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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Thermal Hydraulic Phenomena Subcommittee

BWR T/H Stability Analyses/ABWR ECCS-LOCA Review

San Francisco Airport Hilton

Terrace Room

San Francisco, California

Thursday, November 9, 1989

The above-entitled proceedings commenced at 8:30 o'clock a.m., pursuant to notice, Carlyle Michelson, committee chairman, presiding.

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P R O C E E D I N G S

MR. CATTON: The meeting will now come to order.

This is a meeting of the Advisory Committee on Reactor Safeguard Subcommittee on Thermal Hydraulic Phenomenon.

I am Ivan Catton, Subcommittee Chairman.

We will continue with the meeting. Yesterday we quit before we got to the last item which was the NRC Research presentation. What I would like is if you could shorten it up a little bit and then maybe tell us about the BWR stability analysis that was done by the Finnish Center for Radiation and Nuclear Safety. And maybe put it in proper perspective with respect to what Wolfgang had talked about yesterday.

So who is going to speak? Harold Scott.

MR. SCOTT: My name is Harold Scott.

There hasn't been too much confusion but I just wanted to tell you that I am going to be using the terms core-wide which are synonymous with global, symmetric, uniform, and I use the term asymmetric. Some other people have used regional, nonuniform, out of phase in terms of the types and modes of oscillation that you get.

I'll also be using the word HIPA, and as Wolf told you yesterday we really mean the whole plant analyzer simulation tool.

[Slide]

MR. SCOTT: These are the objectives of our program.

We received a user/need letter last year from NRR, and so the first item is an attempt to address their questions about how high can the flux oscillations become. What are the ultimate limits on the neutron flux oscillations. And another objective as we heard yesterday the FRIGG assessment and comparisons to the LaSalle data.

Particularly we're interested in the ATWS event. There's lot of evidence that there may be cases on different ATWS scenarios will oscillations would either continue or could be begin and we need to find out under what conditions they begin, and does this affect what the operator sees or does or the emergency procedure guidelines. Because when we get all done NRR is going to use these results for looking at each utilities proposed solution or there may be changes in the procedures for ATWS.

[Slide]

MR. SCOTT: The next little chart here is what we call the "code use diagram." I would like to go over particularly when we expect to have results, because I think you saw from the presentations yesterday all the details about these codes and what their capabilities are.

We would expect to finish in November and December here these calculations, and then a draft report would be available in February for TRAC and June for RAMONA. Then, as I said, we want to concentrate on ATWS; we will be doing those probably in December and January. We'll pick a scenario so that we will be working in January and February on the sensitivity. And our plan is to have some draft reports available in March and June.

We have already completed some comparisons between HIPA and LAPUR.

Are you going to show any of that, Jose? Okay, maybe it was in Wolf's slides yesterday.

We may come back to this. I have handed out a separate copy of this if you need to refer to it as we go along.

MR. CATTON: So what's going to be due in March, which of the reports?

MR. SCOTT: The ATWS sensitivity study.

MR. CATTON: Okay.

MR. SCOTT: This is our schedule. Now, if the contractors tell us they can't possibly meet it, then it will be after that. But that's the schedule we believe we can meet, and we sort of need to meet it because we want to finish this program in the summer and the Commission is going to be anxious if two years after the event we haven't got very many answers.

MR. CATTON: Understandably so.

What are you going to deliver in June?

MR. SCOTT: June will be draft reports for these first three items.

MR. CATTON: I got you.

[Slide]

MR. SCOTT: For these amplitude of oscillations. How big can they really get? If there are asymmetric oscillations, can you get some LPRMs that will be giving you 300 percent and not have very high APRM readings? And RAMONA is the only one that can really do that.

[Slide]

MR. SCOTT: Let me talk now about the status of the items that we completed last year.

The Technical Program Group was formed. We've had four meetings and we probably will have another meeting in January. As you saw yesterday, the LaSalle event was simulated with HIPA and RAMONA and in another month here we expect the TRAC simulation.

We've talked a lot about the noting sensitivity studies that were done. Wolf mentioned that we finished the flow reversal and also the drift flux change for SLIP and RAMONA.

We have finished the steady state assessment with both RAMONA and with TRAC, the F-1. And shortly we'll have finished the transients where we did these gain and phase angle studies.

The model is ready at Idaho to run the TRAC calculations and the plant analyzer had made some sensitivity studies to the parameters. And as Wolf showed you yesterday, we have determined with RAMONA some of the cases for when you get core-wide and asymmetric. And Jose March-Leuba has also developed a number of these using LAPUR code in conjunction with his separate model that can calculate the amplitudes.

[Slide]

MR. SCOTT: I'll show you some specific results. Glen Watford mentioned yesterday that just before the reactor scrammed the operators were attempting to restart the recirculation pumps, so the NRC Office of Analysis and Evaluation Operational Data, after the event, had said, gee, would you get a positive reactivity spike if this flow suddenly came on? So the plant analyzer calculation showed that you could get a spike that would trip the reactor but it would not cause any fuel damage or blowing transition.

You could also, if you were careful in adjusting the valve properly or starting up the pump, you could suppress the oscillations without actually getting this large spike that scrammed the reactor.

[Slide]

MR. SCOTT: And as was mentioned yesterday, the LaSalle conditions as calculated lead to the reactor scram and you need all three of these. Obviously, this occurs with whatever probability occurs. And as GE has told us now, this feedwater temperature reduction is sort of a normal situation.

So it's really this axial and radial peaking that if you don't put that in as it was in LaSalle you don't get the instability.

[Slide]

MR. SCOTT: One of the reasons we worked on this was because, if you recall, in the GE original analysis they used sort of idealized power input to the TRAC code from Carousal studies that gave them the flux shapes and we wanted to really see exactly what those flux shapes really looked like.

And as we have said here, the LP arms begin to pick up these various oscillations as it begins.

[Slide]

MR. SCOTT: And with the code we get to asymmetric oscillations that we said side-to-side which we call azimuthal and so-called inside/outside center periphery which we refer to as radial. So as it shows here and RAMONA and March-Leuba have shown that it's rather complex to find out exactly when you're going to get asymmetric and when you're going to get the center outside type; it depends on these parameters particularly.

MR. CATTON: Will a code like RAMONA calculate the right instability shape?

MR. SCOTT: Well, you give it a set of conditions and it calculates a mode, a shape. Since we don't really have too much data we don't know.

MR. CATTON: I guess you have LaSalle?

MR. SCOTT: Yes.

MR. CATTON: Does it calculate the right mode for LaSalle?

MR. SCOTT: It calculates core-wide. Given that, now remember we didn't have an exact representation of LaSalle.

MR. CATTON: I understand.

MR. SCOTT: But as close as we could get we got core-wide. And with slight changes in the conditions we can get other modes of oscillation that are asymmetric. But we did get core-wide with RAMONA.

[Slide]

MR. SCOTT: Let me now show you the results that we're expecting now in this year.

These calculations now will be done with the so-called fixed RAMONA. Wolf talked about this limit of 200 neutron nodes, but we can divide the core up into 191 nodes and then use a super cell or four by four bundles as one -- two by two bundles as one neutron node.

We will be issuing a report here, probably have a draft in the next few months. As I've said, we're going to be using three codes here. Where did my little teeny vugraph go.

The TRAC; we've already seen the HIPA results for this one. Then as I said before, our big important item is the ATWS scenario.

EPRI, you know, has been working in this area, too, to try to find out under what conditions oscillations occur or suppress, particularly when you have lowered water level. This is the calculations we really haven't done yet as to let the water level go down as the emergency procedure guidelines call for.

Then to sort of wrap up we like to have a final report documenting all these results. We'll also try to include any information we get from foreigners. As you mentioned, Finland has been doing some work. And we've talked with Sweden, they're going to do some work with the scan power version of RAMONA. Jerry is doing work with Retran. So whatever results are available we'll try to get those all into a comprehensive report.

MR. CATTON: They will be included in the final report you just mentioned?

MR. SCOTT: If we have them available to us and we think we will, at least from Sweden and other. If EPRI has published something and will show us what they've done, then we can include that information.

The idea was, it was going to be more than just an NRC report, if we could.

MR. CATTON: With all these divergent results I would hope so.

MR. SCOTT: That's one of the ideas, was to try to say, look, here is what we think is the best answer given various codes we're using.

[Slide]

MR. SCOTT: So this is our schedule to complete. And as we said, we don't really have any plans for future work. This is one project we would like to show to NRR that we can actually get them an answer on a schedule that we originally agreed to and, of course, our management would like us to finish a program on the budget that we originally set. So that's it.

MR. CATTON: Thank you.

MR. WARD: Harold, the ACRS wrote a letter on this subject, I don't have the --

MR. SCOTT: June 14th, I think, yes.

MR. WARD: Yes, earlier this year. And the burden, I guess, of advice we were given in that letter was that we thought the NRC should concentrate its attention to worrying about the relationship of BWR instability to ATWS scenarios; we believe that's where the public health risk was.

And we thought that the concern about peak amplitudes, you know, the magnitude of peak amplitudes which might be related only to field damage and not necessarily -- I guess if I owned a nuclear power plant I would sure want GE to tell me whether it was likely I was going to get field damage and how to avoid it in situations that could lead to instability.

But most of the NRC's attention should be directed toward scenarios that might be rare, but might really involve, you know, some public heath risk.

I'm not sure the program is really prioritized in that way in emphasizing. And I realize these aren't completely separable questions, you know, to understand the dynamics of instability here. You're going to learn about both, you know, peak power and total integrated power, I guess. But you can't learn everything about everything in the next few months.

It's not clear to me that the program really emphasizes the ATWS part of it. Do you think it does?

MR. SCOTT: I think it does. Maybe if you -- at the back of the package we gave you the statements of work for both laboratories and I think we have emphasized in there that this Fiscal Year ATWS is the big item. The TPG will be focusing on what ranges of the parameters we want to use in an ATWS study; and that's the scenario we're going to use, an ATWS scenario.

As you recall the questions were, under some ATWS scenarios steam is going to the suppression pool. And if, in fact, and we believe this is quite true now, that if you have large oscillations that increases the core average power. So if you're assuming that, say, the power was at 18 percent, therefore you would get so much steam in the suppression pool and it would heat up at a certain rate. But if those oscillations -- if there are oscillations and they cause that 18 percent power to go up to 23 or 24 percent power that could make possibly some substantial difference in the rate of heatup of the suppression pool.

It may turn out that under those conditions where the MSIV is closed or the bypass is relieving some steam to the suppression pool that we can't really find oscillations under reasonable conditions.

But the intent of the program is to sort of search around particularly with LAPUR code and HIPA which can bang off, as you were told yesterday, just lots of calculations every day and look for these. This is what EPRI is doing, they're looking around to see if you can find the.

The report that GE did for EPRI about a year ago or two years ago came out, the MP5562 indicated that under many scenarios they were close to the instability boundary. And GE's proprietary report from 10 years ago showed cases where there were oscillations. And as you have seen it's quite sensitive, maybe just a little change in the parameter will suddenly give you much larger oscillations; that's what we're looking for.

We are doing other things, too. But I believe we're focusing in on this ATWS question.

MR. CATTON: Larry?

MR. PHILLIPS: Larry Phillips.

I will address that in the NRR presentation later, but just briefly, our research program is really directed almost entirely to the ATWS question. We feel the other questions we have to address, too, because they involve regulatory problems. We do have to meet the regulations.

But we feel that that's pretty well in hand with the Owners Group work and we'll address that.

MR. CATTON: On the other hand, if you can't predict the limit cycles for the more benign circumstances you won't be able to believe them for the ATWS. So you almost have to do them first. It's the only place you have any experimental data.

MR. LEE: I would just add another comment perhaps. In my opinion, the test has been recognized for some time that the instability and the magnitude of potential limit cycle oscillations are very much subject to small variations in each of the conditions. And hence, it will be very difficult to predict with certainty how large the limiting amplitude of oscillations would be. And I have not seen much of an effort in trying to somehow make the boiling water reactor system a little bit more -- a little less susceptible to this kind of instability mode.

I would like to see both vendors -- vendor and NRC look at this problem in that angle. For example, we know very well that the void fraction reactivity, if it can be reduced somewhat in magnitude could make the system less susceptible to oscillation.

And if I understand correctly, again, the foremost coefficients near the exit of the boiling channel, if you can reduce the magnitude somewhat so that you can loosen up the flow oscillation and let the flow relax a little bit more, then you can reduce the susceptibility to this type of oscillation substantially.

These are not the things that you can do next year or this year. But in the long-term these are the things that we need to look at.

I remember after LaSalle event talking to a number of people and some very well versed in some of the reactor problems were genuinely concerned about. I remember talking with Hans Bader about six months ago and he was very much concerned about this particular incident as I recall.

So these are the direction perhaps we need to look at a little bit at the same time.

MR. CATTON: Harold, are you finished?

MR. SCOTT: Yes.

MR. CATTON: Lou, is there anything additional?

I would like to hear about the Finnish work and anything else you might want to tell us.

MR. SHOTKIN: Yesterday you heard Wolfgang Wulff present his HIPA results where he showed that under conditions where there was no scram and you did have an instability he could get ratios of power peaks that were, I don't know, maybe

15, 20 times the initial level. And we just received recently something from Finnish Research Center by a Mr. Valtona where they calculated the TVO event that occurred in their plant in 1987 and they did sensitivity studies using a different code, something they called TRAB, T-R-A-B.

And they also found under ATWS type conditions where, of course, you needed the cold feedwater coming in to introduce reactivity. They also got ratios of power about 20 times the initial level.

I'm not saying that either HIPA or TRAB are correct, but we have -- there's two independent analyses done that show that under ATWS conditions where you have the additional insertion of reactivity from cold feedwater that comes about through some other mistake, that you can introduce large amounts of reactivity. And, in fact, the reactivity that is in this Finnish report shows that after awhile they did go prompt critical, you know, for a very short period of time.

The code that they used, TRAB, was based on RAMONA. And, in fact, most of their calculations ended when they got a flow reversal at the inlet of the core, so we know that was a problem with RAMONA and they have evidently the same problem with their TRAB code.

So we have given you a copy of this. This is a draft report. We understand -- I don't think it has been published yet. It's just to look at for your own edification.

MR. CATTON: Are there any questions regarding the Finnish work?

MR. MICHELSON: While he is up there is this a good time to discuss briefly the advanced boiling water reactor relative to this problem?

MR. CATTON: Certainly.

MR. MICHELSON: What steps are the Staff -- what is the Staff doing differently for the case of the ABWR? A lot of the work clearly applies, but ABWR is a new project, a clean piece of paper to be certified for a long period of time and is worthy of perhaps considerations that you couldn't apply to the present day plants.

MR. SHOTKIN: By ABWR you mean the 1300 megawatts rather than the 600 megawatts?

MR. MICHELSON: Yes, the 13.

MR. SHOTKIN: I can't talk at all about what's going on, on the 1300 megawatt. As I understand it the plant is very similar to the existing boiling water reactors. They've made certain improvements.

MR. MICHELSON: Well, it's a somewhat different core?

MR. SHOTKIN: Yes.

MR. MICHELSON: Well, that's where the changes perhaps could come from.

MR. SHOTKIN: Maybe Larry Phillips from NRR can help answer that.

MR. PHILLIPS: Yes, we did consider that in our review of ABWR and have some questions on it. Basically, for one thing we now have more insight into the stability sensitivities and the core and fuel designs will be done a little better with respect to instability.

The ABWR, of course, has internal recirculation pumps. And they have a feature whereby when two pumps are out there will be a select rod insert. There will be a region of high power, low flow which is automatically excluded from operation; that's the primary improvement.

As I will address later, there's actually proposals on existing reactors which are similar.

MR. MICHELSON: So you are giving it some amount of consideration and maybe at a later time when we get that particular module for final review someone from NRR could come in and tell us a little more just what was finally put in as a requirement to help alleviate this problem.

MR. PHILLIPS: Yes.

MR. MICHELSON: Thank you.

MR. WARD: Would it be appropriate to ask General Electric to comment on the question Carl just asked?

MR. CATTON: I think so.

MR. MICHELSON: I think it would be nice.

MR. WARD: Yes.

MR. CATTON: Maybe you could also comment on these high power ratios that have been calculated by RAMONA and the Finnish code.

MR. SAWYEk: I'm Craig Sawyer from General Electric Company and I have some comments.

We responded to the Staff on one of the questions that was asked on the ABWR review which the ACRS should also have a copy of. Let me spend about five minutes, if you will indulge me, to basically summarize what the response to the NRC's questions were.

[Slide]

MR. SAWYER: The thrust of the questions were: what have we done in the ABWR design to prevent or limit the possibility of limit cycle oscillations? And what have we done in the mitigation area that should they occur against our best efforts, what are we doing about that?

So what I've got basically is a couple of pages to speak to that.

MR. MICHELSON: We don't have copies of this so we have to read it.

MR. SAWYER: I didn't bring extra copies with me, but I can certainly make them available to you; I can send them to you.

What we've done is, we have tightened the inlet orificing and the quantification as we actually doubled the inlet loss of coefficient and that's the single phase area, so that goes in the direction of improving stability.

We opened up the control rod pitch a little bit and what that does is help reduce the void coefficient. And as you probably know, more negative void coefficients make the stability problem a little bit worse.

We have more steam separators, therefore the steam separator bring in the power, so the steam separator pressure drop has been decreased somewhat, and that's in the two-phase region; that also goes in the good direction.

In terms of prevention, the kind of thing that transpired at LaSalle where a single instrument being worked on by an instrument tech was basically the original cause of a pump trip. That can't happen in the ABWR because we have two out of four logic for all of our trip activities. So there is no way that a single manipulation such as took place at LaSalle could initiate an activity such as a pump trip which would tend to get you into the region where you don't want to be.

The recirc pumps themselves are on multiple power supplies and they are grouped in a 23, 23; there are 10 pumps. So that we've supplied the Staff with failure modes and effects analysis for review, but the conclusion of that is that the probability of having all the pumps trip is in the accident range, you know, about the same as a large break LOCA, really. So that the chances during normal plant operation of finding yourself without sufficient number of recirculation pumps running to provide a minimum core flow to keep us out of the excluded region, which Mr. Phillips mentioned is very low.

We are going to implement automatic control logic for power changes to automate major blocks of plant operation. And basically, by doing that we minimize the potential for operator error in trying to drive the plant on power ascension potentially into the wrong region of the power flow map.

We have rod blocks on power ascension to prevent ascension into the wrong area of the power flow map. And there's a minimum pump speed logic that is in both the control system and at the pump itself. So that even if the control system were to demand less than minimum pump speed the pump would refuse to accept that demand and run at minimum speed anyway.

MR. LEE: Mr. Sawyer.

MR. SAWYER: Yes.

MR. LEE: May I ask a question about what you have there.

MR. SAWYER: Please.

MR. LEE: Between these 23 modifications in the design that you're talking about, how much of a difference, for example, in void coefficent reactivity do you anticipate, in the control rod pitch increase and inlet low increase?

MR. SAWYER: Well, let me say it two ways: the void coefficient itself has been reduced in the order of 15 percent relative to what it would have been if we had kept the 12 inch pitch, okay, for the same fuel.

The overall impact of all of these changes for the decay ratio at minimum pump speed line on the rod line relative to not having made any of these changes was calculated by us to be about .4. So a significant tightening of our stability requirements for design.

Does that answer your question?

MR. LEE: Right.

The last question, I thought it played less of a role perhaps than the exit coefficient; am I correct?

MR. SAWYER: Well, they're both important. I don't recall -- if you're interested you can give me a call. We've actually done studies where we've broken down the separate effects of each one of these. But typically, I don't remember anymore because we did the study several years ago. But each one of those changes by itself is worth the order of .1 to .2 in improvement in decay ratio.

MR. LEE: So the steam separator improves the --

MR. SAWYER: Decay ratio, also.

MR. LEE: -- also.

MR. SAWYER: That one is not worth quite as much as the others, but it's order of magnitude about right of what I said.

MR. LEE: Is there anything you can do along this line for the current generation of operating reactors over the next few years?

MR. SAWYER: I don't think so.

Dick, would you like to comment on that?

MR. STIRN: Dick Stirn, General Electric.

I think Larry is going to address it in his presentation. But clearly one of the solutions the Owners Group is now evaluating is excluding region in the power flow map in which it would be susceptible to oscillations and precluding operating in that region. So things like rod block, select rod insert or scram, when the operator enters that region will be under consideration.

But as far as modifying the control rod pitch, things like that cannot be done. Things like changing separator pressure drops cannot be done.

So a lot of the features here are things that you can add to a new plant, but not would not retrofit easily. So the things we are looking at are more into the prevention from entering the region or if you do get into the region you get into oscillations and many type things which I think Larry is going to address in his presentation.

We can retrofit those types of things. We cannot retrofit the major hardware changes that control that.

MR. LEE: But when you go to reload fuel design and have to redesign or refabricate the fuel assembly boxes you might be able to reduce and change around the LOCA descriptions. And then you can also --

MR. STIRN: That is correct. All of our new fuel designs do that. For example, going to low pressure drop, two-phase pressure drop. We're going to a large central water rod to reduce coefficients. We're reducing the pressure drop in the upper type plate and we are also reducing our increase in pressure drop in that fuel type plate. So we are doing those things in new fuel designs. But again, there is just so much you can do with a fixed geometry.

The things I was talking about more were in the area of, we cannot change the plant geometry. Obviously, we can't change the control rod pitch. But we are, and I think is stated in our topical report that we have issued to the NRC, we are maintaining stability margins equal to or better than our past fuel design. That is one of our objectives.

MR. LEE: Thank you.

MR. MICHELSON: Question: in the ABWR how do you measure the unnatural circulation; how do you measure the recirculation flow?

MR. SAWYER: There are two ways that we measure recirculation flow. The most accurate of the two is by measuring pump deck Delta P. Okay.

The other way in which we measure it is --

MR. MICHELSON: With the pumps tripped?

MR. SAWYER: With the pumps tripped what we have to rely on, there still be a pump deck Delta P because the rotors lock. It won't be as accurate, of course, as it is when the pump is running.

We also use core plate Delta P; that's not quite as accurate either. But we frankly have to recognize that when you're down to natural circulation the instrumentation is not as accurate as it is at graded condition.

MR. MICHELSON: An indirect measure is what you're using?

MR. SAWYER: Always. We are using pressure drop as the measurement.

Continuing, just to bring everybody up to speed, this is a page from our SAR submittal where I've added a couple of things here.

This is the what we're calling the excluded region, the region 3 between natural circulation and minimum pump speed, so this is what we're talking about. We're putting in design features such as the rod block that will prevent power ascension unless the flow is greater than minimum pump speed.

And as Larry mentioned, I'm going to talk about that, I have a whole chart on that in a moment. We're taking action so that if more than two pumps have tripped and the flow is detected as being less than the minimum pump speed and the power is greater than given by the 80 percent rod line, then we'll take select rod run-in action.

MR. LEE: Can I ask a quick question?

MR. SAWYER: Yes.

MR. LEE: With your Delta P measurement, how accurate can you predict the recirculation flow rate?

MR. SAWYER: I've forgotten the number. We've done a calculation, but it's somewhere in the 10 to 15 percent range. And we have to account for that between nominal and tech spec limits. So the numbers I'm going to show you here are what I would call the analytical limits; the actual limits which are going to be imposed on the plant are going to be somewhat higher just to make sure that we're covered.

[Slide]

MR. SAWYER: The mitigation, okay, we've alluded to this one and I've got a chart on that in a moment. Flow coastdown scram: one of the mitigation features for the postulated all pump trip event is to have a flow coastdown scram. That is to say that there are sensors that detect the rate of change of core flow and if it exceeds a certain set point then a scram signal is initiated automatically. So that's one feature.

The select rod run-in I'm going to talk about and we've improved the operator interface for LPRM and APRM display monitoring for the operators, so it gives the operator better and clearer information with regard to monitoring the plant.

[Slide]

MR. SAWYER: Let me show you a summary of the logic.

Basically, this is just the monitoring portion of it. The logic portion of it is, if two or more of the pumps are tripped and the power is greater than 30 percent and the flow is less than 36 percent -- this are analytical set points now. The nominal set points are going to have to be somewhat different to account for instrumentation. Then there's a select rod run-in. Of course, there's also manual capability of the operator upon detection.

Now, on power ascensions, if you try to do a power ascension and the power is greater than 25 percent and the flow is less than 36 percent, then you have a rod block which will prevent you from trying to get into that region from below. So that's what the select rod run-in is all about. The rods are selected -- the number of rods are selected by the plant nuclear engineer, basically, for every cycle. And they're selected to make sure that the sufficient worth in those rods to drive the plant below the 80 percent rod line. So that's the basis for the choice of the rods.

MR. CATTON: Thank you.

I believe next on the agenda is Larry Phillips.

MR. PHILLIPS: I am going to discuss the status of the NRR BWR stability review.

[Slide]

MR. PHILLIPS: Now, the regulatory issues that we are faced with are to assure that the automatic protection features and operating procedures will prevent violation of safety limits due to power oscillations. Of course, GDC-12 of the regulations requires this and the primary concern has been asymmetric oscillations which are not protected by scram through the APRM system, and which GE has calculated could violate safety limits. And really, almost the same as the ATWS question, the potential magnitude of such oscillations has not been identified.

The other issue, of course, is ATWS and what we are attempting to do there and what most of this discussion on our research effort has been oriented to is to confirm that existing requirements and procedure guidelines for response to ATWS remain adequate for all potential circumstances of power oscillations associated with ATWS scenarios. And in that respect it really doesn't matter which mode of oscillations we're talking about.

We're concerned with the amplitude of the oscillations, the potential effects on operator response, and the potential damage to the core and suppression pool temperature.

[Slide]

MR. PHILLIPS: The BWR Owners Group proposed resolution with the scram system operable, I will address first. I hadn't realized that this wouldn't be addressed at all to this point. I don't have slides to show some things a little more clearly.

The Owners Group has proposed basically four options, although three options really. Option one is to define a power flow exclusion region for each product line. It's similar to what you saw on the advanced BWR slide. They would provide an automatic control rod insert response to prevent operation in the exclusion region.

In most, this probably will be in the form of a scram for most plants since they can use their current power flow scram design to effect the solution. Rod insert is also part of the proposal.

They would define conditions as an option in this respect, and that's the reason I say it's really -- I call it one option with an option on the option. They would define conditions for bypass of the automatic exclusion actions with continuous surveillance using a stability monitor. This is a noise-based monitor which has been -- one has been developed by Oak Ridge and they are also used fairly extensively in Europe, and it's been tested thoroughly and does a real good job of measuring stability on-line.

MR. MICHELSON: Has the Staff done any kind of a safety evaluation of that device as a supplemental monitor?

MR. PHILLIPS: Yes. We have -- A&F has developed a monitor which is based on the Oak Ridge development and we have reviewed the methods and it's essentially approved. It's currently being reviewed by CRGR, but we expect to issue a safety evaluation report on that very soon.

MR. MICHELSON: Well, that won't be a mandatory device, I guess, for one of the optional ways of handling the problem?

MR. PHILLIPS: Yes. Some of the licensees are planning to propose to bypass when the stability monitor is operable. Now, there's already been one installed at Washington Public Power Supply System. We also are reviewing -- we're also reviewing the implementation of that there. We expect it will become operational in early December. We're looking at the hardware and everything as far as operation goes. They will be doing a rod sequence exchange and be at low power level at that time and they expect to make it operational. It will be a good time to get some measurements in regions where the decay ratio is possibly significant.

MR. MICHELSON: Is this the kind of a feature you would expect to see on the ABWR as well or is it something that isn't needed because of all the other things they might have done?

MR. PHILLIPS: It hasn't been addressed on the ABWR, but I will expect that it will be an option just the same as it is on or is being proposed on current reactors.

MR. CATTON: What is Washington Public Power doing to define the exclusion regions?

MR. PHILLIPS: I'll get into that.

MR. CATTON: Okay.

[Slide]

MR. PHILLIPS: A second option is that for BWR-2 reactors of which there are only two operating: Oyster Creek; and Nine Mile-1. They have quadrant-based flow biased APRM flux scram systems currently. These systems are -- the owners are showing through analyses -- are effective in providing scram protection for both regional and asymmetric oscillations. They expect to show through analyses with a report and they've already done preliminary work and have made some presentations to the Staff. They expect to show that these are sufficient for these particular reactors and no further changes will be required.

The third option is automatic scram action based on specified LPRM signatures. Here again GE has done some scoping studies looking at the sensitivities and the range of -- radial range of sensitivities of LPRMs to local oscillations, and they are looking at a design of this system through selection of the way they would select LPRMs and build them into scram circuitry.

This perhaps is the most radical of the changes being proposed as far as hardware modifications go to current systems. It's probably also the most effective in providing absolute protection on oscillations which would be large enough to potentially violate safety limits.

However, there are some other advantages which I will touch on to option 1.

MR. CATTON: So I guess the analysis becomes relative less important?

MR. PHILLIPS: Well, the analysis -- that's right, except that the analysis, of course, of the -- that's related to the design of the system and that will be submitted and reviewed.

[Slide]

MR. PHILLIPS: The BWROG proposed resolution with respect to ATWS, I think we have two things here. I think we have an official position by the BWROG which maybe is for public consumption, but I don't think they really believe.

The`` have submitted the report, NEDO-31709, which is based on work ..at was completed a year ago where they ran the case of the, I believe it was the LaSalle case with large amplitude oscillations -- there was no scram -- of about 200 percent peak neutron power, and ran them out for some period of time and looked at the increase in average core power.

The report on that was massaged, obviously, for about a year through all the owners, et cetera, et cetera, and it's very ambiguous in its conclusions depending on where you look.

I took the one which I prefer and the conclusion is that the calculations of the core neutron power oscillations to 200 percent of rated resulted in a 7 percent average power increase due to effects of nonlinearities and the system feedwater effects. So I think that's an admission that there's an average power increase just due to the amplitude of the -- related simply to the amplitude of the oscillations as well as, of course, the cold water effects, also.

The BWROG transmittal and report, too, I believe, concludes that previous ATWS evaluations are valid and existing ATWS actions are appropriate. The Staff feels that there's certainly insufficient basis in that report to support that conclusion. We don't necessarily disagree with the conclusion. We don't believe that either the Owners or the NRC has sufficient answers to agree with it at this point.

MR. LEE: I guess I don't understand the implication of what you are saying about the disagreement.

MR. PHILLIPS: What I'm saying is, this is like end-at-all conclusions. It says: "The Owners have concluded that previous ATWS evaluations are valid and existing ATWS actions are appropriate."

If we can support that conclusion, if everybody agrees that that's a conclusion, we can stop work; we don't need anymore. And we don't feel that there are sufficient answers at this point that we can say that this conclusion is necessarily right. We hope it's right. Our belief is that it's right. But there's still a lot of questions to be answered before we can support that through analyses.

MR. LEE: Is there also a lot of disagreement regarding your first bullet?

MR. PHILLIPS: No, I don't think so. I think based on our conversations with the Owners and with GE they're continuing to do work. There's been none officially identified at this point. We expect to identify some. I'll address this a little more, I think, on a later slide.

MR. MICHELSON: I guess what you are saying that thus far you believe that the previous ATWS evaluations are valid and that the existing actions are appropriate; is that correct?

MR. PHILLIPS: Well, to the extent that the procedures and actions would not be changed. That the ATWS evaluations are valid, no, they didn't fully account for -- they accounted for large oscillations; they didn't account for them as large as we think they can get or as we know they can get.

MR. MICHELSON: You don't really agree with that bottom line even now?

MR. PHILLIPS: I don't agree with the first part. However, the difference in evaluation may not change the conclusions.

MR. MICHELSON: It may be that the existing actions are appropriate?

MR. PHILLIPS: Yes.

MR. MICHELSON: Okay. But thus far you believe that the evaluations are not adequate?

MR. PHILLIPS: That's right.

[Slide]

MR. PHILLIPS: Getting to the NRR review status of these items with the scram system operable, we've been conducting this review on an expedited basis you might say. We are looking to resolving it by early next year. We're currently due to report to the Commissioners at the end of February. And we hope that we at least with the Owners are able to complete review of this aspect of the oscillations and present a solution.

The primary things that we have to consider are the methods. One, the methods for definition of the exclusion region. We have had presentations on this and we expect to have at least one more meeting of that nature before the Owners submit a formal proposal.

The main things we were concerned about as far as methods for defining the exclusion region were, of course, the problems that have been experienced with calculation of decay ratio. The main areas of problems have been in -- historically, have been in selecting the proper inputs. Particularly for power distribution, both radial and axial.

The presentations that have been made to us by the Owners, their procedures appear to account for these previous deficiencies. They are also using a larger radial representation than their previous procedures required which we think helps, also, as far as providing a more realistic valued decay ratio. And they are also accounting for the affect of transients -- plant transients on the decay ratio with the main transients of concern being one loss of feedwater heaters where the loss of feedwater heaters have caused plants to go into instabilities.

And two, the loss of flow similar to the LaSalle event. These transients are also accounted for in their analyses to define the exclusion regions.

We, of course, currently under Supplement 1 to Bulletin 8807 require all plants to manually through administrative procedures avoid exclusion regions which we feel are fairly conservatively defined. We have looked at some examples that have been shown to us of the calculated exclusion regions using these new procedures and the region where automatic reactor trip would be required is essentially the same as the regions that are excluded from operation currently.

So we feel that operational experience also supports that this particular region is -- the nonexcluded region is pretty safe as far as stability goes.

So the modeling uncertainties and decay ratio calculation methods were previously reviewed by Oak Ridge and they concluded that there was about a 20 percent uncertainty there. And we feel that evaluation is still valid if you provide the appropriate inputs in the calculation and that's still being used.

MR. LEE: Larry?

MR. PHILLIPS: Yes.

MR. LEE: Is there an attempt being made to see indeed you can augment this power flow map with at least one or two more variables in light of the sensitivities that we've been discussing?

MR. PHILLIPS: No. all the sensitivities are being considered in the calculation of the region. But if we're going to base the automatic action on power flow, then the region, of course, has to be defined in terms of power flow. But all the other sensitivities are being considered in defining the region.

MR. LEE: But, for example, you mentioned the power distribution playing a role.

MR. PHILLIPS: Yes.

MR. LEE: And so instead of having two dimensional map, if you have a three dimensional map that might help define the exclusion region a little more meaningfully.

MR. PHILLIPS: Well, what they do is, they're selecting -- well, first of all, they're looking at transients which shift the axial power to the bottom, which is bad for stability, and they're being accounted for in defining the region of operation. That is, if you're operating in the -- when you look at the exclusion region boundary you have selected conservative conditions of radial power distribution and axial power distribution to define that boundary. You've looked at bad conditions. Assuming that they're bad, that's a fixed power flow region.

So now the question is: under the worst cases of power distribution both radial and axial, if you stay out of this power flow region will you be stable? And all these insensitivities have been considered in defining the region in terms of power and flow.

MR. LEE: I guess what I'm a little bit concerned about regarding that approach is with all the sensitivity uncertainties that one needs to account for the exclusion region might shrink to a very small region that you make that into a lot of spurious rod insert and, if not, outright spurious scrams which may be detrimental to the operation of the plant.

MR. PHILLIPS: Well, the scoping calculations that they have done show that those boundaries are essentially where they exist in the current interim fix. And those are regions where, for most plants, operation is never needed; for a few plants for various reasons they need to operate slightly within those regions. For instance, WPPSS and that's the reason they're using a decay monitor and putting it on their plant.

We have to review the proposed limitations on operation within the excluded region. That, of course, would be how they're proposing to operate with the decay monitor operable to bypass a scram. This would mostly be during situations of plant startup where for some plants due to specific aspects of their design, particularly their pump designs, if they don't -- WPPSS, for instance, have two speeds on their pumps. And if they operate at the higher speed and throttle the flow they get into vibrations.

In order to avoid the vibrations they need to operate at the lower speed without throttling the flow which would get them into the excluded region during a startup. So one of the uses they're proposing for their decay rate monitor is with that operable to facilitate their startups and avoid that type of problem.

MR. MICHELSON: The decay rate monitor is synonymous with the stability monitor or is it something else?

MR. PHILLIPS: Yes.

MR. MICHELSON: They're the same things?

MR. PHILLIPS: Yes. It operates on LPRM, APRM signals and does a noise analysis and spits out a decay ratio.

MR. MICHELSON: You shifted terminology and I wasn't sure if this was the analysis or not.

MR. PHILLIPS: Yes. Well, that's a bad habit of mine.

MR. CATTON: Is the research scheduling a little bit out of sync with what you have to do? I notice that Harold Scott said that ATWS sensitivity would be done in March of '90 and I guess the amplitude of the oscillations would be June of 1990.

MR. PHILLIPS: Yes. We feel that we're at a point where we are attempting to, at least with this aspect with the scram system operable, are attempting to define the solutions by the time that we're scheduled to report to the Commission. We may have to ask for an extension, I don't know. It's very tight but we're working for that now and the Owners are being cooperative and we're attempting to finish this aspect of the review where we can at least say what the long-term solution to put stability to bed except for ATWS is.

Of course, it will then have to be implemented by the individual licensees. But we would like to be able to say, this is the resolution.

And, of course, we need to complete the review of the design and implementation of proposed stability monitor systems; and we expect to complete one of them in December.

The third option or the second option, we need to review the justification for the adequacy of the existing BWR-2 reactor protection system. So there will be a report on that submitted and we will review it.

And for the third option we need to review the design and the associated analyses for the Owners Group proposed instability trip system. We're attempting, as you noted, to complete all these reviews by the end of January. It's a little ambitious but we feel there's --

MR. CATTON: I'm still wondering how you're going to complete it when you won't get Research's input until March.

MR. PHILLIPS: Well, Research's input -- my next slide starts with -- not my next slide.

MR. MICHELSON: Before you go to that, I'm a little puzzled, I guess I didn't track it as well as I should have. But your second option item there refers only to the BWR-2. Were all these statements only relative to BWR-2?

MR. PHILLIPS: No, only --

MR. MICHELSON: Only that particular one?

MR. PHILLIPS: Only that particular one, becuuse the design of their scram system is different from all others.

MR. MICHELSON: The others are somewhat similar, but BWR-2 was different.

MR. PHILLIPS: Right.

MR. MICHELSON: But the other remarks all pertain to the full spectrum of BWRs?

MR. PHILLIPS: That's correct. And the research effort is primarily oriented to the ATWS review, and that's the reason that their schedule does not conflict with this.

To complete that action of it we need to define multiplan action requirements for implementation of the acceptable long-term solutions and prepare Commission paper providing our recommendations and the status.

Then, of course, as the solutions are implemented we will need to review the MPAs.

[Slide]

MR. PHILLIPS: With respect to ATWS I think I already addressed 31709 sufficiently. Work is progressing, as you've seen, on the identification of code limitations and the improvement of stability analytical capability. We do feel that even though it seems slow and I'm sure that it seems confusing, we feel like we certainly have come a long way for one thing in better identifying the stability sensitivities. And it certainly helps to -- we can look at a new fuel design and determine very well, we think, whether it's more or less stable than the previous design.

So unfortunately, we don't feel that the key questions concerning the maximum amplitude and potential consequences of large limit cycle oscillations have been answered at this point. The reasons they haven't been answered is because we have had so many limitations in the code, and I would say the effort to this point has been in large part on code review and code assessment and validation.

As you saw yesterday, even there we have some conflicting views. For instance, INEL TRAC says, well, it looks like you need 48 nodes; and GE says, well, we do it with 24 but our results look good.

Now, actually to answer this type of question I don't think it matters a whole lot because what we're looking at is, under worst type of circumstances for a reactor core how large can the oscillations get.

Now, there are umpty-ump parameters which can be varied as far as core design goes. And the fact the code may have a little bit of error in predicting the exact instability circumstances of a specific core shouldn't keep it from doing a searching type of study of how bad can oscillations get. If you apply it to a specific core and say, when is this core going to go unstable and how large an amplitude going to be, is it going to be in that core? Yes, then you may be concerned about how many nodes you have.

MR. MICHELSON: Well, the goodness of the analysis may be determined by how many nodes you select or the amplitudes may be determined by that; I don't know.

MR. PHILLIPS: Oh, well, I think it probably is.

MR. MICHELSON: Therefore, I think you're saying that they do have to have good models, good analysis because you are interested in what these peak amplitudes might be.

MR. PHILLIPS: Well, I'm saying that if you want to predict exactly LaSalle under the existing burnup conditions, et cetera, and so forth, then you have to have good analysis. But if you want to look at any reactor core, will the noise in the noding there with all the other changes you can make to that core is probably not significant.

MR. MICHELSON: But you certainly don't want to model in things that will tend to attenuate these amplitudes; then you won't see it and you won't necessarily worry when perhaps you should.

MR. PHILLIPS: True.

MR. CATTON: I think you need to get a little bit closer to the question of whether the relative power is 2 or 25.

MR. PHILLIPS: Yes, right. Exactly.

MR. CATTON: I mean, if it's 2.2 or 26 -- 25 or 26 that doesn't matter or 2 or 2.2 that doesn't matter. But if it's 2 or 25 that's a big difference.

MR. PHILLIPS: That's right.

MR. CATTON: And you've got now two code predictions that put it in the 20s, and I guess some code predictions that put it down here 2 and which is right?

MR. PHILLIPS: Exactly. I guess my point is, I don't think the noding is going to have too much of an impact on whether it's -- looking at that sort of a difference whether it's 200 or 2500.

I believe we're at the point --

MR. CATTON: Gary.

MR. WILSON: Gary Wilson, INEL.

The work yesterday shown by INEL implied that 48 nodes are required, in some people's mind, and I would say, please not do that, that is not what we intended to convey. What we intended to convey was that we think the quality of the code projection with respect to stability has two parts. One of them is numerical dampening related; and one of them is the actual physics. How good are the actual physics?

We have done studies in which we have moved to the 48 node type representation to look at the numerical dampening trend. And we looked at a number of them, you know, just find out what the numerical dampening trends were.

The ultimate answer will come when we say we know sufficiently what the numerical dampening contribution is and we know how to quantify that; and then we take the next step where we do the things that GE has done, we go assess against plant data or whatever data we can and look at the real physics and find out how good the answers are. When we know what the numerical contribution is and when we know what kind of nodalization we have to have to well match the experimental data; then we know how many nodes we have. That may well be 24. I would hope it would be six.

So please don't say that the work that INEL presented yesterday says we've got to have 48 nodes; that is not what we intended to convey.

MR. PHILLIPS: Thank you, Gary, I didn't mean to say that, but I did say it.

MR. WILSON: I heard from some other people, so there must be more than just your perception.

MR. PHILLIPS: Yes.

I guess there's one other aspect of some of the things that were said yesterday that I would like to address, too, concerning availability on the part of NRC to data with which we can assess the codes. And I think it may have been a little bleak picture. I believe we have access to all of the data that we want. We may not be able to define exactly what is needed in the sorted detail that GE is on some of their plants and knowing what is available, but we've been getting a lot of data.

The Owners have been working with us to give us what we need as far as LaSalle goes. The Sweds have been very cooperative. We've gotten data on two of their reactors: one where there's been an instability event; and another one where there's been considerable testing. And those have been assessed with LAPUR which Jose will address to some extent.

So we do have data with which to assess the codes. I think it's a question of how we best apply our resources plus a schedule of what we need to do here to get to a solution of the problem.

I'll turn this over to Jose to address a little more in detail some of the work that we have been doing and I'll sum it up after he is done.

MR. CATTON: Looking at the clock, Jose, we have one-hour including summing up by Larry.

MR. MARCH-LEUBA: I don't have a long presentation today.

MR. CATTON: Okay.

MR. MARCH-LEUBA: My name is Jose March-Leuba from Oak Ridge National Laboratory. And it's always difficult for me to make presentations because most everybody in the audience have seen all my work at least four times already, so I never really know what new to say or anything.

I got the idea that this meeting was more oriented towards what codes are available. What are the capacity of the codes. And really, what we know of the stability of what we need to do of the stability. So that's basically how I'm going to make this presentation. I will first give a scope of what my mission is in this area, what I'm getting paid for. Then I'll give some overview of the stability codes as I understand they work.

I'll give a brief description of the LAPUR code which is a frequency of the main code that we all use in Oak Ridge. And a little nonlinear time domain model that I have used for nonlinear studies.

And then I will wrap-up with what I think we know about the stability and what we need to know.

[Slide]

MR. MARCH-LEUBA: Let's start with the scope. Really, I am a consultant to NRR, and that's who pays my bills. Really what I'm being paid for is to know the issues and specifically to review the proposals. Whenever GE or a vendor or somebody presents something to NRR and they need some technical advice then they ask me or they ask somebody else. I mean, I'm being paid to review those proposals and to understand if there are some possible safety issues and I have to raise them with them.

The numerical tools that we work with are LAPUR which is a frequency main linear code. And I have a small reduced order, nonlinear model, which is a very simple, extremely -- some people even laugh at this because it's so simple, but it gives some very nice results and some understanding about the problems.

[Slide]

MR. MARCH-LEUBA: Let's start with the codes.

You all know by now there are two types of codes just for stability. We have the frequency main code; and the timing codes.

Frequency of the main codes just because they're linear they're only good for one thing and this to predict the onset of instability. You cannot do many more things with it except to compare the instability of different designs. About the only thing a frequency of the main code is good for is to tell you, your system is stable or it's unstable. And if you are stable, you are that from away from instability.

I have heard some comments just on the opening remarks that somebody is saying that you need 3-D to calculate the onset of instability. I want to take issue with that. I mean, linear frequency theory is a rigorous statement for the onset of instability.

MR. CATTON: That's 3-D, right?

MR. MARCH-LEUBA: That's my second point. You can make a 3-D of point kinetics.

MR. CATTON: Sure.

MR. MARCH-LEUBA: For some reason whoever admitted point kinetics made a tremendous disservice like calling it point. It is -- point kinetics is better than 1-D and it's better than 2-D and it's better than 3-D. It's reproduces an exact solution of the first mode of the reactor. It is rigorous. It is exact. It's only good for very, very small oscillations around the kinetic point. As long as your power shape is not disturbed.

But as long as that assumption is correct, and that's the only thing we use it for, to bring the instability where we are looking at minute perturbations around the kinetic point. The point kinetic solution is better than a 1-D solution. It is probably better than 3-D solution.

So as long as is perturbation is infinitesimal you don't need 3-D effects. You must have 3-D effects in the thermal hydraulics to compute your activity feedwater, that I grant you. But in neutrons you are exact from the model and you represent all the mathematics of it with point kinetics.

MR. LEE: But if you cannot represent the thermal hydraulic feedback correctly because you do not have 3-D full blown representation, then you point kinetics will not be counted on.

MR. MARCH-LEUBA: That's absolutely correct.

My point was being with respect to the neutrons. The thermal hydraulics you need full blown 3-D neutronics because it is a three-dimensional problem.

MR. LEE: But to the extent that you need to have distributed temperature and dense feedback represented, you do need the 3-D kinetics as well for accurate prediction.

MR. MARCH-LEUBA: No, I don't think so. I mean, you need to be able to condense the 3-D feedback into a 1-D activity.

MR. LEE: But it is time dependent to feedback.

MR. MARCH-LEUBA: But as long as it is very small perturbations so that the shape function does not vary, point kinetics is a exact representation.

MR. CATTON: That's basically --

MR. LEE: That's okay.

MR. CATTON: That's basic to linear stability.

MR. MARCH-LEUBA: Yes, exactly.

MR. LEE: So you do have the 3-D feedback represented in some way.

MR. MARCH-LEUBA: You must have a 3-D feedback representation, 3-D thermal hydraulics on feedback.

MR. LEE: But in your analysis you do not have that capability.

MR. MARCH-LEUBA: Yes, we do.

MR. LELLOUCH: I think you're talking the same thing, but different ways.

My name is Lellouch.

You have to have the steady state three-dimensional power distribution, number one. You must have the representation from the neutronics of the three-dimensional feedback. That is the feedback coefficient to avoid the temperature. And then there is an analytical procedure for collapsing that to the point model. And then if you have the three-dimensional thermal hydraulics with the fixed 3-D power shape, you then have a complete representation if the fundamental mode of neutronics does not change, it doesn't change its shape. You no longer need the 3-D neutronics at that point; you need only quadratures over space in order to feed the point kinetics model.

MR. LEE: I don't have any dispute on that point whatsoever.

MR. MARCH-LEUBA: I guess we agree then.

MR. CATTON: The Japanese study using retrend, they argued that some of the dynamic effects played a role. And the way I read it they actually implied that finite amplitude disturbances could lead to oscillations at points that were outside of the normal exclusion boundary. That says that you need to do something different than linear stability.

MR. MARCH-LEUBA: Your point is well taken.

There is something different of linear stability. The boundary between stable and unstable in a linear sense is a very fine, very thin line. Whenever you go into nonlinear domain you have boundaries that depend on the oscillations.

MR. CATTON: That's right.

MR. MARCH-LEUBA: That cannot be studied in frequency domain analysis.

MR. CATTON: And that needs to be a part of your analysis, too. But that's also a well developed art.

MR. MARCH-LEUBA: It is.

MR. CATTON: I haven't seen it practiced here.

MR. MARCH-LEUBA: It is exceedingly more complex.

MR. CATTON: Of course.

MR. MARCH-LEUBA: I don't think we can -- well, we can probably attempt to tackle it, but we have many more problems than to worry about that.

MR. CATTON: The Japanese said it in their paper that the dynamic effects and disturbances being injected into the core caused the stability boundary to move.

MR. LELLOUCH: It does, but not terribly much.

MR. CATTON: Well, if that's the case then it's not a problem.

MR. LELLOUCH: That's correct. Generally speaking, that is not a problem. But if you actually have infinitesimal perturbations that question vanishes. Only when the perturbations become finite that you then have to look at it.

MR. CATTON: You always have to look at that. Linear stability is nice. It's simple, it's clean, everybody can writes lots of papers. When problems -- when it's subject to finite amplitude instabilities, that's a different problem entirely; it's difficult. But you can't -- and it can shift boundaries a long ways; it may not in this particular case.

MR. MARCH-LEUBA: The typical sample that you're talking about is the axial oscillations, for instance. Actually, the same on the axes are stable when you have them operating, you disturbing half it becomes unstable. So you put a perturbation that is large enough you can excite an instability.

We have not seen that behavior with stability with BWR stability, to my knowledge. I do not think there is an unstable limit cycle around a stable kinetic point. But certainly have not seen any study in that regard either.

MR. LEE: Could you repeat what you said about the magnitude of the perturbation deciding the stability.

MR. MARCH-LEUBA: Yes. Let me give you an example. Hold on a minute.

A very typical example is the center problem. When you have -- you can excite axial center oscillations in a reactor by perturbing the reactor enough, and what happens there in the space is there isn't a kinetic point which is stable. That means trajectories go around and not absorbed by the point.

But somewhere around there, there is an unstable limit cycle. And you could say that you are plotting the end versus the limit, for instance. So if there is an unstable limit cycle and you perturb the solution from that kinetic point in half you might reach a side and then the trajectory is spiral away and you form a stable limit cycle around it.

So that in this case we have the reactor operating normally and then is stable. And if you perturb the reactor in half so that you reach outside that unstable limit cycle, then you would start your center oscillation. That's very typical; it exists in every reactor in the world.

I just don't think that's the case of that instability, but I don't have anything to show and I have not seen a single study that addresses that point.

MR. LEE: I tend to disagree with what you just said about this dotted limit cycle. I think it's normally the limit cycle that you converge to if the system is unstable.

MR. MARCH-LEUBA: No.

MR. LELLOUCH: If the system is unstable to the outer one.

MR. LEE: What is the cycle then? You said it's unstable limit cycle, by definition this is a limit cycle.

MR. MARCH-LEUBA: It is the same thing that the unstable kinetic point, it's something that really doesn't exist. It's a trajectory that repels all other trajectories.

MR. CATTON: I am really enjoying this, but I think we better get back on track.

MR. MARCH-LEUBA: Well, as I said, within some possible exceptions on this large amplitude perturbations, frequency can cause the onset of instability fairly well. And they can compare the stability of different designs.

Time linear codes can do all those things, but they cannot study nonlinear fix like, what would be the limit cycle amplitude whether it will be flow reversal or not. They can predict more easily the impact on the fuel of large limit cycle amplitudes.

And they can also study system effects like controllers and operator actions. Some controllers can be studied in the frequency domain but most of them have dead bands and valves ticking like we were talking yesterday. It's a lot easier to study them on a time domain.

So really the conclusion that we can get is that, again, the frequency domain codes are more accurate numerically and require orders of magnitude less competition. And that is simply because the integration in time has been done. We have integration by the frequency domain. So it is much more easy to compute.

The other negative point of time domain codes is that there is also something difficult to interpret due to system effect and nonlinearities.

So I conclude, at least in my mind, whenever possible it's best to use a frequency domain code. Whenever it's possible is whenever you want to do scoping calculations or relative stability of the changes, and if you want to define a stability in the power to flow map.

[Slide]

MR. MARCH-LEUBA: Here is sort of a who's who in the stability area. It's a list of all the codes that are used in the United States and they're in alphabetical order. And on the frequency domain we have FABLE, that is used by General Electric. LAPUR, that's used by Oak Ridge. And NUFREQ, that was developed by RPI and NRC sponsorship. I wanted it to be used by Westinghouse, but they are out of the business.

On time domain codes we have the COTRAN code, that's nuclear fuels code for stability. HIPA, Brookhaven. RAMONA, that was developed Brookhaven and with some corporation, Scandpower and now they have two different codes. And there are many variations of these codes.

RETRAN is EPRI and the utility uses it for calculations in the Grand Gulf plant. There are many versions of TRAC as we saw yesterday and basically used in the BWR field by INEL and GE.

This really has some history of validations behind them. None is a perfect benchmark or anything like that, but there is a lot -- they have been applied a lot to known cases. For instance, FABLE has been applied to Peach Bottom, Vermont Yankee, LaSalle, and I'm sure many more that are not publishing literature.

The LAPUR code is the one I know. We have a benchmark or at least used it against the Peach Bottom stability test; the Vermont Yankee stability test; Browns Ferry; Susquehanna; Grand Gulf. And specifically, a Swedish BWR that Larry was commenting before in which they were out of phase oscillations. And we were able to benchmark against those tests for in-phase and out-of-phase oscillations because LAPUR can predict out-of-phase type of instabilities.

NUFREQ, at least the NPW version is the latest version. It has been benchmark at least in Peach Bottom and probably against more things, so has COTRAN. And HIPA has been benchmarked individually against NUFREQ stability test, and a little against LaSalle.

We mentioned before, there has been a benchmark between HIPA and LAPUR, a cross-code benchmark.

RAMONA has been certainly benchmarked as I guess FRIGG channel stability. And I know the Sweds Scandpower has used it against some of the tests in Sweden.

[Slide]

MR. MARCH-LEUBA: And RETRAN has benchmarked against Peach Bottom, Grand Gulf. There have been some LaSalle instability event kind of markups.

And TRAC has a very long list of validations, but the ones that were here yesterday, FRIGG for stability, LaSalle. And we saw yesterday Leibstadt.

[Slide]

MR. MARCH-LEUBA: Those are all for linear stability. For nonlinear or for large amplitude limit cycle the picture is a lot worse. There is really not a good benchmark data set that one can use for large amplitude limit cycle. There is some, just a few cases in which you have medium amplitude limit cycles, like we were saying LaSalle and Leibstadt. But there is not a real good benchmark case that would tell us whether there is a factor of 2 or a factor of 20 amplitude limit cycle. And that is really the real question that is worrying us right now is whether the oscillations can be 2,000 percent or 10,000 percent or just 200.

But with available data which is the data at LaSalle, TRAC, HIPA, and RETRAN have been modeled and up to a point all of them have fairly disagreement, what will happen if LaSalle had not scrammed.

[Slide]

MR. MARCH-LEUBA: So the conclusions that I get from this code validations is that, by looking at all the benchmark most codes are very capable of reproducing linear stability results fairly accurately. And this is, once we know the operating conditions, post-test conditions with a very good detailed analysis of what the coefficient was and what the power distributions were, we can get certainly we think 20 percent error estimate into the calculation, which is not bad. I mean, 20 percent is an excellent agreement considering all the sensitivities to all the parameters we have.

The problem comes when we try to define what is the most unstable condition for a fuel cycle for next year in Grand Gulf or in LaSalle.

So I have to conclude that we have a tremendous difficulty in trying to do predictive calculations for a particular plant. We have to do kind of general predictive calculations with sensitivity analysis like saying that the word coefficient can change between 1 and 1.2; then you can bracket it.

What I have really problems is with a maximum amplitude of oscillations. Different codes give different results and some have given like 200 percent, some give 2,000 percent for amplitude limit cycle oscillations. And this is tremendously important on ATWS, because if it is just 200 percent and there is nothing -- I mean, the conclusion that Larry have in this slide is good. There is no problem with ours, with the stabilities

Now, if we have 2,000 percent, that's not to say that puts your average power in 85 to 90 percent of nominal, then we have a problem.

MR. CATTON: Where is the difficulty with the codes, is it in the thermal hydraulics or the neutronics?

MR. MARCH-LEUBA: I guess that's a wise answer.

I think the neutronics will know it fairly well. My impression is the --

MR. CATTON: There are a lot more variables for the neutronics. aren't there? And you worry about power shape and all of these other things. You have rather fixed geometry for the thermal hydraulic.

MR. MARCH-LEUBA: Yes. But you have to be using a lot of correlations and a lot of feeds. I don't know, it really is both. The problem is with both. And there's a third item that you didn't mention is the numerics. It is how you chose to solve the equation.

MR. CATTON: Well, certainly.

MR. MARCH-LEUBA: I don't really have a feeling for what is the problem there. Anything I use is point kinetics.

MR. ROUHANI: I would like to say that there is no technical problem about using neutronic. What I tried to mention, there is a lot of cost involved of going to a number of calculations to provide amplitude. There is no technical problem.

MR. CATTON: Is it 200 or 2,000?

MR. ROUHANI: Regarding the peak? I was not addressing that. I tried to say there is no technical problem in using it.

MR. MARCH-LEUBA: The problem with the peak is that it is not -- for instance, when you have a LOCA analysis you are forcing an external event. You are kind of forcing the result. This is something that just pops up. Nobody is doing anything to it and, yes, the core is oscillation when the reactor vessel is on. By itself it limits.

So you have to have everything very well developed to find out what causes the limit.

In my studies with very simple models I have concluded that the limit cycle is caused by the neutronics, not by thermal hydraulics. It is because of the basic -- the fact that in point kinetics you have th reactivity multiples the power sensitivity. That's the only linearity that causes the limit cycle. You can linearize everything else and you still get the limit cycle of the same amplitude.

Basically, I really don't know how to answer your question. It's like what is the egg or the chicken, everything is together.

So there is a real need to validate, benchmark or verify a time domain for large amplitude limit cycles; and I really don't see how to do that. We have to do the best, as we saw yesterday, use the best available and at least know that it does a good job for the known test points.

Now, just moving along I want to describe briefly how the LAPUR code works. If you are really not interested, there's not that many details, I can skip it. How are we on time, are we all right?

MR. CATTON: My colleagues have mixed emotions.

Why don't you proceed.

MR. MARCH-LEUBA: Skip it?

MR. CATTON: Yes, skip it.

[Slide]

MR. MARCH-LEUBA: Just for the ones that don't have the slides, I was going to talk about LAPUR and how we combine the functions to calculate the in-phase, the out-of-phase and the channel decay ratio to be able to predict those.

Let me just show you, everybody has their own benchmark. I just completed this one a couple of months ago on that Swedish boiling water reactor and how LAPUR can predict the decay ratios, the in-phase decay ratios for this particular reactor. And it had something like 10 points and here we have measurements calculated and they're all fall in a very nice line.

So once we know the condition we can reproduce those calculations fairly well. And this is one of the best benchmark I ever made because they sent me all the information I asked for. I mean, in perfect form and it was very simple to do this benchmark because they did a fantastic job.

The interesting point about this particular set of tests is that all these decay ratios that were calculated were in-phase. They were excited by using pressure perturbations in the steam line. But this particular test for this particular problem was an out-of-phase instability. So they kept moving along the natural circulation and kept cooling control rods and when they got to this point an out-of-phase radial, the one that is 180 degrees out-of-phase just showed up. A very small amplitude, 10 to 15 percent and stopped there.

But for some reason they didn't stop there and they went ahead and they performed the pressure perturbation test anyway. And they were able to measure even though the out-of-phase mode was going on, they were able to excite the in-phase mode at the same time and measure what the decay ratio was in that particular position.

MR. CATTON: So they were superimposed, the two modes?

MR. MARCH-LEUBA: They were completely independent.

So by natural the out-of-phase mode wasn't stable. You saw it, you saw limit cycle. When they introduced a pressure perturbation which is a global variable and it's a core-wide perturbation you excite the fundamental and they were able to see the fundamental superimposed from the second one. And by correlation analysis you can get the difference between the original pressure and the fundamental mode and they were able to calculate and the decay ratio which was stable, which kind of proves that this idea that there are really modes which are completely independent and they're not -- they don't talk to each other. I mean, there is one fundamental mode and one out-of-phase mode that they're completely independent in the linear regime.

I agree that once you become very nonlinear and you have 100 percent oscillations they start to have cross-talk. But in the linear range this test is one of the most interesting tests I have ever seen, which it showed that you could see the stability of two different modes.

LAPUR was able to predict for that particular condition. Now, the out-of-phase decay ratio was more unstable than the in-phase decay ratio. It gave us something like 1.05.

And for some of these tests the in-phase was more unstable than the out-of-phase and then the entire flow, the situation is reversed. So in that particular reactor where there are in-phase and out-of-phase mode, what happened was really an amount of lack in the sense that the in-phase calculation was .9 something and the other phase was 1.

So maybe if they come back tomorrow and do the test the situation is reversed.

MR. CATTON: Does that mean that you have to look for both types separately?

MR. MARCH-LEUBA: Yes.

In the LAPUR analysis we calculate them completely separately. We do more analysis and calculate it. If you have time domain you will have to do your channel arrangements so that you allow for all the modes. All three different calculations in which you force the three modes.

MR. LEE: This test you're talking about is a live start test?

MR. MARCH-LEUBA: It's a Swedish reactor and we promised not to mention the name of the plant.

MR. LEE: It's not Oskarshamn either?

[Laughter]

MR. MARCH-LEUBA: I give you enough information, so if you know the test -- I mean, it was one of those two.

MR. LEE: No, let me pursue. I mean, in this symposium paper the Oskarshamn report is the same event.

MR. MARCH-LEUBA: Yes. Yes. It is exactly the same thing I reported there and then I gave the name of the plant.

MR. LEE: No, not your paper but the Oskarshamn people reported their test.

MR. MARCH-LEUBA: It was Forsmark, the ones that reported it.

NR. LEE: No, it was the Oskarshamn people, I may be wrong. But if I remember Oskarshamn people reported that out-of-phase mode was more unstable than the fundamental mode.

MR. MARCH-LEUBA: Yes, that's Oskarshamn, yes.

You're right, the ABB representative presented something on those concerns, you're absolutely right. So that's the case. And they also in that, the Finnish percent is something on Forsmark, those were in-phase. And we have received the data for those tests and we are going to prepare another benchmark for that.

There is some confusing, also, with the work I do. Some people think that I calculate limit cycle amplitudes with LAPUR. And indeed I do because of this simple one. This is what I call my five equation models. It doesn't have anything to do with momentum of voids or anything; it is just simple five different equations which are just point kinetics.

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Sometimes I use 60 loops and we have one node for the temperature of the fuel and just two nodes for the activity coefficient. The density of the channel which multiple with the coefficient gives you the reactivity feedback.

What I do with this model, it captures basically all the dynamics of the BWR, of the system. And I have a bunch of parameters, A-1, A-2, and K that I feed to the resource of the LAPUR run, so that I run LAPUR with all the detail that it allows me to have. And I generate a transfer function for the reactivity to power that looks something like that one. This is for Vermont Yankee test.

And then I feed the parameters of those to the resource of LAPUR and that's the result of the fit. You can see the crosses of the data and the line, the center is the feed.

So now we have a model that more or less represents the dynamics of a reactor and at least fits very well the linear dynamics that LAPUR predicted. But this more now is a time domain model. It has no linearities, particular that for all times end nonlinearity is the only one it has. So I can drive it and study the effects knowing the effects with it.

MR. LEE: The void reactivity model already assumed something like that's in a solenoid behavior.

MR. MARCH-LEUBA: It is -- let me tell you how I got this model. I got this function or not this one but many, say, 10 or 20 that I run with LAPUR. I run 20 cases. And I calculated how many pulses I needed to fit that other function. And I found out I needed a zero down here in low frequency. And I needed two pulse here. And I certainly needed another two zeros because it was flat and I needed to have the same number of pulse as zeros to be able to fit this function.

So I found out what is a minimum order required to represent these dynamics as LAPUR sees them. Then I did lots of hand waving to show why this zero should be equivalent to a time constant of the fuel dynamics. It's .03 hertz and is of the right order of magnitude. And you can back trace it to be independent to the fuel dynamics.

MR. LEE: I guess my question is more directed to the void feedback model only.

MR. MARCH-LEUBA: Yes.

MR. LEE: And this is the second model that was used many years ago here and there, too, as I remember. And this is the second of the transient function.

MR. MARCH-LEUBA: That's correct.

MR. LEE: That it converted back, in fact, to time domain. So it does not really represent the role of physics. That is, it's a fitted --

MR. MARCH-LEUBA: It's a fit.

MR. LEE: Right.

MR. MARCH-LEUBA: You're absolutely correct.

It does not impose the fact that the reactivity -- the reactivity, as you know, is going to be sinusoidal.

MR. LEE: That physics is represented here.

MR. MARCH-LEUBA: But it's not imposed by that equation, no. It will come up like that in RETRAN or RELAP or high power, any other model that has 12 or 24 nodes.

So this is just a tip on another way to do some analyses that allow me to expand the resource of LAPUR into the nonlinear domain. I have gotten some results from it which are just kind of a scoping calculation. They don't pretend to be accurate to the second order of digit, maybe not even to the first order of digit. But it allows me to do a lot studies. I can run more than 100 times in a day, maybe 1,000. So I can really do a lot of studies with it.

And what I found out with the model is, first, there is a limit cycle that bounds the amplitude; and that's good. And I also determined that the limit cycle is caused by the neutronics. And that's because this model only has nonlinearity and it still produces a limit cycle. And that's the only nonlinearity that's there, so it must be the cause of that nonlinearity.

Another interesting thing we found is that to establish a limit cycle there must be a negative reactivity bias. That means, there has to be an increasing voids, and that can only be accomplished either through average power increase or through a flow reduction.

And we have done some analyses and found out that this average power increase should be of the order of two percent of the peak value. And those numbers have been validated or verified with RETRAN and HIPA. And so this simply more or less predicts some of the behavior of all the more complex modes.

Now, this is of large importance to our ATWS events because really this average power increase, if we have a 20,000 percent oscillation then you have fairly significant average power increase, it hit your suppression pool and all this type of behavior.

Also, I could do some kind of analysis of what is credible or reasonable amplitude. From this model I can get any amplitude oscillation you want. I just have to change the variable K and I get -- you want 2,000, I get 2,000; you want 2 million, I get 2 million. I can give you anything you want.

Now, what is reasonable? What is credible?

What it tells you is that the basic dynamics of the system does not limit oscillation. You can possibly get anything you want.

Now, what can you really get with a real reactor; that's the million dollar question. And I have run some cases with LAPUR and give it reasonable predicting conditions and the highest ratio I get with reasonable predicting conditions is on the order of 1.6. And with the calculation of 1.6, if I fit it into my process I get oscillation of the order of 500 percent nominal.

So within reasonable and credible -- and I'm waving my hands a lot -- I get at least 500 percent of nominal, I believe those oscillations. Now, I can generate runs with LAPUR that put, say, to the 100 degrees Celsius and I can generate oscillations as large as you want.

Whether it is credible or reasonable is up to -- I really cannot tell. So the problem we are having here is, we want to know how large these oscillations can be and the answer is, they can be as large as you want and it depends on how original your operating conditions are.

MR. LEE: This 500 percent peak power case involves reverse load, doesn't it?

MR. MARCH-LEUBA: It more surely will, so it certainly invalidates all my analysis because I don't have any reverse load.

MR. LEE: And that also raises a lot of the question regarding many of the large codes being used for this particular analysis as well?

MR. MARCH-LEUBA: The reverse flow?

MR. LEE: Right.

MR. CATTON: No, no, no, RAMONA has reverse flow.

MR. LEE: I understand, but I don't know whether it has really been validated against any separated test or whatsoever involving reversal.

MR. CATTON: I think TRAC has, but I'm not -- sure, it has reverse flow because they have to deal with the LOCA and the LOCA, it blows it out of both ends of the core, so it can deal with reverse flow.

I don't know if RAMONA has been validated against reverse flow. Certainly TRAC has.

MR. MARCH-LEUBA: From what I understand reverse flow appears whenever you get to 300 percent or so of the oscillations.

MR. LEE: But this is a reverse flow coupled with oscillator behavior, which is somewhat different from the simple LOCA analysis.

MR. MARCH-LEUBA: My gut feeling is that reverse flow with voids is a very difficult thing to model. I'll be very surprised that anybody does well. I mean, reasonable is okay.

MR. WEAVER: Walt Weaver from EG&G.

That is why when I made my presentation I said there is a lack of data, separate effects data for limit cycle. Just exactly addressing the question that you are raising.

MR. LEE: Thank you.

MR. MARCH-LEUBA: And that question of reverse flow was raised at the last ACRS meeting where they can cause channel dryout. I mean, if you have a reverse flow of significant amplitude and you can blow enough steam in there you can keep it dryout for more than an oscillation, maybe 10 or 20 seconds or whatever. I believe the Jens correlation takes care of that, so I've been told. I mean, that it will predict the DMB before it predicts -- but it is one of the mechanisms by which you could get very serious fuel damage if you're still have large oscillations.

MR. SHIRALKAR: Shiralkar from GE.

I wonder if you have large oscillations that you predict, wouldn't the effects of heating loss significantly change your reactor returns and accounting for those things?

MR. MARCH-LEUBA: Yes. I've been trying to with the direct heating would limit the amplitude oscillation. And indeed it does. It acts on extra feedback term that is impossible in-phase with oscillations. But it does not bound it. If you put direct heating in the model and I have put direct heat in this particular model you will still get the amplitudes you want. It does not bound the amplitude oscillation.

I thought maybe it would and that's why I put it there, but it didn't. It will help you. It will certainly help you. Whenever you oscillations of 10,000 percent the doppler should turn you around.

We found out that limit cycles might become unstable or indeed they do become unstable and bifurcate and there's a lot of mathematical theory behind it. But the point behind it is that once you bifurcate you're amplitude increases much more rapidly, so you get a lot more amplitude of bifurcated or for unstable limit cycle that you do from unstabling the cycle.

MR. LEE: Jose, if you're to quote just the one rough credible number where the bifurcation may start in terms of peak power, what would you say?

MR. MARCH-LEUBA: In my model this test of 500 percent.

MR. LEE: So if you would have 2,100 percent peak power --

MR. MARCH-LEUBA: It's bifurcated for sure.

MR. LEE: -- and you should certainly expect it.

MR. MARCH-LEUBA: Oh, yes, you can see the results.

MR. LEE: I think I have seen that, too.

MR. MARCH-LEUBA: Yes.

MR. WULFF: In HIPA we reached these type of peaks after we go through bifurcations. There is a period of doubling bifurcation and then it becomes aperiodic after.

The peaks are controlled by doppler feedback. The directed position comes at the time of the lowest of voids and to the highest power. They have some small influence on returning the power but it is nevertheless controlled by doppler feedback.

MR. LEE: But I have great concern and very little trust in anything that goes beyond 200 percent of peak power in terms of accuracy and reversal and all these things.

MR. WULFF: All of our correlations for quasi study experiments on heat transfer have to be questions, not because of the period of two seconds, but it is really happening at 230 milliseconds to 400 milliseconds that we reach the high power and return to normal. And then we stay at the low power for a low power to make up half Hertz oscillation.

MR. MARCH-LEUBA: So I am getting to the end of my presentation.

I wanted to summarize kind of on a positive mood what we know about the stability. I mean, everything we hear is problems with nodalization; problems with correlations. But we do know a lot of things about it. We know, for instance, we have classified the types of instabilities. We know there are plant type instabilities that have to do with the control system. And Glen Watford described one yesterday that happened during the LaSalle event, this valve that got stuck. And, of course, the instability of this, that it was oscillating by itself. That does not have anything to do with density wave or the channel thermal hydraulic, just a control system instability, just the portion on the outside.

We can have the channel type instabilities, which is purely thermal hydraulic. And then neutronic instability we recognize the in-phase and the out-of-phase region of which there are several modes.

One thing was raised yesterday where there would be axial modes in this type. I happen to believe that there won't because I don't see any advantage for the axial mode thermal hydraulically. The same way that the out-of-phase planner type instabilities will have an advantage because they keep a constant pressure drop across the core.

An axial type of instability would not have the advantage in this thermal hydraulic point of view and will have the disadvantage of being so critical. So I think that maybe some analysis needs to be done, some numbers, but it's not very probable.

I think we understand the physical mechanisms for instability. And we certainly know that LOCA likely have low flows. And when the sensitivities is fairly well, we know what causes instabilities and how we can go to worst conditions. Particularly worst parameters is the high power density. If you have high power density you are very likely to have instabilities; that means power to flow map you go to the left corner. Power shapes is a tremendous and significant parameter, both axial and radial.

We understand the nonlinearities in the reactors and we know that there is a limit cycle that limits the oscillation. But the amplitude is very large and according to some calculations can be up to 2100 percent. As I told you, there is really no limit, it's only an estimate of how reasonable you want to choose the initial conditions.

We also understand that the limit cycle must have a negative reactivity bias which can be accomplished by -- which can only be accomplished by voids to the channel. And that can be accomplished by having a power increase or a flow increase. Doing the worst oscillations one will have a constant feedwater flow and therefore the power cannot increase because the water level will keep reducing until the flow has decreased enough so that the negative reactivity bias has accomplished.

These might have some more implications for the calculations we saw in HIPA at 2100 percent. This negative reactivity bias is nothing significant, it's $7. So our reactor for those types of instabilities of 2100 is $7 super critical. I mean, we're not just critical, we're $7 above it; and we need to compensate with voids for those $7.

So that means if you want to shut down the reactor with boron you need to put at least $7 toward oscillations plus then you have to start shutting down again. So that will increase the time of shutting down the reactor during worst conditions and has to be evaluated. That's one thing people don't realize, an unstable reactor is super critical reactor.

MR. LEE: I thought that there is a negative reactor to bias --

MR. MARCH-LEUBA: To compensate for that super criticality. I mean, while the reactor -- the limit cycle is growing you are kind of diverging, you are super critical. And as the limit cycle grows you are voids to the system to compensate for that super criticality. It's a way of thinking.

MR. LEE: But not by super critical by $7 by any means.

MR. MARCH-LEUBA: Yes, on the average those calculations that HIPA showed are one fraction is $7 higher.

MR. LEE: Could you comment on that.

MR. WULFF: The negative bias is between $6.50 and $7.

MR. LEE: Yes, that I agreed.

MR. WULFF: Yes, it's a negative bias.

MR. LEE: Right.

MR. WULFF: And Jose translates that same number into super criticality. We do get very short times approximately $1.

MR. LEE: Yes, that agree.

MR. WULFF: Maybe 10 cents.

MR. LEE: Sure.

MR. WULFF: So I think it is a way of expressing.

MR. LEE: Okay, fine.

MR. MARCH-LEUBA: It's a way of thinking. The reactor -- the available reactivity to a reactor is never over a dollar, $1.20.

MR. LEE: Thank you.

MR. MARCH-LEUBA: I tend to overspeak sometimes, but it's a way of thinking, okay.

So we also know about the consequences of reactivity. For in-phase oscillations which basically neutronic driven through the void feedback and the fuel axes of filter. That means, all the perturbations come from power and the fuel has a six second time constant, so it has a very negligible effect on fuel for reasonable amplitude oscillations. We are talking a few degrees change in the cladding temperature for very large oscillations.

But we know there has to be this reactivity bias and they are so easy to detect. The in-phase oscillations anybody can see them in the control room.

For the out-of-phase oscillations they are mostly flow driven through the dynamic pressure drop dynamics and can even cause reverse flow at inlet. That shows to me their point of stagnation in the channel. And the safety limits are violated with much small oscillations. And some calculations show that for BWR-6 I believe somewhere between 200 and 300 percent peak power will reach safety limits.

Those are more difficult to detect the in-phase stabilities. This still can be detected because it is kind of global instability; it's in all the channels. Therefore the channel type of instability which is also flow driven and it has exactly the same problems as the out-of-phase. I mean, we will probably see that for 200, 300 percent oscillations you will have the safety limits.

The channel, the worst problem is that it is very difficult to detect. I mean, only one channel out of 800 is oscillating. And if you just don't happen to looking at the LPRM close to the channel you will never see it.

Hopefully, the channels are designed not to become unstable, but under so much pressure conditions or some extreme peaking factors they might become.

[Slide]

MR. MARCH-LEUBA: So what do I feel that we need to know is kind of a recap of Larry presented. And I'm through with my presentation here.

If we have the scram system available we need to validate the effect of fuel integrity of limit cycle oscillations. That basically -- there is only one analysis performed by the Owners Group and GE about what is this effect. And I think NRC needs to -- and that's part of our research effort, is to kind of audit those calculations and find out what is the effect of in-phase, out-of-phase and channel oscillations.

And this is where I put this channel dryout type as a question mark, can it happen and what will be the consequences. We need to evaluate the detectability of the instabilities.

That is, what does the protection plant system in the control room instrumentation see during instability.

And as a third item, we talked about the stability monitor system which is installed in WPPSS and there will be many more installed soon, if option 1-B shows up.

It's only geared to us in-phase instabilities. That does not detect a list in the coolant mode operation, out-of-phase instabilities. The out-of-phase instabilities out of there and they might be detectable. But with the coolant state of the art technology we cannot see them, so there's a lot of work to be done in that area that will be very helpful.

MR. MICHELSON: I'm having a little difficulty with a couple of your statements. You didn't tell us how you might detect these out-of-phase instabilities; you just said it would be difficult.

This channel stability type monitoring systems, perhaps, could monitor, but again, you didn't tell me how you would do that.

Are you just saying you don't know or what is the status of the situation?

MR. MARCH-LEUBA: I do know for to detect instability once a limit cycle occurs, if it is a core-wide instability your APRM will see it; and then your scram will automatically function. The operator doesn't need to do anything. So that's very easy to detect, you don't have to do anything.

If it is an out-of-phase oscillation the APRM in theory doesn't even see it. In practice it sees a little bit. So you have to rely on the operator to hear on the up scale and down scale alarms and recognizing that there is an out-of-phase oscillation.

MR. MICHELSON: How does that work?

MR. MARCH-LEUBA: The LPRM is oscillating widely. But the only scram that's available to a reactor is on the APRMs, the average. The average has not seen anything in theory. But the LPRMs, the way it is detected is either somebody happen to be looking at them or they have this down scale and up scale alarms, so whenever they get up or outside the ranges there is an alarm sounded in the control room. So when the operator hears every two seconds, beep, beep, beep, then he says, there is something wrong here and I will go and watch. So there's a possibility of looking -- of seeing them.

MR. MICHELSON: But you can get some fairly large oscillations without doing that.

MR. MARCH-LEUBA: You must have fairly large oscillation before you reach them. And that's why, really, we are -- all the work is being done by the Owners Group on what we call the long-term resolution is addressed. Because the in-phase oscillation is already solved.

An out-of-phase type oscillation, once you get to 120 or if you have a flow by a scram, you scram.

MR. MICHELSON: Is it possible since there are a limited number of monitors, is it possible you can have oscillations that the monitors are not even seeing effectively, large oscillations?

MR. MARCH-LEUBA: That is the third type of instability, the channel instability. And you are absolutely right. If you have a large flow oscillation in a channel and if you account for a few failed LPRMs, there's a chance you won't be able to see it.

MR. MICHELSON: If you failed once. You don't even have to be failed and you may not see it. I mean, where they're located.

MR. MARCH-LEUBA: Glen has some calculations.

MR. WATFORD: Glen Watford from GE.

For a core of the size of LaSalle there is 176 LPRMs, but I think there is a large number and we have done some calculations for channel oscillations to show that the surrounding detectors, if you take any bundle, within three to four bundles away from it are going to be four strings of detectors which are 16 detectors. And even if just a single bundle is oscillated by the time it gets up to the magnitudes they begin to approach the safety limit, there is going to be considerable detection from the LPRMs.

MR. MICHELSON: I know, that's fine and good. The real problem is, do you presently have monitoring systems that are looking at these local areas instead of looking more at the average?

MR. WATFORD: The LPRM --

MR. MICHELSON: Is the information being processed to bring to the attention of the operator what's happening locally.

MR. MARCH-LEUBA: The LPRM alarms clearly will not protect you to monitoring the channel oscillation if you are far away from it. I mean, until your LPRM sees 120 percent, probably the flow in the channel it might be 300, 400. I mean, that has to be addressed.

As far as your question, the monitor -- the correction actions on there, what is called Co380 calls for the operator to be looking at least 9 LPRMs, whenever they are in region in which instabilities are possible.

So there is no automatic protection, but the operator is supposed to be looking for them; whether he is doing that or not is --

MR. MICHELSON: How does he look for them?

MR. MARCH-LEUBA: He goes to the LPRM channel and selects it.

MR. WATFORD: He has the capability to individually select a monitor from 1 to 4 LPRM strings.

MR. MICHELSON: Somehow he has to be alerted to the fact he ought to be doing this.

MR. WATFORD: He is alerted by being in the region where oscillation is essential. And in today's world there are the interim corrective actions --

MR. MICHELSON: Isn't he kind of busy and he also at the same time has to be manually monitoring; is that right?

MR. WATFORD: Today the requirements is if the reactor ends up in that region, his first action is to get out of the region. In some plants that requires him to scram the reactor, depending upon the plant design. Other plants that includes inserting control rods or if it's possible to increase core flow by increasing core speed. So his first action is, I'm just going to leave this region.

MR. MARCH-LEUBA: And that is the main reason for all this work that has been going in the last year on the Owners Group on the long-term solution is so that the operator doesn't need to do anything; that there is an automatic protection. Because we have automatic protection for only one of three types of instabilities.

[Slide]

MR. MARCH-LEUBA: In ATWS really, again, the key question is: what is the maximum amplitude of limit cycle oscillations. And as it was pointed out before we're not interested on 2.2 versus 2.4, it is 2 versus 20.

The question is: if that amplitude oscillations crosses it with another power increase does it affect suppression pool temperature in ATWS. If the power is not allowed to increase because you control the feedwater flow there is to flow decrease; and the question is, can you still cool your reactor at 80 percent power with this flow. And that, if one needs to -- we need to know, is it possible to have a failure there.

The other question that is kind of more global is, what is the effect on limit cycle oscillation ATWS procedures through the effect of instrumentation. Is the oscillation masking what the operator sees and whether it will cause the operator to act wrongly because he gets scared, he sees an oscillation of 20,000 percent or something and he doesn't know what to do.

That is the end of my presentation.

MR. CATTON: If there are no further questions, and I see none.

Thank you very much.

Larry has to summarize.

MR. PHILLIPS: This will only take a minute.

[Slide]

MR. PHILLIPS: So to summarize, at this stage of our review we feel acceptable methods to provide high assurance of -- that should say, conformance to GDC-12 for evolving core designs have been identified. And we expect that details of design and implementation will be defined in the near term, that is the early 1990.

I would just like to make one other comment concerning this review, I think probably the most difficult aspect of the things that are being proposed does relate to the use of the stability monitor to define the out-of-phase stability decay ratio. It now computes the core decay ratio for in-phase oscillations.

The Owners feel that they have defined a way to convert that to the out-of-phase oscillations through a relationship between the core decay ratio and the channel decay ratio. We need to see details of that and that's probably the most difficult aspect or the most questionable aspect of review that has been identified so far.

We feel that this is a big step forward and that the industry has been very forward in this effort. We feel that they don't want oscillations; they want to remove this as an issue. They don't want the uncertainties involved with oscillations plus a notoriety when a plant goes into large power oscillations; and they have been very positive we feel with this aspect.

I would also like to point out that the solution of the normal operation problem does improve some aspects of the ATWS situation, too. For instance, if you look at the LaSalle event for any comparable loss of flow event, it wouldn't have had to be initiated the same way. The operator, if an ATWS had existed the operator wouldn't have known that until he was seven minutes into the event with large oscillations.

As it stands right now with the proposed solution, particularly exclusion region solution or even the current interim solution, when he goes into the exclusion region he tries to scram the reactor. So that gives him a seven minute head start on a LaSalle type situation. He knows he's got an ATWS and he enters his ATWS procedures. So it does have advantages in that respect, also.

Otherwise as it has been pointed out very much, we've also reached a conclusion on maximum amplitude that it's either 2100, 500 or 200, and although GE doesn't claim that to be a bounding case only for the particular core that they were analyzing, they had all the rods pulled.

We do know -- we feel confident now that the average thermal power increase related to the oscillations is on the order of 2 percent times the neutron flux peak power. In addition, you have whatever effects that cold feedwater or whatever else in the systems is contributing to power rise. So that's a piece of information we feel we have a handle on.

The effect of the oscillations on ATWS is still not determined. More analyses are planned. We haven't completely identified the exact scenarios that need to be evaluated at this point. We have to consider -- and we know that the Owners are doing some work on their own in this regard, also. They're not committed to anything but they haven't stopped.

And we need to identify those scenarios which really place the limits on what suppression pool -- what will heat up the suppression pool temperature to the limit and the limit being either effects on -- usually effects on the NPSH on ECCS pumps and, of course, if it gets higher we need to worry about containment integrity, also.

We have the problem of identifying the amplitude and identifying the scenarios we have to make our application to and also see if the boron injection will shut us down in using the current ATWS procedures and can get us out of the worst oscillation scenarios.

In this regard we have to consider also the effects of the flow, large flow oscillations, large power oscillations; on the instrumentation what the operators are seeing and how this may impact his response to an ATWS.

MR. CATTON: Thank you.

We have a question here.

MR. LEE: One of the concerns I have had regarding this density wave oscillation related to ATWS situation is, not only the uncertainties we have in predicting the magnitude of oscillations and things like I mentioned early-on, but also the possibility that operator, all due to system malfunction, the system takes off on a different transient trap when the reactors aren't doing large amplitude to oscillation, what would be the consequences? I don't think that has been brought up in our discussion today so far.

MR. PHILLIPS: I'm not sure I followed you. When the system takes off -- would you elaborate, please.

MR. LEE: Maybe there is a matter of small insertion on part of reactivity on top of the undergoing oscillation.

MR. PHILLIPS: Yes. Well, that's really the type of thing I had in mind when I said we have to identify the type of scenarios we have to consider in this, also. I'm not sure we can just look at the conventional ATWS scenarios. For instance, LaSalle wouldn't have been covered with that type of scenario. With LaSalle, say you had a scram when you reached 118 percent point, is it really wise -- if you're in very large oscillation is it wise to leave the reactor operating. That's certainly best as far as worrying about suppression pool goes or anything like that. But would you want to leave it open, wide open like that with a very large oscillations and your concern with fuel failures and so forth, and the operator is going to be pretty nervous, I would think.

MR. LEE: So one hypothetical scenario regarding the LaSalle event was, if the operator had indeed been successful in restarting one of recirculation pump momentarily they might have introduced some visible positive reactivity; and then what would happen, I just don't know.

MR. PHILLIPS: Well, I think we have -- Research has looked at that and we pretty much concluded it was scrammed. Yes, it would have been -- for that event it was scrammed.

MR. LEE: But I mean, ATWS situation so it had been scrammed.

MR. PHILLIPS: Yes.

MR. LEE: I get it.

MR. PHILLIPS: Even there with that particular scenario it would have probably introduced a reactivity spike, but the larger flow would have stabilized the reactor.

MR. CATTON: I would like to take a few minutes break before we continue this. I have to check out.

So maybe we'll take a 10 minute break.

[Brief recess]

MR. CATTON: Craig, the schedule shows that you're to give a 45-minute presentation, is that about right?

MR. SAWYER: That's about right.

MR. CATTON: Well, what I would like to recommend then is that we --

MR. SAWYER: I could make it a half-hour if you preferred.

MR. CATTON: We will break for lunch no later than 12:15, so that way we can start at 1:00 I guess for the proprietary session.

So whatever takes over a half-hour is eating into the lunch hour.

MR. SAWYER: This is a little bit of a change of pace because now we're going to be talking about the ABWR, LOCAL evaluations.

MR. CATTON: Yes.

MR. PLESSET: I have a question since we're leaving the question of the BWR instability. I've heard a great deal about this phenomenon when I came to this meeting. First, about the Leibstadt event and I presume that was reported to NRC; I'm sure that it was. I won't ask how it was reported, was it reported as a significant event or not?

MR. PFEFFERLEN: We met with the NRC to discuss the details of all these events as they came up. It was evaluated on the Part 21 and I can't recall exactly what the resolution was. But there were discussions with the Staff on results that we had.

MR. PLESSET: Well, that's something.

Now, I turn to NRR, what was their reaction to this event, did they feel it significant or not?

MR. PHILLIPS: Leibstadt?

MR. PLESSET: Yes. That's over five years ago.

MR. PHILLIPS: Yes. Yes, we did and we had -- the main thing that kept us from reaching an earlier resolution to the generic issue on thermal hydraulic stability was an answer to the question of whether there could be local instabilities, and it got very large and it wouldn't necessarily resulted in an APRM scram.

I guess after Vermont Yankee tests where there were in-phase oscillations we were nearly convinced that we should put the issue to bed like it was. And before we did that, fortunately, Leibstadt event occurred, so we did recognize that here were out-of-phase oscillations which could possibly cause core damage and would not result in an APRM scram and that was considered and accounted for in our resolution but not perfectly.

MR. PLESSET: I guess most of the ACRS is in ignorance on this matter. I presume it was. But it does seem to me a lot of work could have been done in the past five years that was not done.

Would you agree with that?

MR. PHILLIPS: Surely more work could have been done. It was addressed in the B-46 resolution, specifically. As I say, our big problem was that core decay ratios were calculated right. The resolution of B-46 was don't get into instability, stay out of the unstable region. Where more work should have been done was being sure that we did that properly.

The resolution didn't work, but the principle was okay and accounted for the asymmetric oscillations.

MR. PLESSET: That's all I wanted to know, Mr. Chairman.

MR. CATTON: Thank you.

Craig, why don't you initiate your talk.

MR. SAWYER: Okay. As I said, this is a little bit of a change of pace, the subject isn't stability anymore, it's ABWR and the LOCA analysis and the qualification of the program that has backed that up.

The first part of that is nonproprietary which we're covering this morning, which is a description of the ECCS configuration that the ABWR has.

In the interest of time I'm going to skip the lead-in chart which is basically historical.

MR. CATTON: Is there a handout for this talk?

MR. SAWYER: There is a handout and it should be in your package.

MR. MICHELSON: We were back in the book again.

MR. CATTON: Thank you.

MR. MICHELSON: Good; thank you.

MR. SAWYER: The first chart which I'm skipping was basically historical and told you a little bit about how we got there.

[Slide]

MR. SAWYER: What this chart does basically is give you a comparison of the ECCS networks for typical BWR-4s, typical BWR-5 and 6s and the ABWR. The ABWR basically -- it's evolutionary, as you can see, when we went from BWR-4 to 5 we added basically what I call a half a division, which was a single high pressure core spray system powered by its own diesel.

In the ABWR we've gone with three complete divisions with high pressure and low pressure makeup in each division. The RCIC steam driven system has been upgraded to full ECCS and represents the high pressure makeup in one of the three divisions.

Now, usually when I give this talk we've previously described the whole plant and we're not doing that this time, but I should point out that because we have internal recirculation pumps and no external recirculation piping, particularly large piping located below the top of the core as exists in the jet pump plants. The size of the largest LOCA which can occur is significantly reduced and therefore the capacity of the required ECCS, particularly the reflood capacity is also significantly reduced, as you can see from the numbers that are in the ABWR chart relative to the previous plants.

There aren't any large pipes located below the core. The largest single pipe located below top, of course, two-inches in diameter.

The peak clad temperature, we'll get into that in our proprietary discussion. But basically the -- taking advantage of the lack of large recirculation piping attached to the vessel, we don't predict any core uncovery for the design basis accidents.

It has almost N minus 2 capability. There is one combination out of about 500 double failures that you can take wherein you're not covered, but other than that from a -- even from a licensing point of view it's close to being able to claim N minus 2.

[Slide]

MR. SAWYER: This is a little bit simplified PNID which gets into a little more detail of how it's laid out. The two divisions of high pressure core flooder -- for those of you who are maybe confused, we dropped the core spray in the ABWR. Some of the early design studies that you may have been aware of, in those studies we had a core spray system but in the current configuration that Staff is reviewing the core spray has been replaced with a core flooder.

This is high pressure makeup, typically at the safety valve set point. And this is the makeup that you would get typically at around 100 psi. It takes feed either from -- either of these systems, and there are two of them in separate divisions, will take feed either from preferred condensation storage tank or upon automatic logic from the suppression pool for recirculation and injection inboard of the core shroud above the core, but not in the spray mode, just in the flood mode.

The RCIC is pretty typical of RCICs that previous plants have had. Its preferred mode is also from the condensation storage tank, but it can transfer to recirculation from the suppression pool, also. Its injection poured into the reactor is by one of the feed lines.

There is a separation in level here. The RCIC is a dual system in a sense that it response in normal transients, also, to events such as loss of feadwater. And it's the primary transient defense for makeup when you don't have feed pump makeup. This comes on at what GE calls level two which is the first lower level than the scram level in the plant.

These act as backup in case the RCIC fails. And these two systems come on at an intermediate level, a new level which only ABWR has which we call level one and a half; the previous plants only have level two and level one.

So the intent of this coming on at a higher level is to reduce the demand of these ever needing to come on, because this has to fail first; whereas, in the previous plants all of those systems are initiated at the same water level.

Then the intent of this intermediate level is to avoid actually getting to the so-called level one or ECCS action level which would require the full complement of ECCS to come into play.

MR. LEE: What is the significance that you just mentioned about RCIC versus cold flooder, that RCIC would come on first?

MR. SAWYER: The significance is that in the current plants if you have a loss of feedwater you are bound to get to level two, you have no choice.

In the current plants the RCIC steam driven pump and the HPCS in BWR-5 and 6 or HPCI in BWR-4s both will come on because they're both triggered at the same water level. So there's a reduced demand or reduced duty, if you will, on the motor driven makeup systems; they should be required to come on much less often.

MR. LEE: But in terms of containing of potential LOCA, is there a difference in terms of that particular function?

MR. SAWYER: One of the reasons why we were able to lower the initiation level for the HPCS is because of the reduced size of the largest break. So this is a strategy we can employ in this product which we couldn't employ in the earlier products.

[Slide]

MR. SAWYER: The low pressure systems have a capacity of about 4300 gallons per division. Because the reflood rate is much lower relative to current plants, there is a much better match between this flow rate that we need for ECCS duty with the flow rate which you would need just for RHR or other shutdown cooling functions. In the current plants the reflood rate is much higher than the flow you really need for RHR.

The heat exchangers are always in the loop in this design. We've designed this whether it is running in RHR shutdown cooling mode which is shown by this pathway here or whether it is running in LPCI mode which is this pathway here. The heat exchanger and, of course, the secondary side which I didn't show is always in the loop. So this removes the operators dilemma of, should I add water to the vessel or should I remove heat which he has to do in current plants; that choice doesn't have to be made anymore.

I've shown a couple of the auxiliary support that is occasionally called up. For example, fuel pool or the capability for wetwell or drywell spray. The parenthetical 2s there mean that those functions are three divisional. This is triplicated except for where I've indicated a 2, in those functions there is only two divisions of support.

MR. MICHELSON: You made a point that you don't have to decide whether to remove heat now since the heat exchanger is always in the loop but there are two modes of this operation. One is to low pressure flood the vessel; and the other is to cool the wetwell. And, of course, you do have to decide which one you want to do or a combination of what you want to do.

MR. SAWYER: I don't think so. Let me go through that.

Well, first of all, let's ignore the shutdown cooling mode which you only do when you're shutdown. I think you're worried about this line here which is used for suppression pool cooling, which you might use, for example, if you had a leaking relief valve that was heating up the pool.

If you're in true LPCI mode then you are taking the water from the suppression pool, removing heat from it, pumping it through the vessel, it's going to fall out through the break, and then through the vent system not shown in this drawing and it's going to end up back in the pool again.

MR. MICHELSON: That depends on who big the break is as to whether that's an effective means of utilizing your heat exchanger. If the break isn't big enough, then you have to -- and you want to cool the pool, you have to get a larger pool and you have to do that by operator action.

MR. SAWYER: Well, eventually you're going to be at low pressure when you have a break on your hands. And once you get to low -- I see what you're saying, if the break doesn't allow enough water to come out of it; then, of course, you will be cutting back the capacity of these systems to handle the break.

MR. MICHELSON: To handle the heat. I don't handle the break.

MR. SAWYER: Right.

MR. MICHELSON: You're keeping the core flooded, you just aren't taking the heat out.

MR. SAWYER: That's correct. For a small break situation, then eventually the operator is going to have to, at that point, somewhere down stream divide his attention because for a small break you only need one of the systems to keep up with the break anyway. You're right.

[Slide]

MR. SAWYER: This chart shows an elevation view of ABWR versus the BWR-5 and 6, an elevation view of where the nozzles are. ABWR has internal recirculation pumps which the BWR-5s and 6s don't have; they have jet pumps.

Feedwater nozzle, the RHR, LPCI injection, the high pressure core flooder injection elevation is here. Shutdown cooling suction elevation is here. So this pipe here is really the lowest elevation in large pipe. The top of core is here.

[Slide]

MR. SAWYER: This is a rather wordy chart, I don't intend to go into it in detail; it's for your reference. It basically shows azimuthally the PNID layouts for the three divisions. This is one division boundary; another division boundary; and that's the third division boundary. And it shows, for example, whether the suction is taken from the suppression pool as indicated by dropping the line off here because this is the containment boundary. This is the RPV boundary and that's the shroud boundary, at least shown in sketch here. And it just shows azimuthally how this is all arranged and its attachment to the vessel.

MR. MICHELSON: Question.

MR. SAWYER: Yes.

MR. MICHELSON: I didn't notice it before, but why is the ABWR -- the main difference in the length is down in the lower plenum, what was the reason for that compared with the 5, 6? It's a longer vessel and it's all -- apparently, the core is about the same elevation at least in your pictorials; and yet, the vessel is much longer at the bottom. What was the reason?

MR. SAWYER: I guess I don't draw that conclusion from my looking at the sketch. I don't think there is much of a difference. If it is, it's small.

MR. MICHELSON: Yes, I think you can easily see that it's bigger.

MR. SAWYER: It can be an artist --

MR. MICHELSON: It could be just the picture.

MR. SAWYER: Yes, it could be the artist rendition. I don't think that this distance from here to here and this distance from here to here is much different.

MR. MICHELSON: I didn't think it was and that's why I asked the question. It suddenly occurred to me and I didn't notice it.

MR. SAWYER: It could be an optical allusion because of the presence of the recirc pumps.

[Laughter]

MR. LEE: Another question.

MR. SAWYER: Yes.

MR. LEE: Unrelated to ECCS, but is there a standby liquid control system?

MR. SAWYER: Yes, there is. In fact, the nozzle layout drawing shows the standby liquid control system in this division injecting basically by means of one of the high pressure core flooder lines directly into the shroud.

MR. LEE: Thank you.

MR. SAWYER: Let me just summarize in words some of the things that you're going to hear more about after lunch when you see the analysis results.

We've gone to three completely separate mechanical and electrical divisions for the most important functions, which are the core cooling function, the suppression pool cooling function, and the shutdown cooling function. I don't know if you noticed it, it was a little bit subtle at the time, the current plants have for shutdown cooling a single shutdown cooling suction nozzle. And that's one of the reasons why current plants have the so-called alternate shutdown cooling mechanism when you get to low pressure and require your passing water through the valves.

We have three shutdown cooling lines. Each one of those divisions in the low pressure system has its own shutdown cooling suction nozzle. So that's why this function is talked about in terms of three completely divisions.

The heat exchangers are always in the loop, as I said. Some of the things that I didn't talk about which we done is, we've simplified the low pressure network considerably. Steam condensing is gone; RPV head spray has been transferred to the cleanup system; containment flood has been transferred to one of the service water systems, so these functions which were previously appended to the RHR system and therefore made it mechanically a little more complex in terms of the operator's perform, we have just eliminated them from the low pressure ECCS functions completely.

What this has done is reduce the number of valves, the number of pipes in the systems by about a third.

You saw the significant capacity reduction which leads to reduced equipment sizes which came about primarily because of the reduced need for reflood.

We have N minus 2 capability at high pressure, as you saw. We've got three high pressure makeup pumps in addition to feedwater whereas the current products have two.

One of the things that I didn't mention while I was at it was, we've improved the small break response. We've actually done PRA type analyses, best estimate where we've shown that even if the ADS completely fails one of the HPCF pumps is sufficient to keep the core from getting even to

Appendix K limits. So that's an enhancement over what the current products have.

And, of course, for the design basis accidents we don't predict any fuel uncovery.

So in summary, I think that gives you the background which you will need when we get into the LOCA analysis results that we will talk about after lunch.

MR. CATTON: Are there any questions?

We're a little bit early for lunch.

MR. MICHELSON: You mean that's it, that's all.

MR. CATTON: Before lunch it looks like that's all.

MR. PHILLIPS: It would not have been really lost information. Basically, I was to address the status of the Staff review.

MR. MICHELSON: Excuse me, is this a handout?

MR. PHILLIPS: It's on the last page of the previous handout.

We have reviewed the ABWR ECCS performance. We've completed our review and we agree with the conclusions that there is no core uncovery with the ABWR design with only one-inch and smaller pipes below the top of the core. That they do use approved LOCA analysis methods. That the analysis complies with 10CFR50.46 and Appendix K.

In response to some questions that were raised at the last ACRS subcommittee meeting regarding differences between the Appendix K analysis and the best estimate analysis they redid their Appendix K analysis. There were some changes to their Appendix K analysis. I think the main effect was taking stored energy plus two Sigma in the fuel.

On their new calculation the difference was only two degrees between best estimate and Appendix K; I don't recall which was the highest.

But the peak clad temperature is 1149F, and I'm sure they plan to address this in more detail in the proprietary session.

So our conclusion is, it's acceptable.

MR. CATTON: Do they have to do both, Appendix K and best estimate calculations?

MR. PHILLIPS: No.

MR. CATTON: I thought under the new rule they didn't have to?

MR. PHILLIPS: No, they don't have to.

MR. CATTON: They just did it to be good people?

MR. PHILLIPS: Right.

MR. SHIRALKAR: Can I address that. If you're not applying the terminology in SECY-82-472 which was within Appendix K. But also having an evaluation to make a comparison.

MR. CATTON: You have a best estimate capability. The question that I raised last time was, why don't you just use it? You don't want to go through the exercise?

MR. SHIRALKAR: That's correct. Rather than go through the new rule procedures we decided to stay with the improved development.

MR. LEE: Question.

MR. PHILLIPS: Yes.

MR. LEE: Just for the purpose of comparison, what is the peak clad temperature that has been obtained for BWR-6 line?

MR. PHILLIPS: On the old BWR -- I don't recall on BWR-6, but on the BWRs in general the peak clad temperature using Appendix K was approaching 2200 limit on some of them.

What was BWR-6?

MR. SHIRALKAR: It's about the same order of magnitude as ABWR, around 100 degrees Fahrenheit.

MR. PHILLIPS: But that was using best estimate, wasn't it?

MR. SHIRALKAR: No, using that Appendix K.

MR. SAWYER: The difference here in using ABWR because of the analytical assumption is to have all pump trip at times zero. It is not caused from the dryout. I will get into a lot more this afternoon.

MR. CATTON: So you just go into boiling momentarily.

MR. SAWYER: That's correct.

MR. MICHELSON: Question. I gather you have finished the -- well, you've said you have finished the Appendix K type analysis through these conclusions; yet, we haven't seen Chapter 15 safety evaluation report yet. There are a few other things, I guess, in Chapter 15 besides the Appendix K analysis, but I thought most of those had to be done in order to draw the conclusion that's just been drawn. Maybe you or Dino can clarify for me why we haven't had Chapter 15 to evaluate, which was originally scheduled in the first module if the analysis is done?

MR. PHILLIPS: The SER has been completed by the Staff. Dino can maybe say where it is.

MR. SCALETTI: I am Dino Scaletti from NRR.

The fluid problems with Chapter 15 are source term problems and that is one of the reasons why we haven't put the work out yet. There are some other transients that we have to evaluate. We have been looking at them.

I don't think the problems bear on Chapter 5 analysis or excuse me, Chapter 6.

MR. MICHELSON: That's just part of it. Yes. But it will all appear when you do the Chapter 15 SER, I assume.

MR. PHILLIPS: I should have said the reactor systems portions.

MR. MICHELSON: Yes, if you would have said that it would have dawned on me just how far you had gone, because I thought Chapter 15 and you were really saying just Chapter 6 portion or just the reactor systems portion of Chapter 15 is all you have really done.

Okay, that clears it up for me. Thank you.

MR. CATTON: Thank you, Larry.

John, do you have any comments you want to make before you leave?

MR. LEE: Yes. I made most of my points through questions and comments.

But again, I would like to reiterate a point that our ability in predicting large amplitude oscillation is very limited and will remain so for considerable period of time.

So we need to look at the possibility of avoiding these large amplitude oscillations as well as analyzing the consequence of such large amplitude oscillations.

I mentioned in particular the possibility of these dense wave oscillation, these induced transients, possibly coupled with an ongoing transient or possibly triggering some other kind of transient. They may give us a much more difficult problem.

I would like to also suggest that perhaps for validation of some of these nuclear thermal hydraulic coupled calculations one might go back to some of the old reactivity related transient tests performed, for example, at facilities maybe 20, 30 years ago, which for pressurized water reactor environment has seen fairly large power spikes and safety.

Some of these modern production code could accomplish reasonable simulation of reactivity, induced transient with thermal hydraulic feedback.

I misspoke myself about the reliance on exclusion region for immediate resolution of the dense wave oscillation phenomena. I mentioned that inclusion region could shrink -- what I was visualizing was expand considerably with all the uncertainty accounted for. So you may just get into that region so frequently, and that is something that BWR Owners would dearly love to avoid. So we may have to take a slightly different approach in the short-term. That's what I would like to seriously suggest.

And reverse flow with oscillation superimposed on it. As I said, I have very little confidence in our ability to predict, especially with the large amplitude oscillations.

That's all I have.

MR. CATTON: Thank you.

If you could send us a brief note.

MR. LEE: I'll try to summarize a little bit better.

MR. CATTON: I would appreciate that.

Thank you.

It's 12 o'clock, let's break for lunch.

[Whereupon, at 12:00 p.m. a hearing was recessed for lunch to reconvene later this same day.]

REPORTER'S CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission

in the matter of:

NAME OF PROCEEDING: ACRS Thermal Hydraulic Phenomena Subcommittee

DOCKET NUMBER:

PLACE OF PROCEEDING: San Francisco, CA

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.

Dean A. Robinson

Dean A. Robinson

Official Reporter
Ann Riley & Associates, Ltd.

INTRODUCTORY STATEMENT BY THE THERMAL HYDRAULIC PHENOMENA SUBCOMMITTEE
CHAIRMAN'S REPORT
NOVEMBER 9, 1989

The meeting will now come to order. This is a meeting of the Advisory Committee on Reactor Safeguards Subcommittee on Thermal Hydraulic Phenomena.

I am I. Catton, Subcommittee Chairman.

We will continue with the meeting.

INTRODUCTORY STATEMENT BY THE THERMAL HYDRAULIC PHENOMENA SUBCOMMITTEE
CHAIRMAN'S REPORT
NOVEMBER 8, 1989

Copy in

The meeting will now come to order. This is a meeting of the Advisory Committee on Reactor Safeguards Subcommittee on Thermal Hydraulic Phenomena.

I am I. Catton, Subcommittee Chairman.

The ACRS Members in attendance are: J. Carroll, C. Michelson and D. Ward .

We also have ACRS Consultants: C. Corradini, J. Lee, M. Plesset, V. Schrock, H. Sullivan, and C-L. Tien.

The purpose of this meeting is to discuss: (1) the capability of the thermal hydraulic codes to model BWR core power instability, and (2) the key thermal hydraulic design aspects of the GE ABWR related to the ECCS, and LOCA analyses.

Mr. P. Boehnert is the cognizant ACRS Staff Member for this meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the Federal Register on October 24, 1989.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register Notice. It is requested that each speaker first identify himself or herself and speak with sufficient clarity and volume so that he or she can be readily heard.

We have received no written comments or requests to make oral statements from members of the public.

I have

(Chairman's Comments - if any)

We will proceed with the meeting, and I call upon Dr. Berhat Shiralhar of the General Electric Company to begin.

STABILITY MODELING OVERVIEW

(ATWS)

R. C. STIRN

GE NUCLEAR ENERGY

AGENDA

- OBJECTIVE
- ATWS OVERVIEW
- APPROACH
  - METHODS DEVELOPMENT QUALIFICATION
  - METHODS APPLICATION
  - EVENT SIMULATION
- EVALUATION STATUS

OBJECTIVE

O DEMONSTRATE COMPLIANCE TO THE ATWS BASES FOR ATWS EVENTS WHICH COULD POTENTIALLY RESULT IN T-H LIMIT CYCLE OSCILLATIONS.

APPLY:

1) APPROVED ATWS DESIGN/LICENSING BASES

2) TYPICAL PLANT RESPONSE/NOMINAL CONDITIONS

3) EMERGENCY OPERATING PROCEDURES - EPG REV. 4

QUESTIONS:

WHICH ATWS EVENTS ARE POTENTIALLY SUSCEPTIBLE TO LIMIT CYCLE OSCILLATIONS?

IF OSCILLATIONS OCCUR, WHAT ARE THE CONSEQUENCES?

ATWS OVERVIEW

O FAILURE TO SCRAM PROBABILITY

- $< 10^{-7}$ PER DEMAND = GE NEDE-21514
- $3 * 10^{-5}$ PER DEMAND W/O ARI } NRC NUREG-0460 SEC 4.2 VOL I
- $1 * 10^{-5}$ PER DEMAND W/ARI } NRC NUREG-0460 SEC 4.2 VOL I

O ATWS EVENT PROBABILITY

- EVENT PROBABILITY * FAIL TO SCRAM PROBABILITY

O ATWS DESIGN AND LICENSING BASES

- TYPICAL PLANT @ NOMINAL CONDITIONS (NEDE-24222 VOL II)
- BASES (NUREG-0460 SEC 7.1 VOL I)
  = PRIMARY PRESSURE BOUNDARY < EMERGENCY PRESSURE LIMITS
  = PRIMARY CONTAINMENT PRESSURE < DESIGN LIMITS
  = MAINTAIN COOLABLE CORE GEOMETRY
  = RADIOLOGICAL RELEASE WITHIN 10CFR100
    - ASSUME 100% OF FUEL PERFORATES
    - 10% IODINES AND NOBLE GAS RELEASE

ATWS OVERVIEW (CONTINUED)

- ATWS EVENTS
  - 15 EVENTS EVALUATED
  - INCLUDED
    - MSIV CLOSURE
    - TURBINE TRIP
    - GENERATOR LOAD REJECTION
    - PRESSURE REGULATOR FAILURE
    - FW CONTROLLER FAILURE MAXIMUM DEMAND
  - ONE AND TWO RECIRC PUMP TRIPS NOT EVALUATED
    - NON SCRAM EVENT PER FSAR

APPROACH TO ATWS/STABILITY CLOSURE

0 INTEGRATED PROGRAMATICALLY (BWROG, EPRI, GE)

0 INTEGRATED TECHNICALLY

QUANTITATIVE	AS COMPARED TO	QUALITATIVE
DISCIPLINED SYSTEMATIC QUALIFIED		SIMPLIFIED ENGINEERING CALCULATIONS

0 THREE KEY ELEMENTS

- METHODS DEVELOPMENT QUALIFICATION
- METHODS APPLICATION
- EVENT SIMULATION

METHODS DEVELOPMENT/QUALIFICATION

0 STEPWISE APPROACH

- SEPARATE EFFECTS
- COMPONENT
- SYSTEM
- PLANT

0 PHENOMENOLOGICALLY/REPRESENTATIVE TEST DATA

0 PRECISE EVENT SIMULATION

METHODS APPLICATION

0 MODEL SIMULATION SENSITIVITIES

- NUMERICAL SOLUTION

- NEUTRONIC MODEL
  = POINT KINETICS
  = 1-D
  = 3-D

- THERMAL-HYDRAULIC MODEL
  = NUMBER OF CHANNELS
  = CHANNEL GROUPINGS

0 INPUTS

- SEPARATE EFFECTS
  = NUCLEAR
  = THERMAL-HYDRAULICS

- COMPONENTS

- SYSTEMS

EVENT SIMULATION

- 0 PLANT RESPONSE
  - CONSISTENT WITH PLANT DESIGN
  - PLANT MODIFICATIONS PER ATWS RULE
- 0 OPERATOR RESPONSE
  - CONSISTENT WITH PLANT PROCEDURES
    - = NORMAL
    - = EMERGENCY
- 0 APPROPRIATE EVENT SCENARIO

EVALUATION STATUS

- O NUMERICAL METHODS
- O THERMAL-HYDRAULIC STABILITY
- O TRANSIENT MCPR
- O PLANT STABILITY QUALIFICATION *
- O HARMONIC MODES *
- O NEUTRONIC MODEL SENSITIVITY *
- O ATWS SCENARIO DESCRIPTION
- O TYPICAL ATWS ANALYSIS *

* WORK IN PROGRESS

SUMMARY

- O SYSTEMATIC/DISCIPLINED APPROACH FOR ISSUE CLOSURE
  - DEVELOPMENT/QUALIFICATION
  - APPLICATION
  - EVENT SIMULATION
- O ESSENTIALS FOR QUALIFICATION/CLOSURE
  - QUALIFIED MODEL FOR SPECIFIC APPLICATION
  - 3-D CORE SIMULATION
  - NUCLEAR AND T-H INPUTS
  - COMPONENTS
  - SYSTEM EFFECTS
  - EVENT SIMULATION
- O WORK IS IN PROGRESS

TOTAL SYSTEM QUALIFICATION FOR SPECIFIC PHENOMENON APPLICATION

NRC-RES BWR STABILITY RESEARCH PROGRAM

Harold H. Scott
Reactor & Plant Systems Branch

ACRS Subcommittee Meeting
San Fransisco, CA
November 8, 1989

THE OBJECTIVE OF THE BWR INSTABILITY RESEARCH PROGRAM IS TO:

1) PROVIDE A DESCRIPTION OF PHENOMENA RELATED TO INSTABILITY AND THEN DETERMINE THE INFLUENCE OF IMPORTANT PLANT DESIGN AND OPERATING PARAMETERS ON OSCILLATION MODE, AMPLITUDE, AND FREQUENCY TO ANSWER: WHAT ARE THE SENSITIVITIES IN AMPLITUDE AND FREQUENCY TO PERTINENT PARAMETERS, WHAT CONDITIONS RESULT IN LARGE AMPLITUDE LIMIT CYCLE OSCILLATIONS, AND WHAT CAUSES CORE-WIDE VS ASYMMETRIC OSCILLATIONS;

2) REVIEW THE RELEVANT NRC CODES AND THEN ASSESS THEIR USEFULNESS FOR ANALYZING INSTABILITY (SEE CODE MATRIX CHART);

3) DETERMINE CODE VALIDATION REQUIREMENTS AND THEN PERFORM THE CODE ASSESSMENT CASES TO ANSWER HOW WELL DO THE CODES REPRODUCE FRIGG TEST DATA AND THE LASALLE EVENT DATA;

4) EVALUATE BWR RESPONSE TO AN ATWS EVENT, INCLUDING THE APPROPRIATENESS OF ATWS PROCEDURES, TO ANSWER UNDER WHAT CONDITIONS IN A POSTULATED ATWS WILL OSCILLATIONS OCCUR;

5) IMPROVE THE NRC STAFF'S TECHNICAL REVIEW AND AUDIT CAPABILITY FOR INDUSTRY SUBMITTALS RELATIVE TO BWR STABILITY.

CODE USE MATRIX

Use	RAMONA-3B	TRAC-BF1	HIPA(EPA)	LAPUR	M-L Model
Amplitude of oscillation (LaSalle w/o scram)	X	X	X		
Maximum amplitude (core-wide) reactivity & power oscillation	X		X		
Maximum amplitude (asymmetric) reactivity & power oscillation	X				
ATWS scenarios			X		
ATWS sensitivity	X	X	X	X	
Pt vs 1-D kinetics comparison		X			
Stability boundary code comp'n			X	X	
Stability boundary sensitivity			X	X	
Shape/frequency asymmetric osc	X				
Asymmetric vs core-wide osc	X				X

STATUS OF WORK
FY 1989 ACCOMPLISHMENTS

- TPG FORMED; FOUR MEETINGS HELD
- LASALLE EVENT SIMULATED WITH HIPA AND RAMONA-3B
- NODING SENSITIVITY ISSUE ADDRESSED
- FLOW REVERSAL MODELLING DEFICIENCY FIXED IN RAMONA-3B
- ASSESSMENT AGAINST STEADY-STATE FRIGG DATA COMPLETED
- COMPLETED REVIEW OF NRC CODES AND ASSESSED USEFULNESS FOR ANALYZING INSTABILITY
- TRAC-BF1 INPUT DECK COMPLETED FOR LASALLE (POINT KINETICS)
- SENSITIVITY TO KEY PARAMETERS DETERMINED (LASALLE CONDITIONS)
- CORE-WIDE AND ASYMMETRIC OSCILLATIONS DEMONSTRATED

SPECIFIC RESULTS OF ANALYSES

- NRC (AEOD) WAS CONCERNED WHETHER A SEVERE REACTIVITY TRANSIENT CAUSING FUEL DAMAGE COULD OCCUR ON RESTART OF A RECIRCULATION PUMP

- HIPA CALCULATIONS SHOWED THAT RESTART OF THE PUMP COULD LEAD TO A POWER SPIKE THAT WOULD SCRAM THE REACTOR BUT NOT PRODUCE BOILING TRANSITION OR FUEL MELTING. THE POWER SPIKE IS CAUSED BY THE INCREASED FLOW OF SUBCOOLED LIQUID FROM THE LOWER PLENUM AND DOWNCOMER INTO THE CORE

SPECIFIC RESULTS OF ANALYSES (Cont)
CALCULATIONS OF LASALLE SHOWED THAT:

- THE LASALLE CONDITIONS PRODUCE OSCILLATIONS LEADING TO AUTOMATIC REACTOR SCRAM

- THERMAL-HYDRAULIC INSTABILITY AT LASALLE WAS CAUSED BY THE COMBINATION OF 1) AXIAL AND RADIAL POWER PEAKING, 2) FLOW REDUCTION FROM TRIP OF BOTH RECIRCULATION PUMPS, AND 3) FEEDWATER TEMPERATURE REDUCTION FROM REDUCED FEEDWATER HEATING

- THE AMPLITUDE OF POWER OSCILLATIONS REMAINS BOUNDED EVEN IF FAILURE TO SCRAM IS ASSUMED

SPECIFIC RESULTS OF ANALYSES (Cont)

- A SENSITIVITY STUDY WITH THE RAMONA-3B CODE (USING A BWR/4 MODEL) PRODUCED BOTH CORE-WIDE AND ASYMMETRIC NEUTRON FLUX OSCILLATIONS. HYDRAULIC OSCILLATIONS THAT BEGIN IN A FEW HIGH-POWERED FUEL BUNDLES CAN CAUSE ASYMMETRIC POWER OSCILLATIONS DUE TO OUT-OF-PHASE PARALLEL-CHANNEL FLOW INSTABILITIES

- THE OSCILLATION SHAPE IS IMPORTANT SINCE IT EFFECTS THE MEASURED (VIEWED OR RECORDED) LPRM OUTPUT. OUTPUT FROM LPRMs AT PARTICULAR AXIAL AND RADIAL POSITIONS IN THE CORE IS USED TO DEVELOP APRM SIGNALS THAT SCRAM THE REACTOR ON HIGH FLUX

SPECIFIC RESULTS OF ANALYSES (Cont)

- TWO TYPES OF ASYMMETRIC OSCILLATIONS WERE CALCULATED, AZIMUTHAL (SIDE TO SIDE) AND RADIAL (CENTER TO PERIPHERY)

- THE EXCITATION THRESHOLD OF EACH TYPE (MODE) OF OSCILLATION IS A COMPLEX FUNCTION OF THE SUBCOOLING, THE LOCAL CHANNEL (FUEL BUNDLE) INLET FLOW AND BUNDLE POWER, AND THE SUBCRITICALITY OF THE MODE BEING EXCITED

FY 1990 EXPECTED RESULTS

- COMPLETE ANALYSES USING RAMONA-3B (IN THE FULL-CORE MODE) TO DETERMINE ASYMMETRIC OSCILLATION AMPLITUDES FOR NON-ATWS LIMIT CYCLE CASES
- ISSUE REPORT ON CONDITIONS IN BWRS THAT CAN LEAD TO LARGE AMPLITUDE LIMIT CYCLE OSCILLATIONS
- COMPLETE ANALYSES OF LASALLE EVENT WITH NRC CODES, INCLUDING PARAMETER SENSITIVITY STUDIES
- COMPLETE ANALYSES OF RESPONSE OF A BWR TO A SELECTED ATWS SCENARIO, INCLUDING PARAMETER VARIATIONS TO DETERMINE UNDER WHAT CONDITIONS AND OPERATING PROCEDURES OSCILLATIONS WILL OCCUR. THE TPG WILL HELP FORMULATE THE PARAMETER STUDY PLAN
- ISSUE FINAL REPORT DOCUMENTING THE INTEGRATED RESEARCH RESULTS ON BWR STABILITY

SCHEDULE TO COMPLETE & FUTURE PLANS

- ALL BWR STABILITY WORK TO BE COMPLETED IN FY 1990

- NO PLANS FOR FUTURE WORK

FY 90
STATEMENT OF WORK

TITLE:	APPLICATION OF RAMONA-3B AND BNL ANALYZER TO BWR STABILITY	FIN: A3983O CONTRACTOR: BNL SITE: UPTON STATE: NEW YORK
NRC TECHNICAL MONITOR:	H. Scott (FTS 492-3563)	
PRINCIPAL INVESTIGATOR:	W. WULFF (FTS 666-2608)	
BUDGET ACTIVITY:	0601922020	FY90 BUDGET: $400K
FY 1990 WORK PERIOD:	10/01/89 to 9/30/90	

A. BACKGROUND

General Design Criterion 12 (in Appendix A of 10CFR50) states that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillatons which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed. In a memorandum from W. Hodges to L. Shotkin dated June 2, 1988, NRR requested assistance in the form of RAMONA-3B and HIPA calculations. Two questions were posed: (1) What is the potential extent of fuel damage resulting from asymmetric regional neutron flux (power) oscillations if they are not detected and suppressed, and (2) what are the potential implications of instability with respect to ATWS events (where the oscillations might complicate the recovery). A research program was developed to address the questions and to provide an independent review and audit capability for analysis of industry submittals. Four computer codes (each with unique capabilities) are being used, namely, RAMONA-3B, TRAC-BF1, LAPUR, and HIPA. The research is coordinated amongst BNL, INEL, ORNL and University of California at Santa Barbara. The results of the program will be used by NRR to support: 1) review of BWROG solutions for prevention and/or mitigation of power oscillations and 2) review of emergency procedure guidelines for ATWS.

B. OBJECTIVES

The objectives of the BWR instability research program, of which this FIN is a part, are to: 1) provide a description of phenomena related to instability and then determine the influence of important plant design and operating parameters on oscillation mode, amplitude, and frequency, 2) review the relevant NRC codes and then assess their usefulness for analyzing instability, 3) determine code validation requirements and then perform the code assessment cases, 4) evaluate BWR response to an ATWS event, with emphasis on the role of oscillations during the recovery phase and the appropriateness of ATWS procedures, 5) improve the NRC staff's technical review and audit capability for industry submittals relative to BWR stability.

The specific objective of the work funded by this FIN is to assess and apply the BWR Plant Analyzer (HIPA) and RAMONA-3B computer code for understanding of power oscillations in BWRs.

C. WORK REQUIREMENTS

TASKS 1, 2, and 3 were completed in FY 1989.

TASK 4: Program Coordination
Estimated Completion Date. 9/28/1990
Estimated Level of Effort: 2 staff-months

The contractor shall provide the following services: Participate in the Technical Program Group and resolve action items assigned to BNL. Participate in a bilateral meeting with Sweden.

TASK 5: BNL Analyzer Calculations
Estimated Completion Date: 9/30/1990
Estimated Level of Effort: 2 staff-months

The contractor shall provide the following services: Perform calculations with the HIPA code (BWR plant analyzer) as specified in the sensitivity study plan (to be provided by the NRC in December 1989). Submit a draft report on the results of the calculations by March 30, 1990. This report should be prepared to achieve objective number 1 (See Item B above).

TASK 6: RAMONA-3B Calculations for Asymmetric Conditions
Estimated Completion Date: 9/30/1990
Estimated Level of Effort: 10 staff-months

The contractor shall provide the following services: Complete the plan (and cost/time estimate) to utilize LaSalle core neutronics data. Then modify the RAMONA-3B input used for previous calculations and perform full core (191 node) calculations to determine asymmetric oscillation amplitudes for limit cycle cases. Submit a draft report on the results of the calculations by July 2, 1990. This report should be prepared to answer question number 1 (See Item A).

TASK 7: RAMONA-3B Sensitivity Calculations
Estimated Completion Date: 9/30/1990
Estimated Level of Effort: 5 staff-months

The contractor shall provide the following services: In cooperation with the other program participants, assist the NRC in developing a sensitivity study plan. The plan will be developed using process identification and ranking and other elements of the CSAU methodology. Perform calculations with RAMONA-3B as specified in the sensitivity study plan (to be provided by the NRC in December 1989). Perform one additional calculation using the input deck developed in Task 6 to compare with the TRAC-BF1 LaSalle assessment calculation. Submit a draft report on the results of the calculations by June 22, 1990. This report should be prepared to achieve objective number 1 (See Item B on page 2).

TASK 8: Code Documentation and Presentation of Results
Estimated Completion Date: 9/30/1990
Estimated Level of Effort: 3 staff-months

The contractor shall provide the following services: Revise and complete the documentation of the BLEND code. Submit a draft report for review by July 30, 1990. Submit a draft report (by February 27, 1990) on the results of the RAMONA-3B assessment calculations completed in FY 1989. This report should be prepared to achieve objective number 3 (See Item B on page 2). Prepare papers for presentation and/or publication as directed by the NRC Technical Monitor.

D. REPORTING REQUIREMENTS

a. Technical Reports - Final reports from Tasks 5, 6, and 7 should be submitted to the NRC Technical Monitor by

July 31, 1990. These reports will be combined at INEL with reports from the other contractors into a NUREG/CR report. Prepare camera-ready copy of document requested in Task 8 by September 7, 1990.

b. Monthly Business Letter Reports - See Attachment

c. Publications Note - See Attachment

E. FY BUDGET:

	FY89	FY90
PRIOR:	$490K	
OPERATING:		$400K

F. ESTIMATED LEVEL OF EFFORT AND PERIOD OF PERFORMANCE

The overall level of effort is estimated to be 22 staff months over a 12 month period. The starting date for this program is estimated to be October 1, 1989 and the completion date September 28, 1990. (Presentation of results at a professional meeting may occur after this date).

G. MEETINGS AND TRAVEL

To meet the objectives of this FIN, work activities (including participation in TPG meetings) will arise that require travel.
State the number of trips that will be required to perform the proposed work and identify for each trip:
(1) where, who, and how many people;
(2) the length of the stay; and
(3) the purpose of the travel.
If no travel is expected or required, state none. Foreign travel must be addressed separately and approval must be obtained by processing NRC Form 445, in addition to being provided as part of the proposal. The travel identified by the laboratory in their NRC Form 189 proposal, and agreed to by the NRC project manager, shall be considered as approved by NRC execution of an NRC Form 173 accepting the proposal. This approval, however, does not obviate the requirement to submit an NRC Form 445 for foreign travel. Any additional travel to be charged to this project, i.e., travel other than that previously proposed and approved by the NRC, shall be submitted to the NRC project manager for review and approval prior to taking any action which could result in a travel commitment.

FY 90
STATEMENT OF WORK

TITLE: BWR INSTABILITY ANALYSIS

FIN: L10810
CONTRACTOR: INEL
SITE: IDAHO FALLS
STATE: IDAHO

NRC TECHNICAL MONITOR: H. Scott
(FTS 492-3563)

PRINCIPAL INVESTIGATOR: G. Wilson
(FTS 583-9511)

BUDGET ACTIVITY: 0601922020 FY90 BUDGET: $400K

FY 1990 WORK PERIOD: 10/01/89 to 9/30/90

A. BACKGROUND

This program was developed in response to an NRR user request and the results of the program will be used by NRR to support: 1) review of BWROG solutions for prevention and/or mitigation of power oscillations and 2) review of emergency procedure guidelines for ATWS.

General Design Criterion 12 (in Appendix A of 10CFR50) states that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillatons which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed. Following the power oscillation event at LaSalle in March 1988, two questions were posed: (1) What is the potential extent of fuel damage resulting from asymmetric regional neutron flux (power) oscillations if they are not detected and suppressed, and (2) what are the potential implications of instability with respect to ATWS events (where the oscillations might complicate the recovery). A research program was developed to address these questions and to provide an independent review and audit capability for analysis of industry submittals. Four computer codes (each with unique capabilities) are being used, namely, RAMONA-3B, TRAC-BF1, LAPUR, and HIPA. The research is coordinated amongst BNL, INEL, ORNL and University of California at Santa Barbara.

B. OBJECTIVES

The objectives of the BWR instability research program, of which this FIN is a part, are to: 1) provide a description of phenomena related to instability and then determine the influence of important plant design and operating parameters on oscillation mode, amplitude, and frequency, 2) review the relevant NRC codes and then assess their usefulness for analyzing instability, 3) determine code validation requirements and then perform the code assessment cases, 4) evaluate BWR response to an ATWS event, with emphasis on the role of oscillations during the recovery phase and the appropriateness of ATWS procedures, 5) improve the NRC staff's technical review and audit capability for industry submittals relative to BWR stability.

The specific objective of the work funded by this FIN is to assess and apply the TRAC-BF1 computer code for understanding of power oscillations in BWRs.

C. WORK REQUIREMENTS

TASK 1: Program Coordination
Estimated Completion Date: 9/28/1990
Estimated Level of Effort: 3 staff-months

The contractor shall provide the following services: Participate in the Technical Program Group and resolve action items assigned to INEL. Provide input to and coordination of a NUREG/CR report documenting the integrated research results. This report should be prepared to achieve objective number 5 (See Item B above). Participate in a bilateral meeting with Sweden.

TASK 2: TRAC-BF1 Code Assessment
Estimated Completion Date: 3/30/1990
Estimated Level of Effort: 4 staff-months

SUBTASK A:
Complete the assessment calculations of the FRIGG experiments. This includes steady state, transient, and transfer function calculations. Record agreements and differences with the data. Perform comparisons with similar calculations performed at BNL with RAMONA-3B for the purpose of contrasting the TRAC two-fluid model with the RAMONA drift-flux model.
SUBTASK B:
Complete a calculation of the LaSalle instability event by November 28, 1989. Record differences between known (or

assumed) plant parameters and code input values. Compare calculated frequency of oscillations and behavior of power oscillation amplitude with plant data. Then extend the calculation (by assuming a failure to scram) to determine the behavior of the resulting power oscillation limit cycle.
SUBTASK C:
Submit a draft report on the results by February 27, 1990. Include conclusions regarding effects of nodalization and numerical damping. This report should be prepared to achieve objective number 3 (See Item B on page 2).

TASK 3: TRAC-BF1 Code Application
Estimated Completion Date: 6/29/1990
Estimated Level of Effort: 11 staff-months

SUBTASK A:
In cooperation with the other program participants, assist the NRC in developing a sensitivity study plan. The plan will be developed using process identification and ranking and other elements of the CSAU methodology.
SUBTASK B:
Perform code application sensitivity calculations commensurate with the sensitivity study plan to be provided by the NRC. The ATWS calculations will generally require one-dimensional neutron kinetics in order to achieve objective number 4 (See Item B on page 2).
SUBTASK C:
Submit a draft report on the ATWS calculations by March 30, 1990. Submit a draft report on the other sensitivity calculations by June 22, 1990. This report should be prepared to achieve objective number 1 (See Item B on page 2).

TASK 4: Presentation of Results
Estimated Completion Date: 12/12/1990
Estimated Level of Effort: 1 staff-month

The contractor shall provide the following services: Prepare paper(s) for presentation and/or publication as directed by the NRC Technical Monitor.

D. REPORTING REQUIREMENTS

a. Technical Reports
Prepare draft copy of report requested in Task 1 by July 31, 1990 and camera-ready copy by September 21.

b. Monthly Business Letter Reports
Prepare a monthly business letter status report per

the current NRC Manual Chapter 1102 (See Attachment).

c. Publications Note - See Attachment

E. FY BUDGET:

	FY89	FY90
PRIOR:	$400K	
OPERATING:		$400K

F. ESTIMATED LEVEL OF EFFORT AND PERIOD OF PERFORMANCE

The overall level of effort is estimated to be 20 staff months over a 12 month period. The starting date for this program is estimated to be October 1, 1989 and the completion date September 28, 1990. (Presentation of results at a professional meeting may occur after this date).

G. MEETINGS AND TRAVEL

To meet the objectives of this FIN, work activities (including participation in TPG meetings) will arise that require the contractor (and possibly subcontractors) to travel. The NRC Form 189 proposal should state the number of domestic trips that will be required to perform the proposed work and identify for each trip:

(1) where, who, and how many people;
(2) the length of the stay; and
(3) the purpose of the travel.

If no travel is expected or required, state none. Foreign travel must be addressed separately and approval must be obtained by processing NRC Form 445, in addition to being provided as part of the proposal. The travel identified by the laboratory in their NRC Form 189 proposal, and agreed to by the NRC project manager, shall be considered as approved by NRC execution of an NRC Form 173 accepting the proposal. This approval, however, does not obviate the requirement to submit an NRC Form 445 for foreign travel. Any additional travel to be charged to this project, i.e., travel other than that previously proposed and approved by the NRC, shall be submitted to the NRC project manager for review and approval prior to taking any action which could result in a travel commitment.

H. NRC FURNISHED MATERIAL

None

NRC-RES BWR STABILITY RESEARCH PROGRAM

Harold H. Scott
Reactor & Plant Systems Branch

ACRS Subcommittee Meeting
San Fransisco, CA
November 8, 1989

THE OBJECTIVE OF THE BWR INSTABILITY RESEARCH PROGRAM IS TO:

1) PROVIDE A DESCRIPTION OF PHENOMENA RELATED TO INSTABILITY AND THEN DETERMINE THE INFLUENCE OF IMPORTANT PLANT DESIGN AND OPERATING PARAMETERS ON OSCILLATION MODE, AMPLITUDE, AND FREQUENCY TO ANSWER: WHAT ARE THE SENSITIVITIES IN AMPLITUDE AND FREQUENCY TO PERTINENT PARAMETERS, WHAT CONDITIONS RESULT IN LARGE AMPLITUDE LIMIT CYCLE OSCILLATIONS, AND WHAT CAUSES CORE-WIDE VS ASYMMETRIC OSCILLATIONS;

2) REVIEW THE RELEVANT NRC CODES AND THEN ASSESS THEIR USEFULNESS FOR ANALYZING INSTABILITY (SEE CODE MATRIX CHART);

3) DETERMINE CODE VALIDATION REQUIREMENTS AND THEN PERFORM THE CODE ASSESSMENT CASES TO ANSWER HOW WELL DO THE CODES REPRODUCE FRIGG TEST DATA AND THE LASALLE EVENT DATA;

4) EVALUATE BWR RESPONSE TO AN ATWS EVENT, INCLUDING THE APPROPRIATENESS OF ATWS PROCEDURES, TO ANSWER UNDER WHAT CONDITIONS IN A POSTULATED ATWS WILL OSCILLATIONS OCCUR;

5) IMPROVE THE NRC STAFF'S TECHNICAL REVIEW AND AUDIT CAPABILITY FOR INDUSTRY SUBMITTALS RELATIVE TO BWR STABILITY.

CODE USE MATRIX

Use	RAMONA-3B	TRAC-BF1	HIPA(EPA)	LAPUR	M-L Model
Amplitude of oscillation (LaSalle w/o scram)	X	X	X		
Maximum amplitude (core-wide) reactivity & power oscillation	X		X		
Maximum amplitude (asymmetric) reactivity & power oscillation	X				
ATWS scenarios			X		
ATWS sensitivity	X	X	X	X	
Pt vs 1-D kinetics comparison		X			
Stability boundary code comp'n			X	X	
Stability boundary sensitivity			X	X	
Shape/frequency asymmetric osc	X				
Asymmetric vs core-wide osc	X				X

STATUS OF WORK
FY 1989 ACCOMPLISHMENTS

- TPG FORMED; FOUR MEETINGS HELD
- LASALLE EVENT SIMULATED WITH HIPA AND RAMONA-3B
- NODING SENSITIVITY ISSUE ADDRESSED
- FLOW REVERSAL MODELLING DEFICIENCY FIXED IN RAMONA-3B
- ASSESSMENT AGAINST STEADY-STATE FRIGG DATA COMPLETED
- COMPLETED REVIEW OF NRC CODES AND ASSESSED USEFULNESS FOR ANALYZING INSTABILITY
- TRAC-BF1 INPUT DECK COMPLETED FOR LASALLE (POINT KINETICS)
- SENSITIVITY TO KEY PARAMETERS DETERMINED (LASALLE CONDITIONS)
- CORE-WIDE AND ASYMMETRIC OSCILLATIONS DEMONSTRATED

SPECIFIC RESULTS OF ANALYSES

- NRC (AEOD) WAS CONCERNED WHETHER A SEVERE REACTIVITY TRANSIENT CAUSING FUEL DAMAGE COULD OCCUR ON RESTART OF A RECIRCULATION PUMP

- HIPA CALCULATIONS SHOWED THAT RESTART OF THE PUMP COULD LEAD TO A POWER SPIKE THAT WOULD SCRAM THE REACTOR BUT NOT PRODUCE BOILING TRANSITION OR FUEL MELTING. THE POWER SPIKE IS CAUSED BY THE INCREASED FLOW OF SUBCOOLED LIQUID FROM THE LOWER PLENUM AND DOWNCOMER INTO THE CORE

SPECIFIC RESULTS OF ANALYSES (Cont) CALCULATIONS OF LASALLE SHOWED THAT:

- THE LASALLE CONDITIONS PRODUCE OSCILLATIONS LEADING TO AUTOMATIC REACTOR SCRAM

- THERMAL-HYDRAULIC INSTABILITY AT LASALLE WAS CAUSED BY THE COMBINATION OF 1) AXIAL AND RADIAL POWER PEAKING, 2) FLOW REDUCTION FROM TRIP OF BOTH RECIRCULATION PUMPS, AND 3) FEEDWATER TEMPERATURE REDUCTION FROM REDUCED FEEDWATER HEATING

- THE AMPLITUDE OF POWER OSCILLATIONS REMAINS BOUNDED EVEN IF FAILURE TO SCRAM IS ASSUMED

SPECIFIC RESULTS OF ANALYSES (Cont)

- A SENSITIVITY STUDY WITH THE RAMONA-3B CODE (USING A BWR/4 MODEL) PRODUCED BOTH CORE-WIDE AND ASYMMETRIC NEUTRON FLUX OSCILLATIONS. HYDRAULIC OSCILLATIONS THAT BEGIN IN A FEW HIGH-POWERED FUEL BUNDLES CAN CAUSE ASYMMETRIC POWER OSCILLATIONS DUE TO OUT-OF-PHASE PARALLEL-CHANNEL FLOW INSTABILITIES

- THE OSCILLATION SHAPE IS IMPORTANT SINCE IT EFFECTS THE MEASURED (VIEWED OR RECORDED) LPRM OUTPUT. OUTPUT FROM LPRMs AT PARTICULAR AXIAL AND RADIAL POSITIONS IN THE CORE IS USED TO DEVELOP APRM SIGNALS THAT SCRAM THE REACTOR ON HIGH FLUX

SPECIFIC RESULTS OF ANALYSES (Cont)

- TWO TYPES OF ASYMMETRIC OSCILLATIONS WERE CALCULATED, AZIMUTHAL (SIDE TO SIDE) AND RADIAL (CENTER TO PERIPHERY)

- THE EXCITATION THRESHOLD OF EACH TYPE (MODE) OF OSCILLATION IS A COMPLEX FUNCTION OF THE SUBCOOLING, THE LOCAL CHANNEL (FUEL BUNDLE) INLET FLOW AND BUNDLE POWER, AND THE SUBCRITICALITY OF THE MODE BEING EXCITED

FY 1990 EXPECTED RESULTS

- COMPLETE ANALYSES USING RAMONA-3B (IN THE FULL-CORE MODE) TO DETERMINE ASYMMETRIC OSCILLATION AMPLITUDES FOR NON-ATWS LIMIT CYCLE CASES
- ISSUE REPORT ON CONDITIONS IN BWRS THAT CAN LEAD TO LARGE AMPLITUDE LIMIT CYCLE OSCILLATIONS
- COMPLETE ANALYSES OF LASALLE EVENT WITH NRC CODES, INCLUDING PARAMETER SENSITIVITY STUDIES
- COMPLETE ANALYSES OF RESPONSE OF A BWR TO A SELECTED ATWS SCENARIO, INCLUDING PARAMETER VARIATIONS TO DETERMINE UNDER WHAT CONDITIONS AND OPERATING PROCEDURES OSCILLATIONS WILL OCCUR. THE TPG WILL HELP FORMULATE THE PARAMETER STUDY PLAN
- ISSUE FINAL REPORT DOCUMENTING THE INTEGRATED RESEARCH RESULTS ON BWR STABILITY

SCHEDULE TO COMPLETE & FUTURE PLANS

- ALL BWR STABILITY WORK TO BE COMPLETED IN FY 1990

- NO PLANS FOR FUTURE WORK

CODE USE MATRIX

Use	RAMONA-3B	TRAC-BF1	HIPA(EPA)	LAPUR	M-I. Model
Amplitude of oscillation (LaSalle w/o scram)	X	X	X		
Maximum amplitude (core-wide) reactivity & power oscillation	X		X		
Maximum amplitude (asymmetric) reactivity & power oscillation	X				
ATWS scenarios			X		
ATWS sensitivity	X	X	X	X	
Pt vs 1-D kinetics comparison		X			
Stability boundary code comp'n			X	X	
Stability boundary sensitivity			X	X	
Shape/frequency asymmetric osc	X				
Asymmetric vs core-wide osc	X				X

the current NRC Manual Chapter 1102 (See Attachment).

c. Publications Note - See Attachment

E. FY BUDGET:

	FY89	FY90
PRIOR:	$400K	
OPERATING:		$400K

F. ESTIMATED LEVEL OF EFFORT AND PERIOD OF PERFORMANCE

The overall level of effort is estimated to be 20 staff months over a 12 month period. The starting date for this program is estimated to be October 1, 1989 and the completion date September 28, 1990. (Presentation of results at a professional meeting may occur after this date).

G. MEETINGS AND TRAVEL

To meet the objectives of this FIN, work activities (including participation in TPG meetings) will arise that require the contractor (and possibly subcontractors) to travel. The NRC Form 189 proposal should state the number of domestic trips that will be required to perform the proposed work and identify for each trip:

(1) where, who, and how many people;
(2) the length of the stay; and
(3) the purpose of the travel.

If no travel is expected or required, state none. Foreign travel must be addressed separately and approval must be obtained by processing NRC Form 445, in addition to being provided as part of the proposal. The travel identified by the laboratory in their NRC Form 189 proposal, and agreed to by the NRC project manager, shall be considered as approved by NRC execution of an NRC Form 173 accepting the proposal. This approval, however, does not obviate the requirement to submit an NRC Form 445 for foreign travel. Any additional travel to be charged to this project, i.e., travel other than that previously proposed and approved by the NRC, shall be submitted to the NRC project manager for review and approval prior to taking any action which could result in a travel commitment.

H. NRC FURNISHED MATERIAL

None

assumed) plant parameters and code input values. Compare calculated frequency of oscillations and behavior of power oscillation amplitude with plant data. Then extend the calculation (by assuming a failure to scram) to determine the behavior of the resulting power oscillation limit cycle.
SUBTASK C:
Submit a draft report on the results by February 27, 1990. Include conclusions regarding effects of nodalization and numerical damping. This report should be prepared to achieve objective number 3 (See Item B on page 2).

TASK 3: TRAC-BF1 Code Application
Estimated Completion Date: 6/29/1990
Estimated Level of Effort: 11 staff-months

SUBTASK A:
In cooperation with the other program participants, assist the NRC in developing a sensitivity study plan. The plan will be developed using process identification and ranking and other elements of the CSAU methodology.
SUBTASK B:
Perform code application sensitivity calculations commensurate with the sensitivity study plan to be provided by the NRC. The ATWS calculations will generally require one-dimensional neutron kinetics in order to achieve objective number 4 (See Item B on page 2).
SUBTASK C:
Submit a draft report on the ATWS calculations by March 30, 1990. Submit a draft report on the other sensitivity calculations by June 22, 1990. This report should be prepared to achieve objective number 1 (See Item B on page 2).

TASK 4: Presentation of Results
Estimated Completion Date: 12/12/1990
Estimated Level of Effort: 1 staff-month

The contractor shall provide the following services: Prepare paper(s) for presentation and/or publication as directed by the NRC Technical Monitor.

D. REPORTING REQUIREMENTS

a. Technical Reports
Prepare draft copy of report requested in Task 1 by July 31, 1990 and camera-ready copy by September 21.

b. Monthly Business Letter Reports
Prepare a monthly business letter status report per

B. OBJECTIVES

The objectives of the BWR instability research program, of which this FIN is a part, are to: 1) provide a description of phenomena related to instability and then determine the influence of important plant design and operating parameters on oscillation mode, amplitude, and frequency, 2) review the relevant NRC codes and then assess their usefulness for analyzing instability, 3) determine code validation requirements and then perform the code assessment cases, 4) evaluate BWR response to an ATWS event, with emphasis on the role of oscillations during the recovery phase and the appropriateness of ATWS procedures, 5) improve the NRC staff's technical review and audit capability for industry submittals relative to BWR stability.

The specific objective of the work funded by this FIN is to assess and apply the TRAC-BF1 computer code for understanding of power oscillations in BWRs.

C. WORK REQUIREMENTS

TASK 1: Program Coordination
Estimated Completion Date: 9/28/1990
Estimated Level of Effort: 3 staff-months

The contractor shall provide the following services: Participate in the Technical Program Group and resolve action items assigned to INEL. Provide input to and coordination of a NUREG/CR report documenting the integrated research results. This report should be prepared to achieve objective number 5 (See Item B above). Participate in a bilateral meeting with Sweden.

TASK 2: TRAC-BF1 Code Assessment
Estimated Completion Date: 3/30/1990
Estimated Level of Effort: 4 staff-months

SUBTASK A:
Complete the assessment calculations of the FRIGG experiments. This includes steady state, transient, and transfer function calculations. Record agreements and differences with the data. Perform comparisons with similar calculations performed at BNL with RAMONA-3B for the purpose of contrasting the TRAC two-fluid model with the RAMONA drift-flux model.
SUBTASK B:
Complete a calculation of the LaSalle instability event by November 28, 1989. Record differences between known (or

FY 90
STATEMENT OF WORK

TITLE: BWR INSTABILITY ANALYSIS

FIN: L10810
CONTRACTOR: INEL
SITE: IDAHO FALLS
STATE: IDAHO

NRC TECHNICAL MONITOR: H. Scott
(FTS 492-3563)

PRINCIPAL INVESTIGATOR: G. Wilson
(FTS 583-9511)

BUDGET ACTIVITY: 0601922020 FY90 BUDGET: $400K

FY 1990 WORK PERIOD: 10/01/89 to 9/30/90

A. BACKGROUND

This program was developed in response to an NRR user request and the results of the program will be used by NRR to support: 1) review of BWROG solutions for prevention and/or mitigation of power oscillations and 2) review of emergency procedure guidelines for ATWS.

General Design Criterion 12 (in Appendix A of 10CFR50) states that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillatons which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed. Following the power oscillation event at LaSalle in March 1988, two questions were posed: (1) What is the potential extent of fuel damage resulting from asymmetric regional neutron flux (power) oscillations if they are not detected and suppressed, and (2) what are the potential implications of instability with respect to ATWS events (where the oscillations might complicate the recovery). A research program was developed to address these questions and to provide an independent review and audit capability for analysis of industry submittals. Four computer codes (each with unique capabilities) are being used, namely, RAMONA-3B, TRAC-BF1, LAPUR, and HIPA. The research is coordinated amongst BNL, INEL, ORNL and University of California at Santa Barbara.

July 31, 1990. These reports will be combined at INEL with reports from the other contractors into a NUREG/CR report. Prepare camera-ready copy of document requested in Task 8 by September 7, 1990.

b. Monthly Business Letter Reports - See Attachment

c. Publications Note - See Attachment

E. FY BUDGET:

	FY89	FY90
PRIOR:	$490K	
OPERATING:		$400K

F. ESTIMATED LEVEL OF EFFORT AND PERIOD OF PERFORMANCE

The overall level of effort is estimated to be 22 staff months over a 12 month period. The starting date for this program is estimated to be October 1, 1989 and the completion date September 28, 1990. (Presentation of results at a professional meeting may occur after this date).

G. MEETINGS AND TRAVEL

To meet the objectives of this FIN, work activities (including participation in TPG meetings) will arise that require travel.
State the number of trips that will be required to perform the proposed work and identify for each trip:
(1) where, who, and how many people;
(2) the length of the stay; and
(3) the purpose of the travel.
If no travel is expected or required, state none. Foreign travel must be addressed separately and approval must be obtained by processing NRC Form 445, in addition to being provided as part of the proposal. The travel identified by the laboratory in their NRC Form 189 proposal, and agreed to by the NRC project manager, shall be considered as approved by NRC execution of an NRC Form 173 accepting the proposal. This approval, however, does not obviate the requirement to submit an NRC Form 445 for foreign travel. Any additional travel to be charged to this project, i.e., travel other than that previously proposed and approved by the NRC, shall be submitted to the NRC project manager for review and approval prior to taking any action which could result in a travel commitment.

TASK 6: RAMONA-3B Calculations for Asymmetric Conditions
Estimated Completion Date: 9/30/1990
Estimated Level of Effort: 10 staff-months

The contractor shall provide the following services: Complete the plan (and cost/time estimate) to utilize LaSalle core neutronics data. Then modify the RAMONA-3B input used for previous calculations and perform full core (191 node) calculations to determine asymmetric oscillation amplitudes for limit cycle cases. Submit a draft report on the results of the calculations by July 2, 1990. This report should be prepared to answer question number 1 (See Item A).

TASK 7: RAMONA-3B Sensitivity Calculations
Estimated Completion Date: 9/30/1990
Estimated Level of Effort: 5 staff-months

The contractor shall provide the following services: In cooperation with the other program participants, assist the NRC in developing a sensitivity study plan. The plan will be developed using process identification and ranking and other elements of the CSAU methodology. Perform calculations with RAMONA-3B as specified in the sensitivity study plan (to be provided by the NRC in December 1989). Perform one additional calculation using the input deck developed in Task 6 to compare with the TRAC-BF1 LaSalle assessment calculation. Submit a draft report on the results of the calculations by June 22, 1990. This report should be prepared to achieve objective number 1 (See Item B on page 2).

TASK 8: Code Documentation and Presentation of Results
Estimated Completion Date: 9/30/1990
Estimated Level of Effort: 3 staff-months

The contractor shall provide the following services: Revise and complete the documentation of the BLEND code. Submit a draft report for review by July 30, 1990. Submit a draft report (by February 27, 1990) on the results of the RAMONA-3B assessment calculations completed in FY 1989. This report should be prepared to achieve objective number 3 (See Item B on page 2). Prepare papers for presentation and/or publication as directed by the NRC Technical Monitor.

D. REPORTING REQUIREMENTS

a. Technical Reports - Final reports from Tasks 5, 6, and 7 should be submitted to the NRC Technical Monitor by

B. OBJECTIVES

The objectives of the BWR instability research program, of which this FIN is a part, are to: 1) provide a description of phenomena related to instability and then determine the influence of important plant design and operating parameters on oscillation mode, amplitude, and frequency, 2) review the relevant NRC codes and then assess their usefulness for analyzing instability, 3) determine code validation requirements and then perform the code assessment cases, 4) evaluate BWR response to an ATWS event, with emphasis on the role of oscillations during the recovery phase and the appropriateness of ATWS procedures, 5) improve the NRC staff's technical review and audit capability for industry submittals relative to BWR stability.

The specific objective of the work funded by this FIN is to assess and apply the BWR Plant Analyzer (HIPA) and RAMONA-3B computer code for understanding of power oscillations in BWRs.

C. WORK REQUIREMENTS

TASKS 1, 2, and 3 were completed in FY 1989.

TASK 4: Program Coordination
Estimated Completion Date: 9/28/1990
Estimated Level of Effort: 2 staff-months

The contractor shall provide the following services: Participate in the Technical Program Group and resolve action items assigned to BNL. Participate in a bilateral meeting with Sweden.

TASK 5: BNL Analyzer Calculations
Estimated Completion Date: 9/30/1990
Estimated Level of Effort: 2 staff-months

The contractor shall provide the following services: Perform calculations with the HIPA code (BWR plant analyzer) as specified in the sensitivity study plan (to be provided by the NRC in December 1989). Submit a draft report on the results of the calculations by March 30, 1990. This report should be prepared to achieve objective number 1 (See Item B above).

FY 90
STATEMENT OF WORK

TITLE: APPLICATION OF RAMONA-3B AND BNL ANALYZER TO BWR STABILITY

FIN: A39830
CONTRACTOR: BNL
SITE: UPTON
STATE: NEW YORK

NRC TECHNICAL MONITOR: H. Scott (FTS 492-3563)

PRINCIPAL INVESTIGATOR: W. WULFF (FTS 666-2608)

BUDGET ACTIVITY: 0601922020 FY90 BUDGET: $400K

FY 1990 WORK PERIOD: 10/01/89 to 9/30/90

A. BACKGROUND

General Design Criterion 12 (in Appendix A of 10CFR50) states that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillatons which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed. In a memorandum from W. Hodges to L. Shotkin dated June 2, 1988, NRR requested assistance in the form of RAMONA-3B and HIPA calculations. Two questions were posed: (1) What is the potential extent of fuel damage resulting from asymmetric regional neutron flux (power) oscillations if they are not detected and suppressed, and (2) what are the potential implications of instability with respect to ATWS events (where the oscillations might complicate the recovery). A research program was developed to address the questions and to provide an independent review and audit capability for analysis of industry submittals. Four computer codes (each with unique capabilities) are being used, namely, RAMONA-3B, TRAC-BF1, LAPUR, and HIPA. The research is coordinated amongst BNL, INEL, ORNL and University of California at Santa Barbara. The results of the program will be used by NRR to support: 1) review of BWROG solutions for prevention and/or mitigation of power oscillations and 2) review of emergency procedure guidelines for ATWS.

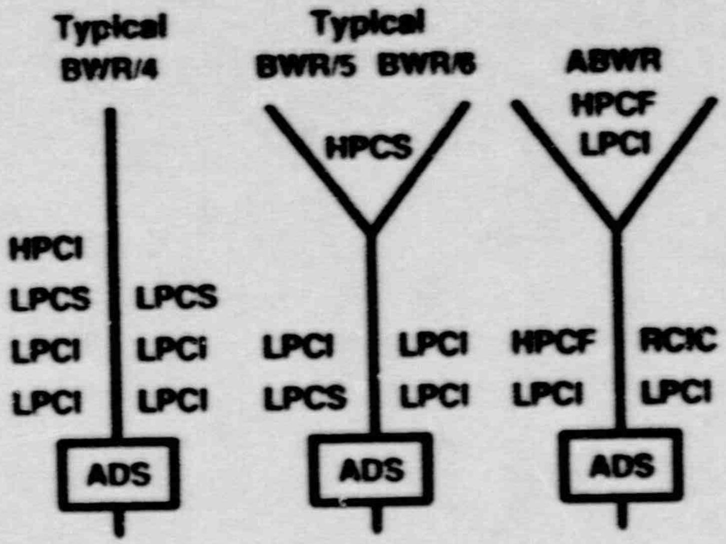
ECCS CONFIGURATION

SIPE6:CDS:FL891101:JDW

CORE COOLING SYSTEM OPTIMIZATION

- Several Proposed ECCS Configurations Considered
  - 2, 3, & 4 High Pressure Systems
  - 2, 3, & 4 Low Pressure Systems
  - Separate and Combined Heat Removal System and Low Pressure ECCS
  - Manual and Automatic Core Cooling Injection for Low Pressure Systems, if Used Also for Heat Removal
  - Core Spray Cooling
- Selection of Optimized System Based on Cost and Associated Benefits
- Benefits
  - Transient Performance Results
  - LOCA Performance Results
  - Probabilistic Risk Assessment Results (Core Damage Frequency)
  - Simplicity

ECCS KEY PERFORMANCE FEATURES



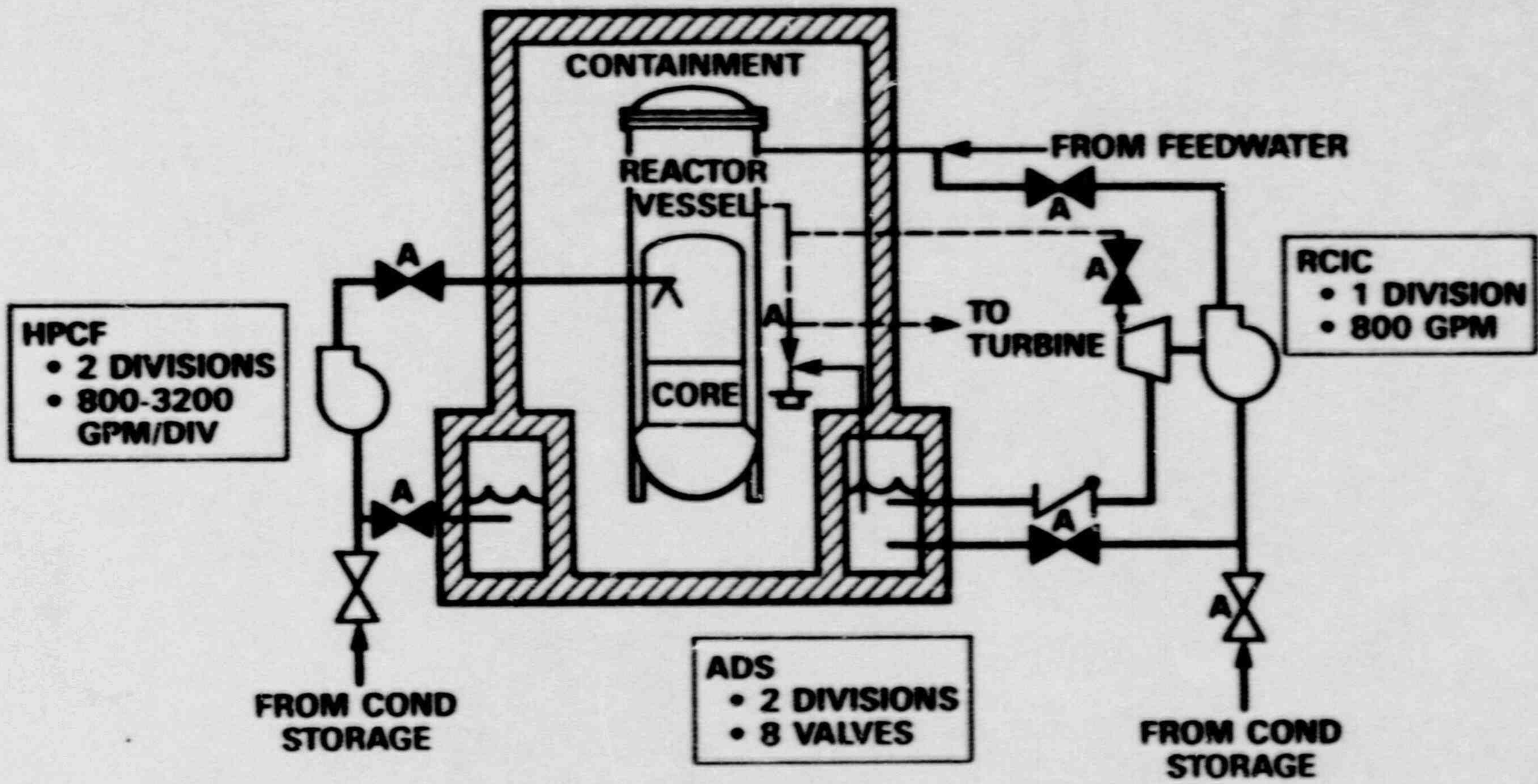
	Typical BWR/4	Typical BWR/5 BWR/6	ABWR
High Pressure* Capacity	5000	1900	2800
Low Pressure** Capacity	42000	29000	19000
No. of Large Pipes Below Core	12	12	0
Peak Clad Temp (APP K), °F	1600	1100	No Uncovery
N-2 Capability	None	All but HPCS Break	All but LPCI (A) Break

* @1100 psi
** @ 100 psi

A2084.25

ABWR EMERGENCY CORE COOLING SYSTEMS

High Pressure



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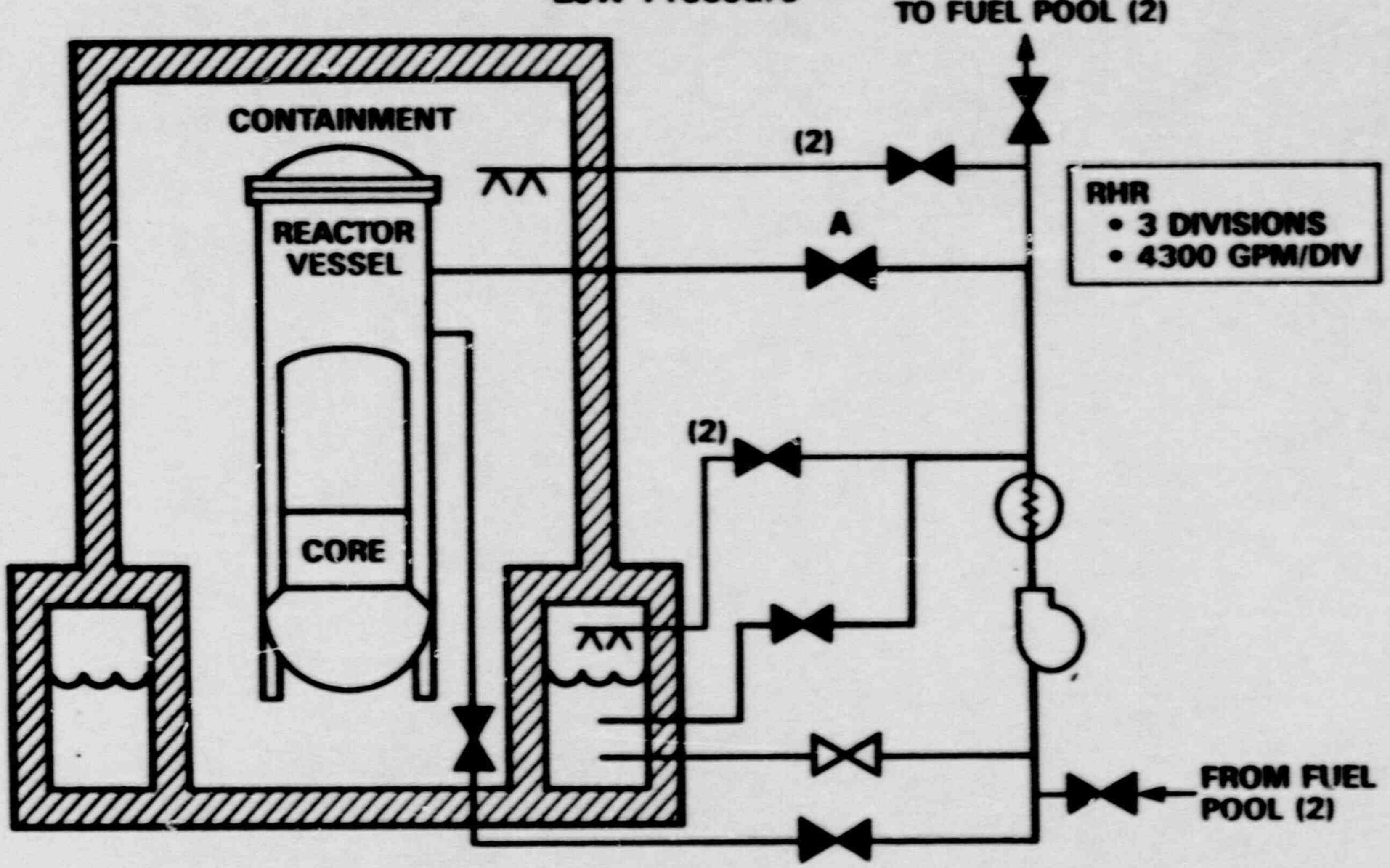
ABWR EMERGENCY CORE COOLING SYSTEMS

Low Pressure



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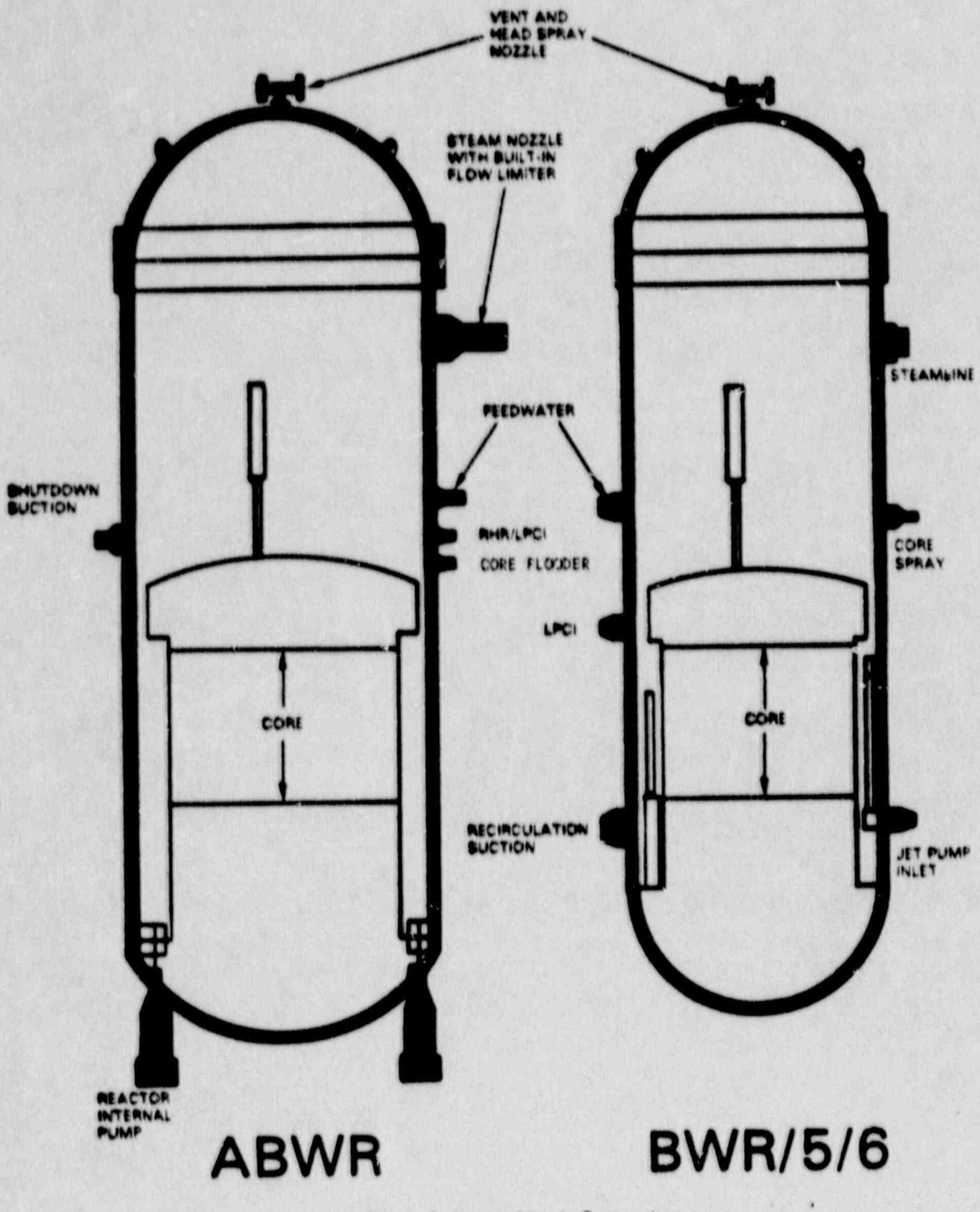
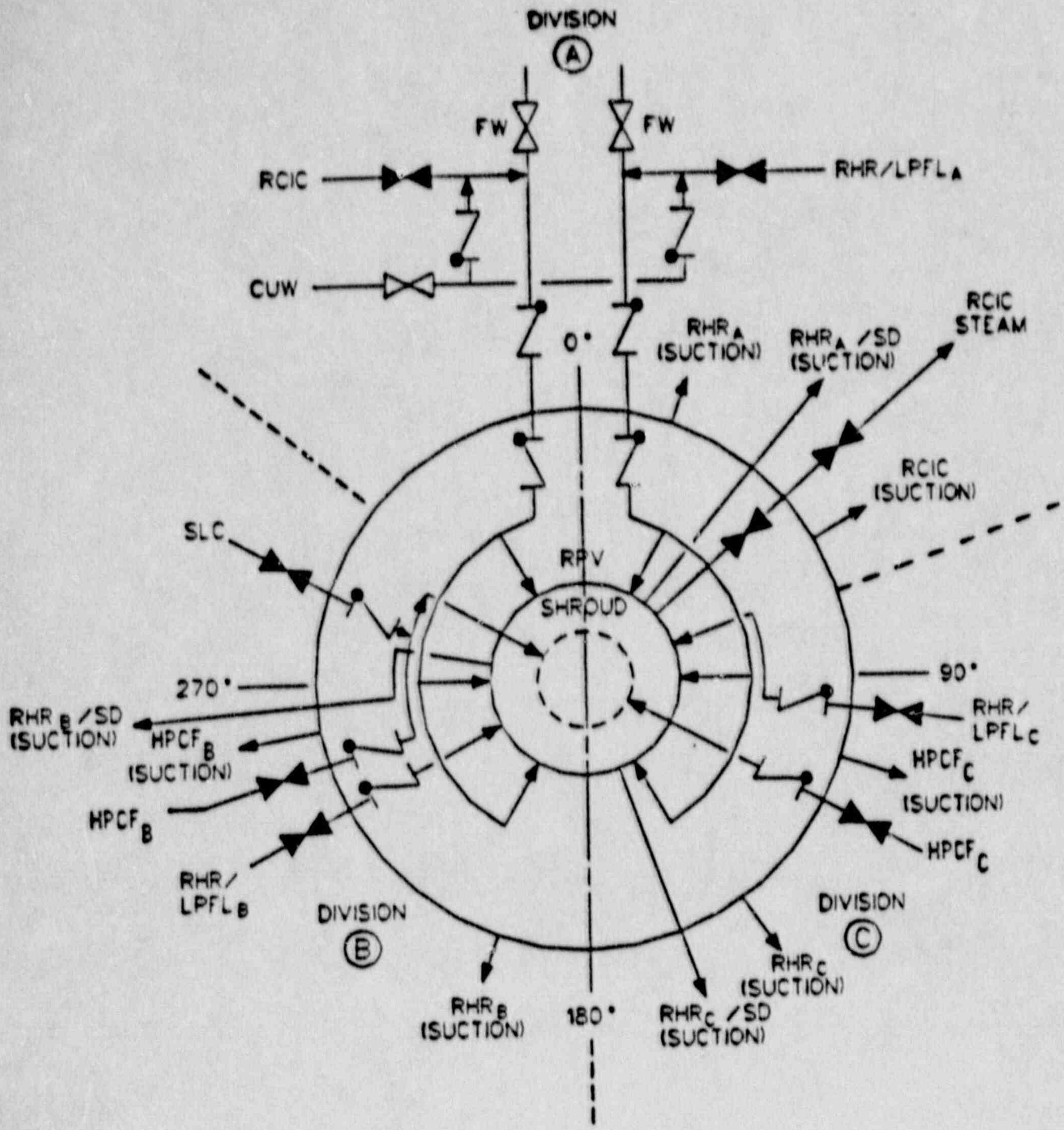


Figure 3. Vessel Nozzle Comparison



ECCS VESSEL NOZZLES (PLAN VIEW)

ABWR SAFETY SYSTEM IMPROVEMENTS

- Three Completely Separate Mechanical & Electrical Divisions for Most Important Functions
  - Core Cooling
  - Suppression Pool Cooling
  - Shutdown Cooling
- Automation of Post-LOCA Pool Cooling
  - Heat Exchangers Always in the Loop
- Elimination/Transfer of Complex Modes
  - Steam Condensing
  - RPV Head Spray
  - Containment Flood
  - Reduced Valves, Pipes by One-Third
- Significant Capacity Reduction
  - Reduced Equipment Sizes
- Greatly Reduced Duty During Transients
  - N-2 Capability at High Pressure
- Improved Small Break Response
  - Reduced Needs for ADS
- No Fuel Uncovery for Any Pipe Break

STATUS OF NRR BWR STABILITY REVIEW

PRESENTATION
TO ACRS T & H
PHENOMENA SUBCOMMITTEE

BY
LARRY PHILLIPS

NOVEMBER 9, 1989

REGULATORY ISSUES

(1) WITH SCRAM SYSTEM OPERABLE

- ASSURE THAT AUTOMATIC PROTECTION FEATURES AND OPERATING PROCEDURES WILL PREVENT VIOLATION OF SAFETY LIMITS DUE TO POWER OSCILLATIONS

(2) ATWS

- CONFIRM THAT EXISTING REQUIREMENTS AND PROCEDURE GUIDELINES FOR RESPONSE TO ATWS REMAIN ADEQUATE FOR ALL POTENTIAL CIRCUMSTANCES OF POWER OSCILLATIONS ASSOCIATED WITH ATWS SCENARIOS

BWROG PROPOSED RESOLUTION

(1) WITH SCRAM SYSTEM OPERABLE

* DEFINE POWER/FLOW EXCLUSION REGION FOR EACH PRODUCT LINE

* PROVIDE AUTOMATIC CONTROL ROD INSERT RESPONSE TO PREVENT OPERATION IN EXCLUSION REGION

* DEFINE CONDITIONS FOR BYPASS OF AUTOMATIC EXCLUSION ACTIONS WITH CONTINUOUS SURVEILLANCE USING A STABILITY MONITOR

BWROG PROPOSED RESOLUTION (CONT'D)

(1) WITH SCRAM SYSTEM OPERABLE

II * USE EXISTING QUADRANT-BASED FLOW BIASED APRM FLUX SYSTEM FOR AUTOMATIC DETECTION AND SUPPRESSION OF BWR 2 REACTORS (OYSTER CREEK AND NMP 1)

III * AUTOMATIC ACTION FOR SPECIFIED LPRM SIGNATURE

BWROG PROPOSED RESOLUTION

(2) ATWS

* NEDO-31709 - INCREASE IN AVERAGE CORE POWER DURING LARGE LIMIT CYCLE OSCILLATIONS

  - CALCULATIONS OF CORE NEUTRON POWER OSCILLATIONS TO 200 % OF RATED RESULTED IN A 7 % AVERAGE POWER INCREASE DUE TO NONLINEARITIES IN THE OSCILLATIONS AND SYSTEM FEEDWATER EFFECTS

  - BWROG CONCLUDES THAT PREVIOUS ATWS EVALUATIONS ARE VALID AND EXISTING ATWS ACTIONS ARE APPROPRIATE

NRR REVIEW STATUS

(1) WITH SCRAM SYSTEM OPERABLE

I
- REVIEW BWROG METHODS FOR DEFINITION OF EXCLUSION REGION (NOV 1989 - JAN 1990)

- REVIEW PROPOSED LIMITATIONS ON OPERATION WITHIN THE EXCLUDED REGION (NOV 1989 - JAN 1990)

- REVIEW THE DESIGN AND IMPLEMENTATION OF PROPOSED STABILITY MONITOR SYSTEMS (NOV 1989 - JAN 1990)

II
- REVIEW JUSTIFICATION FOR THE ADEQUACY OF EXISTING BWR 2 RPS (NOV 1989 - JAN 1990)

III
- REVIEW DESIGN AND ASSOCIATED ANALYSES FOR BWROG PROPOSED LPRM INSTABILITY TRIP SYSTEM (NOV 1989 - JAN 1990)

NRR REVIEW STATUS (CONT'D)

(1) WITH SCRAM SYSTEM OPERABLE

* DEFINE MULTI-PLANT ACTION REQUIREMENTS FOR IMPLEMENTATION OF ACCEPTABLE BWROG LONG TERM SOLUTIONS (LTS) (FEB 1990)

* PREPARE COMMISSION PAPER PROVIDING STATUS AND STAFF RECOMMENDATIONS FOR LONG TERM SOLUTION WITH SCRAM OPERABLE (FEB 1990)

* REVIEW UTILITY MULTI-PLANT ACTIONS FOR LONG TERM SOLUTION SELECTION, IMPLEMENTATION AND TECH SPEC CHANGES (SCHEDULE TO BE DETERMINED)

NRR REVIEW STATUS

(2) ATWS

* NEDO-31709 RESULTS AND NRC STUDIES ARE NOT SUFFICIENT TO SUPPORT THE BWROG CONCLUSION THAT PREVIOUS ATWS EVALUATIONS REMAIN VALID

* WORK IS PROGRESSING ON IDENTIFICATION OF CODE LIMITATIONS AND IMPROVEMENT OF STABILITY ANALYTICAL CAPABILITY

* KEY QUESTIONS CONCERNING THE MAXIMUM AMPLITUDE AND POTENTIAL CONSEQUENCES OF LARGE LIMIT CYCLE OSCILLATIONS HAVE NOT BEEN ANSWERED

SUMMARY AND PRELIMINARY CONCLUSIONS

* ACCEPTABLE METHODS TO PROVIDE HIGH ASSURANCE OF PERFORMANCE TO GDC 12 FOR EVOLVING CORE DESIGNS HAVE BEEN IDENTIFIED

  - DETAILS OF DESIGN AND IMPLEMENTATION ARE EXPECTED TO BE DEFINED IN THE NEAR TERM (EARLY 1990)

SUMMARY AND PRELIMINARY CONCLUSIONS (CONT'D)

* MAXIMUM AMPLITUDE OF NEUTRON FLUX POWER OSCILLATIONS IS UNCERTAIN

- HiPA	2100 %	
- LAPUR + ANALYSIS	500 %	
- TRACG	200 %	(not bounding)

* AVG THERMAL POWER INCREASE

= 1.5 to 2.0 % X NEUTRON FLUX PEAK POWER

EXCLUDING SYSTEM EFFECTS

* EFFECT OF OSCILLATIONS ON ATWS NOT DETERMINED - MORE ANALYSES ARE PLANNED

NRR REVIEW

OF

ABWR ECCS PERFORMANCE

* NO CORE UNCOVERY
* USE APPROVED LOCA ANALYSIS METHODS
* COMPLIES WITH 10 CFR 50.46 AND APPENDIX K
* PCT 1149 F (< 2200 F)

José March-Leuba

Oak Ridge National Laboratory*

ACRS Meeting

November 9, 1989

San Francisco, CA

* Research sponsored by the U.S. Nuclear Regulatory Commission under Interagency Agreement No. 1886-8947-8A and performed at Oak Ridge National Laboratory, operated by Martin Marietta Energy Systems, Inc., for the U.S. Department of Energy under Contract No. DE-AC05-84OR21400. The views expressed in this paper are not necessarily those of the U.S. Nuclear Regulatory Commission

OUTLINE

- Scope of ORNL BWR stability work
- Overview of U.S. BWR stability codes
  - Applications
  - Validation efforts
- Brief description of the LAPUR code
- Nonlinear studies with a reduced-order model
- What do we know about BWR stability?
- What else do we need to know?

SCOPE OF ORNL STABILITY WORK

- Main job is as consultants to NRR
  - Know the issues
  - Review industry proposals
  - Raise possible safety issues
- Numerical tools
  - LAPUR. A frequency domain code
  - A reduced order BWR dynamic model.
    A time domain nonlinear code.

U.S. CALCULATIONAL CAPABILITIES IN THE BWR STABILITY AREA

Frequency Domain Codes

• FABLE	GE
• LAPUR	ORNL/NRR
• NUFREQ	RPI/NRC

Time Domain Codes

• COTRAN (COTRANSA2)	ANF
• HIPA (EPA)	BNL/NRC
• RAMONA	BNL/NRC SCANDPOWER
• RETRAN	EPRI MSU
• TRAC	INEL/NRC GE

CODE APPLICATIONS

Frequency Domain Codes

- Predict the onset of instability
- Compare relative stability of different designs or operating conditions

Time Domain Codes

- Predict the onset of instability and relative stability
- Study nonlinear effects
  - Limit cycle amplitude
  - Flow reversal
- Predict impact on fuel of large limit cycles
- Study system effects
  - controllers
  - operator actions

CODE APPLICATIONS (cont)

- In general frequency domain codes are more accurate numerically and require orders of magnitude less computational effort than time domain codes.

- Results from time domain codes are sometimes difficult to interpret due to system effects and nonlinearities.

  --> Use Frequency domain codes whenever possible

  - Scoping calculations
  - Relative stability of design changes
  - Power/Flow stability map

CODE VALIDATION EFFORTS FOR LINEAR STABILITY

FABLE	Peach Bottom Vermont Yankee Caorso Leibstadt LaSalle, ...
LAPUR	Peach Bottom Vermont Yankee Browns Ferry SLO Susquehanna-2 Grand Gulf Swedish BWR (out-of-phase)
NUFREQ-NPW	Peach Bottom, ...
COTRAN	Peach Bottom, ...
HIPA	Frigg (channel stability) LaSalle
RAMONA	Frigg (channel stability) ...

CODE VALIDATION EFFORTS FOR LINEAR STABILITY (cont)

RETRAN	Peach Bottom Grand Gulf LaSalle, ...
TRAC	Frigg (channel stability) LaSalle, ...

CODE VALIDATION EFFORTS FOR LARGE AMPLITUDE LIMIT CYCLES

- No good benchmark data
  - LaSalle up to the scram point. Modelled by TRAC, HIPA, and RETRAN.

CODE VALIDATION CONCLUSIONS

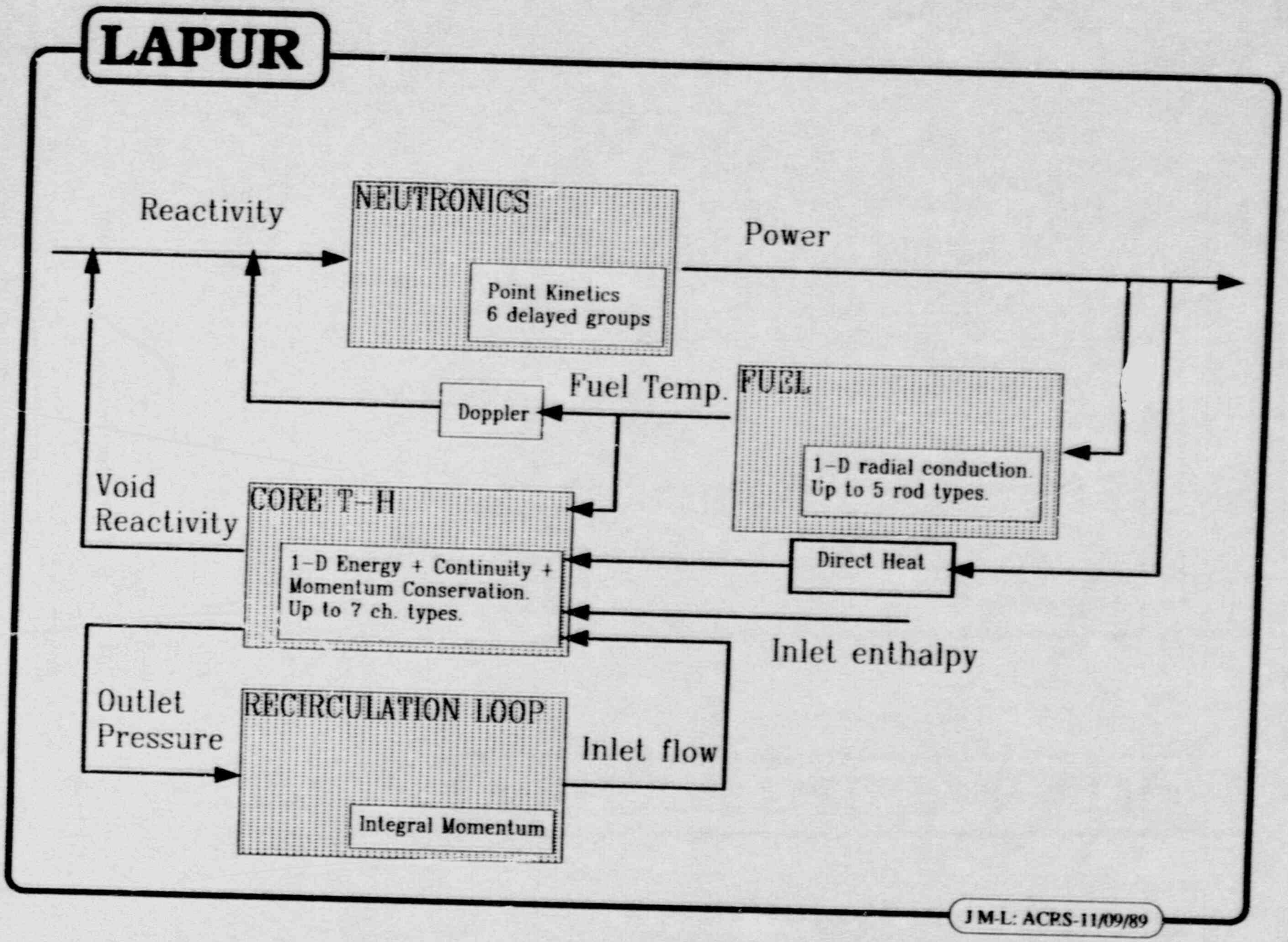
- Most codes are capable of reproducing linear stability test results fairly accurately (within ±20%)

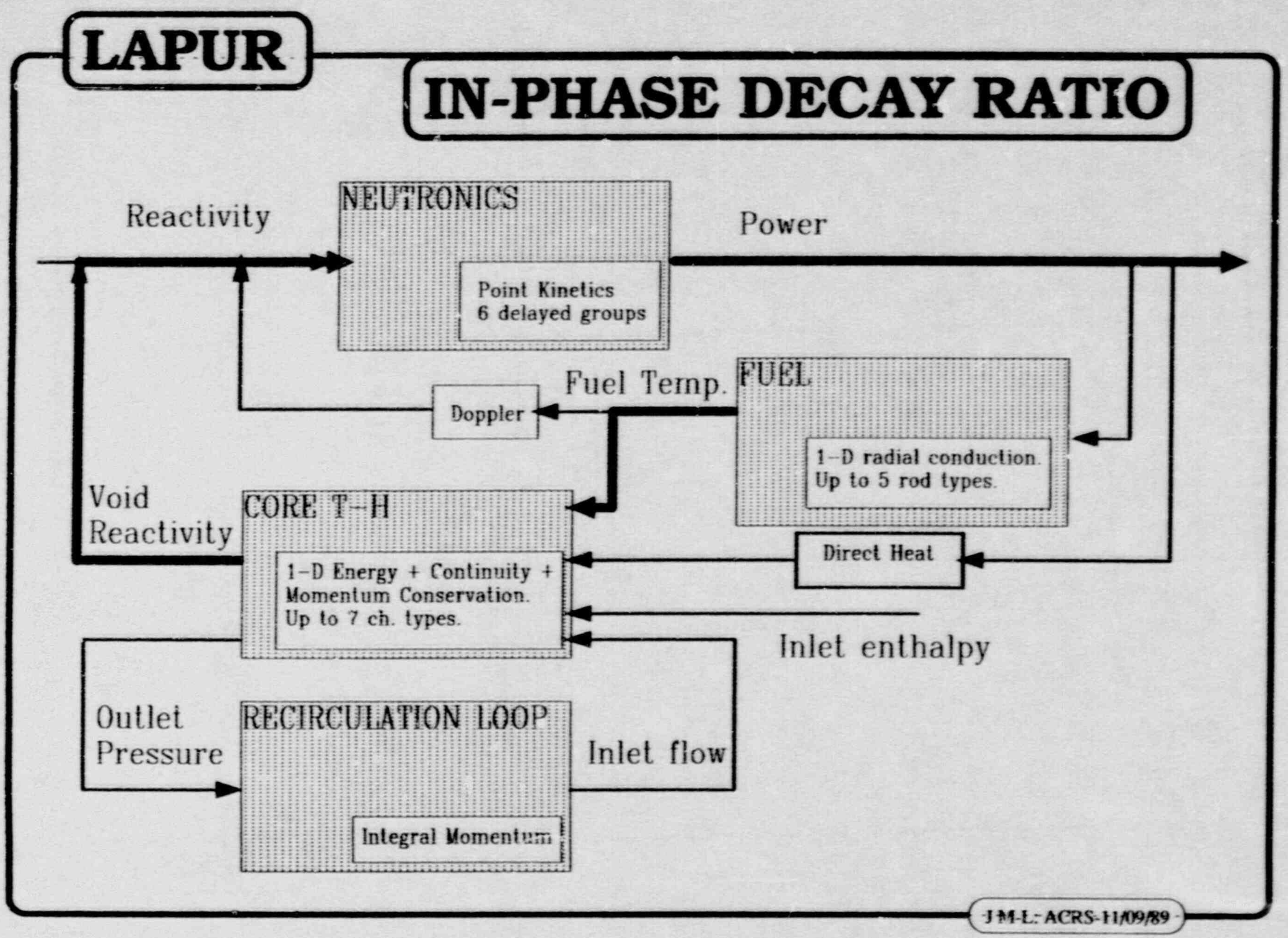
- Everybody has problems trying to define the most unstable conditions for a whole fuel cycle

  ==>Codes are difficult to apply for predictive mode calculations

- Different results when applied to defining maximum oscillation amplitude (200% to 2100%)

  ==>We need to validate/verify/benchmark time domain codes for large amplitude limit cycles


LAPUR
Reactivity
NEUTRONICS
Point Kinetics
6 delayed groups
Power
Fuel Temp.
FUEL
Doppler
1-D radial conduction.
Up to 5 rod types.
Void
Reactivity
CORE T-H
1-D Energy + Continuity +
Momentum Conservation.
Up to 7 ch. types.
Direct Heat
Inlet enthalpy
Outlet
Pressure
RECIRCULATION LOOP
Inlet flow
Integral Momentum
J M-L: ACRS-11/09/89


LAPUR
IN-PHASE DECAY RATIO
NEUTRONICS
Point Kinetics
6 delayed groups
Reactivity
Power
FUEL
1-D radial conduction.
Up to 5 rod types.
Fuel Temp.
Doppler
Void
Reactivity
CORE T-H
1-D Energy + Continuity +
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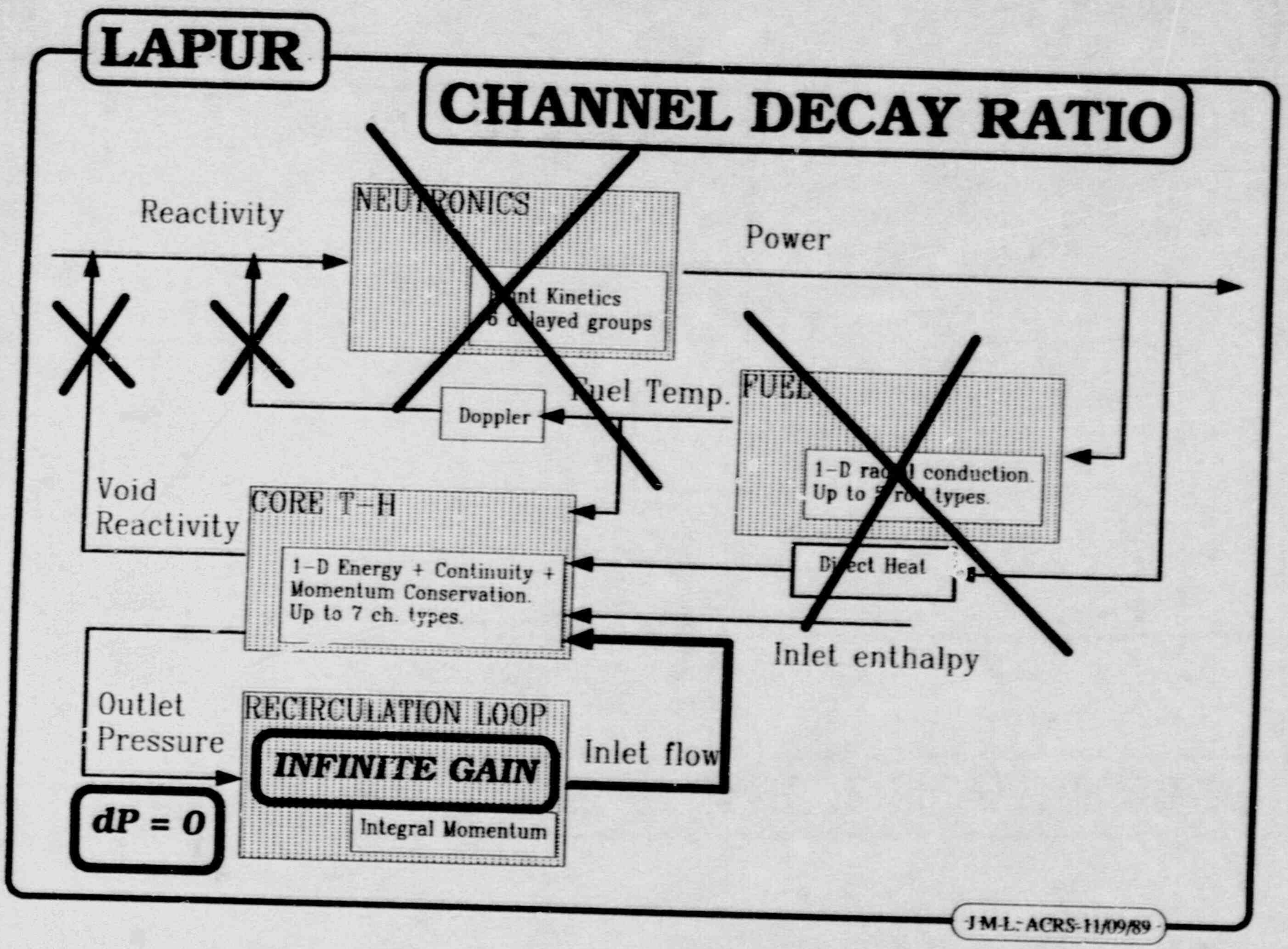
Direct Heat
Inlet enthalpy
Outlet
Pressure
RECIRCULATION LOOP
Integral Momentum
Inlet flow
JML: ACRS-11/09/89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Reactivity
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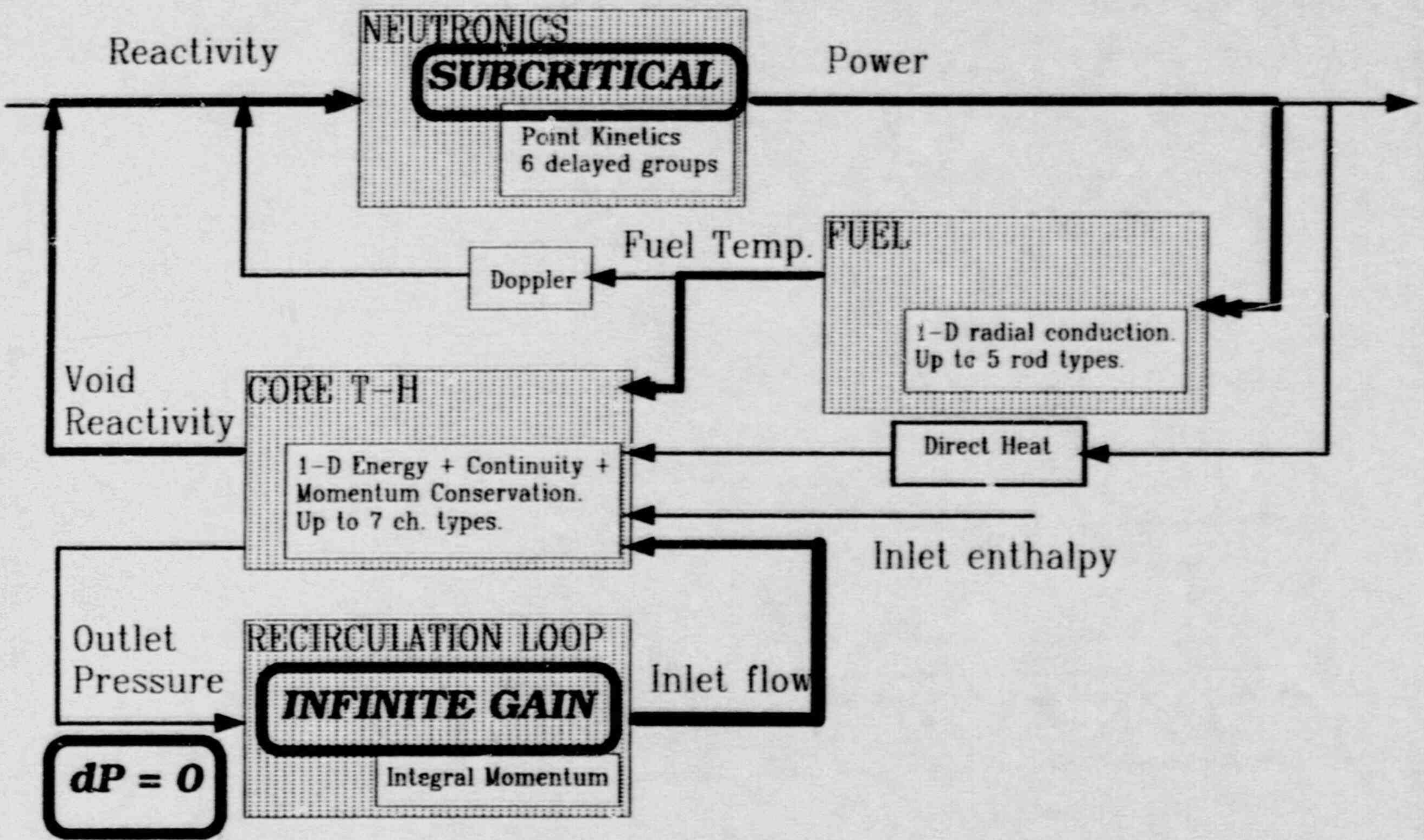
Void
Reactivity
CORE T-H
1-D Energy + Continuity +
Momentum Conservation.
Up to 7 ch. types.
ect Heat
Inlet enthalpy
Outlet
Pressure
RECIRCULATION LOOP
INFINITE GAIN
Integral Momentum
Inlet flow
dP = 0
J-M-L- ACRS-11/09/89

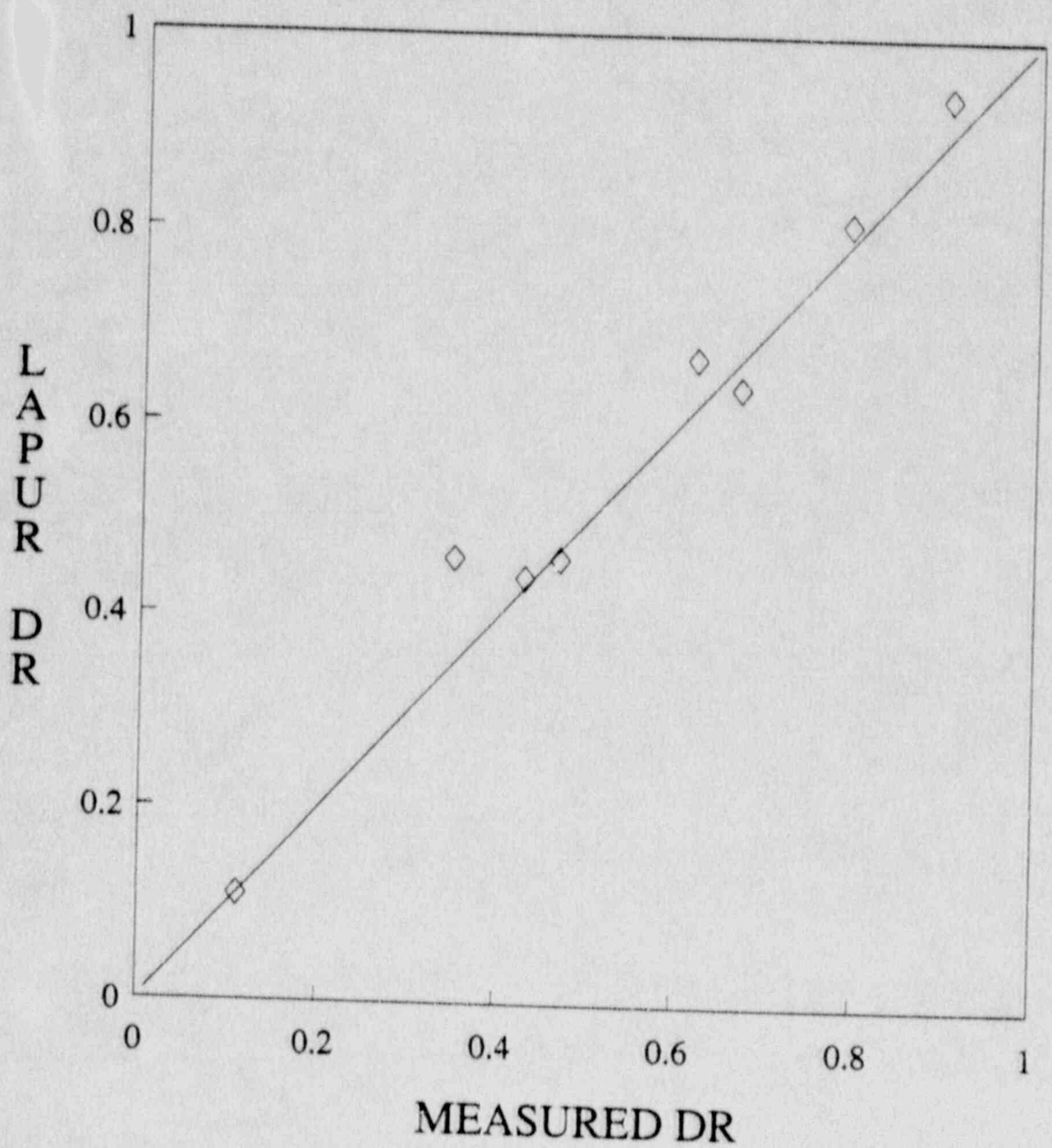
LAPUR

OUT-OF-PHASE DECAY RATIO



J-M-L: ACRS-11/09/89

LAPUR/FOREIGN_REACTOR BENCHMARK
IN-PHASE DECAY RATIO



ORNL REDUCED ORDER NONLINEAR MODEL

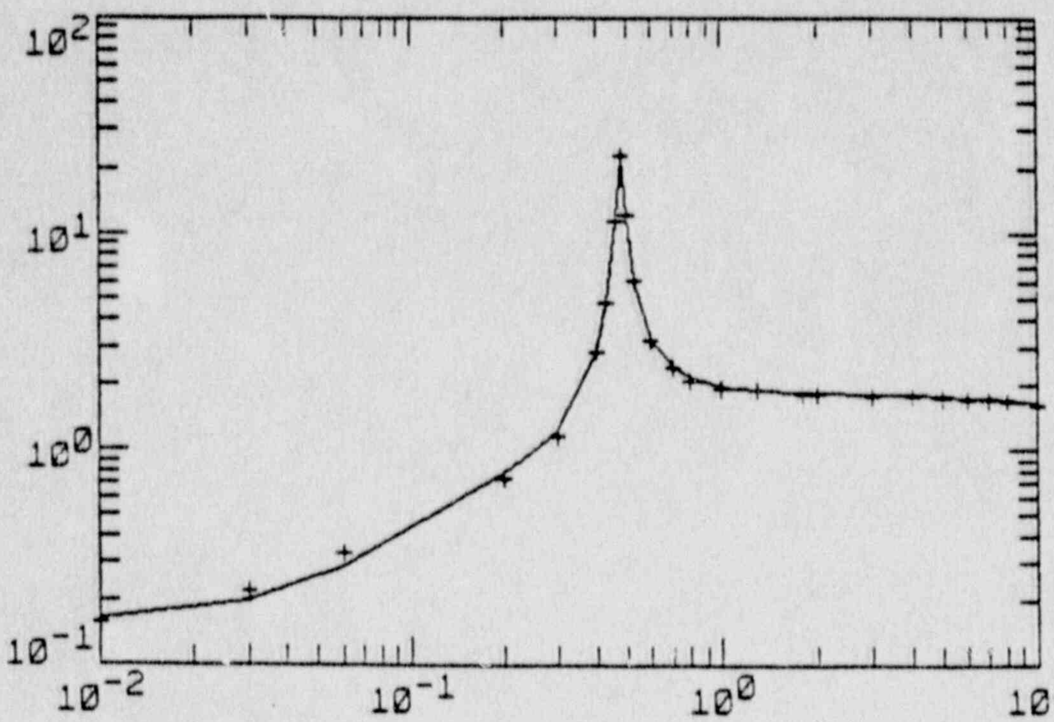
$$\frac{dn(t)}{dt} = \frac{\rho(t)-\beta}{\Lambda} n(t) + \lambda c + \frac{\rho}{\Lambda}$$

$$\frac{dc(t)}{dt} = \frac{\beta}{\Lambda} n(t) + \lambda c$$

$$\frac{dT(t)}{dt} = a_1 \; n(t) - a_2 \; T(t)$$

$$\frac{d^2\rho_\alpha(t)}{dt^2} + a_3 \frac{d\rho_\alpha(t)}{dt} + a_4 \; \rho_\alpha = k \; T(t)$$

$$\rho(t) = \rho_\alpha(t) + D \; T(t) \quad ,$$



RESULTS FROM REDUCED ORDER BWR DYNAMIC MODEL

- Limit cycle bounds oscillation amplitude when BWR becomes unstable

- Limit cycle is caused mainly by neutronics (term rho-times-n in point kinetics)

- To establish a limit cycle there must be a negative reactivity bias (increased voids):

    - Average power increase (2% of peak)
    - Flow reduction

- Limit cycle amplitudes not bounded. "Credible" amplitudes of least 500%_nominal (DR = 1.6)

- Limit cycles might become unstable and "bifurcate", increasing its amplitude

WHAT DO WE KNOW ABOUT BWR STABILITY?

- Types of instabilities
  - Plant (control system)
  - Channel Thermohydraulics
  - In-Phase (core-wide)
  - Out-of-Phase (regional)

- Physical Mechanisms

- Instabilities are more likely at low flows

- Sensitivity to parameters
  - High pcwer density
  - Power Shapes

- Nonlinearities limit oscillation amplitude
  - Limit cycle
  - Amplitude may be very large
    (200% to 2100%)

- Limit cycle --> Negative reactivity bias
  - Power increase (1.5% to 2% of peak)
  - Flow decrease

WHAT DO WE KNOW ABOUT BWR STABILITY? (cont)

CONSEQUENCES

- In-Phase
  - Neutronically-driven (void feedback)
  - Fuel acts as a filter
    --> Negligible effect on fuel for reasonable amplitudes
  - Negative reactivity bias
    --> Average power increase
    --> Flow decrease
  - Easy to detect

- Out-of-Phase
  - Flow-driven (dynamic pressure drop)
  - Reverse flow at inlet
    --> Safety limits may be violated with relatively small oscillations (200% -300%)
  - Relatively difficult to detect

- Channel
  - Flow-driven (dynamic pressure drop)
  - Reverse flow at inlet
    --> Safety limits may be violated
  - Very difficult to detect

WHAT DO WE NEED TO KNOW?

Scram System Available

- Evaluate the effect on fuel integrity of limit cycle oscillations

  - In-phase oscillations
  - Out-of-phase oscillations
  - Channel TH oscillations
    --> Channel dryout?

- Evaluate the detectability of the three instability types.

  - What does the plant protection system and the control room instrumentation see during an instability?

- Improve on-line stability monitoring systems to detect out-of-phase and channel instabilities

WHAT DO WE NEED TO KNOW? (cont)

ATWS Scenarios

- What is the maximum expected amplitude of limit cycle oscillation?
    - What is the associated average power increase?
        --> Does it affect severely suppression pool temperature during ATWS events?
    - What is the associated flow decrease?
        --> Does it affect severely core cooling during ATWS events?
- What is the effect of limit cycle oscillations on ATWS procedures?
    - Instrumentation
    - Operator errors

PUBLIC NOTICE BY THE

UNITED STATES NUCLEAR REGULATORY COMMISSION'S

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

DATE: Thursday, November 9, 1989

The contents of this transcript of the proceedings of the United States Nuclear Regulatory Commission's Advisory Committee on Reactor Safeguards, (date) Thursday, November 9, 1989, as reported herein, are a record of the discussions recorded at the meeting held on the above date.

This transcript has not been reviewed, corrected or edited, and it may contain inaccuracies.