAIRMAN ABDEL-KHALIK: So a rod ejection accident would not fall in that category?

MEMBER MAYNARD: Well, let's keep in mind here that we are talking about a beyond design basis accident by having common cause failures in things. You have systems to take care of all of your design basis accidents and -- for beyond design basis accidents, there were coping studies that were done when some of these original requirements were being established as to the key things that could get you into problems from the real high offsite doses.

And, you know, I believe that this is consistent with the criteria for like the other diverse actuations, the AMSAC systems, the ATWS, and stuff like that.

MR. BEACOM: Well, I think it is -- in fact, it is an ATWS system.

VICE CHAIRMAN ABDEL-KHALIK: I guess I will have to think about it, because I think something like a rod ejection accident would have to be included in the kind of things that we would have to protect against with this diverse system.

MEMBER BLEY: Well, there is a bit of a risk here, and I have to go back to the scoping, so -- but it's -- you're backing up -- you already have a redundant trip system to cover all of the accidents. Now you are saying, "For some that might happen, I want to be able to have another backup system beyond that." And you don't need to do that for the rarest of all events; you need to do that for some of the most severe. I think that is the basic philosophy behind it. So we need to think about that a little more.

VICE CHAIRMAN ABDEL-KHALIK: Okay.

MEMBER MAYNARD: I thought I covered the beyond design basis type stuff. That's a little --

MEMBER BLEY: Well, the main scram system not working is beyond --

MEMBER MAYNARD: Yes, yes.

MEMBER BLEY: That's where we are, and you already have redundancy there to cover you.

MEMBER MAYNARD: But I believe that this is consistent with the current regulatory requirements, is what needs to be protected with a diverse system.

MR. BEACOM: Well, okay. Like Ken said, that's a very good point for addressing Part 100 requirements, as far as offsite radiation release, knowing that -- something much different than in Chapter 15.

MEMBER MAYNARD: Okay. John, you've got a question?

MEMBER STETKAR: Yes. I raised this in the subcommittee meeting, and I was just thinking in the sense of Dr. Apostolakis' earlier question. As it is designed right now, and presented in the topical report, there is a difference in the protection criteria for loss of feed to a single steam generator.

The normal reactor protection, PSMS, system protects against loss of feed to any steam generator, and DAS requires loss of feed to two steam generators.

And when we questioned this in the subcommittee meeting, I believe MHI's response was that the coping analysis would show that loss of feed to a single steam generator, or equivalently single steam generator dryout, was within whatever -- the bounds of the coping analysis. We haven't seen that.

However, if the topical report covers a -- the design concept for any DAS, however it is supplied, I didn't notice any ASAI items in the SER in particular related to this issue. There is a lot of alphabet soup there. I apologize for it, but --

MR. BEACOM: No. There is not an ASAI. Yes, it -- that was brought up after. So we can address that through a DCD RAI or something to that effect, as far as what --

MEMBER STETKAR: But the SER effectively -- if the SER is issued, the SER, at least on a generic basis, would accept this difference. Is that correct?

MR. BEACOM: We wouldn't have an application for it. Once you get the application, the DCD always has precedent over the topical report.

MEMBER MAYNARD: For the DCD, they would have to show that the topical report is applicable to that --

MR. BEACOM: Correct.

MEMBER MAYNARD: -- particular application. And if the coping study came out and showed something different, then this topical report would --

MEMBER STETKAR: I was thinking in the broader design context that the SER is essentially approving the fact that the DAS has two trains, that it is a two-out-of-two logic, that it has -- uses common signals, that the PIF interface devices are all hard-wired analog, you know, that type of -- it is a concept. This is part of -- that logic is part of that concept.

MR. SCAROLA: Excuse me. This is Ken Scarola again. The logic for all of the actuation functions in DAS are only described in the topical report as typical, only so that there is a broader understanding of how we do two-channel functions in voting and everything else.

It states in the topical report that all of the inputs, the functional logic, the actuation algorithms, are plant-specific, and they will be described in licensing-specific documents. So there is no intent for the staff to approve functional algorithms generically.

MEMBER STETKAR: Okay. As long as that's is clear. I must have missed that.

MR. SCAROLA: It's part of -- there's a section in the back on future licensing submittals, and the algorithms are a part of that.

MEMBER STETKAR: Thank you.

MR. BEACOM: Okay. Anyway, we refer to the D3 coping analysis, which discusses the vent -- both event evaluation methods and results of each event evaluation with common cause failure.

I believe it or not, I think that's -- that is it. Now we can go -- ask questions.

MEMBER BLEY: When do we expect us to see the coping analysis?

MR. BEACOM: That is part of the --

MEMBER BLEY: DCD?

MR. BEACOM: -- DCD, the design certification itself, that is currently under review.

MEMBER BLEY: I think that should -- well, that should address a lot of the concerns I think many of us have, and that's a key document I think for us to get a chance to --

MR. BEACOM: It is on the docket. It is available for --

MEMBER BROWN: The coping analysis is -- I've forgotten what the topical report number is, but I know there is a least a Rev 0 of it that --

MR. BEACOM: Yes, actually, Rev 1 is here.

MEMBER BROWN: Is it Rev 1? I have no idea what I've got on my computer, so --

MR. CIOCCO: This is Jeff Ciocco with the NRC. You have -- I mean, we have provided the coping analysis to you, along with the DCD and everything. And it is a technical report, not a topical report. So --

MEMBER MAYNARD: I believe we have all of the technical reports and topical reports.

MEMBER BROWN: What rev is the current rev for that?

MR. BEACOM: The current rev is Rev 1. The number is MUAP-07014, Rev 1.

MEMBER MAYNARD: Well, a couple of things that I think is important to remember about this, is that, again, the SER is for the topical report. Although the topical report is referenced in the DCD, it is during the DCD review that these application-specific action items will be addressed. And also, it is part of the obligation of MHI to show that the assumptions in the topical report are applicable to the US-APWR and the DCD review.

So I have looked at this. Some of my questions in the beginning was, "Well, are we signing off? Are we looking at buying something that if we review it later, and we find it different or not applicable, are we tied to it?" And, again, there are some things that are defined.

It does find -- the staff finds that it's acceptable that the DAS concept, the analog system, is diverse and, you know, different enough from the PSMS and the plant safety system and non-safety systems there that it can be credited for that. But as far as exactly what it does, parameter setpoints, algorithms, and stuff, that will have to be looked at in the DCD review to see if that is really applicable to the US-APWR.

MEMBER BROWN: Let me make one other observation about -- you said the future licensing section of your thing talks about them being typical. So I just read the one paragraph, and I don't find the word "typical" in there anywhere. So it just says that there is a table of future licensing submittals, and that is it. So I guess I didn't pick up on the "typical" either, but --

MR. SCAROLA: I think you'll find the word "typical" in the description of the algorithms back in a previous section.

MEMBER BROWN: Okay.

MR. SCAROLA: That section you're looking at says, "Because the others are only typical, the exact algorithms will be in a future licensing submittal." It's an example --

MEMBER BROWN: The other section, not Section 10, but some other section?

MR. SCAROLA: Right.

MEMBER BROWN: Okay.

MR. SCAROLA: So, for example, for the US-APWR, the exact functional algorithms are in Section 7.8 of the DCD.

MEMBER BROWN: Okay.

MR. SCAROLA: Then, the coping analysis demonstrates the acceptability of those functions, including rod ejection events. I did look at it. All the rod ejection events are in there.

MEMBER MAYNARD: And it's certainly on the record now.

MEMBER BROWN: Yes.

(Laughter.)

MEMBER BLEY: And we do have Rev 1.

VICE CHAIRMAN ABDEL-KHALIK: Now, where and when will the bypassing of the DAS during startup and shutdown -- the justification for bypassing of the DAS during startup and shutdown be provided?

MR. BEACOM: The justification?

VICE CHAIRMAN ABDEL-KHALIK: Right. As far as --

MR. BEACOM: Well --

MR. SCAROLA: Can I address this, Royce?

MR. BEACOM: Go ahead.

MR. SCAROLA: There is no analytical justification for bypassing DAS during shutdown. The philosophy goes back to the SRM to SECY-93-087, where at that time, back in 1993, this was viewed as a beyond design basis event, and, therefore, the SRM says you can use best estimate methods for analyzing your ability to cope with common cause failures.

The interpretation of "best estimate" is normal operating conditions, not the extremes of any operating conditions, and not shutdown conditions, the thinking that you are in these shutdown modes for a very, very short period of time; therefore, the likelihood of CCF during these shutdown modes is very, very low.

So the interpretation of "best estimate" has been normal operating. So there is no analytical justification for bypassing the system, only an interpretation of the SRM.

MEMBER MAYNARD: I would also point out that this is consistent with the ATWS AMSAC systems. They are not required to be functional until -- I forget what ours was, I think it was like 50 percent power or so -- before going above 50 percent power, that you had to ensure that you had the AMSAC system actuated there, or --

VICE CHAIRMAN ABDEL-KHALIK: But that's based on an analysis that shows below 50 percent power as being that category.

MEMBER MAYNARD: I don't believe it was based on any --

(Laughter.)

VICE CHAIRMAN ABDEL-KHALIK: -- which is that peak pressure that you get from a loss of --

MEMBER BLEY: I mean, it was almost the other way around. There were specific conditions under which it was found to be needed --

VICE CHAIRMAN ABDEL-KHALIK: Right.

MEMBER BLEY: -- and something we didn't have. And that is when it went in for --

VICE CHAIRMAN ABDEL-KHALIK: Yes.

MEMBER BLEY: -- for ATWS. So the analysis to show when you needed it kind of showed when you didn't need it.

VICE CHAIRMAN ABDEL-KHALIK: Right. But still, I find this answer sort of unsatisfactory.

MEMBER BLEY: Well --

VICE CHAIRMAN ABDEL-KHALIK: The logic of bypassing the DAS during startup and shutdown.

MEMBER BLEY: Look, but, you know, if you think about --

VICE CHAIRMAN ABDEL-KHALIK: Our normal --

MEMBER BLEY: -- when you shutdown, you have to bypass SI, because you are going to drop in pressure, and you don't want it to come on and flood things. And once you are shut down you don't need the trip part of it, so you are -- the main things that the DAS would take care of --

VICE CHAIRMAN ABDEL-KHALIK: How about startup?

MEMBER BLEY: -- you don't need. Again, in startup, as you are coming up, you have safety injection bypassed until you get up to pressure. Otherwise, you can't operate. You do have reactor trip --

VICE CHAIRMAN ABDEL-KHALIK: Right.

MEMBER BLEY: -- of course, but I suspect -- coping analysis does not look at these conditions. So there are some things on startup with having the backup to reactor trip that I think the coping analysis could show, but it doesn't I guess, right?

MR. SCAROLA: That's the important point. I'm not saying that you cannot analytically demonstrate that the system is not needed. I'm saying we have not done that based on the interpretation of "best estimate."

MEMBER BLEY: I don't understand the "best estimate" piece of that, but my intuitive feeling is you probably don't need it. But an analysis would be --

MEMBER MAYNARD: Well, I don't know if the staff has anybody here. We probably have the wrong people here to address the ATWS requirement of the -- I'm not -- yes. Because I know there was a history behind that as to when it was deeded, and what agreements were made, but I do not believe that MHI is inconsistent at all with what the current requirements are and what the current fleet is using.

But I don't know what the evaluation or the analysis was that provided that. I don't think any of them are required to have their diverse backup system operational during a reactor startup or shutdown there. And I think it's usually around 50 percent is when they -- so I don't -- if it's something that we need an answer to, we need to try to get the right people here for what the requirement is.

VICE CHAIRMAN ABDEL-KHALIK: I certainly would like to know, because I'm not sure I understand the meaning of the word "best estimate" and whether that is used to eliminate, you know, startup and shutdown conditions.

MR. SCAROLA: I think it's important that there is no definition of "best estimate," and that's one of the reasons why we submitted this topical report, because we in this topical report provide our definition of "best estimate," because there is no -- there is none.

MEMBER MAYNARD: Well, I think that probably what we need is more an answer from the staff as to --

VICE CHAIRMAN ABDEL-KHALIK: Right.

MEMBER MAYNARD: -- why this is acceptable. I think that's the real key there.

VICE CHAIRMAN ABDEL-KHALIK: Right.

MR. BEACOM: So do you want --

MEMBER MAYNARD: Pardon?

MR. BEACOM: Go ahead. Explain what --

MEMBER MAYNARD: I don't know when we can get that. I know we are not going to get that right now today, but if you can potentially look into that or try to get something back to us tomorrow morning or something sometime.

MR. CIOCCO: Do we know the exact question that we want to get answered?

MEMBER MAYNARD: I think it is a justification for having -- being -- allowing the DAS to be bypassed during startup and shutdown.

MEMBER SIEBER: That's it.

MEMBER BROWN: For my edification, I would presume, based on you have to shut -- you have to have certain functions bypassed when you are starting up, you know, the low pressures, the turbine trips, blah, blah, blah, whatever those things are. But if your normal systems theoretically are operational during a startup, and theoretically the -- any transient or accident that could be actuated during a startup or a shutdown, should be protected by your normal systems. It's just you don't have your diverse system in play in that circumstance.

Now, that -- am I -- even though your low pressure trip or your low whatever the turbine trip thing is --

PARTICIPANT: Bypass.

MEMBER BROWN: -- something else will be taking care of you during the startup mode. Whether it's a power thing or whether it's some other function, that it's not sensitive to the actual pressure temperature or --

MEMBER BLEY: I kind of suspect, but you need some analysis, clearly --

MEMBER BROWN: We need more time for manual response, and you still have the manual DAS capability.

MEMBER BLEY: I mean, I am presuming that there is an analysis to cover accidents during a startup.

MEMBER BROWN: Oh, yes.

MEMBER BLEY: Using your normal system.

MEMBER BROWN: Oh, yes. Hopefully.

MEMBER BLEY: All right. Thank you.

MEMBER MAYNARD: Are there any other questions or --

(No response.)

If there is no other questions, Mr. Chairman, I will turn it back to you.

CHAIRMAN BONACA: Thank you very much.

And looking at the agenda, we have a break now. You guys gained back the time.

MEMBER MAYNARD: Thank you.

CHAIRMAN BONACA: Congratulations. That's super.

MEMBER MAYNARD: Somebody has to take care of Sam.

(Laughter.)

MEMBER ARMIJO: Thank you, Otto.

CHAIRMAN BONACA: Let's take a break until five after 5:00. Okay?

(Whereupon, at 4:50 p.m., the proceedings in the foregoing matter went off the record.)

NIST Response to Open Item

ACRS Meeting

June 3, 2009

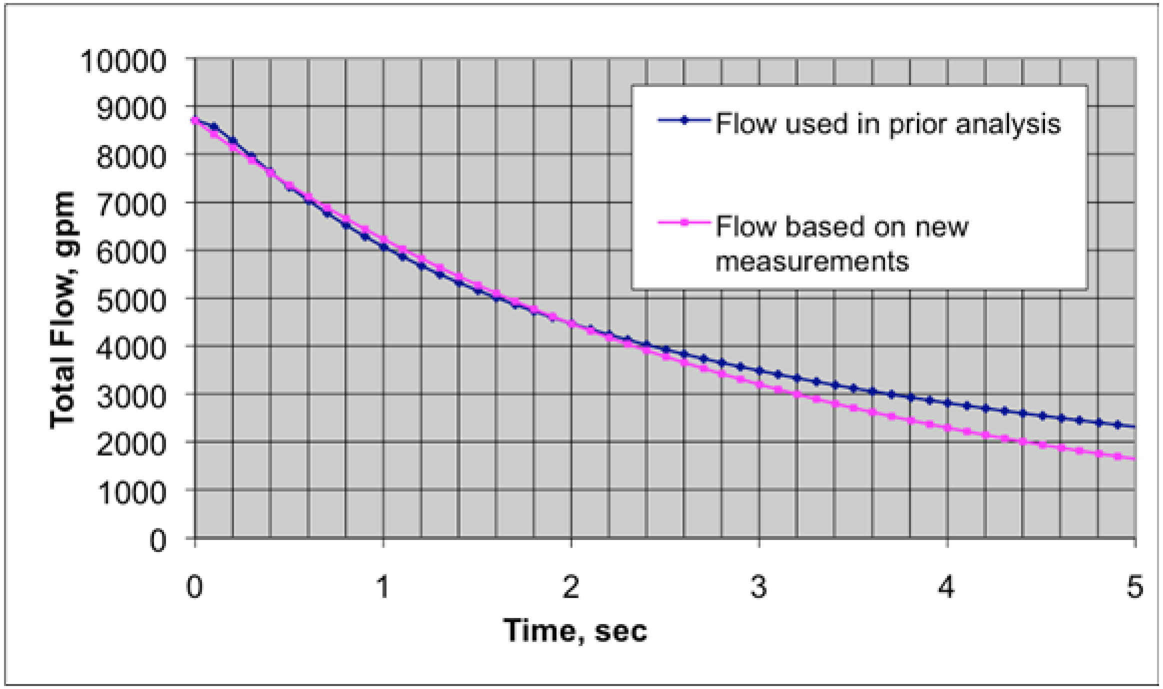
NIST PARTICIPANTS

- Dr. Robert Dimeo, Director NCNR
- Dr. Wade Richards, Chief ROE
- Dr. Robert Williams, Section Head Nuclear Analysis
- Dr. Mike Rowe, Special Advisor to NCNR Director
- Mr. Thomas Myers, Chief Reactor Ops.
- Mr. David Brown, Supervisor Health Physics

Open Item From Meeting of 4/3/2009

- While responding to a question raised at an earlier ACRS Subcommittee meeting, NIST identified an issue with pump coast-down
- It was noted that the pump coast-down curve used for the RELAP analysis was compared to the data measured under different conditions
- Although the curve used in the analysis was very conservative, a new curve was measured under appropriate conditions for comparison

Results



Comparison between prior flow model and new model, which was conservatively based upon new measurement

RELAP ANALYSES

- The minimum CHFR occurs at approximately 1.5 s, where the two curves coincide
- As a result, the MCHFR of 2.17 is unchanged within error from the earlier value of 2.19
- Detailed analyses out to 30 s show that the system progresses to a stable natural circulation state
- The fuel temperature remains below 137°C, substantially below the safety limit

Conclusion

- The limiting loss of flow accident (with failure of shutdown pumps to start) has been extensively re-analyzed
- The results show a substantial margin against DNB (MCHFR > 2)
- This accident poses no danger of fuel damage
- The SAR will be updated to reflect this revised analysis

U.S.NRC

UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

Advisory Committee on Reactor Safeguards (ACRS) License Renewal Full Committee

National Institute of Standards and Technology National Bureau of Standards Test Reactor License Renewal

June 3, 2009

William B. Kennedy, Project Manager

Office of Nuclear Reactor Regulation

Open Item

- In addressing the concerns of the ACRS subcommittee members, the licensee identified an unrelated error in a measured flow coastdown data set
- The flow coastdown data set was used to benchmark the RELAP model used to analyze the loss-of-offsite power accident
- The licensee promptly reported the error on March 30, 2009, to the NRC project manager

Initial NRC Response

- The staff performed a preliminary independent review and calculation to assess the safety significance of the error
  - safety margin reduced, but still adequate
  - isolated error
  - staff's calculation in close agreement with licensee's initial assessment
- The staff discussed the significance of the error with the licensee and a plan to update the flow coastdown data set and the accident analysis

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UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

NRC Staff Review

- The licensee submitted a revised loss-of-offsite-power accident analysis on April 22, 2009
- The staff compared the updated flow coastdown data set against the erroneous data set and found them to be nearly identical
- The staff reviewed the assumptions used in the updated accident analysis and found them to be as conservative as those used in the original analysis

Updated Safety Evaluation

- The minimum critical heat flux ratio (safety margin) at the hot spot on the fuel cladding decreased from 2.19 to 2.17
- The maximum fuel temperature is 137 degrees Celsius (the safety limit is 450 degrees Celsius)
- The staff concludes that there is reasonable assurance that a loss of offsite power will not result in fuel damage and that the consequences of the accident are bounded by the MHA

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Revision of RG 1.21 (Effluents) and RG 4.1 (Environmental Monitoring)

Presentation for:

ACRS Committee Meeting
June 3, 2009

Richard Conatser & Steve Garry
NRR Div. Inspection & Regional Support

Outline

- Introduction (People & Project)
- History (Drivers for Change)
- Documents
- Reg Guide Update Initiative
- Reasons for revising RGs
- Considerations: Backfit, Consistency, Delay Publication
- Closure and Questions

Introduction: Project & People

- Team formed in 2006
  - HQ: NRR, NRO, FSME, RES
  - Regions: I-IV
  - Some are here today
- Progress
  - FRN Oct and Nov 2008
  - Public Meeting in January
  - Office Concurrence and ACRS Sub.: May
  - OGC and ACRS

History (Drivers for Change)

- H-3 in Ground Water
  - Salem – 2003, SFP Leak
  - Braidwood – Mar 2005, H-3 in Well
  - Indian Point – Sep 2005, Crack in SFP
- Lessons Learned Task Force Report
  - Sep-2006, Total of 26 Recommendations
  - 10 Recommendations → RG 1.21
  - 4 Recommendations → RG 4.1

Documents

- RG 1.21 (Effluents)
  - Measuring, Evaluating, Reporting Effluents
  - Abnormal Releases, C-14, Sampling, Surveys, Principal Nuclides, LLD
- RG 4.1 (Environmental)
  - Monitoring Radioactivity in the Environs
  - Exposure Pathways, Routes of Exposure, Samples, Spills, Reports

RG Updates

- 476 Reg Guides to Revise
- NRC Chairman Memo, Jun-2006
- Phases 1 thru 3, ECD Dec-2009
- RG 1.21 and RG 4.1 are in Phase 3

Benefits of Revising RGs

- RG Update
- Lessons Learned Task Force Rec.
  - Ground Water Issues (Surveys, etc)
- Dated Guidance (RGs 35 years old)
- Incorporate OE & Lessons Learned
  - TEDE, Direct Rad, C-14, LLD, etc
- NEI, EPRI, ANI issued new guidance
- Updated NRC guidance is needed

Public Comment: Back-fit

- RGs are not regulations
- RGs describe acceptable methods
- Licensees may continue to use Rev. 1
- Licensees are not required to commit

Public Comment: “Inconsistencies”

- Discussed at ACRS Subcommittee
- NUREG-1301 and 10 CFR 50
  - Semi-annual vs Annual Reports
- 10 CFR 20 and 10 CFR 50
  - TEDE vs Whole Body Dose
- NUREG-1301 and RG 1.21
  - NUREG silent C-14, RG includes C-14
- NUREG-0543 and RG 1.21
  - Calculating EPAs 40 CFR 190 Dose

Public Comment: Delay RGs

- Discussed possibility at ACRS Sub.
- ICRP-103 dose methodology pending
  - SECY-08-197
  - Engage Stakeholders
  - May take many years to complete
- Plants not required to commit to RG 1.21

→ Staff Recommendation: Issue RGs consistent with RG Update Initiative

Questions

?

Regulatory Guides 1.21 & 4.1 Issues (DG 1186 & DG 4013)

George Oliver
June 2009
ACRS

DG-1186 & DG-4013 Issues
Industry & Staff Efforts

- Industry Contribution From 30+ Individuals
- Many Detailed Technical Comments
- Professional & Productive Relationship With Staff
  - January 15, 2009 Workshop Productive
- Emergence Of SECY 08-0197
  - 40 Guidance Documents Impacted
    - An Integrated Approach Is Needed

NEI

DG-1186 & DG-4013 Issues
Need For Integrated Approach

- DG-1186 & DG-4013 Duplicate & Inconsistent With Other Guidance
  - Several Guidance Documents Related To Groundwater
- SECY 08-0197 Offers A Real Opportunity
  - Benefits Of Consolidated Guidance

NEI

DG-1186 & DG-4013 Opportunities

- The Existing Guidance Should Remain Applicable
  - The Licensing Basis Is Not Impacted
- Clarification Of Solid Radioactive Waste Reporting
- Elimination of On Site Radiological Monitoring Programs From DG-4014
- Additional Flexibility
  - Calculate C-14 Effluents

NEI

Meeting Objective

- Assess the risk of PCI/SCC fuel failures during BWR Anticipated Operational Occurrences at EPU conditions.
- ACRS Letter, Dec. 20, 2007 Susquehanna Extended Power Uprate

  ”The staff should develop the capability and perform a thorough review and assessment of the risk of Pellet-Cladding Interaction (PCI) fuel failures with conventional fuel cladding during anticipated operational occurrences.”

Topics

- Background and Reasons for Concern
- PCI Basics
  - Power/burnup dependence
  - Appearance, Mechanism
- PCI failure powers, failure strains and times-to-failure.
- Conclusions and Recommendations

Background

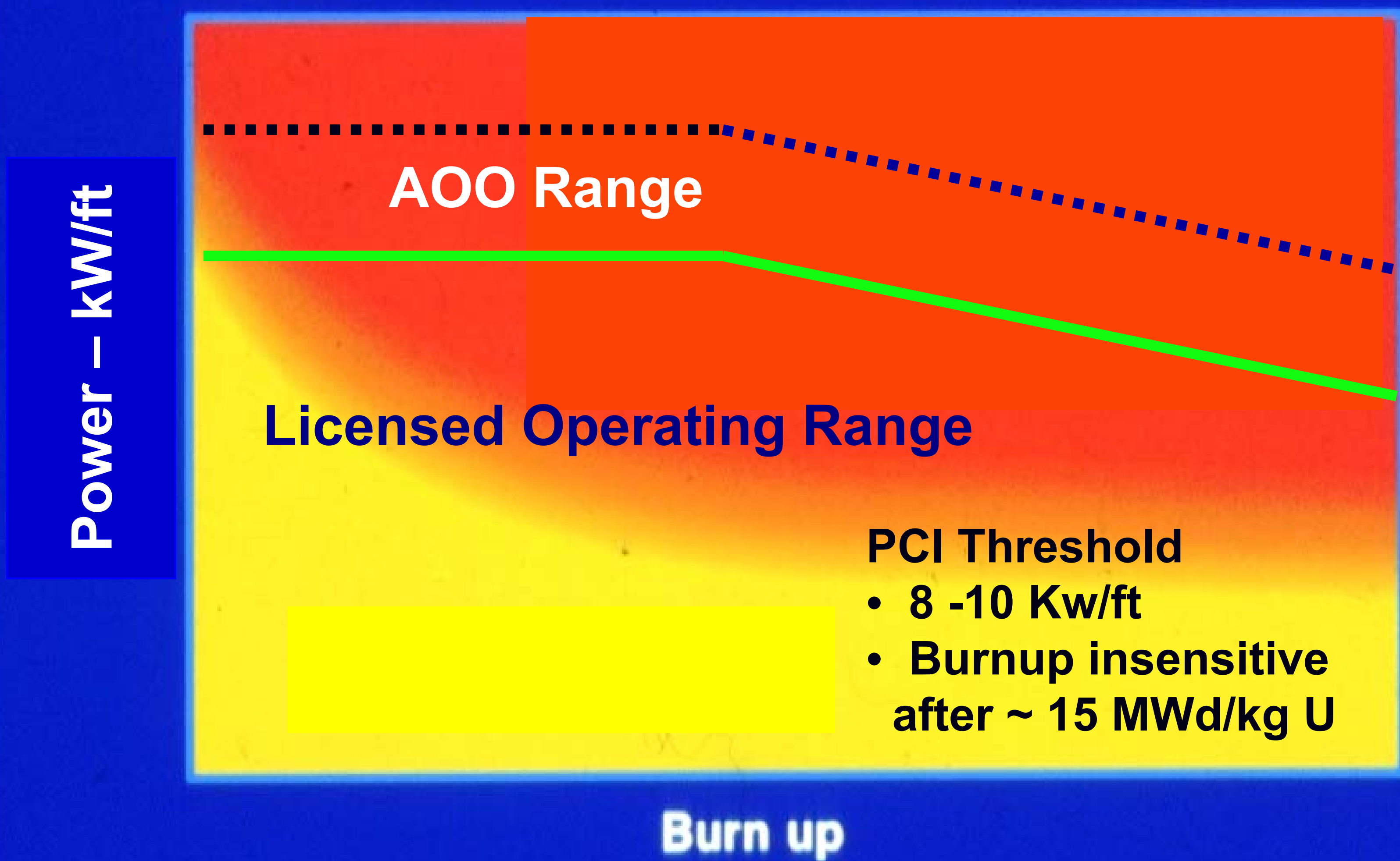
- NRC analyses in late 1970s early 80s
  - Notified vendors ...ready to introduce PCI fuel failure analyses into plant safety analyses.
- PCI mitigations introduced in late 70s, early 80s
  - Operating restrictions, 9x9 and 10x10 fuel bundles introduced
  - PCI resistant design licensed and in service.
- Technical assumptions:
  - existing thermal-mechanical licensing limits were sufficient to prevent PCI during AOOs
  - transients were over too quickly to cause PCI failures during AOOs

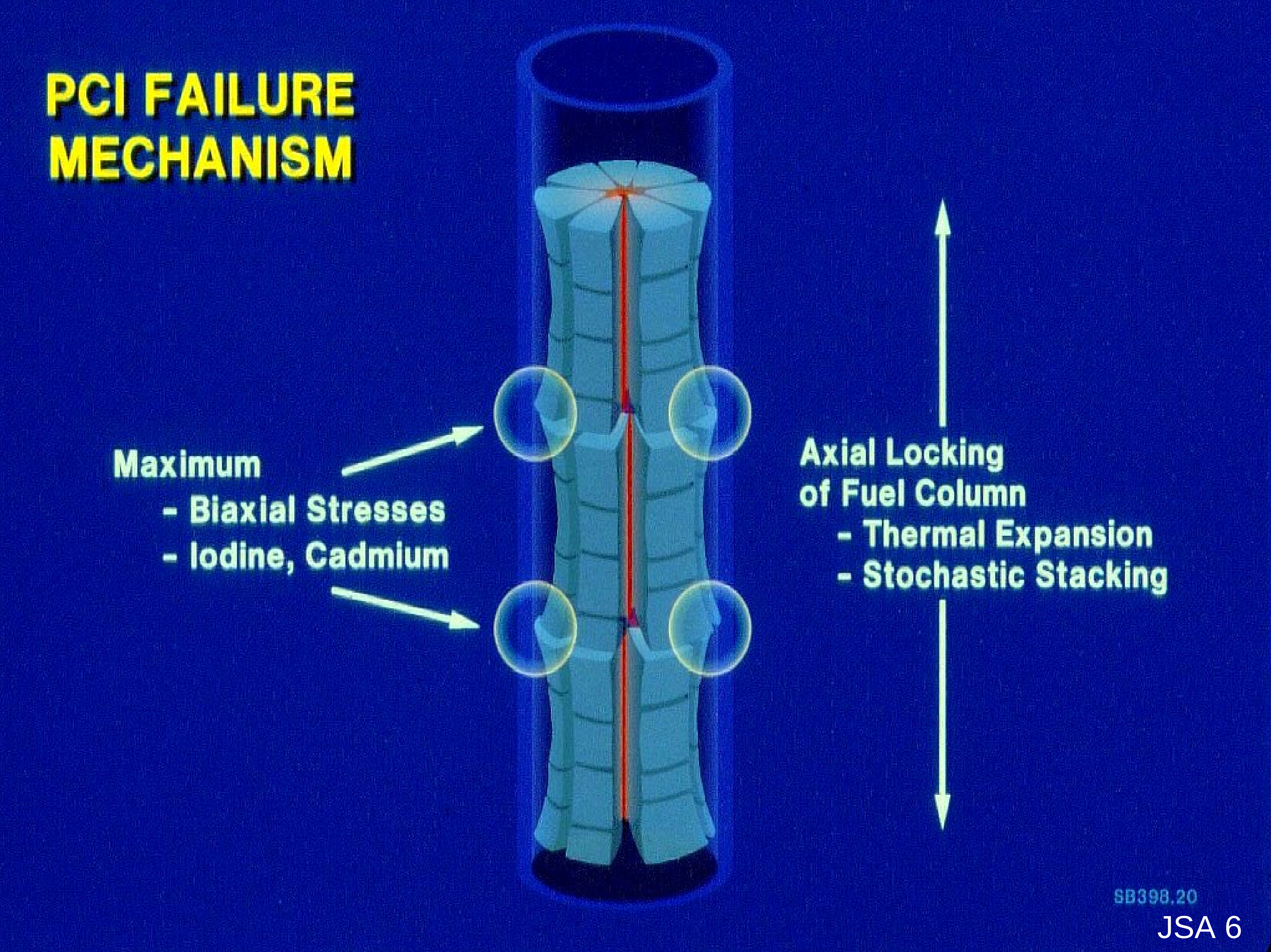
No incentive for PCI-specific regulatory changes

Reasons for Current Concern

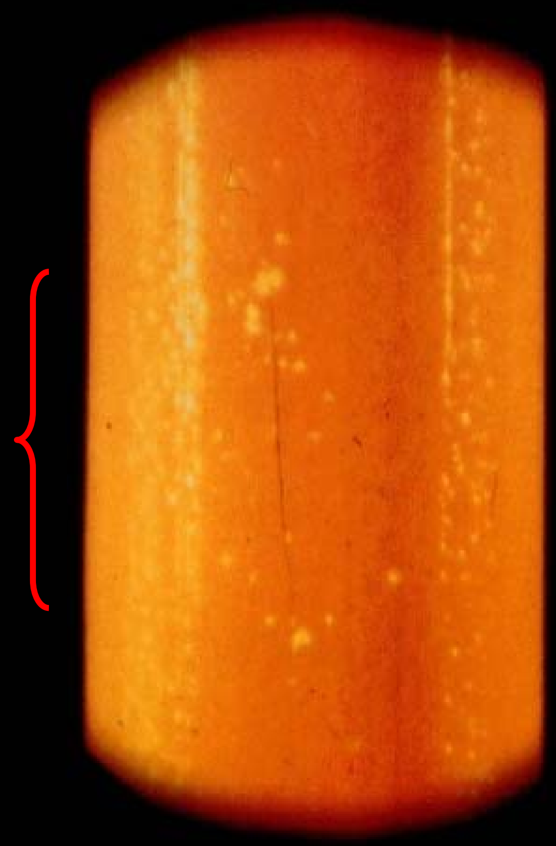
- Margins gained by design changes introduced in the1980s are disappearing.
  - Peak LHGRs of today's 10x10 fuel designs are the same as old 8x8 designs.
  - Number of fuel rods at risk during AOOs increasing in proportion to magnitude of EPU.
  - Use of non-PCI-resistant fuel increasing
- PCI failure strains are much lower than the <1% strain acceptance criteria.
- PCI failure times are very short at AOO power levels.

PCI-MAP


Power – kW/ft
AOO Range
Licensed Operating Range
PCI Threshold
• 8 -10 Kw/ft
• Burnup insensitive after ~ 15 MWd/kg U
Burn up

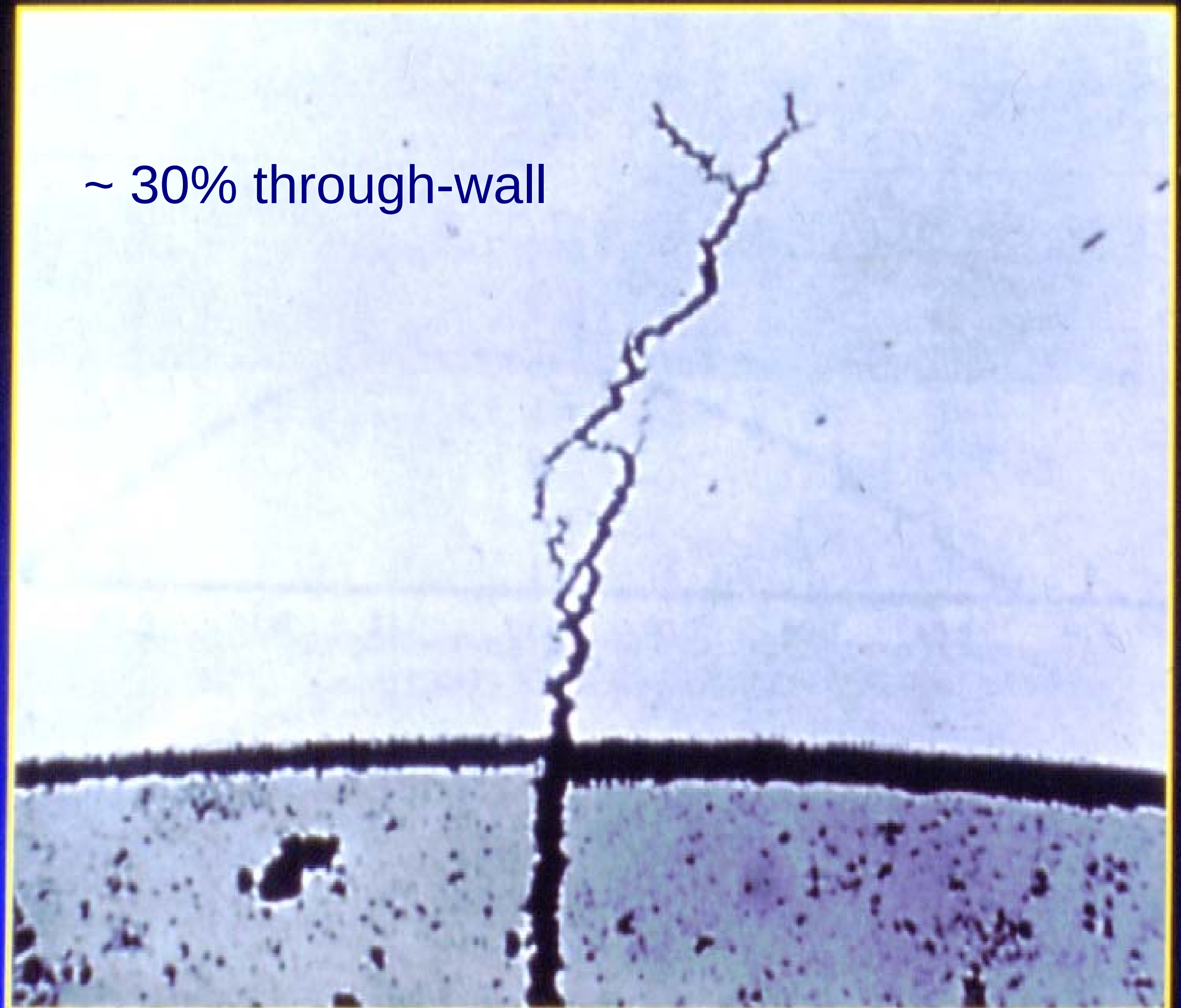

PCI FAILURE MECHANISM
Maximum
- Biaxial Stresses
- Iodine, Cadmium
Axial Locking of Fuel Column
- Thermal Expansion
- Stochastic Stacking
SB398.20

PELLET CLAD INTERACTION CRACK

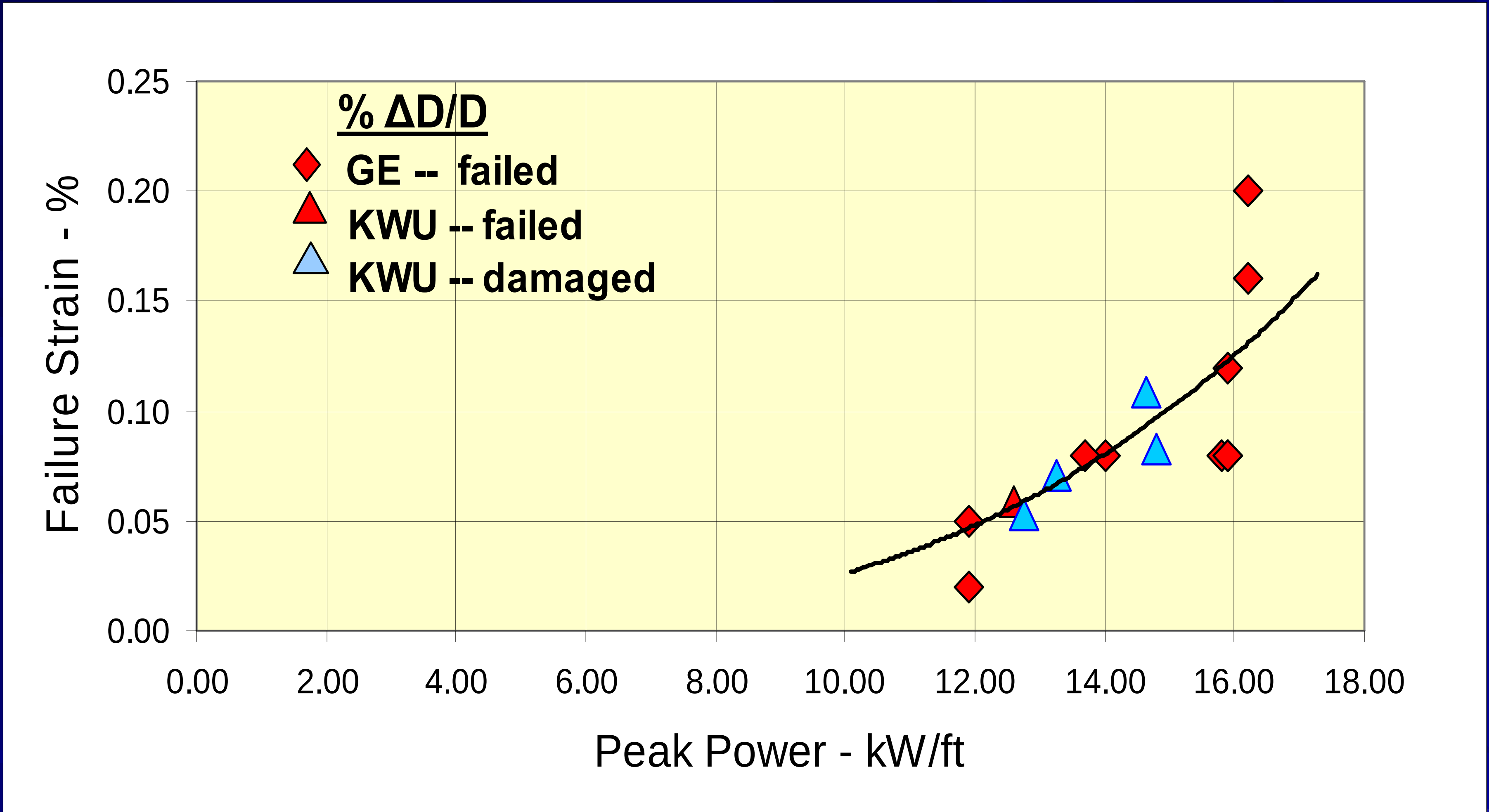


- BWR fuel rod
- Typical axial crack
- << 1% plastic strain

Branching in Incipient PCI Crack

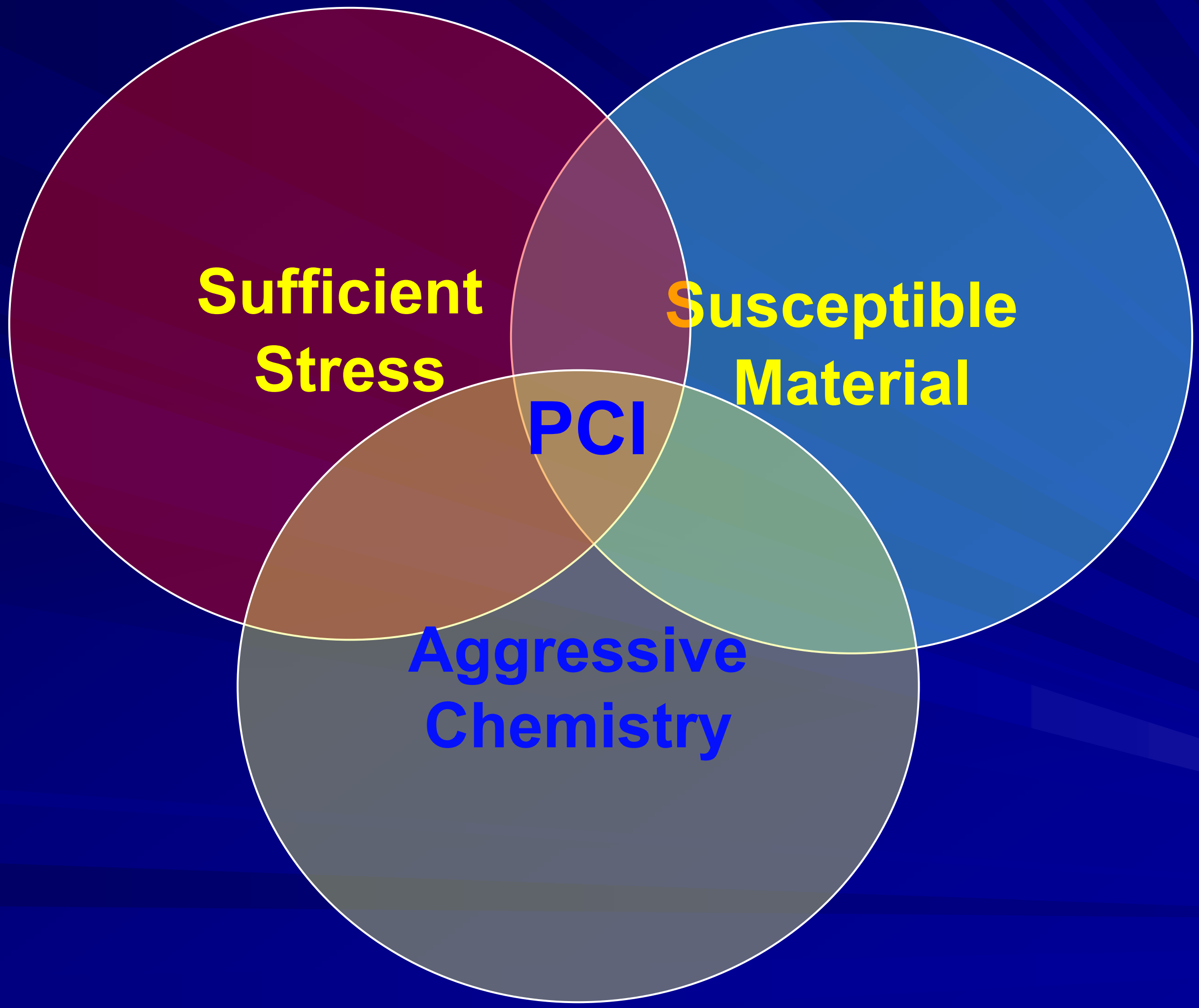


GE and Demo Ramp II PCI Failure Strains

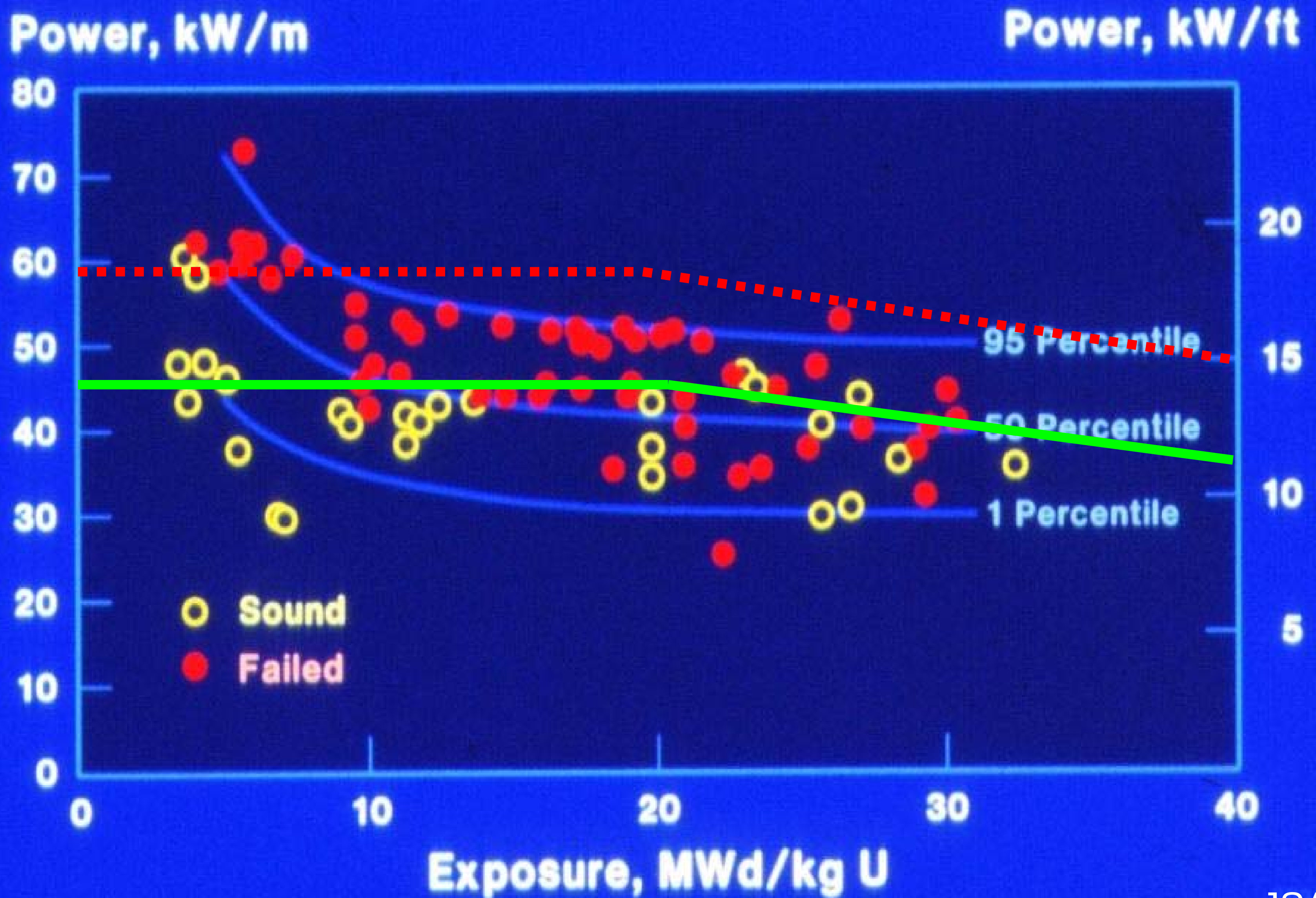


- All rods failed or damaged by PCI
- All strains much lower than 1%

Requirements for Stress Corrosion Cracking



BWR STANDARD FUEL



Oskarshamm 1 Event

- Control rod withdrawal test in Oskarshamm 1 BWR in 1975
  - Performed by ASEA-ATOM to demonstrate PCI resistance of standard 8x8 Zr-2 fuel cladding.
  - Single control blade withdrawn in 10% steps with 2 hour holds.
- Peak powers at failure nodes ranged from 9.1 to 11.3 kW/ft.
- 45 fuel rods in 14 bundles failed by PCI.

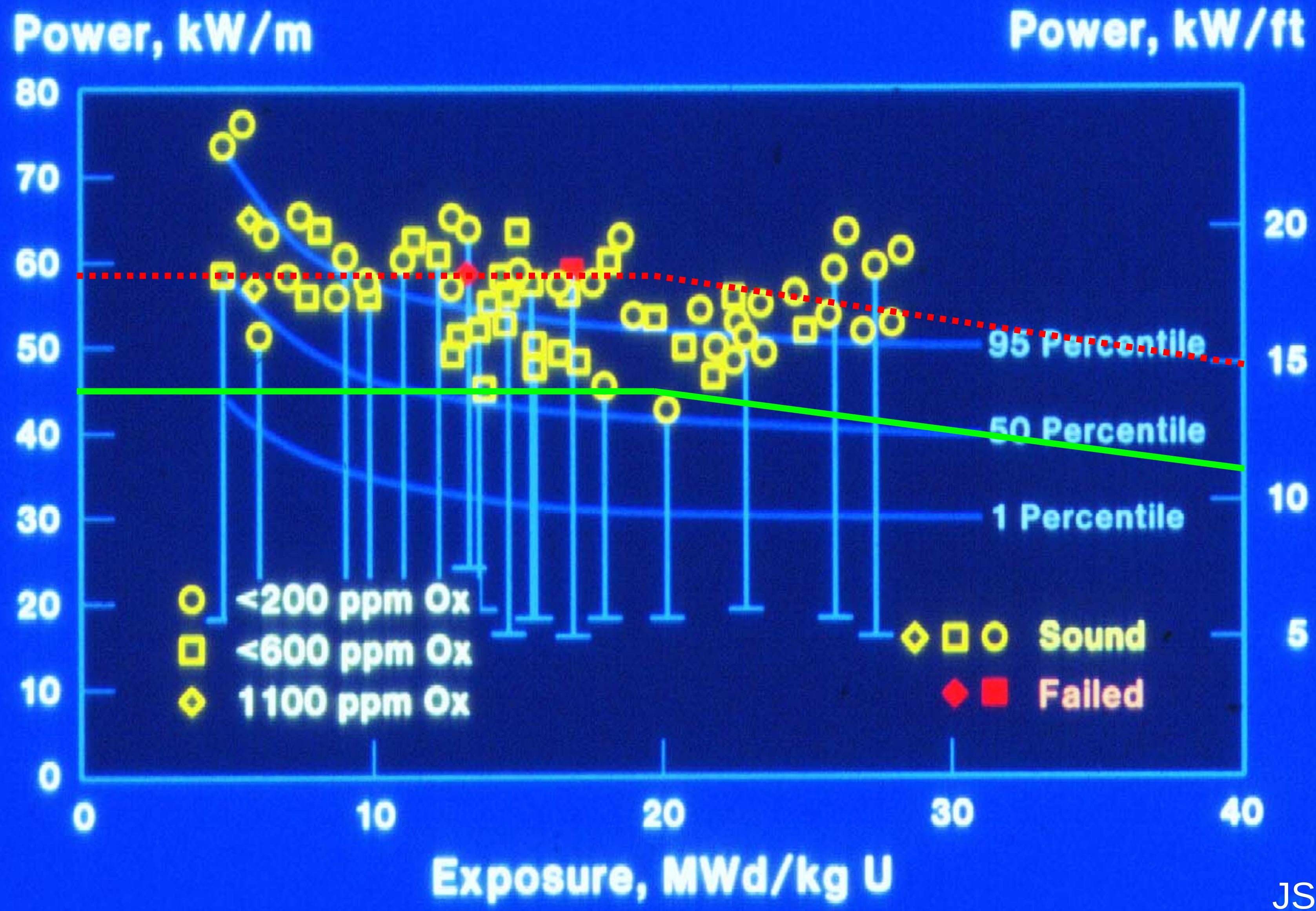
BWR Loss of Feedwater Heater

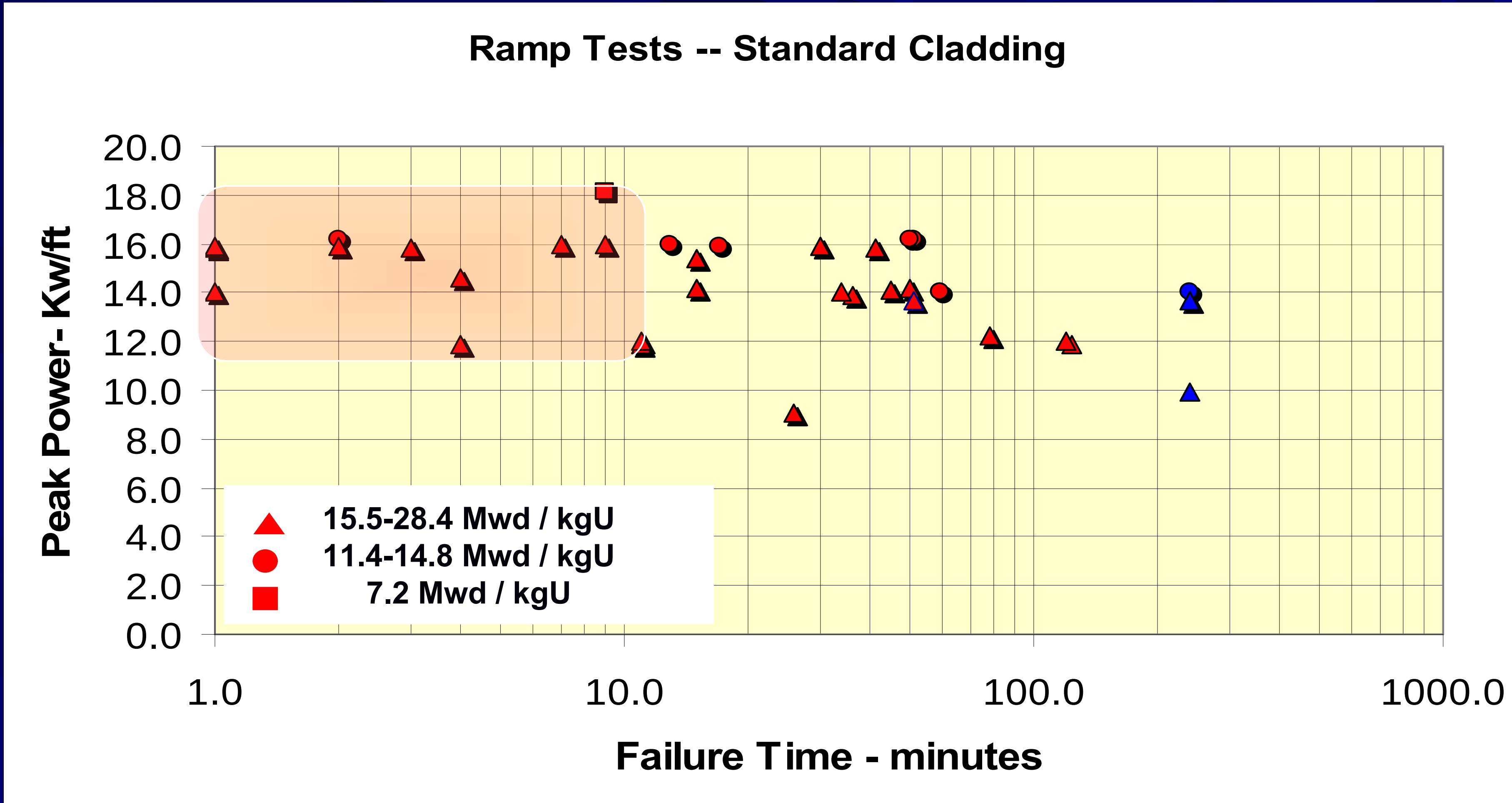
- Loss of feedwater heating results in core power increase due to core inlet subcooling.
- Most severe if feedwater heaters are bypassed
- Core power increases to ~120% of rated in one minute and is maintained until terminated by operator action.
- All fuel rods in the core are affected; peak rods can reach powers up to16 kW/ft depending on fuel design and plant state.

PCI Mitigation Options

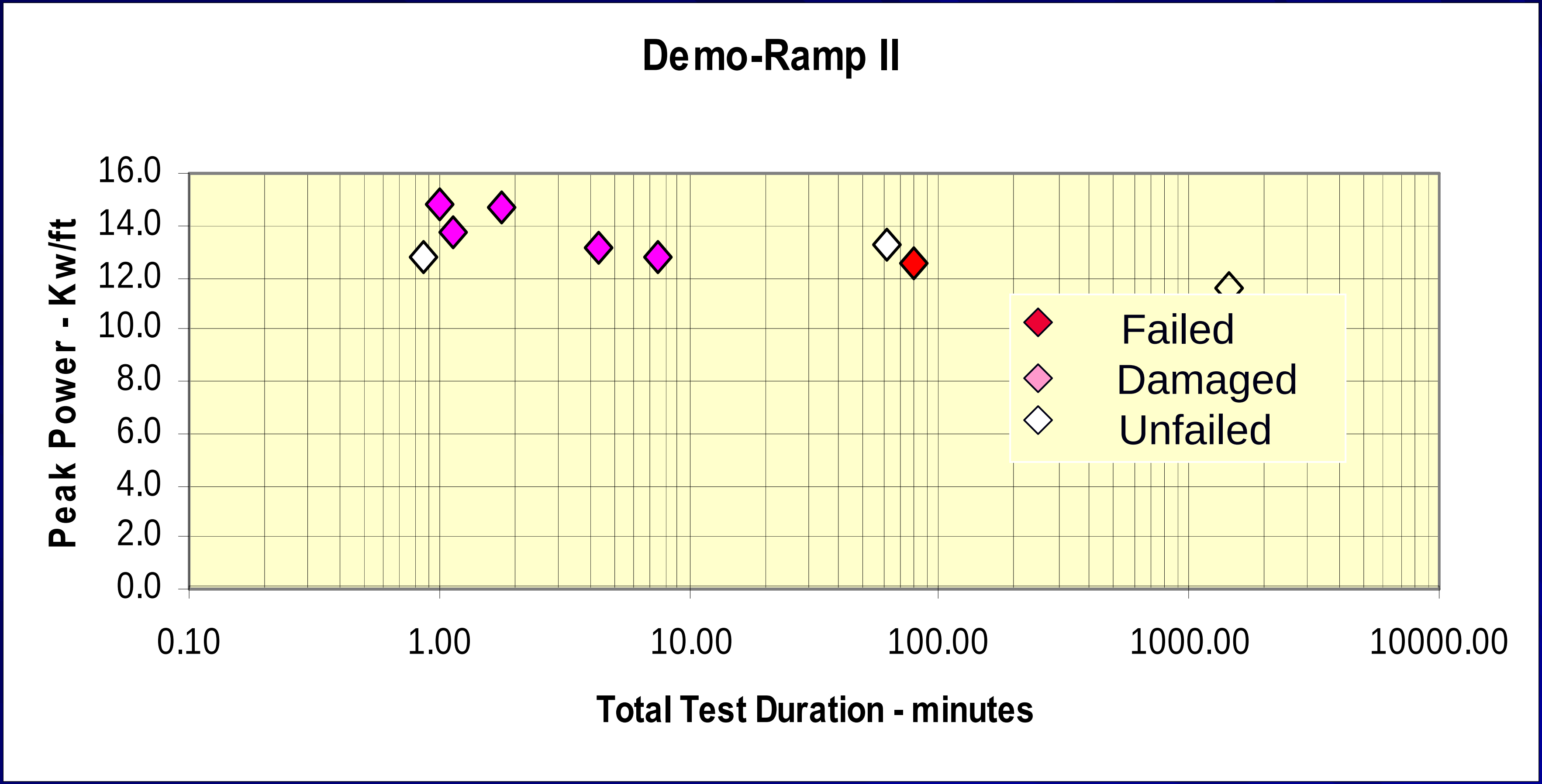
- Normal Operation
  - PCI resistant fuel
  - Preconditioning
- AOOs
  - PCI resistant fuel
  - Prompt operator action

PCI-Resistant Fuel



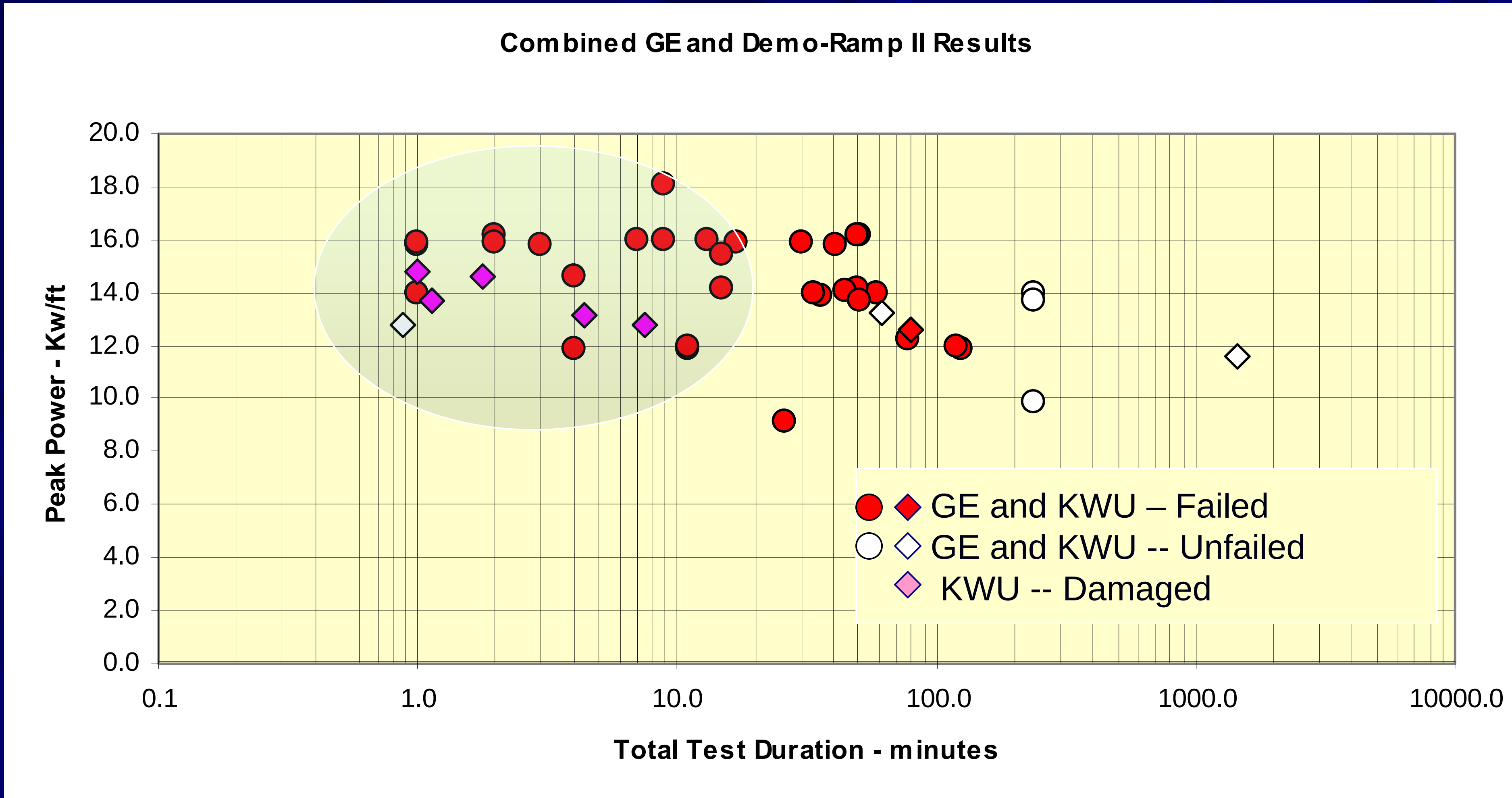


- GE Fuel Rods
- Irradiated in power reactors at low power
- Power ramped in R2 reactor
- 5/25 (19%) failed in 1 to 3 minutes



KWU Fuel Rods

- Irradiated in power reactors at low power
- Power ramped in R2 test reactor. Ramps intentionally terminated
- Five partial failures -- 10-60% thru-wall, in 1 to 7 minute tests
- One non-failed during 0.87 minute test



Combined Ramp Test Results

- Performance of GE and KWU test rods consistent
- Of the 36 rods tested:
  - 8 (22%) failed or were damaged within 3 minutes
  - 1 was not damaged during 0.87 minute test

Conclusions

- PCI failures are driven by chemistry and stress, not by strain.
- Strains required to cause PCI failures in conventional fuel are much lower that the 1% strain criterion.
- Current T-M regulatory criteria do not protect conventional fuel from PCI failure during AOOs
- PCI crack nucleation and propagation rates are fast enough to cause large numbers of conventional fuel failures during AOOs within one to three minutes.
- The number of fuel rods at risk increases with EPU.

Recommendations

- PCI failure criteria should be based on measured failure powers and failure times, not calculated failure strains.

- PCI resistance of specific fuel designs should be determined by power-ramp testing.

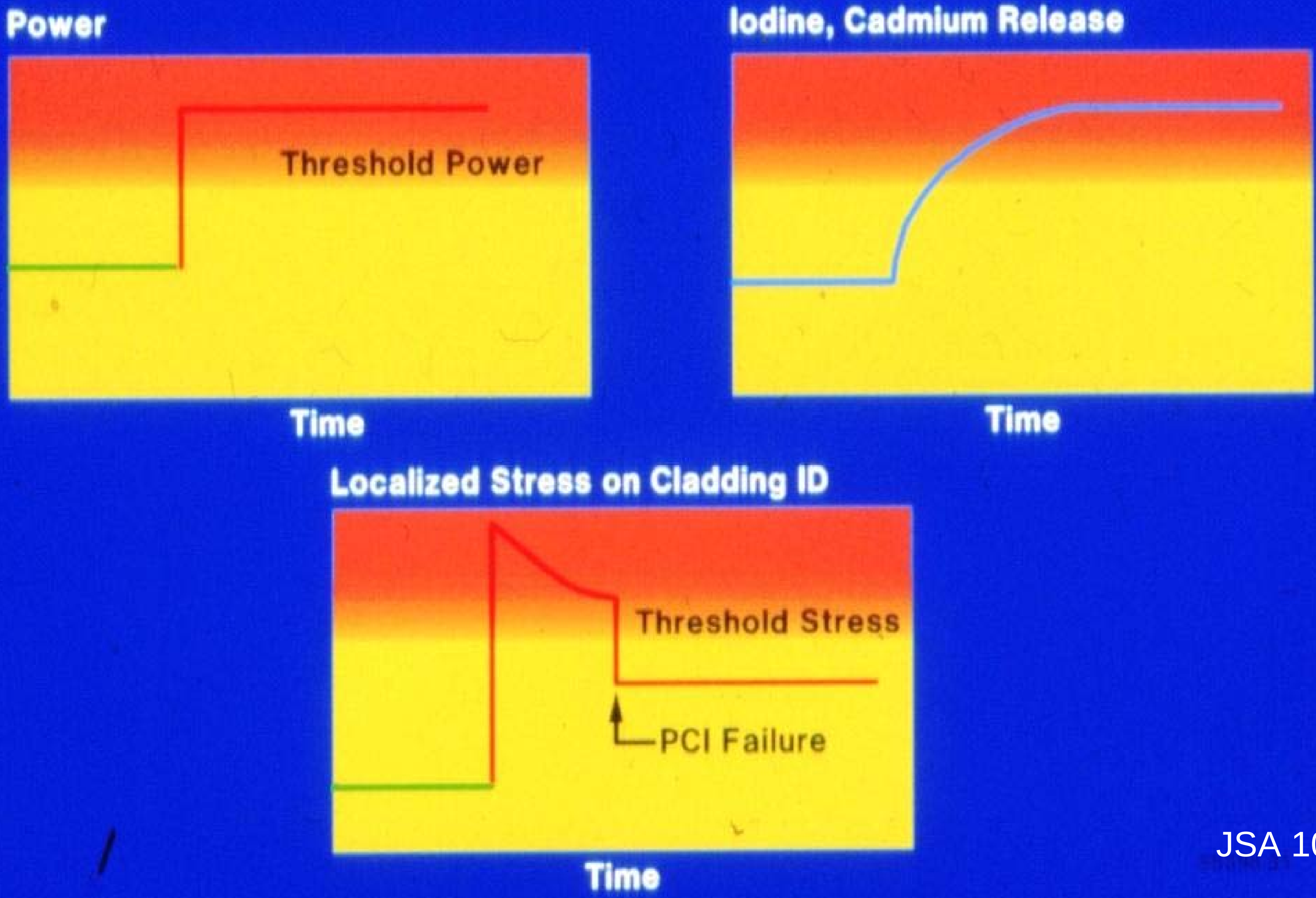
- Failure powers and failure times should be determined from statistically significant numbers of tests performed at conditions (power increase, peak power, time at peak power and burnup) expected during bounding AOOs

Backups

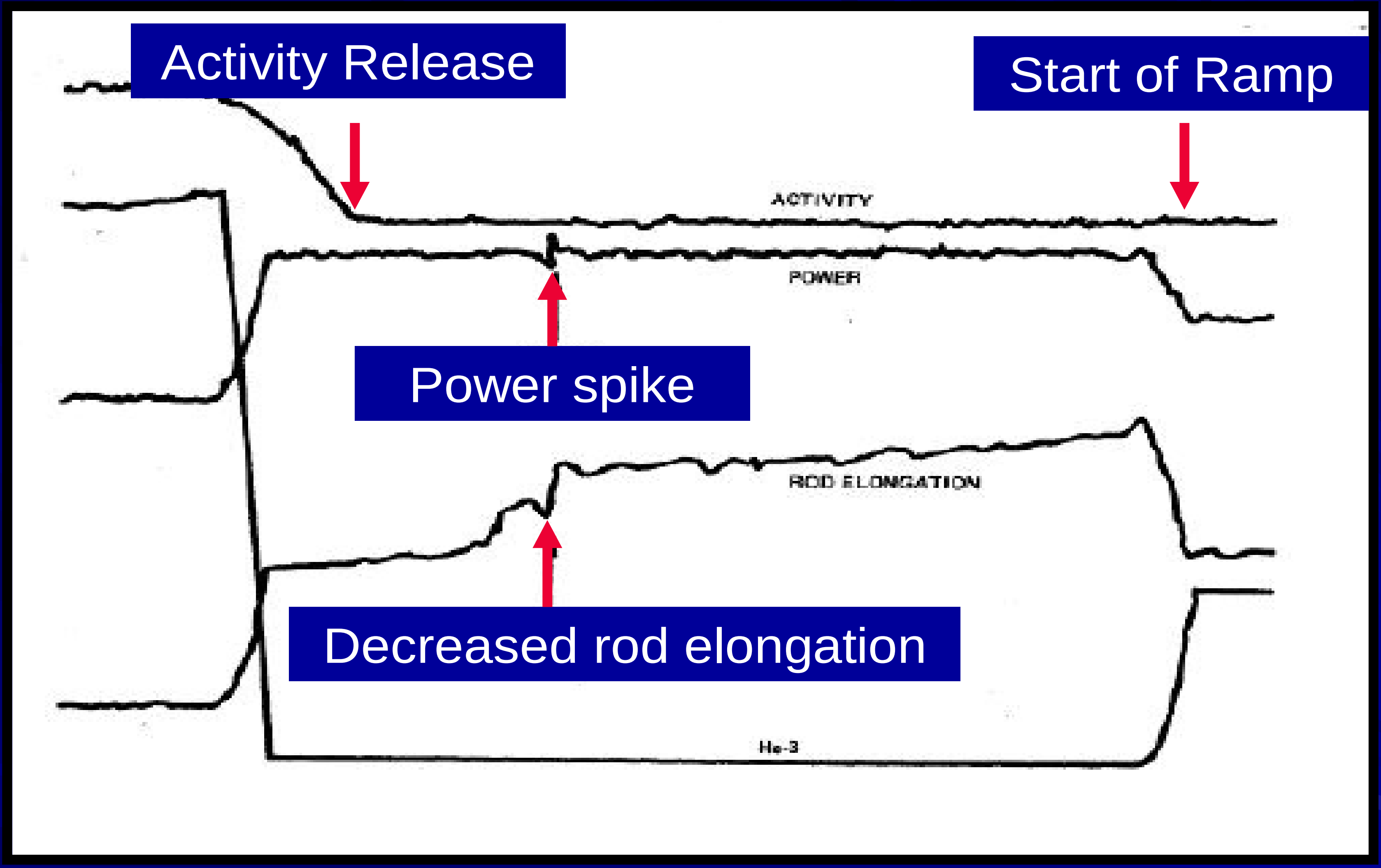
Tokar report to the ACRS on PCI -1979

- “Current plant safety analyses are, therefore, deficient in the sense that they do not, in general, account for PCI, which is now well recognized as a significant fuel failure mechanism.”
- “As the result of our past and on-going efforts on PCI, we believe that the time is right to start introducing PCI fuel failure analyses into plant safety analyses.”
- “...a major segment of the LWR industry holds that PCI failures will not occur during the type of power increasing transients and accidents addressed in Chapter 15 of the Standard Review Plan because the time at the increased transient power is too short.”

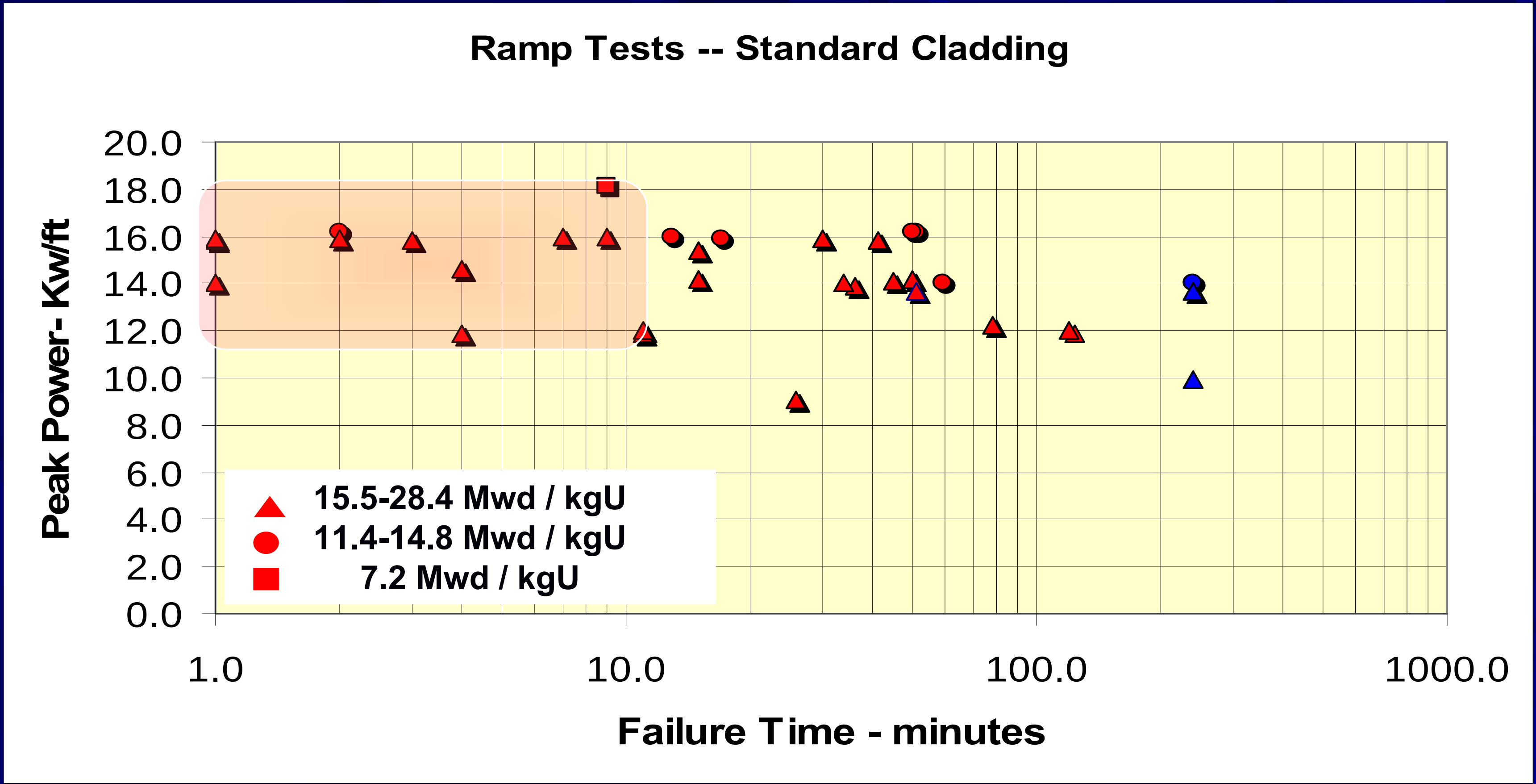
POWER RAMP EFFECTS



JSA 10

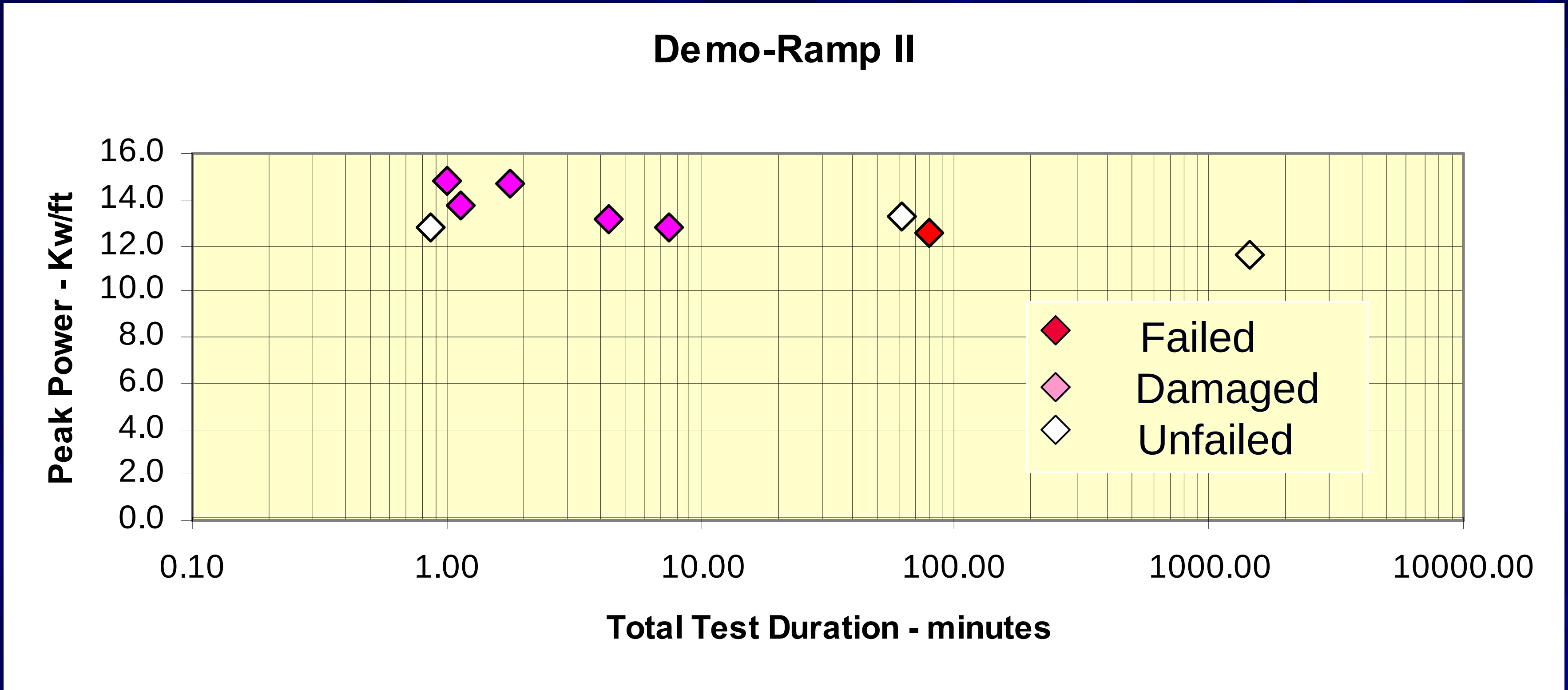


Ramp test time-to-failure detection methods



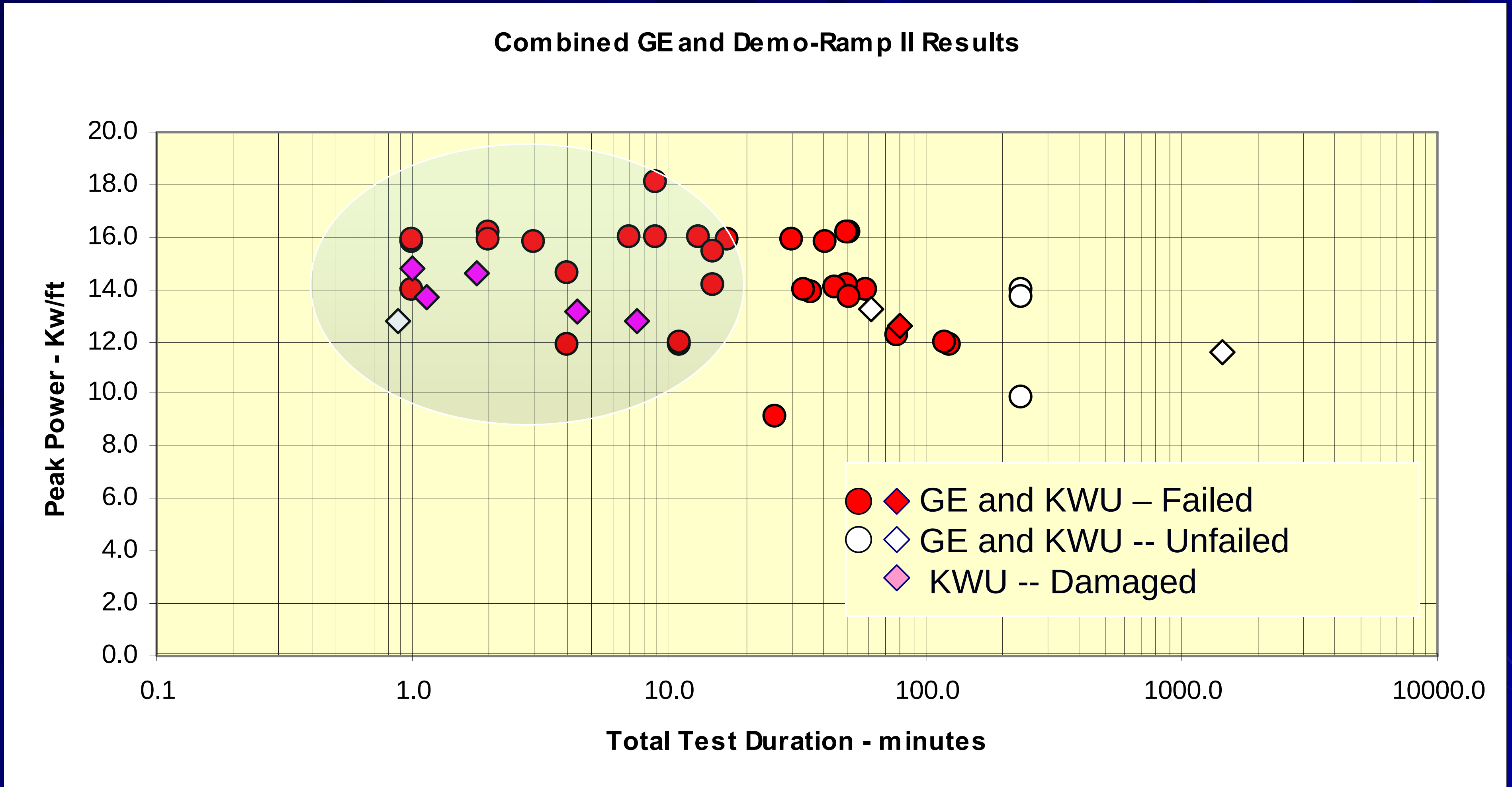
GE BWR Test Fuel Rods

- Irradiated in power reactors to burnups of 7- 28 Mwd / kgU at powers of 4-6 Kw/ft
- Pre test conditioned at 8 to 9 Kw/ft
- Power ramps of 2 to 8 Kw/ft at 2 to 100 Kw/ft-min
- 10 thru-wall PCI failures during 1 to 9 minute tests
- 17 thru-wall PCI failures during 10 to 110 minute tests
- 3 non-failed after 240 minute tests



KWU BWR Test Fuel Rods

- Irradiated in power reactors at 5 to 9 Kw/ft to burnups of 25 to 29 Gwd/t
- Pre-ramp conditioned at 9 Kw/ft
- Power ramps of 1.7 to 5.6 Kw/ft at rates of 1.6 to 10 Kw/ft-min
- One thru wall failure – during 79.8 minute test
- Five partial failures -- 10-60% thru-wall, during1 to 7 minute tests
- One non-failed during 0.87 minute test
- One non-failed during 61 minute and 1440 minute tests (same rod ramped twice)



Combined Ramp Test Results

- **Performance of GE and KWU test rods comparable**
- **Of the 16 tests with durations less than 10 minutes**
  - **9 failed with thru-wall PCI cracks – 5 failed within 3 minutes**
  - **6 had PCI cracks 10 to 60 % thru-wall – deepest occurred within 2 minutes**
  - **1 was not damaged during 0.87 minute test**

GEH Proprietary - Backups

GEH LFWH Analysis

NEDE-32538P-A
GE Proprietary Information

Table 4
LFWH Core Power and ACPR

Parameter	TRACG	PANACEA (Inlet Conditions from TRACG)
Core power to moderator for a 100°F LFWH (% of rated)	118.0 at 50 seconds	117.7
ACPR for a 100°F LFWH (M adjusted to the safety limit 1.07)	0.103 at 50 seconds	0.115

Browns' Ferry Plant Safety Analysis

FSAR BFN 16 Table 14.4-1Summary of Abnormal Operational Transients

Undesired Parameter Variation	Event Causing Transient	Scram Caused by
Nuclear system pressure increase	Generator trip without bypass	Turbine control valve fast closure
Nuclear system pressure increase	Turbine trip without bypass	Turbine stop valve closure
Nuclear system pressure increase	Main steam line isolation valve closure	Main steam line isolation valve closure
Nuclear system pressure increase	Loss of Condenser vacuum	Turbine stop valve closure
Nuclear system pressure increase	Bypass valve malfunction	Reactor vessel high pressure
Nuclear system pressure increase	Pressure regulator malfunction	Reactor vessel high pressure
Reactor water temperature decrease	Shutdown cooling malfunction decrease temperature	High Neutron flux
Reactor water temperature decrease	Loss of feedwater heater*	None
Reactor Water temperature decrease	Inadvertent pump start*	None

- Amendment 7 of GESTAR II SER transmittal letter states:
  - ***Should our criteria or regulations change so that our conclusions as to the acceptability of the report are invalidated, GE and /or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation."***

Fuel Design Criteria Licensing Clause

- Item 4 of Amendment 22 to GESTAR II states:

  " New-fuel-related licensing issues identified by the NRC will be evaluated to determine if the current criteria properly address the concern; if necessary, new criteria will be proposed to the NRC for approval."

- CONCLUSION

  – ***Obligation of licensees and fuel vendors to demonstrate that their fuel designs as operated will preclude known damage mechanism and meet GDC-10 and GDC-12 requirements***

PCI Summary

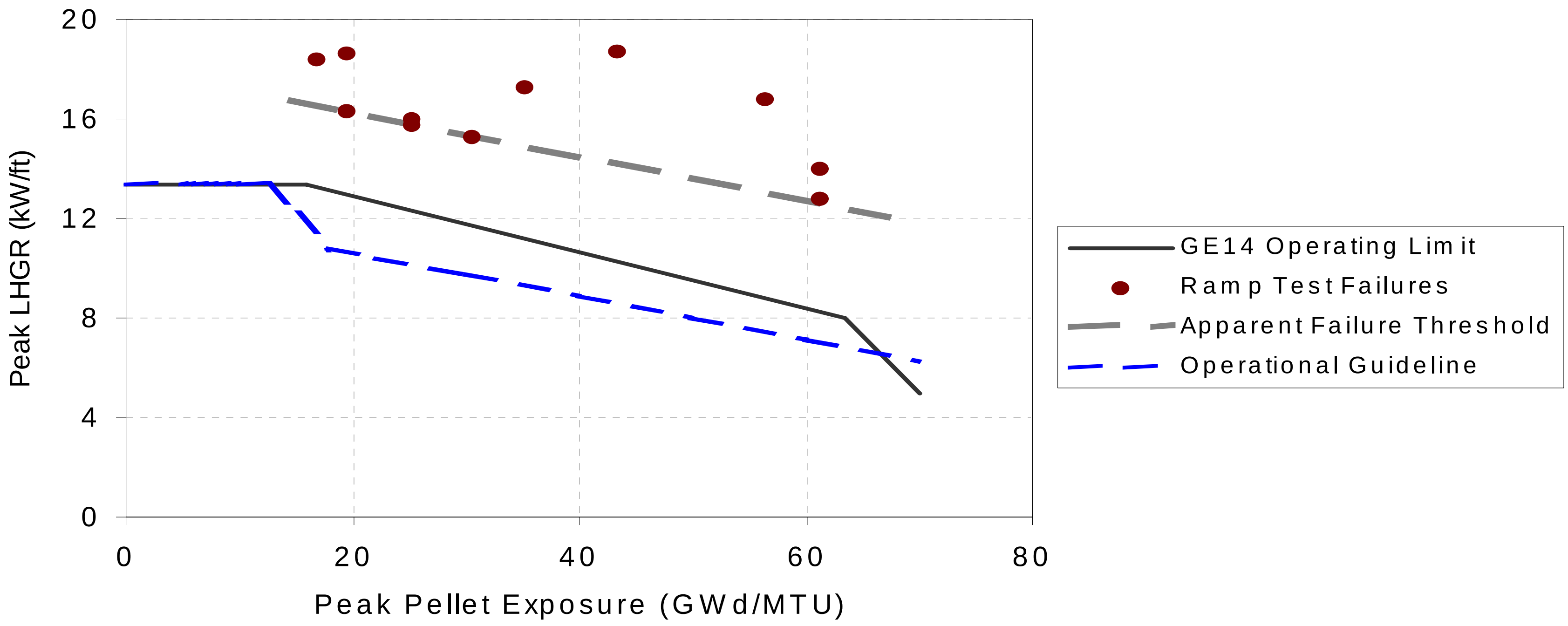
GE14 and SVEA-96+ fuel designs include barrier cladding which significantly reduces the PCI/SCC failure potential

No change in fuel duty/margin to the LHGR for the pre-EPU versus EPU conditions

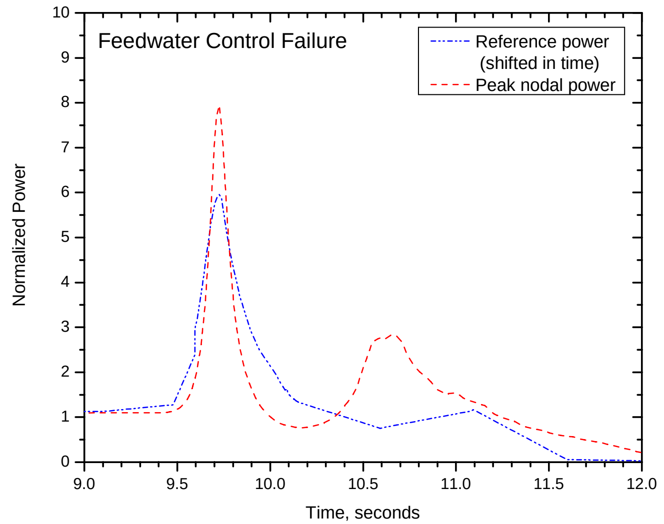
Operational guidelines provide additional margin to avoid PCI/SCC type fuel failures

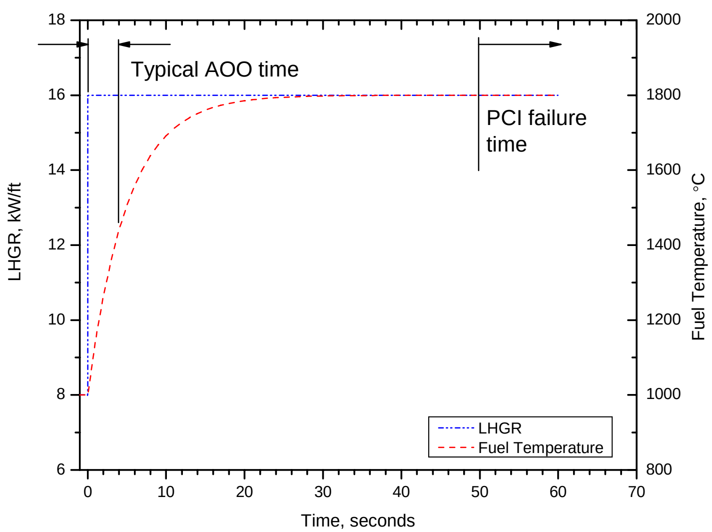
Hope Creek Uses Operating Guidelines to Reduce the Potential for PCI/SCC Type Fuel Failures

- Significant margin compared to ramp tests
  - Apparent failure stress threshold is ~60 ksi
  - Calculated stress at the typical operational guideline threshold is ~15 ksi



PCI Failure during AOOs



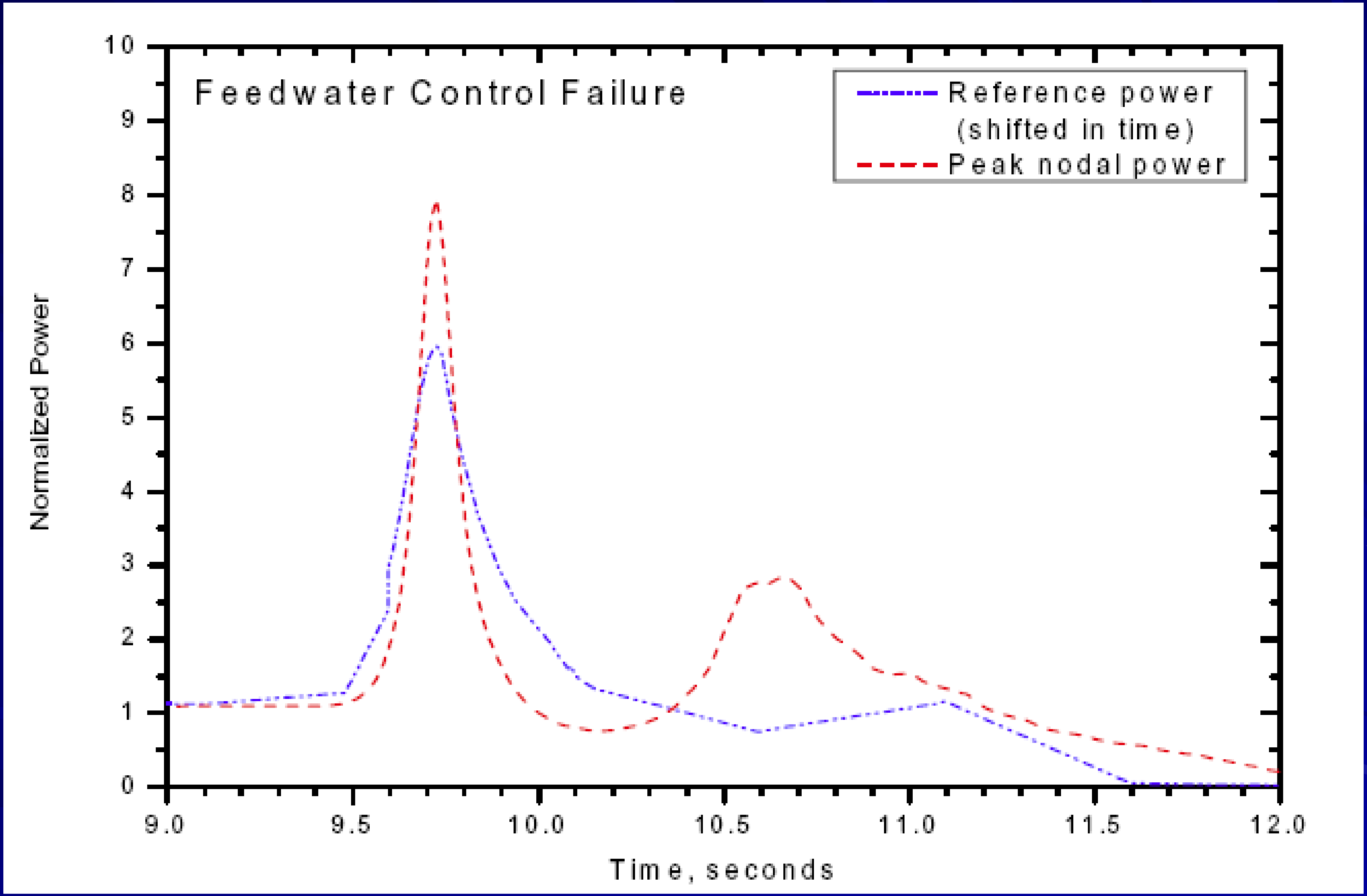


PCI failures can result from elastic loading, without plastic strain

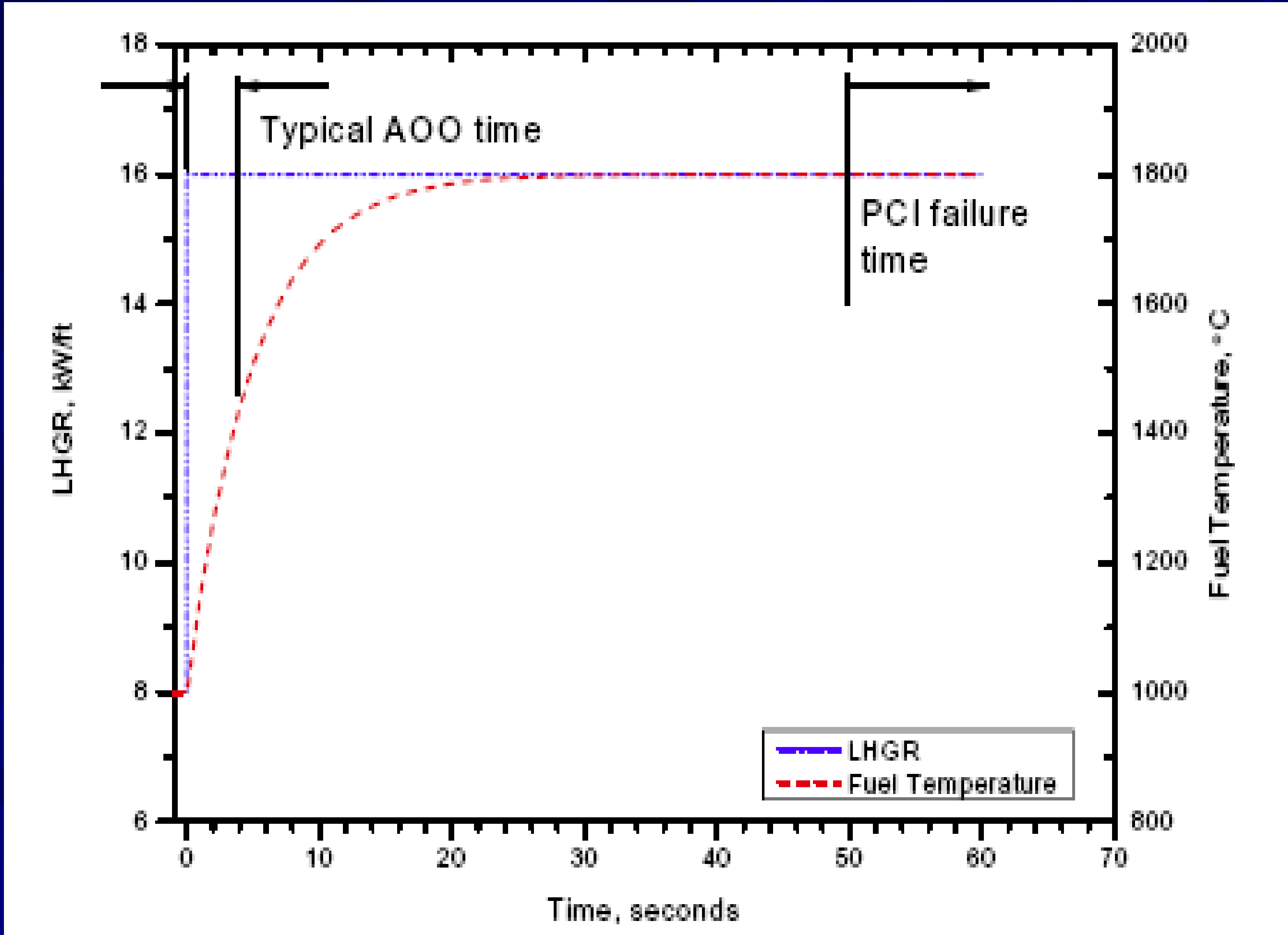
- Requires time, temperature and stress
- Very localized and stochastic in nature

AOOs are events of short duration

- Not enough time for the corrosive fission products release and cause PCI/SCC



Feedwater Control Failure
Reference power
(shifted in time)
Peak nodal power
Normalized Power
Time, seconds
0
1
2
3
4
5
6
7
8
9
10
9.0
9.5
10.0
10.5
11.0
11.5
12.0



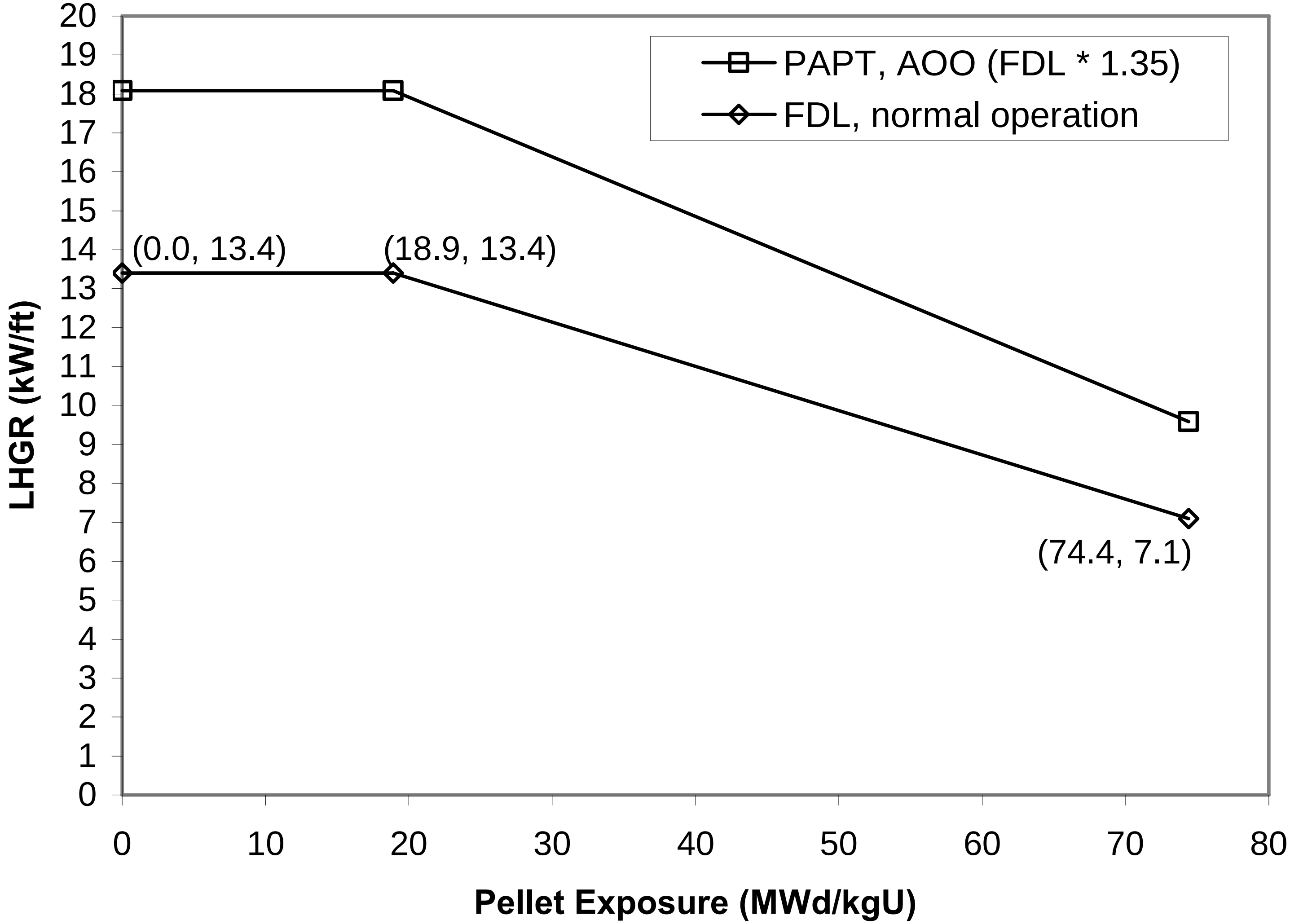
Typical AOO time
PCI failure time
LHGR
Fuel Temperature
LHGR, kW/ft
Fuel Temperature, °C
Time, seconds
18
16
14
12
10
8
6
2000
1800
1600
1400
1200
1000
800
0
10
20
30
40
50
60
70

AREVA Proprietary -- Backups

Thermal Mechanical Methods

Michael Garrett
Manager, BWR Safety Analysis

LHGR limits for ATRIUM™-10 fuel



Thermal Mechanical Methods

- Fuel thermal mechanical limits remain unchanged for Susquehanna CPPU operation
- Fuel rod linear heat generation rate (LHGR) limits are established using NRC-approved thermal mechanical methods
  - The Fuel Design Limit (FDL) LHGR ensures that fuel thermal mechanical design criteria (e.g., rod internal pressure) are not exceeded during steady state operation
  - The Protection Against Power Transients (PAPT) LHGR limit ensures fuel SAFDLs (<1% cladding strain and no centerline melting) are not exceeded during anticipated operational occurrences (AOOs)
  - Neutronic design criteria ensure gadolinia rods are not limiting with respect to thermal mechanical criteria
  - Same FDL and PAPT limits for liner or non-liner cladding

Thermal Mechanical Methods Liner Cladding and Standard Cladding

- Susquehanna uses standard (non-liner) cladding
  - Majority of ATRIUM-10 fuel supplied is non-liner
  - ATRIUM-10 failure-free operation in both Susquehanna units since introduction in 1997
  - FDL and PAPT limits are unchanged from pre-CPPU operation
- Use of liner cladding provides less restrictive maneuvering (power ramp rate) guidelines
  - Liner does not impact FDL or PAPT limits
  - Liner does not provide additional protection for SAFDLs

Thermal Mechanical Methods

- Cycle specific transient analyses performed to establish an operating limit LHGR that ensures PAPT limit is not exceeded during an AOO
- Potentially limiting AOOs were analyzed at CPPU conditions for Susquehanna
  - Limiting event for normal operation (loss of feedwater heating) resulted in overpower ratio of [24% (PAPT based on 35% overpower ratio)]
- The operating limit LHGR is specified in the COLR
- The core monitoring system is used to ensure the core is operated within the operating limit LHGR

PCI/SCC Regulatory Approach

ACRS Full Committee Meeting

June 3, 2009

Paul M. Clifford
Division of Safety Systems
Nuclear Reactor Regulation

ACRS Letter on Susquehanna EPU (December 20, 2007)

- The staff should develop the capability and perform a thorough review and assessment of the risk of pellet-cladding interaction (PCI) fuel failures with conventional fuel cladding, during anticipated operational occurrences (AOOs).
  - The staff should develop qualified analytical tools to demonstrate that operator actions will assure an acceptably low number of failures. If this can be demonstrated by analysis, then the required operator actions should be incorporated into the regulatory process through commitments or inclusion in the updated FSAR.

Staff Response to ACRS Letter (January 17, 2008)

- In response to recommendation 6, the NRC staff will investigate current computational capabilities to model the complex phenomena associated with non-uniform fuel pellet expansion and stress-corrosion cracking (SCC). As necessary, the staff will develop guidance related to an application methodology and regulatory approach for implementing a PCI/SCC fuel failure criteria.

Staff Concerns with ACRS Direction

Staff concerns with the specific direction:

- PCI/SCC phenomena difficult to model and requires tacit assumptions on chemical effects and initial crack depth.
- All domestic fuel designs susceptible to PCI/SCC
  - Various design features (e.g. doped pellets, low alloy Zr barrier, natural Zr barrier) provide varying levels of PCI/SCC resistance
  - Barrier fuel design provides PCI/SCC resistance, but not immune from failure during power maneuvering or AOOs
- Crediting prompt operator action in UFSAR Chapter 15

Important points to consider moving forward:

- Regulations specify performance requirements
  - Does not impose specific design features
- Regulations apply universally
  - Not restricted to a particular fuel of cladding design
- PCI/SCC not strictly an EPU issue or BWR issue

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PCI/SCC Work Priority

- PCI/SCC may yield fuel rod cladding failure (i.e., through wall crack releasing fission gas within plenum)
  - No challenge to core coolable geometry
  - No challenge to pressure vessel integrity
  - No challenge to containment integrity
  - No challenge to systems designed to mitigate transient and minimize offsite activity releases
- PCI/SCC safety significance does not warrant immediate action nor higher priority in staff workload planning than ongoing regulatory improvements.
  - Revision to 10 CFR 50.46(b) ECCS Acceptance Criteria
  - Revision to RG 1.183 Gap Source Terms
  - Revision to RG 1.77 RIA Acceptance Criteria

Change in Staff Position

- No regulations or Regulatory Guides specifically address PCI/SCC.
- Cladding failure mechanisms and SAFDLs defined within approved topical reports and captured within each plant's licensing basis via Technical Specifications and UFSAR.
- Any change to the treatment of PCI/SCC would constitute a **change in a regulatory staff position**.
  - Consider 10 CFR 50.109 "Backfitting" requirements.
  - Complete Regulatory Analysis (NUREG/BR-0058, Rev.04).

Proposed Strategies for PCI/SCC

Alternative strategies:

1. Maintain current approach
2. PCI/SCC protection based on empirical failure threshold
3. PCI/SCC protection based on analytical models

Proposed Strategy #1

Maintain Current Approach

- PROS:
  - Current approach provides reasonable level of protection during core-wide AOOs.
  - Staff resources devoted to more substantial regulatory improvements.
- CONS:
  - Potential fuel cladding breach during certain BWR AOOs due to PCI/SCC.
  - Lack of specific PCI/SCC guidance and regulatory criteria for future fuel designs.

Proposed Strategy #2

PCI/SCC Protection based on Empirical Failure Threshold

- Revise SRP-4.2 guidance on PCI/SCC fuel failure mechanism and level of qualification to demonstrate no fuel failures during AOOs.
  - Quantification of PCI/SCC resistance of all fuel designs under AOO conditions.
  - Empirically derived fuel rod failure threshold based on change in rod power and elapsed time.
  - Calculated rod powers remain below empirical failure threshold during UFSAR Chapter 15 AOOs.
- PROS:
  - Strict compliance with GDC10
  - High confidence predictions on rod power history
  - Consistent with reactivity-initiated accident regulatory approach.
- CONS:
  - Will require empirical data from power ramp testing (facilities limited).
  - May necessitate changes in Operator procedures and training and/or PPS reactor trip setpoints.
  - Implementation costs expected to be high for industry and NRC.

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Proposed Strategy #3

PCI/SCC Protection based on Analytical Models

- Revise SRP-4.2 guidance on PCI/SCC fuel failure mechanism and level of qualification to demonstrate no fuel failures during AOOs.
  - Verification and validation of analytical models capable of predicting, at high confidence levels, crack tip propagation and cladding failure under combined mechanical loading and chemical attach.
  - Calculated cladding stresses remain below analytical failure predictions during UFSAR Chapter 15 AOOs.
- PROS:
  - Strict compliance with GDC10
- CONS:
  - Will require empirical data from power ramp testing (facilities limited) to calibrate analytical models.
  - PCI/SCC phenomena difficult to model and requires tacit assumptions on chemical effects and initial crack depth.
  - No well-verified analytical models exist.
  - Standard modeling approach (95/95) likely to yield overly burdensome requirements.
  - May necessitate changes in Operator procedures and training and/or PPS reactor trip setpoints.
  - Implementation costs expected to be <u>very</u> high for industry and NRC.

Backfitting

- 10 CFR 50.109 “Backfitting” represents a regulatory hurdle for implementing changes in staff positions to currently licensed facilities.
- The rule requires a substantial increase in the overall protection of the public health and safety **and** that the direct and indirect costs of implementation for that facility are justified in view of this increased protection.
- It would be difficult to justify an exception to this rule under “compliance” or “adequate protection”.

Forward Fitting

- No regulatory expectation that requirements or staff positions remain stable for future requests for agency action/approval.
  - Expanding fuel failure mechanisms to explicitly account for PCI/SCC for future fuel designs is not a backfit.
- Due to ongoing fuel design enhancements, implementing forward-fit PCI/SCC requirements likely to encompass a majority of the fleet in a reasonable timeframe.
  - Application of forward-fit PCI/SCC requirements to licensing actions (e.g., EPUs) involving existing, approved fuel design?
- Regulatory Analysis needed to justify change in staff position.

QUESTIONS?

Backup Slides

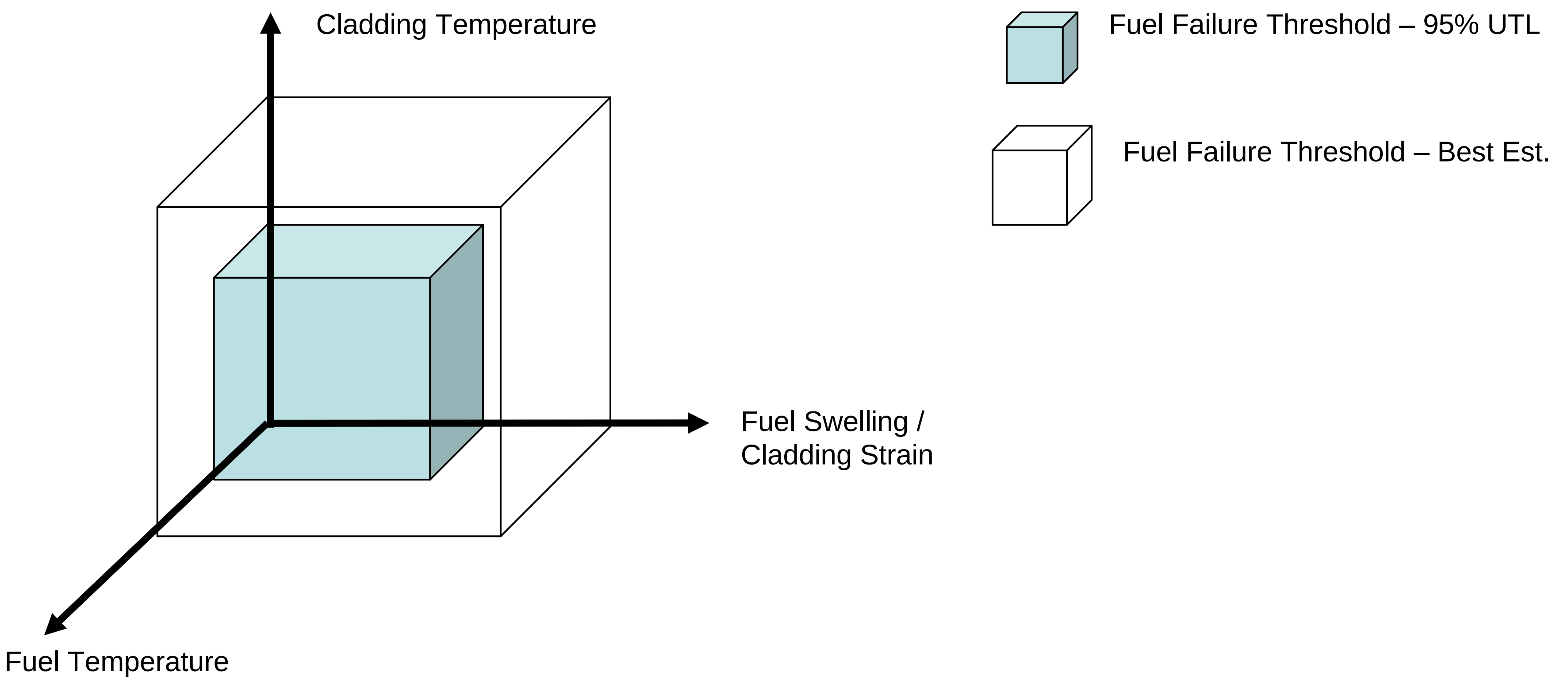
Low Safety Significance

- PCI/SCC may yield fuel rod cladding failure (i.e., through wall crack releasing fission gas within plenum)
  - No challenge to core coolable geometry
  - No challenge to pressure vessel integrity
  - No challenge to containment integrity
  - No challenge to systems designed to mitigate transient and minimize offsite activity releases

Low Probability of Occurrence

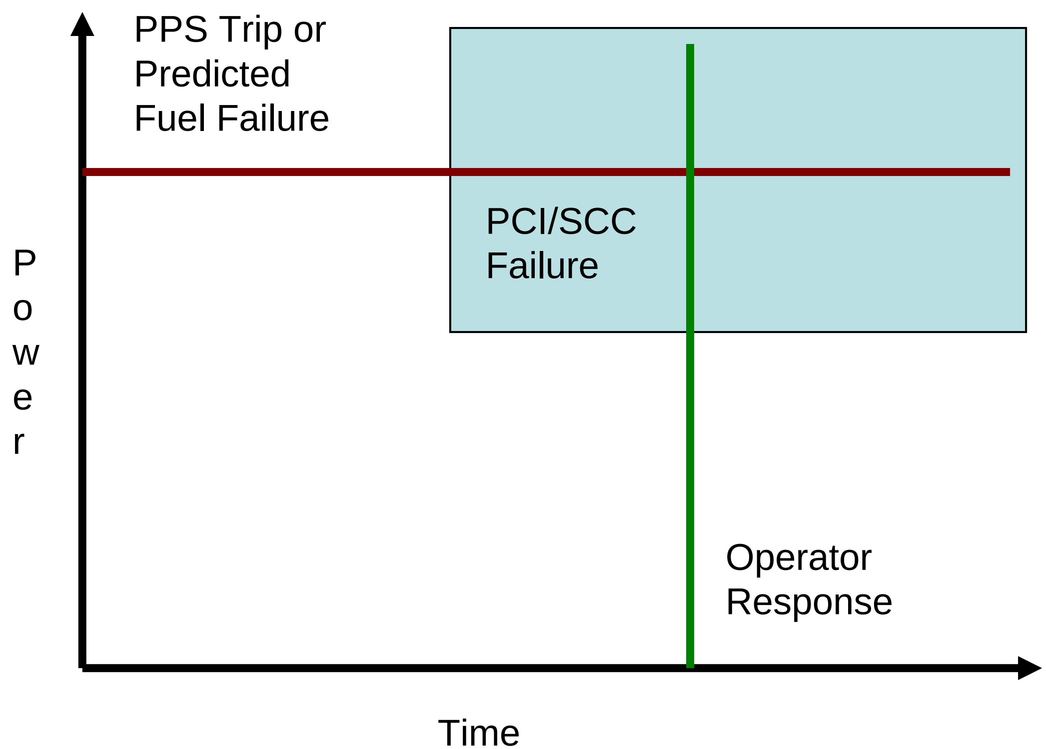
Limited envelope on magnitude of power excursion

- Power level must remain below automatic trip setpoint.
- Power level must remain below level which results in predicted fuel failure calculated using conservative analytical models along with conservative assumptions and initial conditions.



Limited envelope on duration of power excursion

- Duration beyond time necessary for PCI/SCC crack growth.
- Duration below timing for reasonable Operator response.



Current Staff Guidance

Standard Review Plan Section 4.2.II.1.B.vi, “Pellet/Cladding Interaction”

- *Two related criteria should be applied, but they are not sufficient to preclude PCI or PCMI failures. The first criterion limits uniform strain of the cladding to no more than 1 percent. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gauge lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Mechanical testing must demonstrate that the irradiated cladding ductility at maximum waterside corrosion (hydride embrittlement) is well within the 1-percent strain criterion. Although observing this strain limit may preclude some PCI and PCMI failures, it will neither preclude the corrosion-assisted failures that occur at low strains nor the highly localized overstrain failures introduced by pellet chips on the outer fuel diameter. The second criterion states that fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the cladding. Avoiding fuel melting can preclude such a PCI. Note that item 1.B.iv above invoked this same criterion to ensure that overheating of the cladding would not occur.*
- *Fuel vendors have introduced fuel design limits on power maneuvering and rate of power ascension to prevent PCI or PCMI. These design limits have primarily been based on power ramp data from test reactors for a specific fuel design. Recently, however, fuel vendors have been relying more on their predictions of cladding strain and less on their power ramp data to verify that PCMI will not occur. Convincing evidence exists that gaseous swelling and fuel thermal expansion is responsible for cladding strains at high burnup levels and perhaps at even moderate burnups. Therefore, PCI or PCMI analyses of cladding strain for AOO transients and accidents should apply approved fuel thermal expansion and gaseous fuel swelling models, as well as irradiated cladding properties.*

10 CFR 50.109 Backfitting

(a)(1) Backfitting is defined as the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position after:

10 CFR 50.109 Backfitting (con't.)

(3) Except as provided in paragraph (a)(4) of this section, the Commission shall require the backfitting of a facility only when it determines, based on the analysis described in paragraph (c) of this section, that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit **and** that the direct and indirect costs of implementation for that facility are justified in view of this increased protection.

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10 CFR 50.109 Backfitting (con't.)

(4) The provisions of paragraphs (a)(2) and (a)(3) of this section are inapplicable and, therefore, backfit analysis is not required and the standards in paragraph (a)(3) of this section do not apply where the Commission or staff, as appropriate, finds and declares, with appropriated documented evaluation for its finding, either:

(i) That a modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee; or

(ii) That regulatory action is necessary to ensure that the facility provides adequate protection to the health and safety of the public and is in accord with the common defense and security; or

(iii) That the regulatory action involves defining or redefining what level of protection to the public health and safety or common defense and security should be regarded as adequate

US-APWR

Diversity and Defense-In-Depth

Topical Report

Safety Evaluation

To:
Advisory Committee on Reactor Safeguards

By:
Royce D. Beacom
Instrumentation, Controls and Electrical Engineering Branch (ICE1)
Office of New Reactors

Wednesday, June 3, 2009

Agenda

- Diversity and Defense-in-Depth Scope
- Findings and Conclusions
- Listing Sub-Committee Points of Discussion
- Addressing Each Point of Discussion

Diversity and Defense-in-Depth Scope

- Diversity with Safety and Non Safety Systems
  - Protection and Safety Monitoring System (Safety)
  - Plant Control and Monitoring System (Non-Safety)
- Both using the MELTAC Platform

Diversity and Defense-in-Depth Scope

- Functionality of the Diverse Actuation System (DAS) – (analog & non-safety)
  - Provides a defensive measure to cope with Anticipated Operational Occurrence (AOO) or Postulated Accident (PA) concurrent with Common Cause Failure in the PSMS which is beyond design basis

Diversity and Defense-in-Depth Scope

- Provides the ATWS Mitigation Function
- Provides Automatic Actuations where time is insufficient for manual operator action; MHI Proposed: < 10 mins
  - Delay from anticipated PSMS trip
  - Proper actuation of PSMS blocks DAS

Diversity and Defense-in-Depth Scope

- DAS Manual Actuation
  - Separate HSI Panel with conventional Controls and Indicators
  - Proposed < 30 min from Prompting Alarm
- Isolated signals from sensors, shared with PSMS, provided to Non-Safety DAS
- DAS Outputs to discrete portion of Power Interface Module (PIF)

Findings and Conclusions

- LBLOCA Coping Strategy
  - High quality, high reliability, measures of MELTAC within the RPS/ESFAS design
  - Low frequency of AOO and PA events
  - Supplemented with DAS leak protection
- LBLOCA Strategy unacceptable
  - Frequency of AOO / PAs still finite possibility
  - Leak-Before-Break doesn't apply here

Findings and Conclusions

- Protective action – Manual vs Automatic
  - MHI "Target" $\leq$ 10 minutes - Automatic
  - > 10 minutes – Manual Action is assumed
    - Differs from DI&C-ISG-02; < 30 Minutes – Auto
    - Insufficient information to assess manual action between 10 min and 30 min following the event
- Justification for manual actions within 30 minutes – US-APWR HSI Certification

Findings and Conclusions

- The staff concluded that the D3 approach, and the D3 analysis provided per NUREG-6303, had met the acceptable bases for conforming to the requirements and supporting industry standards.
- Subject to satisfactory completion of Application Specific Action Items (ASAI)

Sub-Committee Points of Discussion

The subcommittee meeting identified these points of discussion for the staff:

- How D3 fits in the Overview of US-APWR
- Separate approval of D3 from US-APWR?
- Bypassing DAS; PSMS actuation & Startup/ Shutdown
- Concept of two DAS Subsystems

Sub-Committee Points of Discussion

- ASAI on partial output failure from CCF
- Three DAS Inputs to Rx & Turbine Trip

How D3 fits in the Overview of US-APWR

- New reactor design only – US-APWR
  - MHI intent was for Operating Fleet also
- These Topical Reports are stand alone and will have separate SER's:
  - Safety I&C Sys. Description & Process, D3, MELTAC, HSI System Design & Process

How D3 fits in the Overview of US-APWR

- The safety evaluations of these Technical Reports will be included in the DCD SER
  - Defense-In-Depth Coping Analysis
  - Software Program Manual (Application SW)
- Application Specific Action Items (ASAIs) can be addressed in the following:
  - Directly in the DCD (Rev); ITAAC; COL Action Item

Separate approval of D3 from US-APWR?

- For attributes approved, level of detail is sufficient
  - Will not expect additional detail in DCD
- Staff is confident ASAIs are sufficient
  - Will address additional D3 Info needed
    - Particularly pointers to Coping Analysis

Separate approval of D3 from US-APWR?

- If Applicant/ Licensee cannot meet ASAIs
  - It is their risk to proceed with design or
  - Take exception to TR
    - Provide alternative path for staff approval

Bypassing DAS; PSMS actuation & During Startup/ Shutdown

- Diverse Actuation System (DAS) is bypassed when PSMS actuates:
  - If proper feedback from actuated components is received
  - Prevents unexpected competition between systems

Bypassing DAS; PSMS actuation & During Startup/ Shutdown

- DAS is bypassed at same time as PSMS
  - Required in Modes 1,2 & 3 and Pressurizer Pressure > P11
- However, enabled by different means:
  - PSMS is automatically interlocked
  - DAS is enabled by manual switch

Concept of DAS Subsystems

- Two subsystems balances two issues:
  - – Reliability & Spurious actuation of the DAS
- There are no 50.55(a)(h)/ IEEE 603 safety requirements applicable since this is a non-1E system
    - i.e. single failure, independence, EQ, quality etc.
  - – GL 85-06, ATWS QA applies (Not App. B)

ASAI on partial output failure from CCF

- Concept captured by the D3 Task Working Group of the DI&C Steering Committee.
- The staff position states that a simple failure of the total system may not be the worst case failure, particularly when analyzing the time required for identifying and responding to the condition. For example, a failure to trip may not be as limiting as a partial actuation of the emergency core cooling system, with indication of a successful actuation

ASAI on partial output failure from CCF

- At least two requirements from 50.59(a)(h)/ IEEE-603 are applicable:
  - **Completion of Protective Action**. The safety systems shall be designed so that, once initiated automatically or manually, the intended sequence of protective actions of the execute features shall continue until completion

ASAI on partial output failure from CCF

– **System Status Indication.** Display instrumentation shall provide accurate, complete, and timely information pertinent to safety system status. This information shall include indication and identification of protective actions of the sense and command features and execute features.

ASAI on partial output failure from CCF

- Staff has not proposed a method for addressing this issue – nor should they
- All digital upgrades eventually will have to address this issue

Three DAS Inputs to Automatic Reactor and Turbine Trip

- The DAS has three diverse automatic actuation functions to shut down the reactor and to achieve secondary system core heat removal.
  - High Pressurizer Pressure
  - Low Pressurizer Pressure
  - Low SG Level

Three DAS Inputs to Automatic Reactor and Turbine Trip

- If Chapter 15 event credits a specific reactor trip, an automatic DAS trip would occur on the same trip function
- Refer to the D3 Coping Analysis (MUAP-07014)
  - Discussion of event evaluation methods
  - Results of each event evaluation with CCF