UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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PERIODIC MEETING WITH THE ADVISORY

COMMITTEE ON REACTOR SAFEGUARDS

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PUBLIC MEETING

Nuclear Regulatory Commission

One White Flint North

Rockville, Maryland

Thursday, September 8, 1994

The Commission met in open session, pursuant to notice, at 1:30 p.m., Ivan Selin, Chairman, presiding.

COMMISSIONERS PRESENT:

IVAN SELIN, Chairman of the Commission

KENNETH C. ROGERS, Commissioner

E. GAIL de PLANQUE, Commissioner

STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

KAREN D. CYR, General Counsel
JOHN C. HOYLE, Acting Secretary
DR. THOMAS S. KRESS, Chairman, ACRS
DR. IVAN CATTON, Member, ACRS
JAMES C. CARROLL, Member, ACRS
CHARLES J. WYLIE, Member, ACRS
WILLIAM LINDBLAD, Member, ACRS
DR. WILLIAM J. SHACK, Member, ACRS
CARLYLE MICHELSON, Member, ACRS

P R O C E E D I N G S

[1:30 p.m.]

CHAIRMAN SELIN: Good afternoon, ladies and gentlemen.

The Commission is pleased to welcome once again the Advisory Committee on Reactor Safeguards to brief us on items of mutual interest. This Committee provides an invaluable service to us by giving us independent advice on safety aspects that are proposed in the existing nuclear facilities. Therefore, the Committee's activities are important to us in helping to solve technical problems in licensing and regulation and also sometimes just to give us an overview of things that we may sometimes miss from being so much tied in with the day to day activities.

So, today we are pleased to hear your views, Doctor Kress, on a wide range of issues.

I understand copies of the Committee's letters to the Commission on today's topics are available at the entrance to the room.

Commissions?

The floor is yours, Doctor Kress.

DR. KRESS: Thank you very much for those kind words. We find these meetings very valuable to us as a feedback mechanism to see what you're most interested in.

Before we start, I would like to note that we have

a new Committee member that this is his first meeting with the Commissioners, Doctor Dana Powers. I'd like to introduce him.

CHAIRMAN SELIN: Welcome, Doctor.

DR. KRESS: You probably know him.

With that, I won't spend too much time in small talk introduction. I'll go right to the agenda since we are limited in time.

The first item has to do with our reviews of the passive plant designs. Most of the activities we've had have involved the Thermal Hydraulic Subcommittee and the thermal hydraulics issues, so we'll turn it over directly to Ivan Catton and let him give you a status report.

DR. CATTON: Thank you. I'm not quite going to follow the order that's on the paper in front of you.

CHAIRMAN SELIN: Why am I not amazed to hear that?

DR. CATTON: Well, see, I didn't put it together. I would have arranged it differently.

In any event, both Westinghouse and GE need to have computational tools that predict the behavior of their reactors with reasonable assurance that the uncertainties are known. To accomplish this they have various test programs and they're composed of small scale separate effects tests as well as some integral test systems.

The purpose of these regs is to generate the data

they need to confirm the quality of their computational tools. What this does is it leads you to the generic question of scaling of large complex thermal hydraulic systems. As near as I can tell at this point, neither the vendors nor the staff with their two programs have completed the task. There's still pieces left out.

We actually started looking into the test programs I guess almost two years ago. During that period, both have come a long way. We felt several subcommittee meetings and I first will talk a little bit about the simplified boiling water reactor.

We met last week on August 24th to review the new GE plan for test and analysis programs. I think their presentation reflected sincere planning. They've done a good job in demonstrating how the various phases of a LOCA are covered with their test facilities. We had a number of comments about the completeness of what they're doing. Primarily it's their scaling analysis and, although a good start, it's still incomplete. In particular, attention needs to be given to parallel flow paths and their dynamics. You've got a lot of tanks that are connected together with pipes and these things. You need to be sure that they're going to behave in the proper way so that you get the information you need to qualify your codes.

In this area there is an RES program that includes

the PUMA facility at Purdue and a version of RELAP5 that's dedicated to BWR modeling. The purpose of the PUMA facility is to obtain -- and I don't know what PUMA stands for, but it's to obtain confirmatory data for the assessment of important SBWR-specific phenomena. Scaling was a major part of the presentation and this subcommittee meeting was on August 26th.

An enormous effort went into what is called bottom-up scaling. You sort of work your way around in the bottom and look at rentals numbers and so forth. But there was not the top down scaling which allows you to decide what is important. We felt that if you don't do that, you're going to become overwhelmed by detail and you never get the problem straightened out.

We're not happy or were not happy with how the overall program leading to a computational tool with known uncertainties was structured, but we've been assured that we will be when they present it to us at our October meeting. At that time we will write a letter.

CHAIRMAN SELIN: Would you repeat that last sentence? You were not happy with --

DR. CATTON: We were not happy with the way the program was structured. See, your focus needs to be this code that you're going to use to predict the behavior of the AP -- well, in this case the SBWR, but also the AP600.

Somehow to have an experimental program that is already sort of underway, granted they will probably scale it right, you still haven't really evaluated the needs of your code. In our view, you need some kind of a structure over this that's putting all of the pieces together because you may find that you don't necessarily want to run a particular experiment because it won't teach you much that you need for the code. You may want to do something that doesn't quite look like the reactor because you're scaling tells you this is what you need. That part of it was missing, the sort of integration.

MR. CARROLL: You also said something about a future meeting and a report.

DR. CATTON: Yes. With respect to the RES program, there will be a presentation to the full Committee in October and that will cover their program plan and at that time we will comment on it. We also plan to have a further meeting with GE. They indicated that they had taken a lot of our comments to heart and that they would come back and that they would work with us to get it squared away.

Since we last discussed the AP-600 with you, and I think that's quite awhile ago, we've met with Westinghouse twice and we also had a recent meeting on the ROSA facility. We met with Westinghouse last March to discuss their separate effects test. This is where they look at their

core CMT. I don't remember what the M stands for, and their -- core makeup tank and their PCCS, which is the passive containment cooling system. We weren't too happy with their program. The PCCS test program was to generate data that would confirm their containment analysis tools. Unfortunately, the testing is completed before any attempt at scaling is carried out. So, it wasn't clear to us that the test was run with the right set of conditions.

A lot of work was noted that had just begun and we were very happy to hear that the reg. had been mothballed so that if scaling analysis led them to conclude more testing was needed, they could do it. And I think the staff people who were there with us at the time sort of agreed with our perception of the program.

COMMISSIONER ROGERS: What test rig was that again?

DR. CATTON: The passive containment cooling system. It's sort of a hemisphere set out in a field somewhere and they pour water on the top of it and dump steam inside. The concern was the water flow on the outside of it may not have been appropriate for the test. There were a number of questions, detailed questions like that that came up and we were assured that they were going to look into them and that we would meet when they were ready. I believe they have contacted the ACRS office to try to

schedule a meeting. So, it should be soon.

The CMT also suffers from a lack of scaling analysis. But in the case of the CMT, there are a number of them. They have a tall skinny one with the facility in Italy. They have the separate effects tests at Westinghouse. They have a short fat one at Oregon State and you also have the ROSA facility. So, there's a wide band of scales and I think that if they do an analysis and they come up with the proper kinds of initial conditions, they can run the hell out of these things and then they can be sure that their codes will do the job.

Westinghouse made a large number of commitments. I just recently looked through the minutes and they go on and on. I haven't received any of them and it was last March, but I assume we will one of these days.

CHAIRMAN SELIN: Doctor Catton, your remarks leave me a little bit puzzled. You're saying if they do these tests with the right initial conditions --

DR. CATTON: Well, see, initial conditions is key.

CHAIRMAN SELIN: I understand that. That's not what --

DR. CATTON: Okay.

CHAIRMAN SELIN: What I'm missing is are you neutral as to whether they're going to do this? Do you believe they will do this with the right conditions? Are

you skeptical that they'll do them with the right -- what are you telling us about what you expect them to tell us?

DR. CATTON: I haven't seen anything that they have put in writing. I've heard a lot of probablys and maybes. So that's why I'm not being too positive about it. In one respect, we're really not sure what those initial conditions should be. You need to do this kind of a scaling analysis to tell you what they should be. Once you've got them, you can exercise the facilities against them and then practice with your code.

I don't really see this happening. When you ask Westinghouse whether or not they're going to do calculations of ROSA, they don't say yes. They don't say no, but they don't say yes. So, I'm just not sure.

CHAIRMAN SELIN: So, you give me the impression -- my impression is you're saying there's a range of facilities which could support a good test program.

DR. CATTON: Yes.

CHAIRMAN SELIN: But you haven't seen a good test program yet.

DR. CATTON: That's right.

CHAIRMAN SELIN: Okay.

DR. CATTON: I haven't seen everything put together yet.

COMMISSIONER ROGERS: Excuse me. On that last

point though. Westinghouse hasn't said they'll do ROSA calculations.

DR. CATTON: No.

COMMISSIONER ROGERS: But we've said we'll do ROSA calculations, isn't that right?

DR. CATTON: Yes.

COMMISSIONER ROGERS: So they are being done. It's just that they are going to be done, but they're not going to be done necessarily by Westinghouse. Is that right?

CHAIRMAN SELIN: They have to be done --

DR. CATTON: Well, but you see, the code that you have to depend on is the Westinghouse computer code, not your own. Your own is a separate issue. If Westinghouse --

COMMISSIONER ROGERS: Well, the code that's being used the analyze the ROSA experiments right now.

DR. CATTON: That particular code is RELAP5.

COMMISSIONER ROGERS: Right.

DR. CATTON: That's an NRC code.

COMMISSIONER ROGERS: Right. Now, those calculations are being done.

DR. CATTON: Calculations are being done, that's correct.

COMMISSIONER ROGERS: But you don't have a

commitment or nobody's got a commitment from Westinghouse saying, "We'll use their code on the ROSA facility results?"

DR. CATTON: That's correct.

MR. CARROLL: Your point, Ivan, is that it's their code that's going to be the licensing basis for the plant.

DR. CATTON: That's right.

MR. CARROLL: RELAP5 is simply confirmatory.

DR. CATTON: And the system is complex. There's lots of pipes hooked together and levels move up and down and I guess there was even one case with ROSA where the CMT started to drain, then stop and sat there for an hour before it started to drain again. Well, this may well be due to the atypicalities of the ROSA facility, but on the other hand it may not. I think what you buy is the necessity to explain it and to show that you can indeed calculate it. What this leads you to I think is a much more robust code when you're done, but it may be pain that you well could have done without.

We met with Westinghouse on their COBRA/TRAC program and we had the usual complaints about the codes. But for the most part it's my belief that it's acceptable to begin their validation process. Our primary concern when we met with them was how they're going to treat uncertainty in the predictions. What Westinghouse wanted to do was just to sort of lump everything into a square root of the sum of the

squares and they would actually look at data across the channel where temperatures are higher and temperatures are low and sort of treat everything that wasn't on the mean as an uncertainty. That's not an uncertainty. Those measurements are repeatable, they're not uncertainties.

We had a lot of argument about that and no agreement when we left. To me, other than the comparisons with the actual test facilities, that's the biggest hurdle to get over, is to convince them that indeed they have to treat that as a variable.

With Westinghouse we really found no show stoppers, but I think there's a lot of work that still has to be done.

CHAIRMAN SELIN: Would you care to speculate as to how much -- how the work that has to be done compares with the schedule that we're trying to carry out?

DR. CATTON: No, because what happens is we had the meeting, then we don't hear from them for a long time. I know they're running the OSU facility practically day and night, but we have yet to see any of the data. I understand that Research has some of the data, but I haven't had an opportunity to talk to them. So, I just don't know.

On the 25th, we reviewed the NRC research program at ROSA and it was at that meeting we came to the conclusion that the really -- again, much like when it was not a

systematic research program, the focus is really not on the code and its uncertainties. They're not, at least that we could tell, are not meshed well enough. Again, we were assured that this will be remedied and that we're going to be told about it at our October meeting.

There are some interesting preliminary results from ROSA. The strongly coupled dynamic interacting components shows in the recent tests, particularly the hang-up of the CMT.

I guess to close, we'll be writing a letter on the RES program, both PUMA and ROSA, following a meeting with them at our October meeting.

COMMISSIONER ROGERS: Yes, before you leave the ROSA topic, some time ago you expressed great concerns about the ROSA facility being adequate.

DR. CATTON: Yes.

COMMISSIONER ROGERS: Not modeling the AP600 design, many features that you described in detail that were different and so on. The question in my mind is to what extent if the NRC RELAP code, in fact, does predict the results, the experimental results from the ROSA rig, whether the difference then of the ROSA rig from the AP600 design is a scaling issue. In other words, would that be a scaling concern, that difference? Could they adjust the RELAP code to correct for the differences between whatever is in it

that makes it work very well for ROSA but is inappropriate for the AP600 detail design?

DR. CATTON: I think what you -- I'm not quite sure how to address the question, so let me try to just give you an answer and if I don't hit on it, we can try again.

The ROSA is different and it has atypicalities.

COMMISSIONER ROGERS: Right.

DR. CATTON: Right now today we don't know, or at least I don't know and haven't been shown, how that behavior compares with the AP600 or how I get from one to the other. So, you have to use the code to bridge the gap. Now, you've got SPES, OSU and ROSA, and somehow you have to show, and this is what you do with the scaling, that you're going to capture everything you need into that code that you're going to exercise against these different facilities to convince people that when you predict AP600 the predictions are meaningful. If you just practice against ROSA, that's not going to do it because what happens is that you'll wind up adjusting a nodalization or something will happen and you get good predictions. You need to know why you got good predictions.

COMMISSIONER ROGERS: Yes, absolutely.

DR. CATTON: And then you need to go to the other facilities and they all have different kinds of problems. Although some of my colleagues on my subcommittee don't

agree with me, my feeling is that if you did a damn good job of predicting all three without really changing the code in any way, I would feel reasonably comfortable with the prediction of AP600. But you haven't proven it until you do the scaling.

COMMISSIONER ROGERS: That would be wonderful if it turned out that way, but I'd be surprised, just offhand, whether that code without some adjustments could, in fact, deal with those very different scale --

DR. CATTON: If you can't deal with those three facilities, then how do you say anything about your prediction to the behavior of the AP600? You can't. You have to know what's going on in order to make this kind of a prediction and that's a problem. Fortunately, I think the OSU facility is well scaled, but there's still this problem.

By the way, the OSU scaling report doesn't deal with this top down either, but we have been assured that they will. That will help you better come to grips with initial conditions because you can do things with it. You could change how you run your experiment on OSU in order to capture other phenomena if you know what you're looking for.

COMMISSIONER ROGERS: Yes.

DR. CATTON: This is the piece that's still missing.

COMMISSIONER ROGERS: Do you see -- well, you've

been told --

DR. CATTON: We're headed in the right direction.

COMMISSIONER ROGERS: Yes. Do you see a clear set of steps that could be taken to take care of that?

DR. CATTON: Yes.

COMMISSIONER ROGERS: Yes.

DR. CATTON: The first thing you need to do is to complete the scaling and I think this needs to be done for ROSA. Now, we've been told that it was done for ROSA, but they didn't put the things together right for us and so forth and it may well be true. But it needs to be done for all of the facilities so that you can begin to look at the results of this effort and then compared with what you have for AP600.

See, one of the difficulties, I think, is that we're educated to use a different approach, to start with the detailance or to work our way up. Well, if it's a simple problem like a pipe and a Reynolds number, that's not a problem. But when you have this complex system and you've got different heights to worry about, different diameters, different kinds of processes and things that don't scale, because after all you're using the same fluid in all scales. So, some of the thermal physical properties don't change and you have a problem with that, and you sort of have to figure out how to put it together and that's foreign to a lot of

engineers. We learn to come from the other direction.

COMMISSIONER ROGERS: Okay. Thank you.

DR. KRESS: Okay? Thank you, Ivan.

The next topic comes out of our reviews of the evolutionary plant designs. In the process of our reviews, we asked a lot of questions and got a lot of answers that left us with some items that, although they were fixed to our satisfaction for the evolution plants, we think there still are some lessons learned and William Lindblad is our leader on this one.

MR. LINDBLAD: Thank you.

As Doctor Kress was saying, yes, in July, upon completion of ABWR and System 80+ reviews by the Committee, and after the letters were written on those two projects, the Committee sat and talked about what might we bring to the attention of the staff. So, in July, we wrote to the Executive Director a letter identifying seven items that we thought he might want to have his staff give consideration to how operating nuclear plants were handling the same issues or how their review of future projects might one look at it as well.

We suggested that there was no particular way these items might be addressed. He might address them as a research item or as a change to the standard review plan or consider them and decide they need not be given any greater

attention than that. We have not yet heard back from Mr. Taylor on this and so, while it's accepted, we expect to hear momentarily how he intends to deal with our suggestions. If you have any questions about clarifying some of the items, we'd be pleased to attempt to do that. But we really can't tell you much more than that.

CHAIRMAN SELIN: I had a question different from the items, but from the overall process as opposed to what we've learned about designs. Did you end up with strong feelings one way or another as to the whole Part 52 certification process having been through this stage at this point?

MR. LINDBLAD: Well, it's a new process, new for the staff, new for the Committee, and it's not a complete process yet either because we're just certifying and we haven't seen a buyer and a combined license application yet. So, yes, there are the reservations among members of the Committee how the process is all going to work out. But we'll say we hope that there will be buyers and we hope that we will see the process completed to see how it works out. But we do believe that the reviews -- the staff gave the proposals and the reviews that the Committee gave to the proposals were very thorough.

CHAIRMAN SELIN: Thank you.

DR. KRESS: I think we've got a feeling that the

certification process with the ITAACs and DACs will work and this looks like it can be a good process.

COMMISSIONER ROGERS: Did you have any opportunity or did it occur to you to think of how the lessons learned might be useful to us in reviewing a CANDU reactor design?

MR. LINDBLAD: I don't think we paid specific attention to that. But in the past year we did address the status of CANDU with the staff and we're still waiting to hear from the staff how we might review that at some time.

COMMISSIONER ROGERS: All right.

CHAIRMAN SELIN: Thank you, Mr. Lindblad.

MR. CARROLL: Just skimming through them quickly, I'd say virtually all of them are applicable to CANDU.

MR. LINDBLAD: On the specific ones in our letters, yes, but we didn't look for new ones that might be specific to CANDU because we don't know that much about that proposal yet.

DR. KRESS: We're doing quite well on time here.

MR. CARROLL: Average.

DR. KRESS: Average on time. That's quite well.

COMMISSIONER ROGERS: So far so good.

DR. KRESS: The next item is mine. It has to do with our protective action guidelines letter that we wrote on July the 13th this year. You might recall that that letter was prompted by a request from one of the

Commissioners. I think it was Commissioner Remick, but I'm not even sure of that. When System 80+ came in with the calculated design basis accident having dose values at the site boundary that were less than protective action guidelines for the new source terms, the question was asked if we thought there were any implications of that and what they were.

So, we took at look at it and we didn't think there were any particular significance to this with respect to the new source terms. It just so happens that System 80+ has a very good containment and you would expect as containments get better and better eventually one of them is going to have this calculated result. It's just to be expected as plants get better and better. So, it really didn't have much to do with the source terms.

But in the process of our review, it occurred to us that this may come up again with newer plants, advanced plants, that may be even better than System 80+ are likely to be and at some point you may come in with a plant that has a risk profile that is so good that the question will be asked, "Well, could we relax the emergency planning zones and the associated PAGs?" In order to address that question, we thought that what you needed was to go back to the emergency planning criteria of the PAGs and the emergency planning zones in particular and develop a risk

basis for them because the basis that they now have is only very loosely and ill stated in terms of risk.

With that risk basis, which may not be hard to come by, you could have an understanding and maybe guide your thinking on what might be a potential way to possibly relax these for newer plants that may really be very safe and have a risk profile that is acceptable without them. You may very well want to keep them for defense in depth purposes or other political reasons, but at least you would know what the risk basis was and hove a correlation of some sort to relate to the size of the emergency planning zones.

Now, this is not a straightforward thing because the independent variable is the size of the emergency planning zone, but that's tied in with population, winds, other meteorological conditions and how effective your emergency actions are as a function of that. So, it's not a straightforward conversion to risk and we thought this would be a good chance to take a look at how you might formulate a purely risk-based regulation and use it to guide your thinking.

As a matter of fact, the rationale letter that's later on in the agenda was really originally a paragraph in this letter because we thought that it addressed an overriding issue that has to do with all regulations, not just these particular ones. Some of the members thought it

was a subject that detracted from our PAG letter and ought to be a letter of its own and that's why it just sort of showed up out of the blue at one time. It addressed the same subjects, but was broader and we thought you ought to go back and look at the various regulations and see if there was a way to put them on a risk basis just as -- not because we want to -- think you can replace the body of regulations, but to give you a supplementary risk rationale to them to guide the thinking and to guide future uses of them. So, that's what that was about.

MR. CARROLL: Tom, in your discussion you used the phraseology "defense in depth and other political considerations," implying that defense in depth is the political consideration. Do you want to correct that?

DR. KRESS: No, I didn't -- I would like to correct that, if that's the impression I left. I think defense in depth is a very good concept. It's not political at all. I think the political considerations might be for perception purposes you want an emergency plan, not from risk basis at all. You may want one anyway. That's what I had meant, but not defense in depth.

CHAIRMAN SELIN: The question that your remarks raised really has more to do with your eighth item than to do with that specific item on the EPZs. That is -- I have to tell you I'm a little bit confused. Do you think we're

going too fast or too slow towards trying to do some probabilistic basis in our regulation or both? In the past you've chided us for going too slowly and your remarks today would go with that. But on the other hand, a couple of your letters give you the feeling that you thought we hadn't really thought out very clearly what we were going to do with the probabilistic results when we got them.

DR. KRESS: That's my --

CHAIRMAN SELIN: What would you like to see us do?

DR. KRESS: The ACRS hasn't discussed this fully in committee. So, when I'm speaking, I'm speaking for myself and some of my colleagues may differ. I think there's a need to go back and pretend like you didn't have the regulations that we now have and say if we were going to have a fully risk-based and perhaps performance-based regulation system, what would it consist of and how would it be formulated and how would one do it using PRAs and using acceptance criteria?

Starting from that, one could devise what would be needed to have this risk basis and then one would have to address the body of regulations that we do have and see how they conform to that which is not an easy task. You have to ask how each of the regulations affect risk and that means asking how they affect things like core damage frequency, containment failure probability, the source term and those

sort of things, and you have to translate that into things that are useable with the PRA. Then you do something, I think, that is quite similar to what you've done for implementing the safety goals where you use the PRA in a delta fashion to look at a delta core melt frequency and a delta containment failure criteria.

I think you'll do that with a -- you're doing that with new regulations. I think you can go back and do a similar thing with existing regulations, but go even further because you have to deal more with source terms and other issues that don't show up in that process.

But I think this overall look at the body of regulations is what's missing. I think the PRA implementation and the thing is being done piecemeal and looking at particular things. I mean it's being used very nicely and regulatory analyses is being used very nicely in the IPEs and other things, but it's not, I don't think, being looked at for the whole body of regulations and that's what we had in mind.

Was that helpful?

CHAIRMAN SELIN: Yes. I'll be blunt about it. I'm trying to figure out if I think you meant exactly what you said or what you might have said differently. What you basically told us is to rewrite all our regulations from scratch as if we were starting today and I don't think you

quite mean that.

DR. KRESS: No, no, I didn't mean that. I meant --

CHAIRMAN SELIN: I think what you mean is we've got to take a look at the overall regulatory body and say, "If we were starting today, what would it look like? How different is that from what we have today?"

DR. KRESS: That's more closely what I'm --

CHAIRMAN SELIN: Equivalent of Doctor Catton's top down analysis. Where should we be putting our effort because it's too different from -- the current regulations are too different and where should we just sort of let things stand?

DR. KRESS: Well, the regulations, I think, are quite adequate in assuring safety of the plants. I don't think you can start all over and throw them out. I wouldn't advocate that for a second. I think it would be good to go back on a systematic basis and try to see what is a good underlying risk rationale for them and not as a replacement for the regulations, but as guidance to your thinking on how to interpret them for new plants and when you make changes how to interpret them and to perhaps address this question of coherence in the regulations.

I think the underlying thread that ties all the regulations together ought to be risk and that's the way to

get coherence into it. That's more the tone of what I meant. I think you have to live with the body of regulations you have and you have to somehow get risk into the system and bring them along together and keep them both in place. There ought to be a way to do that.

DR. CATTON: I think you've missed a couple of opportunities. I think this Appendix J leak testing, at least from what we've seen, no matter what you do, it doesn't change risk very much. So, it ought to be relaxed. Another area is --

MR. CARROLL: And that's exactly what the staff is proposing to do.

DR. CATTON: Well, I'm not sure.

MR. CARROLL: Oh, yes.

DR. CATTON: Another area might have been in the Thermo-Lag issue could have been carried into a risk-based fire protection. Well, I'll comment on that in a few minutes.

CHAIRMAN SELIN: Thank you.

COMMISSIONER ROGERS: Before we leave this, on your point 4, I guess it was, on acceptable risk, this is what we started out with before we got into the larger issue. Can you give some indication to me of what relationship you see between our safety goals and determining an acceptable level of risk? How do you relate

those?

DR. KRESS: Yes. Let me --

COMMISSIONER ROGERS: What's different about them to begin with?

DR. KRESS: Let me use an example and that would be the protective accident guidelines, for example. My feeling is that we have an inverted view of where the risk level of adequate protection lies compared to the safety goals. There is a widespread view, in my opinion, that adequate protection is at this risk level and the safety goals are at this risk level. I think they're not. I think they're the other way around. The safety goals are a higher risk level than what we have achieved by the body of regulations that we have and what we call adequate protection. So --

CHAIRMAN SELIN: I think --

COMMISSIONER ROGERS: But we don't really have a measure of that yet, do we?

DR. KRESS: Yes. I think that's what NUREG-1150 in a sense is.

COMMISSIONER ROGERS: Well, for some plants.

DR. KRESS: Well, yes. If you accept NUREG-1150 with a significant difference between safety goals and make some adjustment for the body of plants or maybe look at the IPEs, we don't really have a measure to correct. I'm giving

you -- my feeling, is that the body of regulations have resulted in a risk level that is considerably below the safety goals.

Now, given that, if you're going to go back and do a risk-based set of regulations, one might think a starting point was you have to have an acceptable risk and keep below that. That's the essence of risk-based regulation. One might think safety goals is the starting point for that. I think that would be a mistake because you're already well below those and I think you should actually look at the body of regulations and use the risk level we've achieved as an acceptable level of risk.

COMMISSIONER ROGERS: Well, you know, I think that's a very interesting point of view. It's really quite a dramatic statement.

DR. KRESS: It is, yes.

COMMISSIONER ROGERS: We don't really know that --

DR. KRESS: We don't know.

COMMISSIONER ROGERS: -- we have, in fact, achieved the safety goal.

DR. KRESS: We don't know.

COMMISSIONER ROGERS: We don't really know that. I think there have been suggestions from time to time from ACRS that we try to measure that.

DR. KRESS: Yes. That has been -- and this is one of the reasons for it, is --

COMMISSIONER ROGERS: And my own feeling is that we ought to get -- we ought to come to grips with that if we could.

DR. KRESS: Yes, and we thought you could use the IPEs for that with some enhancing --

COMMISSIONER ROGERS: If you do a level 3 PRA.

DR. KRESS: Yes. Yes.

COMMISSIONER ROGERS: And everybody isn't doing a level 3 PRA.

DR. KRESS: I know. We thought there may be some ways to do something with them though.

COMMISSIONER ROGERS: Yes.

DR. KRESS: Using bounding analyses for the site characteristics and things. But that's my feeling. For example, if you were to put a measure of the risk that's acceptable with the emergency planning zones, as an example we now have. It's the risk we now are accepting, which is lower than the safety goals in my opinion. You wouldn't want to start with the safety goals if you're going to develop a new set of regulations that are risk-based. Not only that, the safety goals do not, in my mind, allow you to deal properly with the uncertainties. I think the safety goals should have been written in terms of some uncertainty

levels. What that allows you to do, for example, is allow regulations that might be construed just to reduce the uncertainties, not to reduce the risk.

Nowhere in the regulations do we have a system that allows that. So, I would have formulated the safety goals in terms of certain level of confidence rather than the actual means.

DR. CATTON: It's not an easy problem.

DR. KRESS: It's not an easy problem.

DR. CATTON: They're struggling with this in Australia where they're trying to go from prescriptive fire regulation to performance based fire regulation. One of the things they're having a great deal of difficulty with is baselining what they've got because the new regulations can't make it less safe and it's a problem.

MR. LINDBLAD: Tom has suggested that PAGs is where we might visit risk-based regulation. I really think that may be one of the more difficult places because with PAGs is where we meet two other agencies who may not be on the bandwagon as much with risk-based regulation as those two agencies are.

DR. KRESS: I suggested that one because with your SRM to the staff you actually requested that they take a look at that and I think the ACRS has some ideas that we could give to the staff on our thinking on that, how to

actually do that and that's why I suggested that as --

COMMISSIONER ROGERS: Well, this might not be the best place to interject this thought, but I'm going to do it anyhow. That is we heard the other day a presentation from our staff on where they're using probabilistic risk analysis and it was very interesting. At the end of the discussion, I raised the question of whether it might be appropriate to draw a distinction that many countries have done between risk analysis and safety analysis, PSA versus PRA. As you know, in Europe and in Japan the analyses are done and called safety analyses rather than risk analyses and, in fact, in most cases don't involve a level 3. They don't really involve the actual consequences that you have to add onto the probabilities to give you risk.

I wonder if you -- my understanding is that you folks haven't been too comfortable with a change in terminology here and I wonder whether since you've just touched on this issue of other agencies, other considerations being involved when you start to look at the consequences, which are what you're talking about when you look at the protective action guidelines, that focusing on the probabilities not on the public health and safety consequences directly, but the probabilities of core melt or containment failure with some release of radiation, but not going that further step and calling that a safety analysis

and focusing upon the mechanics of that might be a useful way to approach as a first step this kind of a review.

DR. KRESS: There has been some strong feeling among some of the Committee members that it is a risk analysis, not a safety analysis. One is sort of the compliment of the other one. And we would, I think as a Committee, say we prefer PRA, but that's probably because we're insensitive to things like public perceptions and that sort of stuff. But we would have preferred the PRA.

With respect to focusing on core melt frequency and conditional containment failure probability, I think that's a good idea for a lot of things. With respect to the particular example of PAGs, I don't think it is because that happened to be one that encompasses everything and that's why I suggested it as a good start for risk, because it deals with the source terms and those other two things as well as meteorology and siting. It has everything we have in it and that's why it's a good place to start, because you can think of all of those things at the same time.

COMMISSIONER ROGERS: Well, I would hope that you would continue to think about this kind of an approach and these problems because it seems to me that this is something of a mark of the maturity of the technology and the science and the regulations all coming together to be able to do this. We know how we've gotten to where we are and I don't

think we have to make any apologies for it. That was the way life was. We had to make progress. But now is a time for reflection and some introspection and I think that trying to bring together everything we know and everything we've accomplished and putting it on a solid quantitative foundation is a very good thing to do if we can afford to do it.

DR. KRESS. Yes.

COMMISSIONER ROGERS: I think the "afford" is a big question in my mind. I don't know how we determine that, but, at any rate, I think it would be very good for us to try to think of how we could bring these things together because what you've said here today has not been said really, that in your opinion we really are well below the safety goals.

I think there's still a lot of arguments around of whether we're anywhere even close to the safety goals in some people's minds and trying to establish where we are with respect to safety goals. I've never heard anybody question the safety goals as an acceptable level. I've never heard anybody say, well, that's really not good enough. Maybe there are people who say that, but I haven't heard it. And if we're well below the safety goals, that's a very significant --

DR. KRESS: That is a significant finding.

COMMISSIONER ROGERS: -- piece of information that I think everyone ought to appreciate, but it's got to be soundly based to be able to make that statement. And I think how one gets to establishing the credibility of that statement is very important and I would hope that you could give us some guidance and thought on doing that.

DR. KRESS: We certainly will take that -- I have revitalized our strategic planning process and that is certainly high on our list of one of the things to look at and we will be sure it stays there, yes.

MR. LINDBLAD: All of this discussion has been on formulation of the regs. Of course, there is the opportunity to do resource allocation and identify research needs with PRA results and I suspect that that's valuable to you too.

COMMISSIONER ROGERS: Absolutely.

DR. KRESS: I guess we should move on now to the next item. It's yours, Ivan, on the Thermo-Lag issue.

DR. CATTON: I'm not sure how much you want to hear. As you know, the ACRS chose the staff option 1, business as usual, with my added comments recommending option 2.

MR. CARROLL: No, they didn't.

DR. CATTON: Well, it looked like that to me. I actually was the author of the letter before they got

through with it.

I was pleased to see the Commission approved the continuing use of option 1 with option 2 being the basis for exemptions. It's my view that when calculations are done that consider the actual fire loadings in some of these various rooms within a reactor building you're going to find that the present Thermo-Lag application will probably survive the three hours, so I was really pleased to see that.

I think, also, option 2 is a good place to start to develop risk-based fire protection. You first have to do the calculations and then you have to calculate uncertainties and you have to put it all together.

In the same vein, right after that letter was written I attended the Fire Science Safety Meeting in Canada and it turned out its theme was risk based fire protection. I was surprised to see that there were no NRC people there. The first paper was by -- and I have a trip report that you might find interesting. The first paper was by a fellow named Olaf Pederson from Sweden who is a civil engineer that helped develop their program, and they've had risk based -- or, they call it performance based, performance based fire regulation since back in the early '70s. New Zealand also has it and they put their whole -- their entire fire regulations are on a page and a half.

MR. CARROLL: But these are not nuclear plant fire regulations.

DR. CATTON: No, they're not, but, see, one of the questions that was raised during our discussions with the staff was the need for the tools. This fellow from Sweden mentioned that there are four what they see to be acceptable calculational tools for predicting the impact of fire in a space.

Now, their interests are a little bit different. They don't want the building to fall down on somebody. They want it to stay up long enough for them to get out, and that's where the structural guy got involved in it. The beam has to hold the load even though it's half burned away.

There was also a very interesting paper, which I don't have a copy of yet, called "Magic Numbers and Golden Rules." One of the things they cited in this paper was the 20 foot separation that's supposed to represent a three hour barrier and how that was just patent nonsense, that in some places what they actually would do is -- a barrier is a barrier, so that means if it's three feet, four feet, five feet, doesn't matter, and in many cases it actually was above where the thermal layer would build up so the hot gases just pass it right by, but it meets all the regulations. That's the down side of prescriptive regulation and I don't know how many of those kinds of

things are built into Appendix R.

I think you'd be a lot better off to jump right in and this might be a place to practice. The French are actually doing that. They want to go to the fire based -- the risk based fire regulation, but to do it they're actually running experiments, developing the tools and practicing with this process before they actually implement it. I don't see any of us doing that. It might be a good idea to start.

CHAIRMAN SELIN: I guess I should start my answer with we're not complete idiots, you know, and we have two problems.

DR. CATTON: Just partial.

CHAIRMAN SELIN: One is to replace Appendix R with a performance based rule. The second is a remedial problem to do with Thermo-Lag. And, if they hadn't occurred at the same time, I don't think -- I mean, it's not to down play the value of the advice, but I just think that if we try to solve both problems at the same time not only do we have an unacceptable approach specifically but we would be widely seen, and correctly so, as using risk base to justify substandard performance. I haven't been able to figure out a way out of that, other than to first say you have to meet our current standards. When you meet the current standards, then we can talk about going to performance based --

DR. CATTON: But there is a step before you get to the risk base, and that's just do calculations. And that's essentially what the option 2 recommends.

COMMISSIONER ROGERS: Yes, that's right.

CHAIRMAN SELIN: I thought we'd instructed the staff to do that.

COMMISSIONER ROGERS: I think that's exactly what we --

DR. CATTON: And you learn to do the calculations. Your staff learns to accept what comes from calculations, then you'd make the next step. But you've got to start and I was pleased to see that you're going to.

COMMISSIONER ROGERS: No, I think that's exactly the philosophy, but, I think, as the Chairman said, the public perception would be very, very clear that everybody isn't happy about using risk analysis for anything and to put it on top of this situation with Thermo-Lag and come to the conclusion that, well, really, you know, we really don't have to worry because we've just done a new risk calculation. I think it would just be a totally unacceptable way to go for --

CHAIRMAN SELIN: I don't think you're idiots either.

MR. CARROLL: No, we aren't, and in fact we had the same discussion among ourselves.

DR. KRESS: Yes, in fact I think our letter almost --

COMMISSIONER ROGERS: But we decided, yes, go ahead and do the analyses, learn how to do these things.

DR. CATTON: Well, and you're going to allow them to use option 2 as the exemption route.

CHAIRMAN SELIN: That's correct, which is no different from what they could have done last year or the year before or the year before --

MR. CARROLL: Yes, it is.

DR. CATTON: Yes.

MR. CARROLL: The staff was very adamant that, unless you told them that they could expand their base of exemptions, that they were going to hold to what they had historically given.

DR. CATTON: So, actually, this is a major breakthrough in a way, that they will now be able to do analysis of what they have in hand in order to then try to get the exemption. I think that's a major step.

CHAIRMAN SELIN: Okay. I didn't understand it that way.

DR. CATTON: We're very pleased to see it.

CHAIRMAN SELIN: I've understood that the basis for an exemption is supposed to be the performance calculation, but specific to a particular --

DR. CATTON: But, you see, what they're demonstrating is that they have a three hour barrier. And I think that's what your requirements are, that they have a three hour barrier, and they can show by analysis that they've got it.

COMMISSIONER de PLANQUE: From this discussion it's clear why the staff was not at the conference. They were trying to figure out our SRM.

CHAIRMAN SELIN: You've been a great help throughout this whole process, Doctor Catton, and I've personally found it very helpful to discuss with you, as we go through the process, not so much how to write an SRM but what action should be taken at this point. Because, there were a number of -- you know, now it looks sort of simple in retrospect, but there were a number of ideas tried out that we hoped would be a magic solution and they just didn't work out at all. And the evaluation is not so much a process that we followed, that was pretty straightforward, but the evaluation of these other hopeful but not successful approaches was very important and you and the Committee were a bit help to me personally and I think to the Commission as a whole as we went through that process.

MR. CARROLL: One interesting aside, one of our summer interns was given the assignment to talk to the people that run these tests and tell us how much fuel it

takes to run a three hour fire test. Amy came up with a very nice report and I'm sure we can share it with you, but it's a lot of fuel compared to what exists in a real world nuclear plant probably any place but the diesel rooms.

COMMISSIONER de PLANQUE: You mentioned the French program, that they're moving towards the performance based. Is this the nuclear industry?

DR. CATTON: Yes.

COMMISSIONER de PLANQUE: Not just the general fire --

DR. CATTON: Not just general. It's for the nuclear. In the trip report, I have the person's name. He indicated an interest in communicating with us on what they were doing. I can get it to you.

CHAIRMAN SELIN: Thank you very much, Doctor Catton.

DR. KRESS: The next item is the National Academy of Science workshop.

Bill?

CHAIRMAN SELIN: Welcome, Doctor Shack. It's nice to have you.

DR. SHACK: Thank you.

The workshop is sort of a jam tomorrow kind of arrangement. As you know, the proposal hasn't quite been finalized yet, although I got a note from our staff today

that it's imminent. It's always sort of been imminent for months, it appears.

We did have a presentation from the staff of the National Academy of Sciences in July, and, again, one of the virtues of a National Academy study is that it's very independent and we'll really know what's going to happen after the panel is selected and they decide what it is exactly they're going to do, but from the description that we had it seemed to be addressing the kind of issues that we thought should be addressed.

It does still seem to us an excellent route to tap all the expertise that's in other fields on digital systems in critical applications, and, again, from the broad outline of the description that was given by the National Academy of Science person, the proposed make-up of the panel, it does seem as thought it will meet that goal. We hope to interact with that panel after it's selected and, again, that's when the real meat of the study will begin is once the panel is selected. We have expressed an interest in meeting with them and they've essentially expressed a willingness to meet with us to discuss some of our concerns and interests.

Specifically, there was some question as to how deeply this should go into human factors work. Again, it's difficult from the makeup of the panel. It seems to us largely weighted towards hardware and software problems,

which is where we think the study should be. But again, one has to actually see when the panel is appointed and what they really decide to do before one can make a final decision.

DR. KRESS: You can have the next one also.

DR. SHACK: I get the next one too.

The next topic is the voltage repair, the voltage based repair criteria for steam generators. We just finalized our letter this morning on that subject in which we recommended that the proposed generic letter be sent out for public comment, which is of course good because it was sent out for public comment, and we don't have a fundamental disagreement with that. There was a differing professional opinion that examined some of the issues there.

We believe that the voltage based criterion applied to the situation in which it is being applied, that is the outer diameter stress corrosion cracking confined to the tube support plates, is not likely to lead to any significant increased risk in tube ruptures. There is, again, considerably more uncertainty in the leakage associated with allowing the steam generators to operate this way.

It is clear that one has fundamentally changed something here. That is, there is now a reasonable chance that you're going to be operating with the primary coolant

boundary breached and you will have leaks, which again is not something that's been done in the past is tried to maintain that to be leak tight and not operate with cracks. The correlations that are used to determine the leak rates are empirically based, and, again, a limited amount of data, although probably enough to go forward with it.

We recommend in our letter that some more attention be paid to the calculation of the radiation doses associated with operating with the steam generators in this condition. The staff has presented an analysis. Some of the members of the Committee have taken different approaches to looking at the releases associated with this during a main steam line break and all the approaches seem to indicate that one does meet the Part 100 limits, but the margins are a little bit uncertain and it certainly warrants further consideration.

COMMISSIONER ROGERS: Well, have you resolved this question of the staff presentation and the Committee's view of things? My impression was that there was some question about whether the staff's information was correct or not that was presented to the Committee. Has that issue been --

DR. SHACK: That issue has been -- there was an error in the presentation to the subcommittee meeting last month which was identified by the staff and they promptly

notified us that there was an error in their presentation. We had a presentation this morning essentially where that analysis was redone. Again, it really didn't lead to different conclusions, although it did lead to the notion that the margins that they had thought were there were substantially reduced.

DR. KRESS: By the way, we did express our appreciation to the staff. We thought that was highly professional behavior to come forth and they're to be congratulated. We like to encourage that, and so I thought I'd get that on the record here.

COMMISSIONER ROGERS: Good.

CHAIRMAN SELIN: Thank you very much, Doctor Shack.

DR. KRESS: The next item we've already discussed. It's the rationale letter. Unless you have additional questions on that, we can go on to the selection of new ACRS members.

As you know, we have one vacancy now and we'll have another one very soon, so we're trying to fill two positions at the same time. The Committee has tried to put together a set of criteria that are quantifiable, quantifiable by looking at résumés and applications and other things, so that the panel that you select whenever that gets put together has some way to deal with the number

of applications we're getting in a quantifiable systematic way, plus they can also put judgement into that also. We've been wrestling with what those criteria ought to be and how to quantify them. We've come up with some thoughts and are still working on the quantification of these.

In the meantime, the ACRS members themselves, believing that they probably are in position to know who some very top notch candidates might be if you start from the top down and say who are the actual best candidates based on personal knowledge and status in the community and that sort of thing, we've come up with a number of names of our own. I think we had six on our list and we're in the process of contacting some of those, and I think two have already opted out of being considered. It leaves us with four on the list that are amenable and would like to be considered. We're in the process now of prioritizing those and perhaps getting another name, so we might end up sending the panel itself five choices from the ACRS and we may send those independently to you. I don't know if you'll end up with --

CHAIRMAN SELIN: That's not the process. I mean, the process is people have to apply.

DR. KRESS: Yes. I'm not sure what the process --

CHAIRMAN SELIN: I mean, the right avenue for

people you think would be good candidates is to encourage them to apply.

MR. CARROLL: They have applied.

DR. KRESS: They have. That's the first thing we've done, yes.

CHAIRMAN SELIN: But once they're in the flow, then there's a well-defined process for considering the candidates.

DR. KRESS: Yes. The question I would have, then, is you would not expect to see a letter from us with a recommendation?

CHAIRMAN SELIN: I would expect that whatever views those members of the ACRS --

DR. KRESS: Would be transferred to the panel?

CHAIRMAN SELIN: -- would be transferred to the panel. We really have to stick to the panel procedures.

There's nothing wrong with individual recommendations to the panel, recommendations as any public citizens could make, but, if I got a letter on one of these candidates, I would just turn it over to the panel and say, "Please take that into account." I wouldn't independently act on a letter, so --

DR. KRESS: Good. That's helpful. It clarifies our -- what we should be doing.

COMMISSIONER de PLANQUE: Since this is a new

process, I think any feedback that you have on how it progresses would be useful to us. It's too early in the --

DR. KRESS: It's maybe a little early now, but I think we are already developing some feedback thoughts and we would be pleased to give them.

MR. CARROLL: Very cumbersome.

COMMISSIONER ROGERS: I think it's feed forward process, feed forward.

CHAIRMAN SELIN: Feet?

COMMISSIONER ROGERS: Feed forward, rather than feedback.

CHAIRMAN SELIN: Oh, I see.

MR. LINDBLAD: Or best foot forward.

CHAIRMAN SELIN: Okay. Sorry.

DR. KRESS: We have no additional items, unless you have something you'd like to --

CHAIRMAN SELIN: I don't have an item in the sense of a specific question. I would just like to repeat the general admonition that the most useful thing that the Committee can do, at least for me as an individual Commissioner, is to sit back and look for places where you're not just saying did the staff do this piece right or that. It's sort of the analogy of Doctor Catton's, the top-down, you know, a major process, an important issue where it's not that we're doing some of the pieces wrong but we're

leaving a piece out entirely.

The canonical question I always put to you is, is there someplace where even if we answer all the questions we've asked we haven't answered all the questions that are necessary to decide what to do. And the huge error of omission is much more worrisome to me than a mistake, an error of commission along the way, because, if somebody has asked the right questions, sooner or later we'll figure out if they got the right answer or not. But the one thing that doesn't present itself is just a complete omission where we've just missed some whole topic that we should be asking, either a subject that we're not looking at or we're looking at a subject where we're leaving out a major piece.

The one that occurs to me, for instance, and this isn't really a request that you look at it, as we get down to $10^{-\text{Xs where Xs are getting to be large numbers}}$, the probability that somebody just left the containment open or, you know, put in something backwards seems to start to overwhelm all the probabilities that we can calculate. And I really get worried when we get into a question about is the calculation less than the safety goal and therefore we can quit or not. I mean, that's the whole philosophy behind severe accidents, behind blunders, behind defense in depth.

But, if we're going to be serious about performance work, we need to have some kind of a bounding

analysis based on blunders as opposed to a fine calculation based on errors. Using the old difference, an error is something that's statistically inevitable. A blunder is what really happens in real life. And I have no idea about how to go about doing that and that's an issue where if you had some advice as to, you know, sort of the level zero human factor issue -- there probably are others like that as well, so please don't -- just because we haven't been smart enough to ask you that, please don't be shy about looking for issues of that type as opposed to we have this research program or this program and are we doing it quite right or not.

DR. KRESS: As a matter of fact, that is the reason I restarted our strategic planning. It's that sort of thing we're thinking about in there and we're sort of glad to have that in there. It does look like an interesting item.

CHAIRMAN SELIN: Well, that particular question is the one that I worry about more than any other question along the way. As we go towards performance-based, how do we worry about truly incompetent performance, not just stuff that's a little bit off? How do we feed that into what we're doing?

Commissioner?

COMMISSIONER de PLANQUE: Well, just along the

lines of those issues, I was very interested to hear what you said about risk-based regulation and performance-based. And if you are not aware of it, you maybe would like to be aware of some of the other activities that are going on interagency-wide.

I just attended a meeting this morning on a subgroup on risk as part of the regulatory group and this is the direction in which they're going. There is an interagency effort to formulate a policy, and you may have seen it in -- it was leaked to Inside EPA and it's in that document if you want to look at it. It's just a working draft that's being discussed among the agency representatives, but it's clearly moving in that direction.

And if you also look at some of the legislation that's being introduced in Congress right now, there's a lot of interest in risk-based regulation and comparative risk and trying to get some coherence in the big picture, not alone just within one small field. So, I think it's a subject whose time is well overdue and I'm glad to hear that you're doing a lot of discussing on that issue.

DR. KRESS: Thank you. We did receive your package of information on the interagency group and I've passed it out to the Committee.

COMMISSIONER de PLANQUE: Okay.

DR. KRESS: We haven't had a chance to look at it

as a committee yet, but we do want to thank you for that.

COMMISSIONER de PLANQUE: Okay.

CHAIRMAN SELIN: My view is that if somebody really took risk-based regulation seriously and looked at the risk of a nuclear accident compared to a lot of other risks we would be reduced to about an office within EPA.

DR. KRESS: We don't want to go too far with this, right.

DR. CATTON: I think David Okrit came to that conclusion years ago.

CHAIRMAN SELIN: Yes, I mean, the straight calculation. The straight calculation. But on a serious note, I really do think there's too much of a chance of a blunder and doing these calculations and comparing them with the risk of driving a car or -- well, smoking is a part, but, just, you know, things that aren't completely under your control would lead to conclusions that are counter-intuitive, and so I have to worry a little bit about the calculations. I do think it's appropriate that there be a high level organization that really is concerned with nuclear power plant safety and a number of other serious risks.

Commissioner Rogers?

COMMISSIONER ROGERS: No. I think it's been an excellent presentation. Appreciate it very much.

CHAIRMAN SELIN: I am amazed that you got through this program in that schedule. I didn't think there would be --

DR. KRESS: I'm a tough task master.

DR. CATTON: In spite of my going over my allocated time.

MR. CARROLL: We spent 15 minutes before we came over here beating Ivan --

CHAIRMAN SELIN: He did very well. Thank you very much.

DR. KRESS: Thank you.

[Whereupon, at 2:43 p.m., the above-entitled matter was adjourned.]

CERTIFICATE

This is to certify that the attached description of a meeting of the U.S. Nuclear Regulatory Commission entitled:

TITLE OF MEETING: PERIODIC MEETING WITH THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - PUBLIC MEETING

PLACE OF MEETING: Rockville, Maryland

DATE OF MEETING: Thursday, September 8, 1994

was held as herein appears, is a true and accurate record of the meeting, and that this is the original transcript thereof taken stenographically by me, thereafter reduced to typewriting by me or under the direction of the court reporting company

Transcriber: Carol Lynch

Reporter: Peter Lynch

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 1, 1994

MEMORANDUM TO: John C. Hoyle, Secretary of the Commission

FROM: John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards

SUBJECT: ACRS MEETING WITH THE NRC COMMISSIONERS ON SEPTEMBER 8, 1994 - SCHEDULE/BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners on Thursday, September 8, 1994, between 1:30 and 3:00 P.M. at the Commission hearing room at OWFN, to discuss items of mutual interest, including the following. Background materials related to these items are attached.

1.	Introduction (NRC Chairman)	1:30 - 1:35 P.M.
2.	Status of the ACRS Review of the Passive Plant Designs (T. Kress) (pp. 1-14) a. Westinghouse AP 600 Design/ Test Programs (W. Lindblad/ I. Catton) b. General Electric Nuclear Energy SBWR Design/Test Programs (J. Carroll/I. Catton)	1:35 - 1:45 P.M.
3.	Lessons Learned from the ACRS Review of the Evolutionary Plant Designs (W. Lindblad) (pp. 15-19)	1:45 - 1:55 P.M.
4.	Protective Action Guidelines (T. Kress) (pp. 20-26)	1:55 - 2:05 P.M.
5.	Thermo-Lag Fire Barriers (I. Catton) (pp. 27-40)	2:05 - 2:15 P.M.
6.	National Academy of Sciences/ National Research Council Study & Workshop on Digital I&C Systems (W. Shack) (pp. 41-45)	2:15 - 2:25 P.M.

7.	Steam Generator Tube Repair Criteria (W. Shack) (pp. 46-49)	2:25 - 2:35 P.M.
8.	Need for Review of Rationale for Regulation (T. Kress) (pp. 50-53)	2:35 - 2:45 P.M.
9.	Selection of New ACRS Members (T. Kress) (pp. 54-57)	2:45 - 2:55 P.M.
10.	Additional Items (e.g., Future of the NRC-spondored research programs at the DOE Labs.) (if time permits) Closing Remarks	2:55 - 3:00 P.M.

Attachments:
As Stated

cc: ACRS Members
ACRS Technical Staff

ITEM(2): STATUS OF THE ACRS REVIEW OF THE PASSIVE PLANT DESIGNS

Based on the present schedule, the NRC staff will provide the ACRS a draft safety evaluation report (DSER) on the Westinghouse AP600 design in November 1994. The ACRS will begin subcommittee reviews after the receipt of the DSER. The final safety evaluation is scheduled to be provided to the ACRS in May 1996.

Recently, GENE requested the NRC staff to place the NRC review of the design-related portion of the SSAR for the SBWR design in series with completing the test programs, delaying resumption of the staff's review until early 1996. The ACRS will begin its review of the SBWR design after receipt of the DSER.

a) Westinghouse AP600 Test Programs

The Committee and the Subcommittee on Thermal Hydraulic Phenomena have heard presentations, during several meetings, regarding the Westinghouse AP600 passive plant test programs proposed by both Westinghouse and the staff in support of the AP600 design certification. The Committee provided reports dated November 14, 1991 and March 10, April 6, and July 17, 1992 to the Commission in regard to the test programs.

The Committee also reviewed, pursuant to an SRM on this matter, selected portions of the NRC AP600 confirmatory test program being conducted at the Japanese ROSA-V test facility. A meeting of the Thermal Hydraulic Phenomena Subcommittee was held on August 25, 1994, to review the status of the ROSA-V test program. The Committee had previously provided a report dated November 18, 1993, on this matter. The Committee is also continuing to review the staff's companion effort to modify the RELAP5 code for analysis of the AP600 design.

The Committee has been continuing its review of the status of the Westinghouse analytical and experimental programs noted above. Meetings of the Thermal Hydraulic Phenomena Subcommittee were held on March 15-16, and May 18-19, 1994 to continue the review of this matter.

Attachments:

- ACRS report to the Commission dated November 18, 1993, Subject: NRC Confirmatory Test Program in Support of the AP600 Design Certification (pp. 3-5)

- ACRS report to the Commission dated July 17, 1992, Subject: Integral System and Separate Effects Testing in Support of the Westinghouse AP600 Plant Design Certification (pp. 6-10)

b) <u>General Electric Nuclear Energy SBWR Test Programs</u>

The Committee and the Subcommittees on Advanced Boiling Water Reactors/Thermal Hydraulic Phenomena have heard presentations regarding design details and test programs for the General Electric SBWR passive plant. The Committee provided a report to the Commission dated June 10, 1992, regarding the proposed test programs in support of the SBWR design certification. The Committee will continue its review of the ongoing experimental and analytical programs related to the certification of the SBWR design.

Recently, GE Nuclear Energy (GENE) conducted a reassessment of its Test and Analysis Program (TAP) being conducted in support of the design certification. In concert with same, GENE requested that NRR suspend its review of the SSAR, pending the substantial completion of the revised TAP. GENE has requested that the NRC provide, in the near future, formal assurance that the revised TAP will be acceptable, vis-a-vis ensuring compliance with the relevant strictures of 10 CFR Part 52.47.

The Thermal Hydraulic Phenomena Subcommittee held a meeting on August 24, 1994 to discuss the above matters. Based on mutual agreement with the NRR staff, the Committee expects to review the acceptability of the revised TAP later this year. This review will proceed in conjunction with the formulation of the NRC staff's response to GENE concerning same.

A meeting of the Thermal Hydraulic Phenomena Subcommittee was also held on August 26, 1994, to review the status of the NRC Office of Nuclear Regulatory Research confirmatory test programs being conducted in support of the SBWR design certification.

The Standard Safety Analysis Report for the SBWR was received on August 26, 1992. The Committee will continue its discussion of this matter on a schedule consistent with the development of the staff's SER.

<u>Attachment</u>:

- ACRS report to the Commission dated June 10, 1992, Subject: Testing and Analysis Programs in Support of the Simplified Boiling Water Reactor Design Certification (pp. 11-14)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 18, 1993

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: NRC CONFIRMATORY TEST PROGRAM IN SUPPORT OF THE AP600 DESIGN CERTIFICATION

During the 403rd meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1993, we reviewed selected aspects of the NRC Office of Nuclear Regulatory Research (RES) experimental program to be conducted at the Japan Atomic Energy Research Institute's (JAERI's) Large-Scale Test Facility (LSTF) in support of the NRC design certification of the Westinghouse (W) AP600 passive plant. Our Subcommittee on Thermal Hydraulic Phenomena met on October 28, 1993, to review this matter. During this review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

In a September 16, 1992 Staff Requirements Memorandum, the Commission requested that the ACRS review selected aspects of the ROSA-V test program prior to its initiation. Specifically, the Committee was asked to review the test matrix and the facility modifications and additions, including instrumentation and controls. The following comments are offered in response to that request:

- The modified LSTF has been designated as ROSA-V. Despite the modifications, a number of atypicalities and scaling distortions exist in the ROSA-V configuration relative to the AP600 design. Some of the atypicalities in ROSA-V are: the use of one cold-leg per reactor coolant system (RCS) loop instead of two; the geometry and heat transfer characteristics of the steam generators; the existence of a four foot loop seal in the RCS; excess metal mass (in particular, for the core makeup tank (CMT)); the volume and geometry of the in-containment refueling water storage tank (IRWST); the primary residual heat removal (PRHR) system; and the configuration of the pressurizer surge line. RES staff representatives stated that they understand the impact these atypicalities will have on system performance. The RES staff has not, however,

presented a convincing argument that it understood the impact. RES should do so and document the results.

- Despite the facility shortcomings, we believe that ROSA-V will generate useful data to support validation of the relevant computer codes. This validation, however, may be inconclusive given the above atypicalities, especially those existing in the CMT, the PRHR system, and the IRWST. We recommend that the staff be urged to resolve the issues resulting from the atypicalities discussed above by additional analyses and, if necessary, by separate effects tests.

- The instrumentation proposed in support of the planned test program appears adequate for code assessment when dealing with single-phase phenomena. It is not clear that it is adequate for the measurement of key phenomena under conditions of two-phase flow. It is inadequate for determining some of the heat transfer characteristics of the PRHR system.

- The AP600 automatic depressurization system (ADS) will be activated by decreasing water level in the CMT. This level will be measured with heated junction thermocouples (HJTCs). The three AP600 integral system test facilities (ROSA-V, APEX—Advanced Plant Experiment—and SPES-II) will use differential pressure (DP) cells to measure this level. Activation of the ADS using DP cells rather than HJTCs could result in significant test distortions, given the inherent time delay associated with the use of HJTCs. The RES staff believes that these differences can be addressed. We were told by RES that JAERI has installed HJTCs of its own design at ROSA-V. We recommend that the RES staff use these HJTCs for ADS control for at least one properly chosen test, even if they are of a different design from those planned for use on the AP600.

- The ROSA-V test matrix is based on examination of transients and design-basis accidents for existing PWR designs. A number of the tests in the ROSA-V Phase I matrix have counterparts in the test matrices of the W SPES II and APEX facilities. These three facilities are scaled differently and have atypicalities of differing natures. We believe that the data obtained from these facilities will prove adequate for the necessary computer code validation by providing a broad range of challenges for simulation, given that the separate effects test programs supply sufficient information for code model development.

- Recently, RES modified the Phase I test matrix in response to a request from NRR to include some very small breaks and some "beyond-DBA" type events. We support this modification, but note that the capability of the relevant computer codes to

model very small-break LOCAs is weak. This may lead to difficulties when code validation is attempted.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

References:

1. U.S. NRC Report, NUREG/CR-6066 (Draft), "Analysis of LSTF Scaling for AP600 Testing," M. Ortiz, et al., June 11, 1993 (Draft Predecisional)
2. Memorandum dated December 23, 1992, from G. Rhee, NRC, to P. Boehnert, ACRS, transmitting INEL Report by T. Boucher, et al., "Description of Design Requirements for ROSA Modifications to Simulate AP600 Phenomena" (Revised September 1992)
3. U.S. NRC Report, NUREG/CR-5853, "Investigation of the Applicability and Limitations of the ROSA Large-Scale Test Facility for AP600 Safety Assessment," M. G. Ortiz, et al., December 1992
4. ACRS report dated July 17, 1993, "Integral System and Separate Effects Testing in Support of the Westinghouse AP600 Plant Design Certification"
5. Staff Requirements Memorandum dated September 16, 1992, from S. J. Chilk, Office of the Secretary, to J. M. Taylor, EDO, "SECY-92-219 - NRC-Sponsored Confirmatory Testing of the Westinghouse AP600 Design"
6. SECY-92-219, Memorandum dated June 16, 1992, from J. M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: NRC-Sponsored Confirmatory Testing of the Westinghouse AP600 Design

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 17, 1992

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: INTEGRAL SYSTEM AND SEPARATE EFFECTS TESTING IN SUPPORT OF THE WESTINGHOUSE AP600 PLANT DESIGN CERTIFICATION

During the 387th meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 1992, we discussed the programs of integral system and separate effects testing being planned by both Westinghouse and NRC to support the certification effort for the Westinghouse Electric Corporation's AP600 passive plant design. We held discussions on this matter during our 381st through 384th (January-April 1992) meetings, inclusive. Our Subcommittee on Thermal Hydraulic Phenomena held meetings on December 17, 1991, March 3, 1992, and June 23-24, 1992 to review this issue. During these meetings, we had the benefit of discussions with representatives of the Westinghouse Electric Corporation and the NRC staff. We also had benefit of the referenced documents. We have previously reported to you on this matter in our letters of March 10 and April 6, 1992.

BACKGROUND

Appropriately validated thermal hydraulic computer models must be relied on to support the safety assessments required for certification of the AP600. Westinghouse has indicated that it plans to use its more mechanistic assessment code, WCOBRA/TRAC, only for large-break LOCA analyses, and will rely on its evaluation model, NOTRUMP, for analyses of all other design-basis events. The NRC plans to use RELAP5/MOD3 to support its assessments.

The NOTRUMP code is an evaluation model code that is based on 10 CFR Part 50, Appendix K, requirements. The other two codes, WCOBRA/TRAC and RELAP5/MOD3, are more mechanistic codes that have been qualified as best-estimate tools only for large-break LOCAs. All of these analysis tools will be required to simulate the AP600 behavior in regimes where the codes are known to be weak. These regimes include phenomena such as horizontal (perhaps countercurrent stratified) flows, interface movements, thermal

stratification, rapid "shock" condensation, boron mixing, and low-pressure gravity-driven flows.

To develop the necessary data for improvement and validation of these models for AP600 assessment, Westinghouse now has plans for conducting a number of separate effects tests at several different facilities, and integral system tests. The integral system test programs are to be conducted in a low-pressure facility now nearing final design at the Oregon State University (OSU) and in an existing high-pressure facility, SPES (in Italy), to be modified to better simulate AP600.

The NRC has proposed to conduct high-pressure confirmatory testing by modifying and using the existing ROSA-IV facility at JAERI in Japan. The modified facility will be referred to as ROSA-V. The NRC has no specific plans for additional separate effects testing. The staff does plan to conduct low-pressure integral system testing in the OSU facility after the Westinghouse program has been completed.

At this time, we have the following comments and recommendations regarding various aspects of these planned and proposed efforts.

<u>WESTINGHOUSE PROGRAM</u>

We believe that, with certain enhancements, the Westinghouse program will be adequate for the certification process. We have the following specific comments and recommendations:

- We are concerned that Westinghouse plans to rely primarily on its NOTRUMP evaluation model (EM) code. It is a step backwards to use computer codes of only EM sophistication and capabilities to evaluate the thermal hydraulic behavior of new nuclear power plants.

- The Westinghouse separate effects tests of most importance to the certification of AP600 are the Core Make-up Tank (CMT) tests and the Automatic Depressurization System (ADS) tests. The test matrices for these do not cover ranges of conditions that are broad enough to yield an adequate data base for the required model development. We recommend that pressure disturbances of the types that would be caused by either ADS valve actuation or by rapid steam condensation when cold CMT fluid is injected into the downcomer region be part of the test program.

- An additional separate effects test facility is needed to investigate the asymmetric effects associated with the downcomer and with the cold-side plenum of the steam generator.

- SPES is generally a good choice for conducting full-height, full-pressure integral system tests. However, in addition to the scaling problems associated with a high ratio of surface area to fluid volume that plague small-scale simulations of this kind (and must be dealt with), the proposed modified version, SPES-II, has two important scaling defects that should be eliminated: (a) the aspect ratio (height to diameter) of the simulated pressurizer is different from that of the AP600 and (b) the cold leg configuration is not geometrically similar to that of AP600.

  We recommend that Westinghouse be required to preserve the scaling of the pressurizer and the geometrical configuration of the cold legs, to better simulate AP600 behavior (this would include simulation of a reactor coolant pump in each leg).

- The method proposed for simulating steam generator tube ruptures in SPES-II is flawed in that it does not appear to allow the break flow from the primary system to be from both the hot and cold sides of the tube. We recommend that Westinghouse develop a better simulation method.

- The OSU low-pressure integral system testing facility is well conceived. We commend Westinghouse for its efforts with respect to this facility. Our evaluation of the scaling rationale for the facility design (discussed during the subcommittee meeting of June 23-24, 1992) is that it is soundly based. Further, the 400 psia design capability should allow considerable simulation of high-pressure effects, while providing the more important low-pressure behavior.

NRC PROGRAM

Our understanding of the justification provided by the NRC staff for its proposed confirmatory high-pressure integral system testing in the ROSA-V facility is as follows:

- Because ROSA-V is considerably larger than SPES-II, such confirmatory testing would provide an additional check on the adequacy of the scaling capabilities of the codes, and would help confirm that important effects have not been overlooked.

- The confirmatory test program would provide the opportunity to maintain the staff's thermal hydraulic expertise and up-to-date knowledge in this field.

While we agree that the above considerations have some merit, we have not been persuaded that confirmatory high-pressure testing by the staff is needed before the AP600 design certification and, even if this were the case, we have significant reservations about the

adequacy of the ROSA-V facility for this purpose. These positions are based on the following observations:

- The NRC staff has not presented convincing arguments supporting its needs for confirmatory testing, particularly at high pressures.

- The SPES-II facility appears to be sufficient to meet all the high-pressure integral system testing needs. The NRC will be able to use the SPES-II facility for its confirmatory testing needs just as it plans to use the OSU facility.

- The desired staff experience will come from pre-test and post-test evaluations of the various tests using the RELAP5/MOD3 code. This experience can just as easily be obtained by evaluating the SPES-II and OSU tests and results.

- The ROSA-V facility contains several atypicalities that will manifest themselves in difficult-to-explain behavior relative to that expected for AP600 (the sensitivity of the ROSA-V thermal hydraulic behavior is well documented in the INEL report, NUREG/CR-5853).

- The tests would be in a distant location. There would be a very limited number of tests, because of the expense involved. In addition, we are concerned that the adequacy of instrumentation (for example) might have to be compromised in order to reduce overall program costs.

For the above reasons, we believe that NRC resources would be better used by focusing on three areas: (a) possible additional separate effects testing to support the modeling needs for RELAP5/MOD3, (b) participation in the pre-test and post-test analyses efforts associated with the SPES-II and the OSU test programs, and (c) consideration of utilizing the SPES-II facility for high-pressure confirmatory testing needs in the same way the staff plans to use the OSU facility for its confirmatory low-pressure testing needs.

To accomplish the above objectives, we believe that the staff should consider the establishment of a task force of experts in related fields to assist it in the development of the analytical and experimental programs necessary for timely certification of the AP600 passive plant design.

Sincerely,

Paul Shewmon

Paul Shewmon
Acting Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG/CR-5853, "Investigation of the Applicability and Limitations of the ROSA-IV Large Scale Test Facility for AP600 Safety Assessment (Draft)," dated May 1992
2. T. Boucher, Idaho National Engineering Laboratory, et al., "Scaling Issues for a Thermal-Hydraulic Integral Test Facility," Paper transmitted via a memorandum from L. Shotkin, NRC-RES, for P. Boehnert, ACRS, dated June 29, 1992
3. Oregon State University Report, OSU-NE-9204 (Draft), "Scaling Analysis for the OSU AP600 Integral System and Long Term Cooling Test Facility," J. Reyes, Jr., dated June 1992 (W **Proprietary Report)**
4. Letter dated January 22, 1992, from G. Saporano, ENEA, Italy, to E. S. Beckjord, NRC, transmitting documentation on SPES test facility
5. Memorandum dated June 13, 1991 from S. Modro, INEL, for L. Shotkin, NRC-RES, transmitting draft report, "Evaluation of Scaled Integral Test Facility Concepts for the AP600" by Modro, et al.
6. U.S. Nuclear Regulatory Commission, SECY-92-219, "NRC-Sponsored Confirmatory Testing of the Westinghouse AP600 Design," dated June 16, 1992 **(Predecisional)**
7. U.S. Nuclear Regulatory Commission, SECY-92-219A, "Addendum to SECY-92-219 - Providing Additional Information to Justify Sole Source Procurement," dated July 9, 1992 **(Predecisional)**
8. Memorandum dated April 21, 1992, from S. Chilk, Secretary, for J. M. Taylor, EDO, and W. Parler, General Counsel, Subject: SECY-92-037 - Need for NRC-Sponsored Confirmatory Integral System Testing of the Westinghouse AP600 Design
9. Westinghouse Topical Report, WCAP-13277, "Scaling, Design and Verification of the SPES-2, the Italian Experimental Facility Simulator of the AP600 Plant," dated April 1992 **(W Proprietary Report)**

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 10, 1992

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: TESTING AND ANALYSIS PROGRAMS IN SUPPORT OF THE SIMPLIFIED BOILING WATER REACTOR DESIGN CERTIFICATION

During the 385th and 386th meetings of the Advisory Committee on Reactor Safeguards, May 6-9 and June 4-5, 1992, we reviewed the testing and analysis programs in progress and proposed by GE Nuclear Energy (GE) in support of the certification effort for the Simplified Boiling Water Reactor (SBWR) passive plant design. Our Subcommittee on Thermal Hydraulic Phenomena held meetings to discuss this topic on April 23 and June 2, 1992. During these meetings, we had the benefit of discussions with representatives of GE and the NRC staff. We also had the benefit of the documents referenced.

GE will use its best-estimate code, TRACG, to evaluate the SBWR thermal hydraulic behavior under accident conditions ranging from ATWS with instabilities to long-term behavior of the Passive Containment Cooling System (PCCS). GE representatives presented a very good analysis of processes and phenomena important to accident scenarios postulated for the SBWR. The results were summarized in tables which are to be used by GE to validate the TRACG computer code. However, these same tables appear not to have been used to guide the design and operation of the experimental facilities that are to support the code validation process.

The GE experimental program consists of three elements:

1) Laboratory scale experiments to obtain fundamental heat transfer data,

2) Separate effects tests to obtain data for parts of the total system and full-scale components where necessary, and

3) Integral system tests to obtain system data.

Although we were shown some comparisons of TRACG predictions with data from GE's integral system tests (GIST and GIRAFFE facilities), the question of whether or not the facilities can scale the important phenomena was not addressed in either GE's presentation or in the documents supplied to the ACRS by GE. A rigorous scaling analysis is needed if integral system test data alone are to be used to demonstrate that a TRACG calculation is meaningful.

We have some comments about the elements of the GE test plan. The initial conditions for the integral system tests are based on conditions assumed to exist some time after vessel depressurization. These conditions include an initial drywell and PCCS nitrogen mass fraction of 15 percent. The nitrogen concentration could be much higher. GE should develop a basis for its choices of initial conditions or broaden its test matrix to include some tests at much higher values of the nitrogen concentration, both in the drywell and in the PCCS.

Separate effects tests to be conducted in the PANTHERS facility will yield the data needed to characterize heat exchanger behavior under a variety of expected conditions. In particular, GE has agreed to add instrumentation to the individual heat exchanger tubes to obtain local heat transfer data. This will make the GIRAFFE integral system experiments more meaningful. We believe GE has been very responsive to issues raised by both the ACRS and the NRC staff in this regard.

The oscillatory behavior observed in the GIRAFFE integral system tests needs more detailed study to ensure that the suppression pool does not overheat due to steam bypass of the PCCS through the suppression pool top horizontal vents. The steam flow rate will be low which could lead to a stratified condition. The suppression pool is not a very effective heat sink when this process occurs. This may well require a separate effects study to obtain data for development of a low steam flow model for the horizontal vent. Further, review of the GIRAFFE facility instrumentation is needed to ensure that the resulting data will support TRACG model validation.

The SBWR has full pressure isolation condensers (IC) capable of removing 4.5 percent of full power decay heat at full system pressure. The behavior of isolation condensers is well understood and introduces no new processes. GE has indicated that it will collect relevant IC operating data for staff review. The SBWR is automatically depressurized when the vessel water level drops to some prescribed value by a staged opening of squib-type valves. Further, GE has had a great deal of experience with automatic depressurization and only the squib-type valve itself is of a new design. As a result, we do not believe that full-height, full-

pressure integral system testing is required for certification of the SBWR design.

The GE program includes conduct of integral system testing at the PANDA facility located in Switzerland. The NRC staff would like GE to obtain data from this facility in time to support its design certification review of the SBWR. To do so, GE would have to accelerate its schedule by six months. We agree with the NRC staff that further integral system testing of the PCCS is needed prior to the final design approval. It has not been demonstrated by GE that existing data obtained from GIRAFFE or GIST testing are sufficient for validation of the TRACG code, nor that the PANDA test facility will yield the needed data. A more definitive assessment by GE is needed; this assessment should include both the scaling rationale for the GIRAFFE, GIST, and PANDA facilities, and a demonstration of how the effects of test facility scaling distortion impact the important processes and phenomena outlined by GE in its evaluation of TRACG. As a part of such an effort, it may be possible to show that one can obtain the needed data by some combination of additional separate effects tests and judicious use of the GIRAFFE and GIST facilities.

To summarize, we agree with the NRC staff views that full-height, full-pressure integral system testing is not needed to support the SBWR design certification. Further, we agree that early integral system testing of the PCCS is essential to meet the present design certification schedule. We have not, however, seen evidence that the PANDA facility is adequate to obtain the needed data.

Sincerely,

David A. Ward
Chairman

<u>References</u>:

1. Memorandum dated February 26, 1992, for the Commissioners from James M. Taylor, Executive Director for Operations, transmitting Advance Copy of proposed Commission paper, "Evaluation of the General Electric Company's (GE's) Test Program to Support Design Certification for the Simplified Boiling Water Reactor (SBWR)"

2. Letter dated February 3, 1992, from R. C. Mitchell, GE Nuclear Energy, to U.S. Nuclear Regulatory Commission, Subject: GE Response to Request for Information on SBWR Testing Program

3. Joint Study Report, "Feature Technology of Simplified BWR (Phase I) GIRAFFE (Final Report)," dated November 1990, The Japan Atomic Power Company, et al. **(GE Proprietary Information)**

4. GE Nuclear Energy, GEFR-00850, "Simplified Boiling Water Reactor (SBWR) Program Gravity-Driven Cooling System (GDCS) Integrated Systems Test - Final Report," A.F. Billig, dated October 1989 **(Applied Technology Restriction)**

5. "ALPHA – The Long Term Passive Decay Heat Removal and Aerosol Retention Program at the Paul Scherrer Institute, Switzerland," by P. Coddington, et al., Paul Scherrer Institute, undated

6. Paper from the Proceedings of The International Conference on Multiphase Flows '91 – Tsukuba, Japan, September 24-27, "Condensation in a Natural Circulation Loop with Noncondensable Gases Part 1 – Heat Transfer," K. M. Vierow, GE Nuclear Energy, and V. Schrock, University of California

7. GE Draft Report: "Test Specification for IC & PCC Tests," undated **(GE Proprietary Information)**

8. Paper submitted to the Department of Energy, "The Effect of Noncondensable Gases on Steam Condensation Under Forced Convection Conditions," M. Siddique, Ph.D. Thesis - Massachusetts Institute of Technology, dated January 1992

ITEM(3): LESSONS LEARNED FROM THE ACRS REVIEW OF THE EVOLUTIONARY PLANT DESIGNS

As a result of the ACRS review of the ABWR and System 80+ designs, the Committee identified some potential areas for staff action for operating nuclear power plants and the review of future plant designs.

The following are some issues the ACRS believes the staff should address as generic issues, as technical specification improvement program issues, as revisions to the standard review plan, or as additional research needs;

- Turbine inspection requirements
- Technical specification requirements for onsite power sources
- Reactor water cleanup system safety
- Review of Chilled-water systems
- Filters or water separators for the hardened vents installed on operating BWR containments
- Fuel-coolant interactions
- Adequacy and use of PRA

The ACRS issued a letter to the EDO on July 13, 1994, regarding the above-mentioned issues

Attachment:

- ACRS letter to Mr. James M. Taylor, EDO, dated July 13, 1994. Subject: Some areas for potential staff considerations for operating nuclear power plants and the review of future plant designs resulting from the ACRS review of the evolutionary light water reactors (pp. 16-19)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 13, 1994

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: SOME AREAS FOR POTENTIAL STAFF CONSIDERATION FOR OPERATING NUCLEAR POWER PLANTS AND THE REVIEW OF FUTURE PLANT DESIGNS RESULTING FROM THE ACRS REVIEW OF THE EVOLUTIONARY LIGHT WATER REACTORS

During the 411th meeting of the Advisory Committee on Reactor Safeguards, July 7-8, 1994, we completed our discussion related to the results of our recent reviews of the General Electric Nuclear Energy (GENE) Advanced Boiling Water Reactor (ABWR) (Reference 1) and the ASEA Brown-Boveri Combustion Engineering (ABB-CE) System 80+ (Reference 2) applications for design certification from the perspective of potential areas for staff action for operating nuclear power plants and the review of future plant designs. These reviews provided us with an opportunity to consider present regulatory practices and procedures vis-a-vis the "state-of-the-art" design requirements for these evolutionary light water reactors (ELWRs).

The following are some issues that we believe the staff should address as Generic Issues, as Technical Specification Improvement Program issues, as revisions to the Standard Review Plan, or as additional research needs.

1. <u>Turbine Inspection Requirements</u> - In the course of reviewing the potential for turbine rotor failure related to the ABWR and System 80+ designs, we learned that the staff has not prepared an appropriate set of preoperational and inservice inspection, evaluation and acceptance requirements for turbine rotor, other than those employing shrunk-on disks.

   Some current licensees have replaced, or are planning to replace, shrunk-on disk rotors with rotors of a different design. We believe that the staff should develop appropriate positions for the various designs on a priority basis.

2. Technical Specification Requirements for Onsite Power Sources - In our letter to you dated February 17, 1994, concerning three issues relating to the 10 CFR Part 52 design certification process for ALWRs, we recommended that the staff resolve the matter of credit for ELWR alternate AC sources when 1E emergency diesel generators are out of service during power operation. We suggested that Technical Specification requirements for such onsite power sources be based on appropriate probabilistic considerations. Subsequently, ABB-CE requested such credit for System 80+ and the staff has granted an allowable outage time for a 1E emergency diesel generator of up to 14 days when the combustion turbine-generator is available. We now recommend that the staff expand this concept to include operating nuclear power plants.

   It is our understanding that Technical Specification requirements for onsite power sources will be incorporated into the Shutdown and Low Power Operations Rule.

3. Reactor Water Cleanup System Safety - The Reactor Water Cleanup (RWCU) System is of safety concern for boiling water reactor plants because it is a high-energy, non-safety system, portions of which may be located inside of the secondary containment. The secondary containment also houses numerous engineered safety features and the Fuel Pool Cooling System. For operating plants, the RWCU System supply line from the reactor vessel is usually a 6-inch pipe. A rupture of this pipe inside of the secondary containment results in a loss of reactor coolant which may create a serious environmental disruption throughout the secondary containment before it can be isolated.

   An ACRS staff report (Reference 3) identified a number of safety-related deficiencies in a similar system for the ABWR. Subsequently, GENE developed a requirement for environmental qualification of all safety-related components and the Fuel Pool Cooling System inside of the secondary containment. The qualification was based mostly on the adverse atmosphere created before complete closure of the isolation valves following a supply line pipe break. Generally, operating plants do not provide a comparable level of environmental qualification.

   Another GENE change was the addition of a second isolation valve in the supply line inside of the primary containment. This valve isolates the reactor vessel from the supply line pipe break in the event that isolation is not achieved by

closing the two primary containment isolation valves under blowdown flow conditions. The added valve is not capable of blowdown isolation. It is closed by manual actuation after the blowdown is completed, thereby achieving reactor vessel isolation and interruption of any prolonged release of Emergency Core Cooling System (ECCS) water to the break which is outside of primary containment. Operating plants may not have a similar capability. We recommend that this issue be investigated for operating BWRs.

4. Review of Chilled-Water Systems - A number of operating plants use large Chilled-Water Systems to provide essential environmental cooling. Because there is no Standard Review Plan (SRP) for these systems, the staff has used other guidance such as SRP 9.2.2 (Reactor Auxiliary Cooling Water Systems) when evaluating the safety of such systems. However, this guidance is not appropriate for the evaluation of refrigeration systems.

   In determining plant safety, the NRC staff needs to evaluate the performance of Chilled-Water Systems under various accident heat loads and during loss-of-offsite-power events, and to consider the ability of such systems to restart and function after tripping or after a prolonged station blackout. We urge that the staff develop better guidance and positions with which to enhance the scope and quality of its plant reviews of Chilled-Water Systems.

5. Filters or Water Separators for the Hardened Vents Installed on Operating BWR Containments - A great deal of analysis was done to demonstrate that the ABWR Containment Overpressure Protection System is adequate without filters or water separators. We are not aware that such an analysis has been done for those operating BWRs with hardened vents. We believe their need for filters or water separators should be reevaluated.

6. Fuel-Coolant Interactions - We are concerned that the safety case with respect to fuel-coolant interactions is based mostly on arguments of low probability of occurrence. It concerns us that neither the industry nor the NRC staff is able to predict limits to the energetics (below purely thermodynamic limits) based on either first principles or sufficient empirical evidence. We believe additional research is needed on this issue.

7. Adequacy and Use of PRA - We are concerned that there are no clear regulatory criteria for what constitutes an acceptable PRA. By accepting the PRAs which have already been submitted, the staff is essentially establishing the regulatory criteria by precedent rather than by promulgating specific requirements. We believe consideration should be given to establishing minimum requirements for PRAs.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. ACRS Report dated April 14, 1994, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Report on Safety Aspects of the General Electric Nuclear Energy Application for Certification of the Advanced Boiling Water Reactor Design
2. ACRS Report dated May 11, 1994, from T. S. Kress, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Report on the Safety Aspects of the ASEA Brown Boveri-Combustion Engineering Application for Certification of the System 80+ Standard Plant Design
3. Advisory Committee on Reactor Safeguards Report by S. E. Mays and M. E. Stella, "ABWR Reactor Water Cleanup System Review," July 30, 1992

ITEM(4): PROTECTIVE ACTION GUIDELINES

At the March 10, 1994 meeting with the NRC Commissioners, the ACRS agreed to consider the implications of the results reported in the ABB-CE System 80+ standard safety analysis report that the calculated doses for the design basis accidents (DBAs), using the new source terms and a hypothetical site, were less than protective action guidelines (PAGs) at the site boundary.

During the June 9-10, 1994 ACRS meeting, the NRC staff briefed the Committee regarding the use of PAGs in emergency planning. In addition, the Commission in the July 30, 1993 SRM, directed the staff to submit to the Commission recommendations for proposed technical criteria and methods to use to justify simplifications of existing emergency planning requirements.

On July 13, 1994, the ACRS issued a report to the Commission supporting the directive from the Commission to the staff and recommending that the staff be directed to develop firm risk-based criteria for emergency planning zones (EPZs) for use with advanced plant designs.

Attachments:

- ACRS report to the Commission, dated July 13, 1994. Subject: Emergency planning zones, protective action guidelines, and the new source terms (pp. 21-24)

- Staff Requirements Memorandum dated March 18, 1994, from Samuel J. Chilk, Secretary, to J. Ernest Wilkins, Jr., ACRS Chairman, and James M. Taylor, EDO. Subject: Staff Requirements--Periodic Meeting with the ACRS, March 19, 1994 (pp. 25-26)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 13, 1994

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: EMERGENCY PLANNING ZONES, PROTECTIVE ACTION GUIDELINES, AND THE NEW SOURCE TERMS

During the March 10, 1994 meeting with the Commissioners, the ACRS agreed to consider the implications of the results reported in the ASEA Brown-Boveri Combustion Engineering (ABB-CE) Standard Safety Analysis Report for System 80+ design that the calculated doses for the design basis accidents (DBAs), using the new source terms and a hypothetical site, were less than protective action guidelines (PAGs) levels at the site boundary. During our 410th meeting on June 9-10, 1994, we had the benefit of a staff presentation on the use of PAGs in emergency planning. We also had the benefit of the referenced documents.

Calculated doses associated with the DBA prescription are sensitive to parameters associated with the DBA specifications, the containment design, and the site characteristics. These parameters include, for example, the source term itself (amount, timing, and chemical form), the effectiveness of engineered and natural aerosol mitigation processes (e.g., sprays and containment dimensions), containment volume and leak rate, the associated DBA pressure source, and specified meteorological conditions.

The items that appear to be major contributors to the low dose values calculated for System 80+ are:

- the large volume of the containment,
- an effective spray system design,
- an annular containment design that routes leakage through a filtered vent,
- the new specification for the source term contained in draft NUREG-1465 (particularly the timing), and

- the use of "medium" meteorological conditions as taken from the EPRI Utility Requirements Document for a hypothetical site instead of "worst-case" conditions.

The implication of the low value of the calculated DBA dose at the site boundary is that it points to a need to revisit the technical basis and rationale that underlie the present regulatory guidance on emergency planning - particularly with respect to the extent of Emergency Planning Zones (EPZs). This is an opportunity to develop a trial application of the concept of risk-based regulations.

The existing regulations require that emergency response plans be established and the guidance calls for including provisions for sheltering and/or evacuating within a 10-mile radius (i.e., plume exposure EPZ) around the reactor site in the event that doses anywhere in that region during an accident in progress are projected to exceed the PAGs. In addition, a 50-mile ingestion pathway zone is called for such that protective measures are available in the event that projected doses exceed additional PAG values in that zone.

The rationale for these requirements seems to be defined in NUREG-0654, from which we cite the following:

> "... it would be unlikely that any protective action for the plume exposure pathway would be required beyond the plume exposure EPZ."
>
> "... the likelihood of exceeding ingestion pathway protective action guide levels at 50 miles is comparable to the likelihood of exceeding plume exposure pathway protective action guide levels at 10 miles."
>
> "Projected doses from most core melt sequences would not exceed PAGs outside the [10-mile] EPZ."
>
> "For the worst core melt sequences, immediate life threatening doses would generally not occur outside the [10-mile] EPZ."

This is a good example of the type of regulatory basis that has concerned the ACRS for years. It has the "right-sounding" words but is lacking in real substance and is inflexible for new designs. In particular, it has only a loose risk basis rooted primarily in the results from WASH-1400, is specific only for contemporary LWRs, and uses qualifiers such as "unlikely," "likelihood," "most," and "generally." We believe the regulations related to emergency planning deserve better.

We believe the current regulatory extent of the EPZs as applied to existing nuclear plants implies an underlying level of "accepted

risk." If a comparable risk basis were to be applied to advanced plants, then the associated resulting EPZs would be expected to be smaller, possibly shrinking to the size of the site boundary.

The Commission, in the July 30, 1993 SRM, directed "... the staff should submit to the Commission recommendations for proposed technical criteria and methods to use to justify simplifications of existing emergency planning requirements." We support this directive from the Commission and note that, as part of the draft PRA implementation plan, the staff intends to proceed with efforts in that direction. We recommend that, as part of this effort, the staff be directed to develop firm risk-based criteria for EPZs for use with advanced plant designs. We believe developing such criteria would first require developing answers to the following questions:

- What level of risk is being "accepted" for currently operating LWRs with their existing EPZs?
- Is this level of "accepted" risk appropriate? If not, what should it be?
- For the advanced plant designs, what would be the size of the EPZs based on a level of risk comparable to the "accepted" value? What are the implications of this result?

We recognize that developing criteria based on "acceptable risk" would be conceptually as difficult as was development of the Safety Goal criteria. We also recognize that defense-in-depth might be a sufficient regulatory basis for the present extent of EPZs. Nevertheless, we believe that now is the appropriate time, and that the guidance on EPZs is the appropriate subject, for a trial effort on risk-based regulation to begin.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Memorandum dated March 18, 1994, from Samuel J. Chilk, Secretary, to J. Ernest Wilkins, Jr., ACRS Chairman, and James M. Taylor, EDO, Subject: Staff Requirements - Periodic Meeting with the ACRS, March 10, 1994
2. U.S. Nuclear Regulatory Commission, NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," December 1978

3. U.S. Nuclear Regulatory Commission, NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," February 1980

4. Staff Requirements Memorandum dated July 30, 1993, from Samuel J. Chilk, Secretary, for James M. Taylor, Executive Director for Operations, Subject: SECY-93-092 - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

IN RESPONSE, PLEASE
REFER TO: M940310

March 18, 1994

REVISED

MEMORANDUM TO: J. Ernest Wilkins, Jr., Chairman
Advisory Committee on Reactor Safeguards

James M. Taylor
Executive Director for Operations

FROM: Samuel J. Chilk, Secretary /s/

SUBJECT: STAFF REQUIREMENTS - PERIODIC MEETING WITH THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), 2:00 P.M., THURSDAY, MARCH 10, 1994, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Reactor Safeguards (ACRS).

The Commission reaffirmed its guidance to the Committee that schedular delays by one vendor should not interfere with the review of another vendor's design certification application.

The Commission requested that the ACRS expand its review from a study of fire barrier material testing to an evaluation of a spectrum of technical and regulatory approaches for dealing with fire barriers in the overall fire protection scheme and provide advice to the Commission.

The Committee agreed to specifically address in their review of the SYSTEM 80+ design the applicant's evaluation that, for the worst credible accident, the dose at the site boundary (one-half mile from the reactor) will remain below EPA's lower Protective Action Guideline of 1 rem. The Commission requested the staff to address this issue at the next meeting of the ACRS.

The Committee agreed to comment on the National Academy of Science's proposed workshop agenda on digital I&C. In particular, should the human factors issues be addressed at the same workshop, as suggested by the Nuclear Safety Research Review Committee, or separately? A concern was raised whether addressing both issues at the same workshop would divert attention and resolution from the issues raised by the ACRS.

cc: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
OGC
OCA
OIG
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR - Advance
DCS - P1-24

ITEM(5): THERMO-LAG FIRE BARRIERS

On August 12, 1992, the Inspector General issued a report that criticized the NRC for the handling of the review of Thermo-Lag test results. The NRC staff completed a review of the fire protection program and issued SECY-93-143, "NRC Staff Actions to Address the Recommendations In the Report On the Reassessment of the NRC Fire Protection Program," on May 21, 1993.

On November 19, 1993, the Auxiliary and Secondary Systems Subcommittee reviewed the technical aspects of the NUMARC Thermo-lag testing program. Subsequently, on December 16, 1993, the ACRS issued a letter expressing concern "...about the use of standards and practices that are based on fire protection standards developed for other industries."

On January 31, 1994, the Executive Director for Operations issued a letter in response to the December 16, 1993, ACRS letter. The response indicated that the staff would address any specific concerns identified by the ACRS, but that the present standards were adequate.

In the March 18, 1994, Staff Requirements Memorandum, the Commission requested that the ACRS expand its review of fire barrier material testing and evaluate a spectrum of technical and regulatory approaches for dealing with fire barriers in the overall fire protection scheme.

After an Auxiliary and Secondary Systems Subcommittee meeting, the ACRS issued a letter on June 14, 1994, supporting the staff's recommendation described as Option 1 in SECY-94-127, "Options For Resolving the Thermo-Lag Fire Barrier Issues." Option 1 required licensees to comply with existing NRC fire barrier requirements. Also, the Committee supported the staff's plan, described in SECY-94-090, "Institutionalization Of Continuing Program For Regulatory Improvement," to develop risk-based and performance-oriented fire protection regulations. The Committee recommended that any such regulatory framework include consideration of fire risk during shutdown conditions, and that the staff and industry work toward the development of generic guidelines for using performance-based approaches to justify exemptions to the requirements. Dr. Catton, in his additional comments, recommended that efforts be undertaken to develop guidance for rating fire barriers using realistic combustible loadings for fire endurance tests, as suggested in Option 2 of SECY-94-127.

In a June 27, 1994 letter to the Executive Director for Operations, the Commission provided direction to the NRC staff on the Thermo-Lag fire barrier issues.

Attachments:

- Letter dated December 16, 1993, from J. Wilkins, Jr, Chairman ACRS, to I. Selin, Chairman NRC, Subject: Thermo-Lag Fire Barriers (pp. 29-30)

- Letter dated January 31, 1994, from J. Taylor, Executive Director for Operations, to J. Wilkins, Jr., Chairman ACRS, Subject: THERMO-LAG Fire Barriers (pp. 31-34)

- Memorandum dated March 18, 1994, from S. Chilk, Secretary for the Commission, to J. Wilkins, Jr., Chairman ACRS, Subject: Staff Requirements - Periodic Meeting with the ACRS 2:00 p.m., Thursday, March 10, 1994 (pp. 35-36)

- Letter dated June 14, 1994, from T. Kress, Chairman ACRS, to I. Selin Chairman NRC, Subject: Thermo-Lag Fire Barriers (pp. 37-38)

- Letter dated June 27, 1994, from J. Hoyle, Acting Secretary for the Commission, to J. Taylor Executive Director for Operations, Subject: SECY-94-127 - Options for Resolving the Thermo-Lag Fire Barrier Issues (pp. 39-40)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 16, 1993

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: THERMO-LAG FIRE BARRIERS

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, in response to the referenced Staff Requirements Memorandum, we discussed with representatives of the NRC staff, NUMARC, and industry the technical differences between NUMARC and the NRC staff on the NUMARC test program related to Thermo-Lag fire barriers. Our Subcommittee on Auxiliary and Secondary Systems discussed this matter during a meeting on November 19, 1993. We also had the benefit of the documents referenced.

At the beginning of our review of the Thermo-Lag fire barrier issue, there were several differences between the staff and NUMARC on how the tests should be instrumented and configured to demonstrate compliance with Appendix R. The differences were in the placement of the thermocouples, whether or not cables should be used in the cable trays during testing, and in post-test evaluation of the cable condition. NUMARC has now agreed to use the thermocouple placement suggested by the staff, and the staff appears to have agreed to some testing with cables in the cable tray. How the test results will be used remains open.

The principal concern of the staff is that the limited number of tests will not yield enough data for extrapolation to the large number of specific configurations needing evaluation. The difficulty is compounded by incomplete characterization of the thermophysical properties of Thermo-Lag. The data from the planned tests can be made much more broadly applicable by additional temperature measurements and engineering analysis. In particular, we recommend that the Thermo-Lag cold side surface temperature be measured and that several identical Thermo-Lag configurations be tested with different cable loadings, including no cable. The resulting data and analysis should allow plant-specific cabling and ampacity factors to be dealt with. It should also be possible to resolve NUMARC concerns about excessive conservatism.

Thermo-Lag provides protection from a fire, in part, by material ablation. This suggests to us that aged material may not perform as well as new material. We recommend that at least one test be duplicated with in-service aged Thermo-Lag.

Our interest in fire protection goes beyond the Thermo-Lag issue. We are concerned about the use of standards and practices that are based on fire protection standards developed for other industries. Their utilization for nuclear power plant application should be specifically evaluated. The move towards risk-based regulation leads us to question present fire risk methodologies, and the adequacy of fire science talent within the agency. We look forward to being kept informed by the staff and NUMARC when they reconsider current fire protection regulations.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

<u>References:</u>

1. Staff Requirements Memorandum, dated November 15, 1993, to J. M. Taylor, EDO, and J. T. Larkins, ACRS, from S. J. Chilk, Secretary, regarding the October 29, 1993 Commission Briefing on Thermo-Lag
2. Memorandum, dated November 10, 1993, to J. T. Larkins, ACRS, from A. Thadani, NRR, regarding ACRS Subcommittee Meeting on Thermo-Lag
3. Memorandum, dated October 8, 1993, for the Commissioners from J. M. Taylor, EDO, Subject: Quarterly Updates of the Thermo-Lag and Fire Prot..tion Task Action Plans

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 31, 1994

Dr. J. Ernest Wilkins, Jr., Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Wilkins:

SUBJECT: THERMO-LAG FIRE BARRIERS

We have reviewed your letter of December 16, 1993, regarding the Nuclear Resources Management Council (NUMARC) Thermo-Lag fire barrier test program. We appreciate the Advisory Committee on Reactor Safeguards prompt review of the technical differences between the U.S. Nuclear Regulatory Commission (NRC) staff and NUMARC. I believe that ACRS played an important role in NUMARC's decision to install thermocouples in accordance with the NRC staff's recommendations.

In your letter, you addressed Thermo-Lag fire barriers and several other fire protection issues. Our initial thoughts on these issues are in the enclosure to this letter. The NRC staff is interested in meeting with the ACRS during March 1994 to fully discuss these and any other fire protection issues of interest to the ACRS.

Please have your staff contact Ms. Helen Pastis, the Office of Nuclear Reactor Regulation/ACRS Coordinator, to arrange a meeting schedule that is convenient with you.

James M. Taylor
Executive Director
for Operations

Enclosure:
As stated

cc: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
SECY
OPA
OCA
OGC
J.T. Larkins

ENCLOSURE

PRELIMINARY RESPONSES TO FIVE POINTS RAISED IN
LETTER OF DECEMBER 16, 1993
FROM J.E. WILKINS, ACRS, TO THE CHAIRMAN

1. *Recommend that the Thermo-Lag cold side surface temperature be measured and that several identical Thermo-Lag configurations be tested with different cable loadings.*

The ACRS recommendations to measure the Thermo-Lag cold side surface temperature and that several identical Thermo-Lag fire barrier test specimens be tested without a cable load and with different cable loadings is a good approach towards testing raceway fire barrier systems. The staff agrees that this approach, if performed in a manner that would bound conduit and cable tray sizes, cable loadings and cable types, could produce the additional data needed to support engineering analyses and would be a better method to broaden the applicability of fire test results. However, the staff, in an effort to find a timely and cost effective solution to the Thermo-Lag fire barrier technical issues, has approached the problem by applying fire endurance testing acceptance criteria which are consistent with existing NRC fire protection regulatory guidance and guidance. The staff considers that its proposed position (Generic Letter 86-10, Supplement 1) on raceway fire barrier testing and the specified test specimen thermocouple placement will provide sufficient thermal data to evaluate the adequacy of these test specimen to perform their fire resistive function under the specified fire test conditions.

Recently, the staff has issued 10 CFR 50.54(f) letters to those licensees which use Thermo-Lag fire barriers. In these letters, the staff requested additional information regarding Thermo-Lag fire barrier configurations and amounts; fire barrier construction techniques, cable fill and construction, and raceway bounding parameters; and ampacity derating. The staff considers that the thermal data as determined by fire tests preformed in a manner which is equivalent to its proposed position, the additional information from individual licensees, and the technical adequacy of the NUMARC application guide are key to a timely resolution of the current Thermo-Lag fire barrier technical issues.

2. *Recommend that at least one test be duplicated with in-service aged Thermo-Lag.*

During May 1992, the National Institute of Standards and Technology (NIST) conducted chemical analyses of six Thermo-Lag samples for the NRC staff. The samples consisted of old (1981), middle-aged (late 1980's), and new Thermo-Lag materials. NIST concluded: "Except for the water content of [the trowel-grade material], all six samples are similar in composition and behavior. Differences have been observed only in the minor components. All of the components detected, except 2-phenoxyethanol, are those found, or closely related to those found, in fire retardant materials, especially intumescent fire resistant paints." This information, coupled with the fact that the fire barriers are

subject to routine surveillance inspections which lead to repairs of degraded barriers, indicates to the staff that aging may not be a significant issue.

Finally, it may not be possible to remove intact sections of Thermo-Lag from existing in-plant barriers for full-scale testing. As a result of the plaster like texture of fully cured Thermo-Lag fire barrier materials, structural damage to an existing barrier system may result in disassembling one of these fire barrier systems. Even if a section of Thermo-Lag protected cable tray could be removed from a decommissioned plant the Thermo-Lag fire barrier assembly would have to be completely disassembled in order for it to be properly instrumented with thermocouples. This would require that the fire barrier test specimen be reconstructed with new materials (e.g., stainless steel banding, trowel grade Thermo-lag 330-1 fire barrier material). In addition, it should be recognized that the conservatism in fire resistive performance of these fire barrier systems as required by the current regulations should be sufficient, in most plant applications, to compensate for any aging and provide an adequate level of fire safety under actual plant fire conditions.

3. *Standards and practices based on fire protection standards developed for other industries should be specifically evaluated for use in nuclear power plant applications. The move toward risk-based regulation leads us to question present fire risk methodologies.*

The ACRS did not provide specific examples of industry standards and practices that should be evaluated for nuclear industry application. The staff will address any specific standard or practice of concern to the ACRS. However, it is the staff position that the consensus fire protection standards that are referenced in the Standard Review Plan (SRP), such as those issued by the National Fire Protection Association, when used in accordance with the SRP, are adequate to develop nuclear power plant fire protection programs. The staff considered the relevance of the referenced standards to nuclear power plant design, construction, operation, and maintenance when it prepared the SRP. The staff also reviews for endorsement fire protection standards that are not referenced in the SRP, as needed and appropriate.

The fire protection standards referenced in the SRP address distinct fire protection practices, features, and systems that are applicable to a variety of industries. For example, the NFPA standard for sprinkler systems is not industry specific, but covers essential elements of systems design, installation, and maintenance for any sprinkler installation. The SRP provides bases and staff guidance for determining where sprinklers should be installed to give adequate coverage. The standard is used after a decision has been made to install sprinkler protection to ensure that the system is properly designed, installed, and maintained.

Staff assessment of fire risk methodologies will be integral to its efforts to develop a risk-based fire protection regulation. The Office

of Nuclear Reactor Regulation (NRR) has a technical assistance contract with Brookhaven National Laboratories (BNL) and RES has a contract with BNL and the National Institute of Standards and Technology (NIST) to assist with this effort. The staff will meet with ACRS as appropriate during the rulemaking process and will address any specific concerns or comments.

4. *The move toward risk-based regulation leads us to question the adequacy of fire science talent within the agency.*

Regulation using fire risk methodologies and risk-based fire protection is an emerging concept. The development and application of risk-based methods will present considerable challenges to both the staff and industry. This effort cannot be successfully completed by fire protection engineers alone, but will require the coordinated and integrated efforts of fire protection engineers, systems engineers, and probabalistic risk assessment (PRA) specialists. NRR has recognized the importance of having fire protection engineers with significant experience with fire protection practices. In order to further enhance our capability NRR recently hired an additional senior fire protection engineer well versed in the state of art of fire protection engineering. NRR is also in contact with the international community to remain cognizant of relevant information. We also expect the PRA expertise of the NRR engineers will mature along with the risk methodologies. It is also expected that there will be mutually beneficial transfer of knowledge between the disciplines involved in the development and application of these methodologies. The staff will also use the technical assistance of contractors, as appropriate.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

IN RESPONSE, PLEASE
REFER TO: M940310

March 18, 1994

REVISED

MEMORANDUM TO: J. Ernest Wilkins, Jr., Chairman
Advisory Committee on Reactor Safeguards

James M. Taylor
Executive Director for Operations

FROM: Samuel J. Chilk, Secretary /s/

SUBJECT: STAFF REQUIREMENTS - PERIODIC MEETING WITH THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), 2:00 P.M., THURSDAY, MARCH 10, 1994, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Reactor Safeguards (ACRS).

The Commission reaffirmed its guidance to the Committee that schedular delays by one vendor should not interfere with the review of another vendor's design certification application.

The Commission requested that the ACRS expand its review from a study of fire barrier material testing to an evaluation of a spectrum of technical and regulatory approaches for dealing with fire barriers in the overall fire protection scheme and provide advice to the Commission.

The Committee agreed to specifically address in their review of the SYSTEM 80+ design the applicant's evaluation that, for the worst credible accident, the dose at the site boundary (one-half mile from the reactor) will remain below EPA's lower Protective Action Guideline of 1 rem. The Commission requested the staff to address this issue at the next meeting of the ACRS.

The Committee agreed to comment on the National Academy of Science's proposed workshop agenda on digital I&C. In particular, should the human factors issues be addressed at the same workshop, as suggested by the Nuclear Safety Research Review Committee, or separately? A concern was raised whether addressing both issues at the same workshop would divert attention and resolution from the issues raised by the ACRS.

cc: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
OGC
OCA
OIG
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR - Advance
DCS - P1-24

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555

June 14, 1994

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: THERMO-LAG FIRE BARRIERS

During the 410th meeting of the Advisory Committee on Reactor Safeguards, June 9-10, 1994, we discussed the proposed staff approach for resolving Thermo-Lag fire barrier issues with representatives of the NRC staff and Nuclear Energy Institute (NEI). Our Subcommittee on Auxiliary and Secondary Systems reviewed this matter during a meeting on June 8, 1994. We also had the benefit of the documents referenced. This report is in response to the March 18, 1994 Staff Requirements Memorandum.

We agree with the staff's view that an immediate order to require upgrading of inadequate Thermo-Lag fire barriers is not needed based on defense-in-depth arguments and the fact that compensatory measures are already in place at those plants that have not resolved their Thermo-Lag problems.

In SECY-94-127, the staff describes the following four options for resolving the Thermo-Lag fire barrier issues:

Option 1 - Require Compliance with Existing NRC Fire Barrier Requirements

Option 2 - Develop Guidance for Rating Fire Barriers Based Upon a Range of Combustible Loadings for Fire Endurance Tests

Option 3 - Develop a Performance-Based Approach Using a Lead Plant

Option 4 - Develop a Performance-Based Fire Protection Rule

We support the staff recommendation described as Option 1, which includes provisions for plant-specific exemptions as permitted in the current regulations. However, we believe that exemptions under Option 1 should not be limited to those permitted by precedent.

Fire-analysis techniques have advanced substantially since the current fire protection regulations were promulgated. These advances justify a reexamination of the bases for granting exemptions. We recommend that, in the near term, the staff and industry work toward the development of generic guidelines for using performance-based approaches to justify exemptions.

We are advocates of risk-based regulation and therefore support the staff's plan, described in SECY-94-090, to develop risk-based and performance-oriented fire protection regulations and recommend that any such regulatory framework include consideration of fire risk during shutdown conditions.

Additional comments by ACRS Member Ivan Catton are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Additional Comments of ACRS Member Ivan Catton

While I agree with some of what is said in the above report, I do not understand why the implementation of Option 2 is considered to be so complex. The computational tools are available to support the selection of Option 2 as a means to resolve the Thermo-Lag issues without resorting to a large number of exemptions. There are examples of how this can be done. Further, most of what must be done will support the effort to achieve a performance-based fire protection regulation. I believe it is time to follow the lead of other countries (e.g., Sweden, Australia, and others) in moving toward realistic performance-based fire protection regulation.

References:

1. SECY-94-127 dated May 12, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Options for Resolving the Thermo-Lag Fire Barrier Issues
2. SECY-94-128 dated May 12, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Status of Thermo-Lag Fire Barriers
3. Memorandum dated March 18, 1994, from Samuel J. Chilk, Secretary, to J. Ernest Wilkins, Jr., ACRS Chairman, and James M. Taylor, EDO, Subject: Staff Requirements - Periodic Meeting with the ACRS, March 10, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 27, 1994

DISTRIBUTED TO ACRS MEMBERS

DISTRIBUTED TO ACNW MEMBERS

9410121

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John C. Hoyle, Acting Secretary /s/

SUBJECT: SECY-94-127 - OPTIONS FOR RESOLVING THE THERMO-LAG FIRE BARRIER ISSUES

The Commission (with all Commissioners agreeing) has approved the continued use of Option 1, which requires compliance with existing NRC requirements and permits plant-specific exemptions where justified. The Commission does not intend to limit the staff's consideration of requests for exemptions currently permitted by regulations.

The Commission (with the Chairman and Commissioners Remick and de Planque agreeing) requests that the staff, in consultation with industry, consider possible new exemptions to Appendix R based on state-of-the-art fire protection methodology and technology and proceed to evaluate the feasibility of developing new guidance for rating fire barriers on the basis of representative plant fire hazards as described in Option 2. The responsibility for developing the technical basis for any new exemptions should rest with the licensees.

Commissioner Rogers disapproved proceeding with Option 2. In particular, he felt the staff should not proceed with the development of a Regulatory Guide in support of this option. However, he did not object to the staff's initiating a research project to investigate the feasibility of either developing standard fire curves or using realistic fire loads. This information would be applicable to a performance-based approach to a fire protection rule.

The Chairman and Commissioner Rogers approved the staff recommendation not to proceed with the development of a performance-based approach to resolve the Thermo-Lag issue as

SECY NOTE: SECY-94-127 WAS RELEASED TO THE PUBLIC ON MAY 20, 1994. THIS SRM AND THE VOTE SHEETS OF ALL COMMISSIONERS WILL BE MADE PUBLICLY AVAILABLE 10 WORKING DAYS FROM THE DATE OF THIS SRM.

ACRS OFFICE COPY / 39

described in Option 3. They believe that the performance based approach should be applied to support development of the fire protection rule. Commissioner Remick and Commissioner de Planque had no objection to pursuing Option 3 if the responsibility and initiative came from industry.

The Commission (with all Commissioners agreeing) has approved the staff recommendation to proceed as planned with the development of a performance-based fire protection rule. This should be pursued by the staff as part of its continuing program for regulatory improvement and/or once a request for rulemaking is received. The Commission felt that the new rule should not be considered a means to resolve the Thermo-Lag issues.

cc: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
OGC
OCA
OIG
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)

ITEM(6): NATIONAL ACADEMY OF SCIENCES/NATIONAL RESEARCH COUNCIL STUDY AND WORKSHOP ON DIGITAL I&C SYSTEMS

Following an extended ACRS review (2+ years) of the state of digital technology and its use in nuclear power plants, ACRS wrote a March 18, 1993 letter to the Commission recommending that it seek the advice of the National Academies of Science and Engineering (NAS&E) regarding the type and extent of regulatory requirements needed for this technology as applied to nuclear power plants. Following a staff-sponsored workshop on digital reliability conducted jointly with NIST, the ACRS again repeated its recommendation in a letter dated November 16, 1993. On December 16, 1993, the Commission directed the staff to ask the NAS&E to conduct a workshop and perform a study to assist in establishing the appropriate approach for regulating the use of digital I&C systems in nuclear power plants. Following an ACRS briefing to the Commission in March 1994, the Commission requested the ACRS to review and comment on the NAS&E proposal for a study and workshop.

Following a presentation by NAS&E representatives during the 411th ACRS meeting in July 1994, the Committee issued a letter dated July 14, 1994 commenting on the NAS&E proposal. As noted in this letter, the ACRS anticipates meeting with members of the study panel during the course of the study.

Attachments:

- Staff Requirements Memorandum dated March 18, 1994, Subject: Staff Requirements - Periodic Meeting with the Advisory Committee on Reactor Safeguards (ACRS), 2:00 pm, Thursday, March 10, 1994, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to Public Attendance) (pp. 42-43)

- ACRS Letter dated July 14, 1994, Subject: Proposed National Academy of Sciences/National Research Council Study and Workshop on Digital Instrumentation and Control Systems (pp. 44-45)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

IN RESPONSE, PLEASE
REFER TO: M940310

March 18, 1994

REVISED

MEMORANDUM TO: J. Ernest Wilkins, Jr., Chairman
Advisory Committee on Reactor Safeguards

James M. Taylor
Executive Director for Operations

FROM: Samuel J. Chilk, Secretary /s/

SUBJECT: STAFF REQUIREMENTS - PERIODIC MEETING WITH THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), 2:00 P.M., THURSDAY, MARCH 10, 1994, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Reactor Safeguards (ACRS).

The Commission reaffirmed its guidance to the Committee that schedular delays by one vendor should not interfere with the review of another vendor's design certification application.

The Commission requested that the ACRS expand its review from a study of fire barrier material testing to an evaluation of a spectrum of technical and regulatory approaches for dealing with fire barriers in the overall fire protection scheme and provide advice to the Commission.

The Committee agreed to specifically address in their review of the SYSTEM 80+ design the applicant's evaluation that, for the worst credible accident, the dose at the site boundary (one-half mile from the reactor) will remain below EPA's lower Protective Action Guideline of 1 rem. The Commission requested the staff to address this issue at the next meeting of the ACRS.

The Committee agreed to comment on the National Academy of Science's proposed workshop agenda on digital I&C. In particular, should the human factors issues be addressed at the same workshop, as suggested by the Nuclear Safety Research Review Committee, or separately? A concern was raised whether addressing both issues at the same workshop would divert attention and resolution from the issues raised by the ACRS.

cc: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
OGC
OCA
OIG
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR - Advance
DCS - P1-24

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 14, 1994

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: PROPOSED NATIONAL ACADEMY OF SCIENCES/NATIONAL RESEARCH COUNCIL STUDY AND WORKSHOP ON DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

During the 411th meeting of the Advisory Committee on Reactor Safeguards, July 7-8, 1994, we discussed the proposal by the National Academy of Sciences/National Research Council (NAS/NRC) for a study and workshop on the "Application of Digital Instrumentation and Control Technology to Nuclear Power Plant Operations and Safety." During our review, we had the benefit of discussions with representatives of the NRC staff and the NAS/NRC. We also had the benefit of the documents referenced. This report is in response to a Commission request in the March 18, 1994 Staff Requirements Memorandum.

The proposal focuses primarily on hardware and software issues that arise from the introduction of digital instrumentation and control (I&C) technology in nuclear power plants. Human factors considerations appear to be limited to human-machine interface issues related directly to digital technology. We believe this balance in emphasis is proper. The issues associated with hardware and software are very broad and any significant diversion of effort from these issues is undesirable. In addition, we believe that the staff's Human Factors Engineering Program Review Model and the acceptance criteria used for evolutionary reactors provide reasonable regulatory guidance for human factors issues. The current need is for a corresponding regulatory framework for hardware and software issues associated with digital I&C technology.

We believe the NAS/NRC study panel findings will assist the Commission in providing necessary guidance to the staff for the development of a regulatory framework for digital I&C. While the staff and the ACRS have identified a number of concerns that are believed to be significant, the ACRS strongly urges that the study panel be permitted to select the issues to be considered.

We expect that the NAS/NRC study will make use of knowledge that has been developed in other industries with digital system experience. We are particularly interested in the state-of-the-art of the development of software specifications, verification and validation of software, the potential vulnerabilities of hardware over the spectrum of adverse environments which can occur in nuclear power plants, and the prediction of reliability (including common-mode failure).

We recommend that the staff identify in the background papers provided to the NAS/NRC study panel those applicable NRC regulations, IEEE standards, Electric Power Research Institute Utility Requirements, and vendor information that pertain to safety-related digital I&C system development.

We understand that a visit to the NRC Technical Training Center simulators is planned. It may be more useful for study panel members to visit a nuclear plant digital system vendor to observe developmental mock-ups and to discuss nuclear power plant digital I&C designs. Consideration should also be given to visiting an operating plant that employs digital control and protection systems.

We look forward to meeting with members of the study panel during the course of the study.

Sincerely,

J. S. Kress

T. S. Kress
Chairman

<u>References</u>:

1. Memorandum dated March 18, 1994, from Samual J. Chilk, Secretary, to J. Ernest Wilkins, Jr., ACRS Chairman, and James M. Taylor, EDO, Subject: Staff Requirements - Periodic Meeting with the ACRS, March 10, 1994
2. Memorandum dated March 1, 1994, from James M. Taylor, Executive Director for Operations, NRC, for The Commission, Subject: Nuclear Safety Research Review Committee Report Dated January 14, 1994
3. Memorandum dated May 3, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commission, Subject: Staff Response to Nuclear Safety Research Review Committee Reports Dated January 14 and February 16, 1994
4. ACRS Letter Report dated March 18, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations
5. ACRS Letter Report dated November 16, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations

ITEM(7): INTERIM STEAM GENERATOR TUBE REPAIR CRITERIA

The NRC staff approved interim tube plugging criteria based on voltage readings of non-destructive ultrasonic tests for licensees of Farley, Trojan, Braidwood, and Kewaunee. The approved interim tube plugging criteria are less restrictive than the traditional minimum tube-wall thickness criteria, which are specified in each licensee's technical specifications.

The proposed Generic Letter, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes," provides details of the additional testing and analysis necessary to support revising the technical specifications. Dr. Hopenfeld, RES, submitted a formal differing professional opinion based on the perceived non-conservative basis for the interim tube plugging criteria. The Executive Director for Operations, requested that the ACRS comment on the differing professional opinion.

During the August 3, 1994, Materials and Metallurgy Subcommittee meeting, the NRC staff presented incorrect information on the off-site dose that might result from the conditions allowed by the proposed Generic Letter. For the corrected off-site dose calculations to meet 10 CFR Part 100 limits, the staff changed some initial assumptions. The staff changed the assumed primary leak rate from 100 gpm to 25 gpm, the assumed iodine concentration from a peak value over two hours to an average value, and the dispersion factor.

The Committee plans to complete a report to the Commission regarding this matter during its September meeting.

Attachments:

- Memorandum dated July 14, 1994, from J. Taylor, Executive Director for Operations, to J. Larkins, Executive Director for ACRS, Subject: ACRS Review of proposed Generic Letter 94-XX, Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes (p. 47)

- Memorandum dated August 17, 1994, from J. Calvo, Acting Director, Division of Radiation Safety and Safeguards, to J. Larkins, Executive Director for ACRS, Subject: Revision to Slides Used by Staff During August 3, 1994, Subcommittee Briefing on Steam Generator Alternate Repair Criteria (pp. 48-49)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 15, 1994

MEMORANDUM FOR: John T. Larkins
Executive Director for the Advisory Committee on Reactor Safeguards

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: ACRS REVIEW OF PROPOSED GENERIC LETTER 94-XX, VOLTAGE-BASED REPAIR CRITERIA FOR WESTINGHOUSE STEAM GENERATOR TUBES

Reference 1: Differing Professional Opinion Regarding Voltage-Based Interim Repair Criteria for Steam Generator Tubes, dated July 13, 1994.

I am enclosing Reference 1 which seems material and relevant to the actions proposed within the subject generic letter that ACRS will be considering at its August 3, 1994 subcommittee meeting and with the full Committee on August 4, 1994. The author of Reference 1 has advised my staff (by 7/14/94 telephone conversation) that he agrees to sending his concerns to the ACRS and to the CRGR for its consideration during deliberations on the proposed generic letter. Further, the author has also agreed to appear at the August 3, 1994 ACRS Subcommittee meeting and contribute his professional views on this matter to the degree the subcommittee chairman determines appropriate in completing the ACRS review of the proposed generic letter. Particular views or comments that the ACRS may wish to offer on any of the issues contained in Reference 1 would be appreciated - these will be factored into my final decisions on dispositioning. Should you have further questions on this matter, please contact M. Taylor of my staff at 504-1722.

James M. Taylor
Executive Director
for Operations

Enclosure:
As stated

cc w/enclosure:
E. Jordan, CRGR Chairman

cc w/o enclosure:
J. Hopenfeld
E. Beckjord

ADVANCE COPY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

AUG 17 1994

MEMORANDUM FOR: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

FROM: Jose A. Calvo, Acting Director
Division of Radiation Safety
and Safeguards

SUBJECT: REVISIONS TO SLIDES USED BY STAFF DURING AUGUST 3, 1994, SUBCOMMITTEE BRIEFING ON STEAM GENERATOR ALTERNATE REPAIR CRITERIA

The purpose of this memorandum is to call to your attention the fact that Thomas Essig of my staff unknowingly presented certain erroneous information to the ACRS Materials and Metallurgy Subcommittee during his August 3, 1994, briefing. Mr. Essig's briefing material focussed on the calculations for radiological consequences of a main steam line break associated with a steam generator with degraded tubes. Two versions of the corrected slides are enclosed -- a red-line/strike-out version and a revised version. Our conclusion remains essentially unchanged, except to note that margin in the calculations, i.e., the flexibility to increase certain parameters and retain a particular outcome, has been reduced.

The error that was made involved the λ_e term -- the sum of λ_c plus λ_r (please refer to the enclosed slides for further detail regarding these terms). During the evaluation of the post-trip release rate, λ_e was set equal to λ_r, since after the depressurization event, λ_c was assumed to be zero (letdown isolated). When the pre-trip release rate ($\lambda \times A_o$, the pre-trip activity) was calculated, the wrong λ was used. Since RCS cleanup would have been occurring (a normal operational activity), λ_e should have been used -- not λ_r alone. This caused the release rate of ^{131}I from fuel to be underestimated by a factor of 28. Beyond correcting this error, other changes have been made in the enclosed slides, as follows:

- An adjustment was made in the algorithm to facilitate the calculation of the <u>average</u> concentration over the post-trip interval, rather than the concentration at the <u>end</u> of the two-hour interval as shown in the August 3rd presentation.

- The primary to secondary leak rate was decreased from 100 gpm to 25 gpm to be more consistent with values used in licensing actions to date, but yet remain somewhat of a bounding value.

- The atmospheric dispersion parameter (χ/Q) was increased from 5 x 10^{-4} (corresponding approximately to the median value for 95th percentile dispersion parameters) to 1.8 x 10^{-3} sec/m^3, which is near the upper end of the range of χ/Q values for existing sites; it

is also the value used for design basis accident calculations in NUREG-1477 (Table 4.1.2-1).

We would also like to take this opportunity to correct an impression which Mr. Essig may have given to the Subcommittee during his response to a question from Dr. Powers. Dr. Powers inquired as to the relative magnitude of the iodine spiking compared to the available gap inventory (and subsequent release) of iodine. Mr. Essig responded that of the 1×10^7 Ci of ^{131}I in the core, about 5 percent is in the gap, and with a 0.5 percent cladding failure, approximately 2500 Ci could be released to the coolant. In his response, Mr. Essig incorrectly recalled the core inventory; it should be around 8×10^7 Ci for a 3000 MWt core. This would result in about 20,000 Ci of ^{131}I potentially being released to the primary coolant. The spiking model, on the other hand, would suggest that approximately 9000 Ci of ^{131}I would be released to the RCS during the two-hour interval considered.

We sincerely apologize for this error and other adjustments to the slides, and regret any inconvenience it may have caused the Committee.

José A. Calvo

Jose A. Calvo, Acting Director
Division of Radiation Safety
and Safeguards

Enclosures:
As stated

ITEM(8): NEED FOR REVIEW OF RATIONALE FOR REGULATION

During the 407th ACRS meeting in March 1994, the Committee issued a report regarding the need to review the traditional bases for plant licensing and regulation. An informal meeting with some Commissioner's Technical Assistants was held in June 1994, to discuss and clarify the intent of the Committee's report. Dr. Tom Kress documented some of his ideas on this matter in an undated memorandum that was used as a basis for the discussion.

Attachments:

- ACRS report dated March 15, 1994, Subject: Need for Review of Rationale for Regulation (p. 51)

- Memorandum by Dr. T. S. Kress, undated, Subject: Further Explanation/Expansion on our "Rationale for Regulation" Letter (pp. 52-53)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 15, 1994

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: NEED FOR REVIEW OF RATIONALE FOR REGULATION

During our review of source-term issues at the 407th meeting of the Advisory Committee on Reactor Safeguards, March 10-12, 1994, we were struck once again by the need to review the traditional bases for plant licensing and regulation. The source-term issues are part of the general question of the proper role for the standard design basis accidents, which is part of the issue of the often-mentioned bottom-up review of the General Design Criteria. These basic reviews of the continuing rationale for regulation never seem to assume a high priority within NRC, relative to short-term matters. We think they should.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

Further Explanation/Expansion
on our
"Rationale for Regulation" Letter

T. S. Kress

The "Rationale" letter was our way of calling for increased coherence in regulations by putting them on a risk rationale - particularly with respect to Design Basis Accidents (DBAs) and the General Design Criteria (GDC).

What we mean by this is best explained by example. Consider the DBAs for containment design associated with 10 CFR Part 100. There is an implied assumption when we use such DBAs that the resulting plant design will have acceptable risk *for all accident sequences*. What the regulator is basically saying then is that the core melt frequency (CMF) that results from other regulatory requirements, combined with the conditional containment failure probability (CCFP) that results form our DBA specifications, gives a risk that is acceptable.

Our "Rationale" letter is asking for an evaluation of such regulations to see if, indeed, the above would generally hold true.

How would such a study be done or how would one write a coherent set of regulations if one were to be able to start all over?

We are not altogether certain of how to approach this, but we have a feeling that something along the following lines could be done:

1. We must first articulate what is meant by acceptable risk. We have the Safety Goals to guide us here but we may want to reinterpret them to consider the 95 "percentile" instead of the "mean" as a way to incorporate the uncertainties.

2. We recognize that the risk of present plants is dominated by large source term releases from failed containments. Therefore, the next step is to use a high ST (that would include MCCI) and some sort of bounding site characteristics (wind and population) to determine a representative bounding consequence.

3. Using the definition of risk, ($\Sigma_i P_i \bullet C_i$) we next define an *acceptable* probability as being the acceptable risk divided by the bounding consequence.

4. At this point, we could address the various regulatory requirements for scram, ECCS, redundant power supplies, etc., to show what level of CMF results from these—using all the initiators. For each sequence, you must then determine

the conditional probability for early containment failure, CCFP.

Summing over all sequences.

$$\Sigma_i \ (CMF)_i \times (CCFP)_i$$

has to give a value that is less than the "acceptable probability" determined in step 3 above.

5. The DBAs, in conjunction with GDCs should be evaluated to show that strict adherence to them will satisfy (4). If this cannot be shown, then the regulaitons need to be revised.

ITEM(9): SELECTION OF NEW ACRS MEMBERS

The attached documents will be used for discussion.

Attachments:

- Memorandum dated June 16, 1994, from J. Hoyle, Acting Secretary, to J. Larkins, Executive Director for ACRS, and J. Taylor, Executive Director for Operations, Subject COMSECY-94-018 - Advisory Committee Member Selection (p. 55)
- Memorandum dated May 4, 1994, from J. Hoyle, Acting Secretary, to J. Larkins, Executive Director for ACRS, and J. Taylor, Executive Director for Operations, Subject: COMIS-94-003 - Expanded input in Advisory Committee Selections (p. 56-57)

OFFICE OF THE SECRETARY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

June 16, 1994

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

John T. Larkins
Executive Director, ACRS/ACNW

FROM: John C. Hoyle, Acting Secretary

SUBJECT: COMSECY-94-018 - ADVISORY COMMITTEE MEMBER SELECTION

The Commission (with all Commissioners agreeing) has approved the proposed Federal Register notice and press release for solicitation of nominations for a new ACRS member. The press release and announcement should provide some sense for the time frame for submittal of nominations if a candidate is to be considered for the vacancy. Consistent with the changes discussed below, the announcement and press release should be changed to request the nominations be sent to the Office of Personnel.

The Commission is providing the following additional guidance on the process for obtaining nominations for NRC's Federal advisory committee positions:

a) The Designated Federal Official for the committee with the vacancy should prepare the Federal Register notice and press release for Commission approval;

b) The Office of Personnel (OP) should convene the appropriate screening panel for review of nominations and provide it with the necessary administrative support; and

c) The appropriate individual from outside the agency should be invited by the NRC Chairman to join the panel.

With OP convening and supporting the screening panel all nominations and resumés should be sent directly to OP.

cc: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
OGC
OCA
OIG

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

May 4, 1994

OFFICE OF THE SECRETARY

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

John T. Larkins, Executive Director,
Advisory Committee on Reactor Safeguards and
Advisory Committee on Nuclear Waste

FROM: John C. Hoyle, Assistant Secretary

SUBJECT: COMIS-94-003 - EXPANDED INPUT IN ADVISORY COMMITTEE SELECTIONS

The Commission (with all Commissioners agreeing) has approved the following procedure for the future selections of new members on advisory committees:

1) The Commission will be provided a draft Federal Register notice and proposed press release and a list of the professional societies/technical organizations for the solicitation of nominations. These documents will indicate what specific expertise/skills are being sought for the opening. The specific expertise/skills will be chosen in consultation with the advisory committee which has the opening.

2) At the time of publication of the Federal Register notice and press release, notification of the search for nominations will be given to appropriate professional societies/technical organizations. The advisory committee with the opening should be specifically invited to suggest candidates.

3) A screening panel will be established to review the resulting nominations. The panel will be composed of:

   a) a representative of the Commission or the principal staff office with whom the committee works,

   b) a (full-time federal employee) representative of the advisory committee with the existing or anticipated vacancy, and

   c) an individual (full-time federal employee) identified by the Commission, preferably from outside the agency,

who possesses the expertise/skills being sought.

4) Each screening panel will:

a) Review and rate the nominations for the selecting official using as benchmarks the specific expertise/skills being sought for the opening, as well as the individual's breadth of knowledge and ability/experience in applying his/her skills to problems outside of their specific field of expertise. The panel's report should list all the qualified candidates, and it should rank at least the best qualified candidates. A brief narrative should be provided identifying the criteria and rationale for the best qualified rankings.

b) In carrying out the provisions of a) above, the panel may seek the advice of other individuals whose views may be useful to the screening panel.

c) Submit a copy of the panel's report to the appropriate committee for its independent recommendation on the nominees, as well as submit a copy to the Commission (or to the designated selecting official for the particular advisory committee).

5) The advisory committee should submit its selection recommendations to the screening panel, and/or the Commission (or the designated selecting official) as they see fit.

This process should be implemented for the selection of advisory committee members for all cases which begin after the issue date of this SRM. For those selections which are currently in process, the previcus process should be used.

NRC's Advisory Committee Management Officer, with input from the advisory committees and from the screening panels, should provide an evaluation of the functioning of this procedure after a two-year period.

cc: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
OGC
OCA
OIG
OP