UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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MEETING WITH ADVISORY COMMITTEE ON

REACTOR SAFEGUARDS (ACRS)

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PUBLIC MEETING

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Nuclear Regulatory Commission
Commissioners Conference Room
One White Flint North
11555 Rockville Pike
Rockville, Maryland

Friday, May 24, 1996

The Commission met in open session, pursuant to notice, at 9:35 a.m., the Honorable SHIRLEY A. JACKSON, Chairman of the Commission, presiding.

COMMISSIONERS PRESENT:

SHIRLEY A. JACKSON, Chairman of the Commission

KENNETH C. ROGERS, Member of the Commission

STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

J. HOYLE

T. KRESS

G. APOSTOLAKIS

J. CARROLL

I. CATTON

M. FONTANA

W. LINDBLAD

D. MILLER

D. POWERS

R. SEALE

W. SHACK

C. WYLIE

P R O C E E D I N G S

[9:30 a.m.]

CHAIRMAN JACKSON: Good morning. It is always a pleasure to meet and hear from you, Dr. Kress, and other members of your Committee, the Advisory Committee on Reactor Safeguards.

MR. KRESS: Thank you.

CHAIRMAN JACKSON: We have quite a few topics of interest to discuss this morning. However, before we begin because I understand this is the last Commission meeting for ACRS member James Carroll, I would like to pause for a few minutes and to have Commissioner Rogers and me present you with two tokens of the Commission's appreciation for your eight years of service to the Committee and to the Commission.

So I would like first to present you in a plaque form with a copy of a letter of appreciation from the Commission and we have a photographer here.

MR. CARROLL: I had guessed.

[Laughter.]

MR. KRESS: That is why he wore a tie.

[Laughter.]

CHAIRMAN JACKSON: That's not all. It says, "Presented to James C. Carroll upon completion of eight years of exemplary service to the Advisory Committee on

Reactor Safeguards and to the Nuclear Regulatory Commission.

[Applause.]

MR. CARROLL: Thank you.

MR. CATTON: Jay, are you sure you are leaving now?

[Laughter.]

CHAIRMAN JACKSON: Over the years, the ACRS has provided valuable advice to the Commission on the safety aspects of the proposed and existing nuclear facilities and we always feel fortunate to be able to draw upon your expertise. Now I understand that today's briefing will cover the following topics, uses of IPEs in the regulatory process, fire protection issues, proposed final revisions to 10 CFR Parts 50 and 100, digital I&C, the ACRS review of standard plant designs and conformance of operating plans with NRC safety goals.

If that is your understanding, Dr. Kress, Commissioner Rogers and I are happy to welcome you to the meeting and look forward to hearing what you have to say. Before you begin, I would like to say that Commissioner Dicus is unable to join us this morning and she sends along her apologies. Dr. Kress.

MR. KRESS: Thank you and I don't intend to waste any time by making speeches. We will jump right into the agenda item and the first item is the use of IPEs in the

regulatory process and that is George Apostolakis' area.

MR. APOSTOLAKIS: Good morning. We wrote a letter on the IPEs dated March 8 and basically what we said there was that the program has been very successful in meeting the intent of the generic letter. Both the utility staff and the NRC staff have developed an appreciation of PRA methods now, all the units have done an IPE.

In addition, we were asked to provide comments on the possible use of the IPEs in the regulatory process which was not part of the original generic letter and the main problem there is that it is not clear to what extent the subjective judgments and use of methods and models have influenced the results.

In other words, if I look at an estimate of core damage frequency say from one particular IPE, I will have to go into the IPE and look at the details of the models to really be able to say that yes, this seems to be a realistic estimate again in the sense that it is up to the standards of the state of the art or it is driven by certain assumptions that perhaps are not justified.

What makes matters more complicated in my opinion is that the review process was done under very severe constraints and I don't know why that was but it certainly did not help in identifying again to what degree assumptions and judgments have influenced the results.

CHAIRMAN JACKSON: Could you be more explicit about what you mean there?

MR. APOSTOLAKIS: For example, there are certain areas in PRAs where there are a number of models out there, for example, the rate estimation, common cause failures and so on and as I understand the rules of the game if a particular licensee selected the method or a model that had been used by other PRAs or that was in a report from a major organization then all the reviewer had to do was to make sure that that model was applied as it was intended, the reviewer was never to question the model itself was applicable. That in my opinion is a severe limitation.

When I have participated in reviews of major efforts both by the industry and national laboratories and the reviews there were very different. Every model, every assumption was scrutinized and the analysts had to defend those. In this case, they did not have to do this.

Now with respect to their use, the Committee felt that there was a lot of useful information in the IPEs and that as issues now come up between the staff and the licensees and if the licensee chooses to use the IPE, then the staff would have to make sure that the methods and models that are being used for that particular issue are up to the current standards.

So we are not recommending a massive effort to

upgrade the IPEs, however if they are to be used for individual issues, then they have to be updated.

CHAIRMAN JACKSON: So let me paraphrase you. Are you saying that in order for a regulatory decision to be based on them, then for that particular decision that the model then should be examined that undergirds, the potential use should be examined from that perspective at least?

MR. APOSTOLAKIS: Yes.

CHAIRMAN JACKSON: All right.

MR. APOSTOLAKIS: Now with regard to risk-informed and performance-oriented regulation, we had two subcommittee meetings with the staff and we also had presentations to the full committee.

CHAIRMAN JACKSON: Let me back you up. We have always said risk-informed performance based so you are migrating in language so before you can tell us the results of your discussion, you have to tell us about your migration in language.

MR. APOSTOLAKIS: Why it is performance-oriented and not performance-based?

CHAIRMAN JACKSON: Why you are using that terminology.

MR. APOSTOLAKIS: I think the staff used it actually. I am not sure. There is no deep thinking behind it.

CHAIRMAN JACKSON: Fine.

[Laughter.]

MR. APOSTOLAKIS: Well, as we said in the letter there are several issues that are intellectual and practical issues that are very difficult and they have to be resolved. The staff, we felt, made a good starting effort with a framework document and the pilot projects they have selected. As usual, we had some comments.

The first one which I think is really a very important one is that we have to have a big picture as we embark on this new effort and by big picture I mean that we have the Commission's safety goals and then we have targets such as the frequency of large releases, core damage frequency, and now with the maintenance rule the licensees are allowed to determine their own performance criteria regarding trains or systems.

Somehow we have to know or we have to understand how all these things come together. If I have subsidiary targets, why this particular value and not something else, how does everything come together? If we think in terms of a level-3 PRA, go top down, how do all these things come together using logic to be consistent with the top level goals that the Commission has promulgated? So we asked the staff to do this or we actually suggested that they do it.

Then second, before again we start specifying

intermediate or subsidiary goals, it seems to me that we have to think about the philosophy of the whole approach. For example, shall we try to set those goals at the highest level possible and what does that mean or shall we say, "Well, it is easy to set the goal regarding the availability of this major component so we will do that." Well, maybe we ought to think about putting it at the higher level if possible and if we cannot, why not.

There are a few other principles that we have listed in our letter. I don't need to go through all of them. Now one major issue that arose was the issue of performance, what does performance mean.

In fact, that was one of our criticisms of the framework document, that we did not use the word "performance" at all. Now as we state in the letter there seem to be two extremes here and I am sure eventually we will settle on something in between.

On one extreme, we look at only hardware and we say, "Well, do we have any statistical records to support a particular estimate of availability or unavailability and that is the measure of performance." We don't have to use any models because we don't trust the models and so on.

On the other extreme, we have a group of people who believe that, for example, the core damage frequency could be a measure of performance if you also state what has

been left out and so on. So that would include more things than just the performance of hardware and a lot of people have problems with that, of course, because that gets into issues such as organizational factors and so on.

I don't think any one of us claims that we do know the answer. This is a very difficult concept but before we talk about performance-based regulation, we have to understand what we mean by performance or maybe define it, that in this context this is what it is going to mean.

Then with respect to the pilot projects, each individual project seems to be fine and it will be very useful but again what is missing is the big picture. Was there a design, what we call in statistics an experimental design, done beforehand to tell us, "Look, on the way to RIPOR, we will have these issues, we have these questions to answer and if we do such and such a project, we will get the answers."

Now we don't know that. We don't know that such a thinking actually took place and these pilot projects may overlap in certain things and they may leave other questions unanswered. So we would like to see again in the big scheme of things how these pilot projects will help us answer some of the questions that we expect will come up.

CHAIRMAN JACKSON: From your discussions with the staff on the pilot projects, is it your judgment that it is

possible to revisit these questions that are of concern to you in a way that what the staff intends to do relative to them can be quote/unquote, "backfit" for lack of a better term, to be able to address some of what you consider to be these difficult issues?

MR. APOSTOLAKIS: Well, the staff has told us that it is not easy to go back and establish new projects because it takes time.

CHAIRMAN JACKSON: No. I am talking about within the context of the projects as they are.

MR. APOSTOLAKIS: Oh, as they are?

CHAIRMAN JACKSON: Right.

MR. APOSTOLAKIS: What kinds of questions will be answered by these projects, you mean or change them a little bit?

CHAIRMAN JACKSON: What you thought may have been missing from the beginning, to what extent can one go back and try to address some of these issues? That is really the question I am asking you and if you made recommendations to the staff along these lines.

MR. APOSTOLAKIS: No, we have not made specific recommendations because I think it is not obvious what the issues will be so somebody has to sit down and think about them.

One obvious thing is that there is very little on

the level-2 issues, for instance. All the pilot projects really deal with level-1 issues. The other question is how do you define a pilot project that deals with level-2 issues because the issue of performance there is something that is not obvious. I mean, what is performance when it comes to level-2 issues?

External events, I don't think there is any project that really deals with that. But these are the sort of obvious ones. There may be other, more esoteric questions that will not be answered and maybe ten months from now we will find that boy, this would have been nice to have something on this. So that is what we meant by that.

Basically, I think this covers the highlights of what we have done.

CHAIRMAN JACKSON: Thank you. Dr. Catton, did you have a comment you wanted to make?

MR. CATTON: No.

CHAIRMAN JACKSON: Commissioner Rogers.

COMMISSIONER ROGERS: Do you think we ought to have questions on each of the individual presentations rather than have to go back?

CHAIRMAN JACKSON: Yes, I think so. Otherwise, it will hard to keep up.

COMMISSIONER ROGERS: I thought it was very interesting your comment about using IPEs for regulatory

purposes, about having to review them for appropriateness of the models that are involved if we want to make a decision of some sort but it brings me back to the standard review plan.

You did indicate that you thought that the standard review plan, a PRA standard review plan now being developed by the staff, can serve as a template for judging the quality and acceptability of individual plant PRAs for the proposed application.

Now how does that relate to your comment about the specific models? In other words, would the standard review plan as you understand it now being proposed by the staff involve a review of the appropriateness of a model?

MR. APOSTOLAKIS: Yes. I would like to see that, yes.

COMMISSIONER ROGERS: Has that been communicated explicitly to the staff in this regard? I think as they develop the standard review plan, it would be nice to see these points of view come together.

MR. APOSTOLAKIS: We have told the staff that we would like to see a list of acceptable and even unacceptable assumptions and models as part of this plan and they were a little bit concerned about the unacceptable assumptions but they didn't seem to have any objection to listing acceptable methods and models.

Now the reason why unacceptables is important is because after the review of all these IPEs, I think the staff now really know where the major pitfalls are, where people really can make judgments and assumptions that are really unacceptable.

Also, after 20 or so years of doing PRAs there is some standardization especially how to develop event trees and what assumptions to make there and the fault trees. So I think it would be useful and some of their statistical methods really were atrocious.

COMMISSIONER ROGERS: I am sure they could give some examples of unacceptable approaches but, of course, the list of unacceptables is infinite.

MR. KRESS: It is non-ending.

COMMISSIONER ROGERS: So it can be by example but it can be a definitive list obviously.

MR. APOSTOLAKIS: But they can draw on their experience from reviewing the IPEs.

COMMISSIONER ROGERS: Right.

MR. APOSTOLAKIS: That is really the point.

COMMISSIONER ROGERS: I just think that if you can convey those thoughts as the standard review plans are being developed, I think that would be very important and very useful.

The point I wonder if you could elaborate a little

bit on relates to page three of your Part B.1 and you list the RIPOR points and the fourth bullet there, "The relationship between RIPOR and defense-in-depth should be explained. The role of defense-in-depth in the determination of performance criteria to accommodate uncertainty and incompleteness in risk assessments should be established." I wonder if you could say a little bit more about that because the defense-in-depth features are part of what is included in the PRA. They are not an add-on. They are part of it.

MR. APOSTOLAKIS: Right.

COMMISSIONER ROGERS: So what do you have in mind there? I am just trying to grasp the concept a little bit better.

MR. APOSTOLAKIS: It refers to the determination of performance criteria of subsidiary targets. I don't think the Committee has a view on this because we have not really gone into details but I can give you my view because different people interpret this in a different way.

To some, defense-in-depth using PRAs means you look at what is left out that the PRA models do not handle and again the typical example is organizational issues and you say, "Okay, then I will do something about that independently of the subsidiary goals that I may have" because, for example, the core damage frequency does not

reflect that. It reflects it to some extent, of course, because of the equipment and so on.

Now in my view though there is another implementation of the concept that it is very important. Precisely because the PRA results are so uncertain, in fact, let me put it in a different way, if we had high confidence in level-3 estimates all we would need would be the Commission's goals, nothing else.

Then somebody comes and says, "Well, the individual risk if ten to the minus eight, compare it with the Commission's number, it is fine." But we know that is not the case. We know the uncertainties are there. So then we say that well, we would really like to have a goal on the large early release and then we realize even that is not enough and another goal on core damage frequency.

Some European countries, for instance, feel that that is not the way to do it, let's put the goals on the safety function frequency availabilities or we can go all the way down to systems and put the goals on the reliability of individual systems.

To me, that is an implementation of defense-in-depth in the probabilistic domain because of the large uncertainties in the result you say, "Well, I had better make sure that these critical intermediate events have low frequency themselves."

Now on top of that, of course, some of them are so important in themselves like core damage that you would like to have a goal there anyway. That was the thinking behind this and that is why we say to accommodate uncertainty which is what I just explained and incompleteness. So you look at both.

CHAIRMAN JACKSON: The incompleteness has to do with not having goals set at these subsidiary levels?

MR. APOSTOLAKIS: In risk assessment. So we know that the core damage frequency was not calculated including everything that is relevant so now we have to do something about what is left out.

CHAIRMAN JACKSON: All right. Commissioner Rogers.

COMMISSIONER ROGERS: That is helpful. I have to think a little bit more about it. I think there are interesting aspects of this but that was helpful.

MR. KRESS: The Committee is still batting this issue around. There are some thoughts we have discussed but haven't arrived at firm positions on. There are things like how does the apportioning between core damage frequency and early large release, that is a defense-in-depth concept and how does one arrive at the proper value for that.

Another concept we have batted around is there ought to be certain functions or systems like the safety

systems we now have, systems important to safety that could be declared as systems important to defense-in-depth and be excluded from the risk-based considerations when you go into a PRA and so forth. Just because you may find out one system by the PRA, if you exclude it or design it differently it may not have that much effect on risk but by intuition and by experience and so forth we know it is an important feature and we could exclude that from being treated in a risk-based space.

For example, just the containment or the shutdown systems that we have so we have to have those, we have to have redundancy, we have to have diversity, the various defense-in-depth concepts associated with them and it doesn't matter what the risk calculation tells you about them, we will have to require those anyway. That is a thought we have batted around but we haven't arrived at a firm position on these things yet.

CHAIRMAN JACKSON: So that means that it is a ripe topic for the next time we talk with you.

MR. KRESS: Yes.

CHAIRMAN JACKSON: All right. Why don't you move on and talk about fire issues.

MR. KRESS: Ivan, that's yours.

MR. CATTON: In response to your request we commented on the PRA fire model developed by BNL for

evaluating fire risk during a self-induced station blackout and the BNL scoping analysis of degraded fire barriers. We were not happy with either.

Further, it came as a surprise to some of us, not all of us but some of us, that a blackout would be self-induced. The first study focused on the effectiveness of the procedures to mitigate a given fire and did not address the probabilistic treatment of fires themselves.

Fire was taken as a given with a predetermined frequency. The scope of the study did not include a number of factors like human error that could impact conclusions. Here, I will let my colleagues expand on these issues if necessary.

My own particular concern is the lack of treatment of the fire itself. Certain conclusions are assumed about it and then it is put into a PRA framework. As a result, we don't believe one can draw substantive conclusions from the study. The uncertainty swamps the results.

The degraded fire barrier study suffered from similar oversimplification. The scoping analysis was oversimplistic in that fire propagation, detection and suppression were not considered. This ignores the fundamental competition between time to damage and time to detection and suppression.

It was simply assumed that the fire barrier had a

given probability of failure. The time to damage would have entailed estimating the impact of a fire on a protected cable tray or other target.

One could estimate the probability of a fire of a given magnitude and the probability of an important protected target being in the vicinity. The calculation tells you whether or not it might be lost. One could then relate a given amount of protection to risk of loss of the target and this was not done.

It is my view that risk-informed fire regulation will require more than what was done by BNL. A number of issues ranging from approximate appropriate fire initiation data base that includes ignition and fuel separately, the modeling and thermal physical data will have to be obtained or addressed somehow. At present, I do not see these issues being addressed by NRC or the industry.

CHAIRMAN JACKSON: Commissioner Rogers.

COMMISSIONER ROGERS: Are you saying that you think that more studies have to be done and more data has to be accumulated?

MR. CATTON: I think a lot of the data is out there. For example, when you look at -- and I need to be careful moving into the PRA arena, that is not my ballgame, but my observation is that the fire frequency data is around but what has not been done is it has not been split into

probability of an ignition source and probability of fuel being in the vicinity because they are really two separate issues.

As a result a lot of the fire frequency data is inappropriate and I think that really needs to be addressed. What is really lacking, too, is given that you can treat that probabilistically which is what you probably have to do, I think you need to do some kind of computation. You can't just put numbers into a PRA and expect the bottom line to be of substance.

CHAIRMAN JACKSON: Are you saying that you need to do some kind of computation?

MR. CATTON: There is phenomena occurring, there is phenomena that we understand that is occurring and I think that needs to be incorporated into the PRA in a meaningful way. This is not happening.

CHAIRMAN JACKSON: How does the BNL fire risk modeling compare with what you see in the IPEEEs?

MR. CATTON: That is a question I have to defer to my colleagues. I haven't looked much at the IPEEE but I think both Bill Lindblad and George could address that.

MR. LINDBLAD: The IPEEE is operating on a review schedule that follow IPE for internal events and we have just finished the internal event review. We really haven't seen much of the IPEEE results or review from the staff so I

really can't say.

MR. APOSTOLAKIS: The study that was presented by BNL was not a new fire PRA model. They used some of the existing models to address a different issue, right, the self-induced SBO.

MR. CATTON: Yes, that's right.

MR. APOSTOLAKIS: So we can't really comment on something as being a new methodology for doing fire risk assessment. So they used simplifications of existing models. Coming back to what Professor Catton just said, I do agree that there is a need for some additional modeling.

MR. CATTON: But not fully.

[Laughter.]

MR. APOSTOLAKIS: But there are some parts of that issue, I believe, of the whole assessment of risks from fires that will never be resolved. The issue of large fires, for example, the issue of having the fire in the right place. In my opinion, these are the weak spots of the analysis. Unfortunately, they are necessary. In other words, you can't assume that a fire anywhere will do damage. It has to be in the right place. Now you can't expect to have statistical information on that. So in my opinion, that is something that will always be with us.

But I do agree with Ivan's recommendations because even the data, I mean we keep hearing about the data doing

this and that and I don't think we ever had really an opportunity to look at the data, what is available and pass judgment.

CHAIRMAN JACKSON: How much and what we have here is fairly generally written, how much specificity have you passed back to the staff relative to what needs to be done to move forward?

MR. CATTON: We have had these kinds of discussions with the staff but the only thing that gets passed is the letter. It is formal. Informal, we have had a lot of discussions.

CHAIRMAN JACKSON: I see. Commissioner Rogers.

COMMISSIONER ROGERS: You mentioned the work done by the insurance industry. Has that been directly related to nuclear power plants or are these more generic studies?

MR. CATTON: It has not been directly related to the nuclear industry but I don't think that the phenomena changes because of one place or another. Your boundary conditions and other things change and what you calculate changes depending on what the source, the transport process and the target are but the methods and the physics don't change.

COMMISSIONER ROGERS: No but I am really talking about the accumulation of data that is appropriate.

MR. CATTON: Oh, there is quite a bit of data that

is appropriate to the nuclear industry that is available like cable trays. Cable trays are not just in nuclear power stations so there is a lot of this kind of data and I think there should be some effort to gather this together and somehow ferret out what is important and begin to bring it to bear.

COMMISSIONER ROGERS: It seems to me that what you are saying is that we really haven't really taken a comprehensive look at this problem. It is sort of like your comments that Professor Apostolakis made in his remarks, that we need to really get an overview of the situation and as we approach it, it looks like there are bits and pieces that have been done but not a comprehensive look in some way.

MR. CATTON: I am going to speak a little bit out of the area that I probably ought to but the fire PRA itself, the structure, to me looks all right. Where the thing starts to come unstuck is when you say, "Well, gee, I have a barrel of something or other in this room, what is it going to do?" There the analysis process gets very weak and I think more needs to be done to bring this thing together. What are you going to have in a given room? That is always going to be probabilistic.

But you ought to be able to assume something and you certainly can go in and say, "Gee, the maximum I could

have is, it could burn instantaneously or it could burn at some rate" and you could begin to put these things together.

As long as we just do this sort of overview and stuck numbers into a matrix, I think you are going to have a result that is not believable, at least not believable by people like myself. You have to put some substance into it somehow.

COMMISSIONER ROGERS: Do you think that the result of that is always going to be an overly conservative approach or not?

MR. CATTON: It could be either way. Let me give you some examples. I have heard some interesting things about cable trays and how they burn. They burn through walls where they are supposed to not burn. They burn down cable trays. All sorts of things happen.

The results that are coming out of the HDR containment in Germany say that we really don't know how these things burn. We don't know how to relate a given heat source to the evolution of gases that will burn. These are just simple physics but you need to look at them and until you do, whatever the PRA practitioner is going to do is going to be some sort of an estimate or guess.

Now is he going to be high or low? You can get examples both ways. I think you don't know and that is sort

of what led us to statements about uncertainty which we sort of put down whenever we can.

CHAIRMAN JACKSON: So you are basically saying that the underlying analysis, physics, engineering that would really need to undergird and make this a robust or end up with a robust technique is just not there.

MR. CATTON: It is weak.

CHAIRMAN JACKSON: Weak.

MR. CATTON: It is very weak.

CHAIRMAN JACKSON: All right.

COMMISSIONER ROGERS: I don't dispute that. The question is, what is the possibility of really finally coming to closure on something like this and whether an analytical approach is the best way to deal with it or perish the thought, a purely regulatory approach that says, "You shall not have in this area certain things" and you don't know whether they are going to be there or not but if you make a rule that says they can't be there, then at least you have some little confidence that it is unlikely that they will be there. It doesn't totally exempt that from happening as we found to our dismay sometimes.

But in other words, is this an area where it is not purely engineering to solve the problem in that you can engineer a system that you feel is going to have a bottom line PRA number that you believe in and feel comfortable

with or will you have to mix this with some requirements that just take care of those aspects of it that are going to be somewhat uncertain.

MR. CATTON: But the first thing you probably ought to do is find what those things are whose uncertainty you can't reduce and then you have to deal with them but to even think that you might be able to do it completely analytical, I think, that is foolish. You just can't do that.

You can in a simple room with something nice but as soon as it gets complicated, you can't. That doesn't mean that you can't do engineering kinds of calculations for this complex system. We do that all the time. What I don't see is an incorporation of this into the PRA structure and I think that is what is needed.

CHAIRMAN JACKSON: That is also probably true at the level of PRAs that are not fire.

MR. CATTON: I wouldn't just focus on fire. It is just that the fire PRA is kind of interesting in that over the years we have been exposed to people who don't believe the results of the fire PRA because they say that the numbers are too pessimistic and there are other people who believe the other way and yet when you look at the PRA, the way it sits in front of you, you can't put your finger on what the reason is except some don't believe and some do

believe.

I think if it is going to be risk-based, you have to develop a risk-informed or whichever word, you have to develop faith in this tool that you are using. Now how do you do that? I think you have to remove the numbers that are stuffed into it when you can.

COMMISSIONER ROGERS: Are there any data similar to epidemiological data in other areas that tell us what kind of fires actually do occur in nuclear power plants? We have several hundred now around the world operating for several decades. Is there any comprehensive data base on fires even little ones that one might be able to look at in terms of what seems to have actually happened?

MR. CATTON: I think that is a necessary step is to do that. There have been some pretty exciting fires. I mean the one in India that you probably know about and there was one in Spain and they have been here, there and everywhere. Somehow these ought to be brought together.

CHAIRMAN JACKSON: Not to mention Browns Ferry.

MR. CATTON: Yes.

MR. CARROLL: Sandia does maintain such a data base and so does EPRI. The Sandia data base to me is as Ivan described it overly pessimistic. A wastebasket fire turns into a core damage event very quickly in the approach that they use.

MR. APOSTOLAKIS: I think if one goes only by what has happened the fire issue essentially goes away.

MR. CATTON: Overstatement.

MR. APOSTOLAKIS: The PRA guy has a problem there because you start out by identifying what we call critical locations, where control cables come together or power cables. So if you go into a room that is under strict administrative controls, the regulatory part that you mentioned earlier, and you identify a location like that.

You do your simple calculations and you realize that under normal conditions, there will never be enough fuel there to cause any damage. So what do you do now? You screen out the location or you look at the evidence again where administrative controls have been violated occasionally.

So you say, "Well, gee, there will not be under normal conditions five gallons of oil or the equivalent. However, I cannot exclude the possibility." So now you have to put in an additional probability there that this amount of fuel will be there and that is when the debate begins and that is why a lot of people don't believe the results. They say that you will never get that fuel there to do all this damage that you are calculating later.

In my opinion, that is an unresolved issue. The evidence cannot resolve it because you do have over the

history of nuclear power, you do have incidents where people found unauthorized amounts of fuel in areas where they were not supposed to be. In fact in one case some inspector from the insurance industry was telling me that he found dynamite that was left there overnight because it was raining outside.

I don't think that any statistical evidence will help us resolve that issue and these are the various factors that I mentioned earlier in addition you have to say that it is exactly what it is supposed to be to do the damage and so on.

Now the second part, I think the physical models, I thin it is interesting to note that the basic tool, computer tool, that is being used right now is the Masters Thesis of a student of 16 years ago and what needs to be done is to revisit that because the fire safety community has done a lot of work outside the nuclear arena developing models and doing experiments and so on and I think what needs to be done is for someone to put together a new model that uses the latest and the best available models.

I don't think we need to start from scratch. I don't think we need a major research program that starts with experiments and let's understand how this works unless we have deemed that information critical to what we are doing and it is not available anywhere else. I think that

is very important.

I have dealt with several fire safety researchers in other fields and it is true that the field has advanced tremendously in the last ten or 15 years. So I would not criticize the existing model as being inadequate and all that because it was never intended to be used by the whole industry. All of a sudden during the Zion and Indian Point PRAs we found out that fire was important and people said, "Well, what do we do? Well, there is this model, let's use it" and it acquired a life by itself but it was never really a serious major effort to develop a tool to be used by an industry.

That historical background, I think, is important and a lot of people have criticized it but I don't think there have been any advances.

MR. CATTON: Just to pursue this a little further, the evaluation of a number of models that took place in Germany, the result was that the biggest problem was the thermal physical properties of the cable itself. How much gas was released when you heated it? Simple things like that and if you can't get the inputs, you can't get the answer.

So some of the basic data is missing and if you look back, you will see that people just didn't measure that. They took a torch to it and it was more qualitative

than quantitative.

These are not hard things to do but they just haven't been done. So part of the data base is incomplete and again, I would agree with George. It shouldn't be a major research effort but there should be some sort of focus on where are the weak links in putting this thing together.

CHAIRMAN JACKSON: All right. Dr. Kress.

MR. KRESS: The next item is the proposed final revisions to 10 CFR Parts 50 and 100 and Bill Lindblad will address that.

MR. LINDBLAD: Thank you, Tom. As you know to a large degree the revisions had to do with reordering where certain provisions would be found and that in itself didn't involve safety issues.

The safety issues that arose had to do with geotechnical considerations and in that regard the revisions really reflected the current state of the art and what had been previously accepted by the staff and the committee with regard to doing probabilistic studies for sites particularly in the eastern United States where it was difficult to identify tectonic structures that would be of interest.

The Committee found that that was certainly representative of what we believe to be proper safety. The revisions did incorporate perhaps a bias by referring to if one were to select a new site, hopefully it was a site with

low probabilistic risk rather than the field that the current plants are in and how that will work out in the future time will tell whether ones will seek those out preferentially or not.

In regard to the other safety aspects of the revision, an important one was incorporating the new source term that had previously been reviewed by the committee and found to be more realistic, more mechanistic and was considered to be a substantial advance in safety evaluation for reactors. A particular issue that came up about using the source term and the dose criteria associated with it is what window of time would be used to evaluate the maximum dose. For some reason that none of us have been able to hammer out a two-hour window has been used in the past but which two-hour window.

There were as you know alternate approaches proposed to the Committee and we listened to both arguments and our letter states that while there was not a great deal difference in the risk profile, we did feel that the main proposed provision of any two hours or the worst two hours, however you choose to do it, was preferable.

CHAIRMAN JACKSON: A question I had for you on that issue and as you say picking say two hours as opposed to three, these things have historical precedent but nonetheless, it seems that there is this in-house difference

of opinion in terms of the first two hours after fuel failure versus worst two hours and one could argue that the one relates more to design and the other relates perhaps to issues related to emergency response because one could say on the one hand if one focused on the worst two hours, that that has clear design implications for a facility.

On the other hand, by focusing on the first two hours that does address the issue of when people are most likely to be around and it has implications in terms of emergency response, people getting to the site as well as evacuation issues in terms of how a dose would build up not to mention that the worst two hours if you are looking at it from the point of view of dose to an individual has to assume that the person comes in, say if it is at the fourth to sixth hour, at the fourth hour and leaves at the sixth, so the issue almost then and maybe this muddles the issue but I am interested in your thoughts, how could one from a public policy perspective not say look at an integrated dose over a larger window that would incorporate the time when one would think that there is dose, but if there is dose and no one is there and there may be less dose but people are there, more likely to be there, why should not one do a calculation that is an integrated calculation over a larger window that takes that into account?

MR. LINDBLAD: I think that is done, of course,

for the population dose.

CHAIRMAN JACKSON: That's right.

MR. LINDBLAD: That is done. But as you point out in the early response to a casualty there is a great reliance on what is already in place called the design provisions of the plant and as one goes farther down the time scale, there is a presumption that the planned emergency response will be put into operation by the licensee as well as the governmental bodies and this agency and that has been thought out recognizing the specifics of the plant and measured up against the society's desire to protect its public. So I believe it is done in that regard but Dr. Kress is our expert in this area and you should hear from him.

MR. KRESS: If I may, thank you, this is one of the places where we get a collision between design basis accidents and reality. Design basis accidents are historically there so that if you go by them and use them to design your plant and features in it, then you will end up with a plant that will be over all safe for the whole spectrum of accidents. One should not confuse the source terms used for design basis accident with real source terms although they are intimately related and one should have some relationship to the other.

When it comes to emergency response type

activities, one ought not to deal with design basis concepts. One ought to use the PRA, real accident spectrums. One ought to formulate his emergency response plans based on what real accidents might occur.

So the source term there ought to be realistic source terms that involve the full time spectrum, the full spectrum of accidents, the types of things one might get and that ought to be the way one deals with emergency response.

Going back to design basis, there is a weak link between the actual risk one ends up with and the form of the design basis accident one chooses. There is a link but it has never been established if you follow this type of design basis considerations, you will end up with a plant that has this level of risk. That nexus has never been made.

We have empirical evidence that it has worked because we have plants that we have now done PRAs on and IPEs, et cetera, and we say, "Well, they are very safe. They meet the safety goals and it is because they are designed according to these design basis concepts." So there is empirical evidence that it has worked.

But one cannot look at these design basis things and say that it is because we did this form of the source term or it is because we had this, we had that, it is the whole bunch of it taken together. So one ought to be careful about trying to use design basis things for real

risk-informed decisions. That was my problem with it.

MR. CARROLL: I would have to add that using the design basis prescriptions, we find sometimes have actually produced a less safe plant.

MR. KRESS: Absolutely.

MR. CARROLL: That is also something to be concerned about.

MR. KRESS: Some of the design basis concepts allowed us to end up with if I may say so an ice condenser containment which personally I don't like and it is because it is allowed within the design basis concept.

MR. CARROLL: Or diesels that were forcing to start too fast or isolation valves.

MR. KRESS: Or valves that were forced to close too fast and those things actually increase the risk.

MR. CATTON: In the fire business, this is called magic numbers and golden rules.

MR. CARROLL: Since we are on fire protection I did confirm something, I believe. The graduate student that invented this thing was one of George's.

[Laughter.]

CHAIRMAN JACKSON: He was speaking so knowledgeably I had that feeling.

MR. CARROLL: I just wanted to make sure you knew.

[Laughter.]

CHAIRMAN JACKSON: But net-net, you support the idea that the radiological doses, the evaluation of radiological doses, should be for the worst two-hour time period.

MR. KRESS: Yes, because it ends up with a more robust design and in design basis space, I think you are looking for that attribute.

MR. LINDBLAD: I would like to comment that while we approved that we did recommend that the careful definition of the total effective dose equivalent limits should be consistent with the way the organ dose weighting factors are found in Part 20 of the regulations that we think that there is an appropriate consistency that should be applied but otherwise, we accepted the basic proposal.

CHAIRMAN JACKSON: Does that mean then that you support the 25 rem?

MR. LINDBLAD: I don't know what number it would work out but whatever the number is, it ought to be consistently used with Part 20.

CHAIRMAN JACKSON: Commissioner Rogers.

COMMISSIONER ROGERS: Isn't the TEDE a well-defined entity now? Whatever it is, it is defined. Are you suggesting that it should be adjusted in some way?

MR. LINDBLAD: There was some statement I believe from industry that suggested that the development of the

specific number that was used in the discussion used an organ dose weighting factor different from that of Part 20. We didn't really determine that ourselves but on just the statement that it was, we suggested that it be consistent. We have not yet a staff's response to our letter. Maybe they will explain the inconsistency at that time but we haven't followed up on it beyond that.

CHAIRMAN JACKSON: All right.

COMMISSIONER ROGERS: Fine. You recommended issuing the rule on the seismic aspects in your letter of April 22 but does that mean not the siting and source term aspects? Does that mean only the seismic aspects?

MR. LINDBLAD: I believe in developing our letter we partitioned the discussion into first seismic and geologic and gave that a do pass and then we approached the radiological and it may be that our letter was inclusive but we did intend, I am sure, we intended that the material, that the final rule be processed.

COMMISSIONER ROGERS: Including all aspects?

MR. LINDBLAD: Yes with the recommendation on the TEDE limit which we are not sure of but we would hope would get refined.

COMMISSIONER ROGERS: I see. You are asking for a review of that before it is issued.

MR. LINDBLAD: We recommended one, yes.

COMMISSIONER ROGERS: I see. All right. Thank you very much.

CHAIRMAN JACKSON: Dr. Kress.

MR. KRESS: Dr. Miller, the next item is yours, digital I&C.

MR. MILLER: Thank you, Tom. The item here is the status of review of regulatory guidance on digital instrumentation and control systems and the primary activity right now, of course, is the review of the standard review plan and I want to remind the Commission that that plan is actually one that codifies currently regulatory guidance into one single document and really kind of updates it and there is guidance on digital I&C upgrades in the form of the generic letter 95-02 which was in April of 1995 in which that endorsed the EPRI guideline which provides guidance through the 50.59 process for I&C systems in current operating plants.

Now we have completed review with the staff over the last couple of months or the last month, in March and in our most recent meeting, four sections of the standard review plan and also two branch technical positions and a number of regulatory guides.

I want to bring up one point. There is a high probability, I think someone said 99 percent, that a branch technical position will be actually dropped in lieu of a

safety evaluation report which will endorse another EPRI guideline and that is in the area of commercial off the shelf software and that guideline is being developed on the model of the previous guidelines on I&C and one we recently approved and that is in the area of EMI RFI. I think that gives you a look to the future and I think is another example of collaboration and cooperation amongst industry, EPRI and, of course, the NRC staff.

The regulatory guides which are listed in your briefing book have been reviewed and essentially completed by the Committee and these guides have the objective of actually supplementing or providing additional guidance on other regulatory guides which have already been endorsed.

That is regulatory guide 1.153 which is safety systems in nuclear power plants and then 1.152 which is safety system computers in nuclear power plants. So those are meant to provide additional guidance.

The plans as we look to the future are that we will complete our review of the review plan in meetings in August and in September or October and in parallel with that, we will expect the Committee report from the National Academy Study, its Phase Two report, to be also reviewed and we will attempt to incorporate all that together in the October meeting of 1996.

So I think we are pretty much on schedule as we

expect and I believe that it is being done as rapidly as possible. As I kind of closing comment from my point of view, not everything is done yet.

The Committee during the course of the review, individual members have raised issues which I think need to be debated amongst the members of the Committee and we have not brought closure on any of these issues.

I want to make that point and it will be subject to debate over the next several months and those issues include a concern about what I would say is the level of detail provided in the regulatory guides, the lack of guidance on a graded approach, the approach which tends to emphasize in some members' minds process over product and I say that because other members don't all have to agree to that and finally the very generic concern expressed by a couple of members and that is the generalized use of industrial standards as a basis for regulatory guides.

CHAIRMAN JACKSON: Generalized use of what?

MR. MILLER: Of industrial standards as a basis for regulatory guides. Of course, that has been the policy of the NRC since, it goes back to IEEE 279 which is actually incorporated into regulation but from that point on, the majority of regulatory guides in I&C and other areas have used industrial standards as their basis.

CHAIRMAN JACKSON: Why is that a generalized

concern?

MR. MILLER: I guess at this point I was going to invite Committee members to make comments on that because that is not my concern. I would rather others speak to that.

MR. LINDBLAD: With me, it is not an issue of the use of the standard, of an industrial standard, I think that that would be appropriate. I do believe though that the regulatory agency has to identify a rationale for why the standard that is used in the industry is necessary and sufficient to meet the regulatory need.

I guess I have observed that were the standard to be submitted by a licensee to the staff for approval of use, it would result in the staff preparing a safety evaluation report which deals specifically with the issue of necessary and sufficient.

When the Agency on its own initiative decides to endorse an industrial standard, there doesn't appear to be a document equivalent to the safety evaluation report that is as explicit in why the standard meets the requirement.

Now it seems to me that many of the standards we endorse, we are endorsing because they represent best practice and to me, we ought to acknowledge that we are endorsing it because it meets best practice.

On the other hand, there may be some standards

that are intended to resolve a specific safety issue and one that comes to mind is the ANS standard on decay heat released from fission, that is an issue and we have endorsed it and frequently used a 20 percent factor to be sure it is right.

MR. CATTON: To be sure it is conservative.

MR. LINDBLAD: To be sure it is conservative, yes. There may be other standards that are intended to solve a specific safety issue and one should either identify whether we are endorsing it as representing good current practice or we are representing a specific safety issue that needs to be resolved.

CHAIRMAN JACKSON: Let me ask you this. Are you saying that your historical experience has been that such safety evaluations are not done and therefore, there aren't SERs that show that or that there haven't been documentation of what safety analyses and how they relate to regulatory requirements?

MR. LINDBLAD: Basically the Committee worked some documents and testimony and it has only been recently and actually one of the other members has raised the issue and it has appealed to me when he raised it and Dana will speak to it shortly but I believe that it would be appropriate to see a document that looks like a SER.

MR. POWERS: I think Dr. Lindblad has explored the

issue well with you. I would just add into it that we need to be careful when we get this industry standard that is labelled a consensus standard to make sure that we have indeed explored the range of technical opinion and not just a narrow portion of the community that can participate in the development of the standard and that we have, in fact, taken the best practice that really does exist out there and are confident that we have adopted a standard that is going to serve us well and accomplish what we think it will accomplish. That, he thinks, as he said I think that can be accomplished by an explicit safety evaluation report on what you want the standard to do and why you think it will do that and why that is enough.

CHAIRMAN JACKSON: From the point of view of safety?

MR. POWERS: Yes, that's right, from the point of view of safety.

CHAIRMAN JACKSON: Dr. Kress, you had a comment?

MR. KRESS: No. I think this has covered our views quite well on this subject.

CHAIRMAN JACKSON: All right. Do you want to go on?

MR. KRESS: Yes. I guess the next item is the status of our reviews of the standard plan designs and Mr. Carroll is going to lead the discussion on this item.

MR. CARROLL: I am looking at page 34 and that is a summary that was written last week about the design certification rulemaking which has become inoperative since the time it was written.

I guess we have learned that General Electric has submitted ten design changes that the staff currently has under review. I believe ACRS has a statutory obligation to hear about these and write a letter on them on the basis that these design changes are advertised to us as being safety significant and we originally signed off on the final design approval.

We also understand Combustion is at least considering submitting some additional design changes. So I guess we are back in the mode of waiting to see the staff's safety evaluation or supplemental safety evaluation or whatever they call it on the GE ones at least before we will be able to comment on the design certification rules.

CHAIRMAN JACKSON: I would hope that working with the staff as much as possible if you could provide your views before the now rescheduled Commission meeting which has been rescheduled for late August.

MR. CARROLL: They have rescheduled again.

CHAIRMAN JACKSON: Well, it is at the moment rescheduled for the last week of August.

MR. CARROLL: Right, August 23.

CHAIRMAN JACKSON: So basically you are not at this point prepared to give any more specific comments.

MR. CARROLL: That's right.

CHAIRMAN JACKSON: Commissioner Rogers.

COMMISSIONER ROGERS: No, I don't have anything on this.

CHAIRMAN JACKSON: Dr. Kress.

MR. KRESS: There are other parts to this section and I guess Mr. Lindblad is going to cover the next part of it.

MR. LINDBLAD: I am the subcommittee chairman for the Westinghouse standard designs. We have had one previous review meeting on it some months ago and we anticipate that there will be another one next month on the specific issue of level-1 PRA. Most of the activity that the Committee as a whole has been doing is in regard to the thermal hydraulic response of a passive plant and its computer codes and here I am going to defer to Dr. Catton who leads that subcommittee to discuss that. Dr. Catton.

MR. CATTON: Thank you. I will try to just summarize where we are at for both the AP600 and the SBWR. Westinghouse has an experimental program that encompasses a number of facilities with various degrees of design sophistication. Some are well scaled and some are not. All have defects of one type or another.

The data resulting from testing at these various facilities is supposed to support the thermal hydraulic computer code V&V. Establishing whether or not the data base is sufficient is not a trivial task.

At our recent Committee meeting, we concluded that what we would need to be sure the data base was complete enough before proceeding to the codes themselves, that this needed to be done and this will entail an full review demonstrating how all the pieces fit together.

You have one facility, something is missing, where do you pick it up in order to establish some sort of measure of sufficiency of the data. Westinghouse has committed to do this. I don't know if they have formally committed but at least at our subcommittee they did and it is my understanding that the staff expects them to do this also.

Now the computer codes, first is COBRA/TRAC and that has been around for a long time and the version that Westinghouse is using is a modification of one that was actually developed by NRC and Westinghouse plans to use this for the large break LOCA ECCS evaluation. We reviewed the code for application to existing plants and we don't see any surprises so we probably won't have anything more to say about it.

For the small break LOCA, Westinghouse plans to use a different code. It is called NOTRUMP and it will be

used as an evaluation model meeting Appendix K requirements which is not best estimate; it is fairly prescriptive. We know very little about this code particularly today's version and its application to AP600. They supposedly have documentation in the mail. We'll see.

Long term cooling analysis will be based again on COBRA/TRAC calculations. We have some concerns about this because it takes so long to complete a calculation that Westinghouse will probably not do a thorough job and I forget the estimate but it was on the order of a couple of months of continuous computation to get one circumstance run to completion. It is just the wrong computational tool for the job and I can go into more detail on this if you wish.

CHAIRMAN JACKSON: Are there better computational tools that exist?

MR. CATTON: Well, I think just doing a quasi-steady analysis would be the thing to do and to exercise a computer code that is inappropriate for the job to me is just foolishness but it is their money and their computer, I guess. The problem is these big codes were developed for the large break LOCA which is a very fast transient. You do special things for the numerical algorithms. You can get away with a lot of things too because the forcing is so strong, 2000 psi when you start.

When you go to low pressures, long term, slow,

subtle balances between buoyancy forces and other things, you just have to do things differently or should and they haven't done this.

Now the containment. Here the data base is weak and the modeling is inadequate. The concerns evolve about the existence of thermal and concentration stratification within the containment itself.

From what we have seen, I don't believe it has been adequately measured and further that the test facility has been properly scaled and the computational procedure that they are using is based on a code called GOTHIC and it is a lumped parameter code and this type of code will not do the job when it comes to calculating stratification. It just won't.

The thing is, is that it is a huge building and by the time you do it properly, there is a price you have to pay, small nodalization for the accuracy and they are just not doing that.

To summarize our plans in this regard, we will meet with Westinghouse when they have put together a story supporting the view that the data base is complete or sufficient and then this will be followed by a review of the codes themselves and our progress is going to be based by the staff. We decided that we wanted to have in hand a draft SER before we meet with Westinghouse again.

Now the SBWR is a little different. It has been some time since we met with GE. We commented on their code and we had a rather bleak view. They, of course, explained to us that that is because what they were putting up on the screen was not what was in the code and the code was really all right but in any event, we were not very happy.

We met several months ago to discuss the test results obtained from the PANDA facility and the scaling of the PANDA facility and we were impressed both with the scaling effort and with the data that came out of it.

At the outset we were concerned that this condenser type heat removal system would get blocked by nitrogen and we had a lot of discussions with GE on how they ought to run their test.

Well, they out-did us a little. They started it with, I believe, 100 percent nitrogen and it worked. It performed its intended function. My reaction at the time was I don't really need to hear any more.

Of course, depending on what GE does we will do it and write a letter on our views. We have not had a response to our concerns about the code and we have not commented on the final draft of their test and analysis program.

By the way, this test and analysis program is sort of an overview of how they plan to put it all together. This is what we are waiting for also from Westinghouse.

MR. CARROLL: How they planned to put it all together.

MR. CATTON: T-A-P-D, test and analysis program --

MR. CARROLL: No, I am just making a point.

CHAIRMAN JACKSON: E-D, making the plant past tense.

MR. CARROLL: SBWR is past tense.

MR. CATTON: Well, I don't know. GE has requested that we comment on it and I don't know what we are going to do. It is going to depend on the staff because we probably won't do anything until we are requested to do so. If SBWR is going to disappear, we probably ought not bother with any more review.

CHAIRMAN JACKSON: Commissioner Rogers.

COMMISSIONER ROGERS: I don't have a question but I do think your comment on when you had the real data from the PANDA facility how your concerns then were totally allayed. I think it is very important to keep in mind that real data is terribly important and you can do all the computer runs you want to in the world but if you haven't got data that validates those, there is lots and lots of questions.

MR. CATTON: That's right.

COMMISSIONER ROGERS: Just the importance of

having experimental data.

MR. CATTON: You need to have data and you need to demonstrate that the data is appropriate for what you are going to do and this always involves scaling up a tremendous ratio and at this point GE has done a reasonable job of doing that. Their scaling analysis was quite complete. We had a lot of complaints about details but overall, it looked good and the results looked good. We were quite pleased.

CHAIRMAN JACKSON: All right. Dr. Kress.

MR. KRESS: The last item on the agenda is mine and it has to do with possible extension of the IPE/IPEEEs to see if one can determine whether the set of plants, how well they conform with respect to the safety goals.

Of course, you realize that the IPE/IPEEEs were never intended for this so what I say is not a complaint against those. They are really just not up to it because most of these PRAs for the IPEs and IPEEEs did not include fire, seismic in a risk sense, they used the FIVE analysis and a margins analysis and they didn't include shutdown risk. Some did not even do a level-2, most did and hardly any did a level-3 and the safety goals are, of course, a level-3 concept in risk.

In order to make this comparison, you do have to basically have a full level-3 PRA analysis that is acceptable and there are some questions of acceptability for

the standpoint, too.

So the question was can you possible bound some of these things and still make use of the IPEs to give one an idea and that was the intent of the study we heard about during yesterday's meeting of Brookhaven National Laboratory as part of the insights program to see if that could be done.

As far as it went, their study did a very nice job. It was just incomplete in the sense that they didn't even pretend to figure out how to treat fires or seismic or shutdown risk.

They did do some nice things on incorporating those plants that didn't go to a full level-2 in terms of trying to estimate what the early high releases would be and they did do some nice things in trying to incorporate site specific meteorology and population.

So it was a nice study as far as it went but we thought it was still incomplete enough that the use of it to infer whether or not the plants meet the safety goals is still going to be problematic.

We had a study done by one of our fellows that discussed how one might incorporate fire, seismic and even shutdown risk in a bounding way and it was a nice study. There is a reference to it in your handout.

The question is should the BNL study, for example,

be extended to incorporate these things and our overall feeling was that it probably is not worth the effort, that the results are still going to be highly uncertain and problematic and will have all the deficiencies and shortcomings that we have discussed with the IPEs in terms of modeling and so forth and that we probably already know enough from what is done already with the IPEs and with the insights program and with NUREG-1150 to infer with some confidence that most of the plants do meet the safety goals.

Now there may be some outliers that still don't but on an average which is what the safety goals were intended to talk about, they probably do meet them and we won't really know this for certain and we won't really know which plants are the ones that don't and which ones do until we have available full level-3 PRAs for each plant that is acceptable and includes all these missing parts.

We think the Commission probably ought to think about a first step in that direction in possibly extending the NUREG-1150 study. This was really a monumental study that was a very nice effort that did state-of-the-art work and we think some effort to include the seismic results and to include shutdown risk would be probably justified just to get a handle on those things for the same five surrogate plants.

So we think that would be a thing to do but as far as having full level-3 PRAs for each plant which is what you basically need, that is a huge effort and we think that wouldn't be needed in the long run for any kind of risk-based regulatory system.

That is a basic tool that one will need but it is not something that we think should be mandated. We think that is up to the industry to come up with that, to upgrade their IPEs to that level and that it probably will take a lot of time and will come about when the licensees come in for use of these IPEs for some sort of regulatory relief. At that time, they will have to come in with an acceptable IPE and over time these things will probably get upgraded to an acceptable and updated level.

CHAIRMAN JACKSON: An acceptable and updated level.

MR. KRESS: Yes. They use them in their maintenance program and their outage planning programs so eventually it would come about and I don't think it should be mandated and I don't see any urgent reason for us to rush out and try to see or get to the level where you can really make the judgment as to which plants and which do not meet the safety goals and whether they do on the average, I think we can be pretty confident that they probably on the average do and eventually we will have that answer if we wait long

enough.

CHAIRMAN JACKSON: Nonetheless, you have made the argument that there should be perhaps a restatement of Commission policy to allow the use of the safety goals on a plant specific basis?

MR. KRESS: Yes.

CHAIRMAN JACKSON: So that would be consistent with that.

MR. CARROLL: To allow the use of some form of safety goals, not necessarily the ones we have right now.

CHAIRMAN JACKSON: Do you intend to continue discussion about what form of safety goal?

MR. KRESS: Yes, we plan to continue that.

CHAIRMAN JACKSON: So how does the staff currently address plant specific backfits in its regulatory analysis if there isn't a current application of the safety goals on a plant specific basis?

MR. KRESS: Plant specific backfits, I am not sure of the answer to that frankly. I know they have to do a regulatory analysis when it is a generic backfit and that addresses plant types generally not getting very specific in terms of plant specific. Does anybody want to add to that?

MR. SEALE: That may be the engine that drives the upgrading of the existing IPEs by individual utilities.

CHAIRMAN JACKSON: I can't you too well.

MR. SEALE: I'm sorry. That may be the engine that drives the upgrading of the existing IPEs by individual utilities. If they want it, it is in their interest to do it.

CHAIRMAN JACKSON: Were you going to make a comment?

MR. CARROLL: Was your question how does this work in terms of the backfit rule on licensing issues on individual plants because it doesn't. The backfit rule doesn't apply to individual plants.

CHAIRMAN JACKSON: No, I understand that. I am talking about when there are plant specific changes.

MR. CARROLL: But for a grouping of plants.

CHAIRMAN JACKSON: Right.

MR. CARROLL: All right.

CHAIRMAN JACKSON: More along the line of what you were talking about, Dr. Seale. Please go ahead.

MR. APOSTOLAKIS: I think that is really part of the problem, that the goals are really sometimes used in a so-called generic sense which is ill-defined itself and sometimes really are used in plant specific applications. Until somebody says you are using a plant specific application and then say, "No, no, I don't want to do that."

I think part of the reason why we recommended that

the Commission restate the goals on a plant specific basis is to make it clear that we need a new statement that says this is the way these things ought to be used because right now it is not clear to people what exactly you mean by using them in a generic way for a population of plants. So there is some confusion, I think, out there.

CHAIRMAN JACKSON: Is there confusion in your mind?

MR. APOSTOLAKIS: Yes. When I heard the answer to the question, what does it mean to do this, I didn't like them and you can't blame the people because to use something in a generic sense is a little difficult. We don't know how to do that. What if three plants are way above the goal? What do you do? The average population is below but what do you do about these three?

MR. KRESS: Nothing because they meet the definition of adequate protection already. That is the basic answer but one might give it a little more regulatory attention to the one that is high on the list.

MR. LINDBLAD: I think there is a question whether regulatory policy requires all plants to be better than average.

COMMISSIONER ROGERS: Oh, I don't know. I think that sounds like a wonderful idea.

[Laughter.]

MR. LINDBLAD: Garrison Keillor has talked about Lake Woebegone in that regard.

CHAIRMAN JACKSON: Dr. Kress, any further comments that you or members of your Committee would like to make?

MR. KRESS: Does anyone wish to add anything?

[No response.]

CHAIRMAN JACKSON: If not, I just want to thank you for a very informative briefing and a very useful one. I have some follow-on comments if Commissioner Rogers has no questions.

COMMISSIONER ROGERS: Just a little bit further on this level-3 and level-2 and so on and so forth question. Would there be any value in having rather complete level-2 PRAs and then to couple those together with some kind of a generic level-3, in other words, a generic population or a location or a site that would then somewhat settle this question about whether on the average because the average then is a site, deals with a sort of generic site? Would there be any value to doing that to give us any confidence in the regulatory aspects of what has taken place?

MR. KRESS: I personally think not and the reason I think not is because in order to arrive at the characteristics of this generic thing, you have to do the plant specific, you have to work backwards from that to get to generic and as long as you are going to work backwards,

you might as well not bother with it.

If you had a generic description, it would stand and you could do some things to bound various sites in a generic type of characteristics and if that bounding result told you that on the average you were below the safety goals, you have learned something and it would be worthwhile. You could do that without working backwards and as is likely to turn out, this bounding analysis told you that you didn't meet the safety goals.

Then you haven't learned very much because you know it is a bounding analysis and you just don't know how much it bounds, what the margins are. I personally don't think that it would be worthwhile to do it.

MR. CARROLL: Isn't it fair to say that if you have a good level-2, doing the level-3 isn't that big a job?

MR. KRESS: That's true. I think that is probably a true statement.

MR. CARROLL: For a plant specific level-3.

MR. KRESS: Yes.

MR. CATTON: At one time and I am not sure who did it, I believe it was Sandia did a study and they put an average plant on a number of different sites and that led to all sorts of excitement. So it is not a good idea.

COMMISSIONER ROGERS: I think this question of restating the Commission's safety goal has to be given some,

for use on a plant specific basis, I think that requires a lot of thought on exactly how to do that.

MR. KRESS: Yes.

COMMISSIONER ROGERS: I think it has to be more detailed than the Commission just reverses its previous policy that said that safety goals are not to be used on a plant specific basis. I think it has to then include some very specific ways in which it would be acceptable to use it on a plant specific basis.

MR. CARROLL: Agreed.

COMMISSIONER ROGERS: Thank you.

CHAIRMAN JACKSON: Thank you again and thank you for your April 23 letter on the PRA related activities. It is of particular interest because it did contain a fair amount of detail and the substance to help focus on some critical questions in terms of what both we and the staff need to think about in this area.

So I just want to encourage you to continue to follow up on the items that we have discussed today and that you have indicated that you would follow up on including the reviews in the digital I&C area as well as addressing Commissioner Rogers' comment a moment ago.

Unless you have any further comments or questions, I again wish you well, Mr. Carroll, and thank you again. If there are no further comments, we are adjourned.

[Whereupon, at 11:10 a.m., the meeting was adjourned.]

CERTIFICATE

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

TITLE OF MEETING: MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) - PUBLIC MEETING

PLACE OF MEETING: Rockville, Maryland

DATE OF MEETING: Friday, May 24, 1996

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

Transcriber: Marilynn Estep

Reporter: Marilynn Estep

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

May 15, 1996

MEMORANDUM TO: John C. Hoyle
Secretary of the Commission

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

John T. Larkins

SUBJECT: ACRS MEETING WITH THE NRC COMMISSIONERS ON MAY 24, 1996—SCHEDULE/BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners between 9:30 and 11:00 a.m. on Friday, May 24, 1996, to discuss the following items. Background materials related to these items are attached:

A. Introduction (NRC Chairman) 9:30 - 9:35 A.M.

B. Items for Discussion:

1. Use of IPEs in the regulatory process, PRA framework document, pilot applications, and next steps to expand the use of PRA in the regulatory decision-making process (Apostolakis) (pp. 1-20) 9:35 - 9:55 A.M.

2. Fire protection issues, including fire PRA models and PRA-based scoping analysis of degraded fire barriers (Catton) (pp. 21-27) 9:55 - 10:10 A.M.

3. Proposed final revisions to 10 CFR Parts 50 and 100 (Lindblad) (pp. 28-31) 10:10 - 10:25 A.M.

4. Status of ACRS review of Regulatory Guidance documents related to digital instrumentation and control systems (Miller) (pp. 32-33) 10:25 - 10:35 A.M.

5. Status of ACRS review of standard plant designs: (pp. 34-38) 10:35 - 10:45 A.M.

   - ABWR and system 80+ design certification rules (Carroll)

   - AP600 design (Lindblad)

- Test and analysis programs associated with the AP600 and SBWR designs (Catton)

6. Conformance of operating plants with NRC safety goals (Kress) (pp. 39-40) 10:45 - 10:55 A.M.

C. Closing Remarks 10:55 - 11:00 A.M.

Attachment: As stated

cc: ACRS Members
ACRS Technical Staff

ITEM B.1:

USE OF INDIVIDUAL PLANT EXAMINATIONS (IPEs)
IN THE REGULATORY PROCESS, PRA FRAMEWORK
DOCUMENT, PILOT APPLICATIONS, AND NEXT STEPS
TO EXPAND THE USE OF PRA IN THE REGULATORY
DECISION-MAKING PROCESS

(DR. APOSTOLAKIS)

ITEM B.1: USE OF IPEs IN THE REGULATORY PROCESS, PRA FRAMEWORK DOCUMENT, PILOT APPLICATIONS, AND NEXT STEPS TO EXPAND THE USE OF PRA IN THE REGULATORY DECISION-MAKING PROCESS

- Use of IPEs in the Regulatory Process

In the December 27, 1995 Staff Requirements Memorandum (SRM), resulting from the meeting between the ACRS and the Commission on December 8, 1995, the Commission requested the ACRS views on,

> "the extent to which the current spectrum of IPEs can be used in the regulatory process."

In the June 16, 1995 SRM, resulting from the ACRS meeting with the Commission on June 8, 1995,

> "the Commission expressed concern about the lack of consistency among current IPEs and suggested that modifications to the IPE PRAs may be necessary if they are to be used for other regulatory applications. In addition, the Commission noted that more meaningful plant-to-plant or scenario-to-scenario comparisons based on risk could be achieved if PRAs were done on a more standardized, replicable basis."

The ACRS Subcommittee on Individual Plant Examinations met with the NRC staff and its contractors on January 26, 1996 to discuss the IPE review process and related issues. During its February and March 1996 meetings, the ACRS discussed the use of IPEs in the regulatory process and other related matters with representatives of the NRC staff and provided a report to the Commission dated March 8, 1996. In that report the Committee made several points, including the following:

- The quality and consistency of the IPEs vary and the impact of assumptions and analytical models is difficult to assess. On a case-by-case basis, however, additional and extended use of these IPEs is possible. As specific regulatory issues arise, the PRA Standard Review Plan now being developed by the staff can serve as a template for judging the quality and acceptability of the individual plant PRA for the proposed application.

- To achieve consistency, some degree of standardization of PRA models and method will be required. IPEs should be reviewed to identify acceptable and unacceptable assumptions and/or models. Codification of assumptions and models ought not inhibit the continued development of PRA methods. These activities would be a significant first step toward addressing the Commission's statement in the June 16, 1995 SRM,

"that more meaningful plant-to-plant or scenario-to-scenario comparisons based on risk could be achieved if PRAs were done on a more standardized, replicable basis."

- The NRC could make additional use of the present IPEs (except those that the staff has found to use unacceptable methods or models) for a limited number of applications (e.g., regulatory analyses and prioritization of generic issues).

The Committee stated that the IPE program has met successfully the objectives of Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." This program has developed a risk awareness, both in the utilities and the NRC, that will contribute significantly efforts to establish a risk-informed and performance-oriented regulatory system. The plant-specific IPEs are an extremely valuable asset that should not be permitted to languish unimproved and unused.

The Executive Director for Operations responded to the ACRS comments and recommendations on the above subject in a letter dated April 10, 1996.

The ACRS also discussed the IPE Insights report with the staff during its meeting on May 23, 1996. The ACRS Subcommittee on PRA plans to discuss this report in detail as well as the NRC research program on the use of PRA during a meeting on June 11-12, 1996.

- <u>PRA Framework Document, Pilot Applications, and Next Steps to Expand the Use of PRA in the Regulatory Decision-Making Process</u>

In the December 27, 1995 SRM, the Commission requested ACRS views on,

"the PRA framework document, its relationship to pilot applications (SECY-95-280), and the next steps in the process to expand the use of PRA in the regulatory decision-making process."

The ACRS Subcommittee on Probabilistic Risk Assessment met with the NRC staff and its contractors, representatives of the Nuclear Energy Institute (NEI), and two invited experts on February 27-28, 1996, to discuss this matter. The ACRS also continued its deliberations on risk-informed and performance-oriented regulation (RIPOR) during its March 7-9, and April 11-13, 1996 meetings. The Committee provided a report to the Commission dated April 23, 1996. In that report the Committee made several points, including the following:

- The PRA framework document provides a good starting point in the development of RIPOR. The process described in the document is a reasonable way to proceed. The ACRS agrees with the staff that the focus should be on the integration of probabilistic and deterministic approaches to regulation. The document, however, does not articulate an overall philosophy for RIPOR. The Committee believes that such a

philosophy should be developed. Some high-level principles that should be included are given below.

- RIPOR should consider risk from all modes of nuclear plant operations, including full power, shutdown, and transition.

- The Commission's safety goals should serve as the top-level acceptance criteria.

- Subsidiary performance-based acceptance criteria should be determined in a consistent way and must be measurable or calculable. The licensee should be granted flexibility in choosing the means to meet the criteria.

- The relationship between RIPOR and defense-in-depth should be explained. The role of defense-in-depth in the determination of performance criteria to accommodate uncertainty and incompleteness in risk assessments should be established.

- Criteria for the adoption of prescriptive regulations should be clearly delineated.

- The acceptance criteria should be set at the highest level of plant system hierarchy that is consistent with the other principles noted above.

The ACRS also recommended that for each pilot project, attempt be made to establish performance-based decision criteria along with the methods that would be used for demonstrating compliance. The ACRS believes that the NRC needs to take a number of important additional steps, noted below, before a RIPOR environment can be achieved.

- A restatement of the Commission's safety goal policy is needed that will allow the use of safety goals on a plant-specific basis.

- A methodology is needed to determine performance-based criteria for regulatory action that are consistent with the top-level safety goals. An important element should be the preservation of the concept of defense-in-depth. The development of this methodology will also provide the opportunity to reexamine the validity of Level 2 subsidiary goals, which appear to be controversial at this time.

- Developing a RIPOR system should be a participative effort between the staff and the industry. Also, the staff should work with foreign researchers and regulatory agencies.

The ACRS Subcommittee on PRA plans to continue its discussion of this issue and the steps required to expand the use of PRA in the regulatory decision-making process during a meeting on July 17-19, 1996. The Subcommittee will also review the programs for risk-based analysis of operating events and will begin review of the proposed Standard Review Plan (SRP) and regulatory guides.

Attachments:

- Report dated March 8, 1996, from T. S. Kress, ACRS Chairman, to NRC Chairman Shirley Ann Jackson, Subject: Use of Individual Plant Examinations in the Regulatory Process (pp. 5-8)

- Letter dated April 10, 1996, from J. M. Taylor, Executive Director for Operations, to T. S. Kress, ACRS Chairman, Subject: Use of Individual Plant Examinations in the Regulatory Process (pp. 9-10)

- Report dated April 23, 1996, from T. S. Kress, ACRS Chairman, to NRC Chairman Shirley Ann Jackson, Subject: Probabilistic Risk Assessment Framework, Pilot Applications, and Next Steps to Expand the Use of PRA in the Regulatory Decision-Making Process (pp. 11-15)

- Staff Requirements Memorandum, dated December 27, 1995, from J. Hoyle, Office of SECY, to J. Larkins, ACRS (pp. 16-17)

- Staff Requirements Memorandum, dated June 16, 1995, from A. Bates, Office of SECY (p. 18)

- Staff Requirements Memorandum regarding Briefing on PRA Implementation Plan, dated May 15, 1996, from J. Hoyle, Office of SECY, to James M. Taylor, EDO (pp. 19-20)

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

March 8, 1996

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: USE OF INDIVIDUAL PLANT EXAMINATIONS IN THE REGULATORY PROCESS

During the 428th and 429th meetings of the Advisory Committee on Reactor Safeguards, February 8-10 and March 7-9, 1996, respectively, we discussed the Individual Plant Examination (IPE) review process and findings with the NRC staff. Our Subcommittee on IPEs also met with the staff and its contractors on January 26, 1996, to review this matter. We also had the benefit of the documents referenced. This report is in response to the December 27, 1995 Staff Requirements Memorandum (SRM).

In the SRM, the Commission requested "the ACRS views on the extent to which the current spectrum of IPEs can be used in the regulatory process." We interpret this request as referring to potential regulatory uses of the IPEs that were not delineated in Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." This report includes comments on both the Generic Letter goals and the Commission request.

<u>Goals of Generic Letter 88-20</u>

The purpose of the IPE program, as stated in Generic Letter 88-20, was for each licensee:

(1) to develop an appreciation of severe accident behavior

(2) to understand the most likely severe accident sequences that could occur at its plant

(3) to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases

(4) to reduce, if necessary, the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

We note that the IPEs were to be limited to the examination of internal initiating events and internal floods with the reactor at power and that individual and societal risks were not to be estimated. Other programs deal with external events and shutdown risk.

The IPE program has been successful at most utilities in meeting goal (1) and, to a lesser extent, goals (2) and (3) of the Generic Letter. Goal (4) of the Generic Letter also appears to have been achieved. We were told that most licensees discovered weaknesses and took corrective actions. In addition, this program has been beneficial in educating a broader segment of the NRC staff about the issues related to these goals.

We were told by the staff that all licensees submitted a Level-1 probabilistic risk assessment (PRA). Most licensees also submitted a Level-2 PRA, although some addressed Level-2 phenomena in a rudimentary manner. The methods and data sources used by different licensees varied widely. In some cases, the choices appeared to be arbitrary. Some licensees chose to include common-cause failures only for major components, while others chose to ignore them completely.

It is difficult to determine the extent to which the variability in IPE results for similar classes of plants is due to actual plant differences or to modeling assumptions. Although some of the causes for this variability may be immediately apparent, others are not. The latter include assumptions made about success criteria, the assumed dependencies between operator actions, and the level of decomposition in fault-tree analyses. (We note that the fault trees were not requested as part of the IPE submittals.)

An example of a potentially significant impact of modeling differences is the range of core-damage frequencies (CDFs) for BWR 3/4s that the staff has compiled. This range is from about 10^{-7} to about 10^{-4} per reactor-year. Although the staff has stated that such differences are primarily due to plant differences, this range of results seems unrealistic given the similarity among BWR 3/4s.

Use of IPEs in the Regulatory Process

As discussed above, the quality and consistency of the IPEs vary and the impact of assumptions and analytical models is difficult to

assess. On a case-by-case basis, however, additional and extended use of these IPEs is possible. As specific regulatory issues arise, the PRA Standard Review Plan now being developed by the staff can serve as a template for judging the quality and acceptability of the individual plant PRA for the proposed application.

As the agency moves toward risk-informed regulation, there will be an increasing need for full-scope PRAs that incorporate fire risk, external events, other modes of operation, and site-specific consequences. When requests for risk-informed regulatory action arise, the NRC staff should make it clear that a relevant PRA should be used.

To achieve these goals, especially consistency, some degree of standardization will be required. Standardizing PRA models and methods has been a controversial subject. Proponents argue that it would create a basis for comparison of PRA results, while opponents fear that it would inhibit methodological developments. We recommend that IPEs be reviewed to identify acceptable and unacceptable assumptions and/or models. Codification of assumptions and models ought not inhibit the continued development of PRA methods. These activities would be a significant first step toward addressing the Commission's statement in the SRM dated June 16, 1995, "that more meaningful plant-to-plant or scenario-to-scenario comparisons based on risk could be achieved if PRAs were done on a more standardized, replicable basis."

We believe that the NRC could make additional use of the present IPEs (except those that the staff has found to use unacceptable methods or models) for a limited number of applications (e.g., regulatory analyses and prioritization of generic issues).

The staff stated that the CDFs for several PWRs are greater than 10^{-4} per reactor-year. Several BWRs have CDFs that are very close to 10^{-4} per reactor-year and the conditional containment failure probabilities for BWR Mark I containments range from about 0.02 to about 0.6. Although the PRAs have limitations as discussed above, these numbers suggest that an investigation would be warranted to reassess their validity and to verify that the very low numbers reported by some other plants reflect actual plant differences.

Our conclusion is that the IPE program has met successfully the objectives of Generic Letter 88-20. This program has developed a risk awareness, both in the utilities and the NRC, that will contribute significantly to efforts to establish a risk-informed and performance-oriented regulatory system. The plant-specific

IPEs are an extremely valuable asset that should not be permitted to languish unimproved and unused.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Staff Requirements Memorandum dated June 16, 1995, from Andrew L. Bates, Acting Secretary, NRC, to the File regarding Meeting with ACRS on June 8, 1995
2. Staff Requirements Memorandum dated December 27, 1995, from John C. Hoyle, Secretary, NRC, to John T. Larkins, ACRS regarding Meeting with ACRS on December 8, 1995
3. Generic Letter 88-20, dated November 23, 1988, to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, Subject: Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)

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

April 10, 1996

Dr. T. S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: USE OF INDIVIDUAL PLANT EXAMINATIONS IN THE REGULATORY PROCESS

Dear Dr. Kress:

This is in response to the ACRS letter to Chairman Jackson dated March 8, 1996, regarding use of Individual Plant Examinations (IPEs) in the regulatory process. The staff agrees with the ACRS conclusion that "the IPE program has met successfully the objectives of Generic Letter 88-20" and "developed a risk awareness, both in the utilities and the NRC." As the ACRS is aware, the staff is currently developing an IPE insights report on reactor safety and plant performance. Based on the staff's work in this area, there are a few points in the letter that the staff would like to address at this time. These comments relate to clarifying the staff's interpretation of variability of the IPE results, consistency and possible uses of the PRAs, and to the core damage frequency and conditional containment failure probabilities for certain plants.

First, your letter states that "it is difficult to determine the extent to which the variability in the IPE results for similar classes of plants is due to actual plant differences or modeling assumptions.... Although the staff has stated that such differences are primarily due to plant differences, this range of results seems unrealistic." As an example of potentially significant modeling differences your letter cites the range in core damage frequencies (CDF) for BWR 3/4 plants.

As discussed during the ACRS meetings and in your letter, the CDFs for this group ranged from 10^{-4} to 10^{-7} per reactor year. However, when excluding the plants at the lower end of the range, the reported CDFs for the BWR 3/4 plant group only varies by about a factor of 10. This variability is expected and is caused primarily because of the plants' actual design differences. For the other three plants in the BWR 3/4 group, their much lower CDFs are primarily due to modeling differences. These modeling differences are discussed in the IPE insights report.

Next, the staff agrees that consistency in the PRA models and methods is necessary to fully realize the benefits of risk-informed regulation. The staff's review of licensee IPE/PRA submittals has highlighted the need for consistent application of the PRA process. This need is currently being considered by the staff as part of the PRA regulatory guide development. The staff also recognizes that unique insights have been gained from the IPE/PRA reviews. In the cases where the IPE/PRA models and methods are well understood and the results appear reasonable, the staff expects that

some of the IPEs/PRAs can be used for future regulatory applications, such as issue prioritization and regulatory analysis.

Lastly, the staff acknowledges that additional review is warranted for several of the CDFs and conditional containment failure probabilities reported for BWRs and PWRs. The staff's review in this area is being documented in the insights report.

The ACRS will be provided a copy of the IPE insights report within the next few months. Presentations on the report are now scheduled for May 23 and June 11, 1996. Each of the points noted above can be discussed in greater detail during these presentations.

Sincerely,

James L. Milhoan for
James M. Taylor
Executive Director
for Operations

References:
1. Letter dated March 8, 1996, from T. S. Kress, Chairman, ACRS to Chairman Jackson regarding Use of Individual Plant Examinations in the Regulatory Process.
2. SECY-96-051 dated March 8, 1996, from James M. Taylor, Executive Director for Operations to the Commissioners regarding Status of the IPE and IPEEE Programs.

cc: Chairman Jackson
Commissioner Rogers
Commissioner Dicus
SECY
OGC
OCA
OPA
OIP
OCAA
ASLBP
OIG/ACNW

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

April 23, 1996

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: PROBABILISTIC RISK ASSESSMENT FRAMEWORK, PILOT APPLICATIONS, AND NEXT STEPS TO EXPAND THE USE OF PRA IN THE REGULATORY DECISION-MAKING PROCESS

During the 430th meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1996, we continued our deliberations on risk-informed and performance-oriented regulation (RIPOR). We met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) during our 429th meeting on March 7-9, 1996. Our Subcommittee on Probabilistic Risk Assessment (PRA) also met on October 26-27, 1995, with representatives of the NRC staff and of the nuclear industry, and on February 27-28, 1996, with the NRC staff and two invited experts, Dr. D. M. Karydas (performance-based standards for fire protection) and Professor T. G. Theofanous (on the proper formulation of safety goals and assessment of safety margins for rare and high-consequence hazards). We also had the benefit of the documents referenced.

This report is in response to the Staff Requirements Memorandum dated December 27, 1995, in which the Commission requested "ACRS views on the PRA framework document, its relationship to the pilot applications (SECY-95-280), and the next steps in the process to expand the use of PRA in the regulatory decision-making process."

<u>PRA Framework Document</u>

The PRA framework document provides a good starting point in the development of RIPOR. The six-step process described in the document is a reasonable way to proceed. We agree with the staff that the focus should be on the integration of probabilistic and deterministic approaches to regulation.

The PRA framework document, however, does not articulate an overall philosophy for RIPOR. We believe that such a philosophy should be developed. Some important high-level principles that should be included are:

1. RIPOR should consider risk from all modes of nuclear plant operations, including full power, shutdown, and transition.

2. The Commission's safety goals should serve as the top-level acceptance criteria.

3. Subsidiary performance-based acceptance criteria should be determined in a consistent way and must be measurable or calculable. The licensee should be granted flexibility in choosing the means to meet the criteria.

4. The relationship between RIPOR and defense-in-depth should be explained. The role of defense-in-depth in the determination of performance criteria to accommodate uncertainty and incompleteness in risk assessments should be established.

5. Criteria for the adoption of prescriptive regulations should be clearly delineated.

6. The acceptance criteria should be set at the highest level of plant system hierarchy that is consistent with the other principles noted above.

Discussion

It is indicative of the novelty of these concepts that we have spent a considerable amount of time discussing the meaning of "performance" among ourselves and with the staff and NEI. Some interpret performance in a limited way; i.e., its measures are simply the reliability and availability (or related quantities) of plant systems and components. Others take a broader view and interpret it as the overall performance of the licensee, including operations, maintenance, training, and the prevailing safety culture at the plant.

Similarly, the definition of performance criteria varies widely. At one extreme, we have simple measures that are either directly measurable or that involve calculations (e.g., the reliabilities and unavailabilities mentioned above). At the other extreme, performance criteria can be probabilistic or nonprobabilistic and can be set at any level. Observations and statistical or experimental evidence from the plant or other sources in conjunction with models can be used to demonstrate that the criteria have been met. As part of an overall philosophy, the staff needs to resolve the ambiguity in the definition of performance criteria.

Pilot Applications

While we support the staff's use of pilot applications, we are concerned that there seems to be no integrated justification for their selection. We would like to see the development of a list of important issues that are expected to arise on the road to RIPOR,

along with a discussion of how the selected pilot projects will help. The staff has agreed to look into these issues.

We also recommend that, for each pilot project, attempts be made to establish performance-based decision criteria along with the methods that would be used for demonstrating compliance. Such an exercise should provide useful insights regarding the overall feasibility of a performance-oriented approach to regulation.

Next Steps to Expand the Use of PRA in the Regulatory Decision-making Process

We believe that the NRC needs to take a number of important additional steps before a RIPOR environment can be achieved. These are discussed below.

Safety Goals

A restatement of the Commission's safety goal policy is needed that will allow the use of safety goals on a plant-specific basis.

Performance-Based Regulatory Criteria

A methodology is needed to determine performance-based criteria for regulatory action that are consistent with the top-level safety goals, as stated in the high-level principles. A "top-down" approach will ensure that this happens. An important element should be the preservation of the concept of defense-in-depth. The development of this methodology will also provide the opportunity to reexamine the validity of Level 2 subsidiary goals, which appear to be controversial at this time.

Programmatic Issues

Developing a RIPOR system should be a participative effort between the staff and the industry. We believe that the magnitude and significance of the task that the staff has undertaken requires a cooperative effort. Also, we recommend that the staff work with foreign researchers and regulatory agencies.

Conclusion

The intellectual and practical issues that the staff must confront in developing a RIPOR structure are significant. The staff has made a good start, but much remains to be done. We are pleased that the staff has agreed to meet with us periodically. Recent meetings have demonstrated that the staff is receptive to suggestions on how to deal with these complex issues. We applaud this attitude. We will keep you informed as these efforts progress.

Additional comments by ACRS Members Thomas S. Kress and Don W. Miller are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Additional Comments by ACRS Members Thomas S. Kress and Don W. Miller

While we agree with most of the Committee's report on this subject, we find it to lack coherence. The major problem we have with the Committee report is its treatment of the concept of "performance-based" regulation. We conceive of basically two meanings to the word "performance" in this context: (1) the performance of equipment (systems and components) in carrying out the intended function, or (2) the performance of the licensee in performing its function (operation, maintenance, inspection, training, etc.). The first of these could further relate to either the operability of the specific equipment (e.g., does it turn on or off, and, in the case of a pump, for example, does it provide the required flow) or to the reliability/availability of the equipment. In our view, the former does not provide any basis on which to develop a regulatory structure (there are no meaningful acceptance criteria that relate to risk). On the other hand, the latter can clearly be anchored in risk. This, however, would be purely risk-based regulation. The word "performance" in this context becomes synonymous with "risk" and such a regulatory concept should be designated as risk-based and should not be called performance-based.

The second possible meaning of performance, the performance of the licensee, obviously has a nexus to risk. This connotation of performance, however, is what we have been calling organizational factors. To date, a methodology has not been developed by which objective performance measures can be identified and be factored directly into PRA to quantify risk implications. Therefore, at this time, we do not have the capability to develop such performance-based regulations in any coherent manner. This would, however, be an area worth pursuing in the future with additional research.

This leads us to our main point. At this time, we should be striving for risk-based or risk-informed regulations and should relegate the concept of "performance" regulation to being a remote possibility that needs substantial research to determine feasibility.

References:

1. Memorandum dated December 27, 1995, from J. Hoyle, Secretary of NRC, to J. Larkins, ACRS, Subject: Staff Requirements Memorandum dated December 27, 1995
2. Memorandum dated June 16, 1995, from A. Bates, Office of the Secretary, NRC, to File, Subject: Staff Requirements Memorandum dated June 16, 1995
3. Letter dated February 6, 1996, from J. Milhoan, Office of the Executive Director for Operations, NRC, to W. Rasin, Nuclear Energy Institute, Subject: Improving the Regulatory Process through Risk-Based and Performance-Based Regulation
4. Letter dated January 3, 1996, from J. Taylor, Executive Director for Operations, NRC, to Chairman Jackson, NRC, Subject: Improvements Associated With Managing the Utilization of Probabilistic Risk Assessment (PRA) and Digital Instrumentation and Control Technology
5. Letter dated November 30, 1995, from Chairman Jackson, NRC, to J. Taylor, Executive Director for Operations, NRC, Subject: Follow-up Requests in Probabilistic Risk Assessment and Digital Instrumentation and Control
6. SECY-95-280, "Framework for Applying Probabilistic Risk Analysis in Reactor Regulation," dated November 27, 1995
7. Letter dated November 14, 1995, from W. Rasin, Nuclear Energy Institute, to J. Milhoan, Office of Executive Director for Operations, NRC, Subject: Draft report, "Improving the Regulatory Process Through Risk-Based and Performance-Based Regulation"

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

OFFICE OF THE SECRETARY

December 27, 1995

IN RESPONSE, PLEASE REFER TO: M951208

REVISED

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

FROM: John C. Hoyle, Secretary

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), 1:30 P.M., FRIDAY, DECEMBER 8, 1995, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the ACRS on the following topics:

1. Proposed resolution of Generic Issue 78, "Monitoring of Fatigue Transient Limits for the Reactor Coolant System"

2. The NEI petition for rulemaking to amend 10 CFR 50.48, "Fire Protection"

3. Development of improved nondestructive examination (NDE) techniques

4. National Academy of Sciences/National Research Council study on "Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues"

5. Proposed final revision 1 to Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants"

6. Status report of IPEs

The Commission requested feedback from the ACRS on use of RuleNet during the next meeting with the Commission.
(ACRS~~EDO~~) (SECY Suspense: 5/24/96)

The Commission also requested the ACRS views on the extent to which the current spectrum of IPEs can be used in the regulatory process.
(ACRS) (SECY Suspense: 2/23/96)

In addition, the Commission requests ACRS views on the PRA framework document, its relationship to the pilot applications (SECY-95-280), and the next steps in the process to expand the use of PRA in the regulatory decision-making process.
(ACRS~~EDO~~) (SECY Suspense: 4/30/96)

cc: Chairman Jackson
Commissioner Rogers
EDO
OGC
OCA
OIG
Office Directors, Regions, ACNW, ASLBP (via E-Mail)
PDR - Advance
DCS - P1-24

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

OFFICE OF THE SECRETARY

IN RESPONSE, PLEASE
REFER TO: M950608A

June 16, 1995

MEMORANDUM TO THE FILE

FROM: Andrew L. Bates, Acting Secretary /s/

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), 10:00 A.M., THURSDAY, JUNE 8, 1995, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Reactor Safeguards for discussion of the following topics:

1. Thermal hydraulic issues
2. Status of Westinghouse AP600 design review
3. Regulatory analysis guidelines
4. Application of risk analysis in rulemaking
5. Proposed final rule on technical specifications
6. Cracking and fatigue in nuclear components
7. Digital instrumentation and control
8. Operating reactors conformance to the safety goals - status report.

During the discussion on cracking and fatigue in nuclear components, the Commission requested the ACRS to lend assistance to the staff in encouraging industry to undertake research to improve NDE techniques with the aim of more accurately detecting and assessing steam generator tube defects.

During discussion of the last topic, the Commission expressed concern about the lack of consistency among current IPEs and suggested that, modifications to the IPE PRAs may be necessary if they are to be used for other regulatory applications. In addition, the Commission noted that more meaningful plant-to-plant or scenario-to-scenario comparisons based on risk could be achieved if PRAs were done on a more standardized, replicable basis.

There were no requirements identified for staff action.

IN RESPONSE, PLEASE
REFER TO: M960404A

May 15, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John C. Hoyle, Secretary /s/

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON PRA IMPLEMENTATION PLAN, 10:00 A.M., THURSDAY, APRIL 4, 1996, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on the PRA implementation plan.

The Commission requested that the staff provide the following supplemental information.

1. The staff should prepare a policy paper, with recommendations, addressing the resolution of the four emerging policy issues identified in its March 26, 1996 memorandum regarding: 1) the role of performance-based regulation in the PRA Implementation Plan; 2) plant-specific application of safety goals; 3) risk neutral vs increases in risk; and 4) changes to risk-informed inservice testing (IST) and inservice inspection (ISI) requirements, for Commission decision prior to the staff's issuance of any final safety evaluation, position, or guidance. However, the Commission does not wish to inhibit any dialog with the public or industry on the resolution of these issues. The bases as well as pros and cons of alternatives should be included.

2. The staff should provide an update on the implementation and use of subsidiary safety goal objectives including any plans for their use in Standard Review Plans (SRPs) and Regulatory Guides (RGs).

3. The staff should clarify how it intends to address uncertainty in the implementation of risk-informed and performance-based regulation.

(EDO) (SECY Suspense: 9/20/96)

The IPE reviews and pilot applications should be reviewed for any lessons that may be applicable to the development and use of SRPs and RGs.

The use of expert judgment, such as is being applied in the high-level waste area, should be considered as a source of useful guidance to the expert panels in the maintenance rule.

The staff is requested to keep the Commission informed of significant policy issues relevant to risk-informed initiatives.

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

ITEM B.2:

FIRE PROTECTION ISSUES, INCLUDING FIRE PRA MODELS AND PRA-BASED SCOPING ANALYSIS OF DEGRADED FIRE BARRIERS

(DR. CATTON)

ITEM B.2: FIRE PROTECTION ISSUES, INCLUDING FIRE PRA MODELS AND PRA-BASED SCOPING ANALYSIS OF DEGRADED FIRE BARRIERS

In November 1995, the NRC Chairman requested that the ACRS review:

- The draft PRA fire model developed by the Brookhaven National Laboratory (BNL) for evaluating fire risk during a self-induced station blackout (SBO) in the event of an uncontrolled fire either in the main control room (MCR) or cable spreading room (CSR)

- Scoping Analysis developed by BNL to evaluate the effectiveness of degraded fire barriers

The ACRS Subcommittee on Fire Protection held a meeting on February 29, 1996, to discuss these issues and other fire protection related matters. During this meeting, representatives of the NRC staff and BNL briefed the Subcommittee regarding the preliminary results of BNL draft report on the self-induced SBO, the scoping analysis of degraded fire barriers, and the proposed feasibility study by National Institute of Standards and Technology (NIST) to develop alternate time-temperature curves for rating fire barriers. Also, the staff provided status reports on the Fire Protection Task Action Plan and Thermo-Lag Action Plan. The full Committee discussed these issues with the staff, BNL, and NIST during its March 1996 meeting and provided a report to the Commission dated March 15, 1996. In that report the Committee provided several comments, including the following:

- The BNL study focused on the effectiveness of the procedures used to mitigate the fire and did not address the probabilistic treatment of fires. The scope of the study did not include a number of issues that could affect the conclusions. For example, the study addressed neither the effects of fire and smoke on human actions nor the possible damage to sensitive electronic control and safety instrumentation. The study is weak in the areas of modeling human actions for the manual shutdown and restart of electrical equipment after an SBO condition. Because of the limitations of the analysis and the failure to quantify uncertainties, no substantive conclusions can be drawn from this study. The limitations of the analysis should be addressed in Phase 2 of this study. A meaningful uncertainty analysis should also be performed.

- The scoping analysis of degraded fire barriers performed by BNL was based on event tree/fault tree models. Although this is a step in the right direction, the analysis does not use the best available methods for modeling fire propagation, detection, and suppression. It does not model the fundamental competition between the time to damage and time to detection/suppression.

- The need for the proposed feasibility study by NIST to develop alternate time-temperature curves for nuclear power plant fire barrier qualification, which includes development of

models and test methods to simulate barrier response, is questionable. Alternate time-temperature curves have been developed by the insurance industry. Furthermore, a large number of fire models exist, some of which are being evaluated by the Department of Energy.

The EDO responded to the ACRS comments in a letter dated April 24, 1996, stating that the staff anticipates that additional research may be necessary to support certain options. The staff will also request that the Probabilistic Risk Assessment Coordinating Committee review the BNL projects and evaluate the benefits of revising or improving certain aspects of the models developed by BNL and of performing an uncertainty analysis. The NRC staff plans to discuss these issues with ACRS Fire Protection Subcommittee in the near future.

The staff is scheduled to submit a Commission paper on the development of a performance-based rulemaking options to ACRS in June 1996. The Subcommittee on Fire Protection plans to discuss this rulemaking with the staff in July 1996. Subsequent to the Subcommittee review, the ACRS full Committee will consider this matter.

Attachments:

- Report dated March 15, 1996, from T. S. Kress, ACRS Chairman, to NRC Chairman Shirley Ann Jackson, Subject: Review of recent fire probabilistic risk assessment reports by Brookhaven National Laboratory and certain fire barrier issues (pp. 23-25)

- Letter dated April 24, 1996, from J. M. Taylor, Executive Director for Operations, to T.S. Kress, ACRS Chairman, Subject: Review of recent fire probabilistic risk assessment reports by Brookhaven National Laboratory and certain fire barrier issues (pp. 26-27)

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

March 15, 1996

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: REVIEW OF RECENT FIRE PROBABILISTIC RISK ASSESSMENT REPORTS BY BROOKHAVEN NATIONAL LABORATORY AND CERTAIN FIRE BARRIER ISSUES

During the 429th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1996, we reviewed scoping fire probabilistic risk assessments (PRAs) performed by Brookhaven National Laboratory (BNL). We had the benefit of discussions with representatives of the staff, BNL, and the National Institute of Standards and Technology (NIST). Our Subcommittee on Fire Protection discussed this matter during a meeting on February 29, 1996. We also had the benefit of the documents referenced.

At your request, we reviewed both the PRA model that evaluated the strategy of using self-induced station blackout (SISBO) to mitigate the consequences of a fire in the control room or cable spreading room and the PRA-based scoping analysis of degraded fire barriers. We also discussed the development of alternate time-temperature curves for qualification of fire barriers and the status of other fire protection issues.

To comply with Appendix R requirements, eight units have procedures that require initiating a station blackout (SBO) condition. An additional fifteen units have procedures for dealing with fires in critical areas that could result in an SBO. The PRA by BNL evaluated the effects of different schemes for managing the electrical systems in the plant when a fire in the control room has required use of the alternate shutdown panel.

The study focused on the effectiveness of the procedures used to mitigate the fire and did not address the probabilistic treatment of fires. The scope of the study did not include a number of issues that could affect the conclusions. For example, the BNL study addressed neither the effects of fire and smoke on human actions nor the possible damage to sensitive electronic control and safety instrumentation. The study is weak in the areas of modeling human actions for the manual shutdown and restart of electrical equipment after an SBO condition. Because of the limitations of

the analysis and the failure to quantify uncertainties, no substantive conclusions can be drawn from this scoping study. The limitations of the analysis should be addressed in Phase 2 of this study. A meaningful uncertainty analysis should also be performed.

In the analysis of degraded fire barriers, BNL developed core-damage frequencies for fire scenarios involving failures of fire protection features such as cable tray fire barriers, automatic detection and suppression systems, and fire barrier penetrations. The PRA model did not examine degrees of fire barrier degradation.

The analysis was based on event tree/fault tree models. Although this is a step in the right direction, the analysis does not use the best available methods for modeling fire propagation, detection, and suppression. It does not model the fundamental competition between the time to damage and the time to detection/suppression. Most current fire PRAs have adopted the competing processes model.

We also discussed the program proposed to the staff by NIST to develop alternate time-temperature curves for nuclear power plant fire barrier qualification. The program includes development of models, ASTM E119-type full-scale furnace tests, and test methods to simulate barrier response. We question the need for this program. We have been told that alternate time-temperature curves have been produced by the insurance industry. Furthermore, a large number of fire models exist, some of which are being evaluated by the Department of Energy. Although the need for new models is not clear, more validation of these models with experimental data is needed. Some data exist (NUREG/CR-6017). Comparisons with fire model simulations show that the results are very sensitive to input parameters that are not always well known.

The staff summarized the progress of licensee actions to correct deficiencies associated with Thermo-Lag fire barriers. The program appears to be meeting its objectives.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>References</u>:

1. Brookhaven National Laboratory, Draft Technical Letter Report, FIN L-2629, "Risk Evaluation of the Response of PWRs to Severe Fires in Critical Locations," May 30, 1995 (Draft Predecisional)

2. Brookhaven National Laboratory, Technical Evaluation Report, FIN L-1311, "A Risk-Based Approach for Evaluation of Fire Mitigation Features in Nuclear Power Plants," November 21, 1995 (Draft Predecisional)
3. U. S. Nuclear Regulatory Commission, NUREG/CR-6017 and SAND93-0528, "Fire Modeling of the Heiss Dampf Reaktor Containment," September 1995

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

April 24, 1996

Dr. T. S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REVIEW OF RECENT FIRE PROBABILISTIC RISK ASSESSMENT REPORTS BY BROOKHAVEN NATIONAL LABORATORY AND CERTAIN FIRE BARRIER ISSUES

Dear Dr. Kress:

I am responding to the letter you sent the Chairman on March 15, 1996, regarding the 429th meeting of the Advisory Committee on Reactor Safeguards (ACRS) and your review of our ongoing projects related to the development and use of risk assessment methods to address specific fire protection-related problems. We appreciate the insights you provided regarding (1) the Brookhaven National Laboratory (BNL) limited fire-risk scoping analysis for post-fire alternative safe shutdown methodologies (used for a fire in either the main control room or the cable spreading room), which by procedure, place the plant in a station blackout condition; (2) the BNL fire-risk assessment of degraded fire barriers; and (3) the National Institute of Standards and Technology methodology for developing and implementing alternative time-temperature curves for testing nuclear power plant fire-resistive barriers. We will evaluate and consider the comments as we develop these programs.

The staff will request that the Probabilistic Risk Assessment Coordinating Committee review the BNL projects and evaluate the benefits of revising or improving certain aspects of these models and performing an uncertainty analysis.

In a staff requirements memorandum of July 27, 1994, the Commission requested that the staff evaluate the feasibility of developing alternative time-temperature curves for rating nuclear power plant fire barriers. The study is focused on developing a performance-based approach that considers a range of realistic fire problems specific to nuclear power plants. In your letter, you stated that the insurance industry has produced such curves and that there are a number of existing fire models, some of which are being evaluated by the Department of Energy (DOE). We will work with the ACRS staff to obtain additional information from the ACRS members who are familiar with the work completed by the insurance industry and the work underway at DOE.

If you have any questions concerning this letter, please contact K. Steven West, Chief, Fire Protection Engineering Section, in the Office of Nuclear Reactor Regulation, at 415-1220.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: The Chairman
Commissioner Rogers
Commissioner Dicus
SECY

ITEM B.3:

PROPOSED FINAL REVISIONS TO 10 CFR PARTS 50 AND 100

(MR. LINDBLAD)

ITEM B.3: PROPOSED FINAL REVISIONS TO 10 CFR PARTS 50 AND 100

The ACRS reviewed the proposed revisions to 10 CFR Parts 50 and 100 in February 1992, and provided a report to the Commission dated February 14, 1992. In that report, the Committee did not express any concerns or reservations for issuing these revisions for public comment. These revisions were issued for public comment in October 1992. Due to the substantive nature of comments received, they were reissued for public comment on October 17, 1994.

The proposed final revisions to 10 CFR Parts 50 and 100, which reflect consideration of public comments, consist of two parts. First is the final rule revising 10 CFR Part 100, "Reactor Site Criteria," for future plants. The second part is the final rule that would provide an alternative to 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," as the licensing basis for new plants. Both cases address the relocation of plant design criteria from Part 100 to Part 50. Revisions to 10 CFR Part 50 would contain source term and dose criteria (Section 50.34) and earthquake engineering criteria (new Appendix S). In the non-seismic area, there is currently a differing view between the Office of Nuclear Regulatory Research (RES) and the Office of Nuclear Reactor Regulation (NRR) regarding the proposed dose evaluation method in 10 CFR 50.34.

The ACRS Subcommittee on Extreme External Phenomena discussed these revisions and associated Regulatory Guides and Standard Review Plan Sections during a meeting on April 3, 1996. The full Committee discussed this matter during its April 1996 meeting. The RES and NRR staff also briefed the Committee regarding their different views in the proposed dose calculation method in 10 CFR 50.34. The Committee provided a report to the Commission dated April 22, 1996. In that report the Committee provided several comments, including the following:

- The proposed final rule requires that any individual, located at any point on the exclusion area boundary for any two-hour period following the postulated release of the fission products, not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE). Similarly, an individual located at the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the release of the postulated fission products (during the entire period of its passage), not receive a dose in excess of 25 rem TEDE. The Committee recommended that careful definition of the TEDE limits that are mindful of organ dose weighting factors found in 10 CFR Part 20 be included in the final rule.

- The proposed final rule states that the evaluation of the radiological doses should be over the two-hour period of maximum dose. The RES has a differing view and recommends that the proposed final rule be modified from any two-hour period after release of fission products (referred to as the "worst" two hours) to a period of two hours commencing with fuel failure (referred to as the "first" two hours). RES believes that the use of the worst two-hour period in the dose calculation is not justified by risk considerations and

could lead to increased costs for future licensees with no commensurate gain in safety. The NRR staff contends that while the revised dose evaluation in 10 CFR 50.34 is intended for future plants, the staff is concerned that a current licensee might seek to use it to remove or disable existing fission product cleanup systems. This could markedly change the risk profile of the plant from that which was licensed.

The Committee was not persuaded by the rationale provided by RES in favor of the first two-hour dose calculation. The Committee agreed with the position taken in the proposed final rule.

Attachment:

- Report dated April 22, 1996, from T. S. Kress, ACRS Chairman, to NRC Chairman Shirley Ann Jackson, Subject: Proposed revision to 10 CFR Parts 50 and 100 and proposed Regulatory Guides relating to Reactor Site Criteria (pp. 30-31)

April 22, 1996

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: PROPOSED REVISIONS TO 10 CFR PARTS 50 AND 100 AND PROPOSED REGULATORY GUIDES RELATING TO REACTOR SITE CRITERIA

During the 430th meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1996, we reviewed the proposed revisions to reactor siting regulations and associated Regulatory Guides and Standard Review Plan sections. Our Subcommittee on Extreme External Phenomena reviewed this matter during a meeting on April 3, 1996. During this review, we had the benefit of discussions with representatives of the NRC staff, Westinghouse Electric Corporation, and the Nuclear Energy Institute. We also had the benefit of the document referenced.

The staff has proposed final revisions to 10 CFR Parts 50 and 100 and a new Appendix S to Part 50 that deal with both seismic and source term issues for future plants and sites. Many of the implementation details will be found in new Regulatory Guides and in Standard Review Plan sections. The existing requirements of 10 CFR Part 100 and its Appendix A will remain in effect for operating plants.

We recommend that the proposed final rule dealing with the seismic aspects be issued.

The proposed final rule requires that any individual, located at any point on the exclusion area boundary for any two-hour period following the postulated release of the fission products, not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE). Similarly, an individual located at the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the release of the postulated fission products (during the entire period of its passage), not receive a dose in excess of 25 rem TEDE. Consistency within the body of NRC regulations is most desirable. We recommend that careful definitions of the TEDE limits that are mindful of organ dose weighting factors found in 10 CFR Part 20 be included in the final rule.

Radiological doses are to be evaluated over a two-hour period. The proposed final rule states that the evaluation should be over the two-hour period of maximum dose. The Office of Nuclear Regulatory Research (RES) has a differing view and recommends that the proposed final rule be modified from any two-hour period after release of fission products (referred to as the "worst" two hours) to a period of two hours commencing with fuel failure (referred to as the "first" two hours). RES believes that the use of the worst two-hour period in the dose calculation is not justified by risk considerations and could lead to increased costs for future licensees with no commensurate gain in safety.

The staff supporting the proposed rule states that (1) the proposed licensing framework would provide a relaxation of engineered safety feature (ESF) performance requirements commensurate with updated source term and radiological insights, (2) the regulatory requirements for determination of in-containment radioactive material during the two-hour dose evaluation period would be consistent and capable of handling designs substantially different from those analyzed in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," (3) the analysis would be easy to perform and reproducible with confidence, and (4) the technical bases and analytical methods would be defensible. While the revised dose evaluation in 10 CFR 50.34 is intended for future plants, the staff is concerned that a current licensee might seek to use it to remove or disable existing fission product cleanup systems. This could markedly change the risk profile of the plant from that which was licensed.

We are not persuaded by the rationale provided by RES in favor of the first two-hour dose calculation. We agree with the position taken in the proposed final rule, and recommend that the rule and the associated Regulatory Guides and SRP sections be issued.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

REFERENCE:
Memorandum dated March 6, 1996, from T. P. Speis, Office of Nuclear Regulatory Research, NRC, to J. T. Larkins, ACRS, transmitting Revisions to 10 CFR Part 100, Reactor Site Criteria, Revisions to 10 CFR Part 50, New Appendix S to Part 50 (Final Rules) and Associated Regulatory Guides and Standard Review Plan Sections

ITEM B.4:

STATUS OF ACRS REVIEW OF REGULATORY GUIDANCE DOCUMENTS RELATED TO DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

(DR. MILLER)

ITEM B.4: STATUS OF ACRS REVIEW OF REGULATORY GUIDANCE DOCUMENTS RELATED TO DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

During the December 8, 1995 meeting with the Commissioners, the ACRS discussed the issues associated with the use of digital instrumentation and control (I&C) systems in nuclear power plants. The ACRS reviewed the findings of the National Academy of Sciences/National Research Council (NAS/NRC) Phase 1 study which defined the important safety and reliability issues concerning hardware, software, and man-machine interfaces. The ACRS provided a report to the Commission dated October 31, 1995, on the NAS/NRC Phase 1 study report.

The report of the NAS/NRC Phase 2 study, which is to identify criteria and provide recommendations for NRC staff review of digital I&C systems, is expected to be completed by September 30, 1996. The staff is developing Standard Review Plan (SRP) Sections, Branch Technical Positions (BTPs), and Regulatory Guides associated with digital I&C systems; this is being done on an expedited basis as directed by the NRC Chairman. Lawrence Livermore National Laboratory (LLNL) is the contractor assisting the staff in the development of the above documents.

On March 6 and May 22, 1996, the ACRS Subcommittee on Instrumentation and Control Systems and Computers discussed with representatives of the NRC staff and LLNL the following proposed SRP Sections, BTPs, and Regulatory Guides:

- Draft Standard Review Plan, Section 7.0, "Instrumentation and Controls — Overview of Review Process"
- Draft Standard Review Plan, Section 7.1, "Instrumentation and Controls — Introduction"
- Draft Standard Review Plan, Section 7.2, "Reactor Trip System"
- Draft Standard Review Plan, Section 7.9, "Data Communications"
- Proposed Branch Technical Position, HICB-14, "Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Safety Systems"
- Proposed Branch Technical Position, HICB-16, "Guidance on the Level of Detail Required for Design Certification Applications Under 10 CFR Part 52"
- Draft Regulatory Guide, "Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"

- Draft Regulatory Guide, "Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"

- Draft Regulatory Guide, "Software Unit Testing for Digital Computers Software Used in Safety Systems of Nuclear Power Plants"

- Draft Regulatory Guide, "Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"

- Draft Regulatory Guide, "Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"

- Draft Regulatory Guide, "Software Test Documentation for Digital Computer Software in Safety Systems of Nuclear Power Plants"

The full Committee also discussed these documents during its March and May 1996 meetings. The full Committee plans to discuss a proposed ACRS report on this matter during its May 1996 meeting. So far, the ACRS has completed its review of all draft regulatory guides associated with the digital I&C Systems. The Committee plans to complete its review of the remaining SRP Sections and BTPs in October 1996. The Committee also plans to review the NAS/NRC Phase 2 study report and the NRC staff's proposed integration of the insights from the NAS/NRC report into the regulatory guidance documents when they become available.

ITEM B.5:

STATUS OF ACRS REVIEW OF
STANDARD PLANT DESIGNS

(MR. CARROLL)
(MR. LINDBLAD)
(DR. CATTON)

ITEM B.5: STATUS OF ACRS REVIEW OF STANDARD PLANT DESIGNS

- ABWR AND SYSTEM 80+ DESIGN CERTIFICATION RULES

Advanced Boiling Water Reactor (ABWR) — The ACRS reviewed the ABWR design and several related design issues and provided a report on the safety aspects of such design on April 14, 1994. The NRC staff issued its Final Safety Evaluation Report (FSER) in July 1994 (NUREG-1503). The FSER delineated the scope of the technical details considered in evaluating the proposed design. On July 13, 1994, a Final Design Approval (FDA) was issued.

ABB-CE System 80+ — The ACRS reviewed the System 80+ design and several related design issues and provided a report on the safety aspects of such design on May 11, 1994. The NRC staff issued its FSER in August 1994 (NUREG-1462). An FDA for System 80+ was issued on July 26, 1994.

Design Certification — Design certification is essential to achieve the goals of 10 CFR Part 52, and it is fundamental to the early resolution of safety issues necessary to ensure a predictable and stable licensing process. The NRC issues a design certification by means of rulemaking proceedings. There is an opportunity for public hearing at this stage. On April 7, 1995, the NRC issued proposed design certification rules for the GENE ABWR and the ABB-CE System 80+ designs for public comment. The comment period ended on August 7, 1995. The Nuclear Energy Institute (NEI) coordinated the industry comments on these proposed rules. Comments were received from NEI, vendors, utilities, Department of Energy, and a public interest group.

The NRC staff requested Commission guidance on two issues. These are the "applicable regulations," and the "verification of inspections, tests, analyses, and acceptance criteria (ITAAC)." In reviewing the applications for design certifications, the staff proposed new requirements from various Commission papers, such as, SECY-90-016, "Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements." These new requirements deviate from or are not embodied in existing regulations. In SECY-96-077, "Certification of Two Evolutionary Designs," the staff selected the new requirements that were applicable to each evolutionary design and evaluated the design information to determine how those requirements were met. The staff designated these requirements as "applicable regulations." The industry strongly opposes the staff's proposed approach. Currently, NEI and the staff indicate an agreement on the ITAAC verification issue.

The ACRS is scheduled to discuss the design certification rules for the ABWR and System 80+ designs during its June 12-14, 1996 meeting. The staff and NEI will provide their views during that meeting.

- AP600 DESIGN

Westinghouse briefed the ACRS regarding the AP600 passive plant design in June 1991, January 1995, and May 1995. Westinghouse issued both the AP600 Standard Safety Analysis Report (SSAR) and the AP600 Probabilistic Risk Assessment (PRA) on June 26, 1992. Westinghouse issued the latest revision to the SSAR on March 29, 1996, and the latest revision to the PRA on November 16, 1995. The staff issued the Draft Safety Evaluation Report (DSER) on November 30, 1994. Items in the DSER were insufficiently developed to support a Committee review of the technical issues.

In a letter dated November 15, 1995, Westinghouse requested that the review of the AP600 design be extended beyond the dates in SECY-94-177, "Revised Review Schedule For the Design Certification Applications." In a letter dated February 2, 1996, Westinghouse requested that the staff expedite its review of the AP600 design in order to support Final Design Approval by April 1997. The ACRS Subcommittee on Westinghouse Standard Plant Designs plans to begin reviewing the AP600 PRA in June 1996. Since September 1994, the Thermal Hydraulic Phenomena Subcommittee has held several meetings regarding the Westinghouse AP600 test and analysis program.

The ACRS was informed that the staff is preparing a proposed Commission paper containing staff positions on the following policy issues concerning the AP600 design:

- Prevention and Mitigation of Severe Accidents
- Post 72-Hour Actions
- Safe Shutdown Modes
- External Reactor Vessel Cooling
- Spent Fuel Pool Cooling System

The ACRS Subcommittee on Westinghouse Standard Plant Designs plans to hold a meeting in July 1996 to discuss this matter. Subsequent to the Subcommittee meeting, the full Committee plans to review this matter during its August 7-9, 1996 meeting.

Commissioner Rogers requested that the ACRS provide guidance on a previous Westinghouse proposal to reference the 1994 Addenda of the ASME Code for piping support design, and on the inspections, tests, analyses, and acceptance criteria (ITAAC) for AP600. Subsequently, Westinghouse decided to reference the NRC-endorsed 1989 Addenda of the ASME code in the SSER. The ACRS plans to review the 1994 Addenda after the ASME working group, which is reviewing the Addenda, completes its evaluation in late 1997. The ACRS plans to begin reviewing the AP600 ITAACs after they are received in July 1996.

Attached is a list of past and proposed ACRS meetings concerning the AP600 design. The schedule is contingent on Westinghouse and the staff submitting information and completing design reviews prior to these meeting dates.

- TEST AND ANALYSIS PROGRAMS ASSOCIATED WITH THE AP600 AND SBWR DESIGNS

AP600 — Westinghouse completed the major tests associated with the AP600 Test and Analysis Program (TAP), and is developing codes (e.g., COBRA/TRAC, NOTRUMP, GOTHIC) for modeling the behavior of nominal plant design. The ACRS Subcommittee on Thermal Hydraulic Phenomena has initiated its review of the Westinghouse TAP. The Subcommittee expects to hold a series of meetings over the next several months to review the Westinghouse final TAP reports. The ACRS reviewed the Westinghouse Best-Estimate Loss-of-Coolant-Accident (LOCA) Analysis methodology and concluded that the use of this methodology was satisfactory for modelling large-break LOCA for currently licensed plants. Westinghouse plans to use the methodology in support of the AP600 design.

Simplified Boiling Water Reactor (SBWR) — General Electric Nuclear Energy (GENE) recently redirected its plant development program from completing the design certification of the SBWR to licensing a 1000 MWe plant design. GENE requested that the NRC staff conduct an orderly close-out of the SBWR review. GENE also requested that the ACRS comment on the SBWR program TAP document, and the work done to support scaling the results of the test program to the nominal SBWR design. The ACRS Subcommittee on Thermal Hydraulic Phenomena plans to consider the GENE review request after receiving the associated staff safety evaluation reports. Subsequent to the Subcommittee's review, the full Committee will consider this matter.

- PAST AND PROPOSED ACRS MEETINGS CONCERNING THE AP600 DESIGN

The proposed ACRS meetings are for planning purposes to support the staff schedule for issuing the final design approval in April 1997. This schedule is based on the assumption that Westinghouse and the NRC staff will provide, in a timely manner, necessary documents (e.g., FSER, test and analysis program report) that are complete.

Dates	Description of Activity
8/25-26/94	Thermal Hydraulic Phenomena: AP600 and SBWR Confirmatory Test and Analysis Programs (TAP)
11/30/94	**Staff Issued DSER**
1/11/95	Westinghouse (W) Standard Plant Designs: Overview of the AP600 design
2/15-16/95	Thermal Hydraulic Phenomena: COBRA/TRAC codes for W AP600
3/27-28/95	Thermal Hydraulic Phenomena: NRC research program to modify the RELAP5/MOD3 code for W AP600

3/29-30/95	Thermal Hydraulic Phenomena: TAP for AP600 Passive Containment Cooling System
5/31/95	Westinghouse Standard Plant Designs: Commission paper on status of technical and policy issues related to the AP600 design
7/26-27/95	Thermal Hydraulic Phenomena: Documentation for the W COBRA/TRAC code for W AP600
11/15/95	Westinghouse letter requesting extension of the NRC review schedule beyond the dates in SECY-94-177, "Revised Review Schedule for the Design Certification Applications"
1/18-19/96	Thermal Hydraulic Phenomena: Documentation for the W COBRA/TRAC code for W AP600
2/2/96	Westinghouse letter requesting expedited NRC review schedule
5/3/96	**Staff Issued DSER Supplement: Testing Program and Computer Code Development**
5/9-10/96	Thermal Hydraulic Phenomena: Overview of W test and analysis program
6/5/96	Probabilistic Risk Assessment: AP600 Level 1 and Shutdown Probabilistic Risk Assessments
7/96	Westinghouse Standard Plant Designs: Staff Policy Positions Contained in Commission Paper
7/96-1/97	Thermal Hydraulic Phenomena: W test and analysis program and NRC DSER Supplement (Number of meetings to be decided)
8/96	Thermal Hydraulic Phenomena: NRC-RES confirmatory test and analysis program
10/95	Westinghouse Standard Plant Designs: Chap. 2: Site Characteristics Chap. 3: Design of Structures, Components, Equipment, and Systems Chap. 4: Reactor Chap. 5: Reactor Coolant System and Connected Systems Chap. 11: Radioactive Waste Management
11/96	Westinghouse Standard Plant Designs: Chap. 6: Engineered Safety Features

	Chap. 7: Instrumentation and Controls Chap. 8: Electrical Power Chap. 9: Auxiliary Systems Chap. 10: Steam and Power Conversion System
12/96	Westinghouse Standard Plant Designs: Chap. 12: Radiation Protection Chap. 13: Conduct of Operations Chap. 16: Technical Specifications Chap. 17: Quality Assurance Chap. 18: Human Factors Engineering
12/96	**Staff Issues FSER**
1/97	Chap. 14: Initial Test Program Chap. 15: Accident Analyses Probabilistic Risk Assessment

<u>The following meetings will be held as needed to complete the design review</u>:

1/97	Westinghouse Standard Plant Designs: FSER Chapters 1-10
2/97	Westinghouse Standard Plant Designs: FSER Chapters 11-18 and Probabilistic Risk Assessment
3/97	Full Committee Review of the FSER

ITEM B.6:

CONFORMANCE OF OPERATING PLANTS WITH NRC SAFETY GOALS

(DR. KRESS)

ITEM B.6: CONFORMANCE OF OPERATING PLANTS WITH NRC SAFETY GOALS

In the September 20, 1994 Staff Requirements Memorandum, resulting from the September 8, 1994 meeting between the Commission and the ACRS, the Commission requested,

> "further guidance and insight from the ACRS on determining where the current population of operating plants, both individually and collectively, fall in relation to the safety goals."

The Commission and the ACRS met on June 8, 1995, and discussed the status of ACRS activities. At that meeting, the ACRS presented approaches for:

- extrapolating level 2 probabilistic risk analyses (PRAs) results to offsite dose consequences, and

- estimating seismic core damage frequencies from seismic margins plant specific high confidence of low probability of failure values, characteristics of seismic fragility distributions, and seismic hazard curves.

The ACRS also noted its efforts to develop approaches for estimating containment accident progression characteristics for groups of sequences not included in individual plant examinations (IPEs) or individual plant examinations of external events (IPEEEs), and for providing insights and guidance for shutdown risk and for the risk associated with other external event initiators. The ACRS still believes the above is possible but the bounding results would not make the effort worthwhile as there is no urgent need for the information.

The Committee believes that the results from NUREG-1150, "Severe Accident Risks: An Assessment of Five U.S. Nuclear Power Plants," NUREG 4832, "Analysis of the LaSalle Unit 2, Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP)," NUREG 6134, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry Unit 1," NUREG 6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf Unit 1," and the insights from the IPE/IPEEE programs and prior plant specific PRAs provide evidence that the mean risk for the population of commercial nuclear plants does fall below the quantitative safety goals. This will not be definitely known, however, until there are full scope level-3 PRAs of acceptable quality available for most of the plants.

In its April 23, 1996 report to the Commission on "Probabilistic Risk Assessment Framework, Pilot Applications, and Next Steps to Expand the Use of PRA in the Regulatory Decision-Making Process," (page 11) the ACRS stated that the safety goals should serve as the top-down acceptance criteria. The ACRS noted that a restatement of the Commission's safety goal

policy is needed that would allow the use of safety goals on a plant-specific basis. The top-down approach to defining criteria and goals would provide incentives for industry to extend and improve current risk analyses. The ACRS plans to discuss a proposed report to the Commission on this matter during its May 1996 meeting.