There's some tests that has been done already in VERCORS in France, up to like 41 gigawatt days per ton. It's about three pallets only, and they are also starting a new facility in Cadarache that we don't know much about, but this one will have a longer length rod, about maybe 60 -- six centimeters long.

And we don't know what the test matrix look like in terms of when the MOX test will be coming in because this facility is supposed to replace this town in the near future. So they will shut down all of the hot cells and those type experiments at Grenoble in France, and then, of course, research will assist licensing in terms of review any technical issues that will be rising.

DR. MEYER: Could I add something here?

DR. LEE: Yes.

DR. MEYER: It's Ralph over here.

I didn't seen Cabri on your slide, but there are two MOX tests in the Cabri water loop, and there have been. Did you have that? I'm sorry if you had it on there.

DR. LEE: No, I didn't put it in here. I mentioned it in here that we need.

DR. MEYER: Oh, okay.

DR. LEE: I didn't put that on here.

So our activities just started not too long ago. So we don't have any results to tell you, but on the source term area, next year, this coming fiscal year, we will start to initiate the validation for the codes that are going to be used for source term analysis, and that's the first one we're going to do.

CHAIRMAN POWERS: When is an appropriate time for us to hear about what you're doing with PARCS?

DR. LEE: I think by May time he will be able to do some demonstration on using the type of analysis that he has.

CHAIRMAN POWERS: So maybe some time in the fall?

DR. LEE: Some time in the fall, yes.

CHAIRMAN POWERS: Yeah, I think the Committee would be --

DR. LEE: -- MOX calculation was the difference between UO-2 versus MOX.

CHAIRMAN POWERS: I think it's been a long time since the Committee has looked at some of these neutronic things, and since it's an important part of TRACK M maybe the Fuel Committee and the Thermal Hydraulics Committee might want to get together and

just focus on that, say, for half a day, just that particular topic.

MR. ROSENTHAL: Because we're using this also just plain the UO-2 RIA issues.

CHAIRMAN POWERS: Sure, yeah. I mean, it's a fairly important code.

MR. ROSENTHAL: Sure.

CHAIRMAN POWERS: I like the way you guys went about selecting to use it and whatnot. I thought that was a very analytic process, but when it came in, there was this list of challenges I would say in interfacing and shortcomings that the code had for modern things, and it would be nice to see how it all came out.

DR. MEYER: By the way, we had a small task in our program with Kurchatov Institute with IPSN involvement as well to do some MOX calculations for the reactivity transients.

MR. ROSENTHAL: And that's really good because everything we have traces back to NDEF (phonetic), you know, NDEF E6 or 7, and that's independent.

Can I just make a summary statement? And that is that I'm relatively new in the current branch, just a little bit over a month, and so I go to Richard

and I go to Ralph all the time. In fact, Ralph's office is next to mine.

And we were talking, and I think it's important to make the following point. If I go to the RIA, okay, what we ultimately will discover is that the speed limit that we thought was appropriate for decades is probably incorrect and, you know, maybe 280 becomes 100 or 80, some other number, and at the same time when we do 3D space-time kinetics, we're pretty comfortable that people will be able to demonstrate that they can live with a revised lower speed limit. So you don't have a big safety issue, having done all that work and recognized that.

And I said, "Yeah, but shouldn't this give us some humility?"

(Laughter.)

MR. ROSENTHAL: Okay? That here was something that, you know, we thought of and didn't question, and now we have a different perception.

And if it's giving us some humility with respect to the enthalpy deposition, then it's fair to say, well, what other surprises are there in stock for us as we go to high burnup or new alloys or your MOX, and that sense of, well, what other surprises are in stock for us, and maybe a little humility, leads us

to, in fact, fund fuel work and research as a truly sensible fraction of the total research budget.

I just wanted to leave you with that.

DR. APOSTOLAKIS: Now, the view for McGuire and Catawba, are these considered changes in the licensing basis, the use of MOX?

DR. LEE: Sure, sure. It would have to be.

DR. APOSTOLAKIS: So what if someone --

DR. LEE: Specific licensee.

DR. APOSTOLAKIS: What if someone decided to use regulatory guide 174 to argue for or against?

DR. LEE: I think the same question would arise, that phrase when Margaret was asked about 1.174.

DR. APOSTOLAKIS: The question will arise, but --

DR. LEE: Yes.

DR. APOSTOLAKIS: -- it says here MOX research, and I don't hear you doing anything about it. Why aren't you looking into it?

DR. LEE: I think that is up to the plant, what they want to do it under the regular 1.1 -- 1.7.

CHAIRMAN POWERS: I guess I'm confused, George. I mean, if the program includes an

examination of the source term, and so I'm a little questioned -- I mean, maybe you can say there's some core degradation work that --

DR. APOSTOLAKIS: If Tom is right and the left values are not the right ones, you have to modify them. Shouldn't somebody look into that? Does that come naturally from this?

CHAIRMAN POWERS: Yes. I mean, that would be the whole point. If somebody came back and said, "Look. This" --

DR. APOSTOLAKIS: What does that -- point to me to that.

CHAIRMAN POWERS: If the source term is going to be different from that, then once you had that, that's when you would have to reexamine your derivation to get from the quantitative health objectives to get to the acceptance value of worth.

DR. KRESS: They're putting together a PIRT now just to look at that. You know, they don't define the program yet. They just want to say what are the likely phenomena; what are the issues; what research should we do.

DR. LEE: The source term PIRT is that we're going to look into what are the issues that we have to deal with for NUREG 1465. What do we need to

do for that for MOX.

And then in the model developments, we're going to validate our models. We're going to use -- for example, I'm going to take a core, and I'm going to have an analysis of all uranium fuel assemblies, analyze and look at inventories, and I'm going to take another core which is one third or 40 percent loaded with MOX, and I look at the two source, and I will do my consequent analysis, and I would like to compare what are the consequence, what are the difference from there.

Now my mother has to be validated (phonetic).

DR. APOSTOLAKIS: Now, when you say do your consequence analysis, what do you mean?

DR. KRESS: There's a design basis space he's talking about.

DR. LEE: For the design.

DR. KRESS: Chapter 15.

DR. APOSTOLAKIS: But LERF was not developed.

DR. KRESS: No, no. He'll have to do more than 1465 --

DR. APOSTOLAKIS: Yeah.

DR. KRESS: -- to get to that stage.

They'll have to have more detailed fission product, release models, and --

MS. SHOOP: This is Undine again.

I would just like to add that as part of our user need memo we have requested the Office of Research to look not only into the source term, but how that will impact the different levels of the PRA, and I believe that right now that's being looked into, and I'm sure that when Richard comes back here to talk about our further research in the future after we're done with the source term, then we'll be able to go into more detail on the additional research we're doing.

CHAIRMAN POWERS: Okay.

DR. LEE: Oh, Dana, one thing that I think we should also know, that the French is launching a PHOEBUS 2K (phonetic), which also has a MOX component in it, and they want to look at is there any sudden core degradation phenomenon that we don't know about that is vastly different between UO-2 versus MOX.

And also in the LOCA arena, they are also looking into doing LOCA as a series of looking at the loss of cooling accident for high burnup fuel, but I don't know whether MOX is included in that.

CHAIRMAN POWERS: They're going to have to

jerk their driver core here pretty soon, aren't they?

DR. LEE: Yes.

CHAIRMAN POWERS: Now maybe they're going to run out of oomph in the driver core.

DR. LEE: I think they need to refurbish that entire thing. The driver core is only good for the current series of tests, and after that they completely have to refuel the whole driver core for the following improvement.

CHAIRMAN POWERS: So there will be an examination of the core degradation aspects.

DR. LEE: That's what they would like to do, yes.

CHAIRMAN POWERS: Right. Any other questions of the speaker?

(No response.)

CHAIRMAN POWERS: Okay. We have a treat. Dr. Lyman from the Nuclear Control Institute is here with us again. Dr. Lyman has spoken to us before. He'd like to have a word with us.

He didn't tell me what he was going to talk about, but I'll bet it's on MOX fuel.

DR. LYMAN: Thank you.

CHAIRMAN POWERS: Put it on your tie probably is a better --

DR. LYMAN: How's that?

CHAIRMAN POWERS: Yeah.

DR. LYMAN: Okay. Well, you're right. Since the top was MOX fuel and that's one of the main concerns of my organization, the Nuclear Control Institute, so I thought it might be a good time to come back.

Actually I've never spoken to the ACRS before on MOX. Two years ago I gave a briefing to interested NRC staff on a study I had done, a preliminary study which was actually a consequence assessment, exactly what was just being discussed, of the use of MOX fuel in light water reactors and actually a regulatory guide 1.174 approach to how you might risk inform the use of MOX fuel.

And so I'd like to actually go over those again. I've since refined the report, and it's going to be published. I wish I had a final version. This is a penultimate version, and it should be available very soon in the Journal of Science and Global Security, which comes out of Princeton University, and it will be on their Web site.

So as soon as that's out, I'd be happy to point people to it if they're interested.

Okay. The title of my talk is "MOX Fuel

Safety, a Need for Research," and I'm very glad that there's finally money in the NRC budget for doing some MOX research since there hasn't been for a long time, even though this program has been coming for a while.

My organization has been very concerned about the way the Department of Energy has dealt with the issue of MOX fuel. From the beginning, their environmental analysis, the whole way in which they made decisions regarding weapons plutonium disposition without really looking hard at some of the safety issues that were going to be coming down the pike with MOX.

I wish they'd involved the NRC earlier, and there is still time to deal with these issues, but it's starting to run out.

So just briefly I'd like to give some of the overall, the general concerns I have with the way the MOX program is evolving, including some very recent developments, and then I'd like to talk about some of the detailed safety issues that I think are of concern in this program.

One is the issue of the source term impact on severe accident consequences and risk, and then the impact on transience, including the over cooling accident, pressurized thermal shock, and then RIAs,

and then finally some troubling issues concerning the MOX qualification plan which has been laid out by the licensee, Duke, Cogeme (phonetic) with Stone & Webster, or DCS.

So starting with the MOX program concerns, I think the question came up before why are ice condenser plants the best suited for using MOX fuel, and the answer is they are the only ones that are willing to do it. There was no real choice for the mission reactors. There was no real competitive bid that was worth anything. There were only three consortia that competed. Two of them didn't even meet the basic requirements. So they were eliminated right off the bat, leaving on the Duke Power consortium, which originally had Virginia Power. They dropped out, I believe, because they would have had to modify their control rod systems in North Anna, and they didn't want to do that.

So for better or for worse, we're stuck with the ice condensers, and I'll talk about our concerns about that a little later.

The second great concern we have with the MOX program is the fact that the timetable is dictated by international agreement and not by safety requirements. The U.S. and Russia signed an agreement

last fall or late last summer that commits both sides to starting to use MOX fuel in light water reactors by the end of 2007, and our concern, of course, is because of the political pressure, because this is a nonproliferation program, that NRC is going to have a very hard time raising substantive issues that might cause delays in the schedule, and they run the risk of being accused of being obstructionist and interfering with important nonproliferation programs.

And so I feel this is a potential tension that might influence the ability of NRC to do a fair assessment of MOX safety issues.

Related to this are the DOE budget cuts which are impending. The MOX program apparently, according to news reports, is not going to get the increases that it expected under a potential Gore administration, since it was Gore who was shepherding the bilateral plutonium disposition talks.

And the fact is that a reduction in budget for MOX is only going to increase pressure that any safety review for MOX be abbreviated, and that there will be less DOE resources available for helping NRC to resolve some of these technical issues.

This could lead to heavy reliance on proprietary foreign data, which for many reasons our

organization doesn't think is going to be appropriate or adequate for resolving the issue of MOX use in U.S. reactors.

And finally, the impending cancellation of the other plutonium disposition track, which was a mobilization of plutonium in a ceramic and disposal of high level waste, this program apparently is being zeroed out by the Bush administration, and that means that there will be at least an additional eight and a half tons of plutonium which will be heading toward the MOX program for disposition in roughly the same time period, and it's not clear how DOE is going to address that at that point, but again, it will increase the burden on MOX as the only route for achieving disposition.

So with those political pressure in mind, I'd just like to review some of our concerns about the safety of MOX, and the biggest contributor I think to the enhanced risk of using MOX in light water reactors is the additional source term that comes mainly from an increased transuranic inventory in the core.

Now, according to the calculations that I did using the scale code, you find for the DCS core, which has a 40 percent MOX core fraction and an aqueous processing which will remove the americium

that's been building up in the plutonium pits since they were last recycled; that if you remove the americium, then at end of cycle I find that you'll have about two times more of the isotopes like Plutonium 239, Americium 241, Curium 242.

Plutonium 238 is a little bit less, but that doesn't have a big safety impact, and also, since I know the Committee has been interested in ruthenium lately, incidentally, for a given MOX assembly you have more than twice the amount of Ruthenium 106. So an average of the core and into cycle, I find you have about 45 percent more Ruthenium 106, which might play a role in events where there's the risk of air oxidation source term, as the Committee has discussed, a PTS event, or a spent fuel pool accident.

Finally, after I first put out my study in spring of '99, DOE revised its EIS calculations, accordingly did a better job, but there are still flaws in the values that are outstanding in the environmental impact statement, and one of those comes from the fact that they assumed for some reason that in the reactors in the U.S. you have three or you divide the core into three equal fractions, and each burnup interval is an equal burnup interval, which is not the case in a reactor with an 18 month core

loading like Catawba or McGuire.

So they actually underestimate the burnup of the second cycle MOX fuel.

So what are some of the impacts on severe accident consequences from the increased true source term using the MAX-2 code, suitably revised after I discover an error in it?

You find that for early containment failure, for a typical early containment failure source term, which in this case what I have here corresponds to about a one percent overall low volatile release; you find that there's a 25 percent increase in latent cancer fatalities as a result of the initial plume. That's not looking at the chronic, long term consequences, but only what's in MAX-2, in what's called the early module, and that's because I don't really trust the chronic module in MAX.

As far as prompt fatalities go, there's a very small or practically no increase, only about four percent for early containment failure because the particular isotopes that are greater in MOX cores don't really influence that much. Again, the results will be available in this paper.

Now, I just looked recently at the possibility of the high ruthenium release that might

correspond to a pressurized thermal shock accident, and I found that that has a bigger impact on the prompt fatalities. In that case, this is preliminary, but there's about a 30 percent increase then in both latent cancers and prompt fatalities for a 75 percent ruthenium release.

DR. KRESS: What was the nature of the error you found in MAX?

DR. LYMAN: It turns out for very high releases, you could have more cancer fatalities than there were people.

DR. KRESS: Oh, okay. It was in the dose consequence.

DR. LYMAN: Right. It was not normalized properly, and so they fixed that, and it will be in the next release.

DR. APOSTOLAKIS: Now, when you're saying 25 percent, four percent, and so on, you're obviously referring to some point value.

DR. LYMAN: Oh, I'm sorry.

DR. APOSTOLAKIS: Is that the mean value of something or best estimate?

DR. LYMAN: You mean --

DR. APOSTOLAKIS: What does the 25 percent refer to?

DR. LYMAN: Oh, I'm sorry. Compared to the exact same source term applied to an only uranium fuel. So in other words, I --

DR. APOSTOLAKIS: So you did both calculations?

DR. LYMAN: Right. You look at the consequence analysis for a particular source term for a uranium fuel, and then you keep the release fractions all the same, which may not be a correct assumption for MOX because there may be greater volatile releases for MOX fuel, but if you assume all of the source term, the release fraction is the same. Then you just look at the impact of the additional actenites (phonetic), for example.

DR. APOSTOLAKIS: Okay.

DR. LYMAN: But I did it over the entire spectrum of isotopes.

And again, of course, there are different release fractions for different accidents. That's a kind of stylized early containment failure, which was derived from NUREG 1150.

Okay. So what about the impact on risk? Well, you can look at a set of a kind of complete set of accidents leading to a large early release, and basing on a NUREG report, which binned a whole lot of

severe accident scenarios into a small number. I was able to do a rough estimate of what is the impact on the average population risk within one mile, which is the parameter cited in the quantitative health objectives.

And so that actually tracks the consequences pretty well, about 25 percent increase for the DCS core in average risk to the public within a mile of the reactor. That's latent cancer fatality risk.

So then I asked if you wanted to risk inform, sine it's quite likely that when there's a submittal for a license amendment for using MOX fuel, then it will meet all of the design basis requirements, but the question is: will it have an impact on risk, which could be something you need to consider?

And now that the staff has the authority to use risk information either in a license submittal that's not risk informed, I thought this might be something that the staff might want to look at since this could be one of the biggest impacts. The biggest impacts of using MOX is not on design basis actions, but on beyond design basis.

But then this question arises, which the

Committee has discussed frequently, is the 1.174 assumes a particular release, and only looks at change in LERF, and so the question is: how do you deal with the situation where the actual frequencies may remain roughly the same, but the inventory changes?

So I did a quick and dirty -- I'm a former physicist. So that's what we do, is try to work with what you've got, and quick and dirty way of using 1.174 was simply to derive what I call an effective LERF, which is let's say you have an accident, two different accidents and only the consequences change. That's associated with a change in risk.

So what's the equivalent change in LERF that would lead to the same change in risk? And so it's just a way of using the scale which is provided in 1.174.

And incidentally, this is also a useful way for evaluating what's an extended power up rate, and the issue does arise if you have the 17 percent extended power up rate. That's going to lead to a significant increase in consequences from severe accident, and if that's acceptable, then this increase associated with MOX will also be.

But inversely, if one isn't, then neither will be the other. So this could be a way of

addressing at least until the formalism is fixed, to address this, a way of addressing things like the risk impact of an extended power up rate.

DR. APOSTOLAKIS: So fixing it probably will mean not to deal with a LERF anymore.

DR. LYMAN: Possibly. I mean --

DR. KRESS: If you had delta R you wouldn't need a LERF really.

DR. LYMAN: Right, and that's what this is just saying. Delta R is the same for both.

DR. APOSTOLAKIS: Because neither the large or the early change, as you said.

DR. LYMAN: Right.

DR. APOSTOLAKIS: Nor the F.

DR. LYMAN: But if this equation isn't right, and it may not be because, you know, you end up with small fractional increases in risk, and you know, the error bars might be big enough that it washes those out, but if that's the case, then if this isn't correct, then the overall 1.174 --

DR. APOSTOLAKIS: So R is the risk.

DR. LYMAN: Right. In other words, probability times consequences summed over all the accidents that contribute to LERF.

DR. APOSTOLAKIS: For whatever risk you

have in mind. I mean prompt fatalities.

DR. LYMAN: Right. In this case I looked at latent cancer.

DR. APOSTOLAKIS: So you do have delta R then.

DR. LYMAN: Right. You can calculate it if you know everything.

DR. APOSTOLAKIS: If you had it or you have it.

DR. KRESS: You have to do some sort of a PRA. Now, he --

DR. APOSTOLAKIS: But look. Lyman says that we should use this to define an effective delta LERF. Therefore, you must have delta R.

DR. KRESS: But he used sort of --

DR. LYMAN: Right.

DR. KRESS: -- an abbreviated --

DR. APOSTOLAKIS: And he did that earlier.

DR. LYMAN: And it's like a Level 3 PRA, except it's very truncated, and it was based on a small set of accidents.

There was a study. I don't have the number with me, but they took, let's say, the Sequoyah NUREG 1150, and they binned. You know, you have thousands of different initiators. They binned them

into a small number of accidents with the same source terms.

So it was manageable. There were three or four different source terms and frequencies associated with that. So you could do a kind of very rough Level 3 and get the risk.

DR. APOSTOLAKIS: Now, instead of doing this, it seems to me since you can do a rough Level 3, what you could do is take the allowed delta F for light water reactors that the NRC staff --

DR. LYMAN: Right.

DR. APOSTOLAKIS: -- has declared is acceptable --

DR. LYMAN: Right.

DR. APOSTOLAKIS: -- ten to the minus seven --

DR. LYMAN: Right.

DR. APOSTOLAKIS: -- and see what the consequences of that are with respect to the acceptable change in prompt fatalities and compare your delta R with that.

DR. LYMAN: That's actually exactly the same thing.

DR. APOSTOLAKIS: It's the same thing?

DR. LYMAN: You're just saying it

differently, yeah.

DR. APOSTOLAKIS: It's not obvious it's the same thing. Is it obvious it's the same thing? I'm not doubting, but --

DR. LYMAN: Well, I have to think about it. I think it's the same.

DR. APOSTOLAKIS: I can't see it's the same.

DR. LYMAN: Because you're just saying what -- you could rewrite this in that way.

DR. APOSTOLAKIS: In other words, what I'm saying is, okay, you can calculate the change in prompt fatalities or cancers and so on, but you don't know what's acceptable, what delta cancers is acceptable, but you have a delta LERF that has been declared acceptable for light water reactors.

Take that and propagate it to the front, the Level 3, and then compare you delta after that.

DR. LYMAN: Yeah. Do you see where it's the same thing? Because you're just saying if you know what the source term is, then you can say, well, I know what the change in risk is going to be associated with that change in LERF. Now, if you can do the Level 3, then you can propagate that through, and then you would get a delta R, which you would

compare.

This is just doing that backward.

DR. APOSTOLAKIS: And so I guess what you're saying is after I take the LERF to the left, I have delta LERF or LERF is delta R over R.

DR. LYMAN: Yes.

DR. APOSTOLAKIS: And there must be some other duplicative factor there that counts as R.

DR. LYMAN: Right, if the source term is the same. Right. It's the same thing.

DR. APOSTOLAKIS: Yeah, yeah.

DR. LYMAN: You know, it's a very obvious, very simplistic --

DR. APOSTOLAKIS: I don't know about obvious. It took me ten minutes to understand.

DR. UHRIG: Are you contemplating a 17 percent increase in power?

DR. LYMAN: No.

DR. UHRIG: I'm not aware of that.

DR. LYMAN: No. What I'm saying is that since the risk that I found associated with using MOX is about, you know, this 25 percent increase. That could be comparable to the increase in risk associated with the power up rate.

DR. UHRIG: Well, the power up rate, the

17 percent typically associated with BWR is not PWR.

DR. LYMAN: No, I'm not saying that it's going to happen. I know Catawba and McGuire are not planning to. I'm just saying that's another example where you could use this.

And, again, if those up rates are approved, then, well, at least it's a way of saying it. It's a way of saying -- well, let me go on to the next slide because at least this shows you in the 1.174 context.

Okay. So what's the risk impact of MOX in ice condenser plants? Now, we know the DCH study that came out last year concluded that ice condensers are substantially more sensitive to early containment failure than other PWRs, and this is precisely the class of accidents in which you would feel the additional risk from MOX because these are the accidents where you would have fuel dispersal and containment failure.

So that in itself is of concern, but here I just did -- this is a rough estimate using the equation from the previous slide where from the McGuire IPE, which is now ten years old, but the total LERF, internal plus external, is 4.7 times ten to the minus six.

So then if you use the delta LERF effective equation from the last page, you get a number 1.2 times ten to the minus six that actually exceeds the reg. guide 1.174 threshold. At least this crude estimate means that it's in the regime where changes would not normally be allowed. So that's the first point.

CHAIRMAN POWERS: Actually, I think it's in the regime. It simply means it's in the regime where it gets increased management attention.

DR. APOSTOLAKIS: That's pointed out here.

DR. LYMAN: Well, the actual language is not normally allowed. It's the top tier. Now, it's close to the boundary, and nothing is set in stone, and you also have permission to use other arguments, you know, quantitative arguments to get out of this hole, but I would say that at least on the scale that's proposed in 1.174, this increase associated with MOX is fairly significant, and I wouldn't write it off.

Now, going back to that, the McGuire IPE does not take into account the Sandia finding that the early containment failure frequency was under estimated by a factor of seven in Duke Power's own IPE and PRA, and this, as Richard Lee said, is still a

matter of controversy.

But if you did take into account the greater early containment failure frequency associated with station blackouts, just again using the IPE numbers, you'd end up with a LERF above ten to the minus five, which is in the regime where no risk increase greater than ten to the minus seventh would be allowed. So that, again, would exclude MOX.

Now, I know that the current PRA for McGuire is about half what it was in the IPE, but I don't know what the station blackout frequency is now, and these are not really publicly available, and so I can't say anything about that. But at least based on what's public, I'd say, again, that the risk is significant.

And, again, the implications for extended power upgrades, I'd say, is one way of looking at if a 25 or 30 percent increase in risk associated with an extended power upgrade, this is a way of evaluating where it fits in the risk informed framework.

And speeding up, now the MOX impact on transience. This is all pretty well known, but I'd just like to point out a few other things.

The PTS screening criteria which are now under review for all plants may not be appropriate for

MOX cores, in other words, the ones that are appropriate for the LEU may not be appropriate for MOX, and one reason is the reduced decay heat immediately after a SCRAM in a MOX core would lead to a more rapid decline in the temperature in the reactor coolant system, and therefore, a more rapid entering into a region, a temperature region where the pressure vessel might be threatened.

Another aspect, well, again, if you have an air oxidation source term with greater fuel and ruthenium releases, then the source term might be more severe for a MOX core in a PTS event, and a final point is that because of the greater fast flux, the embrittlement is going to be somewhat more rapid, and this is not something that Duke Power is planning to take into account at its license renewal time limited aging assessments.

As a matter of fact, Duke made the alarming statement that, well, license renewal comes first, and then they'll evaluate MOX, and if there was a risk that using MOX would impair the ability of their plant to operate safely to the end of the license renewal period, then they won't do MOX.

And when I heard that, I wondered if the Department of Energy knew that was their position, but

considering there's only a two-year, I think, difference between when they're doing their license renewal and when they'd have to do the MOX assessment, it would make sense to do it all at once in my view.

Moving right on in the reactivity insertion, we all know the increased vulnerability of MOX to RIAs or potential increased vulnerability as demonstrated in the REP Na-7 Cabri test is a concern. And a key consideration is the fuel homogeneity and the size distribution of the plutonium agglomerates.

And, you know, this has been known, I think, for decades, and Westinghouse in its consultant's report to DOE recommended -- this is a quote -- "adherence to limits on plutonium agglomerates in the range of 10 to 15 microns should be required."

And in that context, it's pretty alarming to learn that DCS appears to actually be relaxing the existing specification that's in use at the Maalox (phonetic) plant in France, when they should be going in the other direction.

And the Cogema MIMAS plutonium particle distribution that's currently achieved has a mean size of the distribution of the agglomerates of 20 to 40 microns, and the specification is no more than two

percent of the clusters should be greater than 100 microns in size, and the maximum that occurs is around 140, I believe, while the DCS specification, at least in the version of the fuel qualification plan, which we submitted last year, and I understand there's a new version now; so this may have changed, but they specify a mean size of less than 50 microns and a maximum five percent of clusters greater than 100 microns with a maximum size of 400 microns.

So instead of trying to bring this number down to the ten or 15 range that Westinghouse suggested, they seemed to be going in the other direction. I think if this is actually the case that it's something that they need to be called to account for.

Now, on the issue of MOX --

CHAIRMAN POWERS: A 400 micron inclusion, a 400 micron plutonium inclusion would be a fairly significant inclusion, wouldn't it?

DR. LYMAN: Yeah. I mean, it's about the maximum. It was the maximum that was set back when they did those experiments in the '70s or '60s, and hopefully technology has improved since they were making this.

CHAIRMAN POWERS: I'm just trying to

understand what the neutronic effects of a 400 micron -- I mean that's a pretty healthy inclusion, isn't it?

DR. KRESS: It's pretty good.

CHAIRMAN POWERS: I think you would worry about that.

DR. KRESS: I think you'd see it.

CHAIRMAN POWERS: Yeah, I think you would see something.

DR. LYMAN: Well, it's right in the fuel qualification plan if you want to take a look at that number.

Now, generally speaking, we have a lot of concerns about the way the fuel qualification is coming about. First of all, the schedule, I think, is pretty aggressive. They hope to load the LTAs and start irradiating them in McGuire in October 2003.

Then they're going to do it for two 18-month cycles, and so discharge would be around October 2006, and these twice burn LTAs, then they would be subject to some nondestructive analysis, but the first reload batch would be a year later.

So that only gives one year really for doing all of the work that both the licensee and NRC might want to do on these LTAs.

The other aspect is at least according to

the first version of the fuel qualification plan, they wouldn't even be burned up to the maximum discharge burnup that they're proposing for the fuel, but would fall short, and that's another puzzling aspect.

Then the issue of where the LTAs are going to be made is still not determined. As you know, Los Alamos has its contract canceled last year, leaving the program stranded. So the two bad alternatives now are, one, the LTA is manufactured in a European facility, but this raises the issue that they may not be representative if it eventually comes out of a U.S. plant, especially if the fuel qualification parameters are different.

CHAIRMAN POWERS: When you say they're not representative, are you speaking of the fact that they did not have weapons grade plutonium in them or --

DR. LYMAN: Well, no. It wouldn't make sense if they didn't, but where, you know, there had been talk that it might come from England, you know, I don't know the details, but it certainly wouldn't be U.S. weapons grade plutonium that was aqueously purified according to the plan that we have and fabricated according to the specifications that DCS is establishing.

So that has to be looked at. It may not

be that significant an issue, but again, given what we've heard today about the variability and, you know, expectations for fuel, small changes in composition, manufacturing parameters, there seems to be some sensitivity to these things.

And so I would be more confident if the LTAs actually were a product of the plant that's going to be manufacturing them, but the problem with that, which is the other option, is that clearly it's going to cause a delay if the U.S. MOX plan is going to be the source of the LTAs because who knows? They'd have to establish some sort of a pilot line, I guess, and who knows if the fuel coming out of the pilot -- I mean, the first fuel -- is going to be suitable or representative of a later fuel?

So I think there are a lot of issues that are not being dealt with adequately here, and because of this aggressive timetable, NRC's ability to resolve MOX fuel safety issues, I think, is in jeopardy. Again, the time for post irradiation LTA characterization testing is inadequate, forcing a reliance on proprietary find data, which NRC is not going to be able to confirm, and I think the M5 experience, however it plays out, should give pause in this area because whether or not the M5 cladding,

which incidentally is the cladding that's going to be used on MOX fuel, and Framatome is the fuel designer and supplier for the MOX program here; whether or not it turns out to be adequate and meets the existing criteria, I'd have to say that the behavior of Framatome since they were aware that they were doing ring compression tests; they were aware that there was an issue; they were aware of the results. The Germans were making them do these tests.

At the same time NRC was reviewing and approving the M5 cladding without knowing any of this, and the fact is, you know, they didn't ask the questions. So maybe they didn't have to get an answer, but I think if Framatome was completely forthcoming, they would have notified them.

And so I think it raises questions about how reliant we should be on foreign data that's not confirmed independently.

And in this regard, it's especially frustrating that DOE appears to be uncooperative with NRC's Office of Research, and you may not be aware that the Office of Research sent a letter to DOE in December requesting that access be granted to NRC to have some samples of the irradiated lead test assemblies taken to Argonne for NRC's confirmatory

testing.

DOE's answer was basically, "No, thanks. It's duplicative, and you'd have to work that out with the licensee anyway," wouldn't have anything to do with it. It was an evasion.

And this is an example of how I think things are going to play out especially in the context of the budget cuts that we're going to see. DOE is not willing to pay or support any of what it considers additional research, and I think that's a mistake.

I think that both the timetable and the staff resources for MOX safety issue resolution should be based on NRC needs and not DOE needs. You know, in an ideal world, NRC should design the research program it thinks is necessary to answer the questions, give DOE the bill.

(Laughter.)

DR. LYMAN: And then --

PARTICIPANT: In an ideal world.

DR. LYMAN: Right. Well, I'm an optimist.

Cancellation of the immobilization track is going to increase pressure on NRC not to be obstructionist in MOX licensing, and I think this path for MOX approval is not likely to engender public confidence the way things are going.

So I would like to see a tightening up of the goals and the objectives and a good research program addressing some of these concerns.

Thank you.

CHAIRMAN POWERS: Any questions of the speaker?

That was a great presentation. I think we appreciate it when you take the time to come talk to us.

DR. LYMAN: Oh, I appreciate the opportunity.

CHAIRMAN POWERS: Thank you.

DR. SHACK: Let me. What is your argument again about why this is appropriate for the power up rates? You're not arguing that the source terms is increased in the same way. Are you just saying that you should consider the change in source term and use it to modify the LERF?

DR. LYMAN: Well, the source term is increased not in the same way, but some of the --

DR. SHACK: Okay, but your argument is you should consider that change in the source term and modify the acceptance on the LERF. That's what you mean.

DR. LYMAN: Right.

DR. CRONENBERG: The scale and not the source term.

DR. LYMAN: Right. I mean, this is actually discussed here last year where there was some argument how do you risk inform this if you don't have a tool that takes into account change in source term, and I'm saying this is one way to do that.

DR. CRONENBERG: When did the mobilization -- was that really canceled?

DR. LYMAN: Well, they suspended the contract. They had had a request for proposals put out for a mobilization contractor. That's going to be suspended. That money was zeroed out for the coming fiscal year.

They don't say it's been canceled, but everyone I know or what I've heard from people inside the program is it's dead. People have been reassigned. The work is over.

Thanks.

CHAIRMAN POWERS: Ralph, we have some time scheduled for the full Committee tomorrow on this general area of high burnup and MOX fuel. I'll be frank. I did not see anything that I felt a burning need to bring before the full Committee. Is that true or do you have a different perception?

DR. MEYER: No, I think that's okay. I was just wondering what you expected the staff to prepare for tomorrow.

CHAIRMAN POWERS: Well, what I was going to suggest is, I mean, you've basically given us an update on where you stand, that you've gone through your PIRTs. I think that's great.

I was just going to suggest that I'd give a quick summary to the ACRS and let it go at that.

DR. MEYER: Okay.

CHAIRMAN POWERS: I mean, there's nothing for us to write a letter about. So I hope you're not expecting a letter from us.

DR. MEYER: Right, right.

CHAIRMAN POWERS: We need to produce a letter that says to close out one of the GSIs on this high burnup fuel.

DR. MEYER: Yes.

CHAIRMAN POWERS: Okay, and basically what we need to be able to say is everything that's listed in that GSI is being addressed in the research program, and I think we're on safe grounds in saying that.

DR. MEYER: That's correct.

CHAIRMAN POWERS: Okay. So it seems to me

that the only thing we need to do is why don't I just give a summary of what went on at this meeting? You guys can go do your work and actually make some progress.

DR. MEYER: Okay.

CHAIRMAN POWERS: And that's not put -- I mean, I just don't see a need to have a -- I'm sure the Committee members would be very interested in what's going on, but that's all it would be, would just be technical interest and whatnot, and that's the job of the subcommittee. We get the fun job.

DR. MEYER: Okay.

CHAIRMAN POWERS: They've got to work hard.

DR. MEYER: That sounds fine to me. So I don't have to prepare a presentation tomorrow.

CHAIRMAN POWERS: I don't think you need to prepare a thing.

Richard, similar I think on the MOX. You're just getting started. I don't see anything. I think between Med and I we can take your viewgraphs, put together a viewgraph that says, "Here's what we talked about, and our intention is to come back and look again roughly in the fall."

Because that looks like when things were

coming down both from Margaret's perspective and from your perspective; is that right, Ralph?

DR. MEYER: Okay.

CHAIRMAN POWERS: I mean that's all I see to do. I think it was a great update, but I just don't see anything that the Committee needs to act upon, except we need to get that GSI out.

DR. MEYER: Yeah.

CHAIRMAN POWERS: But I think that's --

DR. MEYER: That's a separate.

CHAIRMAN POWERS: It's a separate issue for us, but I think it's -- I mean, I think what we needed from you is the assurance that the research program is covering it.

DR. MEYER: The assurance that?

CHAIRMAN POWERS: The research program --

DR. MEYER: Yes.

CHAIRMAN POWERS: -- is taking into account everything that --

DR. MEYER: It does. It does cover everything that was said.

CHAIRMAN POWERS: And I think that was all that was needed.

DR. MEYER: Yeah.

CHAIRMAN POWERS: Okay.

DR. MEYER: Okay. Great.

CHAIRMAN POWERS: Any other comments people would like to make?

(No response.)

CHAIRMAN POWERS: In that case, I will adjourn this meeting of the Subcommittee with thanks to the speakers. All very interesting, and at the same time somewhat confusing in that there obviously is at least one variable that I don't understand in clad behavior.

(Whereupon, at 3:17 p.m., the Subcommittee meeting was adjourned.)

CERTIFICATE

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

Name of Proceeding: ACRS REACTOR FUELS

Docket Number: (NOT APPLICABLE)

Location: ROCKVILLE, MARYLAND

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

Rebecca Davis

Rebecca Davis
Official Reporter
Neal R. Gross & Co., Inc.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
REACTOR FUELS SUBCOMMITTEE
APRIL 4, 2001
ROCKVILLE, MARYLAND

PROPOSED AGENDA

TOPIC	PRESENTER	TIME
I. Introductory Remarks (ACRS)	D. Powers	8:30- 8:35 a.m
II. Research Activities on high Burnup PIRT	R. Meyer	8:35- 9:45 a.m
III. Framatome Testing and Assessment of LOCA ductility Of M5 cladding	TBD	9:45- 10:30 a.m
*****BREAK****		10:30- 10:45 a.m
IV. Westinghouse Testing and Assessment of LOCA ductility Of ZIRLO cladding	TBD	10:45- 11:30 a.m
V. Summary of OECD Topical Meeting on LOCA fuel safety criteria	R. Meyer	11:30- 12:00 Noon
******LUNCH*****		12:00- 1:00 p.m
VI. Recent Operational Issues and Experience with high burnup fuel	M. Chatterton (NRR)	1:00- 1:30 p.m
VII. Research activities on MOX fuel	R. Lee	1:30- 2:00 p.m
VIII. General Discussion and adjournment		2:00- 2:10 p.m

NOTE: **Presentation time should not exceed 50% of the total time allocated for a specific item. Number of copies of presentation materials to be provided to the ACRS is 35.**

ACRS Contact: **Med El-Zeftawy (301) 415-6889**
E-mail: mme@nrc.gov

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON REACTOR FUELS

APRIL 4, 2001
Date

ATTENDEES - PLEASE SIGN IN BELOW

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Ryan T. Coles	U.S. General Accounting Office
Mitch Nissley	Westinghouse
WILLIAM LEECH	WESTINGHOUSE
DAVID DIAMOND	BNL
Rich Janati	PA DEP/BRP
Michael Aldrich	Framatome ANP
GARRY GARNER	FRAMATOME ANP
Larry Ott	ORNL
Edwin Lyman	Nuclear Control Inst.
Edward Burns	Washington Group International
Skip Copp	Duke Energy

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON REACTOR FUELS

APRIL 4, 2001
Date

NRC STAFF SIGN IN FOR ACRS MEETING

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NAME	NRC ORGANIZATION
Sudhamay Basu	RES
RALPH CARUSO	NRR/SRXB
Steve LaVIE (La Vie)	NRR/SPSB
Muffet Chatterton	NRR/SRXB
Bob Martin	NRR/DLPM
Richard Lee	RES
Harold Scott	RES
Jack Rosenthal	RES/SMSAB
UNDINE SHOOP	NRR/SRXB

RECENT OPERATIONAL ISSUES AND EXPERIENCE WITH HIGH BURNUP FUEL

Margaret S. Chatterton
ACRS Reactor Fuels Subcommittee Meeting
April 4, 2001

Outline of Presentation

- Burnup Extension Activities
- Lead Test Assembly Guidelines
- Recent Fuel Issues
- Current Fuel reviews

Basic Approach for Burnup Extension

- NRC Working with Industry to Develop a Strategy and a Plan
- Industry is Developing the Plan and Guidelines
- Industry Will Do the Testing and Develop the Criteria
- Objective is to Endorse Industry Approach in a Regulatory Guide

Burnup Extension Guidelines

- Address Current Licensing Requirements including LOCA, ATWS and RIA
- Be Risk Informed
- Emphasis on Lead Test Assemblies
- Fuel Performance Monitoring Program

Burnup Extension Activities

- Draft Submittal from NEI - March 2000
- NRC Staff Provided Comments
- NEI/NRC Meeting - December 6, 2000
  - Outlined Approach on RIA
  - NRC Staff Comments - Approach looked reasonable
- Expect Submittal Late Summer 2001

Lead Test Assemblies

- Prototypical
  - Up to Proposed Burnup
  - Power History
  - Type of cladding
  - Flow Conditions
  - Water Chemistry
- Characterized before Irradiation
- Poolside and Hot Cell Examinations
  - After Each Cycle
  - After Final Burnup

Lead Test Assembly Guidelines

- Purpose
  - Consistent Approach
  - Consistent Data Collection
  - Obtain Data
- Outline of Topics
- Progress
  - Meeting with WOG - May 2000
  - Topical Submitted by WOG and NEI - Oct/Nov 2000
  - WOG/NEI/NRC Meeting - December 6, 2000
  - Staff Comments and Acceptance of Topical for Review - January 25, 2001

Lead Test Assembly Guidelines

- Definition
- Characterization
  - Poolside Examination Data
  - Hotcell Examination Data
- Number
- Placement
- Safety Requirements
- Reporting

Recent Fuel Issues

- Oxidation Higher than Predicted
- Axial Offset Anomalies
- Fuel Failures Due to High Fuel Duty
- Adverse Effects of Water Chemistry
- High Crud Buildup
- Accelerated Growth of Rods and Assemblies

Current Fuel Reviews

- Duplex Cladding
  - Extensive Testing and Use in Europe
  - Beginning Review
- Zirlo for CE Plants
  - Reason for Request
  - Timetable for Review
  - Issues to be Examined

March 22, 2001
Slide #1

Ductility Testing of Zircaloy-4 and ZIRLO™ Cladding After High Temperature Oxidation in Steam

Westinghouse Electric Company

Advisory Committee on Reactor Safeguards
Reactor Fuels Subcommittee

April 4, 2001
Rockville, MD

Zircaloy Ductility after High Temperature Oxidation in Steam

- Ductility measurements on Zircaloy oxidized in high temperature steam were used to establish cladding embrittlement criteria of 10 CFR 50.46
  - Peak Cladding Temperature no greater than 2200F
  - Equivalent Cladding Reacted (ECR) no greater than 17%
- Testing consisted of quench tests from temperature and ring compression tests
  - Ring Compression Tests conducted on Zircaloy-4
  - Quench Tests of Zircaloy-2 and Zircaloy-4
- The purpose of the criteria is to ensure the cladding would remain sufficiently intact to assure an easily coolable geometry

BNFL

March 22, 2001
Slide #3

Information Supplied for ZIRLO™ Licensing

- Testing of ZIRLO™ was performed to obtain data on the following areas
  - Material mechanical properties, density, thermal expansion, thermal conductivity, specific heat, phase changes,high temperature creep, high temperature oxidation, and rod burst characteristics.
- Other than phase change characteristics, the properties are essentially equivalent to those of Zircaloy-4
- It was argued that because of the close similarity to Zircaloy-4, the17% ECR criterion continued to apply
- The NRC agreed that the 17% criterion for Zircaloy also applied to ZIRLO™ and 10 CFR 50.46 was amended to state that the acceptance criteria applied to ZIRLO™

BNFL

Results of Tests on Alloy E110 Oxidized in High Temperature Steam (Bohmert, Kerntechnik 57)

- ECR to cause complete embrittlement is about 1/3 the value for Zircaloy-4
- A number of physical differences between the oxide layers of E110 and Zircaloy-4 were observed
  - E110 displayed a heterogeneous appearance of the oxide scale
  - E110 formed two oxide layers that were frequently separated by cracks
  - Multi-layer oxide layers tend to flake
  - Zircaloy-4 always had a glossy black firmly adherent single layer relatively free from mechanical failures
  - E110 showed low hydrogen uptake only if firmly adherent crackless oxide layers were formed
- High temperature steam oxidation tests of ZIRLO™ and Zircaloy-4 produce similar dark adherent oxide layers

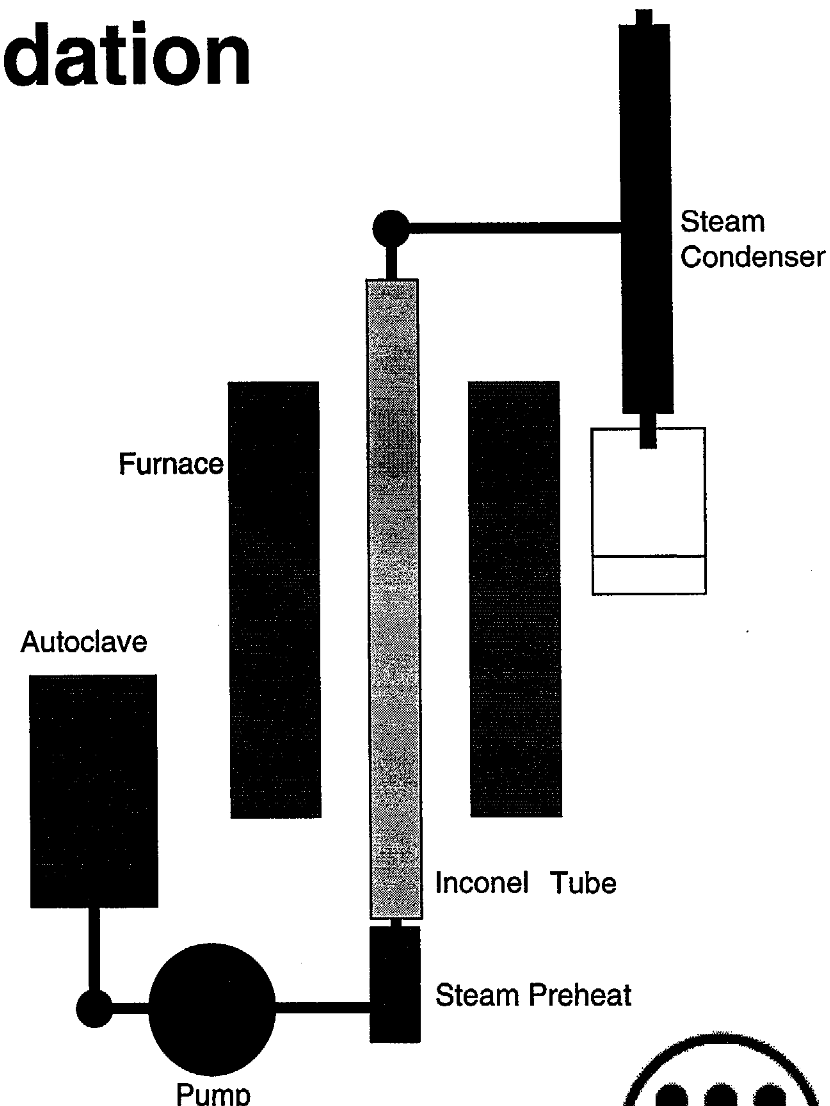
BNFL

ZIRLO™ and E110 Are Not Equivalent

- Both alloys contain 1% niobium
- ZIRLO™ also contains
  - Sn
  - O
  - Fe
- Sn and oxygen are alpha phase stabilizers and raise the transition temperature relative to Zr-Nb binary alloys.
- There are significant differences in the oxide layer structure reported for the E110 alloy and those observed for either ZIRLO™ or Zircaloy-4

Steam Oxidation

- Clam shell resistance furnace.
- Specimens placed in Inconel tube.
- Deaerated water from autoclave pumped into Inconel tube.
- Exit steam condensed by water cooling jacket.



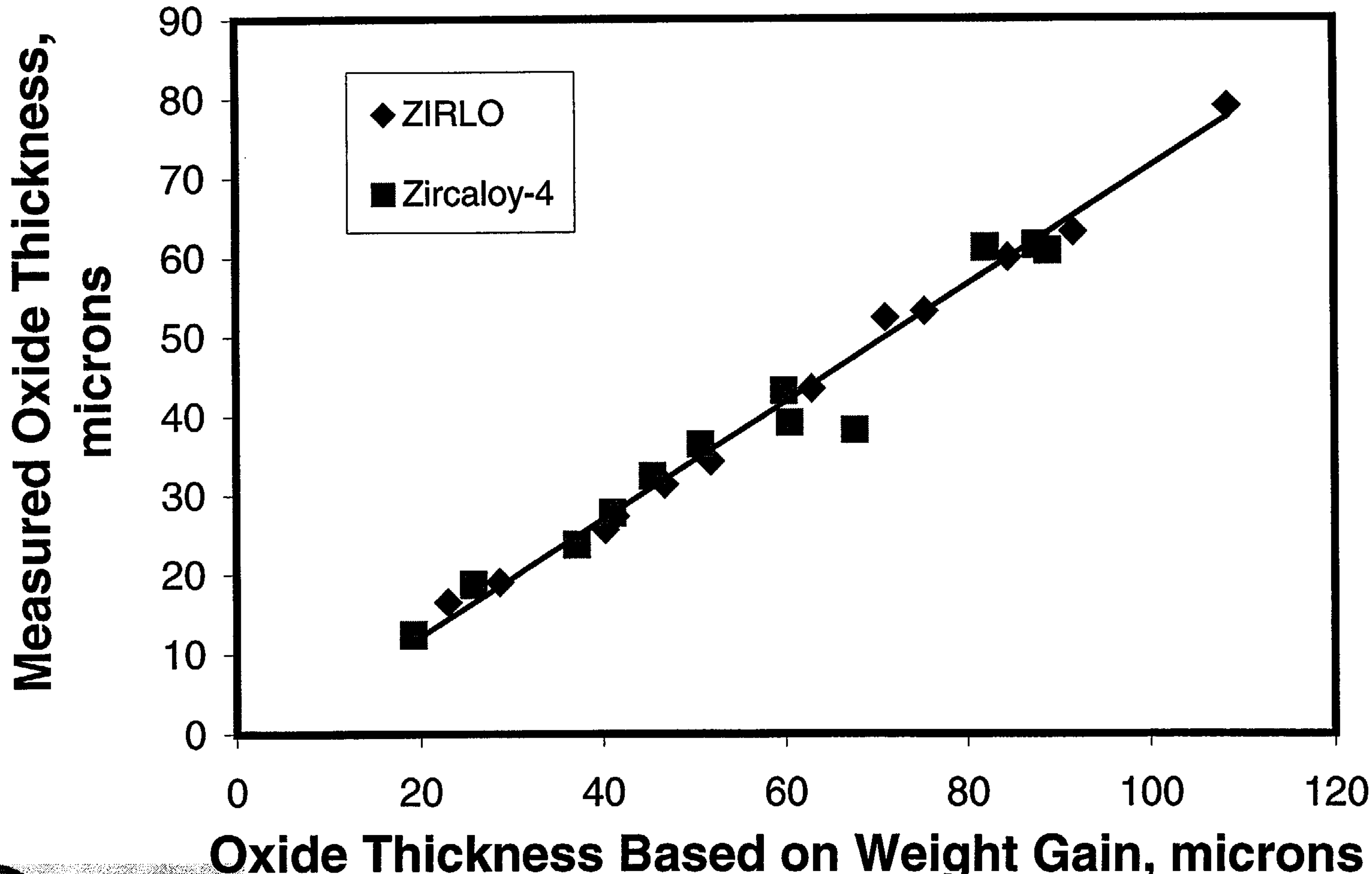
Specimen Evaluations (In Progress)

- Specimen Evaluations
  - Oxide Layer Characteristics
  - Ring compression tests
    - Assess cladding ductility.
    - Room temperature and 275°F.
    - Test performed similar to Hobson & Rittenhouse (ORNL Report 4758) and Böhmert.
  - Optical metallography
    - Oxide thickness, α-stabilized layer, transformed-β layer.
    - Microhardness to assess oxygen penetration.
  - Hydrogen and oxygen concentrations

BNFL

March 22, 2001
Slide #8

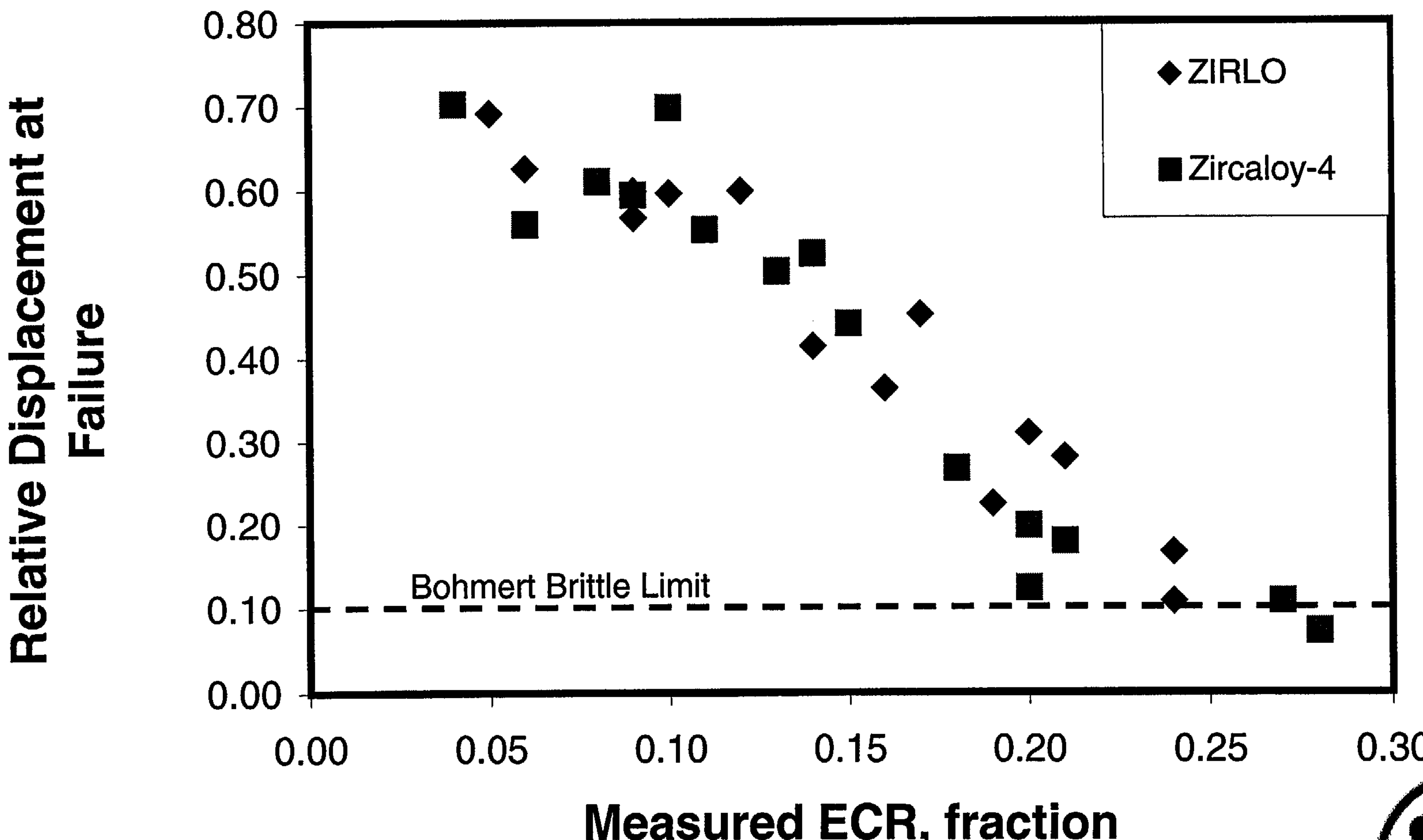
Measured Oxide Thickness vs. Oxide Thickness Based on Weight Gain



BNFL

March 22, 2001
Slide #9

Relative Displacement at Failure vs Measured ECR at a Temperature of 275F (PRELIMINARY)



Comparisons of ZIRLO™ and Zircaloy-4

- Both oxide layers were dark, adherent, and with no laminations
- Both have similar fractions of oxygen in the oxide layer and in the metal
- Ring compression tests show similar values of displacement at failure versus the measured Equivalent Cladding Reacted
- ZIRLO™ and Zircaloy-4 exhibit similar behavior

BNFL

Plan for Project Completion

- Perform remaining sample preparation
- Complete all planned tests
- Document and Review the results
- Meet with the NRC to discuss the results (May)

MOX FUEL SAFETY: A NEED FOR RESEARCH

Edwin S. Lyman, PhD
Nuclear Control Institute

Presentation to the ACRS Reactor Fuels Subcommittee

April 4, 2001

OVERVIEW

- MOX Program Concerns
- MOX Source Term Impact on Severe Accident Consequences and Risk
- MOX Impact on Transients
  - Overcooling and PTS
  - Reactivity Insertion Accidents
- MOX Fuel Qualification Issues

MOX PROGRAM CONCERNS

- No real choice of mission reactors
- Timetable dictated by international agreement and not by safety requirements
- DOE budget cuts will increase pressure for abbreviated MOX safety review
- Heavy reliance on proprietary foreign data
- Cancellation of immobilization track will increase burden on MOX program

MOX SOURCE TERM FOR SEVERE ACCIDENTS

- Compared to LEU core, DCS MOX core (40% MOX core fraction, Am removal) at EOC contains approximately
  - 2 times more Pu-239, Am-241, Cm-242
  - 10% less Pu-238
  - 45% more Ru-106 (important for PTS events, spent fuel pool accidents?)
- DOE EIS inventory calculations flawed

MOX IMPACT ON SEVERE ACCIDENT CONSEQUENCES

- Increased TRU source term in DCS core is important for severe accidents with early containment failure (ECF):
  - 25% increase in latent cancer fatalities (LCFs)
  - 4% increase in prompt fatalities (PFs) (E.Lyman, *Science and Global Security*, forthcoming)
  - Both LCFs and PFs increase by about 30% for high-Ru release fraction

MOX IMPACT ON RISK

- Assuming all initiator frequencies remain the same, average LCF population risk (< 1 mi) also increases by 25% for DCS core
- Risk of MOX use can be assessed using RG 1.174 methodology by defining

$$\Delta LERF_{eff} \equiv LERF \times \Delta R/R$$

- Also useful for evaluation of extended power uprates

RISK IMPACT OF MOX IN ICE CONDENSER PLANTS

- Ice condensers "substantially more sensitive to early containment failure" than other PWRs (NUREG/CR-6427, April 2000)
- Precisely the class of accidents in which additional MOX source term impact is felt
- McGuire IPE LERF (int+ext): 4.7×10^{-6}
  - $\Delta LERF_{eff} = 1.2 \times 10^{-6}$
  - > RG 1.174 1×10^{-6} threshold

RISK IMPACT OF MOX IN ICE CONDENSER PLANTS (cont.)

- Estimate does not take into account Sandia finding that McGuire IPE underestimates ECF frequency by a factor of 7
- If taken into account, McGuire IPE LERF would exceed 1×10^{-5}: no LERF increase greater than 10^{-7} allowed (RG 1.174)
- MOX risk increase may be unacceptable
- Implications for extended power uprates

MOX IMPACT ON TRANSIENTS: OVERCOOLING EVENTS AND PTS

- PTS screening criteria for LEU cores may not be appropriate for MOX cores:
  - reduced decay heat leads to more rapid RCS temperature decrease
  - greater actinide and Ru inventory implies air oxidation source term is more severe
- faster embrittlement from greater fast flux
  - Duke Power not planning to consider MOX use in license renewal TLAAs

MOX IMPACT ON TRANSIENTS: REACTIVITY INSERTION

- Increased vulnerability of MOX fuel to RIAs is a concern (Cabri REP Na-7 test)
- Key consideration is fuel homogeneity and size distribution of Pu agglomerates:
  - Westinghouse (1994) recommended to DOE that "adherence to limits on Pu agglomerates in the range of 10 to 15 μm" be required
  - Yet DCS appears to be proposing a relaxation of the French specification!

PU PARTICLE DISTRIBUTIONS

- Cogema MIMAS Pu particle distribution:
  - mean size 20-40 μm
  - max. 2% of clusters > 100 μm
  - max. size about 140 μm
- DCS specification:
  - mean size < 50 μm
  - max. 5% of clusters > 100 μm
  - max. size 400 μm

MOX QUALIFICATION ISSUES

- Schedule for fuel qualification and licensing is very aggressive:
  - Oct. 2003: commencement of LTA irradiation
  - Oct. 2006: discharge of twice-burnt LTAs
  - Oct. 2007: first MOX reload batch
- Where will the LTAs be made?
  - if in Europe, may not be representative
  - if at the U.S. MOX plant, will cause delay

QUALIFICATION ISSUES (cont.)

- NRC ability to fully resolve MOX fuel safety issues is in jeopardy:
  - Time for post-irradiation LTA characterization and testing is insufficient
  - may force over-reliance on proprietary foreign data without confirmation --- Framatome/M5 experience should give pause
  - DOE uncooperative --- has rejected RES request for access to spent LTA rods

CONCLUSIONS

- Timetable and staff resources for MOX safety issue resolution should be based on NRC and not DOE needs
- Cancellation of the immobilization track will increase pressure on NRC not to be "obstructionist" in MOX licensing
- Current path for MOX fuel approval is not likely to engender public confidence

MOX RESEARCH

Presented to the
ACRS Reactor Fuel Subcommittee
April 4, 2001

by

Richard Lee
Office of Nuclear Regulatory Research
(301) 415-6795

MIXED-OXIDE FUEL

ISSUE

- Utilization of weapons-grade mixed oxide fuel (MOX) in specific U.S. Pressurized Water Reactors (PWRs)

BACKGROUND

- U.S. Department of Energy issued Record of Decisions (1/14/97 and 1/4/00) to pursue a hybrid approach to safely and securely dispose of up to 50 metric tons of surplus plutonium from the U.S.

- The hybrid approach allows for the immobilization of approximately 17 metric tons of surplus plutonium and the use of 33 metric tons in MOX fuel.

- Savannah River Site has been selected for weapons-grade MOX fuel fabrication.

- Weapons-grade MOX are to be used in selected U.S. PWR commercial reactors (McGuire and Catawba).

MIXED-OXIDE FUEL

RES ACTIVITIES:

- NRC/RES is to provide technical support in licensing review of weapons-grade MOX use in PWRs

- Technical support: Improvement to Analysis Codes and Assessment of Environmental Impact of MOX fuel use

  - Neutronics: develop models for MOX, benchmark against critical experiments, computational benchmarks, and plant data

  - Fuel: revised model for MOX, assessment of fuel behavior under normal and abnormal conditions

  - Source Terms: validate model(s) against relevant experimental data, and perform consequence analysis

MIXED-OXIDE FUEL

STATUS:

- Conduct Phenomena Identification and Ranking Tables (PIRTs) for MOX http://www.nrc.gov/RES/PIRT/)

  - PIRT for LOCA and reactivity accident completed
  - PIRT for source term is being initiated and expects to complete by this year

- Neutronics:

  PARCS code development at the Purdue University
  - initiated in November 2000
  - implement and assessment of multi-group, P1 and P3 for X-sections representations
  - collaboration with France - Saclay, comparison of CRONOS vs. PARCS
  - development of a "theoretical" benchmark for reactivity transient for MOX under discussion with OECD/NEA

- Neutronics: (continued)

  Brookhaven National Laboratory

  - independent assessment of PARCS
  - provide feedback to code developer
  - assist in assembling benchmark/assessment problems for PARCS analysis of MOX cores
  - assist NRR in review of technical issues related to MOX licensing as needed (e.g. MOX fuel qualification program)

  Oak Ridge National Laboratory

  - Initiated the development of the NEWT lattice physics code

- Fuel:

  Pacific Northwest National Laboratory

  - initiated modifications of fuel codes for MOX analysis
  - assess code against MOX fuel behavior (e.g., Halden)

- Source Terms:
  - Initiated effort to obtain relevant experimental data (e.g., VERCORS, France; VEGA, Japan) for the assessment of fission products release models for MOX fuel
  - Additional experimental data may be available from the IPSN MAGRAGUE program at Cadarache, France
- Assist in licensing review of technical issues as they arise

LOCA Ductility of M5™ Cladding

Garry L. Garner
Framatome ANP
Lynchburg, Virginia

Outline

- ➢ Review of In-Reactor Operating Experience
  - Alloy composition, fabrication parameters
  - Corrosion/hydrogen properties
- ➢ Review of High Temperature Testing
  - Oxidation Tests
  - Quench Tests
  - Post-Quench Mechanical Testing
- ➢ Conclusions and Summary

In-Reactor Performance

Alloy M5™

- Composition:
  - Sn: An impurity in M5™
  - Fe: Target 250 - 500 ppm (improve corrosion)
  - O: Target value 1250 - 1450 ppm (improve creep)
  - S: Maintain consistent creep behavior
- Thermomechanical processing
  - Low temperature annealing to insure stable microstructure

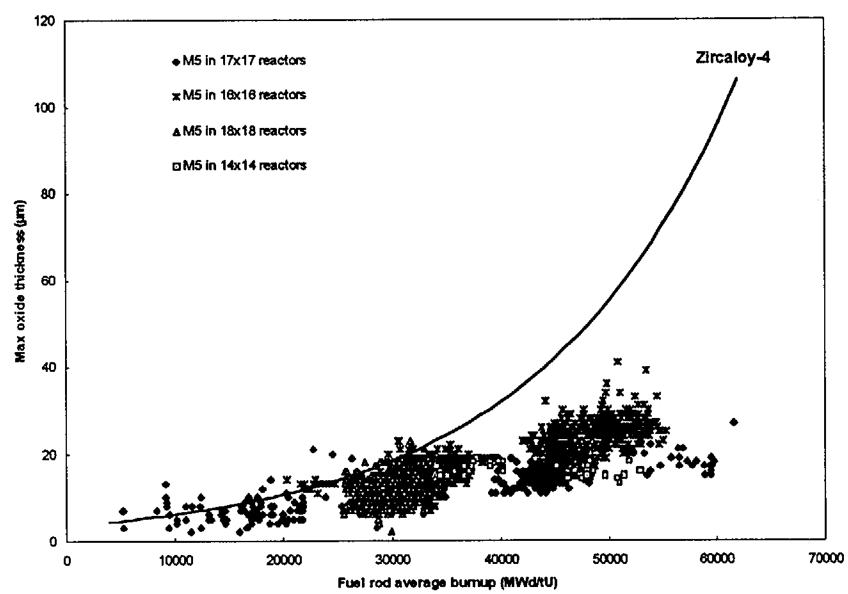
M5™ PWR Corrosion Performance

- Additional data in BU range 50-60 GWd/tU

- Excellent corrosion behavior of M5™
  - for all designs and for all operating conditions

- Thickness < 40 µm for BU up to 63 GWd/mtU

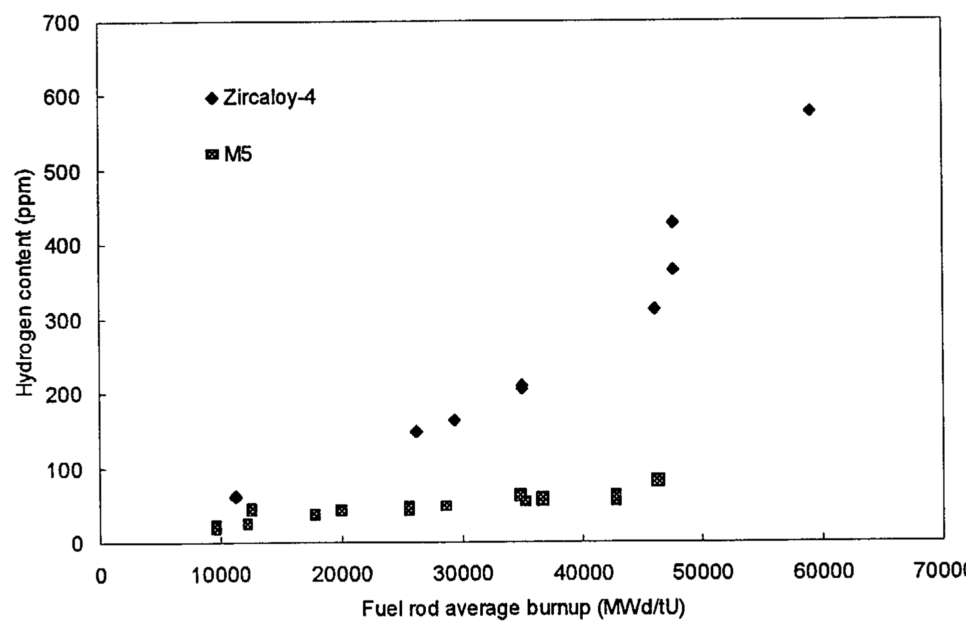
Corrosion behavior of Zirconium alloy claddings



PWR Hydrogen Performance Of M5™

- Significant reduction of clad hydrogen content

- Additional M5™ data at high burnup planned in 2001



FRAMATOME ANP

Summary: M5™ In-Reactor Performance

- Low oxidation rate

- No increase in rate to burnups of 63 GWd/mtU

- Lower sensitivity to temperature and rod power than Zr-4 (reactor duty)

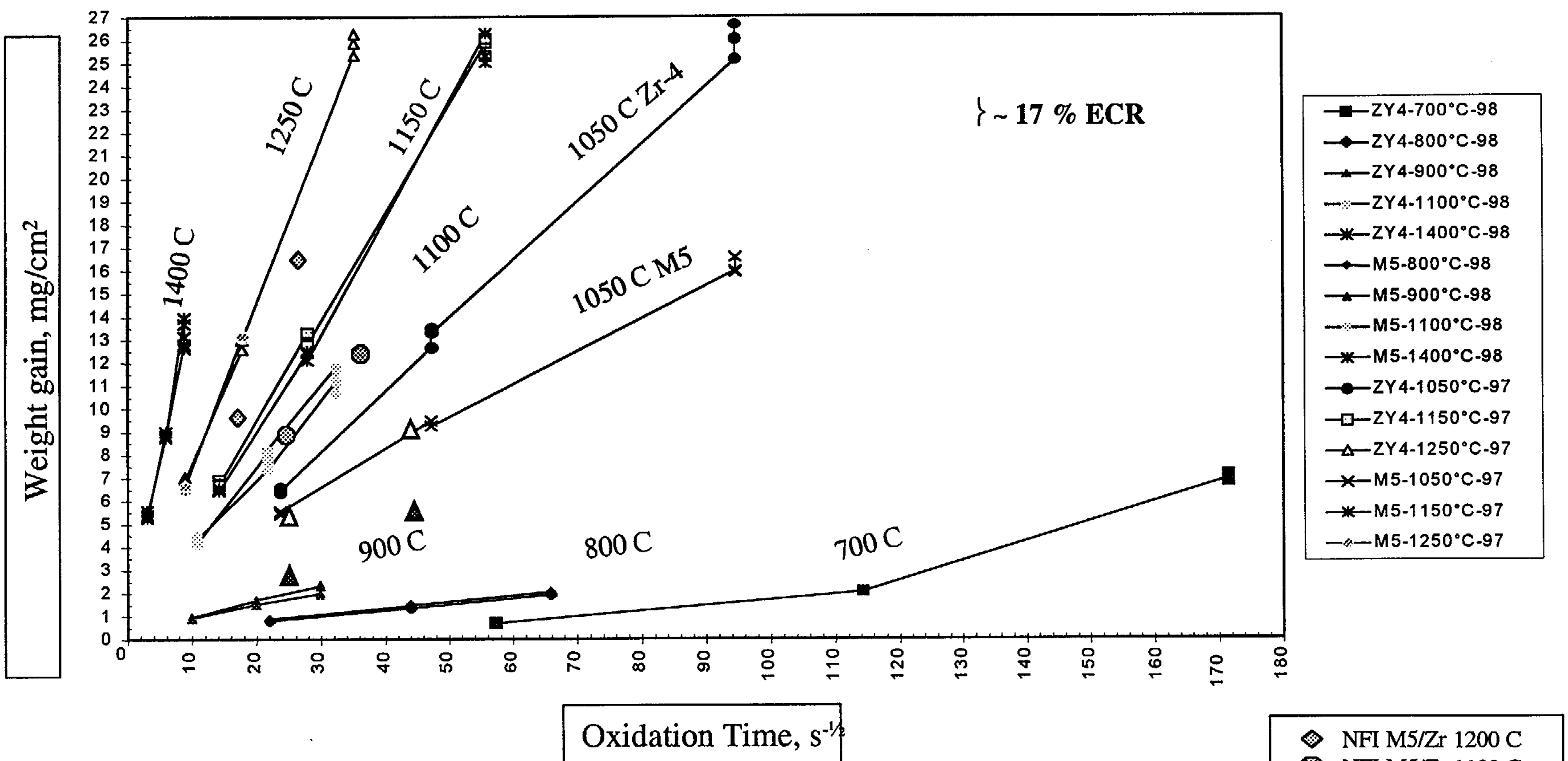
- **Low oxidation rate + low hydrogen absorption = low hydrogen content at high burnup**

CINOG High Temperature Testing

High Temperature Oxidation (CINOG) Test Matrix

- Oxidation Tests (M5™ and Zr-4)
  - Double-Sided Oxidation (L = 20 mm)
  - T = 700, 800, 900, 1050, 1100, 1150, 1250, 1400 C (as manufactured cladding)
  - T = 1200 C for Pre-Hydrided Cladding (200 ppm for M5, 200 and 450 ppm for Zr-4)
  - 3 Oxidation times/Test Temperature → (50, 100, and 200 μm/side)
  - 3 Samples/Test Condition (2 repeat tests)

Oxidation 700 TO 1400°C
Zr-4 and M5™



- M5™ behaves better than Zr-4 at 1050°C
- Zr-4 values are consistent with literature
- M5™ values are consistent with independent Japanese tests

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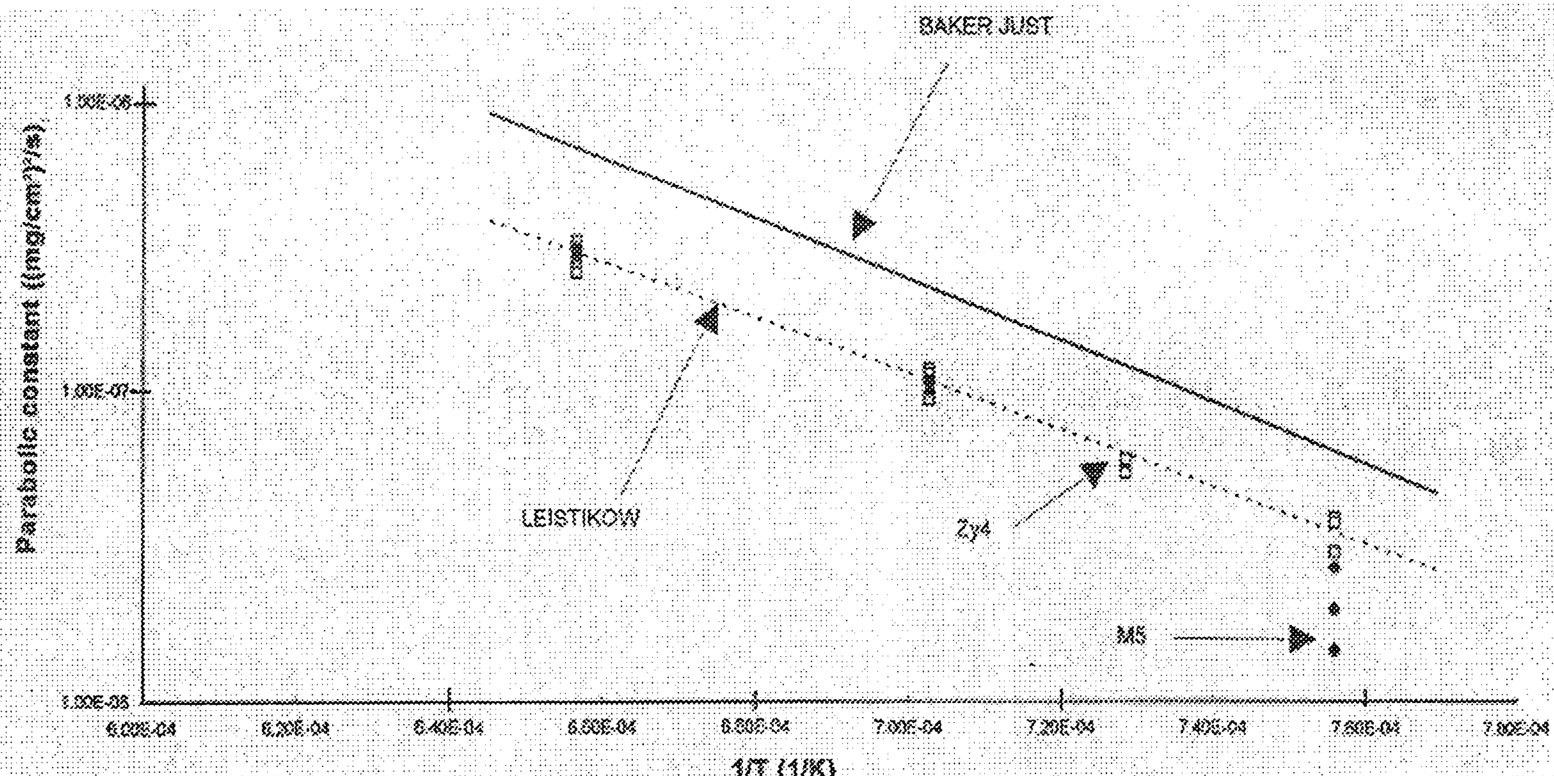
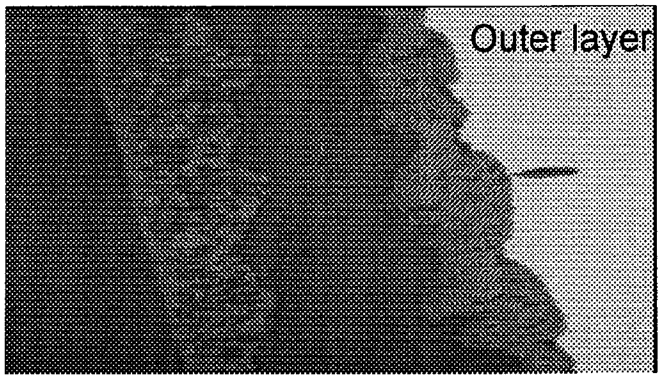
Oxidation Kinetics - CINOG
Comparison with literature results

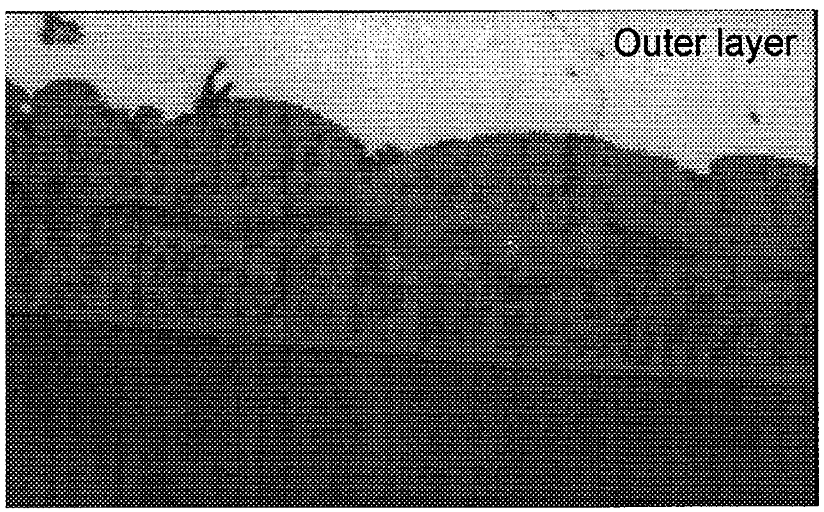


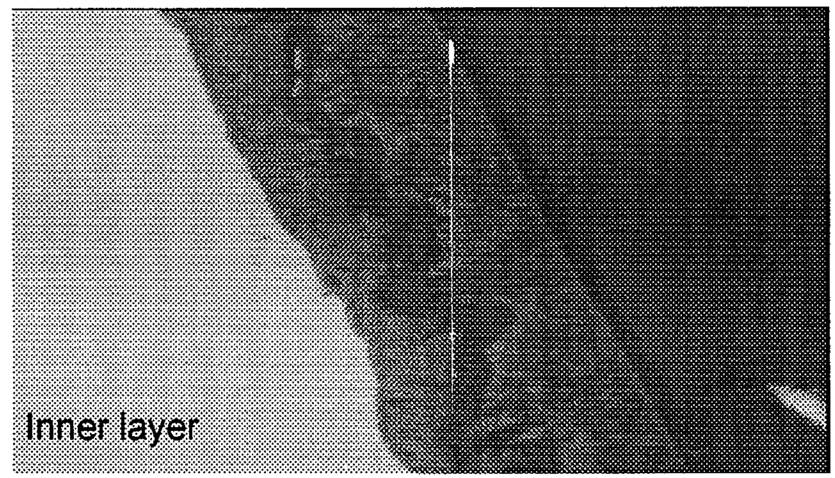
→ BAKER-JUST MODEL IS BOUNDING IN ALL ENCOUNTERED CONFIGURATION

→ LESTIKOW MODEL ACCURATELY PREDICTS ZY4 RESULTS AND IS BOUNDING FOR M5™

Zy-4 Metallographic Observations After Oxidation at 1000°C for 3,270 Seconds



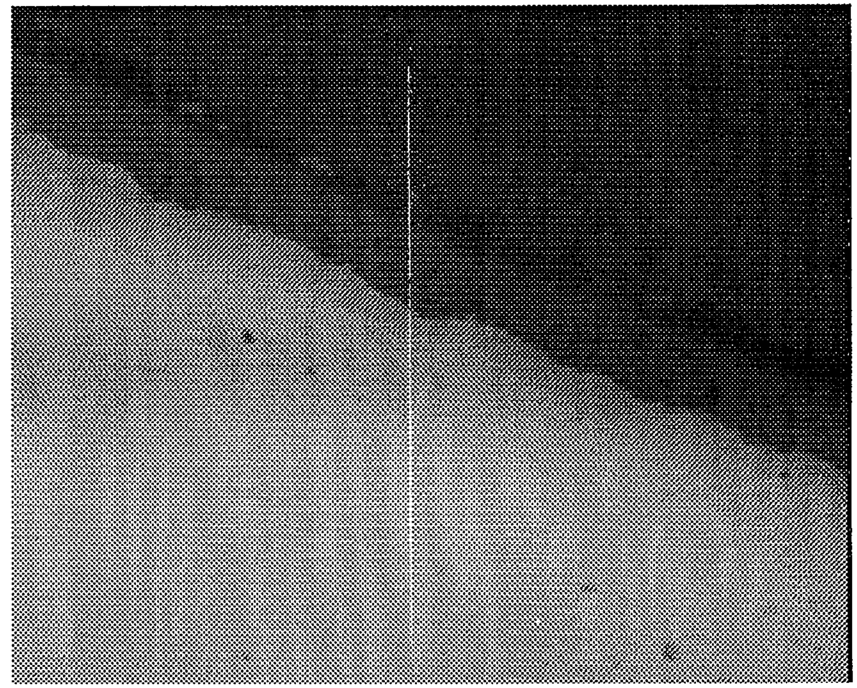




→ Trace of delamination in inner and outer zirconia layers

M5™ Metallographic Observations After Oxidation at 1000°C for 3,270 Seconds

Outer layer



Inner layer

- The inner and outer zirconia layers are homogeneous
- No trace of delamination

FRAMATOME ANP

Metallographic Observations M5™ and Zr-4 After Oxidation at 1000 °C for 3,270 seconds

	Zr-4	M5™
External Zirconia Layer (μm)	55.2 to 61	18.9 to 20.3
External α Zr-O Layer (μm)	53.9 to 71.6	53.9 to 61.9
β_{Zr} Layer (μm)	351 to 379	394 to 409
Internal α Zr-O Layer (μm)	53.5 to 70.4	47.4 to 57.3
Internal Zirconia Layer (μm)	48.7 to 55.8	19.0 to 21.7

CINOG Quench Test Matrix

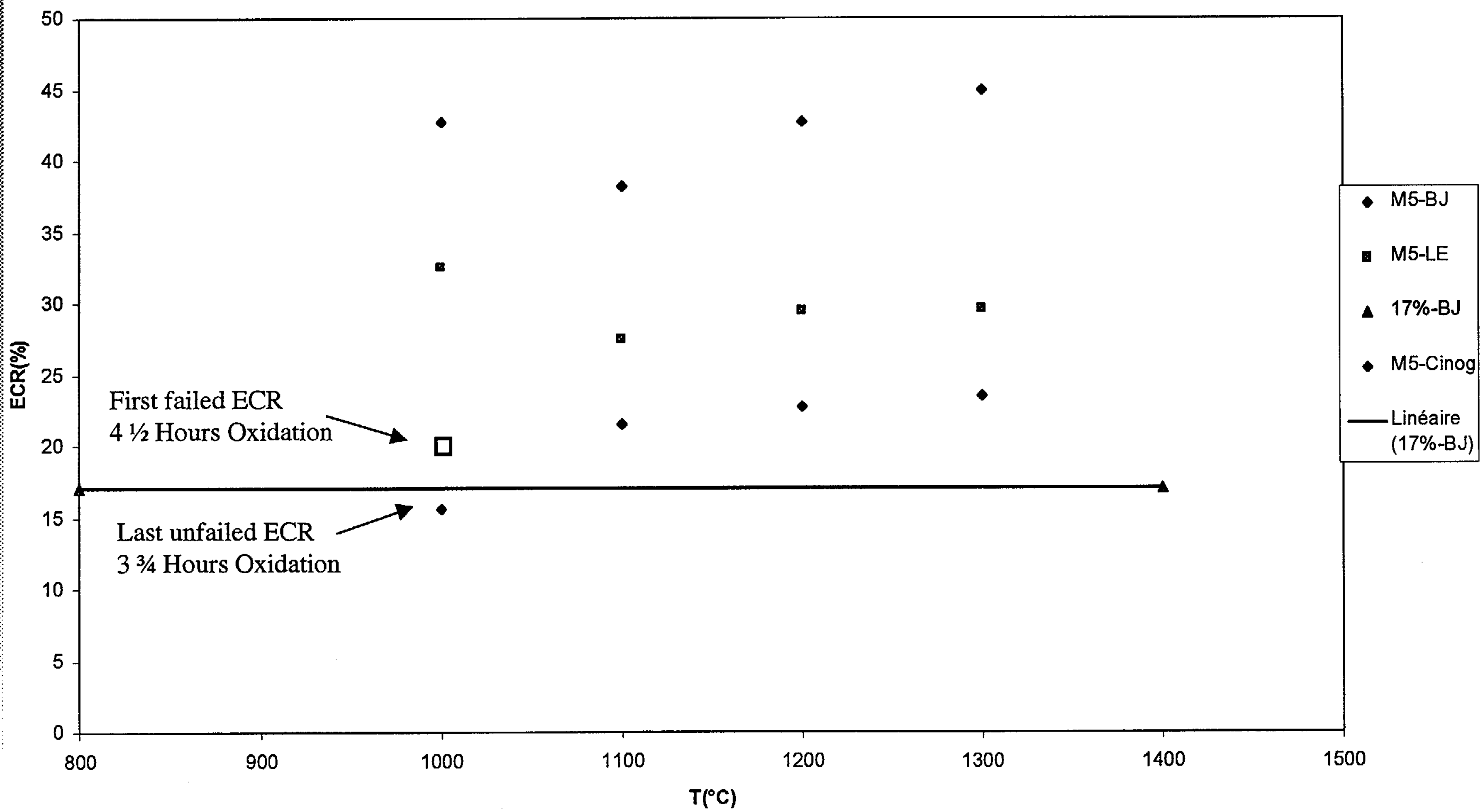
- Quench Embrittlement Tests (M5™ and Zr-4)
  - Double-Sided Oxidation (L = 100 mm)
  - Cladding Failure when Cladding Leaks Air Under Slight Overpressure
  - T = 1000, 1100, 1200, 1300 C for as manufactured cladding
  - T = 1200 C for Pre-Hydrided Samples (200 ppm for M5, 200 and 450 ppm for Zr-4)
  - Generally 5 or more Tests to Establish Cladding Failure Threshold

- Post-Test Metallography and Hydrogen Analysis

Quench Test Results

Alloy	Oxidation Temperature (°C)	Time to Failure (seconds)
Zr-4	1000	6,500
	1100	2,970
	1200	950
	1300	390
M5™	1000	13,500
	1100	2,959
	1200	1,200
	1300	495

CINOG Quench Test
ECR versus temperature



FRAMATOME ANP

CINOG Quench Test

HYDROGEN CONTENT IN Zy-4 AND M5™ AFTER QUENCH

Alloy	Oxidation temperature (°C)	Duration (sec)	(H2) (ppm)
Zy4	1100	2970	24-32-22
ZY4	1200	950	21-22-22
Zy4	1300	390	26-25-25
M5™	1100	2959	18-18-20
M5™	1200	1200	16-19-17
M5™	1300	495	21-24-21

- Maximum oxidation duration before embrittlement similar or higher for M5™
- Slight hydrogen pickup, practically temperature-independent

CINOG Test Results Summary

- High Temperature Oxidation Performance of M5™ is Equivalent or Superior to Zr-4

- M5™ Hydrogen Uptake is Low

- M5™ Accident Survival is Superior to Zr-4

  T > 1100 C M5™ and Zr-4 Have Similar Survival Ability
  T < 1100 C M5™ Survives up to 2 Times Longer than Zr-4

- M5™ Does Not Exhibit Delamination of Oxide

- Using Baker/Just to Establish ECR M5™ Always Meets the 17 % Criterion

- At Moderate Temperatures (1100 C > T > 900 C) M5™ Requires Excessive Oxidation Times to Achieve ECRs near 17 %

- Because M5™ Actually Performs Better During an Accident, The LOCA Criterion Should Remain 17 % Local Oxidation as Calculated by Baker/Just

Post-Quench Mechanical Tests

Post-Quench Mechanical Tests
Test Matrix

➢ Oxidation

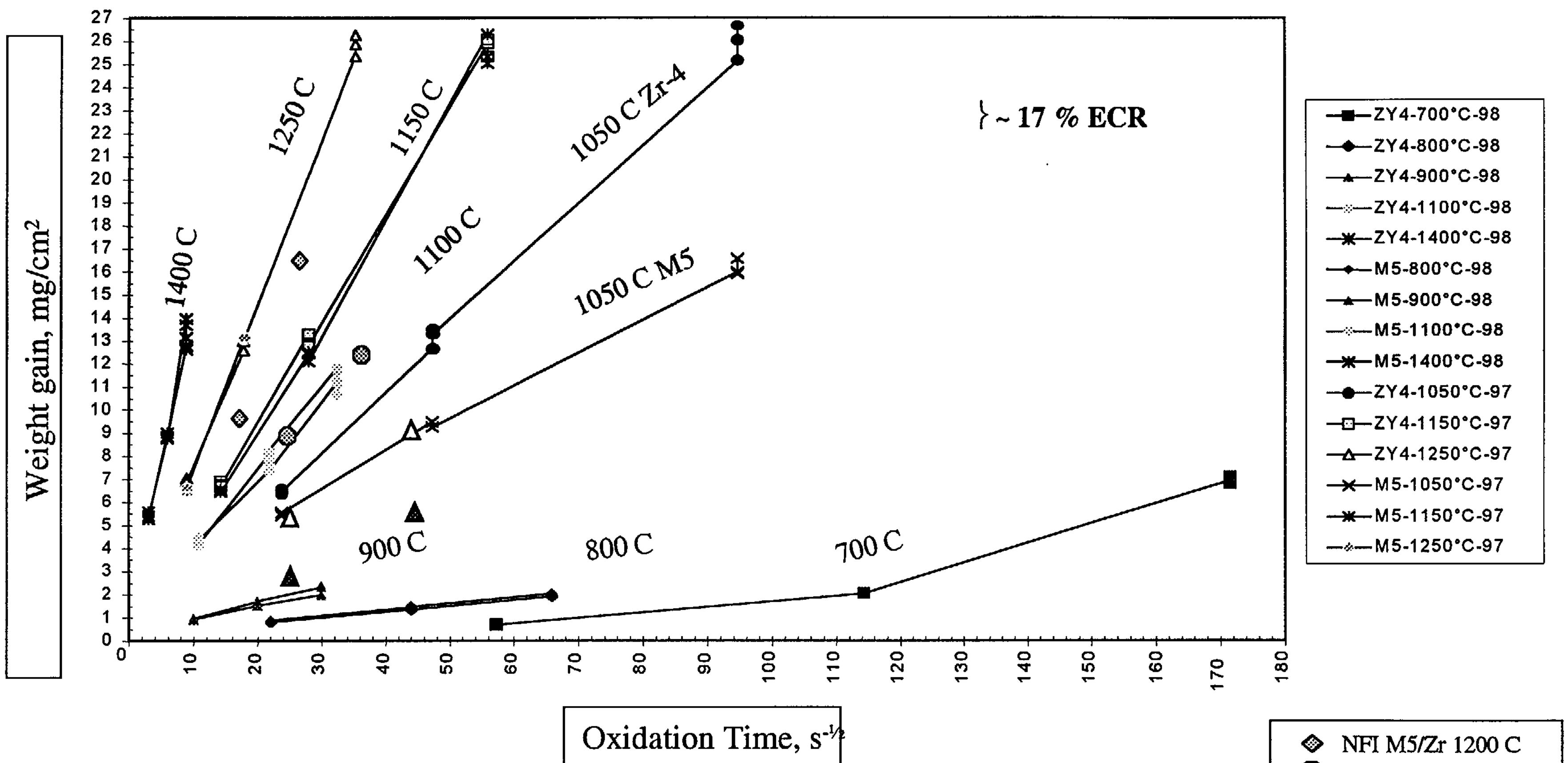
- T=1100 °C
- t → ECR=3, 6, 10 and 17% (Lestikow law)
- Single face oxidation
- As-fabricated M5™ and Zr-4 cladding tubes

➢ Water Quench

➢ Mechanical tests

- Three point bend
- Impact
- Ring compression

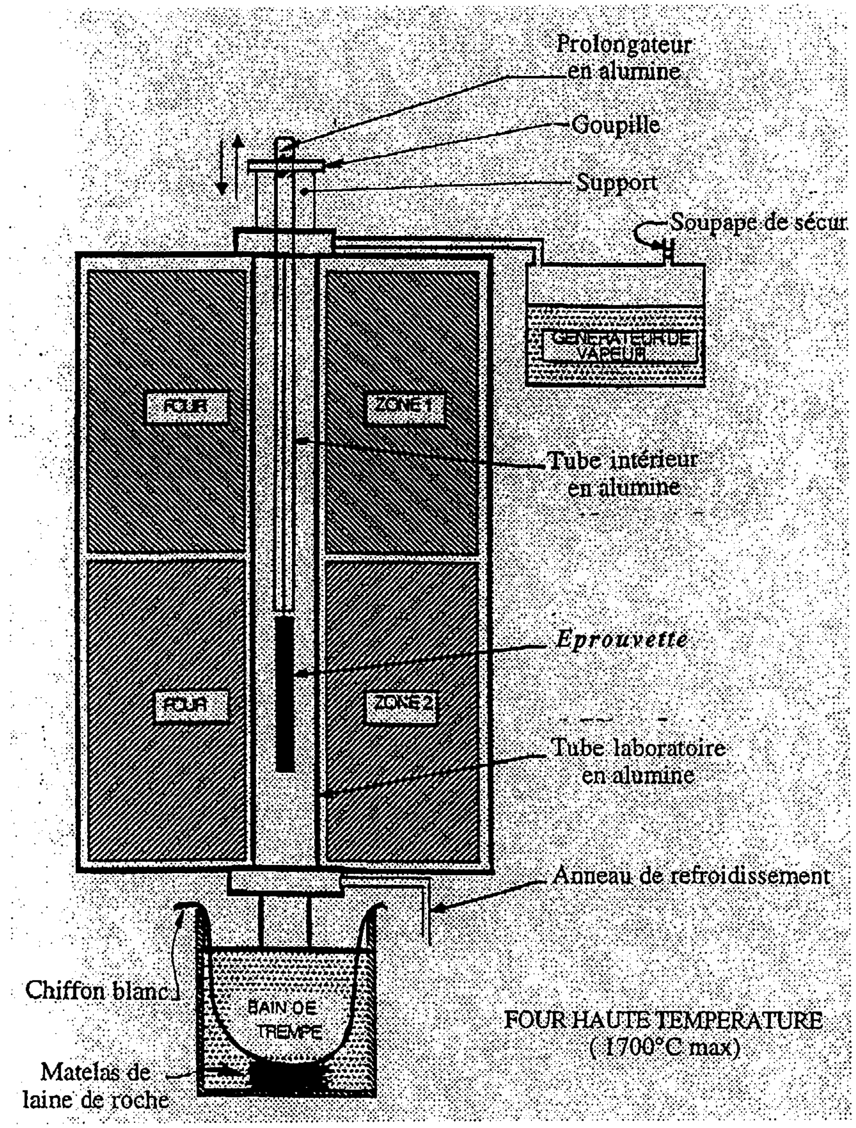
OXIDATION 700 TO 1400°C ZIRCALOY-4 and M5™



- M5™ behaves better than Zr-4 at 1050°C
- Zr-4 values are consistent with literature
- M5™ values are consistent with independent Japanese tests

FRAMATOME ANP

Post-Quench Mechanical Test
Oxidation - Device



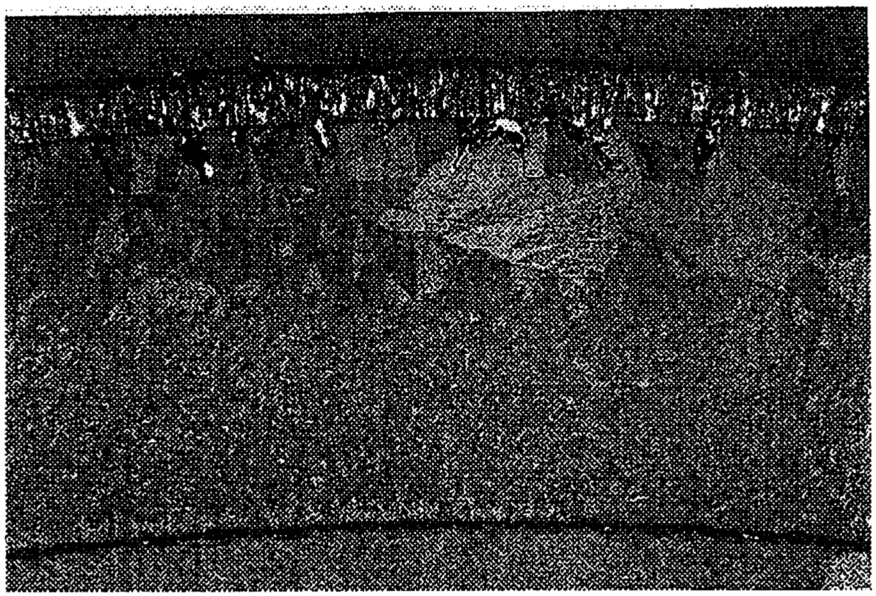
FRAMATOME ANP

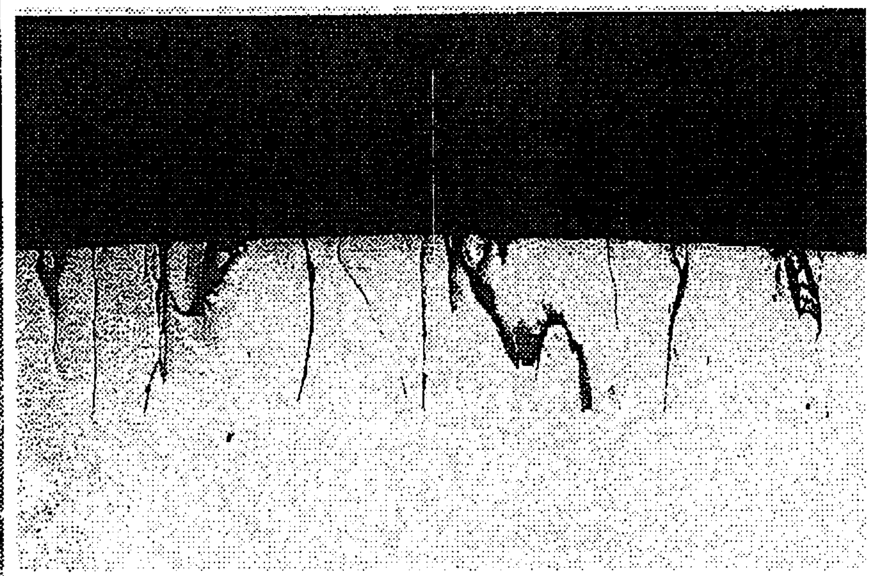
Post-Quench Mechanical Test

Percent of spalled oxide after oxidation at 1100 °C and quench for the longest exposure time (3098 - 3800 s)

Alloy	Test Number	Weight Gain During Oxidation (g)	Oxide Spalled (g)	Oxide Spalled (%)
Zr-4	71	1.0839	0.7215	66.9
	74	1.0799	0.6975	64.6
	77	1.0919	0.9088	83.2
M5	73	1.1634	0.0230	2
	76	1.1544	0.0259	2.2
	79	1.1696	0.0458	3.9

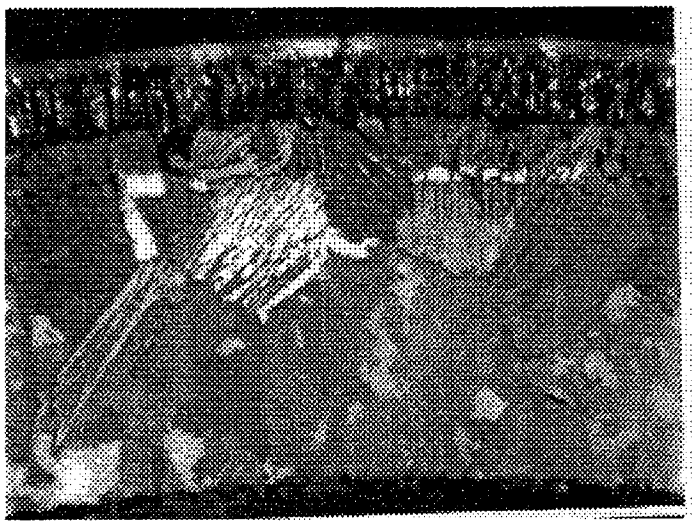
Metallographic Observations Of Low-Tin Zr-4 After Oxidation At 1100°C t = 1349 s and Quenched

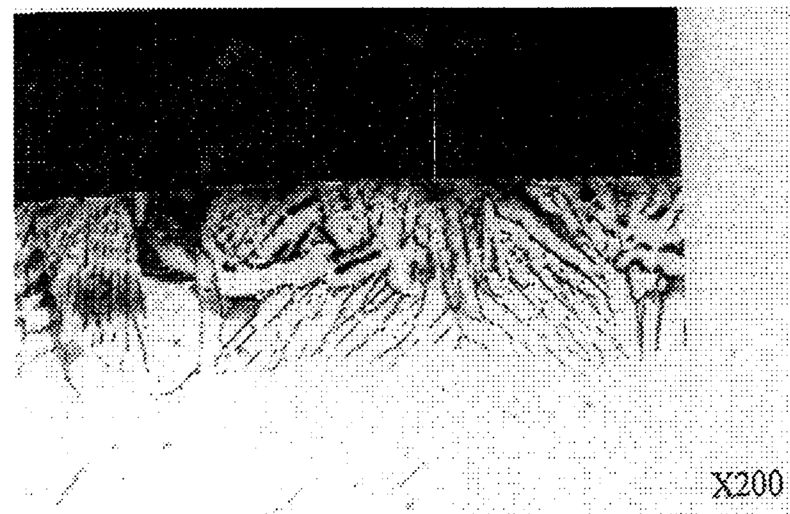




- ➢ α-Zr(0) layer: large α-grains
- ➢ α-Zr(0) layer: cracks

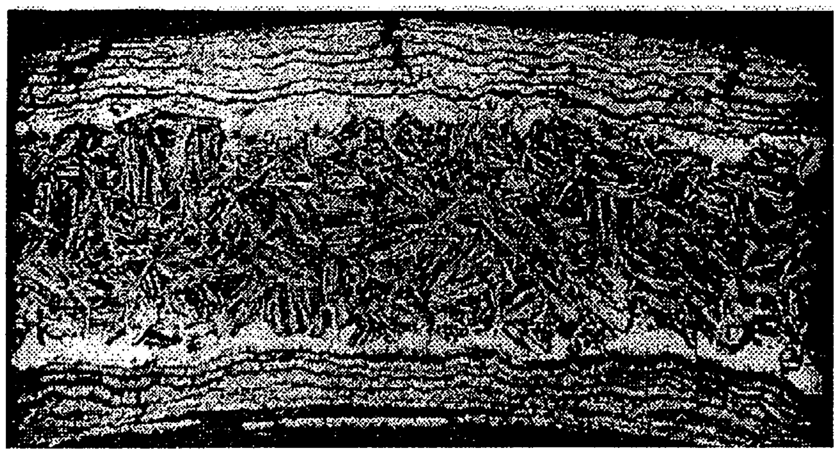
Metallographic Observations Of M5™ After Oxidation At 1100°C t = 3600 s and Quenched





- α Zr (O) layer : Linear distribution of niobium particles in α platelets
- α Zr (O) layer : no cracks

High - Temperature Oxidation
Russian Alloy E-110 Cladding



- Stratified and cracked oxide layer
- Different morphology than M5™

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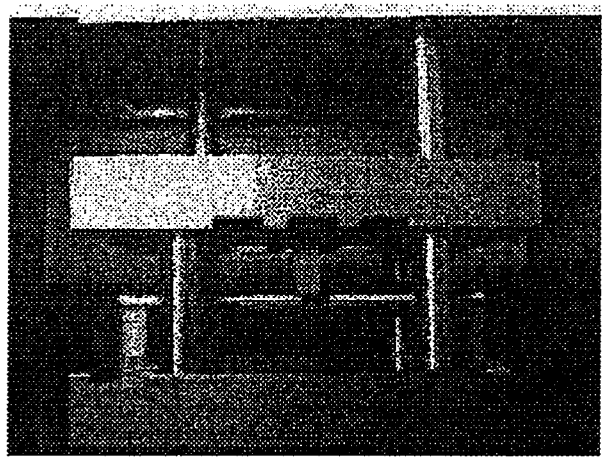
"… at an early stage, multilayer oxide scales are formed which tend to flake."

Böhmert et al. on Russian alloy E110

- M5™ has not exhibited multilayered oxide scale
- M5™ did not flake in quench tests

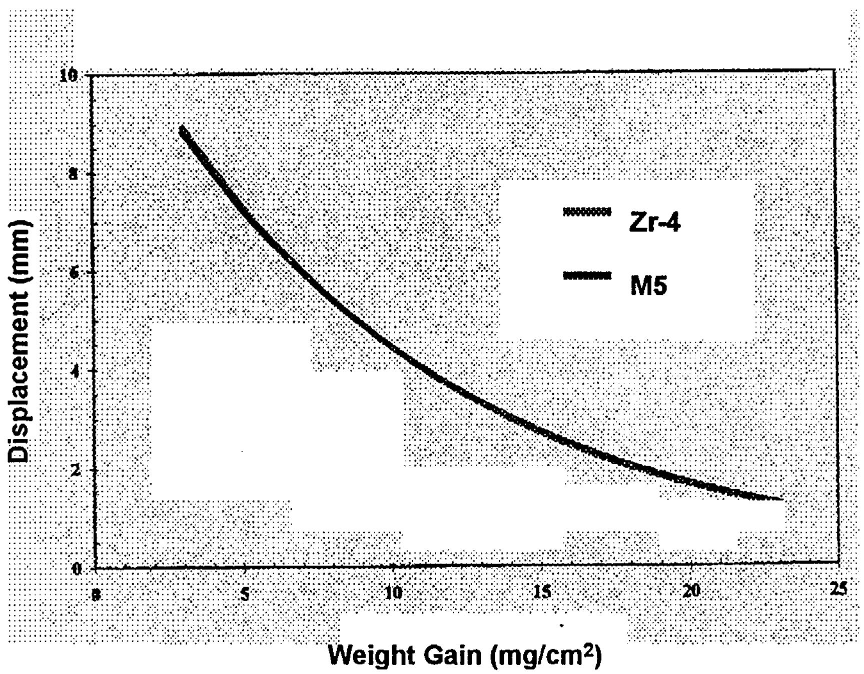
Post-Quench Mechanical Test
3 Point Bend Test Apparatus



Starting position

7.5 mm displacement

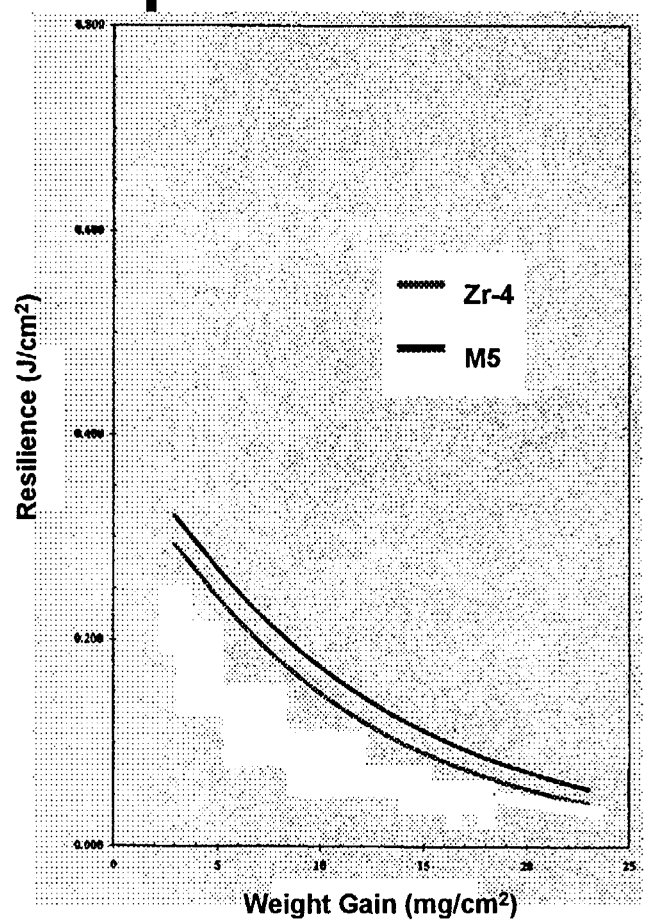
Post-Quench Mechanical Test
3 Point Bend Test Results



- M5™ and Zy4 behave similarly

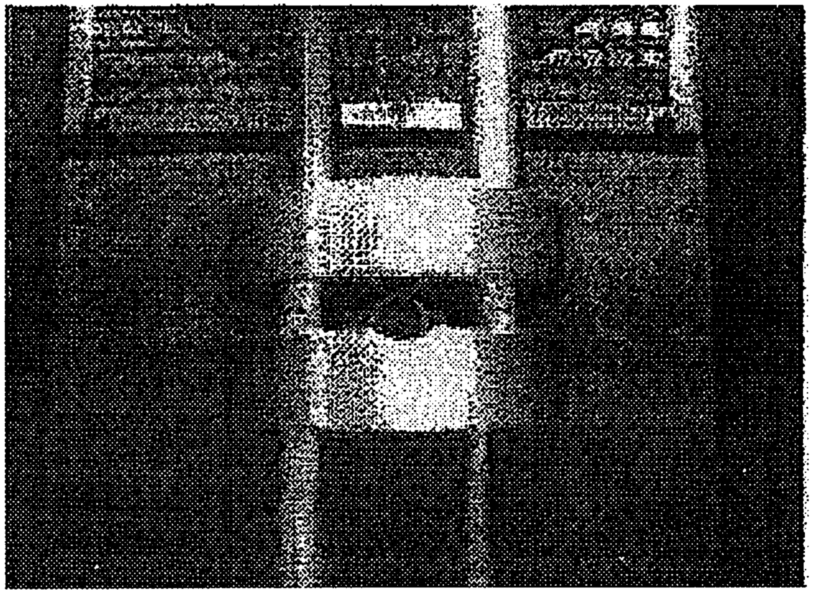
FRAMATOME ANP

Post-Quench Mechanical Test
Impact Test Results



- M5™ behaves slightly better than Zr-4
- Zy4: ductile rupture in ex α-β phase and brittle fracture in α-Zr(0)
- M5™: ductile rupture in ex-α-β phase and quasi-ductile in α-Zr(0) layer

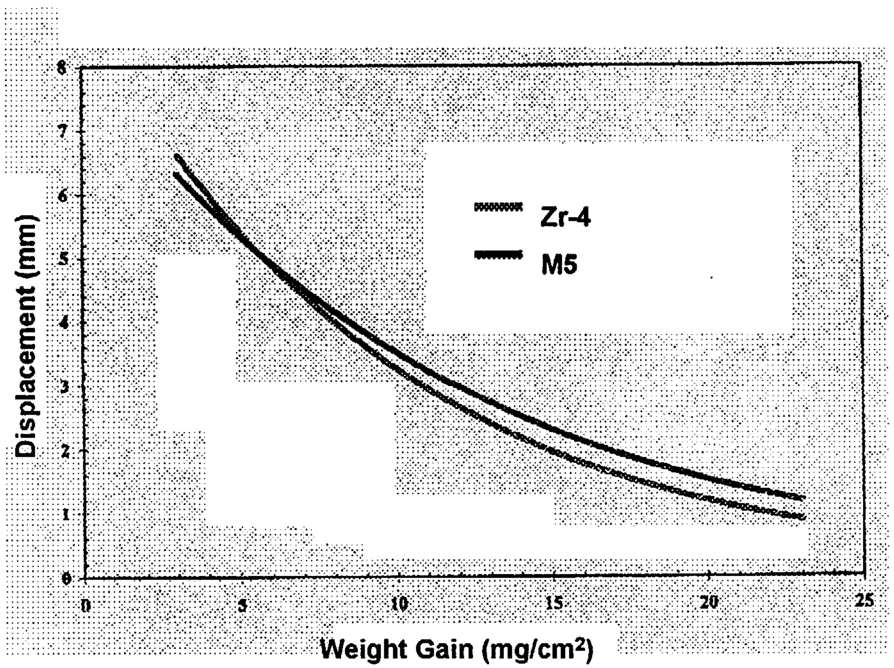
Post-Quench Mechanical Test
Ring Compression Test



Starting position

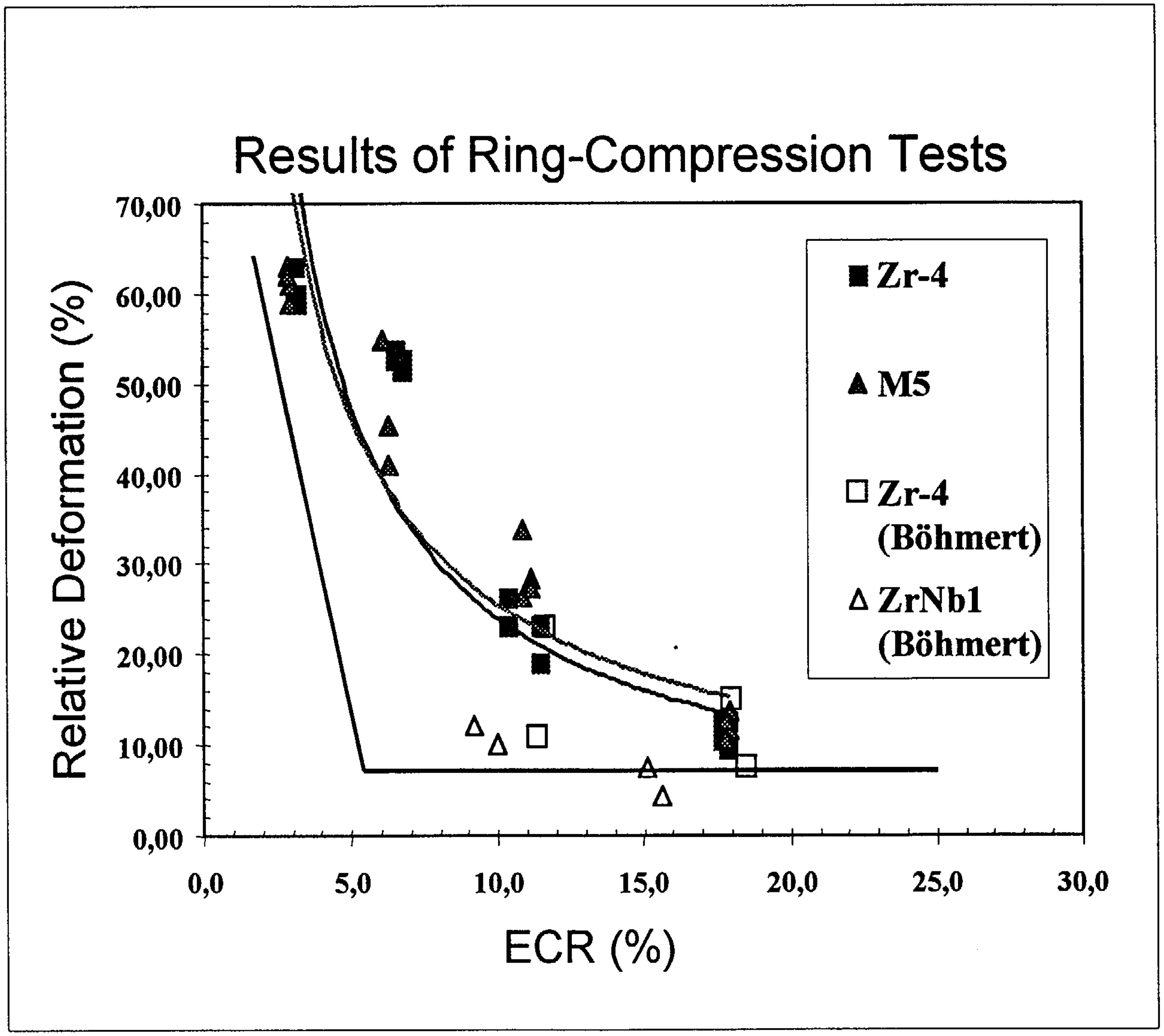
FRAMATOME ANP

Post-Quench Mechanical Test
Ring Compression Test Results



- M5™ behaves slightly better than Zr-4

Comparison With Böhmert's Results at 1100°C



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Conclusions
Post-Quench Mechanical Tests

- M5™ Tested in the Böhmert range with results different than E110
  - Order of magnitude less hydrogen uptake
  - Completely different oxide morphology

- M5™ Performed better than or similar to Zr-4
  - No delamination
  - Similar bend test results
  - Slightly better impact test results
  - Slightly better than Zr-4 and much better than E110 in ring compression tests

- Böhmert's conclusions regarding Zr-1Nb alloy performance may be valid for Russian alloy E110 tested in 1992, but are not valid for M5™
  - Significantly different composition and processing parameters

Summary

- M5™ in-reactor operating performance is superior to Zr4
- M5™ LOCA and post-LOCA oxidation rates are equal to or slower than Zr4
- M5™ LOCA and post-LOCA mechanical performance is equivalent to Zr4
- M5™ LOCA and post-LOCA performance is acceptable and is equal to or better than Zr4 in events of equal duration
- M5™ LOCA and post-LOCA mechanical performance is superior to the Zr-1%Nb alloy tested by Böhmert

FRAMATOME ANP

United States Nuclear Regulatory Commission

SUMMARY OF OECD TOPICAL MEETING ON LOCA FUEL SAFETY CRITERIA

Ralph Meyer
Office of Nuclear Regulatory Research

ACRS Reactor Fuels Subcommittee
April 4, 2001

OECD/NEA/CSNI
SPECIAL EXPERT GROUP ON FUEL SAFETY MARGINS
Wolfgang Wiesenack (Halden, Norway), Chair

TOPICAL MEETING ON LOCA FUEL SAFETY CRITERIA
Georges Hache (IPSN France), Technical Program Chair
Aix-en-Provence, France
March 22-23, 2001

Proceedings to be Published

Post-Quench Ductility
- Background (G. Hache, IPSN)
- Hungarian Paper on E110 (L. Maroti, AEKI)
- 2 Russian Papers on E110 (L. Andreeva-Andrievskaya and N. Sokolov, VNIINM)
- French Paper on M5 (A. Lebourhis, Framatome)
- American Paper on ZIRLO (W. Leech, Westinghouse)

Effect of Axial Constraint during Quenching
- Japanese Paper on Experiments (Uetsuka, JAERI)
- French-American Paper on Calculations (Waeckel, EPRI & EdF)

Relocation of Fragmented Fuel into Ballooned Region
- French Paper on Calculations (M. Lambert, EdF)
- French Paper on Calculations (C. Grandjean, IPSN)

RATIONALE OF THE LOCA 10CFR50.46b CRITERIA FOR ZIRCALOY AND COMPARISON WITH E110 ALLOY

G. Hache (IPSN, France)
Introductory Presentation

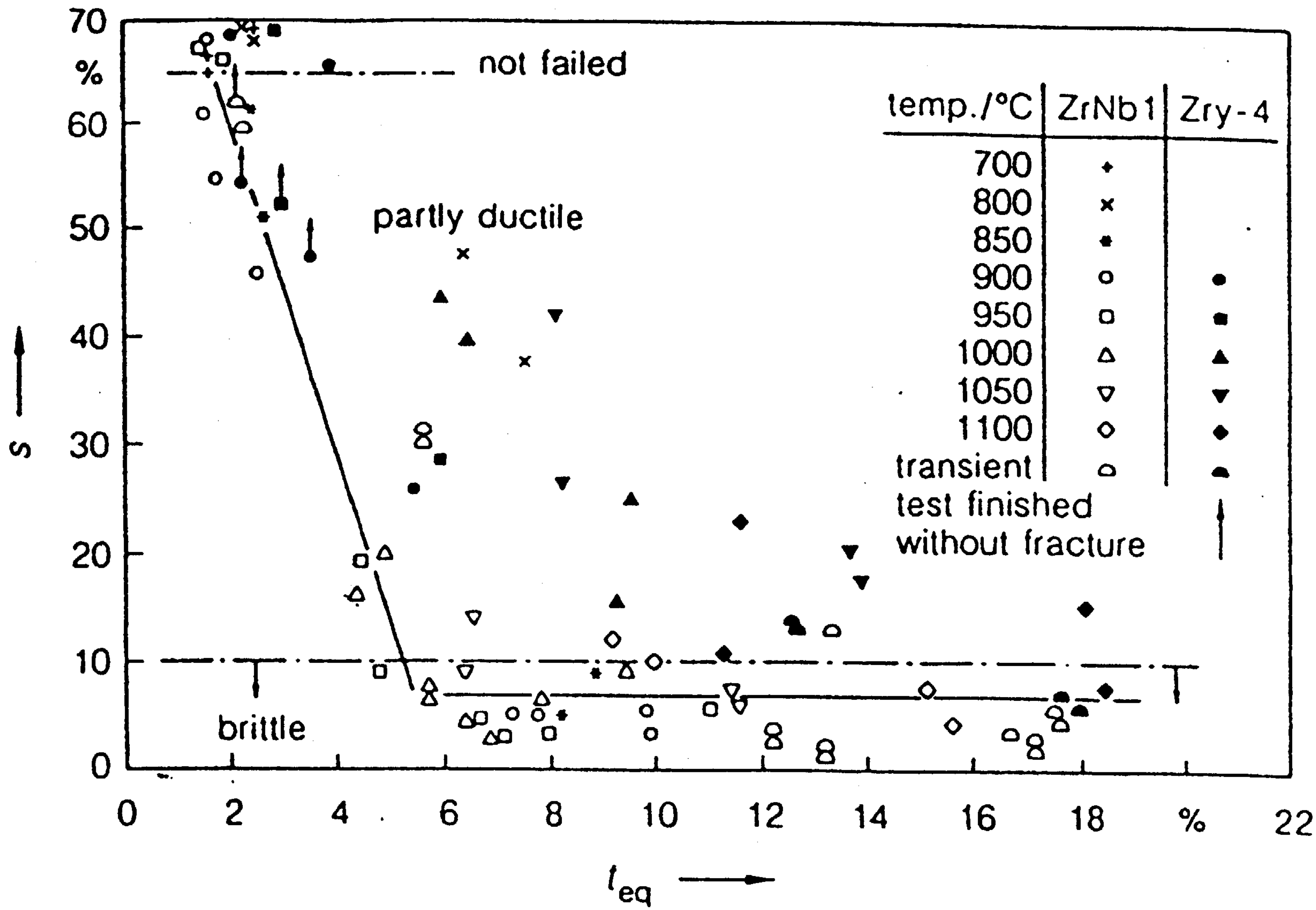
OPINION OF THE REGULATORY STAFF AND COMMISSIONERS (1)
(ECCS Rule - Making Hearing, 1973)

- Reluctance to neglect the effect of mechanical constraints on thermal-shock fragmentation
  - rod-rod interaction due to ballooning or bowing
  - rod-grid interaction due to differential shrinkage between fuel rods and guide tubes

- Justified later by JAERI constraint-quench tests and Phebus LOCA-219 bundle test

OPINION OF THE REGULATORY STAFF AND COMMISSIONERS (2)

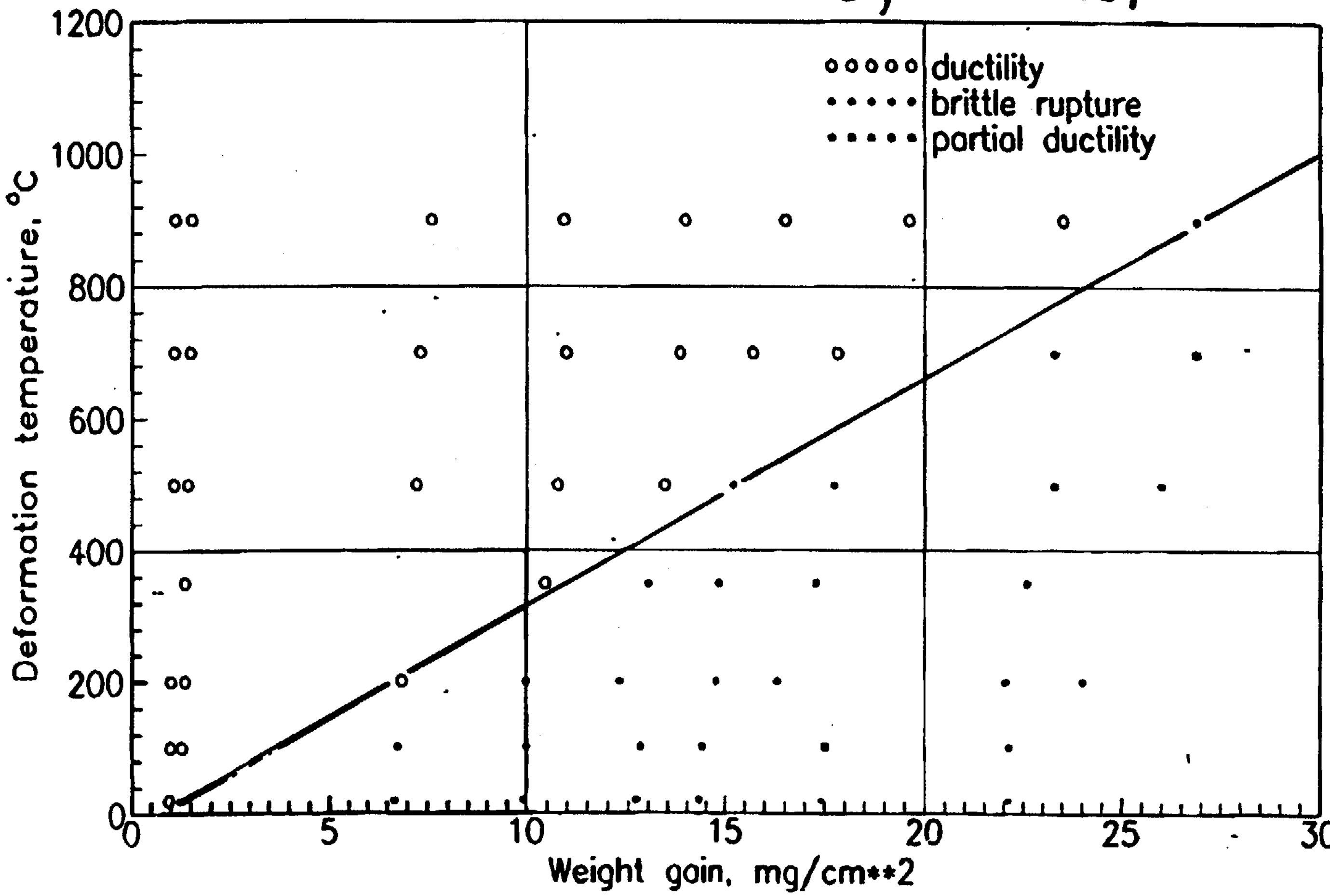
- Retention of ductility is the best guarantee against potential fragmentation under various types of loadings (thermal shock, hydraulic, seismic forces).
- Results from unconstrained quench tests (simple thermal-shock test) were:
  - considered only corroborative and reassuring.
  - Their use for regulatory purposes not accepted.
  - Later studies showed a large margin compared to 17% -ECR and 2200°F -PCT criteria.
- 17% -ECR and 2200°F -PCT criteria are based on results from post-quench ductility test (Hobson's slow-ring-compression tests).



J. Böhmert, "Embrittlement of ZrNb1 at room temperature after high-temperature oxidation in steam atmosphere," Kerntechnik, Vol. 57 (1992) p. 56

DUCTILITY TESTS (2)

Ductility of steam-reacted Zr1%Nb claddings. Ring compression results.
Bochvar Institute, Varna 1994



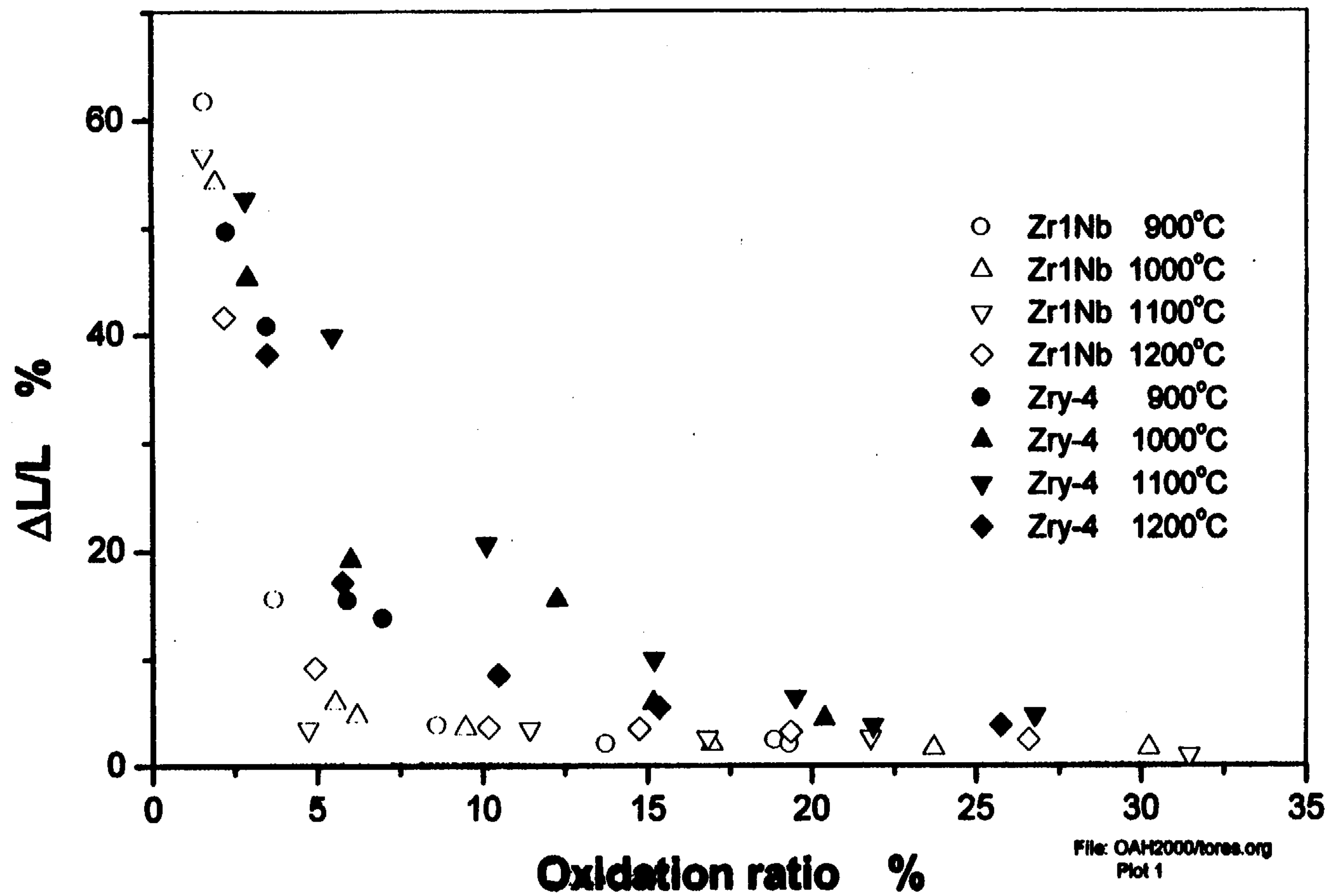
very good linear correlation for ZDT

The line reaches 275F (135°C) for a weight gain of 4.7 mg/cm² that is to say 6% ECR

The application of the hearing methodology leads to ZDT ≤ 275F or ECR ≤ 6%

PRESENTATIONS ON POST-QUENCH DUCTILITY

1. L. Maroti (AEKI, Hungary)
2. N. Sokolov (Bochvar, Russia)
3. A. Lebourhis (Framatome, France)
4. W. Leech (Westinghouse, USA)
5. H. Chung (Argonne, USA)



Ring compression tests with preoxidised Zr1%Nb and Zircaloy-4 claddings

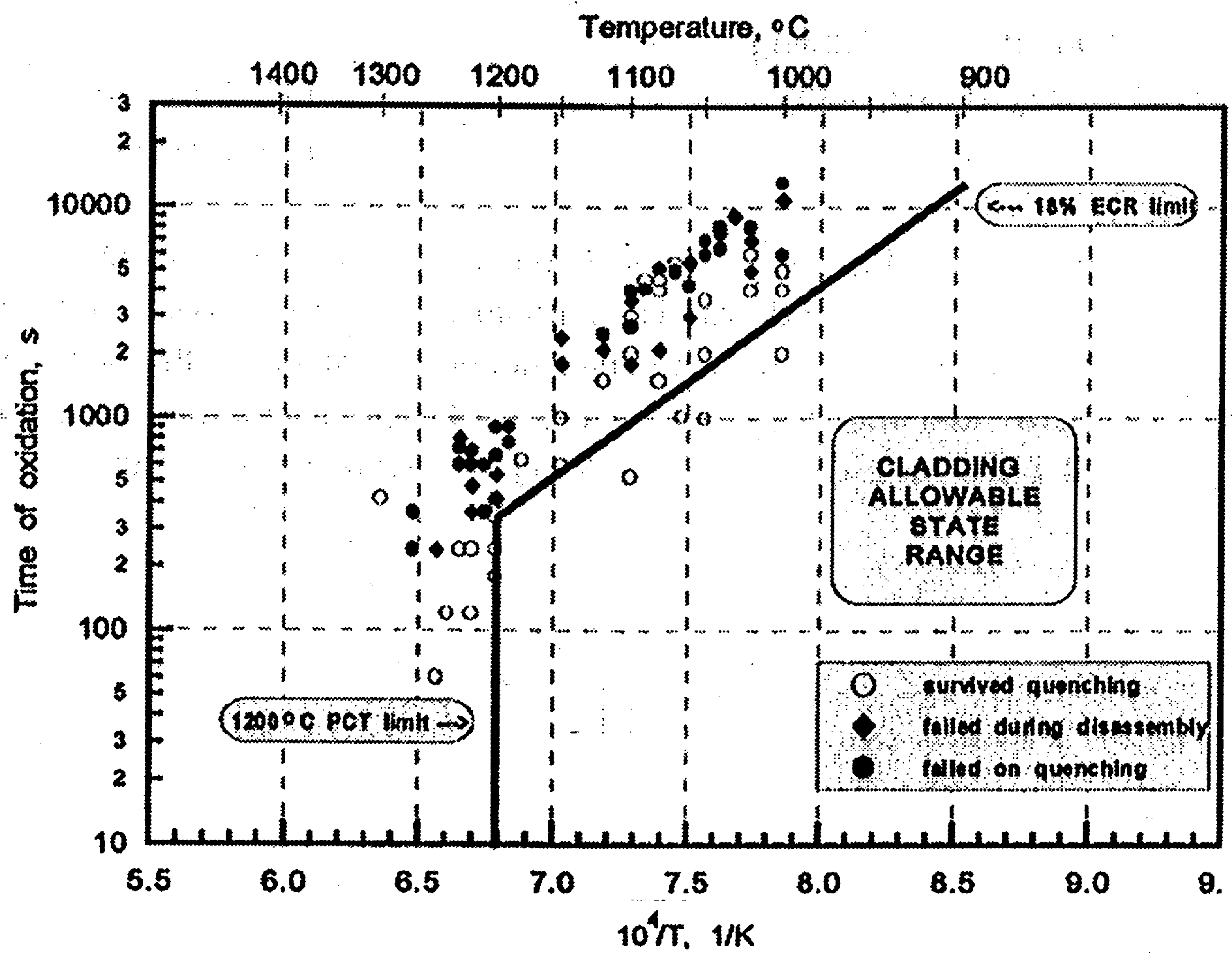
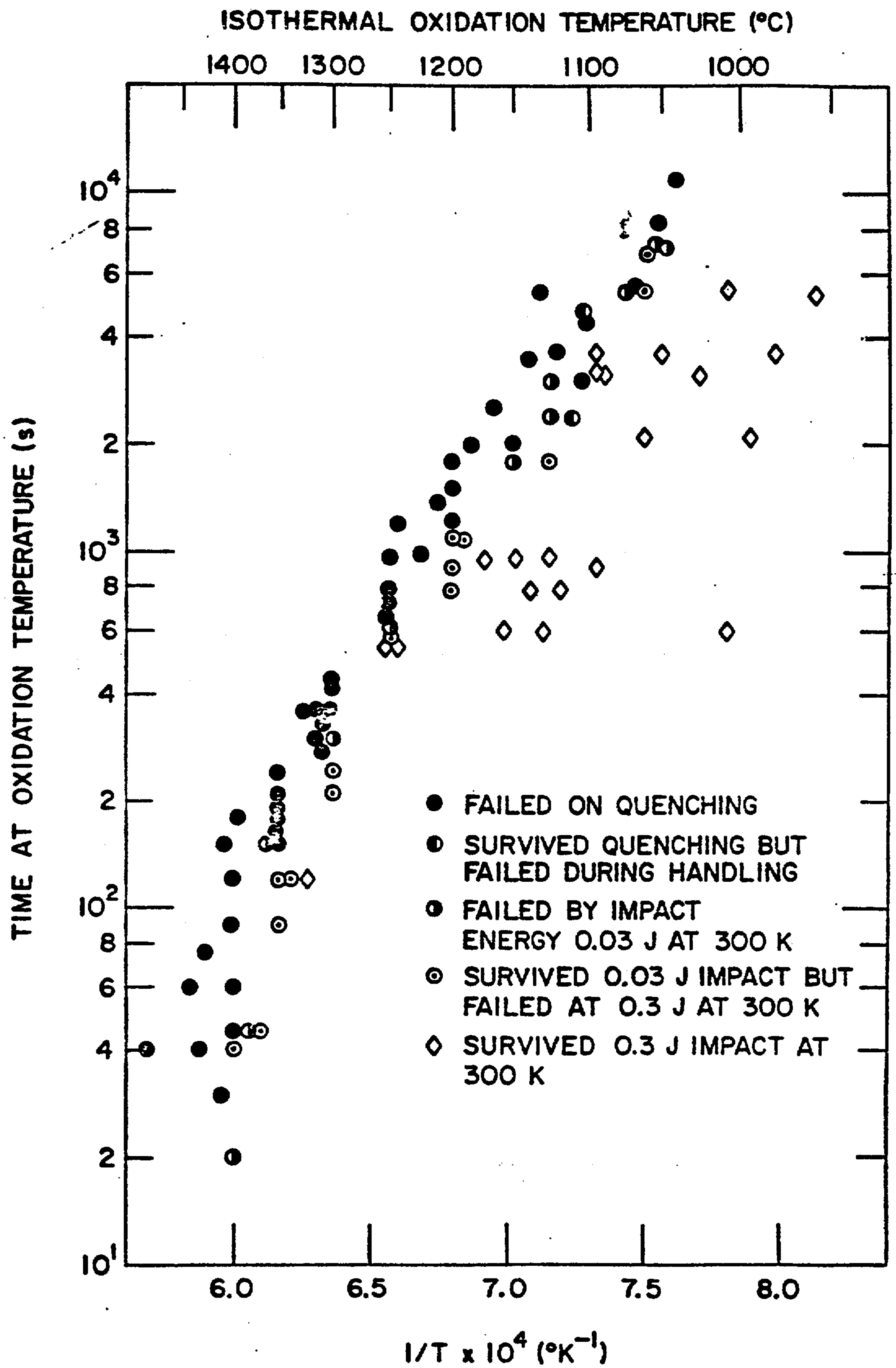
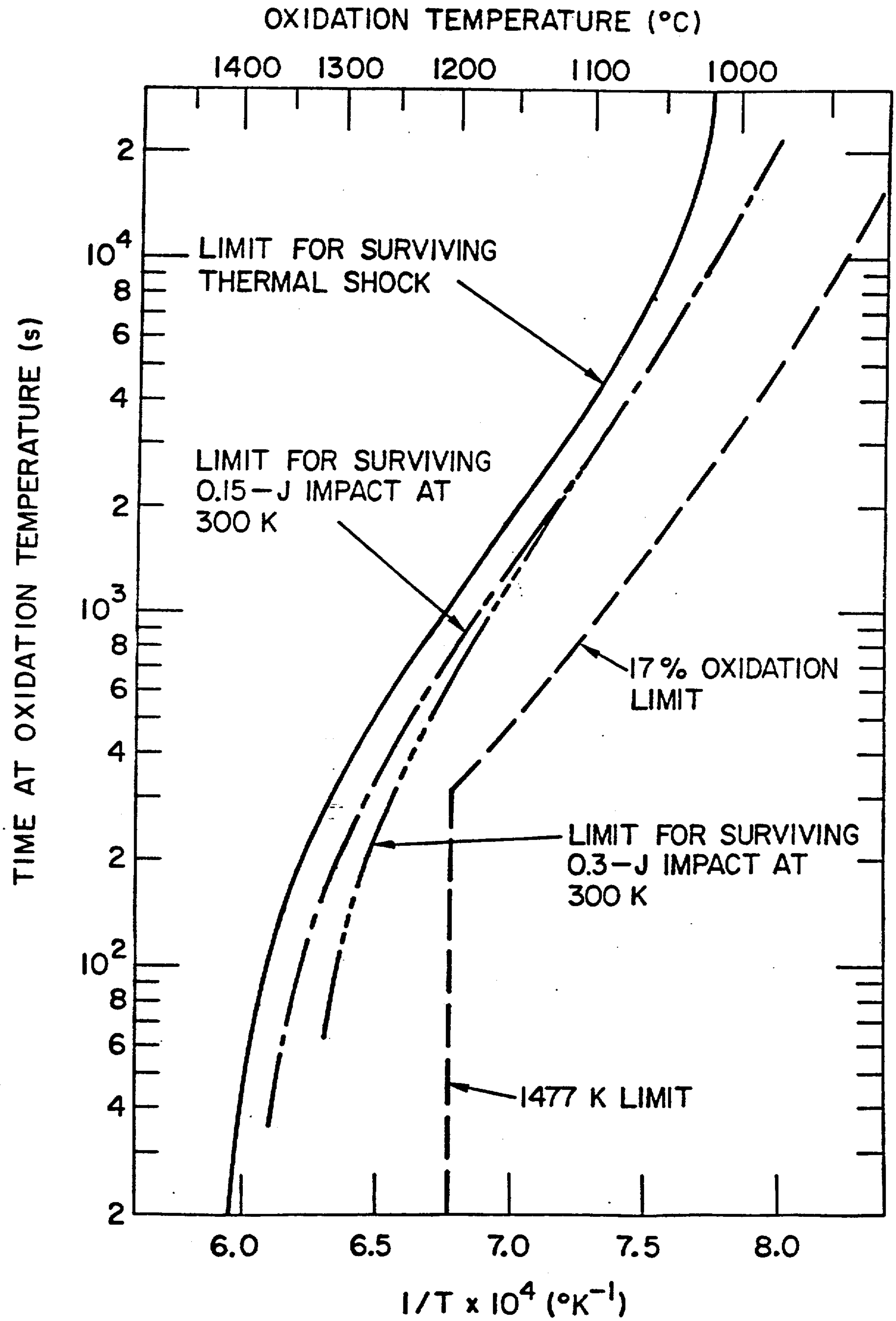


Fig.5 Post-test appearance and failure map.
Simulator of type 1



R. Van Houten, "Fuel Rod Failure as a Consequence of Departure from Nuclear Boiling or Dryout, NUREG-0562 (1979) Fig. 2-A



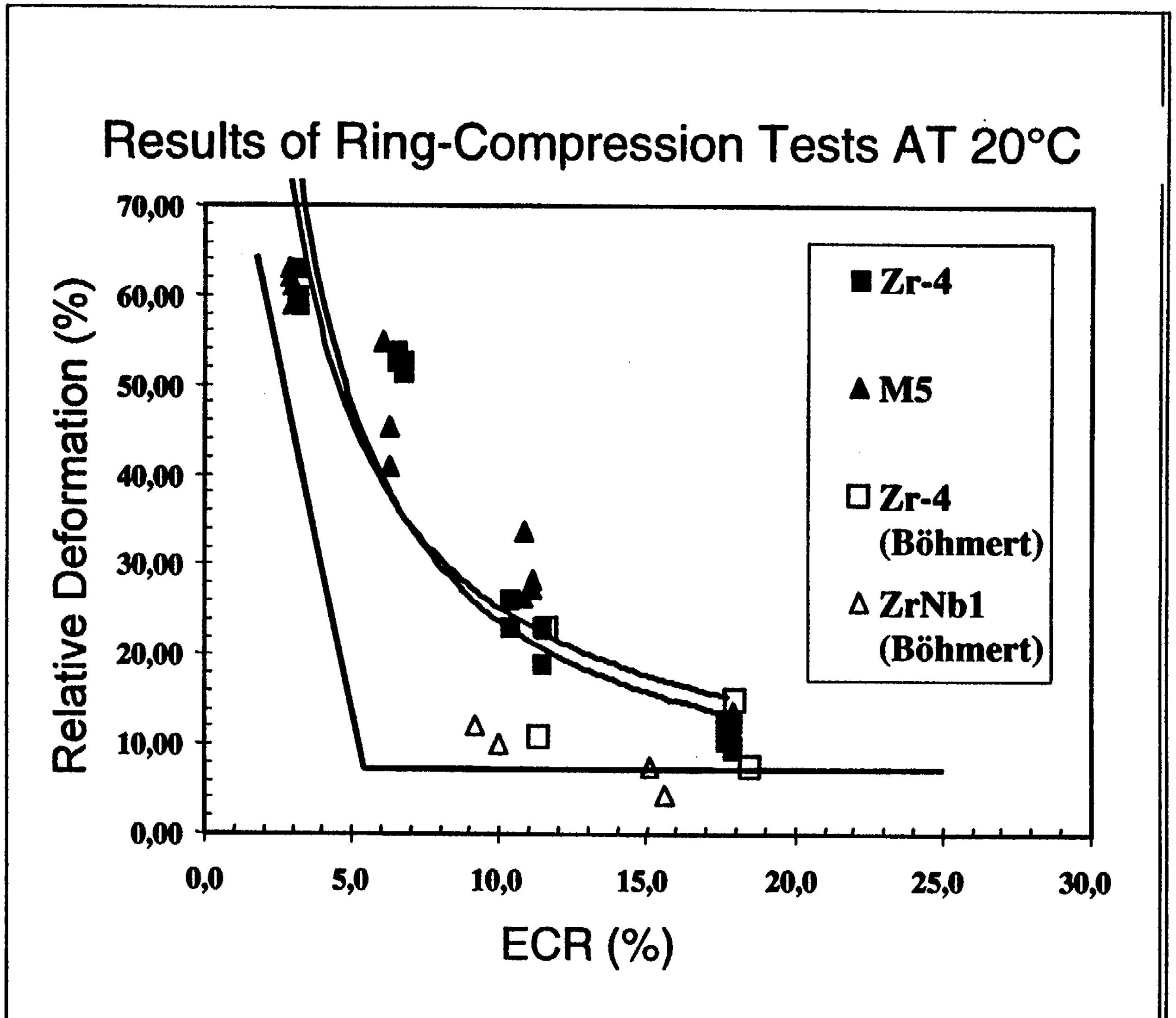
R. Van Houten, "Fuel Rod Failure as a Consequence of Departure from Nuclear Boiling or Dryout, NUREG-0562 (1979) Fig. 2-B

4. QUENCH EMBRITTLEMENT : MAIN RESULTS (1/2)

ALLOY	OXIDATION TEMPERATURE(°C)	TIME TO FAILURE (S)	ECR FAILURE (%)
Zy4	1000	6500	22*
	1100	2970	30
	1200	950	29
	1300	390	29
M5™	1000	13500	16 *
	1100	2959	28
	1200	1200	30
	1300	495	31

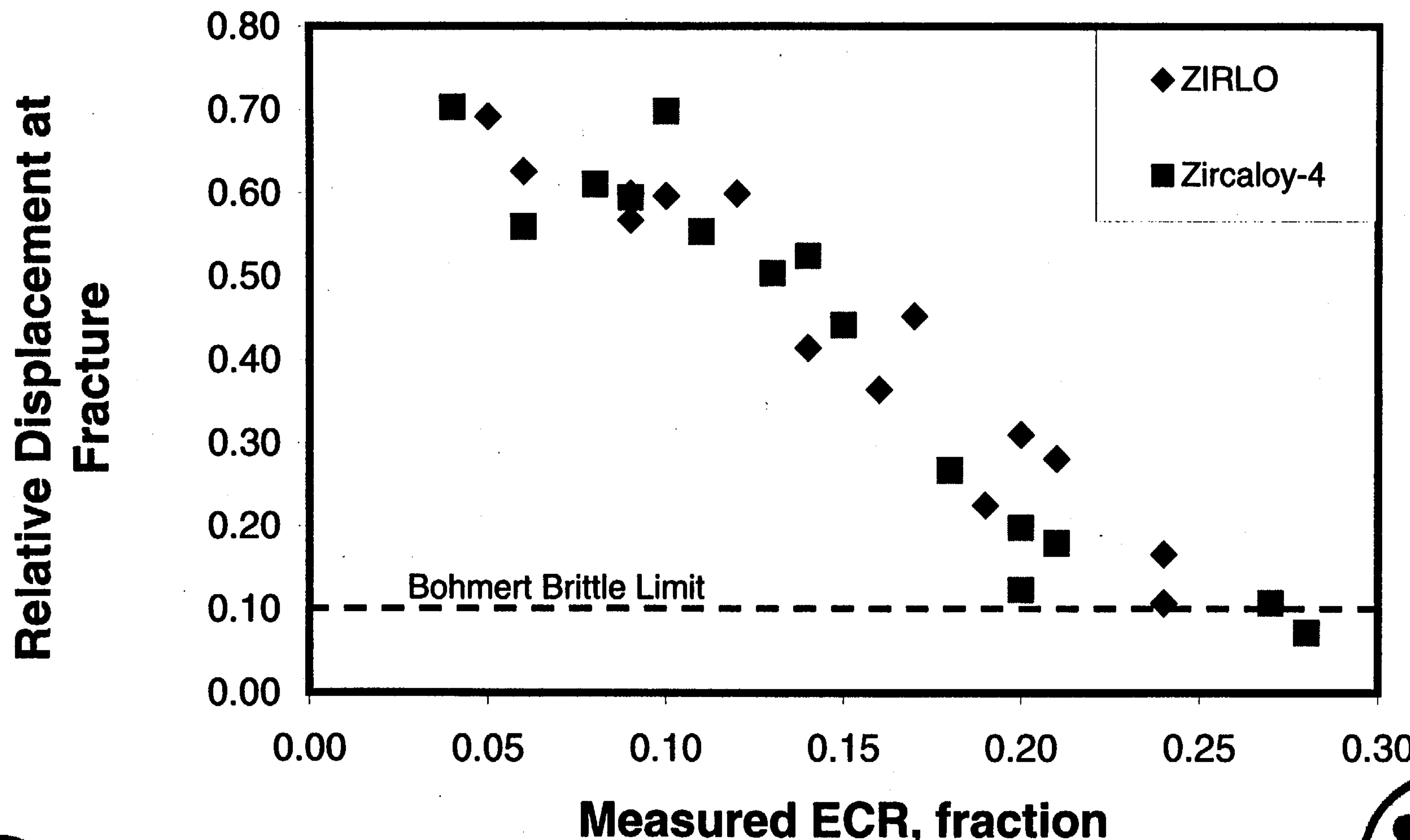
* conservative value

COMPARISON WITH BÖHMERT'S RESULTS AT 1100°C



March 22, 2001
Slide #6

Relative Displacement at Fracture vs Measured ECR at a Temperature of 275F (PRELIMINARY)



BNFL

CONCLUDING OBSERVATION

We need to understand the effect of small materials differences so we don't have to repeat all tests every time the manufacturer makes a small change in the alloy.

G. Hache
(more or less)

Comparison of M5 and E110 Composition
(Both are recrystallized)

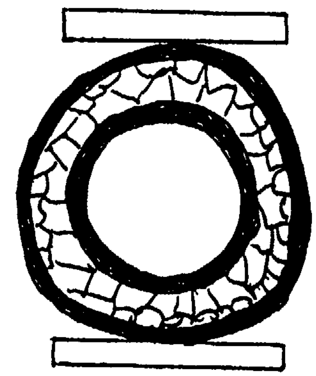
Element	M5 Composition wt%	E110 Composition wt%
Zr	~99 (balance)	~99 (balance)
Nb	0.95 (0.80-1.20)*	0.9-1.1 (0.95-1.05)**
O	0.114 (0.09-0.18)*	<0.1 (0.05-0.07)**
Fe	0.054 (0.015-0.060)*	<0.05 (0.006-0.012)**
Cr	0.0029	<0.02
Si	<0.003	<0.02
C	0.0026	<0.02
Ni	<0.005	<0.02

Data from Halden report HWR-636
* ASTM Toronto, p. 506
** ASTM Garmisch-Partenkirchen, p. 787

Background

Post-Quench Ductility

- Key rationale for LOCA embrittlement criteria--1204°C (2200°F) PCT and 17% ECR limits:
  - **avoid zero-ductility in cladding**
  - ensure coolable core geometry



- Primarily based on Hobson's test 1972-73:
  - Zircaloy-4 tube oxidized at 1100-1315°C on two sides
  - short ring cut, compressed 3.8 mm slowly
  - crack-free adherent oxide, **H uptake low, <150 wppm**
  - **reflects O-induced embrittlement only**
  - **H-induced ductility degradation negligible--unknown in 1973.**

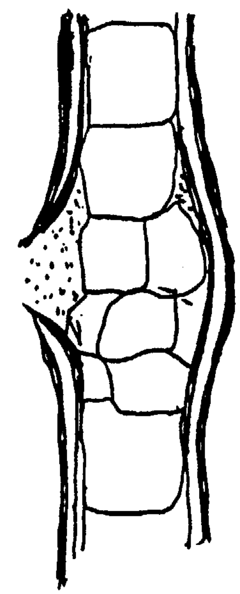
Background (Continued)
Post-Quench Ductility

- H-induced embrittlement at H contents higher than about 600-700 wppm:
  - observed in 1980-1983, ANL & JAERI
  - local regions near burst opening, Zircaloy-4 tube
  - H alone (low O in beta layer) not much deleterious

- Significant effect of H uptake in E110 Zr-1Nb:
  - Boehmert 1992, Griger et al. 1999
  - at H contents higher than about 150-200 wppm

- Effects of 4 factors appear inseparable:
  - oxidation (before and during LOCA transient)
  - H uptake (larger than a threshold amount)
  - high burnup
  - Nb addition (E110, M5, Zirlo, Alloy A)

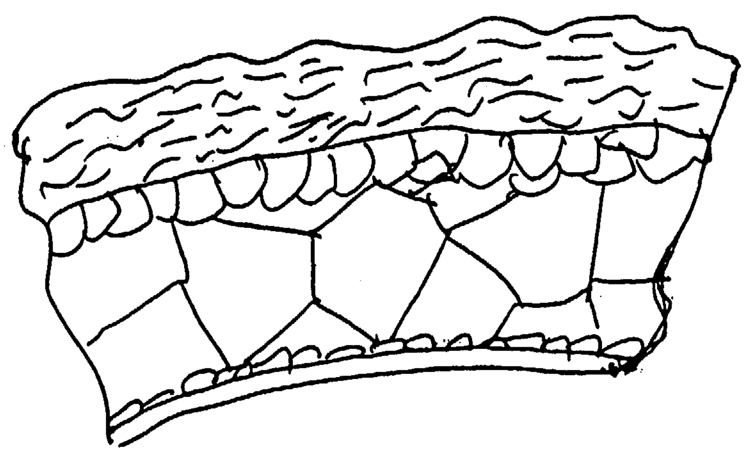
3 Routes for Large Hydrogen Uptake



#1 During normal operation to high burnup (≈62 MWd/kgU)
- standard Zircaloy-4 up to ≈700-800 wppm
- low-Sn Zircaloy-4, Zirlo
- M5

#2 Through "unprotected" ID surface near burst opening

#3 Through "high-temperature breakaway" oxides on the OD surface
- H uptake through normal high-temperature oxide (crack-free, tetragonal, protective) is limited to <150 wppm.



SCOPE OF WORK ON HIGH-BURNUP ISSUES AT ARGONNE
(PIRT Adjusted, EPRI Cooperation)

- Testing in Current ANL Program for Zircaloy-2 and Zircaloy-4 (Target 2003)
  - Integral Test (Ballooning, Rupture, Oxidation, Quench — with Fuel)
  - Oxidation
  - Thermal Shock (to be determined)
  - Phase Relations
  - Mechanical Properties (including Post-Quench Ductility)
  - Post-LOCA Seismic Loading
  - Fuel Relocation (limited to Observation during Integral Test)

- NRC is Interested in Conducting Confirmatory Tests on ZIRLO and M5

- May only need Subset of Tests for Other Cladding Types like ZIRLO and M5
  - Oxidation
  - Thermal Shock (to be determined)
  - Phase Relations
  - Mechanical Properties (including Post-Quench Ductility)

PROPOSED WORK ON UNIRRADIATED ZIRLO AND M5
(Target 2001)

- Review All Test Methods to Determine Test Conditions (Zircaloy Specimens first)
- Agreement on Test Conditions will involve EPRI, Westinghouse, and Framatome
- Post-Quench Standard Test (perhaps Axial Tensile Test) on Unirradiated Cladding
- Post-Quench Ring-Compression Tests (probably also) on Unirradiated Cladding
- Oxidation Rate and Phase Relations as needed to interpret Ductility Results
- No Mechanical Properties or Other Testing at this Time (later in High Burnup Program)
- Proprietary Treatment of Data may be arranged if Requested

PROPOSED COOPERATION

- Pattern after Current ANL Program with EPRI Cooperation
- Westinghouse and Framatome would be Included in all Test Planning
- EPRI is also Interested in further Cooperation (Subject to Approval of RFP)
- Once Agreement is Reached, Start Unirradiated Testing in 2001 and Irradiated Testing in 2003

NRC PROGRAM FOR ADDRESSING EFFECTS OF HIGH BURNUP AND CLADDING ALLOY ON LOCA SAFETY ASSESSMENT

R. Meyer
NRC Office of Nuclear Regulatory Research

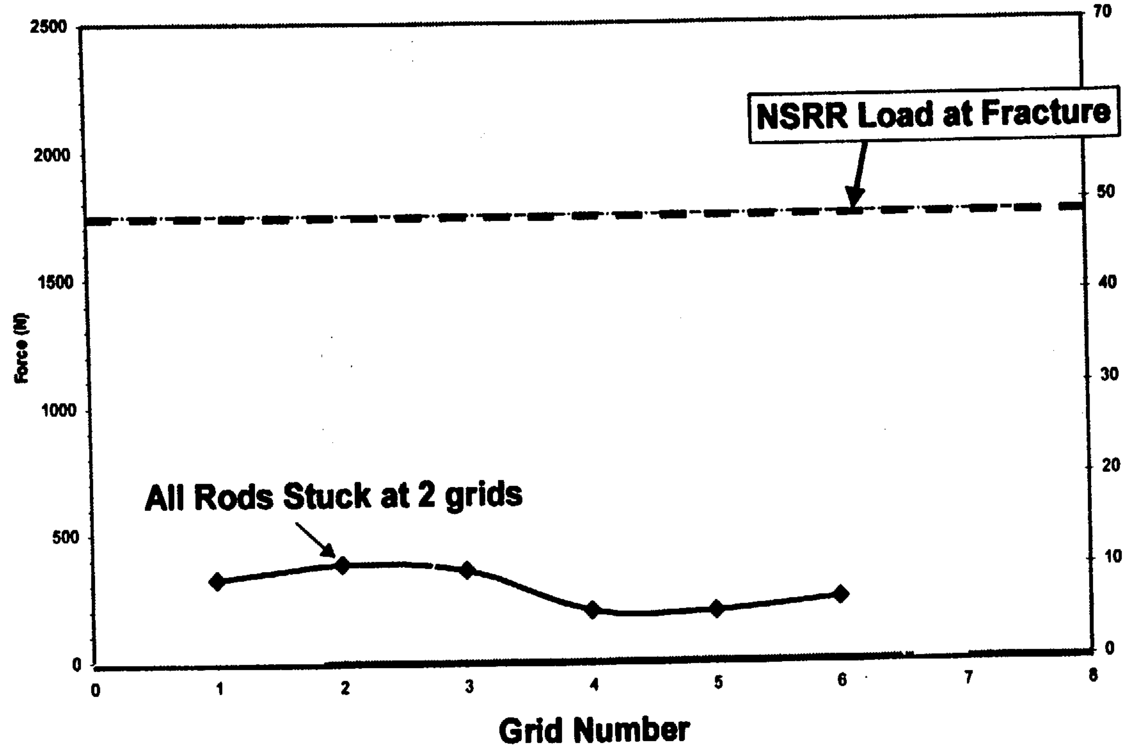
PRESENTATIONS ON AXIAL CONSTRAINT DURING QUENCHING

1. H. Uetsuka (JAERI, Japan)

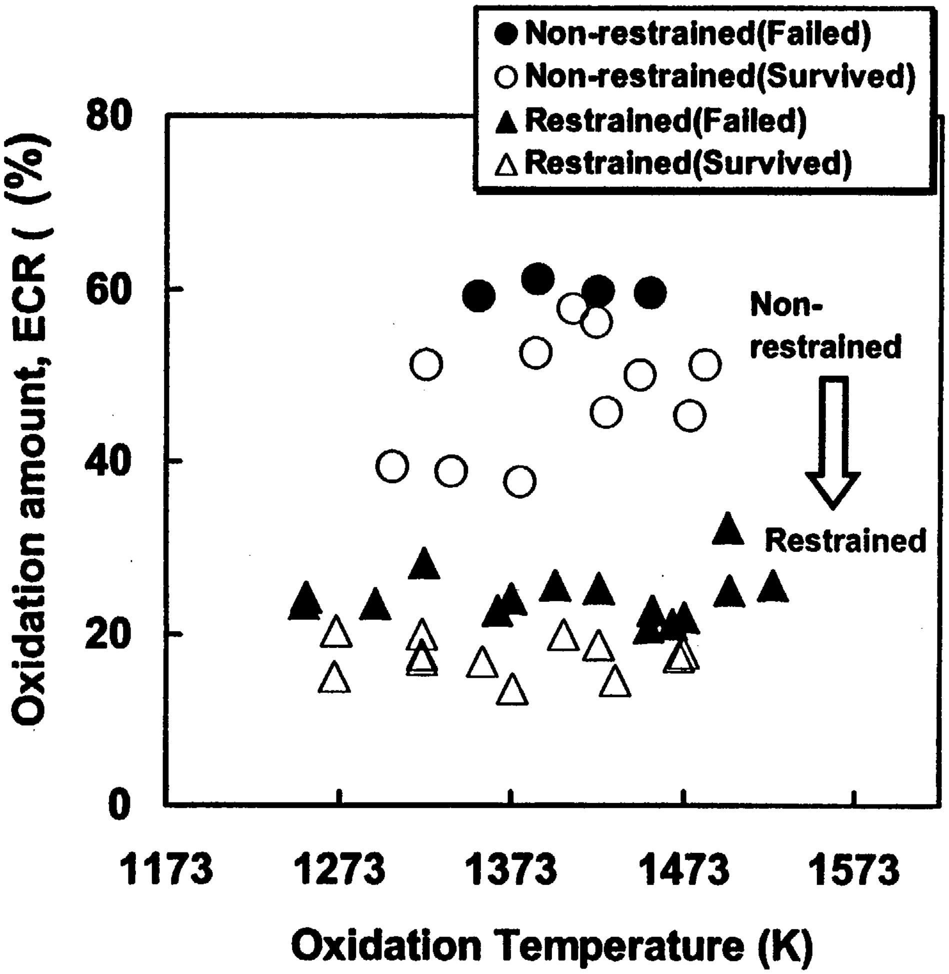
2. N. Waeckel (EPRI-EdF, USA-France)

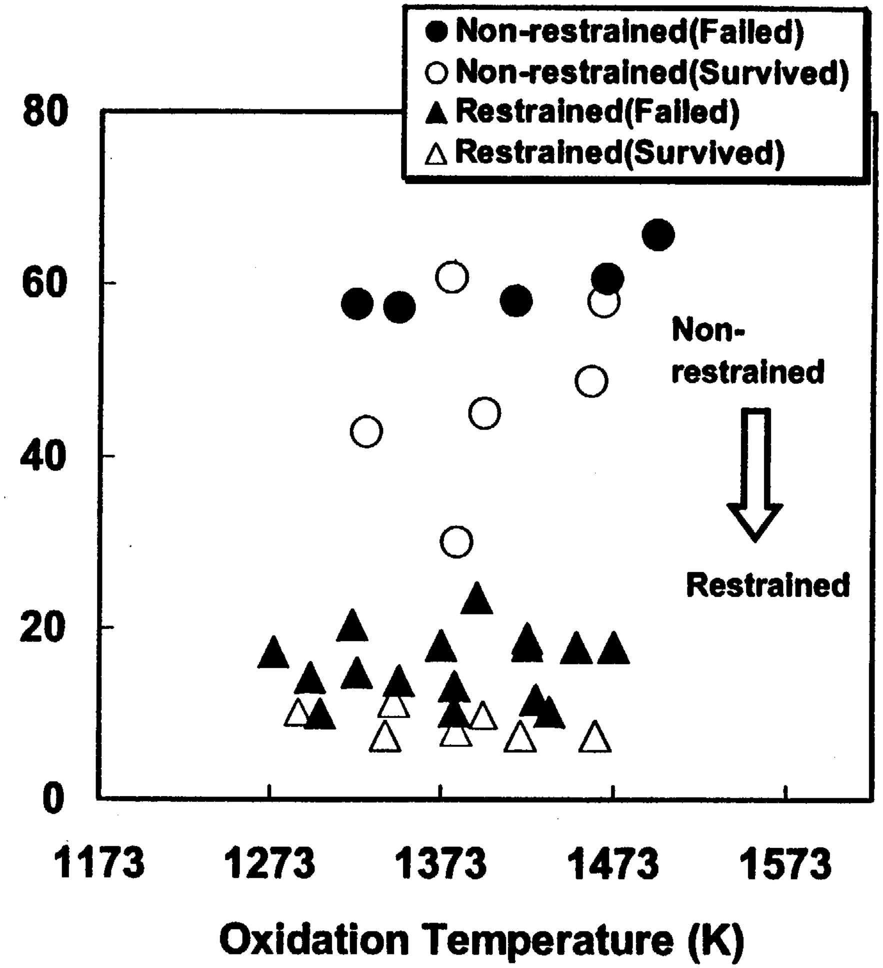
EPRI EDF

Fuel Rod Axial Force Distribution



Failure map(2/2) -Restraint condition-





JAERI

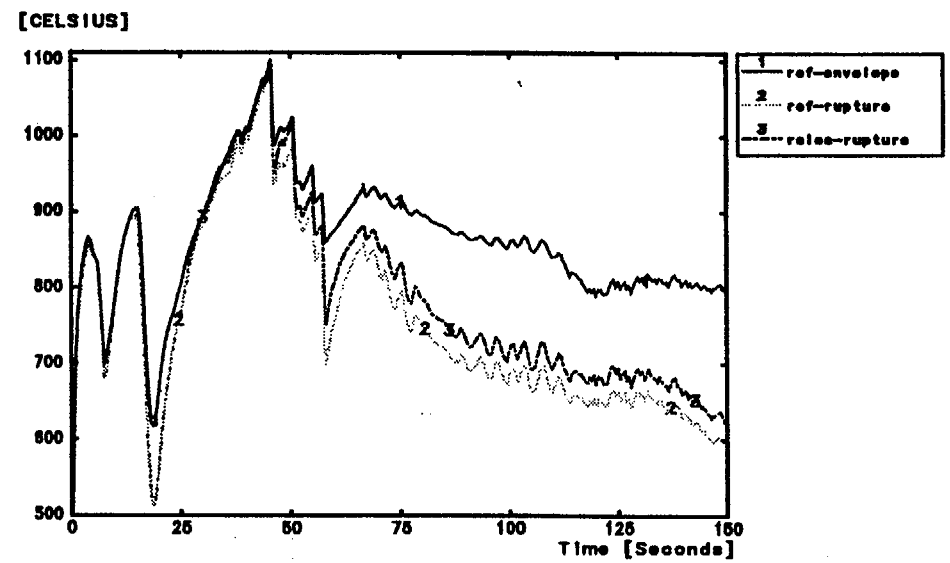
PRESENTATIONS ON FUEL RELOCATION INTO BALLOONS

1. M. Lambert (EdF, France)

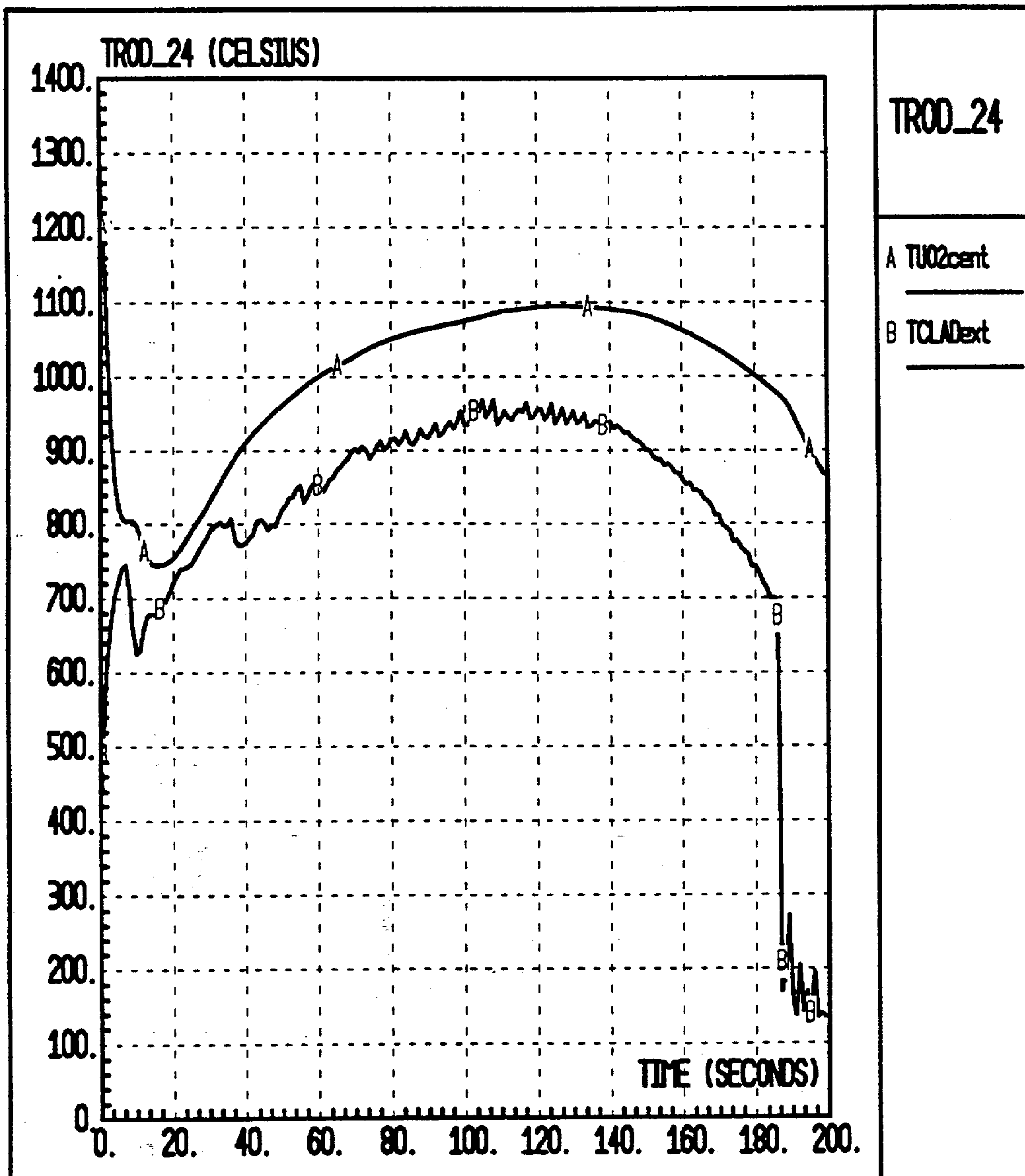
2. C. Grandjean (IPSN, France)

Relocation in Large Break LOCA

Calculation



cladding temperatures

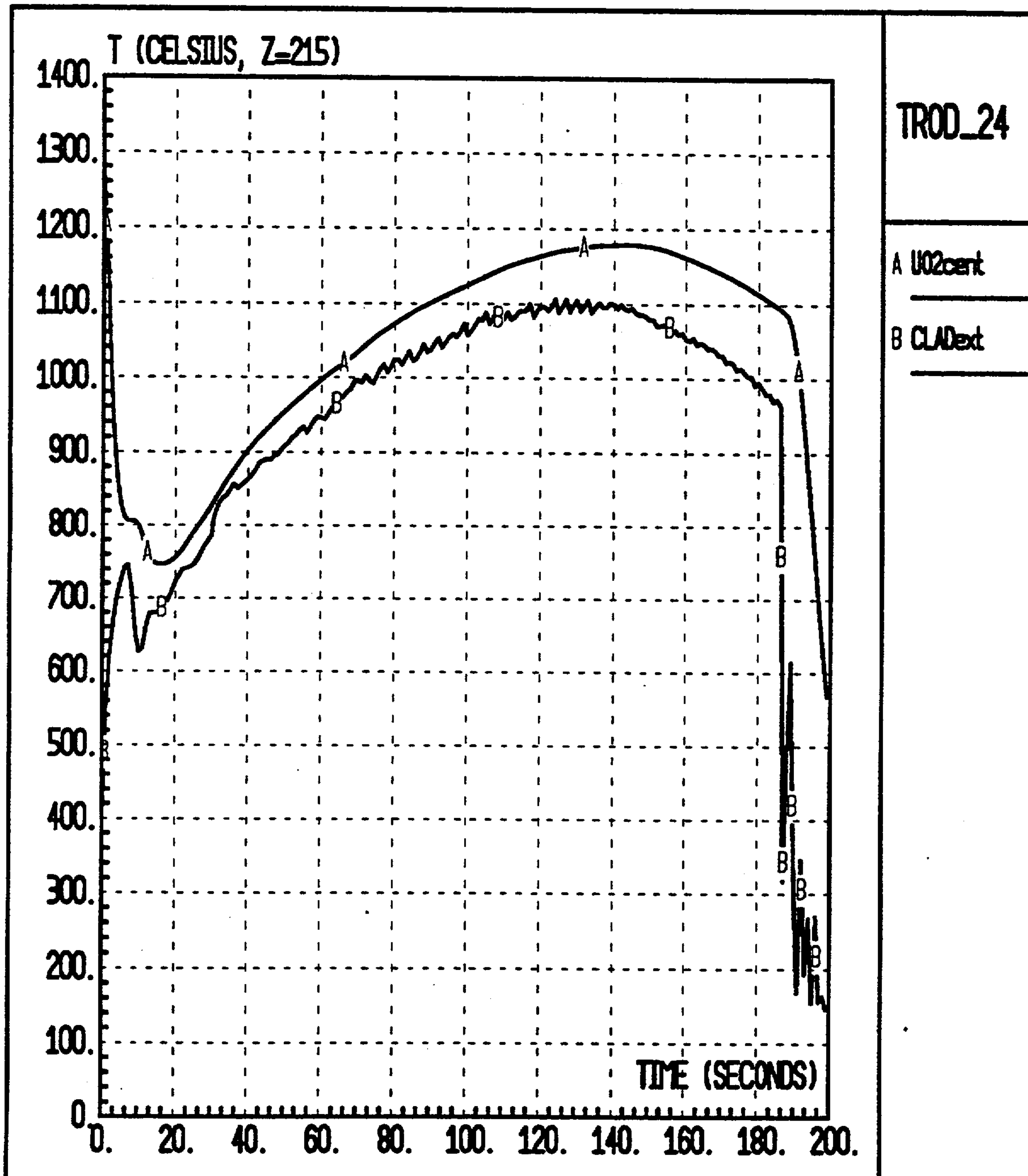


Large Break LOCA. Hot Rod. BU = 57 GWj/tU

No fuel relocation. No protective effect of initial oxide

CATHARE2

V1.3L priv



LB LOCA. Hot Rod. Burnup = 57 GWd/tU

Fuel relocation in rupt. mesh : 61.5% of balloon volume

CATHARE2

V1.3L priv

United States Nuclear Regulatory Commission

RESEARCH ACTIVITIES AND THE HIGH BURNUP PIRTS
(**P**henomenon **I**dentification and **R**anking **T**ables)

Ralph Meyer
Office of Nuclear Regulatory Research

ACRS Reactor Fuels Subcommittee
April 4, 2001

FUEL CLADDING ALLOYS

(Main Objective of Alloy Development is to Reduce Corrosion during Operation)

- Zircaloy (Zirconium with 1.2-1.7% Tin)
- Low-Tin Zircaloy
- ZIRLO (Low-Tin Zircaloy with ~1% Niobium added)
- M5 (Zirconium with ~1% Niobium and no Tin)

FUEL-RELATED SAFETY CRITERIA

- Limited Fuel Damage during Postulated Accidents to Ensure Coolable Core Geometry and Avoid Core Melt
  - Criteria for Overpower Events (Reactivity Accidents)
  - Criteria for Undercooling Events (Loss-of-Coolant Accidents)

- Limited Fuel Damage during Dry Storage to Facilitate Removal from Storage
  - Criteria to Avoid Creep Rupture (Normal Storage Conditions)

- All Safety Criteria were developed for Low Burnup Fuel
  - It was thought that Early-life Conditions were more Limiting

- All Safety Criteria were developed for Zircaloy-clad Fuel
  - It was thought that Alloy Improvements for Operation would also be good for Accidents and Storage

STATUS

- Reactor Operation to 62 GWd/t Burnup Approved for Zircaloy, ZIRLO, and M5
- Specific Questions have been raised about Criteria for Accidents
- Confirmatory Data and Assessment of Accident Criteria for Current Burnup Limit (62 GWd/t) to be provided by NRC
- PIRTs were developed to Help Focus Research Programs and Find Methods to Resolve High Burnup Issues
- Data and Assessment for Extended Burnup beyond 62 GWd/t to be provided by Industry
- Dry Storage to 45 GWd/t Approved for Zircaloy-clad Fuel only
- Dry Storage Criteria for Higher Burnups and Other Alloys to be developed by NRC

PIRT SCENARIOS
(based on 1998 Agency Program Plan)

- PWR Rod Ejection Accident (assumed base case: TMI-1, 15x15 fuel, 62 GWd/t peak rod, hot zero power)
- BWR Power Oscillations without Scram (assumed base case: Lasalle-2, 8x8 fuel, 62 GWd/t, 84%power)
- Loss-of-Coolant Accident (no specific plant assumed, Zircaloy-clad fuel, 62 Gwd/t)

PIRT ACTIVITIES

- ~25 Fuel Experts from Industry, Labs, Universities, and Foreign Agencies
- 8 Meetings (total 25 Days) from August 1999 to October 2000
- 3 NUREG/CR Reports with PIRTS and Related Information (final Drafts)
- Staff Report with Interpretations and Suggestions (Draft)
- Web Site with all Reports and Transcripts (www.nrc.gov/RES/pirt)

PIRT FOR PWR ROD-EJECTION ACCIDENT
(280 cal/g Limit in Reg. Guide 1.77)

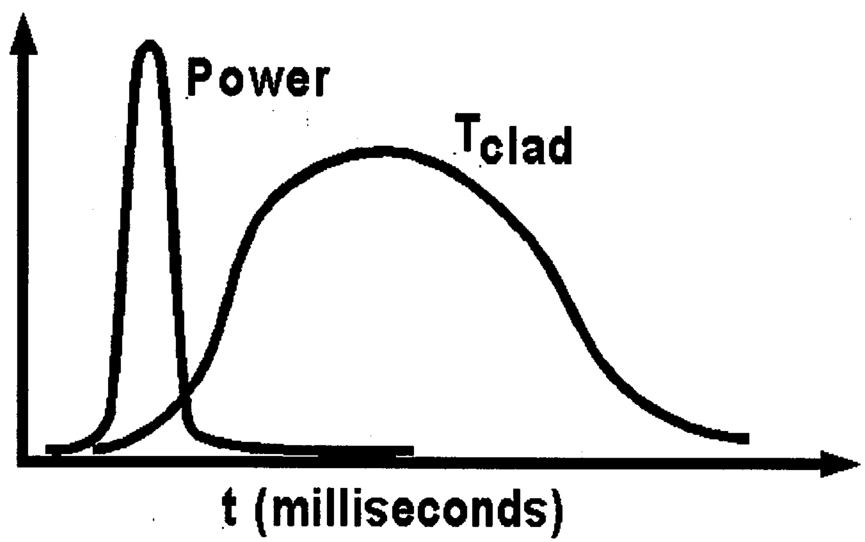


Fig. 1 Qualitative plot of fuel rod power and cladding temperature for a PWR rod-ejection accident

IMPLICATIONS FROM ROD-EJECTION PIRT

- Core Design Changes can Alter the Energy Deposited in the Accident
- Ejected Rod Worth might be used as a Substitute for a Fuel Enthalpy Limit
- Testing in Burnup Range of Interest is Important (Oxidation Phenomena alone will not determine Outcome)
- MOX Rod Testing is Important because of the Pu-rich Agglomerates
- It is Important to Test in Correct Coolant Environment (Water Loop)
- Effect of Different Cladding Alloys not very Important (extrapolate with Mech. Props.)
- High Temperature Ballooning and Rupture might occur for Some Cladding Alloys with high Ductility (i.e., no PCMI Failure)

A METHOD TO RESOLVE ROD-EJECTION ISSUES

- Improve Empirical Correlation with New Data from Cabri and NSRR
- Obtain Mechanical Properties Data for Zircaloy, ZIRLO, and M5 Cladding Alloys
- Use FRAPTRAN Fuel Rod Code to Adjust Correlation for Different Alloys
- Use PARCS 3-D Neutron Kinetics Code for Generic Safety Analysis

Target late 2003 for Confirmatory Resolution at 62 GWd/t using two Cabri Tests (ZIRLO and M5), Initial Tests from NSRR High Temperature Capsule, and Code Analysis.

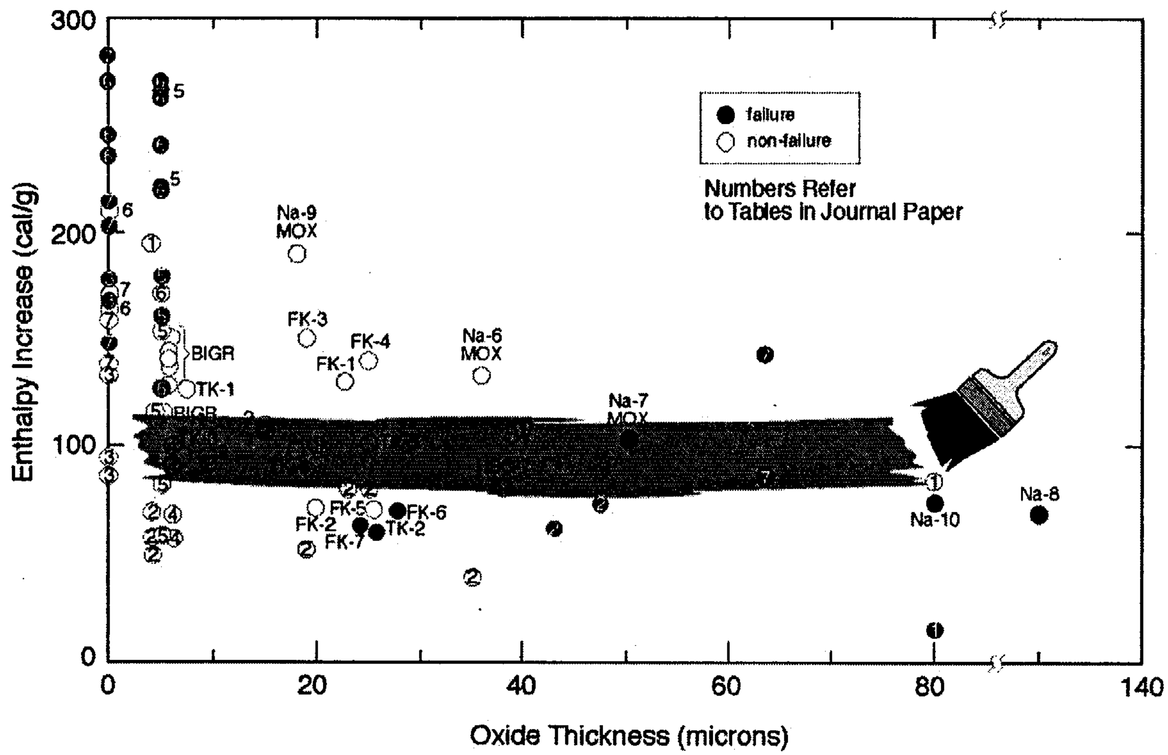


Fig. 2 Fuel enthalpy as a function of oxide thickness for tests described in Ref. 10 (solid symbols indicate cladding failure; open symbols indicate no failure)

PIRT FOR BWR POWER OSCILLATIONS WITHOUT SCRAM
(280 cal/g Limit used by GE)

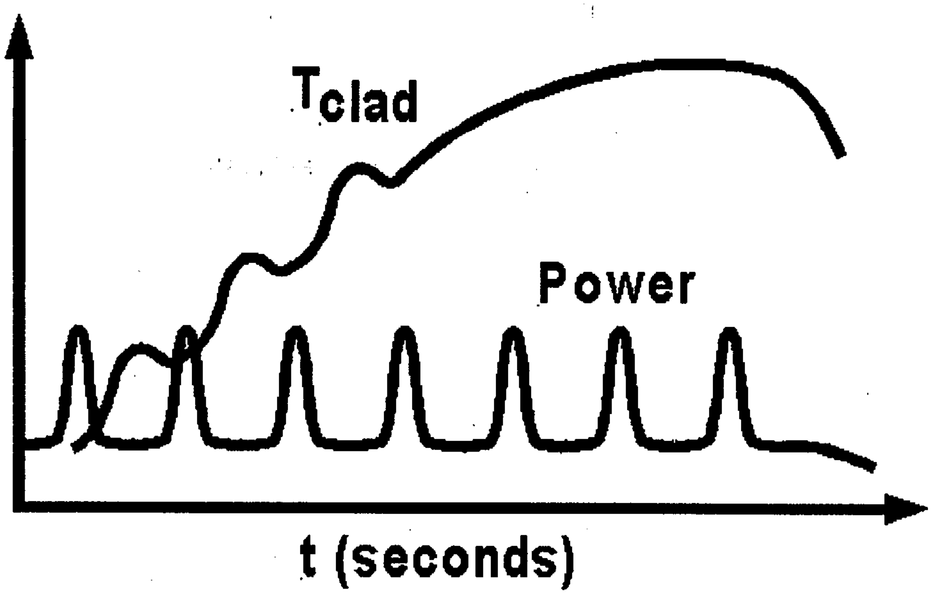


Fig. 3 Qualitative plot of fuel rod power and cladding temperature for BWR power oscillations without scram

IMPLICATIONS FROM POWER-OSCILLATION PIRT

- Pellet-Cladding Mechanical Interaction (PCMI) Cladding Failures are Not Expected
- LOCA-like Oxidation is Expected with possible Ballooning and Rupture
- Cladding Embrittlement will take place at Lower Temperature than Cladding Melting or Fuel Melting
- Runaway Oxidation is Not Expected
- LOCA-like Embrittlement Criteria appear to be Appropriate

A METHOD TO RESOLVE POWER-OSCILLATION ISSUES

- Repeated-Pulse Test Capability in NSRR to address PCMI Failure
- High Temperature Dryout Test Capability in Halden Reactor
- Information from LOCA Work on Embrittlement Criteria
- Generic Calculations with FRAPTRAN-GENFLO (STUK Finland) Hot Channel Code to Compare with Embrittlement Criteria

Target 2004 for Confirmatory Resolution at 62 GWd/t. Depends on Testing that has not been Fully Planned and future Code Developments.

PIRT FOR LOSS-OF-COOLANT ACCIDENTS

(1204°C PCT, 17% ECR Embrittlement Criteria in 10 CFR 50.46)
(Ballooning, Rupture, Oxidation EMs in Appendix K)

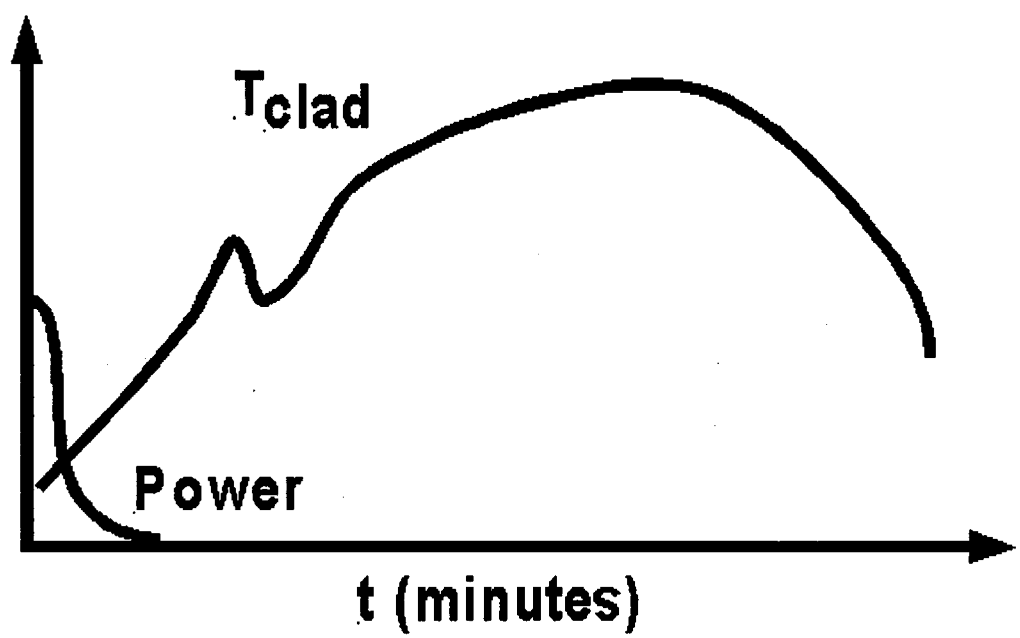


Fig. 4. Qualitative plot of fuel rod power and cladding temperature for a loss-of-coolant accident

IMPLICATIONS FROM LOCA PIRT

- Many Thermal-Hydraulic Models were ranked as Highly Important and Not Well Understood
- A Foreign Member of the PIRT Panel (G. Hache, France) reminded us that NRC's 10 CFR 50.46 Embrittlement Criteria were based on Ring-Compression Ductility Tests rather than Quench Tests
- Cladding Alloy Type was found to be Very Important

A METHOD OF RESOLVING LOCA ISSUES

- Integral Testing at ANL with High Burnup Zircaloy-Clad Fuel (Ballooning, Rupture, Relocation, Oxidation, and Quenching)
- Integral Testing in Halden Reactor with High Burnup Zircaloy-Clad Fuel
- Separate-Effect Testing at ANL with High Burnup Zircaloy-Clad Fuel (Mechanical Properties, Oxidation, Post-Quench Ductility)
- Related Results from JAERI and RRC-Kurchatov Institute
- Limited use of FRAPTRAN Fuel Rod Computer Code for Design and Interpretation of Experiments
- Integral and Separate-Effect Testing of Advanced Cladding (ZIRLO and M5) at ANL

Target Resolution in 2002 for BWR with Zircaloy, 2003 for PWR with Zircaloy, 2004 for PWR with ZIRLO, and 2005 for PWR with M5 depending on Availability of Fuel Rods and Other Factors

"NRC" FUEL RESEARCH

- Argonne Nat'l. Lab. Hot Cells: LOCA-Related Research, Dry Storage Research, and General Mechanical Properties
- Penn. State University: Consulting and Subcontracting to Argonne
- Pacific Northwest Nat'l. Lab.: Fuel Rod Code Development for Steady State and Transients
- Brookhaven Nat'l. Lab.: Reactivity Accident Analysis with 3-D Plant Transient Code
- Halden (Norway) Materials Test Reactor: Steady-State and Transient Properties
- IPSN (France) Cabri Pulse Reactor and Hot Cells: Reactivity Accidents and Mechanical Properties
- JAERI (Japan) NSRR Pulse Reactor: Reactivity Accidents and LOCA-Related Research
- RRC-Kurchatov Institute (Russia) Pulse Reactors and Hot Cells: Reactivity Accidents and General Mechanical Properties

EPRI COOPERATION

- Successfully obtained High Burnup BWR (Limerick) and PWR (H. B. Robinson) Zircaloy-Clad Fuel
- Technical Assistance in Planning Integral and Separate-Effect Tests at ANL
- Expressed Interest in Continuing this Cooperation with NRC in the ANL Program with Advanced Alloys (ZIRLO, M5)