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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Title: PERIODIC MEETING WITH THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

Location: ROCKVILLE, MARYLAND

Date: MARCH 10, 1994

Pages: 73 PAGES

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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PERIODIC MEETING WITH THE ADVISORY
COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

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PUBLIC MEETING

Nuclear Regulatory Commission
One White Flint North
Rockville, Maryland

Thursday, March 10, 1994

The Commission met in open session, pursuant to notice, at 2:00 p.m., Ivan Selin, Chairman, presiding.

COMMISSIONERS PRESENT:

IVAN SELIN, Chairman of the Commission
KENNETH C. ROGERS, Commissioner
FORREST J. REMICK, Commissioner

STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

JOHN HOYLE, Assistant Secretary

KAREN CYR, Office of the General Counsel

DR. J. ERNEST WILKINS, JR., Chairman, ACRS

DR. THOMAS KRESS, ACRS

MR. JAMES CARROLL, ACRS

DR. IVAN CATTON, ACRS

MR. CHARLES WYLIE, ACRS

MR. CARLYLE MICHELSON, ACRS

DR. HAROLD LEWIS, ACRS

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

2:00 p.m.

CHAIRMAN SELIN: Good afternoon, ladies and gentlemen.

We're pleased to welcome the Advisory Committee for Reactor Safeguards who will be briefing us on a number of issues they've recently taken under consideration. Today we'll mostly focus on advanced reactor and design certifica. n issues and this is especially timely since the staff is nearing completion of the review of the first standard plant design under Part 52.

I should say that contrary to common belief, I look forward to most of our ACRS meetings. Today is one that is the cause of considerable mixed emotion because it's Doctor Wilkins last meeting with the Commission. His term ends next month. We very much regret that he has chosen not to seek reappointment.

Doctor Wilkins, on behalf of the Commissioners and the staff, we'd like to express our gratitude and thank you for your role in advising the Commission and in chairing ACRS. Since you joined the ACRS in 1990, you have faced the challenging period. You have done a superb job. You take proper regard of copies of the agenda. There are eight items on this list. I can tell you quite frankly that I have some doubts we can get through all eight of these. But it's my understanding that this is roughly your priorities.

I would like to propose that we interchange items 4 and 5 because there will be some common issues in 3 and 5 that deal specifically with schedules and priorities for ACRS activities. I'd like to raise some of those with you after you shall have heard the discussions of the subcommittee chairmen of those areas.

Then we'll get started and see how far we get. I'm sure that as usual you gentlemen will have questions and we'll do our best to respond to those questions.

We'll start out with these policy, technical and licensing issues related to evolutionary advanced LWR designs and Charlie Wylie is the subcommittee chairman in that area.

MR. WYLIE: Well, the Committee has reviewed the staff's proposed resolution of the policy and technical issues related to both evolutionary and passive advanced light water reactor designs which were identified in SECY-90-016 and SECY-93-087. We

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

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In recognition of your accomplishments, I'm pleased to present to you, if you'll come over here -- since we have a photographer here, we have to get the audio visuals right.

Ernest, we have a letter for you signed by all the Commissioners. We have a plaque for you. Thank you very, very much for all the service that you have provided.

DOCTOR WILKINS: Thank you very much, Mr. Chairman, Commissioners.

CHAIRMAN SELIN: Since you have always been free with your advice and comment, we assume that your imminent departure will not be necessary to make you even more frank with us. We look forward to this session very much.

Doctor Wilkins?

DOCTOR WILKINS: I believe we all have

copies of the agenda. There are eight items on this list. I can tell you quite frankly that I have some doubts we can get through all eight of these. But it's my understanding that this is roughly your priorities.

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We'll start out with these policy, technical and licensing issues related to evolutionary advanced LWR designs and Charlie Wylie is the subcommittee chairman in that area.

MR. WYLIE: Well, the Committee has reviewed the staff's proposed resolution of the policy and technical issues related to both evolutionary and passive advanced light water reactor designs which were identified in SECY-90-016 and SECY-93-087. We prepared seven letters in that review to the Commission and the EDO.

Our last report was to the Commission dated November 10 of last year covering eight issues related to the passive plant designs. We were in general agreement with the staff's proposal, but we offered specific comments on three of those issues, regulatory treatment of non-safety systems, definition of passive failures and the reliability assurance program. In all, we reviewed about 54 issues which, as I said, we had general agreement, but we did offer comments and concerns on some.

So, with that, we're open, I think, for questions.

CHAIRMAN SELIN: Well, the general impression that I have is that the staff has been most responsive to your observations.

DOCTOR WYLIE: I think so.

CHAIRMAN SELIN: Having had the interchange before that, are you satisfied with the responses and the actions that have been taken?

DOCTOR WYLIE: In general, I think that's true. We have commented on those. Well, we reviewed them and we commented where we did not. But I think in general we did.

DOCTOR WILKINS: Yes. As you know, we got a response from the EDO --

CHAIRMAN SELIN: Right.

DOCTOR WILKINS: -- that addresses the concerns that we expressed in these letters and I don't believe there's been any case where the Committee has said that the EDO's response was inadequate or not responsive or something of that sort. I'd need to look around and make sure that my colleagues agree with me on that conclusion.

DOCTOR WYLIE: I think that's true. I think we still had some concerns that we offered suggestions and offered concerns that we had in some areas.

DOCTOR WILKINS: Yes. In fact, among the EDO's possible responses is yes and we'll look into that. That's all we can ask him to do.

CHAIRMAN SELIN: When I read that letter, it occurred to me that it was within the capability of the staff to respond to these. Sometimes you make a recommendation, you'll think smarter, jump higher, run faster. People say, "We'll try," but it's not so easy to see how to do those. But there wasn't, I don't believe, among your recommendations there were any in this particular case.

Mr. Carroll?

MR. CARROLL: I think one exception to that might be that we're still not in agreement with the staff on the approach they're taking on reliability assurance programs. That was the subject of a letter we wrote last month.

DOCTOR WILKINS: Is that related to the fourth item on there?

MR. CARROLL: Yes.

DOCTOR WILKINS: So, we may get to that a little later.

MR. CARROLL: Which has now become the fifth.

CHAIRMAN SELIN: Commissioner Rogers?

COMMISSIONER ROGERS: Well, I'm not sure I'm quite in sync with your order of discussion here, but let me bring up something now that relates to your December 23rd. I have a number of things I'd like to hear a little bit more about and some of your comments. In your December 23rd letter, you expressed disappointment in the limited technical basis provided for several of the requirements relating to severe accidents. I wasn't clear on what kind of documents you were talking about where you found that shortcoming. Was it in the FSER or the URD that you were referring to there?

DOCTOR WILKINS: The December 23rd letter refers to the URD.

DOCTOR WYLIE: That's right.

COMMISSIONER ROGERS: URD?

DOCTOR WILKINS: URD, yes. And that is, in fact, the next item on the agenda.

COMMISSIONER ROGERS: Okay.

DOCTOR WILKINS: Although Charlie Wylie is the guy in charge of that too, so there's no problem if you want to talk about it right now anyway.

COMMISSIONER ROGERS: Well, I may be a little out of sync with your presentation here, but that was the question. Do you have any thoughts about what the nature of the difficulty is there? For example, is more research needed? Is more of an analysis needed? You were questioning the technical basis for the requirements, but what do you think is the solution? What's lacking?

DOCTOR WYLIE: Well, Doctor Catton had the lead on this item. I think it would be best to --

DOCTOR CATTON: Okay. I think it was a little of both. An example is the base mat penetration. Early on, EPRI produced a report and the report was just inadequate to justify the .02 square meters per megawatt thermal, yet it has sort of become the standard. It's not all that difficult to attempt to come to a number where you can be sure that you can quench it by pouring water on the top of it. The Germans have done this and they came up with the number .04 something.

COMMISSIONER ROGERS: Yes.

DOCTOR CATTON: It's not clear that it has to be .04, but they found a bound that you can believe, where EPRI didn't do that.

The basis for their arguing a particular hydrogen concentration was equally weak. It's not that any of the plants haven't done something to accommodate these things like GE and the ABWR is arguing that it can take up to ten days to penetrate the base mat. That sort of is not the same thing as saying .02 and it's coolable.

So, the disappointment is really in that I don't think they really addressed it with the proper kind of intensity. It's not that we don't know how, it's just that they didn't do it, or at least that we don't know about it.

COMMISSIONER ROGERS: But you don't think it's a question of more research needed necessarily or maybe not.

DOCTOR CATTON: Well, I don't need anymore research to come to a number that I can feel comfortable with.

COMMISSIONER ROGERS: Yes.

DOCTOR CATTON: You can calculate this .04 something. But if you want to argue for the .02, you have to do something and the MACE test had been a failure. Now, it's my understanding that the German's new design, and you were there, Tom. I believe they said it was .04 something. This is a number that you can depend on.

COMMISSIONER ROGERS: Okay.

DOCTOR CATTON: Does that help?

COMMISSIONER ROGERS: Yes. I just want to get the flavor of what your concern was and I think I've got that now.

CHAIRMAN SELIN: Your other question?

COMMISSIONER ROGERS: Well, I think I'll wait until I see whether I can ask it at the right time.

CHAIRMAN SELIN: You're going to set a standard.

DOCTOR CATTON: Right. Just ask anyway.

CHAIRMAN SELIN: Why don't you just continue?

COMMISSIONER ROGERS: Well, I've got a bunch of little questions that relate to various parts of some of your letters. For example, let's say, your letter of January 14th, '94. You were quite pleased with the human factor engineering acceptance criteria done by the staff, but you raise some questions about the I&C relationship. The question is really what's involved in those two different areas. I think on page 35 of the packet, which is page 4 of the January 14th letter, you say the staff has not yet formulated an identifiable set of criteria which must be met by digital I&C systems. So, how is it that the human factors criteria are acceptable and in the absence of I&C systems criteria? Aren't they somewhat linked together? They're not exactly the same thing obviously, but they have been coupled together in many considerations.

DOCTOR CATTON: I bet Hal could address that question.

DOCTOR WILKINS: We have a problem here. Hal is the I&C guy and Jay is the human factors guy. Now, I'm not sure which of them actually wrote this or was the original author of this paragraph.

DOCTOR LEWIS: Jay was.

DOCTOR WILKINS: Even though it's a

Committee letter. So, the fact that one of them authored it is not really critical.

MR. CARROLL: I wrote both sections.

COMMISSIONER ROGERS: Well, I don't really care who authored it. I wonder what --

MR. CARROLL: The short answer is in the control room design area I think the human factors people working with industry have come up with a model of what they want in order to do a design certification. In talking to both GE and Combustion, I think they agree that this is a very good way to handle this so-called DAC item.

By contrast, when we're talking about the hardware and software of our reactor control and protection system, although the word "menagerie" comes from Bill Kerr, not me, I think it's a very proper description. We really don't have anything that ties it all together that tells the vendor what is expected of him, what's going to be acceptable. I think that's something that needs some very high priority attention and that's why we said what we did.

COMMISSIONER ROGERS: Do you think you're going to get that from the staff? Are there any -- any of the dialogue that's taking place indicating that they understand what your concern is here? Well, I guess they understand it. The question is do they think they can do anything about it.

MR. CARROLL: Well, I think they are, for example, closing in on backfits for operating plants and coming up with something that's going to help them and the industry do meaningful reviews. I think they've indicated that they're going to try to put together a standard review plan for digital control and protection. I haven't seen much evidence of it yet.

DOCTOR LEWIS: I think maybe I will add a word or two. I tried otherwise, but I didn't succeed.

All of this depends a great deal on faith. I think that I'm not exercised about the human factors things simply because to the extent that there is any art or science out there I think NRC is reasonably on to it and I don't see any glaring holes. The converse is true of the digital stuff. In the case of the digital stuff, the reason I personally was happy to sign off on this paragraph was that I have some confidence in the industry people who are doing the design for the plant whereas I have limited confidence in the staff ability to review it.

It's a hard call and I noticed in a recent letter from the Research Review Committee to Eric

Beckjord that that committee was really extremely explicit in saying that the staff simply doesn't have the technical competence, executive or technical, in the area that it needs to handle these problems. That's something we got balled out for by one of your senior staff members.

DOCTOR WILKINS: Yes. I don't know whether I ought to make sure that the Commission understands that Doctor Lewis is expressing his personal opinions.

DOCTOR LEWIS: I think they always know that.

DOCTOR WILKINS: The Committee has not reached that conclusion.

DOCTOR LEWIS: But anyway, I was willing to sign off for that reason. I have some confidence in GE. I would have preferred, as it says here, if they had done the job from the bottom up instead of from the top down, but it didn't work out that way.

DOCTOR WILKINS: Commissioner Rogers, on page 2 of this letter, which is 33 in the letters, we say that it, and that's the document, this Chapter 18 of the FSER and so on, it also specifies the acceptance criteria by which the staff will evaluate the HFE program elements. On page 4 we say the staff has not yet formulated an identifiable set of criteria and I think that was the point that we wanted to have come through.

COMMISSIONER ROGERS: Yes. Okay.

DOCTOR WILKINS: You may well say, "Well, those criteria for HFE aren't all that great." That's an issue one can debate.

COMMISSIONER ROGERS: But it's a different area, but it's done.

DOCTOR WILKINS: But it's done, yes.

MR. CARROLL: I'd like to add a little follow-up to Hal. Hal mentioned he has confidence in GE and Combustion Engineering and I just wanted to say that we have gotten very well acquainted with the people doing the I&C design from both of those companies in the course of our review and I share his feeling that they have some pretty talented people working on this stuff.

DOCTOR LEWIS: Yes. And isn't it a pleasure to be able to say something nice about somebody?

MR. CARROLL: I always do.

CHAIRMAN SELIN: Well, moving right along.

COMMISSIONER ROGERS: That's fine.

COMMISSIONER REMICK: No question on the

first item.

CHAIRMAN SELIN: Thank you, Mr. Wylie.

DOCTOR WILKINS: Then we'll go back to Mr. Wylie on the utility requirements document for the passive plant designs. I suspect that we've already covered some of this.

DOCTOR WYLIE: Yes, I think so. We introduced that and basically we wrote a fairly favorable report. We were in agreement with the URD and the staff's review, except we did express some concerns again regarding the things we've talked about already, the severe accident issues on hydrogen control and core melt spreadability, coolability, steam explosions, the explosions and also lack of design criteria for containment to withstand severe accidents. So, that's basically all I intend to say about that.

CHAIRMAN SELIN: The point I didn't follow in Doctor Catton's remarks, is the gist of your remarks, Doctor Catton, that in spite of the fact that the example you gave, and perhaps others, have not been supported, they have been approved as part of the utility requirements document and therefore would be accepted by reference in the other certifications and that they should be examined more carefully even now?

DOCTOR CATTON: I'm not sure I quite followed what you're --

CHAIRMAN SELIN: You were saying that there wasn't any basis for these numbers.

DOCTOR CATTON: That's correct.

CHAIRMAN SELIN: But the numbers have become partly accepted.

DOCTOR CATTON: But yet both --

CHAIRMAN SELIN: Does that mean that in the regulatory process we have approved numbers without a basis for them?

DOCTOR CATTON: Well, in a way, yes. What it winds up is you come down and you make a judgment and the judgment really is your own feelings rather than being able to lay something out and walk through it and come to a conclusion that is transparent.

CHAIRMAN SELIN: Doctor Kress?

DOCTOR KRESS: I think if you read the staff's resolution of that particular item, they say that they don't really accept it but they will examine it on a case by case basis.

DOCTOR WILKINS: I think that's right. The utility requirements document is not an NRC requirements document.

MR. CARROLL: That's right.

DOCTOR WILKINS: And the staff has consistently and I think properly taken the position that where they can make sure of the utilities requirement document to support conclusions, fine. But where they can't, do something else.

CHAIRMAN SELIN: And the particular examples that Doctor Catton --

DOCTOR CATTON: GE and the ten days to penetration and I guess Combustion Engineering for dealing with the steam explosions has really beefed up the cavity and they can make arguments that we don't care.

CHAIRMAN SELIN: Okay. Fine. Thank you. That's a reassuring and quite clear answer.

DOCTOR WILKINS: All right. If there are no further questions in these areas, let's move to the ABWR reactor review and that will be followed by the CE System 80+ report. I've asked the subcommittee chairmen, or they've agreed at least, to talk to you about what has been happening. Then when they've both finished, we want to raise some questions concerning schedules because they are impacting each other.

Carl?

CHAIRMAN SELIN: Thank you.

Mr. Michelson?

MR. MICHELSON: Okay. Thank you.

We last discussed this schedule, I think, in September of last year. It's been awhile. Since that time in late September our Severe Accident Subcommittee held a three day meeting in which we poured over very carefully the whole severe accident picture for the ABWR, and I think did a fine job on that. Subsequently we've had four subcommittee meetings with ABWR in which we've tried to work through the last of the amendments and the safety evaluation reports and so forth. The last meeting in that series finished up on the 26th of January.

Now, on December 15th, we wrote you a letter in which we pointed out what we felt the schedule would be and also pointed out there may very well be a potential delay in the schedule because material simply wasn't coming in. In that letter we pointed out to you that we needed to see two important documents, the draft final safety evaluation report from the staff and the final Amendment 34 to the standard safety analysis report from GE.

We have received just in the last few days now the -- before that we had received the draft final FSER. Since that time we have received mark-up copies of pages to that draft to cover up through the

Amendment 34. We have not though seen Amendment 34 from GE. The GE has been negotiating these with the staff and their SER reflects the negotiation, but we have not seen --

CHAIRMAN SELIN: Mr. Michelson, I checked into that when I heard about it and I'm a little confused by the process, I have to say. There seems to be a chicken and egg aspect to this where GE and the staff may be waiting for your final deliberations on the other points since 34 is a wrap-up amendment. Can you explain this to me?

MR. MICHELSON: Yes.

CHAIRMAN SELIN: I'm confused at this point.

MR. MICHELSON: What happened is since September, of course, we've had quite a number of meetings in which we brought up issues and GE explained how they're going to handle them. In many cases they've even given us written material describing how they will handle them. We have not seen that material yet reflected into an amendment to the SSAR that we can read and look at and say, "Yes, that's final and we agree."

CHAIRMAN SELIN: So, from your point of view, it's very clear that the vendor has enough information of the ACRS position on all of the issues to make a --

MR. MICHELSON: Oh, yes, yes, I think. And I don't think there's really any disagreement on any of the issues, we just haven't seen the product.

CHAIRMAN SELIN: I see. Okay.

MR. MICHELSON: And we think we know what the product will contain, but we certainly haven't seen it. Now, when we see the product, we'll simply verify yes, that's what you told us, and we're done.

CHAIRMAN SELIN: Okay.

COMMISSIONER REMICK: Carl --

MR. MICHELSON: And that's Amendment 34.

COMMISSIONER REMICK: -- are these the open issues in the FSER?

MR. MICHELSON: No, no. No. Most of these are ACRS issues that have been brought up for which we have resolutions but have not seen the documentation.

CHAIRMAN SELIN: It's a question of documenting -- okay.

MR. MICHELSON: Now, it's important to see the documentation because in many cases the resolutions in some cases are verbal. In some cases they're slides that show how they're going to do it.

In some cases they're rather detailed written explanations but not to the ones you'll see in the SSAR.

CHAIRMAN SELIN: We'll follow-up on this after this meeting. I didn't understand that properly.

MR. MICHELSON: So that's Amendment 34. Now, the thing that's bothering us a little bit is that it's reported that Amendment 34 is going to be 6,000 pages. A great deal of that, I'm sure, has nothing to do with any of our issues, but our issues are buried like needles in the haystack. When we do receive this document, we have to somehow find our needles and see how they handle them amongst 6,000 pages of problems with errors and inconsistencies and so forth that have been corrected. A lot of it is editorial. They've also gone to the scientific units for the metric system.

All of this has got jumbled together into what apparently is a massive amendment that we never counted on to review and furthermore haven't seen it yet either. We have though seen the final mark-up pages on the FSER. We haven't had time to look at them yet since we just got them a day or so ago. But in principle it looks like yes, I can just take that page and see what changes they made, read them. They're generally quite legible and I think we can act on that, but we have not seen anything comparable to that for the Amendment 34. Have not anything, marked up or not.

CHAIRMAN SELIN: We'll follow-up on that this afternoon. Thank you.

MR. MICHELSON: Now, we originally were promised Amendment 34 in mid-February and that was the basis for our letter of December 15th. We pointed out that yes, we could do that and we could have a subcommittee meeting on March 9th, get it all cleaned up and start writing our letter. Now, what happened is we never got the amendment. We had to cancel the March 9th meeting because we didn't have any agenda for it without an amendment and that's where we're at at the moment.

It now appears that Amendment 34 is going to be here at the end of the month, whatever exactly that means. The end of the month is, of course, just the few days before full committee in early April. When and how we will look through 6,000 pages and draw final conclusions and so forth is not altogether clear.

What we have done though is with the best of our judgment we've tried to put together a final report in which we have assumed everything will come out like we think it will and we're working on that. We have a draft copy of that. The Committee is going to start working on it late this afternoon. But we certainly cannot issue it because it's a check without an amount fill in yet.

CHAIRMAN SELIN: That's very gracious and flexible of the Committee to do this.

MR. MICHELSON: But we were trying to get it ready so that we could move. But now when we get the final amendment so we can clear our report out is uncertain to us.

CHAIRMAN SELIN: That's all that the Commission can ask of you.

MR. MICHELSON: That's about all I have the moment.

CHAIRMAN SELIN: Thank you.

DOCTOR WILKINS: I don't know how relevant it is, but let me just say this anyway. It is unfortunate that GE held up Amendment 34 in order to revise the entire document because of the units. We would have been, I think, perfectly content to say to you, "We understand that they're going to change the units and the staff can check that they've done it, but it has no significant impact on our responsibility to advise you on the safety of this."

MR. MICHELSON: Amendment 34 should have been a technical amendment. Thirty-five should have been the editorial amendment which we wouldn't need to see and could have been issued much later.

DOCTOR WILKINS: But that's water under the bridge.

COMMISSIONER REMICK: What about the open items then? What's the schedule for that?

MR. MICHELSON: As I understand it, the staff told us today, in fact, there's only one open item remaining and it deals with design control and it will be cleared up before our April meeting. So, we will have the final reading on it.

DOCTOR WILKINS: We were handed a package this morning -- maybe the ACRS staff had it yesterday, but the members got it this morning -- which described the resolution of 13 out of 14 or 12 of 13, I've forgotten the exact number, something like that.

COMMISSIONER REMICK: I think there are still 13 open.

DOCTOR WILKINS: Well, if there were 13 on the original list, then 12 of them have been resolved.

COMMISSIONER REMICK: Yes. There were 14, but one of them the staff indicates that's the vessel level, no longer an issue.

MR. MICHELSON: That's been resolved.

COMMISSIONER REMICK: So, there are 13 left.

DOCTOR WILKINS: And they've taken care of 12 of those.

COMMISSIONER REMICK: Taken care of 12.

DOCTOR WILKINS: We have that as of today.

COMMISSIONER REMICK: But you have not looked at them.

MR. MICHELSON: I have the mark-ups which will reflect the resolutions. I just haven't had time to look at them. The members all got it and by April we'll have looked at it.

MR. CARROLL: Those were not necessarily our issues.

DOCTOR WILKINS: No.

COMMISSIONER REMICK: No, I understand, but I assumed you would look at them.

DOCTOR WILKINS: Yes.

COMMISSIONER REMICK: The resolution, yes. How about the -- has the issue of the HVAC been resolved?

MR. MICHELSON: Oh, yes. That's all taken care of now. It turns out there's a bigger problem for which that's just a tail on the dog.

COMMISSIONER REMICK: Have you looked at the --

CHAIRMAN SELIN: Don't let him get off the hook

COMMISSIONER REMICK: All right. Go ahead.

CHAIRMAN SELIN: Do you care to expand on that?

COMMISSIONER ROGERS: That's an interesting way to take care of a problem.

MR. MICHELSON: Yes. What we were concerned with for some time on the common ventilation system was that it became the umbilical cord that tied the environments of three divisions together in case you would rupture that reactor water clean-up system. Well, it turns out -- I don't know. Maybe you haven't been briefed on it, but it turns out in making good calculations on the reactor water cleanup system that it turns out that you cannot isolate in time even with good isolation valves. The system pressurizes the entire secondary containment. So, the common environmental connection makes no difference. It goes through and opens the doors. If they won't open, they'll blow them off.

So, what they have had to do is go back and environmentally qualify everything inside of secondary containment for 15 pounds and 248 degrees fahrenheit.

CHAIRMAN SELIN: Which they have.

MR. MICHELSON: To solve the problem. So, the common ventilation system went away because it just became a trivial issue compared with what turned out to be the real issue. If you break one of those pipes in that system, you can't get a timely isolation, even if it works. They also went back and added a valve inside a containment to isolate the break after the break is over with. To isolate it to keep from losing the ECCS water because there's a very small elevation difference between the takeoff to a reactor water clean-up and the top of the core. So, they fixed that. I believe they've fixed everything.

MR. CARROLL: No, they've still got the drain too, Carl.

MR. MICHELSON: Beg pardon?

MR. CARROLL: They've still got the vessel drain.

MR. MICHELSON: Yes. That they put in separately, yes. They had to put a remote control on it. But that was an easy fix.

COMMISSIONER REMICK: Have you looked at the question of the surface area of the suction strainers on the ECCS system? Is that an issue? Apparently the staff and GE are in agreement.

MR. MICHELSON: Apparently so. We looked at it to some extent and passed on any detailed look at it, yes.

COMMISSIONER REMICK: Yes.

CHAIRMAN SELIN: Thank you very much.

MR. MICHELSON: Take care of it?

DOCTOR WILKINS: Let me ask Jay to talk about the CE System 80+ review.

MR. CARROLL: Okay. Our schedule is on page 77 of the handout. Although we have had a number of meetings before December of '93, we began our serious review of the FSER on December 8th, 1993. You can see through yesterday we've knocked off quite a bit of the material. We presently have one more meeting on April 5 and 6 to complete our review of the Combustion material. I'm hoping in April to have a draft letter probably with some holes in it still, but at least something we can get started on in April and hopefully we'll be able to get out a full committee report in the May meeting. Our commitment is to do it in June, but I think given the excellent cooperation and support we've received from Combustion Engineering on this effort, and from the NRC staff, we seem to be clicking right along. I think it is conceivable we'd have a report out in May.

CHAIRMAN SELIN: Now, the Commission is committed to each of the two applicants for several times, for awhile, that each of them has been on his own schedule. That implies that if for some reason the CE submission is perfect and the GE still has holes in it or vice versa, that the one is not supposed to impact on the other. Is that still true?

DOCTOR WILKINS: Right. That's the subject that I wish to --

CHAIRMAN SELIN: Okay.

DOCTOR WILKINS: -- put on the table after they finish talking about these other things, yes.

MR. CARROLL: Well, that's all I really had to say about the schedule. So, put it on the table.

DOCTOR WILKINS: I would like to -- it's inevitable we have a good guy and a bad guy. Jay is the good guy and says we're going to get this out in May and I'm the bad guy that says we promised you in June. I don't want to change the promise yet.

CHAIRMAN SELIN: Well, that's not what I'm concerned about.

DOCTOR WILKINS: No, I know it isn't.

CHAIRMAN SELIN: I'm concerned about if GE should happen to slip, will it impact on the CE schedule or vice versa.

DOCTOR WILKINS: There is an inevitable impact. If Carl does not receive Amendment 34 in time for the Committee to complete its letter in April, then that letter will slip to May.

CHAIRMAN SELIN: Right.

DOCTOR WILKINS: And May is when we're supposed to work on the CE letter. It just isn't physically possible for us to do both.

CHAIRMAN SELIN: Well then, you have to do the CE letter first. The fact is that we've set up a schedule with a commitment to each of these contractors and one of them keeps his commitment and the other one doesn't and the other one has to slip to the end of the line, not bump the first one. That's just simple --

DOCTOR WILKINS: That certainly is my attitude, but I think it's important for the Committee to hear that from you.

CHAIRMAN SELIN: Fair enough.

DOCTOR WILKINS: And the Commission. I believe it is entirely possible that we will not have this Amendment 34 even by the end of the month.

MR. MICHELSON: Well, it won't do us any good --

DOCTOR WILKINS: It won't do us much good if we get it on April 1st anyway, you know.

MR. MICHELSON: The middle of the month is the very latest.

DOCTOR WILKINS: In fact, when we really ought to have it is next week. That's about as late as we could get it.

CHAIRMAN SELIN: I'm certainly not trying to get into which organization is at fault and all that. But we have set up commitments that are supposed to be sideways and if one slips then the other shouldn't be bumped out of its place just through no fault of its own.

MR. MICHELSON: There is a problem of losing momentum, of course. If you drop something and wait two months to pick it up again, you kind of lose a little bit of the momentum.

DOCTOR WILKINS: I think that's inevitable and it's unfortunate.

CHAIRMAN SELIN: Obviously it would be

highly desirable if both could be done.

DOCTOR WILKINS: There are similar but perhaps slightly different impacts on resources in the NRC staff itself. The NRR people I think would very much like to get the ABWR out of the way. Then they can really focus their attention on the --

CHAIRMAN SELIN: So would I.

DOCTOR WILKINS: -- System 80. But in all honesty, we have to let you know that there is a potential problem. I think it's likely to be a real problem and I think you've given us our policy decision on priorities.

CHAIRMAN SELIN: I need to ask my colleagues if they agree with that.

COMMISSIONER ROGERS: I agree.

COMMISSIONER REMICK: That's consistent, sure.

COMMISSIONER ROGERS: I think we've said that all along.

CHAIRMAN SELIN: Yes, we have given clear and unanimous guidance on that.

DOCTOR WILKINS: Okay.

COMMISSIONER REMICK: Before we leave the System 80+, I have -- ABB-CE System 80+ is the first plant that now is using the updated source term. In their SAR they indicate that at the site boundary they will meet the EPA protection action guidelines for emergency planning. The staff in the advanced FSER indicates that although they haven't done an independent determination of that, they find the approach acceptable. I wonder in your next meeting if you plan to look at that question. I think that's an important question. Does that design for basically a severe accident scenario meet those packs? I think it's an important question.

MR. CARROLL: Okay. That does come up on the agenda at the next meeting.

COMMISSIONER REMICK: It does? Good.

MR. CARROLL: Under accident analysis.

COMMISSIONER REMICK: I would appreciate it if you looked at that.

MR. CARROLL: We will look into that.

COMMISSIONER REMICK: And I hope the staff listening therefore will be prepared to address it at your meeting.

DOCTOR WILKINS: We had a presentation just this morning on the July source term. What's the gentleman's name?

MR. CARROLL: Jay Lee.

DOCTOR WILKINS: Mr. Lee, yes. Mr. Lee.

And Congel. I guess Congel is the division director. We've just had a presentation on that and we got into some of these issues then, but they'll be more specifically addressed at the subcommittee meeting.

COMMISSIONER REMICK: Good. Thank you.

MR. CARROLL: I guess one thing I left out on our schedule on System 80+ is that on St. Patrick's Day we're going to go visit Palo Verde to see what one of these -- or at least something close to this looks like.

COMMISSIONER REMICK: System 80?

CHAIRMAN SELIN: For those of us who are bilingual, we enjoy the idea of going to the green tree on St. Patrick's Day. That's pretty good.

MR. MICHELSON: On the ABWR, there was something that came up this morning that we probably should make a short statement on and that is the staff is still quite concerned that we are somehow going to come up with some show stopper at the last minute and bullox up their process. I think the Committee tried to assure that each member has looked at his particular area and, to the best of our knowledge, we do not think there are any show stoppers, but we don't know until it's confirmed by Amendment 34 that it's been fixed. But we do not anticipate that anything has gone wrong. It's just that we don't know that it's fixed right.

MR. CARROLL: The staff did characterize the critical path as being the ACRS and we kept correcting them and saying the critical path is Amendment 34.

CHAIRMAN SELIN: Well, I suspect that there will be some very careful reconsideration of different things at the end of this meeting to see if it's possible to still make your original schedule.

MR. MICHELSON: The original schedule has already passed in mid-February. That was the promised date for our April final report.

DOCTOR WILKINS: The next issue on our agenda is entitled three issues related to 10 CFR Part 52 design certification. That is number four on the original agenda list. We've shifted it now to be number 5. Jay is again the man to talk about this one.

MR. CARROLL: Our letter of February 17th begins on page 73 and speaks for itself. Does anyone have any questions?

COMMISSIONER ROGERS: Yep, I do. I'm puzzled about your February 17th, 1994 letter and your November 10th, 1993 letter because they don't seem to be consistent.

MR. CARROLL: That's right.

DOCTOR WILKINS: Shucks, we were hoping you wouldn't notice that.

DOCTOR LEWIS: That's the first time that's ever happened.

COMMISSIONER ROGERS: What went wrong? Everything looked hunky-dory in 1993 and in 1994 not so.

MR. CARROLL: I guess I wasn't paying much attention when -- where was that? Did you split that in there, Kress?

DOCTOR KRESS: No.

MR. CARROLL: Somebody did while I was not paying much attention. But we have talked quite a bit since then with the staff about their expectations with respect to RAP and I guess that led to the position on the February 17th letter.

COMMISSIONER ROGERS: Well, yes, you had a lot more information from the staff between November 10th, '93 and -- was that what caused the discomfort, more information?

MR. CARROLL: Yes. I think both letters wonder why this RAP program is something unique and distinct and why it can't be put into the maintenance program and QA program.

COMMISSIONER ROGERS: Right. That wasn't apparent in '93. Is that it?

MR. CARROLL: No, it said that in '93, didn't it?

COMMISSIONER ROGERS: No. We're in substantial agreement with the staff proposal on the liability assurance program RAP.

MR. CARROLL: Last sentence, wherever you are.

COMMISSIONER ROGERS: Well, that's the November 10th, '93 letter that I'm looking at. That's page 6 in your --

MR. CARROLL: Page 6. I thought I looked at that this morning.

COMMISSIONER ROGERS: It does say we continue to recommend that the RAP be integrated with implementation of the maintenance rule, but the beginning sentence in the paragraph says, "We agree with the staff."

In your '94 letter you've got strong reservations have crept in.

MR. CARROLL: And that's because we talked more to the staff and --

COMMISSIONER ROGERS: Really heard what

they were thinking about.

CHAIRMAN SELIN: You're basically saying, "We still continue to think it ought to be integrated into a maintenance rule."

MR. CARROLL: Yes.

CHAIRMAN SELIN: Now that we understand better what the staff is recommending, we see that they're not agreeing with you more than the other way around.

MR. CARROLL: Yes.

DOCTOR WILKINS: May I make a --

COMMISSIONER ROGERS: Yes.

DOCTOR WILKINS: -- personal aside, Commissioner Rogers? It is a real pleasure to me to hear you ask that question.

COMMISSIONER ROGERS: Well, I'm pleased to ask it.

DOCTOR WILKINS: Because we became aware of this situation ourselves last month sometime and there was a question of whether we should write another letter which clarified it, and we decided not to write another letter. But it's always gratifying to know that the Commissioners --

COMMISSIONER ROGERS: That somebody is reading them.

DOCTOR WILKINS: -- read these damn letters.

COMMISSIONER ROGERS: Somebody is, yes.

MR. CARROLL: Well, the author of the words of the earlier letter was Pete, and Pete feels very strongly, and I do too, really, that PRA ought to be an influence in the maintenance programs and whatever, so I think perhaps the "we support" had that flavor to it. And I think we say that again in the second letter, but I guess I'm just troubled that we're creating another maintenance program.

COMMISSIONER ROGERS: Well, I think it's an important point, that we have three now --

DOCTOR WILKINS: Three, yes.

COMMISSIONER ROGERS: -- and you're concerned. I think that's very important for us to worry about.

MR. CARROLL: I also have been troubled by some of the staff FSERs. On reviewing the vendors' DRAP programs it almost sounded like people on the staff believe that, if you get a reliability number out of the PRA, then there's some sort of a handbook you could go to and page through it and find the right pump that's going to meet that reliability. I think the staff is backing off from some of that pie-in-thesky thinking, but I'll know better when I see some of the final final FSER stuff.

Similarly, they seem to be saying with the ORAP program that, given that you know what the required reliability is from a PRA, the plant operator is going to have to figure out some way to demonstrate that he's always got that reliability in his equipment, and that's just a replay of our favorite subject, emergency diesel generators. There's just no way to do that.

DOCTOR KRESS: In fairness, though, I think the staff has said they're not going to require demonstrating a specific reliability, that the maintenance program and their inspection program and associated things will be patterned in such a way that, given the historical reliability of that type of equipment, they can have a high assurance it will be maintained. I think that's a reasonable approach that can be done.

COMMISSIONER ROGERS: Okay. Yes. I was concerned about that statement, you know, that really is just what you've just said, that demonstration is clearly not feasible in that sense that we've -- like for the diesel generator problem.

CHAIRMAN SELIN: Did you want to say

anything else, Commissioner Remick?

COMMISSIONER REMICK: No.

CHAIRMAN SELIN: I'd like to comment on the other -- comment on one. On this Tier 2 Star, I take your concern. I think it's a reasonable concern, but I think what the staff is proposing is actually a sensible procedural piece, something between trying to specify something before it's knowable and just going to the other extreme that says that any analysis that the vendor or even the utility believes can be done to show no safety impact -- staff still believes that there's a subset of those analyses they'd like to see rather than just respond to them.

I don't think the principle is a problem. I read yours as a cautionary note saying "don't abuse this."

MR. CARROLL: Exactly.

CHAIRMAN SELIN: And provided they not abuse this, you don't object to the process itself. Is that correct?

MR. CARROLL: Well, it looks like since we wrote this they've taken a somewhat new tack. They've, at least in some of the Combustion stuff I've been looking at recently, they have said "by definition, any change is an unreviewed safety question," and I don't think legally that's a very good thing to say.

CHAIRMAN SELIN: Any change across the board or in these --

MR. CARROLL: No, on a particular. And if you're going to change a functional requirement in the software, by definition that's an unreviewed safety question. I don't think that's a very good legal position to take, because that sounds like something I could ask for a hearing on. So, it would be better to say "we would require prior staff approval" or something.

CHAIRMAN SELIN: Okay.

As far as the outside power source -- can we go on to that one?

MR. CARROLL: Yes.

CHAIRMAN SELIN: There I'm confused, because my understanding is that Combustion Engineering is -- which is to say that, if the plant is down and one of the diesels is down and then there's a problem with the second diesel, that they could still use the turbine generator in the -- if the plant is down on maintenance, but they're not saying that they could use this as a third train if the plant is operating and they're struck by a hurricane or some such. Am I missing something? Did I misunderstand?

MR. CARROLL: Okay. We talked to the staff and Combustion in considerable detail on this issue yesterday. In modes 4, 5, and 6, the staff is saying that they would require two of three, namely two of -- the three being the two EDGs and the gas turbine AAC, so at mid-loop operation the staff would require two of those three. Combustion has already committed in the tech specs that they will have two of those three in modes 4, 5, and 6, so the staff says "fine, if that's what you want to commit to.

This issue deals with power operation and what the present situation is is that, if I have one diesel generator out for more than 72 hours, I must shut the plant down because a single failure in the other would violate GDC 4.

I think what we're suggesting here is that, given that people were required to put in this gas turbine, they ought to get some credit for it.

CHAIRMAN SELIN: I see. I was reading it backwards.

MR. CARROLL: And that credit ought to be based on some fairly sophisticated probabilistic considerations. The gas turbine is not as good as an EDG and it is not a seismic 1 device or structure, however it's got a fairly high seismic margin of .36 g, apparently. It's not protected against tornadoes and hurricanes. It does not have the rapid start capability. It takes about two minutes to get it up to speed. But, it just seems to me they should be getting some credit for it.

Now, I was very surprised yesterday to learn that the staff is sort of saying, "Well, Combustion hasn't come and asked us for a ruling on this." And Combustion is saying, "Yes, but you know about the issue and you know we're going to come and sit down with you on it, so you better be prepared to tell us what you're going to do." And all us trouble makers on ACRS were doing here was trying to get people to move on this issue, because I think it's a very important one to the viability of this design.

CHAIRMAN SELIN: I misunderstood your position. That's quite a reasonable approach. If the system is there, then its probabilities ought to be calculated and taken into account as appropriate.

MR. CARROLL: Yes. You may want to reduce the credit you get if a tornado was circling around the plant, but ordinarily it's a very viable back-up to one of the diesels.

CHAIRMAN SELIN: I could see Doctor Lewis coming in and saying, "If a tornado warning comes in, the first action to take is to recompute the PRA."

DOCTOR LEWIS: You don't want to see the PRAs on tornadoes.

COMMISSIONER REMICK: But you're only talking about the case where one diesel is down and if you're going to exceed the 72 hours, which gives them ample time to start the gas turbine up and have it running --

MR. CARROLL: Oh, yes, prove it out, sure.

COMMISSIONER REMICK: -- unless there is a hurricane coming or something, then you might not want to take credit for it, but it seems like a reasonable request.

CHAIRMAN SELIN: Thank you for your lucid explanation. I didn't understand.

MR. CARROLL: Well, the frustrating part of this is that we started asking the staff this question back at the time we were reviewing the SP-90 plant which had exactly the same problem and nobody seems to have come to any -- come to grips with it.

DOCTOR WILKINS: Well, we'll keep on mentioning it.

MR. CARROLL: Oh, yes.

CHAIRMAN SELIN: Thank you.

Doctor Wilkins, I think we're ready to move on to the next one.

Thank you, Mr. Carroll.

DOCTOR WILKINS: Let me ask Ivan Catton to address the test programs for the passive plants, the Westinghouse AP600 and the GE SBWR.

DOCTOR CATTON: I guess under this agenda item the November 18, 1993, letter is mentioned. In that letter we reported to you on the ROSA testing and we concluded that, despite the shortcomings that we mention in the November 1993 letter, we believe that ROSA will yield useful data to support validation of the relevant computer codes. Now there's a combination of SPES II, OSU, ROSA IV, and CMT separate effects tests, and that covers a broad range of sizes and scales and so forth. Further, Westinghouse has agreed that they would present a case for the completeness of the set.

The only problem we had is that, with respect to ROSA, was that we've not really seen anything that I think really fully describes its weaknesses, but I think you can live with them.

The other area is that we haven't seen from Westinghouse, or the staff for that matter, a set of calculations that delineates the boundary conditions for the separate effects tests on the CMG. In that regard, we're going to meet with them next week and these things will be discussed.

In our letter we commented to you about the heated junction thermocouples, and in none of the plant testing are they going to use them, and we think they should. However, there has been a change in the design and I understand now that they only will have the first stage and last stage activated by heated junction thermocouples, but nevertheless we believe that they -- that if that's what you have in your plant, that's what you ought to be testing, and we don't see that being done anywhere.

Now if the evaluation of the CMT separate effects testing demonstrates that its operational envelope is properly bracketed and the scaling rationale is sound, I think the program is done.

We visited OSU in September, I guess, and we didn't write a letter at that time on it, but we're really very favorably impressed with the facility. I think it's probably one of the best facilities of its type that we've seen. It's complete. The scaling that's been done is complete and the arguments are really sound.

Now the SBWR is another story. We haven't heard from them since sometime in 1992 and at that time we had a number of concerns about what they were doing. And again what it came down to was that we didn't believe that the conditions under which they were going to do their testing were broadly enough ranged. In this case it had to do with the nitrogen concentration where they were going to initiate the testing of some of the passive elements. We felt that it should be much higher. Of course, we were promised that they would come back with arguments as to why it was satisfactory, but we've not heard from them.

We haven't heard anything about the staff SBWR program either, although as soon as they're ready we will.

Unless there are any questions --

COMMISSIONER REMICK: I have a question, Ivan. It's my impression that we allowed our thermal hydraulic expertise at some of our national laboratories on which we're strongly dependent to deteriorate in the past, recent past, but I'd like to know what your opinion is. Has that been more or less restored? Are you satisfied with the type of activity that the national labs are now able to provide in this area that we're talking about, the passive design?

DOCTOR CATTON: Well, I'm not sure I can

fully address that because we haven't looked into it.

COMMISSIONER REMICK: Okay.

DOCTOR CATTON: We had a meeting two months ago on RELAP5 and for the first time I think the RES did a very good job, but it's difficult for us to evaluate the capability at the labs. We saw the same people with one or two exceptions.

COMMISSIONER REMICK: I'm not --

DOCTOR CATTON: The same people from the national lab, with one or two exceptions.

COMMISSIONER REMICK: That were there previously, you mean?

DOCTOR CATTON: Yes. So, I'm not sure I can address your question very well.

COMMISSIONER REMICK: Okay.

DOCTOR WILKINS: Is it fai-- to say, Ivan, that at this recent meeting those people were better prepared?

DOCTOR CATTON: Oh, absolutely. There was no question. For the first time in a number of years the presentations were actually sound. They were well laid out. People were very capable of addressing any questions that we raised. The problem was there was also a complaint about how much money had to be spent in order to achieve that. What that tells me is that somehow something is wrong. They should have been ready to walk in and do that. They are taking your money.

COMMISSIONER REMICK: But you were satisfied with the presentations?

DOCTOR CATTON: Oh, no question.

COMMISSIONER REMICK: Okay.

DOCTOR CATTON: But there was this complaint about the amount of money that had to be spent to accomplish it. There's a message there.

COMMISSIONER REMICK: Well, it could be money to get them up to speed or --

DOCTOR CATTON: Well, then you have to hope that they stay up to speed.

COMMISSIONER REMICK: That's right. That's my concern.

DOCTOR WILKINS: It is as if -- and I don't claim this is a fact -- it is just as if the individuals of whom I've been speaking don't regard making these presentations or staying up to speed or being able to talk about these things at the drop of a hat as one of their jobs.

COMMISSIONER REMICK: Oh, I see.

DOCTOR WILKINS: Their job is to write programs and to execute those programs and to get answers for people that want to get answers, and so when we come along or anyone comes along and asks questions about the documentation or what is the physics in this program or anything of that sort it is a diversion from their regular duties.

COMMISSIONER REMICK: I see.

DOCTOR WILKINS: And so, it costs money because they still have to do their regular duties or what they perceive to be their regular duties, and that's an issue which I've seen from both sides of the table. I don't think there's any particular secret as to what laboratory Ivan is talking about or that in fact I was at one time employed by that laboratory a long, long time ago and I saw this same issue from the other side of the table and I used to deal with people in the NRC trying to defend the national laboratory employees.

I think the fundamental problem is that the work scope that is given to the national labs is too heavily slanted toward getting the results and not nearly enough attention paid to documenting and recording the results and putting them in shape so that when those people are gone somebody else can come along and do it. And it's my personal view that you need as much money for that activity as you do for the technical activity. Now other people would say you maybe only need a third. The fact of the matter is, when they start spending ten percent, then you start getting complaints of the sort that Ivan said.

DOCTOR CATTON: Well, I would still argue, though, that, if you're not interested in your product and you're only interested in doing the computation, then something is wrong. And I think that's where the problem is.

These codes have been around for a long time. They're basically 1976 technology as far as how things are done. If you're not interested in looking at the answers and trying to understand them, then I think you should find another national lab or do something else or get other people. You have to be interested in what the product is or else you can't be critical.

COMMISSIONER REMICK: Well, is it a question of maintenance of the expertise at such a level that they can do that or that they have to shift their focus depending on -- they have a task to do now and tomorrow they have a different task and therefore there's no continuity.

DOCTOR CATTON: I sometimes have this problem with students. They just really like to run the computer because you don't have to think too much when you do it.

CHAIRMAN SELIN: It's much more gratifying.

DOCTOR CATTON: If it's a robust computer code, you'll always get answers and you can plot them up and do all kinds of fancy things with them, but that's not where it's at. Where it's at is understanding what you've got because these codes are based on small scale experiments or pieces and they have to be extrapolated to a full sized plant. You can only do that if you think about it.

DOCTOR KRESS: I may have a conflict of interest here being a national lab employee, but I think you've touched on an issue that's much broader than just thermal hydraulics. That is maintaining the expertise at the national labs for purposes of technical assistance to NRC. It is a problem, in my observation, at practically all the national labs and it's the result of decreasing research budgets, decreasing work at the lab, particularly experimental work. Eventually people get tired of doing strictly analytical computer work, the good people do, and drift off into other areas. They're not available just like that to come back. I think it's a broader problem than just thermal hydraulics and it's an issue worth thinking about.

DOCTOR WILKINS: I'd agree with Tom on that completely.

COMMISSIONER REMICK: One other question. The Committee's conclusion on the diverse water level indication for the ABWR, does that also apply to the SBWR or have you not looked at that yet?

DOCTOR CATTON: We haven't done much with the SBWR in over a year. So, this issue came up in between.

COMMISSIONER REMICK: So, this relates only to ABWR.

DOCTOR WILKINS: It's my understanding that when we do get around to the SBWR we may want to look at that question.

DOCTOR CATTON: And I suspect my colleagues will overrule again and decide that their opinion for the ABWR will hold.

DOCTOR WILKINS: That's right. I forget we were unanimous.

But it is true, and I guess I should emphasize this fact, that we really haven't -- as a Committee, we really haven't done much with the SBWR. It has been the number fourth of this collection of four reactors and I have no idea what we're expected to do in 1995 or 1996 on this issue. I tend improperly, by the way, to lump SBWR with CANDU in the sense that these are reactors which we'll get around to one of these days.

DOCTOR CATTON: We do plan to look at the RES/SBWR integral system at Purdue. We hope to hear from GE soon. Apparently there are some things going on that have to be settled before we hear from them about their test program. We also have a little concern. They have taken a code they call TRAC G and suddenly increased its area of computation to include the containment as well and this means they have to deal with nitrogen and these codes are not very good at that. So, we plan to address this also. We have just been postponing it because there's nothing happening on the other side.

COMMISSIONER REMICK: Yes. Well, I recall when GE was in here a few weeks ago they indicated they were going to get hopping on the SBWR.

DOCTOR CATTON: Wonderful.

DOCTOR WILKINS: Until they do, we'll wait.

COMMISSIONER REMICK: We'll wait, yes.

DOCTOR WILKINS: All right. Let me move

then to item number 7, risk-based regulation and that's Hal Lewis.

DOCTOR LEWIS: It's not entirely clear to me why this is on the agenda because, as you recall, the history is that you sent an SRM to the staff just about two years ago telling them to consider the problems and feasibility of switching or moving toward risk-based regulation and especially the problems of the transition from deterministic to risk-based regulation that's absolutely probabilistic. The staff generated a paper in the fall of '92, I guess, which we reviewed and wrote you a letter in which we said that the paper was not yet ready for the big time, but that we hoped to see an improved version of it, which -- if we have seen. I don't recall having seen it. So, the problem is more or less in limbo.

We did also say in our letter that the specific issue was probably less important than the overall question of the variety of elements of the Commission that are working on things which are closely related to this and we gave you a list at that time. There's been some improvement in that situation in the sense that the PRA Working Group has come in and the staff professes to a see change in its attitude toward the subject.

CHAIRMAN SELIN: Well, more specifically, they submitted a plan to us and I was hoping we might get your reaction to that plan.

DOCTOR LEWIS: Yes. We're not going to be talking about that, I think, because we don't have a Committee position on it. But in any case, some things seem to be happening. I came here prepared to tell you that the significant item is that the dog didn't bark, that nothing much has happened in the last year and a half, but a little bit has happened. There's been slow progress.

On the other hand, I did, in looking through the papers before coming to this meeting, I noticed a very interesting thing which is a matter of one letter in words. It sometimes can be significant. Your SRM from two years ago spoke of risk-based regulations, plural. Our letter spoke of risk-based regulation, and there's a big difference between the two because we've often complained that the staff thinks of safety, of the job it has to do, namely the generation of regulations which people have to obey, whereas risk-based regulation is really a state of mind, and a single letter does make a difference. It leads me to wonder whether people are, in fact, speaking of exactly the same thing.

I was trying to think of an example for a place where one letter might make a difference and the only thing that came to me on the bus coming over was just think that there's a difference between saying I'm going to pursue a virtuous life and I'm going to pursue a virtuous wife. So, a change of one letter can make a big difference.

CHAIRMAN SELIN: I can give you a lot of other examples.

DOCTOR LEWIS: But in any case, you do know that next week, beginning of next week there will be an ANS executive conference on risk-based regulation.

CHAIRMAN SELIN: Which we are actively participating in.

I wanted to ask you a question. When you talk about risk-based regulation or risk-based analysis or performance-based analysis, which do you see as including the other one? Do you see these as synonyms or quite different or what?

DOCTOR LEWIS: Risk-based regulation and what?

CHAIRMAN SELIN: Risk-based analysis and performance-based analysis.

DOCTOR LEWIS: Well, one is a tool to be

used in the other. Risk analysis or risk assessment has to be used as part of risk-based regulation. I personally -- you asked the question personally -- think of risk-based regulation as a kind of state of mind, like living a virtuous life. That is to say that you should think of everything that you do in terms of its impact on the risk to the public. One shouldn't have to say that in this Agency. In fact, we heard this morning a discussion, just thinking of the single item that happened, of the source term work which ended with a viewgraph that said, "Of course this has very little impact on the health and safety of the public, but we think the work is extremely valuable in guiding the design of reactors in the future." And you know, that's fine and I'm all for research. No professor can be against research. But the fact is that that isn't risk-based regulation because it isn't --

So, I think of risk-based regulation entirely as a state of mind in which you always ask yourself whether what you're doing has any impact on the risk, on the health and safety of the public. In that sense, the search for regulations which can be changed from deterministic to probabilistic is only a tiny part of the game and is probably hopeless because if you look at any given regulation, it more or less has to be deterministic simply because the person who is being regulated has the right to know what it is he's got to do in order to pass. We do that with students too. We have a passing grade and some of us stick to it more rigorously than others do and it's not all good and we all know there's no difference between a B- student and a C+ student, or not much difference. Anyone can become president. In the end, you do have to make the -- well, almost anyway. You do have to make these deterministic decisions and we have recommended to you that one take the issue of bringing, if you like modern decision analysis, into the Agency seriously. That has not happened.

So, I really don't see this as something that can be done overnight, even though the staff professes a much more positive view and probably holds a more positive view toward the general concept than they did. The resources within the Agency to carry it off are not there. My constant complaint that there are very few statisticians and fewer of those are Bayesians and fewer of those have any influence on the senior staff. So, these large changes are not going to be accomplished by edict from the Commission or letters from us.

So, it's long-term and my personal view is not all that optimistic about it. But you asked what I thought, and I told you.

CHAIRMAN SELIN: I did. Thank you.

COMMISSIONER REMICK: Just like Doctor Wilkins was pleased that a Commissioner reads ACRS letters, I'm pleased an ACRS member reads SRMs.

CHAIRMAN SELIN: To prove my Bayesian credentials, I think my letters are going to start from now on, all else being equal. But I appreciate your comments. Perhaps in the future you'll have an opportunity to look at these documents as a Committee and give us your opinion.

DOCTOR WILKINS: Let me say, Commissioner Remick, that one of the first things I did when I became Chairman was to try to find out what were the outstanding SRMs that had anyplace in them the letters ACR and S, in any order.

CHAIRMAN SELIN: Doctor Wilkins, we have a fine chance of making the eighth item.

DOCTOR WILKINS: The eighth item, yes.

CHAIRMAN SELIN: So, why don't --

DOCTOR WILKINS: All right. Let me turn then to it, the Thermo-Lag fire barriers, and Doctor Catton will lead this discussion, which I think will extend somewhat beyond Thermo-Lag.

CHAIRMAN SELIN: Thank you.

DOCTOR CATTON: It could. It could. It depends on what kind of questions I'm asked. We wrote you a letter and our recommendation for the Thermo-Lag testing was a little bit more complete, I believe, than NUMARC and the staff finally settled on. The reason we did that is that there are two problems really or two questions and one is whether the particular thickness of the Thermo-Lag gives you the thermal isolation that you want. The second is if it didn't, why. Well, it was our feeling that what you're faced with is testing everything or run a few tests that are complete. They, of course, have chose not to. In order to do the complete tests, you just instrument it a little bit differently and you don't put any of the cables in that slow down the processes, so that you can more rapidly detect if you do indeed have a rupture of the Thermo-Lag.

Basically that's where we were with that.

Now, we plan to have a meeting. I believe we've scheduled it right before the full Committee meeting in June. At that time we're going to hear a little bit more, I guess, about the results of Thermo-Lag testing and rumors have it that it's kind of bleak in some cases. But we're going to address the question of fire barriers in a more general way.

CHAIRMAN SELIN: There's something that you could do that would be extremely useful. If I might extrapolate a little bit, I think it's pretty clear that there is no such thing as a three hour barrier, at least for raceways, and we've got plenty of evidence that says there's an awful long way to get there. And furthermore, the one hour barrier seems to depend very much on the quality of the installation and not just the amount of Thermo-Lag.

DOCTOR CATTON: That's right.

CHAIRMAN SELIN: And without trying to draw all of the regulatory and safety implication to this, it seems to me that two things have to be done. One is whatever testing is done has to go more towards getting some basic information that we should have had ten years ago about how different configurations react. Not do they meet regulatory standards, but just what are the characteristics of the material.

But the second, which is even more important is once you start getting into trying to describe the behavior of insulation as opposed to different thicknesses of canonical shapes, that gets pretty tough and one is faced with trying to do broad tests on insulation and go in and destructively examine the insulation of particular sites trying to figure out what to do about that, or to hold off on some of these installation tests until we get specific plans from specific sites about what they're going to do about their own configuration and see them try to configure their testing to their own particular problem.

I don't have a question to pose to you yet, but I think we, the Commission, not just the staff, are going to need some help from you in this area as we get into what are really very difficult and tricky questions of -- tricky is the wrong word. I don't mean it that way, but just subtle questions of the interaction between the sense of a regulatory approach in an area where there's a serious problem, where we know some things in the negative sense about the problem, but where we don't really know all of the basics about the material or the sensitivity to installation and the fact that you've got to solve the problem. Standing more fire watches indefinitely is clearly not a solution.

So, I think there are going to be some interesting epistimological questions about how does one go about trying to determine some of the information on insulation as we look more closely what is the regulatory problem we're trying to solve and should we take a generic approach or turn to each of the facilities.

I personally think it's just a waste of time or money to continue the set of tests specifically oriented towards determining whether three hour raceway barriers exist or are close to existing. Results are so bleak at this point that I wouldn't mind being shown wrong, but I think we have to make the going assumption that we're just very unlikely to get from here to there and let the plant sit in limbo for years while we continue to do these tests is not sound regulation.

Your advice could be very useful as we start to try to cut the cloth to fit the problem and not the other way around.

DOCTOR CATTON: Hadn't really planned to do that. We certainly could refocus a little bit. My feeling is that what's lacking is analysis combined with the right kind of experiments. I don't see that coming out of this program and I'm not surprised at the conclusion you're drawing now. But we certainly will attempt to address these questions.

DOCTOR WILKINS: You said, Ivan, that you were planning to have a subcommittee meeting.

DOCTOR CATTON: We've been fussing about most of the morning trying to get a date and I think we settled on right before the June full committee meeting. See, we have another part to this too. We're a little bit concerned about the use of water sprays on oil fires, like in the diesel room. One of our consultants has done an analysis and we wanted to bring that to bear.

CHAIRMAN SELIN: I mean that's the kind of thing I'm talking about. For instance, our rules say that if there aren't three hour barriers, and let's just say there aren't three hour barriers, you need detection and suppression systems. It's absolutely clear that suppression systems are even a benefit let alone --

DOCTOR CATTON: Well, we raised that question at a subcommittee meeting about the equivalents. We weren't able to get an answer.

CHAIRMAN SELIN: But your research on that will be helpful because we're going to have to take a broader view of what it takes to suppress the fires and what's the best way to do it, not just say, well, if you have a system that sprays water, even if it might short circuit the diesels, you need to -- it's not only drugs that have unintended interactions, it's safety systems.

But this is a very big field and I think we're starting to think this out in a more productive fashion than we've been in the last year or two. But there are very sensitive scientific and technical questions, not just engineering.

DOCTOR CATTON: Oh, I agree. The whole area of fire control is -- of course, I'm looking at it from an outsider, but it appears to me to have suffered from strict rules that they go by that were generated as far back as 1900 and I think it's time, particularly in the nuclear business, if your fire barrier doesn't work, the down side is a lot worse. So, if I use a three hour barrier in my house, that's one thing. But a three hour barrier in a nuclear plant that's to the same standard is another. We plan to look into this.

DOCTOR LEWIS: When I served in the Navy, I found the consequences of an uncontrolled fire were worse than the --

CHAIRMAN SELIN: Yes.

Commissioner Remick?

DOCTOR WILKINS: May I take one more minute of your time? I mentioned this to you yesterday, but let me make sure that the other Commissioners are aware of it. Doctor Beckjord, in his capacity as Director of the Office of Research, is making arrangements with the National Research Council to sponsor a -- to hold a workshop on digital I&C systems and what sort of policies and procedures the Agency should adopt. The Committee, at my recommendation, took the position that we did not wish to get involved in the planning of that workshop because we were concerned that we might have to tell you what we thought of the workshop. We didn't want to be -- what's the right word?

CHAIRMAN SELIN: Co-opted.

DOCTOR WILKINS: Captured. Co-opted. That's the right word, yes, co-opted.

CHAIRMAN SELIN: It's the fifth word of our modus operandi.

DOCTOR WILKINS: Yes. Doctor Beckjord did request from three of the members of the Committee as individuals, at least three, I don't know, maybe there were some others, if they had personal opinions to express and two of them did express opinions. The thrust of those opinions was that the NRC was trying to tell the other NRC, the National Research Council, in too much detail, how it was supposed to get its work done. I think, but I'm not really sure, that Eric paid a little attention to that. On the other hand, I suspect that the final work plan is still much more prescriptive than it really ought to be.

Now, the National Academy of Science are big boys and they can protect -- they don't need us to protect them from that. They can certainly take care of themselves in that regard.

CHAIRMAN SELIN: No, we appreciate that advice. But, Doctor Wilkins, I hope the Committee doesn't quite go as far as you had said.

DOCTOR WILKINS: Well, I hadn't quite finished. I think we ought to -- when the plans are finalized, I think we ought to look at them as a Committee officially and give you an opinion as to whether we think the workshop, if conducted along these lines, would accomplish the purpose that we think you intended.

CHAIRMAN SELIN: That would be highly desirable.

DOCTOR WILKINS: Yes.

CHAIRMAN SELIN: To get you involved in the process by which we get to a point is not fair and it's not efficient. But once we're done, since to be blunt about it, a large reason that this workshop is to be held is because of your position, it would certainly be appropriate for them to exactly say that, "If we do what we say we're going to do here finally, would that meet the concerns that you've stated?"

DOCTOR WILKINS: And we're prepared to do that.

CHAIRMAN SELIN: Fair enough.

DOCTOR WILKINS: We are prepared to do that.

One subissue in this is the inclusion in the work scope of activities related to human factors and the Committee has not formulated an opinion on this, so let me just tell you what I personally think and that is that that's a diversion. I'm not saying it's not important. It may be quite important, but I think it will divert the attention and resources and time and energy from the at least as important and perhaps more important issues that were raised by the Committee in the first place. That's my 60 seconds on that subject.

CHAIRMAN SELIN: Thank you very much.

Commissioner Remick?

Well, this has been a terrific meeting. I really think that, on the one hand, it's truly a bitter-sweet thing because to have such a terrific meeting at your last session as Chairman just sets a very tough standard.

DOCTOR WILKINS: All I can say is that Michael Jordan set a good example and, while I don't think I'm as good in my business as he is in his, still it's better to go out while people are asking you why you're going out than to go out when people are asking you, "What? Are you still here?"

MR. CARROLL: You're speaking of Michael Jordan's basketball career or his baseball career?

CHAIRMAN SELIN: Thank you very much.

(Whereupon, at 3:27 p.m., the above-entitled matter was adjourned.)

CERTIFICATE OF TRANSCRIBER

This is to certify that the attached events of a meeting of the United States Nuclear Regulatory Commission entitled:

TITLE OF MEETING: PERIODIC MEETING WITH THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

PLACE OF MEETING: ROCKVILLE, MARYLAND

DATE OF MEETING: MARCH 10, 1994

were transcribed by me. I further certify that said transcription is accurate and complete, to the best of my ability, and that the transcript is a true and accurate record of the foregoing events.

Carol Lynch

Reporter's name: Peter Lynch

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

March 3, 1994

MEMORANDUM TO: Samuel J. Chilk, Secretary of the Commission

FROM: John T. Larkins, Executive Director, ACRS

SUBJECT: ACRS MEETING WITH THE NRC COMMISSIONERS ON MARCH 10, 1994 - BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners on Thursday, March 10, 1994, between 2:00 and 3:30 P.M. to discuss items of mutual interest, including the following. Background material related to these matters is attached:

1. Policy, Technical, and Licensing Issues Related to Evolutionary and Advanced LWR Designs - C. Wylie (PP.3-24)
2. EPRI Utility Requirements Document for the Passive Plant Designs - C. Wylie (PP.25-29)
3. General Electric Advanced Boiling Water Reactor (GE ABWR) - C. Michelson (PP.30-71)
4. Three Issues Related To 10 CFR Part 52 Design Certification - J. Carroll (PP.72-75)
5. ABB-CE System 80+ - J. Carroll (PP.76-77)
6. AP600 and SBWR Test Programs - I. Catton (PP.78-91)
7. Risk-Based Regulation - H. Lewis (PP.92-95)
8. Thermo-Lag Fire Barriers - I. Catton (PP.96-98)

Attachments: As Stated

cc: ACRS Members
ACRS Technical Staff

ITEM 1: Policy, Technical and Licensing Issues Related To Evolutionary and Advanced LWR Designs

The Committee has discussed the resolution of technical and policy issues at various meetings beginning in early 1990. The Committee has provided seven reports to the Commission or the EDO on this matter. The Committee and the Improved Light Water Reactors Subcommittee have also discussed the regulatory treatment of non-safety systems (RTNSS) issues for the passive light water reactor designs. For RTNSS, the basic issue under review is that passive plant designs rely on passive safety systems to meet the regulatory requirements, but also include active non-safety systems as a first line of defense to reduce challenges to the passive safety systems during transient events. The Committee issued a report regarding this matter on November 10, 1993. The Committee will continue its review of additional issues as they are identified.

The following documents are attached:

- ACRS report to the Commission dated November 10, 1993. Subject: Draft Commission Paper, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs" (PP.4-7)

- ACRS report to the Commission dated April 26, 1993. Subject: SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs" (PP.8-11)

- ACRS letter to James M. Taylor (EDO) dated September 16, 1992. Subject: Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs" (PP.12-15)

- ACRS letter to James M. Taylor (EDO) dated August 17, 1992. Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements (PP.16-20)

- ACRS letter to James M. Taylor (EDO) dated May 13, 1992. Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements (PP.21-24)

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

November 10, 1993

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

Dear Chairman Selin:

SUBJECT: DRAFT COMMISSION PAPER, "POLICY AND TECHNICAL ISSUES ASSOCIATED WITH THE REGULATORY TREATMENT OF NON-SAFETY SYSTEMS IN PASSIVE PLANT DESIGNS"

During the 403rd meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1993, we reviewed the NRC staff's positions and recommendations in the subject draft Commission paper, which reflects changes resulting from public comments on an earlier draft. We reviewed this earlier draft during our 400th meeting, August 5-6, 1993. Also, our Subcommittee on Improved Light Water Reactors reviewed this matter during a meeting on August 4, 1993. During this review, we had the benefit of discussions with representatives of the NRC staff and EPRI. We also had the benefit of the documents referenced.

The basic issue under review is that passive plant designs rely on passive safety systems to meet the regulatory requirements, but also include active non-safety systems as a first line of defense to reduce challenges to the passive safety systems in the event of transients or plant upsets. As this represents a departure from the current licensing approach, the draft Commission paper is intended to develop regulatory and review guidance for the AP600 and SBWR certification submittals.

In the draft Commission paper, the staff identified eight issues that pertain to the regulatory treatment of non-safety systems (RTNSS) for passive LWRs. We are in general agreement with the staff's positions and recommendations for resolving these issues, but have the following specific comments on three particular issues.

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A. Regulatory Treatment of Non-Safety Systems

This specific issue has the same name as the general subject because it addresses an overall process for resolving the various issues. The overall process proposed by the staff would make innovative use of PRA to determine the risk significance of active non-safety systems with respect to meeting the ancillary safety goal on core-melt frequency, and a large release goal not fully defined. Reliability/availability "missions" for the active non-safety systems would be developed and regulatory oversight procedures applied that would depend on the assessed risk significance.

In general, we think the proposed RTNSS process is a bold and positive step in the direction of risk-based regulation. We recommend that the Commission approve this general process, and we encourage the staff to proceed with further development, to address some of our specific concerns, and to begin the implementation of the process. Our specific concerns are as follows:

1. The staff is still proposing the use of a "large release" frequency of $1x10^{-6}$/yr as a "safety goal guideline." Since a different segment of the staff previously recommended abandoning this concept (we think for good reason), it is disturbing to see it being resurrected here. We believe the RTNSS process would be better served by use of a conditional containment failure guideline.

2. We believe that the risk significance of the active systems (as developed from the baseline and focused PRA) will be sensitive to the reliability values assumed in the PRAs for the passive systems. We are concerned that there does not exist a sufficient data base to establish appropriate reliability values for use in the proposed process.

3. We were told that the reliability/availability "missions" for the risk-significant active non-safety systems will, in fact, be reliability values. The proposed process is vague about how the review and regulatory audit processes can determine whether or not such reliability "missions" will have been met in the design and maintained during operation. We believe that the proposed review and audit processes, reliability assurance program, and implementation of the Maintenance Rule will not provide assurance that such "missions" have been met.

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4. The document calls for generating uncertainty distributions for the PRA results. Since the only numerical goals mentioned were based on mean values, it is not clear to us how the uncertainties are to be used by the staff.

B. Definition of Passive Failure

The draft Commission paper identifies certain passive failures that could initiate accidents. Included are check valve failures, medium- or high-energy pipe failures, and valve stem or bonnet failures. We note that valve stem or bonnet failures are included as initiating failures for the passive plants. To the best of our knowledge, the staff does not postulate such failures as current licensing practice for evolutionary plants. If such a failure were postulated to occur in the outboard containment isolation valve for the reactor water cleanup system of the Advanced Boiling Water Reactor, and the postulated single active component failure results in a failure to close the inboard containment isolation valve, the final result would be an unisolated loss-of-coolant accident outside of the primary containment.

Concerning check valves, we support the staff position to redefine check valves (except for those whose proper function can be demonstrated and documented) in the passive safety systems as active components subject to the single failure consideration.

C. Reliability Assurance Program
(Issue E in the draft Commission Paper)

We are in substantial agreement with the staff proposal on the reliability assurance program (RAP). It is noted that this program represents a significant commitment of resources by the ALWR vendor and, even more, the COL applicant. The use of modern risk assessment methods in identifying the systems, structures, and components to be covered within this program, and hence the use of these resources, is an important feature of the staff approach. We continue to recommend that the RAP be integrated with implementation of the Maintenance Rule.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

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References:
1. Draft Commission Paper (Undated), from James M. Taylor, NRC Executive Director for Operations, for The Commissioners, Subject: Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs, received July 21, 1993
2. Revised Draft Commission Paper (Undated), Subject: Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs, received November 4, 1993

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

April 26, 1993

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

Dear Chairman Selin:

SUBJECT: SECY-93-087, "POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS"

During the 396th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 1993, we discussed the NRC staff positions, delineated in SECY-93-087, on policy, technical, and licensing issues pertaining to evolutionary and advanced light-water reactor designs. During this meeting, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced. We have discussed these issues during several of our previous meetings and provided comments and recommendations in the reports referenced.

We are in general agreement with the staff's positions in SECY-93-087; however, we have concerns regarding some issues and offer our comments and recommendations as follows. (The section titles and letter designations correspond to those in SECY-93-087.)

I. SECY-90-016 ISSUES

E. Fire Protection

In our April 26, 1990 report, we pointed out that redundant train separation is likely to be the most significant feature leading to reduced fire risk. We recommended that the proposed fire protection enhancements include separation of environmental control systems (i.e., separate heating, ventilating, and air conditioning (HVAC) systems for each train). The staff responded by conceding that separate HVAC arrangements may be needed, although other options may be available to the designer. The Commission endorsed the staff's response.

We remain concerned that a common normal ventilation system (such as that proposed for the ABWR) will be difficult to design to prevent the effluent from a postulated accident in one train of engineered safety features from reaching essential mitigating equipment in the other trains and

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creating conditions that exceed their environmental qualifications. Of particular concern is the capability of ventilation dampers to isolate the effects of high energy pipe ruptures in confined compartments served by the common HVAC system.

G. Hydrogen Control

The staff claims that it has sufficient basis for understanding hydrogen behavior to go forward with licensing criteria. It has not been demonstrated to us that this basis is as extensive, or applicable, as the staff believes. Further, the AP600 and ABB-CE System 80+ designs have containments that are more susceptible to significant damage from hydrogen detonation than most existing and evolutionary plants. This requires that the licensing criteria for this issue be reconsidered.

H. Core Debris Coolability

The staff has weakened the position taken in SECY-90-016 by not requiring that the core debris be adequately quenched. We believe that the present criterion for coolability, namely a cavity floor area greater than $0.02m^2/MWt$, is not soundly based. We recommend that the staff validate containment response to core-on-the-floor accident sequences by independent analyses using, for example, MELCOR, or CORCON and CONTAIN.

J. Containment Performance

We agree with the requirement that containment stresses not exceed ASME Code Service Level C for metal containments, but it is not clear how electrical penetrations through the containment should be considered. Such penetrations utilize nonmetallic electrical insulation as a portion of the containment boundary and need further consideration.

L. Equipment Survivability

We agree that passive plant design features provided only for severe accident mitigation need not be subject to the environmental qualification requirements of 10 CFR 50.45. We believe, however, that such mitigation features must be designed to provide reasonable assurance that they will operate in the severe accident environment for which they are intended and over the timespan for which they are needed.

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II. OTHER EVOLUTIONARY AND PASSIVE DESIGN ISSUES

Q. Defense Against Common-Mode Failure in Digital Instrumentation and Control Systems

The staff's second recommendation is that the vendor or applicant analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR). We recommend that the scope of this assessment include consideration of common-mode failures during all events postulated in the SAR (e.g., fire, flood, pipe rupture, and extensive loss of essential power sources) and not be restricted to those events discussed in Chapter 15, "Accident Analysis."

T. Control Room Annunciator (Alarm) Reliability

The staff's basic recommendation is that the Commission approve the position that the alarm system for ALWRs meet the applicable EPRI requirements for redundancy, independence, and separation. These requirements do not include the use of Class 1E equipment and circuits. The staff also seeks approval of an additional position that goes beyond the EPRI requirements. This position is that "alarms that are provided for manually controlled actions for which no automatic control is provided and that are required for the safety systems to accomplish their safety functions, shall meet the applicable requirements for Class 1E equipment and circuits." We believe that the staff needs to provide clarification and additional justification for this position.

Collectively, our identified issues represent a significant array of incompletely addressed concerns. We urge that they be addressed on a timely basis to ensure their early consideration by the design teams.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. SECY-93-087, dated April 2, 1993, for the Commissioners, from James M. Taylor, Executive Director for Operations, NRC, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors (ALWR) Designs

2. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations, March 18, 1993
3. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs," September 16, 1992
4. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Digital Instrumentation and Control System Reliability, September 16, 1992
5. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements, August 17, 1992
6. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements, May 13, 1992
7. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light Water Reactors Certification Issues and Their Relationship to Current Regulatory Requirements, April 26, 1990

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

September 16, 1992

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

Dear Mr. Taylor:

SUBJECT: DRAFT COMMISSION PAPER, "DESIGN CERTIFICATION AND LICENSING POLICY ISSUES PERTAINING TO PASSIVE AND EVOLUTIONARY ADVANCED LIGHT WATER REACTOR DESIGNS"

During the 389th meeting of the Advisory Committee on Reactor Safeguards, September 10-12, 1992, we reviewed the NRC staff's positions and recommendations concerning the certification issues for evolutionary and passive light water reactor designs contained in the draft Commission paper, which was forwarded to the Commission on June 25, 1992. Our Subcommittee on Improved Light Water Reactors met on September 9, 1992, to review this subject. During these meetings we had the benefit of discussions with representatives of the NRC staff and EPRI. We also had the benefit of the document referenced. We previously provided comments to you on other policy issues related to design certification in our letters of May 13, 1992 and August 17, 1992.

Our comments and recommendations on the proposed policy issues contained in the draft Commission paper are given below. Issues A, B, C, D, E, and G apply to evolutionary and passive plant designs and Issues F and H apply only to passive plant designs. The issue titles and letter designations correspond to those of the draft Commission paper.

A. <u>Defense Against Common-Mode Failures in Digital Instrumentation and Control (I&C) Systems</u>

It is our view that the thrust of the staff recommendations concerning defense against common-mode failures in digital I&C systems as underlined in Issue A of the draft Commission paper is appropriate. We agree with the staff that the applicant should be required to assess the defense in depth and diversity of the proposed designs for the events postulated in the Safety Analysis Report, and demonstrate an acceptable plant response for each. The staff proposes that the instruments, controls, and equipment required to demonstrate an acceptable response be independent of any common-mode failure mechanisms associated with the event. We view this requirement to be essential, but remain open as to the

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best approach. The staff proposes an independent set of safety-grade displays and controls in the main control room. We believe that other arrangements might be shown to be acceptable.

In a separate letter to Chairman Selin dated September 16, 1992, we have provided additional comments and advice regarding the general approach being taken by the staff in its review of digital instrumentation and control systems.

B. Analyses of External Events Beyond the Design Basis

To assist in the closure of severe accident issues, the staff recommends that (1) analyses submitted in accordance with the requirements of 10 CFR 52.47 (concerning the contents of applications for standard design certification) include an assessment of internal and external events and (2) during the design certification review, the staff should evaluate those external events that are not site dependent (e.g., fires, internal floods) and certain bounding analyses. We agree with this staff recommendation.

C. Elimination of the Operating Basis Earthquake from Seismic Design

The staff is still reviewing this issue and has expressed only an interim position. We believe the staff is taking an appropriate approach in its interim position.

D. Multiple Steam Generator Tube Ruptures (MSGTRs)

The staff is recommending that the applicant for design certification perform additional analyses to determine the AP600 response to multiple breaks of up to 5 steam generator tubes. We agree with the staff's recommendation, but believe the staff should have a better technical basis for estimating the frequency of occurrence of such multi-tube breaks.

The staff is also recommending that the applicant for design certification of a passive or evolutionary PWR assess design features necessary to mitigate the amount of containment bypass leakage that could result from MSGTRs. We agree with the staff's recommendation.

E. Probabilistic Risk Assessment (PRA) Beyond Design Certification

The staff is recommending that, throughout the duration of the combined or operating license, the PRA be revised to address significant plant modifications, operating experience, and other developments that may affect previous PRA insights.

We are convinced that it is worthwhile for a plant operator to have an up-to-date PRA and are, therefore, reluctant to recommend against this position. However, if this is to be required, the staff should more clearly specify how it intends to use the updated PRA and what is meant by keeping it current. We think such guidance is part of the overall issue of appropriate use of PRAs in regulation and would be helpful to licensees and to the staff.

F. Role of the Operator in a Passive Plant Control Room

We agree with the first part of the staff's position "that sufficient man-in-the-loop testing and evaluation be performed ... to demonstrate that functions and tasks are integrated properly into the man/machine interface design" of passive ALWR control rooms.

The second part of the staff's underlined position states "that a fully functional integrated control room prototype is necessary for passive plant control room designs to demonstrate that functions and tasks are integrated properly into the man/machine interface design." We pointed out to the staff that the non-underlined last sentence of this paragraph is inconsistent with this language in that it would permit an applicant to "demonstrate that a control room prototype of reduced scope is sufficient." We also pointed out that the non-underlined paragraph preceding the underlined paragraph states that such a prototype "would likely" be required (not would be required) to demonstrate that functions and tasks are integrated properly into the man/machine interface design. We believe that the staff should clarify its intent by reconciling these various statements.

The staff believes that operators of passive plants will be confronted with a new operating philosophy. The staff argues that "the operators of passive plants must understand the operation of 'investment protection' systems and their interfaces with the safety-related passive systems" and that they will be confronted with "new functions and tasks unlike those required for evolutionary plants" (or current plants) "due to the new approach in operational philosophy" and "the increase in automation, and the greater use of advanced technology in the passive plant designs." As a result of our discussions with the staff and EPRI, we believe that the staff may be overreacting to the "newness" of these issues. It appears to us that additional discussion of this issue among the staff and EPRI and the vendors is needed.

G. Control Room Annunciator (Alarm) Reliability

We agree with the staff's position that the alarm system for ALWRs should meet the requirements of the EPRI Utility Requirements Document.

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H. Regulatory Treatment of Nonsafety Systems

We were told that the staff is still engaged in significant on-going discussions and review of this issue and that the associated position and recommendations are subject to modification. We believe the issue is substantial and has broad implications with respect to such items as use of PRAs in regulation, safety goal implementation, and reduction of regulatory burdens, and we expect to have additional future interactions with the staff and the industry. Consequently, we are not prepared to express a position on this issue at this time.

Sincerely,

David A. Ward

David A. Ward
Chairman

Reference:

1. Draft Commission Paper dated June 25, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Review of the Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs"

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

August 17, 1992

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

Dear Mr. Taylor:

SUBJECT: ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTORS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

During the 386th, 387th, and 388th meetings of the Advisory Committee on Reactor Safeguards, June 4-5, July 9-11, and August 6-8, 1992, we discussed with representatives of the NRC staff the staff's positions, recommendations, and resolution schedules concerning the certification issues for evolutionary and passive light water reactors contained in the draft SECY paper dated February 7, 1992. This supplements our letter of May 13, 1992, and provides our comments and recommendations on some of the staff's positions for the passive light water reactors. The section titles and letter designations correspond to those in the draft SECY paper.

I. SECY-90-016 Issues (For Passive Plants)

E. Fire Protection

The NRC staff is seeking Commission approval to use the enhanced fire protection criteria previously approved for evolutionary Advanced Light Water Reactor (ALWR) plants by the Commission's Staff Requirements Memorandum (SRM) of June 26, 1990. This SRM approved the staff's position on fire protection as presented in SECY-90-016 and supplemented by the staff's April 27, 1990 response to our report on the SECY. We recommended separate Heating, Ventilating, and Air Conditioning (HVAC) systems for each division as an important step toward ensuring adequate environmental separation of safety systems. The staff agreed that consideration of smoke, heat, and fire suppressant migration may result in separate HVAC systems, but other options may be available to the designer. Our report to the Commission of April 13, 1992, on the Draft Safety Evaluation Report for the ABWR identified the adequacy of physical separation as a continuing issue for the

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ABWR, due in part to the use of a shared HVAC system for multiple trains of redundant safety systems during normal plant operation.

Our concern with shared HVAC systems is related to the need for adequate isolation of such systems during certain disruptive events (e.g., fires, floods, or pipe breaks). If the isolation is not adequate, the HVAC arrangement may become a pathway whereby effluents from the event are conducted to locations where required safe shutdown equipment is located. This is not a concern if either (1) the HVAC isolation provisions are able to withstand the event consequences (e.g., pipe whip, jet impingement, static and dynamic pressure, and elevated temperature) during and after closure with consideration of single active component failures and acceptable leakage, or (2) the safe shutdown equipment is qualified for the environmental exposure resulting from a release of the adverse environment at any credible location along the HVAC pathway such as duct openings or blowout locations.

Except for the concern with shared HVAC, we support the staff recommendation that the passive plants should be reviewed against the enhanced fire protection criteria approved in the Commission's SRM.

F. Intersystem Loss-of-Coolant-Accident

The staff's position is that designing these low-pressure fluid systems that interface the reactor coolant system (RCS) to withstand full RCS pressure (to the extent practicable) is an acceptable means for resolving this issue. For those systems that have not been designed to withstand full RCS pressure, the staff indicates that other measures will be required. We recommend approval of the proposed staff resolution, provided consideration is given to all elements of the low pressure piping system (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

G. Hydrogen Control

The staff recommends that the evolutionary LWR designs provide a system for hydrogen control that can safely accommodate hydrogen generated by the reaction of steam with 100 percent of the fuel cladding surrounding the active fuel. (Note: This is not 100 percent of the reactive metal in the core.) We support the staff's recommendation.

The staff also recommends that the system be capable of precluding uniform containment concentrations of hydrogen greater than 10 percent. We are aware of analytical work in support of the resolution of Generic Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas," that suggests the possibility of transition to detonation at average concentrations as low as 12 percent. We recommend that the staff do a similar analysis of the impact of hydrogen combustion, and possible detonation including stratification, before establishing a limit for the average hydrogen concentration. This is of particular importance to steel-shell containments.

I. High Pressure Core Melt Ejection

To cope with the possible effects of direct containment heating (DCH), the staff concludes, ". . . that ALWR design should include a depressurization system and cavity design features to contain ejected core debris."

DCH is an extremely improbable event, and we see no need to require two modes of coping with the possibility. Either depressurization or cavity design provisions alone should be adequate. Because of possible safety benefits for other events, reliable depressurization is the preferred approach.

J. Containment Performance

The staff has not yet developed an adequate technical position relating to requirements for containment performance in passive LWRs. We agree that the proposed value of 0.1 for a conditional containment-failure probability (CCFP) is reasonable but, as we stated in our letter of April 26, 1990, regarding "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," this value is defined only within the context of a family of initiating events. It should be used by the staff in the development of its requirements and not merely passed on to applicants.

The deterministic criterion proposed by the staff is not a simple alternative to the CCFP. It could be used more logically as a complement. Using ASME Code Service Level C stress limits is not unreasonable given a known loading for which the containment is to be designed. However, determination of the appropriate loading is the hard part of the problem and the suggested deterministic criterion is essentially meaningless without it. The staff states that "applicants using the deterministic approach will be required to define the challenges considered in this evaluation." The staff takes no position on what those challenges should be or how they are to be quantified. Apparently the intent is to default to a "design specific review." This approach leaves the applicant without any real guidance from the Commission on this important topic.

We acknowledge that it is a very difficult task to establish containment performance criteria but is important. We suggested what we believe to be the best approach in our letter of May 17, 1991, "Proposed Criteria to Accommodate Severe Accidents in Containment Design."

K. Dedicated Containment Vent Penetration

The staff proposes that the decision on the need for a containment vent for passive designs should not be made at this time but should wait until specific plant designs are evaluated. We believe that the Commission should make a generic judgment about the acceptability of containment vents for LWRs. This should be a part of establishing general criteria for containment design as proposed in our letter of May 17, 1991.

L. Equipment Survivability

We agree with the staff's recommendation that features provided only for severe-accident mitigation for the passive plant designs not be subject to the environmental qualification requirements of 10 CFR 50.49, quality assurance requirement of 10 CFR 50, Appendix B, and redundancy/ diversity requirements of 10 CFR 50, Appendix A.

N. In-Service Testing of Pumps and Valves

We support the staff recommendation that the special pump and valve design, testing, and inspection provisions be imposed on all safety-related pumps and valves for the passive ALWRs.

III.E - Control Room Habitability

There were several significant differences between the staff and EPRI at the time the staff drafted this policy issue. EPRI has subsequently made a proposal to modify its Utility Requirements Document to include a requirement for a passive, safety grade, control room pressurization system that would use a bottled air supply to maintain operator doses within regulatory limits for the first 72 hours following an accident. (The regulations require that operator doses be so limited for the duration of the accident.) The pressurization system proposed by EPRI would be designed to be replenished by off-site portable supplies after 72 hours if needed. Accordingly, EPRI has recommended that the staff close this issue.

We discussed this matter with the staff and EPRI during our June 4-5, 1992 meeting. The staff told us that it is currently evaluating the EPRI proposal and is not prepared to close this issue. ACRS had several comments regarding design features of the passive control room pressurization system proposed by EPRI. We believe that the staff should take these comments into account in its evaluation. We may provide additional recommendations after the staff has completed its evaluation.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:

1. Draft SECY Paper dated February 7, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements
2. SECY-90-016 dated January 12, 1990, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements
3. Memorandum dated April 27, 1990, from James M. Taylor, Executive Director for Operations, NRC, for NRC Commission, Subject: Staff Response to ACRS Conclusions Regarding Evolutionary Light Water Reactor Certification Issues

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

May 13, 1992

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

Dear Mr. Taylor:

SUBJECT: ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTORS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

During the 383rd, 384th, and 385th meetings of the Advisory Committee on Reactor Safeguards, March 5-7, April 2-4, and May 6-9, 1992, we discussed with representatives of the NRC staff the staff's positions, recommendations, and resolution schedules concerning the certification issues for evolutionary and passive light water reactors contained in the draft SECY paper dated February 7, 1992. We also had the benefit of the documents referenced. The staff requested ACRS comments on the draft SECY paper. Our comments and recommendations on some of the staff's positions are given below.

I. SECY-90-016 Issues

Item M. Elimination of Operating Basis Earthquake

Appendix A to 10 CFR Part 100 currently establishes the Operating Basis Earthquake (OBE) at a level one-half of the Safe Shutdown Earthquake (SSE). With this specification, the OBE exerts undue influence over the seismic design and requires a full spectrum analysis in addition to that of the SSE. The staff's proposal is to effectively decouple the OBE from design. We agree with the staff's recommendation.

II. Other Evolutionary and Passive Design Issues

Item A. Industry Codes and Standards

We agree with the staff's recommendation to use the newest codes and standards that have been endorsed by the NRC in its reviews of both the evolutionary and passive plant design applications, and its

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recommendation that unapproved revisions to codes and standards be reviewed on a case-by-case basis.

Item D. Leak Before Break

We agree with the staff's recommendation to extend the application of the leak-before-break approach for both evolutionary and passive advanced light water reactors.

Item E. Classification of Main Steamlines of Boiling Water Reactors (BWRs)

We agree with the staff's recommendation for resolution of the main steamline classification for both evolutionary and passive BWRs.

Item F. Tornado Design Basis

Based on a study (NUREG/CR-4661) that compiled a considerable quantity of tornado data, the staff recommends that the maximum tornado wind speed of 300 mph (compared with the present 360 mph) be used for the design-basis tornado. We agree that the best available data should be used, but caution that design-basis specifications have sometimes been established conservatively to provide margins to deal with events not specifically addressed in the design basis. We recommend that the staff's position be approved with a qualification that the staff require assurance that other potential loads that may have been previously subsumed within the tornado design basis be taken into account if necessary.

Item H. Containment Leakage Rate Testing

The staff recommends that the maximum interval between Type C leakage rate tests for both evolutionary and passive designs be increased to a 30-month interval from the 24-month interval now required in 10 CFR Part 50, Appendix J. No significant safety penalty caused by this change has been identified. We agree with the proposed staff position.

Item I. Post-Accident Sampling System (PASS)

The staff is requesting approval of changes in requirements for the PASS currently found in 10 CFR 50.35(f)(2)(viii). These requirements, and the guidance contained in Regulatory Guide 1.79 and in NUREG-0737, resulted from consideration of the TMI-2 accident.

We agree with the staff's proposal but have the following comments:

1. The requirements as contained in the above referenced regulation refer to "the reactor coolant system and containment that may contain TID-14844 source term radioactive materials" and to measurement of these and other materials. In light of source terms now considered in severe accident analysis, it is advisable to revise this obsolete description.

2. The proposal for "Elimination of the Hydrogen Analysis of Containment Atmosphere Samples" is appropriate, given that safety grade hydrogen monitoring instrumentation will be installed.

3. The Electric Power Research Institute (EPRI) proposed elimination of an existing requirement for the capability to sample the reactor coolant at operating pressure in order to measure the dissolved gas and chloride in the coolant. EPRI claims that maintaining the systems on existing plants produces significant exposure of operating personnel, and that given a severe accident, no useful information, not otherwise available, is provided by this capability. The staff proposes to retain the requirement, but to change the time after accident onset at which the capability must be available from 8 to 24 hours. During our discussion with the staff, we were unable to elicit any reason for this requirement other than that it was established following the TMI-2 accident. We cannot endorse continuation of the requirement for high pressure sampling on the basis of information available to us.

4. The staff proposes approval of a position that "would require the capability to take samples for boron and for activity measurements 8 hours and 24 hours, respectively, after the end of power operation." The intent appears appropriate, however, we suggest that it might be better to specify a time at which the information from measurements becomes avail-

able to the operator rather than the time at which samples can be taken. Further, we assume that what is required is boron concentration rather than the presence or absence of boron. Finally, we suggest that the phrase "after the end of power operation" be made more specific.

Item N. Site-Specific Probabilistic Risk Assessment

If, as concluded by the staff, enveloping analyses are practical for both seismic events and tornadoes, it is appropriate that these be part of the submittal at the time of certification. However, enveloping analyses are not as practical for other external events such as river flooding, storm surge, tsunamis, hurricanes, and volcanism. Therefore, the staff recommends that these other types of site-specific PRA information be submitted at the combined operating license (COL) stage. We agree with this recommendation but would like to hear more about how the staff proposes to deal with any unacceptable findings at the COL stage.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:

1. Draft SECY paper dated February 7, 1992, for the Commissioners, from James M. Taylor, NRC Executive Director for Operations, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements (Draft Predecisional)
2. SECY-90-016 dated January 12, 1990 for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements
3. U.S. Nuclear Regulatory Commission, NUREG/CR-4661, Subject: Tornado Climatology of the Contiguous United States, dated May 1986

ITEM 2: EPRI Utility Requirements Document for the Passive Plant Designs

The Committee and the Improved Light Water Reactors Subcommittee have been briefed on the EPRI Requirements Document for the Passive Plant Designs. In October 1993, the Committee reviewed the staff's FSER for Volume III of the EPRI Advanced LWR Utility Requirements Document (URD) for passive plants. In addition, the Improved LWR Subcommittee discussed this matter during a meeting in October 1993. Final Committee deliberations on this matter occurred in December 1993.

The following document is attached:

- ACRS report to the Commission dated December 23, 1993. Subject: Electric Power Research Institute Advanced Light Water Reactor Utility Requirements Document -- Volume III, Passive Plants (PP.26-29)

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

December 23, 1993

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

Dear Chairman Selin:

SUBJECT: ELECTRIC POWER RESEARCH INSTITUTE ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENT -- VOLUME III PASSIVE PLANTS

During the 402nd meeting of the Advisory Committee on Reactor Safeguards, October 7-8, 1993, we reviewed the staff Final Safety Evaluation Report (FSER) for Volume III of the Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) for Passive Plants. Our Subcommittee on Improved Light Water Reactors held a meeting on October 6, 1993, to review this subject. Our final deliberations on this matter occurred during our 404th meeting, December 9-11, 1993. During these meetings, we had the benefit of discussions with representatives of the NRC staff and EPRI. We also had the benefit of the documents referenced.

In the early 1980s, EPRI established the ALWR program to support the United States utility industry efforts to ensure a viable nuclear power generation option for the 1990s and beyond. The overall objective was to establish utility industry policy along with top-tier technical and operational criteria for evolutionary and passive plant designs that would facilitate standardization and combined licensing. The intent of the program was to resolve as many of the policy, technical, and licensing issues as could be identified before specific plant designs were to be submitted, or approved. The remaining specific detailed technical and operational issues were to be resolved during consideration of detailed design information on specific plant design submittals. The program was to ensure that future nuclear power plants would be safer, simpler, more robust with greater margins, more easily operated and maintained, and more certain of being constructed and licensed without delays. The approach was to use utility experience to establish design philosophy, produce design criteria and guidance to achieve the objective, and to address the policies and regulations of the NRC.

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The EPRI ALWR URD is a compendium of technical requirements for the design, construction, and performance of ALWR nuclear power plants for the 1990s and beyond. The URD consists of three volumes:

- Volume I, "ALWR Policy and Summary of Top-Tier Requirements," is a management-level synopsis of the URD, including the design objectives and philosophy, the overall physical configuration and features of a future nuclear plant design, and the steps necessary to take the proposed ALWR design criteria beyond the conceptual design state to a completed, functioning power plant.
- Volume II, "ALWR Evolutionary Plant," consists of 13 chapters and contains utility design requirements for evolutionary nuclear power plants.
- Volume III, "ALWR Passive Plant," consists of 13 chapters and contains utility design requirements for passive nuclear power plants.

We have followed the development of the EPRI ALWR program from its inception and offered suggestions regarding safety improvements on several occasions. We discussed development of the EPRI URD program and the NRC staff reviews during numerous Subcommittee and full Committee meetings. We previously presented our comments to the Commission pertaining to the FSER for Volume II by our report of August 18, 1992.

Volume III is similar to Volume II and many chapters are identical except for the features, requirements, and those policy, technical, and licensing issues unique to the passive plants. Although the Standard Review Plan (SRP) was used by the staff as guidance, the level of detail in the URD did not permit a verification of adequacy. (The SRP was written to support the review of the final safety analysis reports on specific plant designs for which a significant amount of design and construction information is normally available.) The staff conducted its review with the understanding that the EPRI design criteria would meet all current regulations, except where deviations were identified. The staff review of the URD focused on determining whether the EPRI criteria conflict with current regulatory requirements.

In addition, the staff identified a number of policy, technical, and licensing issues which needed resolution in order to complete its review of the ALWRs, including the URD. We provided comments on these issues by our referenced letters. The Commission considered the staff positions on twenty-one of the issues identified in SECY-93-087 pertaining to passive plants.

We believe that the staff has conducted a thorough and comprehensive review. We are in general agreement with the FSER pertaining to Volume III and its conclusion that meeting the URD requirements could result in a reactor design that would not conflict with regulatory guidelines, and that would be responsive to various policy statements. Nevertheless, we are disappointed in the limited technical basis provided for several of the requirements relating to severe accidents - in particular hydrogen control, melt spreading and coolability, and fuel coolant interaction (steam explosion). In addition, we believe additional consideration should have been given to general design criteria for containment to withstand severe accident loads.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

References:

1. SECY-93-087, dated April 2, 1993, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
2. SECY-92-172, dated May 12, 1992, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Final Safety Evaluation Report for Volume II of the Electric Power Research Institute's Advanced Light Water Reactor Requirements Document, including the following enclosures:
   - Draft Safety Evaluation Report for Volume I, "Program Summary of the NRC Review of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992
   - Final Safety Evaluation Report for Volume II, "NRC Review of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document for Evolutionary Plant Designs," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992
3. Electric Power Research Institute, Advanced Light Water Reactor Utility Requirements Document, Volume II, "ALWR Evolutionary Plant," Chapters 1-13 through Revision 4, dated April 1992

4. Draft Commission Paper, undated, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs
5. Staff Requirements Memorandum dated July 21, 1993, from Samuel J. Chilk, Secretary, to James M. Taylor, Executive Director for Operations, Subject: SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
6. Letter dated November 10, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Draft Commission Paper, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs"
7. Letter dated April 26, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
8. Letter dated August 18, 1992, from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Electric Power Research Institute Advanced Light Water Reactor Utility Requirements Document -- Volume II, Evolutionary Plants
9. Letter dated August 17, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, EDO, Subject: Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements
10. Letter dated May 13, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, EDO, Subject: Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements
11. Letter dated April 26, 1990, from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light-Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements

ITEM 3: General Electric Advanced Boiling Water Reactor (GE ABWR)

The ACRS Subcommittee on Advanced Boiling Water Reactors (ABWR) and other subcommittees have held 59 meetings beginning in October 1989, to discuss the NRC staff's Draft Safety Evaluation Report (DSER), the GE Standard Safety Analysis Report (SSAR) for the ABWR, and related matters. The Committee has provided four letters to the EDO and seven reports to the Commission on matters related to this review.

The ABWR Subcommittee visited the GE facility in San Jose, California on June 15 and 16, 1993. The purpose of this visit was to gather information associated with the review of the ABWR SSAR. In addition, the Subcommittee held a meeting on June 17, 1993 in San Jose, CA to continue its review of the SSAR. Since then, the ABWR Subcommittee and other Subcommittees held several meetings. The review of the staff FSER for the ABWR started with an ABWR Subcommittee meeting in October 1993 followed by other meetings in November and December 1993 and January 1994. The Subcommittees on Computers in Nuclear Power Plant Operations, Design Acceptance Criteria, Severe Accidents, and Probabilistic Risk Assessment have met to review FSER areas of special interest to them.

The version of the FSER reviewed by the ACRS covered up to Amendment 33 of the General Electric Nuclear Energy (GENE) SSAR issued on December 7, 1993. It appears likely that an additional SSAR amendment will be needed to take care of a significant number of items that were brought to the attention of GENE during and since the previous ACRS reviews of earlier versions of the SSAR. GENE is planning to issue a final amendment (No. 34). The Committee intends to complete its review of the ABWR design and issue a final report to the Commission in April 1994.

The Committee has also completed its review of the Design Acceptance Criteria (DAC) to be included in the Certified Design Material (CDM) for the ABWR design. The four subject areas addressed by DAC are: Human Factors Engineering, Radiation Protection, Piping Designs, and Instrumentation and Control. The Committee issued its report to the Commission regarding this matter on January 14, 1994.

The following documents are attached:

- ACRS report to the Commission dated January 14, 1994. Subject: Final Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design (PP.32-36)

- ACRS report to the Commission dated December 15, 1993. Subject: ACRS Review of the Advanced Boiling Water Reactor Final Safety Evaluation Report (PP.37-38)

- ACRS report to the Commission dated March 18, 1993. Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule (PP.39-40)

- ACRS report to the Commission dated October 16, 1992. Subject: Second Interim Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design (PP.41-44)

- ACRS report to the Commission dated August 12, 1992. Subject: Inspections, Tests, Analyses, and Acceptance Criteria Program for the GE ABWR Design (PP.45-48)

- ACRS letter to James M. Taylor (EDO) dated August 12, 1992. Subject: ACRS Plan For Reviewing The Application For Certification of the GE Advanced Boiling Water Reactor Design (PP.49-51)

- ACRS letter to James M. Taylor (EDO) dated April 13, 1992. Subject: Review of the Draft Safety Evaluation Reports on the GE Advanced Boiling Water Reactor Design (PP.52-60)

- ACRS report to the Commission dated August 13, 1991. Subject: Additional Comment on Schedules for Advanced Reactor Reviews (PP. 61)

- ACRS report to the Commission dated July 18, 1991. Subject: Schedules for Advanced Reactor Reviews (PP. 62)

- ACRS letter to James M. Taylor (EDO) dated July 18, 1991. Subject: Concerns Related to the General Electric Advanced Boiling Water Reactor Design (PP.63-67)

- ACRS letter to James M. Taylor (EDO) dated November 24, 1989. Subject: Module 1 of the Draft Safety Evaluation Report for the Advanced Boiling Water Reactor Design (PP.68-71)

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

January 14, 1994

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

Dear Chairman Selin:

SUBJECT: FINAL REPORT ON THE USE OF THE DESIGN ACCEPTANCE CRITERIA PROCESS IN THE CERTIFICATION OF THE GENERAL ELECTRIC NUCLEAR ENERGY ADVANCED BOILING WATER REACTOR DESIGN

During the 405th meeting of the Advisory Committee on Reactor Safeguards, January 6-7, 1994, we completed our review of the Design Acceptance Criteria (DAC) to be included in the Certified Design Material (CDM) for the General Electric Nuclear Energy (GENE) Advanced Boiling Water Reactor (ABWR). The four subject areas addressed by DAC are Human Factors Engineering, Radiation Protection, Piping Design, and Instrumentation and Control.

Our Ad Hoc Subcommittee on DAC, in a joint meeting on November 2, 1993, with the Computers in Nuclear Power Plant Operations Subcommittee, reviewed Chapter 7, "Instrumentation and Control Systems," of the GENE Standard Safety Analysis Report (SSAR), the NRC staff Final Safety Evaluation Report (FSER) for this Chapter, and the related DAC. This DAC was further discussed during our November 4-6, 1993 meeting. Our ABWR Subcommittee, during its meeting of November 17, 1993, reviewed the human factors aspects of Chapter 13, "Conduct of Operations," and Chapter 18, "Human Factors Engineering," of the GENE SSAR, the NRC staff FSER for these Chapters and the related DAC for Human Factors Engineering. The DACs on Radiation Protection and Piping Design were discussed during our December 9-11, 1993 meeting. In each of these meetings, we had the benefit of discussions with representatives of the NRC staff and GENE. We also had the benefit of the documents referenced.

In addition to the meetings described above, both ACRS and its Ad Hoc Subcommittee on DAC (which was established to review the DAC process as requested by the Commission in its April 1, 1992 Staff Requirements Memorandum) met on a number of occasions to consider the overall DAC process as it was evolving. We provided two interim reports during this period. With this report, we believe that the Ad Hoc Subcommittee on DAC has now completed its assignment.

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BACKGROUND

Since our last report, considerable effort has been expended by the NRC staff, GENE, NUMARC, and interest d industry participants in the development of the Tier 1 CDM for the ABWR. As described in the GENE CDM submittal of December 7, 1993, the Tier 1 CDM relevant to the four subject areas that use the DAC process is contained in Section 3.0 "Additional CDM." This section consists of those aspects of the certified design that do not lend themselves to the system-by-system coverage provided in Section 2.0 of the CDM for individual plant systems. Each of the four DAC CDM sections consists of a Design Description and associated Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). Certain elements of these ITAAC are designated as DAC because they describe the design process to be used in implementing the design commitments stated in the Design Description. This is in contrast to the general case in which ITAAC will be used to confirm that the as-built plant systems have the design characteristics stated in the Design Description. Both the CDM and the associated Tier 2 material constitute the complete set of requirements for the certified design.

RECOMMENDATIONS AND COMMENTS

With respect to the material in Section 3.0 "Additional CDM" covering the four subject areas historically referred to as DAC, we are generally satisfied that it provides a reasonable basis for the staff final safety determination needed to support Final Design Approval. Our comments on each of these CDM are as follows:

Section 3.1 - Human Factors Engineering (HFE)

This section imposes Tier 1 requirements on the Combined Operating License (COL) holder with respect to the implementation of the human-system interface (HSI) for certified design. All six elements of ITAAC associated with this CDM have been designated as DAC by the staff and GENE.

Our review of HSI covered Chapter 18 of the FSER and the "HFE Program Review Model and Acceptance Criteria for Evolutionary Reactors," both dated December 1993. The latter document provides the technical basis for the staff review of the HFE design process proposed for certification. It also specifies the acceptance criteria by which the staff will evaluate the HFE program elements proposed by an applicant. We commend the staff for the development of this document. It provides much needed guidance to applicants on the staff expectations with regard to HFE for evolutionary reactors.

The HSI scope is limited to the main control room and [illegible]emote shutdown system. We commented, in our report of Ju[illegible] 1992, that the scope of the DAC then under development shou[illegible]panded t include "... transmission switchyard work stations, because of the importance of offsite power to the safety of nuclear power plant operations" and "... incorporation of human factors principles in the design of local panels where instrumentation and controls important to safety are located." Although not included in this section of the CDM, we believe that these issues have been appropriately addressed elsewhere in the CDM.

Section 3.2 - Radiation Protection

This section imposes Tier 1 requirements on the COL holder with respect to the design of radiological shielding and ventilation systems. The scope of this section includes the design of these features for the Reactor Building, Turbine Building, Control Building, Service Building, and Radwaste Building. All six elements of ITAAC associated with this section have been designated as DAC by the staff and GENE.

The Design Description requires that the plant shielding design permit operators to perform required safety functions in "vital areas" of the plant under "accident conditions." The definition of "vital areas" in the Design Description differs from that in 10 CFR 73.2. We believe that other terminology should be used in this Design Description to avoid confusion with the definition used by the nuclear power plant security community.

ITAAC 3 of Table 3.2a contains the design commitment that "the plant shielding design shall permit plant personnel to perform required safety functions ... under accident conditions," and defines the accident radiation source term to be used for the shielding design. We agree that this source term is appropriate for this purpose.

Acceptance Criteria 1.a, b, and c of Table 3.2b distinguish, for purposes of ventilation system design, among "normally occupied rooms," "rooms that require infrequent access," and "rooms that seldom require access." The distinction between 1.b and 1.c is not obvious and should be more sharply drawn.

Section 3.3 - Piping Design

This section imposes Tier 1 requirements on the COL holder with respect to: (1) the design of nuclear safety-related piping systems and certain non-nuclear safety-related piping systems; (2) the analysis of the dynamic effects associated with postulated high energy pipe breaks on structures, systems, and components that are required to be functional during and following a safe shutdown earthquake; and (3) the reconciliation analysis of the as-built piping against the piping design. All three elements of this ITAAC have been designated as DAC by the staff and GENE.

The scope of this section is spelled out in the Design Description. There are, however, a number of additional aspects of piping design and analysis important to nuclear power plant safety which are not covered by this section. These have been discussed in detail with the staff and GENE on a number of occasions. We have been told that these piping design and analysis issues will be included elsewhere in the CDM. We will continue to follow this matter until we are satisfied that these issues have been properly addressed.

Section 3.4 - Instrumentation and Control

This section imposes Tier 1 requirements on the COL holder with respect to: (1) the configuration of safety-related digital instrumentation and control (I&C) equipment encompassed by the Safety System Logic and Control (SSLC); (2) the hardware and software development process used in the design, testing, and installation of I&C equipment; and (3) the diverse features included in I&C system design to provide backup support for postulated worst-case common-mode failures of SSLC. ITAAC 7 through 11 have been designated as DAC by the staff and GENE.

We would have preferred that the staff had based its review and acceptance of this section, the related Section 2.0, and SSAR Chapter 7 on a documented review model and specific acceptance criteria, as was done in the case for the Human Factors Engineering section discussed above. The staff has not yet formulated an identifiable set of criteria which must be met by digital I&C systems. In the FSER, reference is made to a menagerie of NRC regulations and regulatory guides, to a set of industry standards, and to several NRC publications which provide the basis for the staff conclusions concerning the process being followed by GENE. However, an examination of these indicates that most were developed before any significant application of digital technology to reactor safety systems, that only a few are relevant to many of the staff concerns, and that several are obsolescent if not obsolete.

We continue to recommend that the staff produce, on an expedited basis, a soundly conceived Standard Review Plan for digital I&C systems for both ALWRs and operating plant backfits.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

References:

1. GE Nuclear Energy, "ABWR Certified Design Material," Volumes 1 and 2, December 7, 1993

2. GE Nuclear Energy, "ABWR Standard Safety Analysis Report," September 1993
3. Staff Requirements Memorandum from Samuel J. Chilk, Secretary of the Commission, to David A. Ward, ACRS Chairman, dated April 1, 1992, Subject: Periodic Meeting with the Advisory Committee on Reactor Safeguards on March 5, 1992
4. NRC staff Final Safety Evaluation Report for the General Electric Nuclear Energy Advanced Boiling Water Reactor, December 1993
5. NRC staff Final Safety Evaluation Report for the General Electric Nuclear Energy Advanced Boiling Water Reactor, "HFE Program Review Model and Acceptance Criteria for Evolutionary Reactors" (Appendix 18A), December 1993
6. ACRS report dated June 16, 1992, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Interim Report on the Use of Design Acceptance Criteria in the Certification of the GE Nuclear Energy Advanced Boiling Water Reactor Design
7. ACRS report dated October 16, 1992, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Second Interim Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design

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

December 15, 1993

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

Dear Chairman Selin:

SUBJECT: ACRS REVIEW OF THE ADVANCED BOILING WATER REACTOR FINAL SAFETY EVALUATION REPORT

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, we discussed the schedule for completing our review of the NRC staff Final Safety Evaluation Report (FSER) for the General Electric Nuclear Energy (GENE) Advanced Boiling Water Reactor (ABWR) Standard Safety Analysis Report (SSAR). Previous schedules for our review of the ABWR were discussed in the referenced documents.

Our review of the FSER for the ABWR started with an ABWR Subcommittee meeting in October 1993, followed by another meeting in November. (During earlier Subcommittee meetings going back to 1989, we had reviewed ABWR/SER material.) Additional meetings are planned for December and January as advance copies of final draft material become available. Our Subcommittees on Computers in Nuclear Power Plant Operations, Design Acceptance Criteria, Severe Accidents, and Probabilistic Risk Assessment have met to review FSER areas of special interest to them.

The version of the FSER that we are reviewing is thought to cover most GENE submittals through Amendment 31 of the SSAR. This amendment was a reissuance of the complete SSAR in July 1993. Since then, GENE has issued an extensive revision as Amendment 32 and has just issued Amendment 33 on December 7, 1993. The staff intends to update its FSER through Amendment 33 during January 1994.

It appears likely to us that an additional SSAR amendment (beyond 33) will be needed to take care of a significant number of items that we have brought to the attention of GENE during and since our previous reviews of the SSAR (which were based on various earlier amendments). These items include numerous errors and inconsistencies in the SSAR and the absence of certain key information that we believe will be essential to obtaining a favorable Committee report. Some of these items were accommodated in Amendment 32.

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Items brought to the attention of GENE by late November might be covered in Amendment 33. Additional items are likely to surface during the December and January Subcommittee meetings. All of our items must be closed with a final amendment issued by mid-February, reviewed expeditiously by the NRC staff, and considered by our ABWR Subcommittee at a meeting scheduled for March 9, 1994. We intend to complete our review and issue a final report only after the FSER is revised to reflect the final amendment to the SSAR.

On this basis, our ABWR Subcommittee will prepare, for full Committee consideration in March, a draft report on those portions of the ABWR application which concern safety. Barring untimely receipt of needed information or completion of the FSER revision, we expect to issue a final report to you in April 1994.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

References:

1. SECY-93-097, dated April 14, 1993, for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Integrated Review Schedules for the Evolutionary and Advanced Light Water Reactor Projects
2. SECY-93-041, dated February 18, 1993, for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule
3. ACRS Report dated March 18, 1993, to Chairman Selin from Paul Shewmon, ACRS Chairman, Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule

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

March 18, 1993

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

Dear Chairman Selin:

SUBJECT: ADVANCED BOILING WATER REACTOR (ABWR) REVIEW SCHEDULE

During the 395th meeting of the Advisory Committee on Reactor Safeguards, March 11-12, 1993, we discussed the staff's revised estimate of the schedule (proposed in SECY-93-041) for completing its review of the ABWR design. We also had the benefit of the documents referenced.

We note that in SECY-93-041, the time proposed for our review of the Final Safety Evaluation Report (FSER) is one month. In our July 18, 1991, report to you on "Schedules for Advanced Reactor Reviews," we agreed with the staff's estimate of three months for completing our review of the FSER. It is still our view that three months will be needed to perform a meaningful review, given the proposed schedule for transmitting the information to us.

Regarding our present ABWR review status, our work on the ABWR design certification application stalled in November 1992, pending the development of additional technical information by General Electric Nuclear Energy (GE) and decisions by the NRC staff on a number of important areas such as:

- design acceptance criteria/inspections, tests, analyses and acceptance criteria, digital control systems, control room and human factor provisions, and severe accident/probabilistic risk assessment considerations

- interface requirements and representative conceptual designs for uncertified portions of the design

- technical resolution of Unresolved Safety Issues and Generic Safety Issues as required by 10 CFR 52.47

- closure of open and confirmatory items in the October 1992 draft of the FSER

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- closure of open items and concerns from the ACRS Advanced Boiling Water Reactors Subcommittee meetings of August 19, October 21, and November 18-19, 1992

Our subcommittee meetings with the NRC staff and GE were, in general, limited to consideration of the October 1992 draft of the FSER and the initial submittal and first twenty amendments (through March 13, 1992) of the ABWR Standard Safety Analysis Report (SSAR). We have not met with the staff or GE on these matters since November 1992, although we have planned a subcommittee meeting on severe accidents on March 18, 1993.

We will meet again to complete our review when the staff and GE provide us with reasonably complete final documentation for our consideration. There are now several additional voluminous amendments to the SSAR to consider, and extensive revision of the FSER is likely. From the nature of past ACRS open items and concerns on the ABWR and the uncertainty concerning their resolution, we believe that significant problems may still persist.

If it would expedite the schedule, we would be willing to meet with the staff and GE to review portions of the final FSER and associated SSAR beyond Amendment 20 as they are completed and made available. This would ensure a more timely resolution of any remaining concerns and could shorten the three months otherwise needed for our review of the advance copy of the complete FSER package (referred to in SECY-93-041) and preparation of our final report required by 10 CFR 52.53.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:
1. Letter dated February 9, 1993, from Dennis M. Crutchfield, NRR, to Paul Shewmon, Chairman, ACRS, Subject: Review Schedule for the Advanced Boiling Water Reactor (ABWR)
2. SECY-93-041, dated February 18, 1993, for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule

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

October 16, 1992

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

Dear Chairman Selin:

SUBJECT: SECOND INTERIM REPORT ON THE USE OF THE DESIGN ACCEPTANCE CRITERIA PROCESS IN THE CERTIFICATION OF THE GENERAL ELECTRIC NUCLEAR ENERGY ADVANCED BOILING WATER REACTOR DESIGN

During the 390th meeting of the Advisory Committee on Reactor Safeguards, October 8-10, 1992, we continued our deliberations regarding the use of the design acceptance criteria (DAC) process and associated inspections, tests, analyses, and acceptance criteria (ITAAC) in the certification of the General Electric Nuclear Energy (GE) Advanced Boiling Water Reactor (ABWR) design. Our Ad Hoc Subcommittee on Design Acceptance Criteria considered this matter during its October 7, 1992 meeting. This Subcommittee was established to review the DAC process as requested by the Commission in its April 1, 1992, Staff Requirements Memorandum.

During these meetings we considered SECY-92-299, dated August 27, 1992, which is a staff status report on the subject of the development of DACs for the ABWR certification in the areas of instrumentation and controls (I&C) and control room design. It was evident from our meetings that the staff's review of these DACs and preparation of the supporting draft Final Safety Evaluation Report (FSER) chapters will require extensive further work. During these meetings, we had the benefit of discussions with representatives of the NRC staff and GE. We also had the benefit of the documents referenced.

Our first interim report on the DAC process, dated June 16, 1992, focused mainly on the other two DACs proposed by GE for use in certification of the ABWR design, namely, ITAAC 3.7 "Radiation Protection" and ITAAC 3.3 "Piping Design." We concluded that these DACs (with certain clarifications to the language of the drafts we reviewed) can provide an acceptable basis for the staff's final safety determination needed for design certification. We understand that these DACs will be available in final form for completing our review as part of the FSER. The staff is unable at this time to provide a schedule for completion of the FSER.

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This interim report deals with the remaining two DACs - control room design, and instrumentation and controls. In our June 16, 1992 interim report, we indicated that these DACs had not been developed to a point where we could offer an opinion as to their acceptability. We did express concerns to the staff on several aspects of these DACs as they existed at that time. The staff has subsequently responded to these concerns.

Control Room Design DAC

Enclosure 3 of SECY-92-299 contains the DAC (i.e., ITAAC 3.6 "Human Factors Engineering") proposed by GE for the ABWR control room design (human factors aspects), a draft of the staff's FSER for Chapter 18 of the Standard Safety Analysis Report (SSAR), "Human Factors," and a Human Factors Review Model developed by the staff. The staff certification of control room design will be based on the design process described in this ITAAC. The implementation of the control room design process will be the responsibility of the combined operating license (COL) applicant or holder.

The draft FSER contains three open items in this DAC area, all involving documentation issues, that are being completed by GE and will then require the review and approval of the staff. These open items appear to be easily resolvable.

We learned at our meetings that GE had submitted a new revision of ITAAC 3.6 since the issuance of SECY-92-299. It was this new material, which had not been completely reviewed by the staff, that we reviewed. Although we had a number of suggested language clarifications, we conclude that this ITAAC (with appropriate modification) will be able to provide an acceptable basis for the staff's final safety determination needed for design certification. We will complete our review of FSER Chapter 18 and this ITAAC when these documents become available in final form.

Instrumentation and Controls (I&C) ITAAC

Enclosure 2 of SECY-92-299 contains the ITAACs proposed by GE for ABWR I&C and a draft of the staff's FSER for Chapter 7 of the SSAR, "Instrumentation and Control Systems." The staff notes that GE will not have submitted complete design information in the I&C area prior to design certification because this is an area of rapidly changing technology. GE proposes the DAC material be included in the Tier 1 design as one system ITAAC (2.75 "Multiplexing") and three generic ITAACs (3.2 "Instrument Setpoint Methodology," 3.4 "Safety System Logic and Control," and 3.5 "Software Development"). The implementation of the design process described in the Software Development ITAAC would be the responsibility of the COL applicant or holder. Our review focused on the Software Development ITAAC which describes a design process as contrasted to a design.

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The draft FSER includes five open items and 19 confirmatory items in the I&C area that are being completed by GE and will require the review and approval of the staff.

We learned at our meetings that GE had submitted a new revision of ITAAC 3.5 since the issuance of SECY-92-299. It was this new material, that had not been reviewed by the staff, that we reviewed. We had a number of suggested clarifications to the language of this ITAAC. In addition, there are certain characteristics of software which, when specified at the beginning of the development process, make later assessment far easier. We believe that the staff and GE should include this concept in the Software Development ITAAC. We conclude that this ITAAC has the potential of providing an acceptable basis for the staff's final safety determination needed for design certification. We will continue our review as more information becomes available.

Finally, we are concerned about the significant number of post-design certification activities associated with these two DACs - control room design, and instrumentation and controls. The COL applicant or holder will be responsible for carrying out these activities. This will involve extensive future negotiations with the staff. It will also have the effect of diminishing the value of certified designs and seems to us to be contrary to the spirit of 10 CFR Part 52. We believe that the argument that these DACs represent areas of rapidly changing technology is being overplayed by both the staff and GE in justifying the extent to which the DAC process is being used.

We will keep you informed as our review of the DAC process in the certification of the GE ABWR design continues.

Additional comments by ACRS member Harold W. Lewis are presented below.

Sincerely,

David A. Ward

David A. Ward
Chairman

Additional Comments by ACRS Member Harold W. Lewis

I have a reservation about the Committee letter, for the specific issue of software certification. I have already taken (Reference 4) a more relaxed position than the Committee in the general area of DACs. That position reflects my view that we are dealing with a mature industry, not at all inexperienced in the design of modern reactors, and therefore requiring a different style of regulation than may have been the case in an earlier period. The most effective role of NRC is through oversight of the safety of the industry product, rather than on certification of each detail. The DAC process lends itself to this kind of regulation, but only in areas in which the staff itself has the experience and expertise necessary to assume this more global role. I hope that the staff will not inhibit the application of modern technology through excessive specificity, as exemplified by the analog backup controversy, on which the Committee has previously commented (Reference 6).

I have a separate nagging problem with the DAC process, as it is now being implemented, one which is exacerbated in this case. The staff is negotiating with the industry not only the potential applicants' programs for compliance with the (still unclear) acceptance criteria, but also the nature of the very requirements that the applicants will later have to meet. It is important to be very circumspect about the NRC's role in this process, lest NRC independence be compromised.

References:
1. SECY-92-299, dated August 27, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design
2. Staff Requirements Memorandum M920305A dated April 1, 1992, from Samuel J. Chilk, Secretary of the Commission, for David A. Ward, Chairman, ACRS, Subject: Periodic Meeting with the Advisory Committee on Reactor Safeguards on March 5, 1992
3. GE Nuclear Energy, "Tier 1 Design Certification Material for the GE ABWR," dated June 1992
4. Report dated February 14, 1992, from David A. Ward, Chairman, ACRS, to the Hon. Ivan Selin, Chairman, NRC, Subject: Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews
5. Report dated June 16, 1992, from David A. Ward, Chairman, ACRS, to the Hon. Ivan Selin, Chairman, NRC, Subject: Interim Report on the Use of Design Acceptance Criteria in the Certification of the GE Nuclear Energy Advanced Boiling Water Reactor Design
6. Report dated September 16, 1992, from David A. Ward, Chairman, ACRS, to the Hon. Ivan Selin, Chairman, NRC, Subject: Digital Instrumentation and Control System Reliability

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

August 12, 1992

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

Dear Chairman Selin:

SUBJECT: INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA PROGRAM FOR THE GE ABWR DESIGN

During the 388th meeting of the Advisory Committee on Reactor Safeguards, August 6-8, 1992, we reviewed a sample of the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) which are being prepared by GE Nuclear Energy (GE) as a part of its application for certification of the ABWR design. This topic was also reviewed at a joint meeting of our Subcommittees on Decay Heat Removal Systems and Advanced Boiling Water Reactors on August 5, 1992. During these meetings, we had the benefit of presentations by members of the NRC staff and by representatives of GE. Our review has been in response to a request by the Commission made at our meeting with them on March 5, 1992, and confirmed in a Staff Requirements Memorandum dated April 1, 1992. We also had the benefit of the documents referenced.

ITAAC are an important part of Tier 1 submittals which the NRC requires of applicants for design certification under Part 52. They are intended to abstract from the more voluminous source, the Standard Safety Analysis Report (SSAR), the information needed by the NRC staff to make its final safety determination and to ensure that this information is agreed to at the time of design certification and verified in the completed plant. The form and content of individual ITAAC are still being developed by an iterative process between GE and the NRC staff.

There are several types of ITAAC, as described by the staff:

Systems
Generic
Interface
Design Acceptance Criteria (DAC)
Combined Operating License (COL)

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Our present review has been confined to the general program and to the first type, which includes the largest number of individual ITAAC. We were told that the entire plant design can be described in terms of about 140 systems. Of these, GE has proposed that about 85 have sufficient safety significance to be covered by individual ITAAC. These comprise the "Systems ITAAC." We have reviewed 5 of these 85 in some detail, as a means for evaluating the ITAAC process.

We intend to continue our review by investigating examples of the Generic and Interface ITAAC. We were told there are nine Generic ITAAC for the ABWR, covering subjects which apply to many or all systems, such as welding and equipment qualification requirements. We have commented on DAC in an interim report of June 16, 1992. The COL ITAAC, which will be concerned with such matters as operator training, will be developed by a COL applicant after the design certification. We would expect to review these in the future when appropriate.

We conclude from our review that the ITAAC process appears to be generally well founded and can be made to work as the staff and GE visualize. The general form and scope of the individual ITAAC we studied were satisfactory. There is, however, a problem with content of the ITAAC. Although the examples we examined were a part of what was described as the final Stage 3 GE submittal, there was a significant lack of consistency, accuracy, and completeness. We were informed by both the staff and GE that this is a problem beyond the five examples we selected for our review. Both are individually committed to major efforts to improve the quality of the content of all ITAAC.

We were told by the Director of NRR that he plans an extensive and in-depth review of the submitted ITAAC and will not recommend approval of a Final Design Approval (FDA) until the results of the review are fully satisfactory. This could mean a delay in the presently projected date for the FDA issuance. For its part, GE expressed its commitment to respond to problems indicated by the staff review and to conduct its own quality review in parallel. GE intends to ensure consistency among ITAAC and other Tier 1 and Tier 2 documents. In addition, we were told that NUMARC intends to carry out an independent review of the ABWR ITAAC. GE already has comments from utilities on the Stage 3 ITAAC. These will be incorporated into the continuing iterations between the staff and GE.

We are concerned with the structural adequacy of walls and associated penetrations within buildings housing critical systems outside of primary containment during possible fires, floods, or pipe breaks. It was not clear from the material presented to us how structural requirements for these will be verified through the

ITAAC process. We expect to pursue this matter at a future meeting.

A PRA has been performed for the ABWR design and certain conclusions about the safety of the design can be drawn from this. In performing the PRA, many assumptions were necessary about the performance reliability of components and systems. There appears to be no means by which Tier 1 requirements (e.g., ITAAC) will ensure that components and systems in the plant can be expected to have reliabilities which are consistent with those assumed in the PRA. The SSAR provides some information on this, but does not close the loop. We were told that appropriate reliability values for components and systems will be ensured through a reliability assurance program developed by a COL applicant. We believe this matter deserves more study.

In our report to you of September 10, 1991 on ITAAC, we expressed a preference for Option 3 in SECY-91-210 which would allow for completing the ITAAC after issuance of the FDA for ABWR. The staff position is that completion of the ITAAC before the FDA is essential. Given our evaluation of the current status of ABWR documentation, we agree.

We trust the above discussion and comments have been helpful. We expect to complete our review in the near future.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:
1. SECY-91-210, dated July 16, 1991, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Requirements for Design Review and Issuance of a Final Design Approval (FDA).
2. Staff Requirements Memorandum dated April 1, 1992, from Samuel J. Chilk, Secretary, for David A. Ward, ACRS, Subject: Periodic Meeting with the Advisory Committee on Reactor Safeguards on March 5, 1992.
3. Excerpts of Inspections, Tests, Analyses, and Acceptance Criteria from GE Nuclear Energy Report: "Tier 1 Design Certification Material for the GE ABWR," dated June 1992, as follows:

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- Standby Liquid Control System (2.2.4)
- Residual Heat Removal System (2.4.1)
- Reactor Building Cooling Water System (2.11.3)
- Emergency Diesel Generator System (Standby ac Power Supply - 2.12.13)
- Control Building (2.15.12)

4. Report dated September 10, 1991, from David A. Ward, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications.

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

August 12, 1992

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

Dear Mr. Taylor:

SUBJECT: ACRS PLAN FOR REVIEWING THE APPLICATION FOR CERTIFICATION OF THE GE ADVANCED BOILING WATER REACTOR DESIGN

During the 388th meeting of the Advisory Committee on Reactor Safeguards, August 6-8, 1992, we discussed our plan for reviewing the GE application for certification of the Advanced Boiling Water Reactor (ABWR) design. Our goal is to complete this review prior to the issuance of the Final Design Approval (FDA) that is scheduled for December 1992. Subject to receiving relevant information from GE and the NRC staff in a timely manner, we plan to meet this goal. Any significant delay on the part of GE and/or the NRC staff in providing necessary information to support our review will delay the completion of our review.

Our plan for review of the matters associated with the ABWR design is as follows:

I. Final Safety Evaluation Report (FSER), Certain Other Staff and GE Licensing Documents, and Remainder of the ABWR Standard Safety Analysis Report (SSAR) Submittals

NRC Staff's Schedule for Submittal of the FSER - In our April 13, 1992 letter to you regarding the ABWR Draft Safety Evaluation Report (DSER), we stated, "If we are to provide our final report on this subject in December 1992, it will be necessary that we receive a complete and final SER no later than early September 1992." Although the staff plans to issue the FSER by early September 1992, we understand that it will not be complete, and will contain a large number of open items. The staff plans to issue Supplement 1 to the FSER by late October 1992, documenting the resolution of the open items. Resolution of the remaining open items, if any, is expected to be addressed in subsequent supplements. The staff is not sure at this time whether there will be multiple supplements, or on what schedule they will be issued.

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Schedule for ACRS Review - In order to support the NRC's current schedule for issuing the FDA for the ABWR design, we plan to complete our final report to the Commission during our December 10-12, 1992 meeting. Our Subcommittee on Advanced Boiling Water Reactors has scheduled the following meetings to review the ABWR design:

August 19, 1992 - To discuss GE's and NRC staff's responses to the issues included in our April 13, 1992 letter.

September 23-24, 1992 - To start the review of the ABWR FSER, certain other GE and NRC staff licensing documents, and the remainder of the SSAR submittals.

October 21-22, 1992 - To continue the review of the FSER, other licensing documents, and the remainder of the SSAR submittals.

November 18, 1992 - To review Supplement 1 to the FSER and any residual issues.

If we are to complete our final report in December 1992, we will not be able to perform a meaningful review of the supplements issued after October 1992.

II. Design Acceptance Criteria (DAC)

NRC Staff's Schedule for Submittal of the FSER - At the end of May 1992, the staff provided its draft SER (SECY-92-196) on the DACs related to Radiation Protection and Piping Systems. The staff expects to provide its draft SER on the remaining DACs, in the areas of Man/Machine Interface and Control and Protection Systems, by early September 1992.

Schedule for ACRS Review - On June 16, 1992, we provided an interim report to the Commission that included specific comments on the Radiation Protection and Piping Systems DACs; owing to lack of detailed information, we provided only general comments on the Man/Machine Interface and Control and Protection Systems DACs. The staff plans to provide detailed information on the DACs related to Man/Machine Interface and Control and Protection Systems and updated information on the other two DACs by early September 1992. Based on this schedule, our Ad Hoc Subcommittee on Design Acceptance Criteria plans to schedule a meeting during September or early October 1992 to discuss this matter. We plan to complete a final report on these four DACs during our October 8-10, 1992 meeting.

III. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

NRC Staff's Schedule for Submittal of ITAACs - The staff has already provided ITAACs for a number of ABWR systems to the ACRS. The staff is reviewing these ITAACs and identifying areas where additional information is needed from GE.

Schedule for ACRS Review - In the April 1, 1992 Staff Requirements Memorandum (SRM), the Commission requested that we review in some detail representative ITAACs submitted by GE, and provide recommendations to the Commission by August 21, 1992. Accordingly, a Subcommittee meeting was held on August 5, 1992, to review the following ITAACs:

- Standby Liquid Control System (suggested by Commissioner Rogers during the March 5, 1992, meeting between the ACRS and the Commissioners)
- Residual Heat Removal System
- Reactor Building Cooling Water System
- Emergency Diesel Generator System (Standby ac Power Supply)
- Control Building

The full Committee discussed these ITAACs with representatives of the NRC staff and GE during its August 6-8, 1992 meeting and provided a report to the Commission dated August 12, 1992.

IV. Summary

Completion of our review of the above-mentioned items in accordance with the schedule noted above depends upon timely receipt of relevant information and appropriate support by the staff and GE. If the staff has any problem in supporting any of the meetings noted above, we would like to hear from you as soon as possible.

Sincerely,

David A. Ward

David A. Ward
Chairman

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

April 13, 1992

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

Dear Mr. Taylor:

SUBJECT: REVIEW OF THE DRAFT SAFETY EVALUATION REPORTS ON THE GE ADVANCED BOILING WATER REACTOR DESIGN

During the 383rd and 384th meetings of the Advisory Committee on Reactor Safeguards, March 5-7 and April 2-4, 1992, we discussed the Draft Safety Evaluation Reports (DSERs) on the Advanced Boiling Water Reactor (ABWR) design which is described by GE Nuclear Energy (GE) in its Standard Safety Analysis Report (SSAR), as amended, and for which GE has applied for design certification in accordance with 10 CFR Part 50, Appendix O. The DSERs which are the basis for this report were sent to the Commissioners for information as six SECY papers (SECY-91-153, 235, 294, 309, 320, and 355). These generally cover the SSAR and its first eighteen amendments. Our Subcommittee on Advanced Boiling Water Reactors discussed these papers with representatives of GE and the NRC staff during its meetings on September 18 and October 23, 1991 and January 23-24 and February 20-21, 1992. We also had the benefit of the documents referenced.

Our first report to you concerning the DSER for this project was dated November 24, 1989. That report conveyed our comments on Module 1 of the design (former GE designation). We also sent a report to you on July 18, 1991, outlining several ABWR design concerns that developed during subsequent review.

We note a marked improvement in the quality of the staff's DSER evaluations since our November 24, 1989 report. The staff reviewers appear to be following the guidance outlined in the applicable Standard Review Plans (SRPs) to the extent possible, and they are asking good in-depth questions in most areas.

The SECY-91-161 schedule indicates that the Final Design Approval (FDA) is to be issued before the end of Calendar Year 1992. If we are to provide our final report on this subject in December 1992, it will be necessary that we receive a complete and final SER no later than early September 1992. There are now more than three hundred open items in the DSERs, many of which are major. In

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addition, there is a number of important policy issues which are unresolved. With the staff programs in place, it is probable that these issues can be resolved. However, this is a large undertaking, and we have concerns about whether it can be accomplished on the schedule now indicated.

In the course of our review, we have identified technical issues for which resolutions should be achieved before we write our final report. These are listed and discussed as follows:

1. Control Building Flooding

   The proposed ABWR plant design locates the Reactor Building Cooling Water (RBCW) System at the lowest elevation in the control building, with the essential 250 V dc battery rooms and the main control room at a higher elevation, but still below ground.

   Our concern with this arrangement is the potential for control building flooding due to an unisolated break in the Reactor Service Water (RSW) System which provides cooling water from the Ultimate Heat Sink (UHS) to the RBCW System. The proposed UHS is a ground-level spray pond which we assume to be at building grade and likely to contain sufficient water to flood the control building.

   The staff should obtain sufficient information on the interface and conceptual design of the RSW System and UHS to support an adequate evaluation of the flooding potential. The staff's evaluation should include consideration of isolation valve arrangements, the feasibility of and time available for response, and the assumption of a single active component failure during the response. The design information and flooding analysis should be included in the SSAR.

2. Adequacy of Physical Separation

   Pipe breaks, internal plant flooding, and external events such as fire are of major concern if their effects cannot be confined in order to protect required safe-shutdown equipment. We believe that the key to confinement is the provision of appropriate separation barriers. However, a classical barrier such as the 3-hour-rated fire barrier wall and its penetrations (e.g., doors and dampers) may not, of itself, be sufficient to ensure separation under (a) the combined effects of pressure, heat, and smoke from a fire, and the flooding which results from fire mitigation, (b) the effects of pipe whip, jet impingement, or compartment pressurization due to pipe breaks, or (c) the influx of water and hydrostatic pressure buildup due to internal floods.

We believe that the SSAR should describe and the staff should evaluate the adequacy of proposed separation barriers for the full range of events and conditions for which separation must be ensured. We continue to recommend that systems required for safe shutdown not share a common Heating, Ventilating and Air Conditioning (HVAC) System during normal plant operation. The secondary containment HVAC System for the ABWR is such a shared system.

3. <u>Protection of Environmentally Sensitive Equipment</u>

The ABWR makes extensive use of environmentally sensitive equipment (including solid-state electronic components) for essential protection, control, and data transmission functions. Such components are known to be susceptible to adverse environmental changes, particularly temperature extremes. We are concerned that a number of these components may be located in plant areas where postulated events such as pipe breaks, fire, internal flooding, or loss of room cooling may create an adverse environment. Such environments need to be identified in the SSAR to ensure appropriate environmental qualification of the equipment.

4. <u>Review of Chilled-Water Systems</u>

The ABWR uses large chilled-water systems to provide essential environmental cooling, which in turn includes cooling of the solid-state electronic components. Because there was no SRP for chilled-water systems, the staff used other guidance such as SRP Section 9.2.2 (Reactor Auxiliary Cooling Water Systems) when the safety evaluation was performed. However, this guidance is not appropriate for the evaluation of refrigeration systems.

The NRC staff needs to evaluate the performance of chilled-water systems under varying accident heat loads and during loss-of-offsite-power events, and to consider their ability to restart and function after a prolonged station blackout. The DSER sections which should evaluate the performance of large chiller packages do not address these issues. We believe they should.

5. <u>Use of Leak-Before-Break Methodology</u>

It is our understanding that GE will not propose the use of leak-before-break methodology for the ABWR standard plant. Thus, the DSER should be revised to ensure that consideration is given to pipe break effects for all systems and locations. This may introduce additional structural protection and environmental qualification requirements in the SSAR.

6. Use of Integral Low-Pressure Turbine Rotors

In our July 18, 1991 report to you, we recommended that the staff review the issues involved with the use of integral low-pressure (LP) turbine rotors. It is our understanding that this new design for LP rotors will be used for the ABWR. (Rotors of this type are being used in rotor replacement programs at currently operating plants.) The practice of turbine manufacturers has been to bore the centerline of this type of rotor to remove impurity inclusions. We were concerned that the use of unbored rotors was being contemplated. The Electric Power Research Institute (EPRI) has recently added a requirement in its Advanced Light Water Reactor Utility Requirements Document (URD) that LP rotors be center-bored.

7. Cavity Floor Area Beneath Reactor Vessel

The cavity area beneath the reactor vessel is sized to meet the EPRI URD specification of 0.02 m^2/Mwt. The ABWR design includes flooding of the cavity. Little consideration has been given to how this should be accomplished. There is little evidence that the planned cavity area will lead to quenching following flooding or that the ABWR flooding plans will not lead to ex-vessel steam explosions. Further attention needs to be given in the SSAR as to when and how fast the cavity should be flooded in order to avoid exacerbating a core-melt accident if it should occur.

8. Adequacy of the ABWR PRA

It is impossible to determine whether the PRA submitted by the applicant will be adequate for a safety determination absent information on how it is to be used by the staff. In our February 14, 1992 report to the Commission on the Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews, we commented on the need for guidance on the use of PRA in the review of new plant designs. At this point the applicant has submitted a PRA, a contractor has performed an extensive review, and the staff has prepared a DSER. However, the use of the PRA in the design certification process is still undefined.

Presumably, the results of the PRA will be used in the course of the staff's determination that the design is expected to produce a nuclear power plant that has an appropriate response to severe accidents. In the Severe Accident Policy Statement, the Commission indicated that a PRA would be required for each new design, and that the results of this PRA would be part of the information which would guide the staff in its determination that a design is adequate to deal with severe

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accidents. The policy statement published in the Federal Register of August 8, 1985, also states that "Accordingly, within 18 months of the publication of this Severe Accident Policy Statement, the staff will issue guidance on the form, purpose and role that PRAs are to play in severe accident analysis and decision making for both existing and future plant designs...." The Statement says further, "The PRA guidance will describe the appropriate combination of deterministic and probabilistic considerations as a basis for severe accident decisions."

The staff has yet to produce the promised guidance. We urge that the staff formulate a set of criteria that it plans to use in making severe accident decisions. This should include the way in which the results of a PRA are to be used in the process (not just whether the PRA has been done properly).

9. Containment Hydrodynamic Loads

Air-clearing loads on containment structures are the result of a complex process resulting from the drywell air being forced into the wetwell by the primary system blowdown. The water in the vent system is pushed down and out until the horizontal vents are cleared. The water-clearing process produces a jet of water into the suppression pool which causes a load on the outer part of the wetwell wall. This water clearing is followed by an air-steam mixture which creates a large bubble as it exits into the pool. The steam condenses but the air expands forcing the water above it up into the wetwell air space. The wetwell air space is compressed due to the momentum of the water in the layer above the bubble.

The wetwell air space will be subjected to an energetic two-phase eruption as a result of the air-clearing process. The vacuum breakers which are in the vicinity will be exposed to this environment unless protected. The SSAR should describe what the environment will be and what protective measures, if any, are needed to ensure survival of the vacuum breakers. If a vacuum breaker does not close, the suppression pool is bypassed and the wetwell/drywell pressures will rise at a rate dictated by the capability of some means other than the suppression process (e.g., containment sprays) to remove heat and condense steam. The SSAR should contain an analysis of such a situation.

The early work to address problems arising from analyses of the Mark I, II, and III containments is not sufficient to address similar processes that will occur following a LOCA in an ABWR containment. The ABWR is different for two reasons: (a) the volume of the wetwell air space in the ABWR is approximately that of a Mark II, and (b) the impact of the air-clearing loads will be alleviated somewhat because the expected blowdown flows are much smaller than those expected in a Mark I or Mark II. Nevertheless, the combination of a much smaller wetwell and the lower mass flow from the break have not received sufficient attention to be written off by the staff or GE without further analysis or experimental investigation. We are not aware of any testing of the ABWR type geometry. We believe there are sufficient differences in both geometry and LOCA characteristics to require further evaluation of the air- clearing phase of the LOCA by more extensive analysis and/or experimental investigation.

10. Adequacy of SSAR Treatment of the Reactor Water Cleanup System

We performed a review of the Reactor Water Cleanup (RWCU) System using our own staff. This system was chosen because it is a non-safety system located outside of primary containment, but inside the building which houses engineered safety features. It uses pipes up to 8-in. nominal diameter whose rupture would result in a LOCA and a source of serious environmental disruption in the building. This system is not seismically qualified or built to quality assurance standards.

Our review identified a number of deficiencies in the SSAR, some of which are listed below:

- There is little useful information presented in the SSAR that describes how the Japanese codes and standards used for the RWCU System design can be converted to domestic design standards. The Quality Group classifications for certain portions of the RWCU System are inconsistent with the Japanese code-related classifications shown on the Piping and Instrumentation Diagrams. The Safety Class/Quality Group transition between the piping inside primary containment and that outside primary containment is not in accordance with ANSI/ANS safety class standards for BWR fluid systems.

- The questionable ability of system isolation valves to close under large-break-LOCA conditions has been the subject of extensive NRC testing and a Generic Letter (GL 89-10). However, the SSAR specifies no special performance requirements for these valves.

- The safety-grade leak detection and isolation system which actuates the system isolation valves was not described in detail sufficient to support an assessment of its adequacy.

- The ABWR PRA did not evaluate as initiating events RWCU System line breaks (or other LOCAs) outside the primary containment. The exclusion of these breaks was based erroneously on an analysis of the effects of suppression pool bypass events on overall risk. However, the analysis failed to take into account that the bypass path (e.g., RWCU System pipe break) could be the initiator for the core-damage event.

- The PRA analysts took credit for the RWCU System as a heat removal system in all sequences where reactor pressure is assumed to remain high. The analysts assumed that the capacity of the non-regenerative heat exchanger (NRHX) is adequate to remove the decay heat. The capacity appears to be adequate; however, our calculations indicate that the outlet temperatures on the RWCU System side and cooling water side of the NRHX would exceed the design limits for the piping. Furthermore, a temperature sensor between the NRHX and the RWCU System pumps in the present design would automatically isolate the NRHX on high temperature, making it unavailable.

The items mentioned above are among a number of issues that were identified. It is important for the staff to ensure that the shortcomings of the RWCU System and PRA related portions of the SSAR are not indicative of problems in the remainder of that report.

11. Plant Design Life and Aging Management

We recommend that the SSAR clearly define the scope of the 60-year design life for the ABWR and describe a program plan for achieving it. This program should include those aging management measures which are necessary to maintain the plant within its design basis throughout its design life. This program should specify the original design and application criteria and, where required, the projected refurbishment or replacement requirements with appropriate rationale. To the extent applicable, the lessons learned from the NRC's Nuclear Plant Aging Research Program as well as other aging research projects should be incorporated into this program.

We note that the EPRI URD (Volume II, Chapter 1, Paragraph 3.3) includes a requirement for a plant design life of "60 years without necessity for an extended refurbishment outage," and discusses the requirements for its achievement in Paragraph 11.3.

12. Station Grounding and Surge Protection

Chapter 8 of the ABWR SSAR defines the scope of and specifies the requirements for the electrical power systems. The scope is limited to the onsite electrical power systems and to the interface requirements with the offsite electrical power systems.

Notably absent are lightning protection, station grounding systems, and surge protection measures which are necessary to protect plant personnel and equipment during normal and abnormal conditions. These measures are required to eliminate or reduce electrical shock hazards to personnel, and to protect systems and equipment against damage or misoperation as the result of lightning strikes, switching operations, electrical arcs, short circuits, static electricity, etc. These protective measures and their interface requirements should be included in the SSAR.

The ABWR makes extensive use of sensitive solid-state electronic components for essential protection, control, and data transmission functions. These components should be protected from extraneous electrical impulses that will damage them or cause improper performance. To the extent practical, these components should be isolated from potential adverse signals that may be transmitted over control or data links from remote locations, meteorological stations, switchyards, etc.

We note that the EPRI URD (Volume II, Chapter 11, Item 9, "Electrical Protective Systems") addresses requirements for these systems. We recommend that these grounding, surge protection, and isolation features be included in the SSAR.

13. Corrosion Control for Structures

The SSAR should include an interface requirement for a corrosion control program to identify the potential for the corrosion of structures and components and to determine the corrective measures to be taken. The program should commence prior to the completion of the detailed design of building substructures and underground installations. The program should consider the potential for corrosion from galvanic direct currents which may flow as the result of copper ground mats on site, including the electrical switching stations' ground mats. The potential for corrosion of containment building substructures and liners should be considered. The mitigation measures may include coatings, wrappings, cathodic protection, electrical bonding, elimination of galvanic currents, or other mitigation means.

We do not expect to receive a separate reply to the above items if they are covered appropriately in the final SER. We will keep you informed of any additional concerns as our review proceeds.

Sincerely,

David A. Ward

David A. Ward
Chairman

<u>References</u>:

1. GE Nuclear Energy, Standard Safety Analysis Report, "Advanced Boiling Water Reactor," Chapters 1 through 20 (Amendments 1 through 18)
2. SECY-91-153, dated May 24, 1991, for the Commissioners from James M. Taylor, Executive Director for Operations, NRC, Subject: Draft Safety Evaluation Report (DSER) on the General Electric Company Advanced Boiling Water Reactor Design Covering Chapters 1, 2, 3, 4, 5, 6, and 17 of the Standard Safety Analysis Report (SSAR)
3. SECY-91-235, dated August 2, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapters 1, 3, 9, 10, 11, and 13 of the SSAR
4. SECY-91-294, dated September 18, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapter 7 of the SSAR
5. SECY-91-309, dated October 1, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapter 19 of the SSAR, "Response to Severe Accident Policy Statement"
6. SECY-91-320, dated October 15, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Advanced Boiling Water Reactor Design Covering Chapter 18 of the SSAR
7. SECY-91-355, dated October 31, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapters 1, 2, 3, 5, 6, 8, 9, 10, 12, 13, 14, and 15 of the SSAR
8. Electric Power Research Institute, "Advanced Light Water Reactor Utility Requirements Document" (Volume II)/ALWR Evolutionary Plant, Revision 3, Issued November 1991

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

August 13, 1991

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

Dear Chairman Selin:

SUBJECT: ADDITIONAL COMMENT ON SCHEDULES FOR ADVANCED REACTOR REVIEWS

In our report to you of July 18, 1991, on "Schedules for Advanced Reactor Reviews," we noted that the time required for Committee review of the final Safety Evaluation Reports (SERs) and Final Design Approvals will be three months, as stated in the text of SECY-91-161, rather than two months as shown on the bar charts. We failed to note that the three months review time (starting at time of receipt) also applies to the draft SERs. Except for ABWR, the bar charts show only one month for ACRS review. The text is silent on this point.

Sincerely,

David A. Ward

David A. Ward
Chairman

Reference:
U.S. Nuclear Regulatory Commission, SECY-91-161, dated May 31, 1991, from J. Taylor, Executive Director for Operations, for the Commissioners, Subject: Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions

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

July 18, 1991

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

Dear Chairman Selin:

SUBJECT: SCHEDULES FOR ADVANCED REACTOR REVIEWS

During the 375th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 1991, we discussed the staff's proposed "realistic" schedules identified in SECY-91-161 for completing the reviews of the evolutionary and passive advanced light water reactor (ALWR) design certification applications and the review of the Electric Power Research Institute's (EPRI) ALWR Utility Requirements Document. We had the benefit of presentations by and discussions with members of the NRC staff and NUMARC, as well as the documents referenced. Consideration of this matter by the Committee was based on the request of the Commission, as reflected in Staff Requirements Memorandum M910607A dated June 18, 1991.

We believe that, barring unforeseen circumstances, the ACRS will be able to meet these schedules. Note, however, that the time required for Committee review of the final SERs and FDAs will be three months, as stated in the text of SECY-91-161, rather than two months as shown on the bar charts.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:
1. U.S. Nuclear Regulatory Commission, SECY-91-161, dated May 31, 1991, from J. Taylor, Executive Director for Operations, for the Commissioners, Subject: Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions
2. Electric Power Research Institute, Utility Requirements Document, June 1986
3. Memorandum dated June 18, 1991 from Samuel J. Chilk, Secretary of the Commission, for David A. Ward, ACRS, and James M. Taylor, EDO, Subject: Staff Requirements - Periodic Meeting with the ACRS, June 7, 1991

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

July 18, 1991

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

Dear Mr. Taylor:

SUBJECT: CONCERNS RELATED TO THE GENERAL ELECTRIC ADVANCED BOILING WATER REACTOR DESIGN

During the 375th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 1991, we discussed the status of the Advanced Boiling Water Reactor (ABWR) design, described in the Standard Safety Analysis Report (SSAR), for which the General Electric Company (GE) has applied for design certification in accordance with 10 CFR Part 50, Appendix O. Our Subcommittee on Advanced Boiling Water Reactors also discussed this matter during its meetings on October 31, 1990, and May 30, 1991, with representatives of GE and the NRC staff. We also had the benefit of the documents referenced.

Our previous letter to you concerning the ABWR design was dated November 24, 1989, and conveyed our comments on Module 1 of the Draft Safety Evaluation Report (DSER). Since this letter, we have been kept apprised of the design and the status of the review while awaiting receipt of additional DSERs. The staff now says that DSER preparation by modules will be discontinued in favor of preparation by SSAR chapters and Standard Review Plan (SRP) sections.

To ensure the completeness of our review, it will be necessary to account for any additions or revisions to each DSER as forwarded by a SECY subsequent to issuance of our respective comment letter. An arrangement acceptable to us is needed to ensure the identification of any additions or revisions, and we should agree on an appropriate time for their review. Our comments will not be complete, however, until we have submitted a report to the Commission concerning the final SER on which we expect to comment by mid-November 1992.

Our activities subsequent to the completion of our November 1989 letter have focused on several design concerns that were discussed with GE and the NRC staff in an effort to ensure an early awareness and understanding. We believe that it is appropriate to document them here for timely consideration and resolution in appropriate DSER sections. We expect to have additional items later. We do

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not expect separate replies to our concerns provided the staff responds in the appropriate DSER.

1. <u>Control Building Flooding</u>

The proposed ABWR design locates the Reactor Building Cooling Water (RBCW) System at the lowest elevation in the control building with the essential 250-V. DC battery rooms immediately above, and the main control room at the next higher elevation. This arrangement places the main control room below ground grade. Our concern with this arrangement is the potential for control building flooding due to an unisolated break in the open-cycle cooling water piping or components inside the building. The ultimate heat sink (cooling pond) is likely to provide sufficient water to flood the building to near ground grade.

2. <u>Physical Separation Barriers</u>

Internal plant flooding and external events such as fire are of major concern if their effects cannot be confined to a single division of required safe-shutdown equipment. We believe that the key to confinement is the provision of an appropriate separation barrier. However, a classical barrier such as the 3-hour-rated fire barrier may not of itself, be sufficient to ensure divisional separation under the combined effects of pressure, heat, smoke, and flooding which accompany a fire and its mitigation. Also, it would appear from the SRP that the effects of delayed suppression on room temperature, pressure, and barrier leakage need to be considered when determining that safe shutdown can be achieved. We remain unconvinced that divisional separation barriers for the ABWR have been adequately prescribed for the range of events and conditions during which they must provide separation.

Of particular concern is a diesel fuel fire which may be subject to delayed suppression in the ABWR diesel generator rooms which are located inside the reactor building. It is not clear how these rooms will be qualified by design or testing to withstand burning fuel if spread across the floor by a fuel line rupture. Furthermore, it is not apparent how the compartment doors will be qualified for this condition or whether they can confine the fuel to the room. If manual mitigation is required, a fire barrier door must be opened. It is not certain that this can be achieved safely or that the external environmental effects of a prolonged opening of the door have been considered.

3. Environmental Protection for Solid-State Electronics

The ABWR makes extensive use of solid-state electronic components for essential protection, control, and data transmission functions. Such components are known to be susceptible to adverse environmental changes, particularly temperature extremes. We are concerned that a number of these components may be located in plant areas where postulated events such as pipe rupture, fire, internal flooding, or loss of room cooling may create an adverse environment. The response of such components to the environmental change may be unpredictable and lead to unacceptable system interactions or responses. The behavior of solid state electronic components in environments created by off-normal or accident situations needs to be considered before the adequacy of any physical separation and environmental control measures can be evaluated.

4. Review of Chilled-Water Systems

The ABWR makes extensive use of large chilled-water systems to provide essential environmental cooling functions including those for the solid-state electronics. Since there is no SRP for chilled-water systems, the staff uses other guidance such as SRP Section 9.2.2 (Reactor Auxiliary Cooling Water Systems) when performing its safety evaluation. This guidance does not include evaluation of the large refrigeration equipment that is required for chilling the closed-cycle cooling water.

The NRC staff and GE need to evaluate the safety implications of chilled-water systems, including performance under varying accident heat loads, loss-of-offsite-power loading characteristics, and ability to restart and function after a prolonged station blackout. The NRC staff should develop appropriate guidance for such reviews by preparing a suitable SRP now.

5. Use of Leak-Before-Break Methodology Outside of Primary Containment

In our report of March 14, 1989 to then NRC Chairman Zech on "Additional Applications of Leak-Before-Break Technology," we expressed our belief that an avenue for consideration of further extension of the leak-before-break (LBB) concept should exist. This is still our position. We are concerned that the NRC staff is not giving serious consideration to GE proposals to extend the concept to systems outside of the primary containment because the staff feels constrained by General Design Criterion 4 which does not propose review of methodology.

We would like to see a renewed effort by GE and the NRC staff to determine if a real potential for substantial safety and/or economic benefits can be realized in applying properly the LBB concept outside of the primary containment.

6. Use of Integral Low-Pressure Turbine Rotors

The catastrophic failure of a low-pressure (LP) turbine rotor can lead to high-energy missiles that are capable of damaging safety-related equipment. The domestic turbine manufacturers (General Electric and Westinghouse) have been using an LP turbine design for large turbine generators consisting of a relatively small-diameter bored shaft with shrunk-on and keyway locked blade ring disks. The manufacturers are now offering an integral LP turbine rotor machined from a single large-diameter forging. A rotor of this design would operate at much higher stresses than the shaft of a shrunk-on disk rotor.

We were told by the Electric Power Research Institute (EPRI) representatives that a decision has not as yet been made with respect to a requirement in the ALWR Utility Requirements Document for boring the LP turbine rotors. Boring has historically been performed to remove impurity inclusions near the forging centerline. Such inclusions are stress risers and have led in the past to a number of catastrophic turbine and generator rotor failures in fossil-fueled power plants. Modern forging practices minimize such inclusions and present-day nondestructive examination and evaluation techniques provide much greater assurance of the soundness of turbine-generator rotors.

The NRC staff should follow this issue closely since the use of integral LP turbine rotors, particularly if they are not bored, will require the development of an entirely new set of preoperational and periodic operational inspection, evaluation, and acceptance requirements to protect against turbine missiles. (The staff should also consider this issue for LP turbine rotor replacement programs for currently operating plants.)

7. Cavity-Floor Area Beneath Reactor Vessel

The layout of the containment for the proposed ABWR design makes use of a cavity floor area beneath the reactor vessel to deal with core/concrete interaction. This area is based on an EPRI requirement of $0.02m^2$ per MWt. If a larger area is required, major changes to the containment sizing and layout may be needed. Timely development of a Commission position on this issue is important not only to this design but also to the design of all Advanced Light Water Reactor designs.

Sincerely,

David A. Ward
Chairman

References:
1. Letter dated August 17, 1989 from Charles L. Miller, Office of Nuclear Reactor Regulation, NRC, to Patrick W. Marriott, General Electric Company, enclosing Draft Safety Evaluation Report Related to the Final Design Approval and Design Certification of the Advanced Boiling Water Reactor, dated August 1989.
2. Letter dated August 7, 1987 from Thomas E. Murley, Office of Nuclear Reactor Regulation, NRC, to Ricardo Artigas, General Electric Company, enclosing GE Advanced Boiling Water Reactor, Licensing Review Bases, dated August 1987.
3. GE Nuclear Energy, Standard Safety Analysis Report, Advanced Boiling Water Reactor, Chapters 1 through 20.

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

November 24, 1989

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

Dear Mr. Taylor:

SUBJECT: MODULE 1 OF THE DRAFT SAFETY EVALUATION REPORT FOR THE ADVANCED BOILING WATER REACTOR DESIGN

During the 355th meeting of the Advisory Committee on Reactor Safeguards, November 16-18, 1989, we met with representatives of the Office of Nuclear Reactor Regulation (NRR) and the General Electric Company (GE) to discuss Module 1 of the staff's Draft Safety Evaluation Report (DSER) for the Advanced Boiling Water Reactor (ABWR) design. This matter was also considered by our ABWR subcommittee during several meetings, the latest on October 31, 1989. We also had the benefit of the documents referenced.

The staff's DSER relates to the GE application for final design approval (FDA) and design certification of the ABWR design. The DSER is scheduled for completion in four modules. Module 1 is the subject of this letter and addresses Chapters 4, 5, 6, and 17 of the ABWR Standard Safety Analysis Report (SSAR) and corresponding chapters of the Standard Review Plan (SRP), NUREG-0800. Our review of these chapters of the SSAR has been completed through Amendment 7.

A number of the SSAR and DSER sections included in the Module 1 chapters are presently missing and will be issued as SSAR revisions and supplements to the DSER. Even within the included sections, there are a number of open, unresolved, and confirmatory issues and incomplete interface requirements or other information that will delay completion of our review until the revisions and supplements are issued. Comments on such missing or incomplete information will be included with our review of future modules.

Our comments should not be considered complete until we have prepared a report to the Commission concerning the final integrated DSER, which is presently scheduled for late 1990. For now, we are providing the following comments and recommendations concerning Module 1.

GENERAL

1. The staff's ABWR licensing review bases letter to GE (Reference 2) states, "The degree of design detail necessary for providing an essentially complete design is to be that detail that is suitable for obtaining specific equipment or construction bids and to demonstrate

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conformance to the design safety limits and criteria." We believe that the level of design detail in Module 1 falls short of this requirement. For example, we find that while GE has committed to follow applicable codes, standards, and regulatory guides, they have developed internal specifications for materials used in the fabrication of pressure boundary components that have not been submitted for NRC review. We also find that a number of design details (such as those relating to design temperature and pressure and pipe size) are indicated on drawings in the SSAR as "to be established by others" or similar statements. Unless such information is included in the SSAR or other documents that are reviewed by the staff, it is clear that the level of design detail is inadequate. We recommend that the staff revisit the issue of what constitutes an "essentially complete" design. The staff should also consider the question of form and depth of reporting differences between the ABWR being designed for construction in Japan and the ABWR design being proposed for certification.

2. The SSAR chapters contain a number of sections for which there are no corresponding sections in the DSER or SRP, or the subjects of the DSER or SRP sections are different. Also, there are cases wherein the SRP contains sections that do not appear in the SSAR or DSER. We recommend that the DSER sections be referenced by number and title to the corresponding SSAR sections they evaluate. Differences, including the absence of any corresponding SRP sections, should be identified in the DSER.

CHAPTER 4 - REACTOR

3. The fine motion control rod drive system (FMCRDS) materials list discussed in SSAR Section 4.5.1.1 shows Stellite guide rollers and roller pins. Section 5.2.3.2.2.2 states that cobalt base alloys used for pins and rollers in the FMCRDS have been replaced with noncobalt alloys. The list of materials should be corrected.

4. We were told by GE that the design of the integral rod ejection support system for the FMCRDS has been changed from that described in SSAR Section 4.6.1. The staff should determine that their evaluation in the DSER is based on the revised design and the SSAR should be corrected.

CHAPTER 5 - REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5. The SSAR states that the automatic depressurization system (ADS) utilizes safety relief valves (SRVs) each of which is equipped with an air accumulator and check valve arrangement designed to ensure two actuations following failure of the air supply. Although not stated in the SSAR, GE indicated that the accumulators are backed up by the nitrogen supply system. This backup arrangement needs to be described in the SSAR together with how check valve operability will be ensured.

6. The specifications given in the SSAR for the materials of the primary pressure boundary do not meet current "good practice," or the practice GE says they would require in the construction of an ABWR--they should. To clarify this issue, the SSAR should contain answers to the following questions: (1) will the steel in the core beltline be forged rings or welded plate?; (2) will upper limits on sulfur content of the rolled plate in the pressure vessel be those given in the ASME Code SA-533, Specifications for Pressure Vessel Materials (0.04%) or lower values consistent with good modern practice (under 0.015% with shape control)?--an adequate level is specified for forged segments (ASME Code SA-508, Class 3, Specification for Quenched and Tempered Vacuum-Treated Forgings) and is available as an option in SA-533 but not called out by GE; and (3) what will be the upper limit on delta ferrite for cast stainless steel components? The Code's allowed value of 25% should be halved to substantially remove concern about long-term aging.

7. SSAR Section 5.3.3 states that design for vessel annealing is not required because the predicted value of adjusted RT_{NDT} does not exceed 200° F. The DSER states that the integrity of the reactor vessel is ensured because the vessel may be annealed, if necessary. GE stated during our meeting that the vessel is not designed to be annealed. The DSER statement should be resolved with GE.

8. We believe that potential safety hazards (e.g., excessive internal pressure) associated with an uncleared electrical fault inside a reactor internal pump (RIP) should be analyzed and documented in the SSAR.

9. We were told by GE that motor restraint rods are provided to prevent ejection of an RIP. We believe that this important feature should be described in the SSAR and evaluated by the staff.

10. SSAR Section 5.4.6 states that the design basis for the Reactor Core Isolation Cooling (RCIC) system is only 30-minutes of operation during a loss-of-ac power event. We believe that a more complete discussion of the station blackout capability should be included in the SSAR. The DSER should include an evaluation of the 30-minute capability as an acceptable design basis.

11. The DSER contains no specific references to SSAR Sections 5.4.4-5, 5.4.9, and 5.4.12-14. These sections discuss feedwater piping, main steam line flow restrictors, isolation systems and piping, component supports, and valves. There are no comparably numbered sections in the SRP. It is not clear where the staff intends to report its evaluation of these important topics.

CHAPTER 6 - ENGINEERED SAFETY FEATURES

12. The design basis for the ECCS and the conclusions given about its performance do not include the ejection of an RIP (450 cm^2 break).

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The rationale for excluding such an event as a design basis break should be discussed in the SSAR.

13. DSER Section 6.2.6 indicates that inflatable seals will be used for primary containment equipment and personnel air lock penetrations. We believe that an appropriate description of the seals and the air supply arrangement and reliability should appear in the SSAR. The discussion should include the capability of the seals to function under elevated pressure and temperature conditions for prolonged periods of time following a design basis accident.

14. There is a new section 6.5.5 (Pressure Suppression Pools as Fission Product Clean-Up Systems) in the SRP which does not appear in the SSAR or DSER. Why is this SRP section not being used for the ABWR?

CHAPTER 17 - QUALITY ASSURANCE

15. Chapter 17 of the SSAR is intended to describe how GE and its major technical associates (not mentioned by name in the SSAR but we assume to be Toshiba Corporation and Hitachi Limited) engage in the joint development and engineering of the ABWR design. The quality assurance programs used by the technical associates are not described or referenced in the SSAR. We believe they should be.

In conclusion, we believe that significant progress has been made by the staff in its review of the SSAR for the Advanced Boiling Water Reactor. A considerable amount of work remains to be completed before the FDA is issued as expected by the end of 1990. We will continue to review this work as the documentation becomes available.

Sincerely,

FJRemick

Forrest J. Remick
Chairman

References:

1. Letter dated August 17, 1989 from Charles L. Miller, Office of Nuclear Reactor Regulation, NRC, to Mr. Patrick W. Marriott, General Electric Company, enclosing Draft Safety Evaluation Report Related to the Final Design Approval and Design Certification of the Advanced Boiling Water Reactor, dated August 1989
2. Letter dated August 7, 1987 from Thomas E. Murley, Office of Nuclear Reactor Regulation, NRC, to Ricardo Artigas, General Electric Company, enclosing GE Advanced Boiling Water Reactor, Licensing Review Bases, dated August 1987
3. GE Nuclear Energy, Standard Safety Analysis Report, Advanced Boiling Water Reactor, Chapters 4, 5, 6, and 17

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ITEM 4: Three Issues Related to 10 CFR Part 52 Design Certification

At the February 1994 meeting, the Committee discussed three issues that relate to the 10 CFR Part 52 design certification process for ALWRs. These issues are: (1) the staff's implementation of Reliability Assurance Program (RAP), (2) the staff's proposed use of "starred" Tier 2 Certified Design Material (CDM), and (3) Technical Specification requirements for onsite power sources for Evolutionary Light Water Reactors (ELWRs).

The following document is attached:

- ACRS letter to James M. Taylor (EDO) dated February 17, 1994. Subject: Three Issues Relating to the 10 CFR Part 52 Design Certification Process for ALWRS (PP.73-75)

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

February 17, 1994

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

Dear Mr. Taylor:

SUBJECT: THREE ISSUES RELATING TO THE 10 CFR PART 52 DESIGN CERTIFICATION PROCESS FOR ALWRS

During the 406th meeting of the Advisory Committee on Reactor Safeguards, February 10-11, 1994, we discussed three issues that relate to the 10 CFR Part 52 design certification process for Advanced Light Water Reactors (ALWRs): (1) the staff's implementation of Reliability Assurance Program (RAP), (2) the staff's proposed use of "starred" Tier 2 Certified Design Material (CDM), and (3) Technical Specification requirements for onsite power sources for Evolutionary Light Water Reactors (ELWRs). We are commenting on these matters at this time because we believe that they need timely senior staff management attention. We had the benefit of the documents referenced.

ALWR Reliability Assurance Program

During our January 6-7, 1994 meeting, we heard a staff presentation on the RAP that is being required as a part of the design certification of ALWRs. The RAP requires both a design phase and an operational phase reliability assurance program (DRAP and ORAP). In addition, we reviewed your memorandum of August 2, 1993, in response to Commissioner Remick's questions on this subject. We also understand that OGC has concerns regarding the need for the DRAP and ORAP.

In our letter to you dated October 15, 1992, concerning "Proposed Guidance for Implementation of the Maintenance Rule," we noted that the RAP being required of ALWR COL holders ". . . will involve the establishment of a third kind of maintenance program (in addition to the maintenance programs required by the Maintenance Rule and the License Renewal Rule)." We suggested that consistent staff guidance was needed on the elements of an acceptable program that will satisfy these three sets of requirements. We have subsequently learned that a similar situation exists in the relationship between RAP and the quality assurance requirements of Appendix B to 10 CFR Part 50.

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While we agree that PRA insights with respect to the reliability of risk significant structures, systems and components (SSCs) should be a part of maintenance and quality assurance programs for ALWRs, we continue to question the need for a separate RAP. We believe that senior staff management should perform a high level review of the need for the RAP. The objective of such a review should be to determine if it is possible to integrate those unique requirements of RAP that have a valid safety basis into the implementation of existing programs required for ALWRs (the Maintenance Rule, the License Renewal Rule, and Appendix B to 10 CFR Part 50).

The following aspects of RAP are of particular concern to us:

- The staff appears to believe that risk-significant SSCs should be given some sort of "special consideration" during the detailed design and procurement phases of an ALWR plant. It is not clear to us how the design engineering organization of a COL holder will be able to demonstrate that it has given "special consideration" to the procurement of risk-significant SSCs.

- The staff has not made it clear how the COL holder will develop reliability monitoring programs that will demonstrate that risk-significant SSCs are operated and maintained consistent with the PRA assumptions during the operational life of the plant. Demonstration of the reliability of risk-significant ALWR SSCs in any meaningful manner is clearly not feasible.

ALWR "Starred" Tier 2 Material

The staff has recently told us of its plan to designate certain Tier 2 CDM in the certification of the General Electric Nuclear Energy ABWR, and presumably in the certification of other ALWRs, as material which could not be changed by a COL holder under the 10 CFR 50.59 - like process, but would require prior review and approval by the staff. This will, in effect, create a three tier design certification process. Although there may be a valid need for this kind of restriction in certain cases, we recommend that senior staff management review each application of such "starred" Tier 2 CDM to ensure that the process is not being used in an arbitrary and capricious manner by the staff. In our view, the existing 10 CFR 50.59 - like process that a COL holder must use in order to change Tier 2 material generally provides the needed check and balance on changes to Tier 2 material.

ELWR Technical Specification Requirements for Onsite Power Sources

The staff informed us during our ABB-CE System 80+ Subcommittee meeting of December 8, 1993, that it is still considering Technical Specification requirements for onsite power sources for ELWRs. (A

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similar, but somewhat different, issue exists with respect to the onsite power sources for the "passive" LWRs.) We have been interested for some time in the question of what credit will be given for the ELWR Alternate AC (AAC) source when one of the 1E Emergency Diesel Generators (EDGs) is out of service. Unlike the 1E EDGs, the AAC sources in the ELWR plant designs are not seismically qualified nor are they located within a structure hardened against the effects of tornados or hurricanes. This is particularly an important issue for the ABB-CE System 80+, where the onsite power sources consist of two 1E EDGs and a single AAC. If one of the 1E EDGs is out of service for maintenance, loss of offsite power (LOOP) would make the unit vulnerable to the single failure of the remaining 1E EDG under design basis accident conditions. Unless credit is given for the AAC (which may be damaged as a result of a seismic event or tornado or hurricane that caused the LOOP), the unit would have to be shut down whenever extended maintenance is performed on either of the 1E EDGs during power operation.

It appears to us that staff resolution of this matter is long overdue and that senior staff management attention to this issue is needed. Further, we believe that the Technical Specification Requirements for onsite power sources for ELWRs should be based on appropriate probabilistic risk considerations.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, ~~Jr.~~
Chairman

References:
1. Memorandum dated August 2, 1993, from James M. Taylor, NRC Executive Director for Operations, to Commissioner Remick, Subject: SECY-93-087: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs
2. Report dated October 15, 1992 from David A. Ward, Chairman, ACRS to James M. Taylor, NRC Executive Director for Operations, Subject: Proposed Guidance for Implementation of the Maintenance Rule, 10 CFR 50.65
3. U.S. Nuclear Regulatory Commission, "Advance Copy of Safety Evaluation Report Related to the Certification of the Advanced Boiling-Water Reactor Design," December 1993

ITEM 5: ABB-CE System 80+ Design

The ACRS Subcommittee on Advanced Pressurized Water Reactors has held six meetings beginning in April 1990 to discuss the ABB-CE Systems 80+ design features and related issues. The Committee provided a report to the Commission dated November 14, 1990 in regard to the Licensing Review Basis (LRB) document. The staff's Draft SER on the Systems 80+ design was provided to the ACRS on October 1, 1992, and a Subcommittee meeting was held on February 9, 1993 to discuss this document. During the Subcommittee meeting, two major issues were discussed. These were human factors engineering and diversity of instrumentation and control. As these issues were not yet resolved, the Committee did not comment on them.

On April 13, 1993, some members of the ACRS Subcommittee on Advanced Pressurized Water Reactors visited the dynamic mockup of the control room at the ABB-CE facility in Windsor, Conn. The Subcommittee meetings on December 8, 1993 and February 9, 1994 reviewed the following documents for the Systems 80+ design: (1) specific chapters of the Standard Safety Analysis Report, (2) the updated draft FSER, and (3) the ABB-CE submittal of certified design material. Additional meetings have been scheduled for March 8-9 and April 5-6, 1994.

The following document is attached:

- ACRS Plan for the ABB-CE Systems 80+ Review, March 2, 1994 (PP.77)

PROPOSED CE SYSTEM 80+ ACRS REVIEW PLAN

Current as of March 2, 1994

(Note: * indicates as per SECY-93-097, **bold** indicates ACRS mtg)

DATE	ACTION	NOTES
Dec 8, 1993	**ACRS CE80+ Subcom meeting** (NUPLEX 80+ demonstration)	SSAR/FSER Chapters reviewed: 7 - I&C 8 - Electric Power 18 - Human Factors Engr.
Dec 31, 1993	CE submitted final Tier 1, DD, and ITAAC to NRC	As of Dec 31, SSAR Amendment T was most current
9 Feb 94 (Wed)	**ACRS CE80+ Subcom meeting**	SSAR/FSER Chapters reviewed: 4 - Reactor 10 - Steam and Pwr Convers. 11 - Rad Waste Management 12 - Rad Protection 13 - Conduct of Operations 14.2 - Initial Test Program 17 - Reliab Assur & QA
late Feb* 94 or early Mar	Draft FSER submitted to ACRS and Commission	As of Feb 11, SSAR Amendment U was most current
8, 9 Mar 94 (Tue/Wed)	**ACRS CE80+ Subcom meeting** Principal reviewers: Mr. Lindblad - Chapt 2 & 3 Mr. Davis - Chapt 19 PRA Dr. Kress - Chapt 19 SA	Review SSAR/FSER Chapters 2 - Site Envelope Character. 3 - Design Struct/Comp/Equip 19 - PRA(Level I, II, & III) Severe Accident, & SD Risk
5,6 Apr 94 (Tue/Wed)	**ACRS CE80+ Subcom meeting** Principal reviewers: Dr. Shack - Chapt 3, 4, 5, and 10 (materials sections)	SSAR/FSER Chapters to review: 1 - Intro/General Descript. 5 - Reactor Coolant Systems (and Materials sections) 6 - Engrd Safety Features 9 - Auxiliary Systems 14.3 - ITAAC 15 Accident Analysis 16 - Technical Specifications A - Closure of USIs/GSIs
Apr 94	**ACRS Full Committee mtg**	ABB-CE/Staff presentations
May 94	**ACRS Full Committee mtg** DRAFT Committee letter	
June 94	**ACRS Full Committee mtg** FINAL Committee letter	
late June* 94	Publish FSER	

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ITEM 6: AP600 and SBWR Test Programs

• Westinghouse AP600

The Committee and the Subcommittees on Advanced Pressurized Water Reactors/Thermal Hydraulic Phenomena have heard presentations regarding design details for the Westinghouse AP600 passive plant and the test programs proposed by both Westinghouse and the staff in support of the AP600 design certification during several meetings. 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.

More recently, the Committee 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. The Committee 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 discussed the status of the Westinghouse analytical and experimental programs noted above, as well as the staff's review of same, during its November 1993 meeting. A meeting of the Thermal Hydraulic Phenomena Subcommittee is currently scheduled for March 15-16, 1994 to continue the review of this matter.

The following documents are attached:

- ACRS report to the Commission dated November 18, 1993. Subject: NRC Confirmatory Test Program in Support of the AP600 Design Certification (PP.79-81)
- 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.82-86)

Note: General Electric SBWR starts on page 87.

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

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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

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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.

WESTINGHOUSE PROGRAM

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

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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**)

• General Electric SBWR

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.

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.

The following document is attached:

- 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.88-91)

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.

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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

David A. Ward
Chairman

References:

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

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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

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ITEM 7: Risk-Based Regulations

During the November 1992 meeting, the Committee discussed various aspects of risk-based regulations with representatives of the staff and industry. Both efforts by the staff and NUMARC had just gotten under way. The staff provided the following definition for the approach to risk-based regulation:

> "an approach to regulation where quantitative insights derived from a probabilistic risk assessment are used to focus utility and regulatory attention on design and operational issues commensurate with their impact on risk to the public".

Following a meeting with industry on March 10, 1992, the Commission issued a SRM dated March 26, 1992, in which they requested that the staff provide their views on the practicality of risk-based regulations and the feasibility of developing a transition strategy from deterministic based regulations. The Committee discussed a draft Commission paper with the staff during the November 1992 meeting but chose not to comment on the details in the paper as they were in need of much further development.

The following documents are attached:

- ACRS report to the Commission dated November 16, 1992. Subject: Risk-Based Regulation (PP.93-94)

- SRM dated March 26, 1992. Subject: Briefing on Risk-Based Regulations Transition Strategy, March 10, 1992 (PP. 95)

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

November 16, 1992

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

Dear Chairman Selin:

SUBJECT: RISK-BASED REGULATION

During the 391st meeting of the Advisory Committee on Reactor Safeguards, November 5-7, 1992, we reviewed a draft Commission paper on Risk-Based Regulation. The paper responds to the Staff Requirements Memorandum (SRM) dated March 26, 1992. During this meeting, we had the benefit of discussions with representatives of the NRC staff, and of the document referenced.

We interpret the Commission's charge to the staff as reflecting a recognition of the increasingly sophisticated and widespread use of analytical risk assessment techniques in the nuclear enterprise, a natural evolution of a process that began with the 1975 publication of the Reactor Safety Study, WASH-1400. Since it is now possible to make informed and quantitative statements about many (but not all) of the contributors to nuclear risk, it is correspondingly possible to optimize the deployment and use of the regulatory resources available to the Commission. The SRM directed the staff to both examine the feasibility of such a risk-based approach to regulation and to suggest means by which it could be implemented. The draft paper on which we were briefed is the preliminary response to that charge.

We would prefer not to comment in detail on the paper itself, except to note that it needs a great deal of work before it can be considered responsive to the Commission's charge at the level of sophistication demanded by the importance of the question. The staff is still working on the paper, and we expect to see a later and improved version. It is simply not yet ready for public comment.

Far more important to us is the issue of coherence of the various efforts now in progress in various parts of the staff to develop and implement activities that could be collected under the name of risk-based regulation. We have commented earlier about the Maintenance Rule, Regulations Marginal to Safety, and other

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initiatives involving the use of risk analysis, and have at this meeting heard about Risk-Based Regulation, revision of the Regulatory Analysis Guidelines, and the Prioritization of Generic Safety Issues. Each of these requires informed use of quantitative risk information and appropriate attention to the Commission's safety goals, yet each is being analyzed by an independent group, with an independent perspective on the NRC's needs. In addition to this, there is the PRA Working Group, whose progress we have been following closely. We are unable to find any focal point for all these efforts, except at the level of the EDO.

We continue to call for increased coherence in the treatment of all these matters, bound to each other by the common need to weave the threads of the safety goals (the expression of the ultimate objective of regulation) and quantitative risk assessment (the tool that makes more directed risk management possible) into the NRC fabric. If it is not done at the level of the EDO it will not be done, and resources that could be devoted to assuring nuclear safety will be squandered.

In the past we have suggested strong measures to address this problem. While not pushing any particular solution, we still believe that the collection of issues discussed here is important to the future performance of the agency. The coherence problems will not be solved by an incoherent effort.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

Reference:
Memorandum dated October 16, 1992, from Warren Minners, Office of Nuclear Regulatory Research, NRC, for Raymond F. Fraley, ACRS, transmitting Draft SECY Paper (undated) from James M. Taylor, Executive Director for Operations, for The Commissioners, Subject: Risk-Based Regulation (Predecisional)

OFFICE OF THE SECRETARY

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

March 26, 1992

IN RESPONSE, PLEASE REFER TO: M920310B

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON RISK-BASED REGULATIONS TRANSITION STRATEGY, 2:00 P.M., TUESDAY, MARCH 10, 1992, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed on a transition strategy to the use of risk-based regulations. The information was presented by:

Mr. John C. Brons
President and Chief Operating Officer
New York Power Authority (NYPA)

Mr. Herschel Specter
New York Power Authority

Mr. William H. Rasin
Director, Technical Division
Nuclear Management and Resources Council (NUMARC)

The Commission requested that the staff provide their views on the practicality of risk-based regulations and the feasibility of developing a transition strategy from deterministic based regulations. Consideration should be given to the need for a threshold which ensures some attention to low-priority items.
(EDO) (SECY Suspense: 12/18/92)

cc: The Chairman
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
Commissioner de Planque
OGC
OCAA
OIG
ACRS
PDR - Advance
√DCS - P1-24

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ITEM 8: Thermo-Lag Fire Barriers

In response to a SRM dated November 15, 1993, the Auxiliary and Secondary Systems Subcommittee held a meeting on November 19, 1993 to review the technical differences between NUMARC and the staff over NUMARC's test program for Thermo-Lag fire barrier adequacy. The ACRS reported the results of its review to the Commission in a report dated December 16, 1993.

In SECY-93-362, December 30, 1993, the staff indicated that it was reviewing the ACRS comments and would provide the results of its review to the Commission when completed.

The following document is attached:

- ACRS report to the Commission dated December 16, 1993. Subject: Thermo-Lag Fire Barriers (PP.97-98)

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.

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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

References:

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 Protection Task Action Plans

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