UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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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, December 8, 1995

The Commission met in open session, pursuant to notice, at 1:34 p.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, SECY

K. CYR, OGC

T. KRESS, ACRS

R. SEALE, ACRS

D. POWERS, ACRS

I. CATTON, ACRS

C. WYLIE, ACRS

G. APOSTOLAKIS, ACRS

D. MILLER, ACRS

J. CARROLL, ACRS

W. LINDBLAD, ACRS

W. SHACK, ACRS

M. FONTANA, ACRS

P R O C E E D I N G S

[1:34 p.m.]

CHAIRMAN JACKSON: Good afternoon, everyone. The Commission is very pleased to welcome Dr. Kress and the members of the Advisory Committee on Reactor Safeguards. Since this is the first Commission meeting, I believe, with the ACRS since I was appointed Chairman I would like to recognize as I did with you earlier but recognize for this meeting the importance the Commission places on the recommendations of the ACRS.

The ACRS has made significant contributions to the NRC and continues to play a substantial role in providing an independent technical perspective to the Commission on important issues, policies and technical questions.

The issues on today's agenda are of importance and the Commission is looking forward to hearing from you. Before we begin, Commissioner Rogers, do you have any opening comments?

COMMISSIONER ROGERS: Nothing to add, thank you.

CHAIRMAN JACKSON: Dr. Kress, you may proceed.

MR. KRESS: We certainly thank you for those words and recognize that for us to have any inputs at all it is through us giving the best possible technical advice that you can use so that is our objective.

You do have an agenda. The way we propose to proceed through this is to have the cognizant ACRS member on each of these issues give a short discussion on it and you can ask questions as we go along. So the first item is the Proposed Resolution of Generic Issue 78 on the Fatigue Transient Testing Limits and Dr. Powers is our cognizant member on that.

MR. POWERS: As you are aware many of the older reactor vessels were designed to a different version in the ASME code an in particular, they were designed before there was a requirement for an explicit fatigue analysis. So concerns as the plants age arise about the issues of fatigue of the major components in the primary coolant system and generic issue 78 came out of those concerns over fatigue.

The issue of monitoring transient loads during the operations, however, has been subsumed into the fatigue action plan that I believe you have seen and approved. Generic issue 78 has now evolved into a probabilistic assessment of what fatigue could lead to cracking of major components in the pressure vessel.

The staff has undertaken a very sophisticated analysis of the probability of initiating cracks and having those cracks propagate through the walls of major components and lead to failure of those major components.

They have in addition bolstered that very sophisticated analysis with a much more classic analysis in which you hypothesize the existence of cracks with a distribution of sizes and use the available technologies to calculate the probability that those cracks will propagate through the wall and lead to component failure.

In the course of the presentations on these models the ACRS has raised a number of questions about the treatment of uncertainty and the quantification of uncertainty in those analyses. Staff has revealed an excellent understanding of that issue, a sophisticated understanding.

They are currently in the process of reevaluating their codes and have presented to us some information on quantifying those uncertainties. They are still not in a position that they would like to finalize this issue pending examination of some of the details of their codes, what I characterize as dotting "i's" and crossing "t's."

The preliminary best estimate analysis certainly show that these are very flaw tolerant structures that we have and you get very low probabilities of component failure from these analyses.

To finish the analyses the staff plans to undertake some detailed peer reviews of their codes so we have tabled writing a letter concerning the resolution of generic issue 78 until they have had a chance to complete this analysis, what I think will be a fine resolution of this issue and that does not mean that the issue of fatigue is closed. That continues on in the fatigue action plan and the plans the staff has for addressing the issues of fatigue in connection with license renewal.

CHAIRMAN JACKSON: Have you developed a position on whether it is necessary to monitor reactor coolant system transient cycles?

MR. POWERS: I don't think the Committee has developed a position on that per se. I think the general sense is that fatigue of the major components is not our biggest issue when it comes to aging but a specific position I don't believe the Committee has come up with on that. The staff is examining that as part of the fatigue action plan.

CHAIRMAN JACKSON: Commissioner Rogers, questions?

COMMISSIONER ROGERS: Yes. Is there any need for more data here or is this really a study, an analytic study or computer study, that can be carried out with existing data?

MR. POWERS: You know that staff has developed some specific data to supplement their generic data base because of concern over testing samples that have been exposed to fatigue cycles in the air as opposed to fatigue cycles in the actual reactor coolant system.

Part of the outcome of the probabilistic analysis would be to, of course, identify and the associated uncertainty analysis would be to identify if you do need any data to bolster your confidence levels in these analytic approaches.

Right now I think that if the best estimates hold up, we are not going to see a big need for additional data. It will be more a question of the uncertainties. If the uncertainty analysis that we have seen up to today hold up, then it looks like additional data probably isn't needed as crucial in this area as many other areas we could think of.

COMMISSIONER ROGERS: We are coming to closure I believe with the industry on a reliability data base and I don't know, I haven't raised this question with the staff, but I don't know whether there is any intention of including in that data base failures due to fatigue of the type that are involved with this study. Do you think that some thought should be given to making sure that there is adequate reporting of failure data in the accumulation of data for that data base that would be useful here?

MR. POWERS: I think there are fine mechanisms already available for reporting failures. I think the repository for the issues of fatigue, the information there, is really the ASME codes. I think that stands apart from the data base issue you are talking about.

COMMISSIONER ROGERS: Yes, it is just coming from a broader context.

MR. POWERS: Yes and it gets a broader view and a broader, more technical, in-depth, as a universal standard than it would in the data bases we are talking about.

COMMISSIONER ROGERS: Good. Fine. Thank you.

CHAIRMAN JACKSON: Can I take your remarks then to mean that you don't expect that the fatigue limit will be exceeded for most plant components during the 40-year life of a plant?

MR. POWERS: That's right. I think we are going to find that for the major components that the 40-year life time we are in pretty good shape on. One of the issues that has to be confronted in the fatigue action plan is what about those plants that have license renewal and now you are talking about a 60-year lifetime and that is what the fatigue action plan is to address and there clearly is the potential of exceeding the fatigue lifetimes based on the relatively conservative kind of analyses that have been done up until now.

CHAIRMAN JACKSON: But you think that the fatigue action plan will allow the development of a methodology for evaluating fatigue during a license renewal process.

MR. POWERS: I think that is the intention, to have the process you follow and to address the question of what happens if you exceed the cumulative usage factor of one for a plan. Does that negate the possibility of license renewal? I think no. But what do you do about that? What is the appropriate response to those questions. But for the 40-year life time, it looks like we are going to be in fairly good shape based on what we have seen up to now.

CHAIRMAN JACKSON: All right. Dr. Kress.

MR. KRESS: The second issue on our list is the NEI Petition for Rulemaking in the area of fire protection and Dr. Catton is our lead person on that.

MR. CATTON: Thank you, Tom. Some of what I will say is my own view and interpretation of our letter. We are not all in full agreement. We initiated a review of the NEI petition for rulemaking to allow for performance based alternative to Appendix R because we thought it would be a good opportunity to further the cause of risk based regulation.

We were not very happy with what we found. It is our view that you have to have clearly stated goals, freedom in how you achieve them and means to demonstrate that you have achieved them and the NEI Petition was somewhat deficient in all three.

The NEI Petition states that fire modeling and PRA will be used to identify pertinent performance criteria and this is stated without a clear, quantitative performance criteria in the face of a fire threat. My colleague, George, can go into some detail on some aspects of this if you wish.

One cannot calculate probabilities of an outcome without a detailed analysis and detailed analysis in my view must include fire modeling. The uncertainties in the contributors to the calculated result and those due to the model itself must be treated in a meaningful way if the results are to be substantive and NEI does not plan to do this. As a matter of fact, in reading some of the responses to the staff on questions, they were very resistant. That might be putting it mildly.

Information exists. May models can be found in the literature and I believe one has even been blessed by the Swedes for such use but simple things like wire in a cable tray and how it burns and under what circumstances flashover occurs are not really fully understood.

I believe research is needed to bring some of these elements together but in spite of this, I believe you can do an uncertainty analysis. There are techniques available to address it and here I am referring to the approach that was taken for the large break LOCA. There is a means to deal with it. If you don't like the answer, you do the research that is needed to reduce the uncertainty but you can get at it.

At the time we wrote our letter we were convinced that neither the staff nor NEI had any plans for doing what we felt was needed to support the fire PRAs. Methods for validating a given fire model are needed.

Methods for dealing with smoke do not seem to exist. There is a limited fire initiation data base and one still cannot separate the possibility of transient combustibles being where they shouldn't from the fire events. This is mainly because what they do is just record fire events and then count and you may well have had a few gallons of something that you shouldn't in some location.

We have scheduled a subcommittee meeting with the staff to address a number of these issues, in part, I believe because of your request and we have a subcommittee meeting scheduled for the end of February.

At that time we will discuss the status of the task action plan for fire protection, the BNL Fire PRA and also fire barriers and their reliability. I added smoke and modeling to the list to sort of round it out and we are also scheduling a subcommittee meeting for sometime in April where we will really dig into performance based fire regulation. At that time we should hear about what is being done around the world, for example, in New Zealand, Australia and the Swedes.

We support the use of PRA in the development of a performance based rule but want to note that it will take some time to do it and there will be resources associated with the process. There are some difficult questions that have to be answered and the first, what is the safety goal to be achieved by a given rule and how will it be stated.

You could argue that some fraction of the CDF, for example, that could be a goal that you are trying to achieve. Second, can the desired goal be cast into functional requirements?

In other words, how do you get from this safety goal to some thickness of insulation on a particular device that has to be protected and thirdly, what models are available for determining whether or not a functional requirement has been met and how good are they which gets back to the uncertainty.

I would be delighted to address any questions you might have or turn it over to George.

CHAIRMAN JACKSON: My first question was going to be to George. Dr. Apostolakis, I am particularly interested in any particular comments you have from an overall PRA perspective and particular uncertainty analysis and its role.

MR. APOSTOLAKIS: The overall objective, the stated objective, of the petition, of the proposed Appendix S, was that under conditions of fire, the plant systems would be able to perform their safety functions and bring the plant to a stable condition.

So this statement does not include anything probabilistic in it. Then as we read the appendix it says that the licensees will have a number of ways of demonstrating that they have achieved this objective and one of them is PRA. Now if you do a PRA, you produce, of course, probabilistic results, the probability that something will happen and so on.

So what is missing in the proposed appendix is guidance on what to do with those results. In other words, if I am to use PRA to demonstrate that I have achieved a certain objective, I have to have some guidance and some criteria with which I will judge the probabilities that I will produce. Otherwise, my results are inconsistent with the objective of the appendix, of the rule.

So that was the major objection. Now when it came to details where the issue really begged for some probabilistic analysis, the authors of the appendix decided to go to a more traditional language, regulatory language, and talk about credible scenarios, for instance.

Well, what is credible? I mean in the PRA space, you have to define that because there isn't such a thing. In probability theory, there is no term like credible. Now if you say, "Well, any event whose probability is less than 'x' is considered incredible," then at least I have some guidance so when I do my PRA I will know what to declare as credible and incredible.

So these are the things that were missing from the proposed rule and the committee noted that in the letter. That was the main objection really.

CHAIRMAN JACKSON: All right. Thank you. In your September 15, 1995 letter to the EDO, I noted that several ACRS members felt that the use of performance based rules for fire protection is frustrated by the conventional attitudes of regulators and that engineering solutions supporting performance requirements are subjected to a disproportionate standard of proof, I think you said, whereas the prescriptive requirements embodied are accepted without proof, et cetera.

What is the basis for those comments and how does this tie into what we are discussing?

MR. CATTON: Well, it is called a ratchet and part of it came from observations at a 1994 international fire science meeting and there was a talk given by some people just on this aspect and I could get more specific if given the opportunity to write them down but I can't pull them out but I will give you one example.

CHAIRMAN JACKSON: I will tell you what would be helpful to me is if you could marry the comments in that letter vis-a-vis prescriptive regulation to what you feel needs to be done relative to a PRA framework in this area.

That would be very helpful.

MR. CATTON: Well, you have to believe it and that really is what that was getting at. If you don't believe the PRA, then all you are going to wind up doing is ratcheting the industry because then they will get to do both and until they do it the way you want them to do it, you don't accept what they have done.

That is the kind of thing we are trying to get at. If you don't believe in analysis, you won't accept the results of analysis. So it stays prescriptive and everything you learn gets dumped in on top of it and that is the ratchet.

So you discover something via the analysis that is good, you can't take advantage of it because people don't believe it. If you discover something bad through this analysis, you ratchet up the level of regulation. That is a fear that everybody has and we felt that it would be a good idea to mention it, at least some of us did.

CHAIRMAN JACKSON: Dr. Apostolakis.

MR. APOSTOLAKIS: Well, yes, if you don't want to implement a new approach, then it is natural to attack the tools of that approach and say that the uncertainties are very large, this tool doesn't do this, doesn't do that and so on and that is really the wrong comparison.

What you should be doing is compare these tools and the approach with the tools that you are already using not with an abstract idealized approach to the problem because then the tools will come out short and we saw some of that at some of the meetings where there was a presentation where this tool, for example, had so many uncertainties and we don't know what to do with it and therefore, we don't want to approach the problem this way and I think that is where the genesis of this sentence of the disproportionate scrutiny came from.

CHAIRMAN JACKSON: I will tie it together. Are you saying that on the one hand what that letter was meant to deal with was an over conservatism but on the other hand you are saying that the approach as outlined as you understood it in the NEI petition didn't address the sorts of issues that would need to be addressed to make the PRA credible, the PRA as applied in the area credible? Is that what I should take away from the two things?

MR. APOSTOLAKIS: Yes.

CHAIRMAN JACKSON: Let me ask this following question. Should the revision of fire protection regulations be approached in phases where portions of the regulation that might relax requirements and one has to be careful with this, that are marginal to safety and I use that word in quotes, be revised while further research continues or is it something that does not lend itself to that kind of incremental approach?

MR. CATTON: I think you can proceed straightaway but I think you have to do it understanding the drawbacks or the weaknesses in the tools you are using. For example, there is a demonstrated process to deal with the modeling. You can use that. If the uncertainties are too large then you have to back up and do it again.

One of the things that we toyed with in putting the letter together is whether or not we should recommend that there be a pilot program of some kind so that we could get these issues on the table and deal with them.

CHAIRMAN JACKSON: All right. Let me ask you the following question. You mentioned resources and again it ties in in some sense to all of these questions, do you have any comments relative to whether the current NRC staffing and expertise give us the capability to proceed in this area or what we would need to do to have that expertise?

MR. CATTON: By resources, I am referring to you may have to put some dollars into some kind of a research program. That is one. The second is personnel. Things are changing. Our first interaction with the fire protection people was the old standard prescriptive approach. They are changing. Each time we meet it is better. So some of my concerns in that regard have been alleviated just by the interaction we have been having.

I believe you management is pushing very hard from the top to get this change to take place. There is still a little bit of resistance that is detectable. Now I suggested to you earlier today that one area that you might look into is what the foreign people are doing.

The French, for example, have actually run tests where they are trying to put together risk based or performance based regulation. So they run the test and see how it works. They actually have built a rather large facility to light fires in.

The Canadians have an international fire center where they are supposedly focusing on risk based fire regulation and a lot of people from all over the world spend time at this particular facility.

CHAIRMAN JACKSON: All right.

MR. KRESS: With respect to your question on whether it should be phased in or some other approach taken, the Committee is not always of one mind on such things as that. That is one of the reasons we end up with Added Comments and the question can be taken to be a general one with respect to risk informed performance based regulations. How to get there and in what manner do you do it?

There are a number of us on the committee that are pretty strong believers in defense-in-depth and don't want to see that sacrificed in the name of other types of regulation. We think it is possible to keep both.

The thought process needs to be done to be sure that when one moves to either a purely risk base or performance oriented that you preserve that defense-in-depth. So we do look for pilot programs.

Personally, I don't want to see a wholesale change of the regulations all at once. I think an easier, more phased approach would be a lot better.

CHAIRMAN JACKSON: I am taking notes on your comments. Commission Rogers, do you have questions?

COMMISSIONER ROGERS: Over the years in many times in my visits to plants I have seen some strange looking situations that when I asked about them was told, "Well, we had to do that to satisfy Appendix R" and it was clear that often fire protection really in satisfying appendix R really resulted in inaccessibility of systems for monitoring their status and so on and so forth and the problem seemed to me to be that with a very heavy emphasis on fire protection and not accompanied with a sufficient concern about how the optimum fire protection would impact on the optimum operation and safety of the plant, one could get into some rather ridiculous situations.

I wonder whether in anything you have been thinking abOut and talking about here, whether you are putting that into your thinking as well because that seemed to be one of the deficiencies of Appendix R, that when you turn the fire protection experts loose on a system and tell them to make it as fire proof as possible, it may be very difficult to do anything with that system after it is so fireproof.

I think that it comes back to taking a look at the entire system and its functionality and, of course, that is what we like to look to you folks for because you have such a breadth of experience and interest that when you are looking at something like fire protection, you are also I am sure mindful of the fact that you are trying to protect an operating plant that also has to be protected from other kinds of challenges. It comes back to the defense-in-depth issue.

It could very well be that really doing a super job on fire protection in fact deteriorates defense-in-depth in some ways. I wonder to what extent you have been talking about this issue in connection with fire protection.

MR. CATTON: We have certain members who are very sensitive to these kinds of things and some who are less so like myself. There is immediate reaction from those people to try to keep us honest with respect to the overall system. But I think the fact that you do a PRA addresses that if it is done properly.

But there is an interesting paper that you might be interested in. It was entitled, "Magic Numbers and Golden Rules." That is what Appendix R is, magic numbers and golden rules. Those can get you in trouble and sometimes the rule that you have to abide by actually makes the thing worse but it is built into history.

COMMISSIONER ROGERS: Yes.

CHAIRMAN JACKSON: It is interesting to me as I listen and I am interested from the PRA experts perspective as to why a risk informed approach can't allow you to get at the kind of optimization which is really what we are talking about, what Commissioner Rogers is talking about.

MR. APOSTOLAKIS: I believe it can and I believe this is another area where our members need to be a little better educated and maybe come to have the same views. There is such a thing as defense-in-depth in the PRA space, I believe, and in fact I plan to raise the issue at one of our upcoming PRA subcommittee meetings. Defense-in-depth is not something that is exclusively the property of the deterministic approach. These are certainly very valid concerns.

Now when a fire risk assessment is done the analysts do not necessarily look at the issues such as would the system be available for something else but that is something that I think can be corrected by giving the proper guidance to the people who will do this kind of analysis because the people who do the fire analysis, of course, it involves systems people but unless you tell them they have to worry about the bigger picture they may not do that and it also involves heat transfer people and so on who really don't think about these things.

But I do believe that in a risk informed performance based regulatory framework you can include all of these concerns and, in fact, that is why we started this series of meetings with the staff and the industry to discuss these things, to make people aware of certain things. But we are still discussing among ourselves what performance based regulation means.

CHAIRMAN JACKSON: In the end, is it not fair to say that if one sits and focuses on one aspect of a plant and says I am going to PRA it to death and then I am going to focus on some aspect of the plant and I am going to PRA it to death without ever having looked at how one system or one part interacts with the other system interaction, maybe even apparently disparate system interaction but looking at how the function of one affects the functioning of another, that you haven't really done an overall risk assessment.

MR. APOSTOLAKIS: Exactly.

MR. CATTON: That's right.

CHAIRMAN JACKSON: But you are the expert.

MR. APOSTOLAKIS: And I agree with you. I think this is true and I think the probability of actually catching something that affects another part of the analysis in a negative way is higher when you do a risk assessment because the way it is done is both the input and the output comes from the bigger picture, the event trees and the fault trees.

So the guys who are responsible for these trees, when they get the results from the PRA, they might ask some questions. I mean, you are telling me this and this and that, but these are not the people though who are regulating the facility.

CHAIRMAN JACKSON: PRAs can't be done in a vacuum.

MR. APOSTOLAKIS: No, that's true.

CHAIRMAN JACKSON: Otherwise, it is mathematics and statistics.

MR. APOSTOLAKIS: True.

CHAIRMAN JACKSON: Any other comments?

COMMISSIONER ROGERS: I have just a couple more little things I would like to just touch on. I am curious as to why the state of modeling seems to be in such an unsatisfactory state of affairs. There are so many situations that I am sure draw parallels in other parts of the industry that I don't know that a nuclear plant is that unique and its susceptibility to fire. In fact, it may be less susceptible than many other industrial situations. Why is it that there really is I take it from what you said, Dr. Catton, a really serious need for additional modeling.

MR. CATTON: I didn't mean to say that. The thing that is different is the focus of the modeling that has been done and the availability may be of experimental data in bringing it to bear on the modeling that has been done.

We know how to write the Navier Stokes equations. We know how to do combustion calculations. I think we know how to do these things but there is the process of validation and acceptance of a given model. That is what is missing.

I was surprised when I read a recent report on the HDR containment that the modeling that was done, what was missing is they didn't know the relationship between the fire and heat release and how the cables decomposed.

The other thing that they were surprised at is that when they moved the cable tray around in some circumstances they got flashover and in others they didn't and they really weren't sure why.

Now I am not a fire expert, maybe fire experts could answer that question for me but I was surprised to see that in the report and I will look into that. It is data that is missing.

MR. APOSTOLAKIS: I would like to make a comment. I think we should distinguish between the state of the art in fire science and in fire risk assessment as practiced in the nuclear industry.

In fire science, they have very sophisticated models to model smoke propagation and all sorts of things, compartment fires and so on. Fire risk assessment for our industry have not really attracted the attention they should have over the last ten or 15 years.

If I look at the first risk assessment that is being done today it differs very little from what we did for the Zion and Indian Point PRAs in 1980 and the code that is being used was developed by me and a graduate student and that is not very good.

It has been 15 years and one would expect a much more sophisticated code to be used now but it is not being used. So that is part of the frustration that the added comments tried to reflect.

COMMISSIONER ROGERS: The other point was do you have anything to say about the pilot RuleNet project that we have launched on deriving comments on RuleNet?

MR. CATTON: Oh, RuleNet.

COMMISSIONER ROGERS: RuleNet, yes.

MR. CATTON: I just found out about it. Actually there was an article in the Los Angeles Times and I haven't had an opportunity yet to test it but the article in the Los Angeles Times said that it is a fog index. We don't understand all this technical stuff yet people tell me that it is the ultimate in simplicity to read it. So I don't know. But next time we meet, I will let you know.

MR. KRESS: Several of us intend to get on the RuleNet.

COMMISSIONER ROGERS: It is a new NRC initiative and it is something that we are just trying on a pilot basis. I commend you to look at it at any rate.

MR. CATTON: We will.

MR. KRESS: It certainly sounds like a good idea.

COMMISSIONER ROGERS: Through your home computer or office computer and see what it looks like. I have asked our General Counsel whether she saw any particular problems with you as individuals actually participating in RuleNet as individuals and so far I take it none.

MS. CYR: None.

COMMISSIONER ROGERS: So you can even get in the game.

MR. CATTON: We can nit-pick from a distance.

COMMISSIONER ROGERS: Yes, so I would call it to your attention at any rate because the fire protection rule is the first pilot project that we are trying with this and we are very interested to see how this broader mechanism for public participation is going to work and you are very welcome to join in the fun.

MR. CATTON: I'll do it.

CHAIRMAN JACKSON: Dr. Kress.

MR. KRESS: The next item on the agenda is the development of improved nondestructive examination techniques and Dr. Seale is our lead person on that. Bob.

MR. SEALE: Thank you. Commissioner Rogers requested that the ACRS examine advances in the techniques and possible availability of new ideas in this field at our June meeting and that was later confirmed in a June 6th SRM.

On September 7th, the Committee heard a presentation from the staff and an industry team headed by the EPRI advisory group on NDE. I might say that was one of the most well coordinated and integrated presentations that I have heard in a long time from the two different sides, if you will. They are working the problem and not each other it seemed.

The industry group included B&W, Westinghouse, ABB-CE and ZETEC which is a contractor which also has been developing some of the detectors I understand. Considerable progress has been made in resolving circumferential cracks and toward developing techniques for detecting steam generator tube defects.

The eddy current detector remain the mainstay for the identification of likely crack sites while ultrasonic methods which are slower provide in some cases better resolution of individual cracks once they have been located.

Much of the recent effort has been directed towards speeding up the process of conducting a full inspection of all the tubes in a steam generator and this is increasingly the normal practice during refueling outages for many plants.

In many cases the time that has been allocated to steam generator tube inspection is the second largest or longest block of time in the outage and I might add if we want to think in risk space that is mid-loop operation which is not exactly our most comfortable place to be in terms of risk. So it is time we want to ration as much as we can.

The use of multiple detector heads on the tube probe on line real time digital processing of probe data and the availability or the ability to conduct simultaneous inspections of several tubes at one time have all been integrated.

One point about the real time digital processing of the data is that it not only speeds the process up and allows immediate identification of the places to look to get better resolution and so forth but it tends to make the data evaluation process more uniform.

So you have to believe that there is a QA enhancement that goes with digital processing of those data. So those are all important parts of the speed-up process.

There have been some detectors also that have been developed which have both eddy current and ultrasonic heads in the same probe. So that you can without having to go through all the spaghetti work, you can do both parts of the inspection at the same time.

I might add that one of the recent times when we have discussed this with the Commission the comment was made that Hal Lewis had questioned a person about how many leads came out of the probe and they sort of looked like something you would find on the lab bench in a physics lab, you know.

They used to be pretty one-of-a-kind almost things and some of the probes now are just incredibly well done and rugged. There has just been a lot of progress toward making equipment that would tend to engender confidence in the results that you get.

Currently the industry is focusing on developing techniques and qualification procedures to reliably size identified circumferential cracks. They sizing of the defects can confirm that you are less than 40 percent through the wall with such cracks and that 40 percent by the way is the criterion for tube plugging in some cases.

So at the present time the industry need is not focusing on the need for a better detector, rather the feeling is that the crack location techniques that are now in use detect the cracks and that has been confirmed by the sacrificial examination of tubes that have been identified as having cracks in them and certainly under current regulatory practice the industry is required to plug any tubes that show a circumferential crack.

So the real concern is that we are running out of tubes or we have the potential for running out of tubes. Cracks are identified which may not have further growth in them or they are very early in the growth cycle. In these cases, it can be argued that an immediate plugging is not required for safety. Rather, continued crack monitoring with end of cycle inspection to measure indicated crack growth would be an appropriate action.

The contention is that too many tubes have been plugged too early and perhaps some didn't really need to be plugged at all. So the industry is worrying now about a new steam generator rule.

The staff in the letter that we wrote on this rightly pointed out that an extensive data base will be necessary to support any changes in the existing rule. So we need to work to resolve these requirements. I will say the staff has accepted alternate repair and plugging criteria for axial cracks in the tube support plate crevice region which is a step in that direction.

The techniques for detecting the cracks seem to be pretty good, very good, and I think there is growing confidence with them partly because a lot more data that has consistency in the sense that there has been a lot of sequential looking at steam generator tubes between or as you go through different life cycles so that you really feel that you are looking at the same thing in different stages of its evolution and so on and that has given a lot of confidence to the results that have been detected or that they have so far.

COMMISSIONER ROGERS: Now really what you are talking about though are principally it sounds to me like largely on the laboratory scale though. It wasn't long ago that had the Maine Yankee steam generator circumferential crack problem in which a licensee followed NRC regulations and in doing their tube examination and didn't see anything.

When they went back using an improved probe, these cracks then started to turn up that hadn't been seen before or at least they weren't aware of them before. Now that wasn't very long ago.

So I am a little concerned about whether these new improved techniques that you are referring to are actually getting extensive use in the field or whether this is something that looks like it has great promise in the laboratory but hasn't really been translated into standard practice in the field.

MR. SEALE: Our impression is that it is getting out into the field and is being used and has been used in several cycles now by some people.

COMMISSIONER ROGERS: That is very important.

MR. SEALE: Bill, did you want to make any comments on that?

MR. SHACK: Maine Yankee isn't all that long ago.

COMMISSIONER ROGERS: Right.

MR. SHACK: But I believe Dr. Seale is correct, that there has been a great deal of progress in that and it is reflected in what one might call an inspection transient. Steam generators are now finding lots of circumferential cracks and I don't really believe that they suddenly appeared.

COMMISSIONER ROGERS: Yes.

MR. SEALE: Yes.

MR. SHACK: It was just that your detections have improved. I know Commonwealth Edison has a substantial number of circumferential cracks at one of their reactors.

MR. SEALE: But nobody is arguing about, or at least not too many seem to be arguing about how many tubes they are going to inspect. They are generally inspecting them all.

COMMISSIONER ROGERS: Yes, so I understand.

MR. SEALE: It seems to be very much the rigor these days.

COMMISSIONER ROGERS: The implications of these with respect to the aging question, I think, is very important.

MR. SEALE: Real serious.

COMMISSIONER ROGERS: There is where one really needs to have a data base, I think, to go ahead. In your letter of September 15th you said, "We believe that adoption of a new steam generator rule with realistic requirements for demonstrating tube integrity could provide the industry with a strong economic incentive to develop more effective NDE techniques." Could you just elaborate a little bit on what you mean by more realistic requirements?

MR. SEALE: Well, it really has to do with the present practice of plugging a tube if you detect a circumferential crack and that is really regardless of wall penetration.

The thought is that you very quickly begin to use up the margin that is built into the steam generator and now you have de-rate and at some point you have to decide whether or not you really want to run with an 85-percent steam generator or whether you happen to be at at that point.

So it is a question of extending the life of the tubes if you had a set of criteria that you had confidence in in allowing you to continue to use those tubes although you had detected the beginnings of a crack.

MR. KRESS: There is a disincentive to developing that.

COMMISSIONER ROGERS: I understand the disincentive. I am trying to reach for the incentive and I am having a little trouble with it. I can see the disincentive.

MR. KRESS: The incentive would have to be developing a more risk-based criteria for plugging tubes than just through wall. It has to do with the potential for real burst and potential for real leakage which is on the books. The plans are on the books to look at that and that is a bit like what we had in mind.

COMMISSIONER ROGERS: It seems to me that this requires a very careful connection though between those criteria and the NDE techniques so that adopting the technique does buy a benefit with respect to let's put it as a relaxation in some way of some tube requirements and not adopting the technique doesn't buy you that relaxation. You have to have something that really is a clear incentive and not just a disincentive. Thanks.

CHAIRMAN JACKSON: Dr. Kress.

MR. KRESS: The next item has to do with the National Academy of Sciences Study on Digital I&C. Dr.

Miller.

MR. MILLER: Thank you, Dr. Kress. Actually I am going to put the next two items together in kind of one package because I didn't feel that I could effectively separate them and I have also provided a little diagram. I guess every professor has to have a crutch.

I will kind of walk us through this. I originally developed that more for my own benefit but I thought I might share it with you. It is listed as "draft" so any errors in there means it is evolving still.

To put these two issues in perspective I think it is of value to look briefly at history. In the 1992-1993 time frame several things occurred that impact the National Academy Study and also Reg Guide 1.152.

During that period the ACRS through a series of workshops and meetings looked at the issue of regulation of digital I&C and expressed some concern that the regulatory process was inhibiting implementation of digital I&C in nuclear power plants.

Out of that the ACRS recommended a study take place which is the National Academy Study and that study was initiated in the autumn of 1994. Now parallel with that two other things occurred that are kind of coalescing on one objective here.

One, the NRC staff issued a generic letter in

August of 1992 that basically said all digital I&C changes or upgrades would be unreviewed safety items. During the public comment period, industry had the obvious concern about that approach and with that the generic letter then was withdrawn on a temporary basis and industry with the leadership of EPRI and consultation with NRC staff developed what is called the Guideline on Licensing Digital Upgrades and EPRI published that guideline as a report in December of 1993 which was then later endorsed with a couple of exceptions by the NRC in April of 1995.

Again at the same time the staff initiated a study or an evaluation, should we update the standard review plan for licensing of power plants, particularly the I&C aspect of that and that is chapter seven of NUREG-0800.

That process began again in January of 1995 and that process is being primarily taken, the lead on that is taking place by Lawrence Livermore. So if you look at those three issues in that context, I would like to kind of walk you through a little bit of this diagram.

It kind of shows how we are addressing the issues that are identified in the National Academy Study. Now I would remind you that the National Academy Study has now reported phase one and that was done to the ACRS in September of 1995 and to the Commission, I believe, about the same time and the staff in October of 1995.

In that Study they identified eight issues, two strategic and six technical. So what this little diagram shows you hopefully is how we are looking at all those eight issues and I always have to start with way back in history and it demonstrates sometimes the brilliance of our founding fathers so to speak.

IEEE 279 which was really issued as the standard in 1981 was kind of the foundation. I do this diagram from the bottom up because I look at what is the foundation and there are still many individuals who still are practicing I&C including one on the NRC staff who were part of that original standard. That evolved into IEEE 603 which is a very key standard, it is kind of like the pivotal standard for almost everything that is done for nuclear power plant safety systems and I&C.

That standard is endorsed by Reg Guide 1.153 and the revision of that standard is now out for comment for endorsement and that should come back in the winter of 1996. In that diagram anything that is in bold indicates the timing of an NRC action. So September, 1995 Reg Guide 1.153 was issued for public comment.

Now we go up to Reg Guide 1.152 which was endorsed in October of 1995 and Reg Guide 1.152 is basically the criteria for digital computers in safety systems in nuclear power plants. The purpose is interesting. It says, "In conjunction with criteria in IEEE 603, establish minimum functional and design requirements for computers used as components of a safety system." So Reg Guide 1.152 now becomes the key standard for software basically in digital I&C.

Those little arrows then indicate that going over to the left, the EPRI/NUMARC Guideline which was endorsed by Generic Letter 95-02 is very strongly supported, or another way to look at it, leans heavily on Reg Guide 1.152 or IEEE 7432 as does the standard review plan which is upgrading a number of reg guides and also developing branch technical positions. That will lean heavily on Reg Guide 1.152.

Now if you look at the issues coming out of the National Academy Study, a key strategic issue is case-by-case licensing. The EPRI/NUMARC Guideline addresses that in that it provides a guideline for upgrading through the 50.59 process where you avoid the case-by-case studies if you do your upgrades correctly.

It did introduce a slight change, I would call it an ambiguity in the sense that there is a little different of opinion between the industry and the staff but I think we are going to close that in the next year or two and I will mention that briefly in a moment.

The standard review plan also addresses several, I guess, five of the technical issues; software quality assurance, common mode software failure, systems aspects, safety and reliability and commercial off the shelf.

However commercial off the shelf hardware and software is not an easy issue to address. There are several other groups listed over to the left there. EPRI has a program going on with that as does ISA.

They are all going to be addressing that issue from a different perspective both the laboratory perspective at Lawrence Livermore and the industry perspective. So commercial off the shelf hardware or those issues are not easy issues to address and the industry and the staff of the NRC and Lawrence Livermore are all addressing those hopefully in a common way.

Going off the right kind of different issues that don't quite get supported by standards, a strategic issue is the Technical infrastructure. There is concern that there is lack of expertise both within the regulatory group and I think also within industry.

The NRC is addressing that at this point by developing a NUREG which provides for guidance for professional development of the staff and also through a series of what is called regulatory perspectives workshops, the first of which occurred this week and I had the pleasure of becoming a student instead of an advisor and I attended at least a day and a half of that workshop. It was a three-day workshop.

I thought it was a very positive situation. It demonstrated to me that our staff has some strong people, probably not enough of them but some very good people out there doing digital upgrades and working with the industry doing that.

One of the objectives of this is to have those individuals who have the experience and the expertise to be in teaching those who have less experience and expertise. I saw a lot of young faces sitting around and a lot of new people who have just been through the process and just coming off of different parts of the NRC to begin working as inspectors.

I also gained a good perspective of the inspector life versus the staff life versus the ACRS life. They are different. We should all go cycle through all three of those probably. It was an enjoyable two days.

With that then, they are addressing the technical infrastructure issue and I think the industry is doing the same through various in-house training programs. Another technical issue is human factors. That one we looked at as the ACRS on NUREG-0700. We wrote a report on that in November of 1995.

In addition to that, there are two IEEE standards that address human factors plus there is the draft human performance plan which we will be looking at in February of 1996.

So you can see all those activities are kind of coalescing on addressing the issues raised by the National Academy Study. Then the National Academy Study is going to address those issue with their own perspective and that will probably lend interesting differences and hopefully interesting commonalties and with that, we will have synergism hopefully to reach a common goal on a number of issues.

The interesting thing about the issues that were raised at least from the perspective of the ACRS are those issues are common to other industries where software is required for dependability and system operation.

There were really no new issues raised that we as ACRS or NRC staff were not aware of. There were just some perspectives on those issues which caused, I shouldn't say cause, but resulted in some dialogue amongst the ACRS and the National Academy panel and also later interaction between the staff and the National Academy panel.

But I think it is all coalescing on what I hope you would see in the next year or two a situation where the regulatory process unlike in the past where I think it was either real or perceived was inhibiting digital upgrades and digital implementation to a position where the regulatory situation is no longer inhibiting and maybe at times even encouraging digital upgrades.

That would be, I think, our objective for the next two years to see that happen and now that Chairman Jackson has indicated there is going to be an overall study with the staff and you introduce another term or a same term for a different issue, a standard review plan, I think all those will come together in the next two years and hopefully we can sit here two years from now and say regulation is leading and not inhibiting this introduction of this important technology.

CHAIRMAN JACKSON: You mentioned looking at how other industries where safety is of paramount importance have introduced the use of digital technology and you said that there were no new issues raised. Do you see in all of this a methodology for bringing insights from other industries into how the transition occurs and particular things to look out in the transition from one type of technology to the other?

MR. MILLER: I think that is where the National Academy Study we would hope would introduce some insights on that issue. That study panel as you know in meeting those individuals has a breadth of perspective has three or four from the industry, nuclear industry, but five or six from other industries, aviation being the one that is most often cited. We now have planes that are flying in the air which are totally computer controlled.

So we would hope to look at those and I do know that our staff visited Boeing, for example, I believe a few months ago and obtained their perspective. We have an interesting report, trip report, from that group. Boeing has not it the same way we are doing it. They developed a little more in-house capability and are dependent somewhat less on industrial and consensus standard building.

CHAIRMAN JACKSON: Does PRA have a role in assessing the risk implications involved with the use of digital I&C technology and safety related equipment?

MR. MILLER: I believe it does but I think I will defer to my colleague at the other end of the table who has some pretty strong opinions on both software and George is not reticent on strong opinions by the way and also on how it might affect digital I&C. We are planning a workshop together sometime down the line.

CHAIRMAN JACKSON: Is that why you are at opposite ends of the table?

[Laughter.]

MR. MILLER: Well, he has moved from California to MIT and skipped Ohio. I think that is why he is over there but George may have particular comments.

MR. APOSTOLAKIS: There are three of my colleagues who have not made comments about me. Would you care to?

[Laughter.]

MR. KRESS: I think he is a very fine person.

CHAIRMAN JACKSON: Sounds like a Chairman's comment.

MR. APOSTOLAKIS: Well, as you know in non-nuclear industries there is a fairly large group of people who have been doing work on software reliability and there is an equally large group of people who think that this is nonsense, that the fundamental issue is one of specification errors which are design errors in our parlance and there is no theory really in reliability that tells you how to handle design errors. Even for standard PRAs we do not address that issue.

Now when we say is there a role for PRA, well, PRA, of course, is not only the "P," it is not only the probabilities, it is this structured approach with trees and logic diagrams and all that so that part, I think, can be used immediately especially since we are not really looking only at the software in isolation but the software as part of the bigger system.

So we have to combine now the performance of software with the hardware and the humans and so on and the event trees, the fault trees that we have and other more advanced tools are good tools for that.

Now when it comes to probabilities though I don't know. I mean that is where we really have to think very hard as to how to quantify that impact. I don't think that anybody really knows right now how to do that unless we result of course to good old expert opinion.

MR. POWERS: The systematic frameworks tend to be used for code design and things like that and come under the category of systems engineering and it looks much like fault tree and event tree sorts of engineering and does not have a probabilistic aspect of it. But certainly the Boeing aircraft now use the system engineering concept and most of the software companies are using that.

CHAIRMAN JACKSON: But you are saying as far as the application here it is not there yet.

MR. APOSTOLAKIS: It is not there.

MR. MILLER: One of the issues on software and I have gone to a couple of workshops on software development and so forth as George pointed out the majority of the errors are design errors or translating system requirements into design requirements of the software which in a sense are human factors type errors or human errors.

It was interesting to sit through the workshop this week and listen to a couple of the NRC inspectors talk about digital upgrades and they have a long list, there are a lot of digital upgrades that have already taken place.

Of course, the majority are relatively small systems and so forth but once again, they go through as an inspector and look and ask very pointed questions, have you taken the requirements for your system, how have you translated into these software requirements and do you as the operator of the plant, not really the I&C engineer of the plant, understand how that is done and don't leave it up to the vendor to do that transition because that transition has to have a lot of plant input.

So we can see already our inspection people who are out in the field understand that is where the majority of the problems occur. Even in small, very simple systems, they can occur right there. So to coalesce overall on something that can be used for reliability, I guess in two years maybe I can answer that question better. I am not convinced we are going to do it very easily and thus use it in PRA.

MR. KRESS: I certainly am not an expert on digital I&C or PRA but it occurs to me that it is not that different from some of the things we do in PRA. The question is can you ascribe a probability to a software function that it will not function as intended and will give you some wrong behavior or behavior that is not intended.

I think it is probably possible to do that. I may be the only one on the whole committee that believes that so

I want to be sure I give you that qualification but I think it has to do with looking at the whole output space that you get under severe accident conditions from a software given the input space.

I think it will be possible to look at output space for different accident sequences comparing that to the total output space and develop a probability that you will be in a region that is likely to give you an error given the qualifications and the systems approach you took to developing the software in the first place and some experience on the number of errors you might expect in such a software depending on how complex it is and how many lines it has.

I don't think the process of developing probabilities of failure for software has been explored enough. I think there are possibilities there that people need to think about. We have dismissed it out of hand and I don't think we should.

MR. MILLER: The whole process is not different than hardware. It is different in the sense that in hardware you still have the same problem of translating design requirements into hardware requirements. The difference is in hardware, I will put it this way, Mother Nature tends to be the predominant failure mechanism, things wear out and things fail. In software, I think the human becomes the major failure mechanism and that is in that process of translation and that is more difficult but I will not say impossible to categorize by any means.

CHAIRMAN JACKSON: George.

MR. APOSTOLAKIS: I certainly agree that in principle one can do this and I did not mean to imply earlier that it is impossible. What I meant was that there is a whole new class of problems that we have to think about now and that it is not simply a matter of taking tools, analytical tools, that we already have and using them or adapting them a little bit to the software. We have to think about it.

For example, one of the most common cause for failure and I don't mean common cause failure that we have observed is unusual inputs. Of course, in principle if you could figure out what is the probability of these unusual inputs you can do it but doing it in practice is an entirely different story and I think Dr. Kress' outline is really right on the mark, that one would start with event trees and fault trees, the conditions that would be created but then translating that into a set of undesired inputs to the software that have a certain probability is really something that I don't think will happen in the next two or three years, I really don't think so.

COMMISSIONER ROGERS: Just a couple of points, I really don't know an awful lot about this although I try to follow it but it does seem to me that this question of what the distribution functions are of these various inputs is the really tough problem and that you can do a point analysis and it looks pretty good but then how do you pick the right kind of a distribution function for the inputs to study the sensitivity of this result to those and there is where some very, very strange things can happen and certainly do come out of PRAs in some that I have seen. So maybe that is really where the big difficulty is in these unusual events, what is the distribution function for those. It is very difficult.

MR. APOSTOLAKIS: Exactly.

CHAIRMAN JACKSON: Or how sensitive is the analysis to the kind of distribution that you use.

MR. APOSTOLAKIS: It is not really what kind you use. It is what distribution, what is the probability of these events really. That is a fundamental input. If you don't know that, you can't get at it.

CHAIRMAN JACKSON: It is related.

MR. APOSTOLAKIS: Right now we really don't know and I have seen a series of actual incidents from the space industry and like the Phobus Satellite and that was a Soviet satellite and they sent a command to the satellite to do something and they skipped one of the digits.

Then they realized that the remaining sequence was meaningful to the satellite. Normally it would not be but it was meaningful. It commanded it to go to a self-testing mode and it is still doing it. It has isolated itself from the earth and it is still testing itself.

That is a wierd thing. Then there is another example from the Bruce reactor in Canada where the code executed a certain subroutine and then came back to their own place and there was a partial melt of the fuel.

So these are things that we are not used to handling in the PRA space and they all seem to be so unique. I mean you are trying to generalize a little bit and say that there is a class of problems there. The only class we can see comes back to the debate back in the 1970's on ATWS, things that we haven't thought of but what does that mean. It is easier to talk about things that we haven't thought of than sitting down and putting some probability distribution on those things.

CHAIRMAN JACKSON: At the same time you can't walk away from it.

MR. APOSTOLAKIS: That is certainly true.

CHAIRMAN JACKSON: Because it has happened.

MR. APOSTOLAKIS: That is certainly true.

CHAIRMAN JACKSON: And these things are being put into plants that were designed with completely different kinds of control systems, et cetera, in mind. So I leave you with that to say that it is an important area to really try to focus on and again the staff is going to be doing some work related to this. Commissioner Rogers.

COMMISSIONER ROGERS: Just one other point. You mentioned some concerns with environmental stressors as to whether there has been adequate attention to that in, I guess, the NAS phase one. Do you feel more comfortable on that count now or is that still a concern?

MR. MILLER: I think it was a matter of emphasis. Several members of the Committee, I would say a majority of the committee, believed that the various environmental stressors should have been an issue equivalent to the technical issues raised by the Academy Study. The Study Panel felt it was not quite an equivalent issue. It was a matter of degree of emphasis more than a matter of importance.

I think everybody believes various environmental stressors which really are the same stressors we had with previous systems but do influence digital in different ways and should be carefully looked at.

COMMISSIONER ROGERS: Can't be forgotten about.

MR. MILLER: Definitely can't forget them.

MR. KRESS: We did not disagree with the Academy's thought that it did not need to be raised to the level of a key issue in their study because we thought it was a different kind of animal. It belonged in the equipment qualification and it wasn't different than normal equipment qualification issues are. It is just the effects on these may be different.

COMMISSIONER ROGERS: Fine. Thank you.

MR. WYLIE: Well, I note here though that the ISA is addressing this in their standard and I think the IEEE also has done this.

MR. MILLER: EPRI as a corollary to their guideline has another guideline on EMI/RFI and my understanding is that staff is going to recommend endorsement of that guideline sometime next year on EMI/RFI so I think everybody realizes that they are important. It is a matter of which box you put it into in a sense.

CHAIRMAN JACKSON: Thank you.

MR. KRESS: The last item on the agenda is a status report on the IPE examinations. George Apostolakis.

MR. APOSTOLAKIS: We are planning to write a letter on the IPE program to you by the end of February. We owe you a letter and in planning for that we are also trying to fulfill our desire to become more active early in the process of developing the standard review plan that the Chairman requested the staff to develop and also to contribute to the development of a risk-informed performance based framework.

We are having a series of subcommittee meetings, PRA subcommittee meetings. We started last October and we heard from the staff, of course, but also we had NEI representatives, EPRI representatives and several utilities and owners groups. I think the subcommittee was very pleased to see that the industry is utilizing the experience from the IPEs and the IPEs themselves and they are doing a lot of things to improve their operations and, of course, to become more effective.

The conceptual problems that bother me a little bit were not addressed at that meeting and they bother me in the sense that I think we should address them as soon as possible before we develop a standard review plan or we go ahead with something more specific.

So we are planning to have another subcommittee meeting this coming January, the 10th and 11th of January we will discuss some of these issues. One of them is again what is the role of defense-in-depth in the PRA space but also others, how good are the models that we have and how do we develop a mix of probabilistic and deterministic criteria. Are we ready to go to performance based regulation for certain issues and then for others we are not ready. We need to do more work and so on.

Then the following day, the third day, the 12th, we will have a briefing by the staff on the IPE reviews and the insights they have gathered and some broader discussion as to where we think now after NUREG-1150 and after the IPE exercise, what are the risks that these 108 or 109 plants pose to the public and this will be, of course, a very important input to our letter to you a month later.

We had a planning meeting yesterday with the staff and it went very well. So I think we are on our way.

CHAIRMAN JACKSON: It sounds like all of the questions I have here to ask you are ones that presumably you are going to be addressing.

MR. APOSTOLAKIS: Well, I am ready to take notes.

CHAIRMAN JACKSON: One is what changes are necessary in the current spectrum of IPEs and PRAs in order for them to be acceptable for regulatory applications, which ones? Second, given the current state-of-the-art of these as well as industry's proposals for their use such as on-line maintenance ranging from that to license amendment justification, how should the issue of what limits there should be on the use of PRAs in regulatory decision-making be addressed?

I assume that you have reviewed the PRA framework document.

MR. APOSTOLAKIS: Yes.

CHAIRMAN JACKSON: And if you haven't that you will and if so, what are your observations on it and what follow-up activities might your committee take up in looking at that and if there is any particular aspect of the PRA implementation plan that the ACRS views as particularly important in furthering the agency's capabilities in this area.

MR. APOSTOLAKIS: I think, yes, all these questions will be addressed.

CHAIRMAN JACKSON: So they will be addressed. You would not like to address any of them this afternoon.

MR. APOSTOLAKIS: I think it would be premature. It is only three weeks until that meeting.

CHAIRMAN JACKSON: All right.

COMMISSIONER ROGERS: Just a little bit on that, what is your thinking with respect to the approach that is being taken towards the development of PRA models in the U.S. versus elsewhere in the world, in particular Europe, because I guess that is really where else it is being practiced and whether there are any common features that one could identify that could lead to the development of some kind of guidelines for establishing models for PRA analysis of different plants.

What I have in mind is a concern that there seems to be some degree of arbitrariness in how one actually sets up a PRA model for a given plant and how confident can a regulator be that results of a particular model are pretty representative of what to expect for the whole plant and whether there is some need for agreement on at least some general guidelines in setting up models.

Once the model is there then I think the technology takes over but whether there is something that could be some useful ground to be explored with respect to and I hate to use the word standardized because I don't really want a cookie cutter approach but some rather general guidelines that somehow or other give one the confidence that a PRA that has been done for a particular plant really is going to produce or is going to uncover, let's put it that way, uncover the important weaknesses in the plant that would be uncovered by another PRA done by somebody else.

In other words, is there some commonality here that one can rely on? I think everybody realizes the bottom line number is not really the important output of a PRA. It is really the process itself to some degree is the most important aspect of it but relative sensitivities or relative probabilities are very valuable as outputs from a PRA. Is there something that one can begin to look at in the way of a standardized approach here?

MR. APOSTOLAKIS: Yes. The Community of petitioners have resisted any form of standardization and the argument has been that it is too soon, 15 or 20 years, that the methods are still evolving and so on.

On the other hand, if you look at some things you can see that there is some standardization. For instance, the set of initiating events now that the PRAs are looking at. I mean these are more or less standard. There are about 20 or 25 initiating events.

In the sense that if you look at the PRA and it does not address two or three of those, immediately the question will come up why not? I mean this is the standard list. It used to be that people would do a so-called and that is way back now, random failure analysis and then a common cause failure analysis. That is now inconceivable. I mean the analysis has to include common cause failures and it seems that most people are settling on using the multiple Greek letter method.

So some of the standardization is taking place but to what extent we can do that and formalize it is not obvious and that is something that needs to be done in the next maybe few months so that the SRP can acquire some shape.

CHAIRMAN JACKSON: It can't take 15 years.

MR. APOSTOLAKIS: It cannot take 15 years. That's right. In fact, I think two years now is what the community of analysts think is a reasonable time.

[Laughter.]

MR. APOSTOLAKIS: Now I didn't quite follow your comment about Europe and the U.S.

COMMISSIONER ROGERS: Just whether there is anything different, whether there is anything such as a uniquely European approach to developing models for PRAs or a uniquely American approach here or whether basically, of course, we do find I know in talking to people that many of the European plants have been analyzed by American companies.

MR. APOSTOLAKIS: That's right.

COMMISSIONER ROGERS: It is really not that different from that point of view but I was just searching in my own mind in a recent visit overseas to see whether anyone saw any value in somehow or another writing down some guidelines for the development of PRA models.

The reason that I have an interest in this is that I am worried about PRAs being done by just a small group of contractors. The whole value of the PRA is that the plant does it itself, not that it goes and hires a contractor and produces a shelf of volumes that go up on a shelf that they can point to and never really look at or understand.

But then if the plant is going to do that PRA, then they have to develop some expertise and they have to get started in some way and one would hope that there would be some guidelines to get them launched on this that represent a common knowledge base here.

MR. APOSTOLAKIS: I think that most of the advances in methodology have come out of this country. We have had, of course, the two major pioneering studies especially the first one, the Reactor Safety Study and then NUREG-1150 and also major research programs where models were developed. With the exception perhaps of the human reliability area, I think most of the work has been done here and not in Europe.

Another thing we are doing here which is uniquely, I think, American and it also addresses the issue of the unavailability of standardized methodology is we peer review our studies sometimes to death. We spend more money reviewing them than actually doing them.

I am not sure that that happens in other parts of the world or if it does happen, it happens in a closed room. You don't have a committee. So that was one of the ways, a major way though, that some quality control was imposed on the process.

But coming back to the standardization, I think it can be done. It can be done up to a point though. I would not standardize human reliability.

MR. POWERS: It is my impression that the European countries when you go to level three probabilistic risk assessments have superior technology to what is being employed in the United States. It is also my impression that the staff is now pursuing that.

MR. CATTON: One of the areas that is interesting is how do you incorporate phenomenological models into a PRA in a consistent way. I know that GRS has a program to do this.

CHAIRMAN JACKSON: I know. I just happened to speak with Dr. Birkhoffer.

MR. CATTON: Good.

CHAIRMAN JACKSON: He brought that up.

MR. CATTON: That is a very interesting arena and if you are ever going to go level two and beyond, I think that some of these issues are going to have to be addressed as to how to accomplish it.

COMMISSIONER ROGERS: Thank you very much.

CHAIRMAN JACKSON: Dr. Kress.

MR. KRESS: Did you want to share with us anything that Dr. Birkhoffer said to you?

CHAIRMAN JACKSON: No.

[Laughter.]

MR. KRESS: All right. We have nothing further.

CHAIRMAN JACKSON: I would like to thank the ACRS very much for a very informative briefing to the Commission. Obviously the topics we have discussed are crucial for maintaining and improving the Agency's ability to be effective in our regulations and I would encourage you to continue to provide the Commission with your independent perspectives on issues important to our mission and with that unless Commissioner has some further comments.

COMMISSIONER ROGERS: No, just very thankful for a very fine meeting.

CHAIRMAN JACKSON: Thank you.

[Whereupon, at 3:07 p.m., the meeting was concluded.]

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, December 8, 1995

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

December 4, 1995

MEMORANDUM TO: John C. Hoyle
Secretary of the Commission

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

SUBJECT: ACRS MEETING WITH THE NRC COMMISSIONERS ON DECEMBER 8, 1995—SCHEDULE/BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners between 1:30 and 3:00 p.m. on Friday, December 8, 1995, to discuss the items listed below. Background materials related to these items are attached.

A.	Introduction		1:30 - 1:35 p.m.
B.	1.	Proposed Resolution of Generic Safety Issue 78, "Monitoring of Fatigue Transient Limits for the Reactor Coolant System" – (Dr. Powers) (pp. 1-6)	1:35 - 1:50 p.m.
	2.	The Nuclear Energy Institute Petition for Rulemaking to Amend 10 CFR 50.48, "Fire Protection" (Dr. Catton) (pp. 7-12)	1:50 - 2:05 p.m.
	3.	Development of Improved Nondestructive Examination (NDE) Techniques (Dr. Seale) (pp. 13-19)	2:05 - 2:20 p.m.
	4.	National Academy of Sciences/National Research Council Study on "Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues" – Phase 1 (Dr. Miller) (pp. 20-24)	2:20 - 2:35 p.m.
	5.	Proposed Final Revision 1 to Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants" (Dr. Miller) (pp. 25-36)	2:35 - 2:50 p.m.
	6.	Status report on Individual Plant Examinations (IPEs) (Dr. Apostolakis) (pp. 37-47)	2:50 - 2:55 p.m.

C. Closing Remarks 2:55 - 3:00 p.m.

Attachment: As stated

cc: ACRS Members
ACRS Technical Staff

Item B.1:

Proposed Resolution of Generic Safety Issue 78, "Monitoring of Fatigue Transient Limits for the Reactor Coolant System"

(Dr. Powers)

ITEM B.1: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE 78, "MONITORING OF FATIGUE TRANSIENT LIMITS FOR THE REACTOR COOLANT SYSTEM"

The original scope of Generic Safety Issue (GSI)-78, "Monitoring of Fatigue Transient Limits for the Reactor Coolant System," was to determine whether additional regulatory requirements for transient monitoring (cycle counting) of components in the reactor coolant pressure boundary are necessary at operating plants. In June 1993, the NRC staff developed the Fatigue Action Plan to resolve issues associated with fatigue failure of components in the reactor coolant pressure boundary. The original scope of GSI-78 was subsumed in the Fatigue Action Plan.

During the September 7, 1995 ACRS meeting, the NRC staff briefed the Committee on a draft Commission paper, "Completion of the Fatigue Action Plan," which was later issued on September 25, 1995, as SECY-95-245. In SECY-95-245, the NRC staff did not recommend additional licensee actions to address transient monitoring at this time. In a report to the Commission dated October 16, 1995, concerning the Fatigue Action Plan, the Committee stated that no immediate staff or licensee action is needed based on the work done by the staff and industry. The Executive Director for Operations responded to the Committee's October 16, 1995, report in a letter dated November 14, 1995.

The current scope of GSI-78 is focused on the evaluation of risk from fatigue failure of components in the primary coolant pressure boundary. The NRC staff completed a probabilistic risk assessment parametric study. The staff proposes closing GSI-78 based on the results of the parametric study.

During the September 7, 1995 ACRS meeting, Dr. Powers questioned the NRC staff concerning the influence of uncertainties on the final results of the parametric study and the uncertainties associated with the assumed values for the estimated cumulative usage factor, the stress amplitude, and through wall crack propagation rate. He also questioned the validity of the assumed independence between the probabilities of crack initiation and crack propagation.

On November 3, 1995, the NRC staff met with Dr. Powers and some members of the ACRS to provide additional information related to the above questions. The Committee plans to complete a letter to the EDO regarding GSI-78 during the December ACRS meeting.

Attachments:

- Report dated October 16, 1995, from T. S. Kress, Chairman, ACRS, to Shirley A. Jackson, Chairman, NRC, Subject: Fatigue Action Plan (pp. 3-5)

- Letter dated November 14, 1995, from J. M. Taylor, Executive Director for Operations, to T. S. Kress, Chairman, ACRS, Subject: Fatigue Action Plan (p. 6)

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

October 16, 1995

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

Dear Chairman Jackson:

SUBJECT: FATIGUE ACTION PLAN

During the 425th meeting of the Advisory Committee on Reactor Safeguards, October 5-7, 1995, we completed our deliberations on the Fatigue Action Plan that we started during our 424th meeting, September 7-8, 1995. We had the benefit of discussions with representatives of the NRC staff regarding this matter and of the documents referenced.

The Fatigue Action Plan was developed to help resolve Generic Issue 166, "Adequacy of Fatigue Life of Metal Components." It was intended to address three specific issues: (1) the margin against fatigue failure of older nuclear power plants with reactor coolant pressure boundary components designed to ANSI B31.1 requirements rather than the newer ASME Code Section III, Class 1 fatigue requirements; (2) the effects of reactor coolant environments on fatigue life; and (3) the appropriate staff actions when components have cumulative usage factors (CUFs) greater than 1.

The work done on the Fatigue Action Plan by the staff and the additional work supported by the Department of Energy and the Electric Power Research Institute have shown that, even after including environmental effects, the CUFs for almost all reactor components which were originally designed to ASME Code fatigue requirements will still be less than 1. It also showed that the nuclear piping, which had been designed to the ANSI B31.1 requirements, in general has margins against fatigue failure comparable to those achieved by using the ASME Section III, Class 1, fatigue requirements. Although fatigue failures have been experienced in nuclear plants, these failures have been due to unanticipated loads and not to inadequate design margins for the anticipated cyclic loads.

Based on a probabilistic parametric study, the staff concluded that even if fatigue cracks were initiated, rupture of reactor coolant piping as a result of fatigue crack growth would be a low-probability event. We anticipate commenting on this parametric study at a later time.

The summary of the Fatigue Action Plan provides only general guidance for the appropriate actions to be taken when the CUF is greater than 1. However, the supporting documentation suggests that the proposed nonmandatory appendix to Section XI of the ASME Code provides evaluation methods which may be acceptable to the staff. These methods provide a choice of either the traditional CUF approach or a "flaw-tolerance" approach similar to that widely used in the aerospace industry. We agree that these types of evaluations would be appropriate.

We agree with the staff that maintaining the integrity of the reactor coolant pressure boundary is an important element in defense-in-depth, and that fatigue is a potentially significant mechanism which can degrade the integrity of the pressure boundary. But, on the basis of the work done by the staff and industry, no immediate staff or licensee action is needed.

Dr. William Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>References</u>:

1. Draft Commission Paper, received August 30, 1995, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Completion of the Fatigue Action Plan (Predecisional)
2. U. S. Nuclear Regulatory Commission, NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," published March 1995
3. SECY-94-191 dated July 26, 1994, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Fatigue Design of Metal Components
4. Staff Requirements Memorandum dated May 21, 1993, from Samuel Chilk, Secretary of NRC, to John T. Larkins, Executive Director, ACRS, Subject: Periodic Meeting with the Advisory Committee on Reactor Safeguards, Friday May 14, 1993

5. Letter dated August 17, 1992, from David A. Ward, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Related Branch Technical Position On Fatigue Evaluation Procedures

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

November 14, 1995

Dr. Thomas S. Kress, Chairman
Advisory Committee for Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington D.C. 20555-0001

SUBJECT: FATIGUE ACTION PLAN

Dear Dr. Kress:

I am writing in response to your October 16, 1995, letter concerning resolution of the subject Fatigue Action Plan. We appreciate your review, and agree that no immediate staff or licensee action is needed on fatigue.

With regard to the committee's anticipated comments on the probabilistic study, I note that the staff met informally with ACRS members on November 3, 1995, to discuss a series of questions on that study. It is my understanding that the questions were satisfactorily answered by the staff at that meeting.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: Chairman Jackson
Commissioner Rogers
SECY

Item B.2:

The Nuclear Energy Institute Petition for Rulemaking to Amend 10 CFR 50.48, "Fire Protection"

(Dr. Catton)

ITEM B.2: THE NUCLEAR ENERGY INSTITUTE PETITION FOR RULEMAKING TO AMEND 10 CFR 50.48, "FIRE PROTECTION"

The NRC staff and the Nuclear Energy Institute (NEI) began discussing rulemaking to amend 10 CFR 50.48, "Fire Protection," in April 1993. NEI submitted a petition for rulemaking on February 2, 1995. The NRC staff published the NEI petition and thirteen questions for public comment in the *Federal Register* on June 6, 1995. The staff is using RuleNet, a Fedworld pilot electronic public forum to elicit interactive public participation.

The NEI petition provides an alternative to the prescriptive requirements of the current rule. The petition proposes that plant-specific changes to fire protection programs would be performed by licensees based on advances in fire science, fire modeling, and probabilistic risk assessment techniques. Changes would be documented but would not require NRC approval with sufficient justification. The NEI petition envisions NRC regulatory guides would accept industry guidance documents, which describe acceptable methods for complying with the proposed rule. The NRC staff intends to use SECY 94-090, "Institutionalization of Continuing Program for Regulatory Improvement," as the basis for reviewing the NEI petition.

In June 1995, the ACRS discussed the NEI petition and performance-based regulations with NEI and the NRC staff. In a letter to the Executive Director for Operations (EDO) dated September 15, 1995, the Committee stated that the NEI petition was deficient in the following elements, which should be included in any performance-based regulation:

- clearly stated objectives with demonstrable performance requirements
- flexibility in the methods that licensees are permitted to use to meet the performance goals or criteria
- valid means for the regulatory body to establish that the performance criteria have been met

The Committee supports the use of probabilistic risk assessment in the development of a performance-based rule, but believes it will take time and resources to develop a well-written rule. The Committee noted that regulations of this type should include the right mix of criteria that are based on good engineering practice and on probabilistic risk assessments, which are refined by fire research tests.

The EDO responded to the ACRS in a letter dated October 30, 1995, noting that the staff anticipates that additional research may be necessary to support certain options. The NRC staff plans to forward the rulemaking plan to the Committee. The ACRS Subcommittee on Fire Protection plans to meet with the NRC staff and nuclear industry in the future to discuss a proposed NUREG and rulemaking plan, which support development of a performance-based fire protection rule.

<u>Attachments</u>:

- Letter dated September 15, 1995, from T. S. Kress, Chairman, ACRS, to J. M. Taylor, Executive Director for Operations, Subject: The Nuclear Energy Institute Petition for Rulemaking to Amend 10 CFR 50.48, "Fire Protection" (pp. 9-11)

- Letter dated October 30, 1995, from J. M. Taylor, Executive Director for Operations, to T. S. Kress, Chairman, ACRS, Subject: The Nuclear Energy Institute Petition for Rulemaking to Amend 10 CFR 50.48, "Fire Protection" (p. 12)

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

September 15, 1995

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Taylor:

SUBJECT: THE NUCLEAR ENERGY INSTITUTE PETITION FOR RULEMAKING TO AMEND 10 CFR 50.48, "FIRE PROTECTION"

During the 424th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 1995, we completed our discussion regarding the subject rulemaking petition. Our Auxiliary and Secondary Systems Subcommittee met on June 7, 1995, to begin the review of this matter. During these meetings, we had the benefit of discussions with representatives of the staff, the Nuclear Energy Institute (NEI), and the Electric Power Research Institute (EPRI). We also had the benefit of the documents referenced.

The NEI petition for rulemaking proposes to amend 10 CFR 50.48, "Fire Protection," by adding an Appendix S, which is described as a "performance-based" alternative to the existing prescriptive Appendix R. NEI believes that the recommended addition to 10 CFR 50.48 will be "safety neutral" and that considerable cost savings will result.

We support risk-based regulations. It is not clear, however, how performance-based regulations should be developed from risk consideration. It is our perception that such regulations should include the following elements:

- Clearly stated objectives with demonstrable performance requirements, expressed either in deterministic or probabilistic terms.
- Flexibility in the methods that the licensee is permitted to use to meet the performance goals or criteria. These methods should be supported by operational experience and experimental results.

- The regulatory body must have a valid means to establish that the performance criteria have been met.

Unfortunately, the proposed rule in the NEI petition is deficient in all these elements.

The objective of the proposed rule is to assure "that the safety functions required to safely shut a plant down and maintain it in a safe condition are maintained during and following a fire." It is further stated that fire modeling, as well as PRAs, may be used to identify the pertinent performance criteria. The proposed rule, however, avoids setting probabilistic requirements and uses non-quantitative language. Thus, there are references to "credible" fires and "credible" scenarios, as well as to "adequate" time for completing safety functions. These concepts need to be defined in quantitative, probabilistic terms. For example, we would expect a quantitative performance requirement for the probability that fire will compromise safe shutdown equipment and lead to core damage.

Some of the issues that the proposed rule raises could be naturally resolved in a PRA context. Examples are the inadvertent actuation of automatic suppression systems and the relevance of the current requirements regarding the concurrent occurrence of a fire and loss of offsite power. In addition, the proposed rule does not address the issue of transient fuels. PRAs have shown that, in some cases, transient fuels are required to produce fires of severity sufficient to damage redundant safety systems. Such transient fuels have been found in controlled areas in the past. Not only are transient fuels not addressed, the proposed rule suggests that some administrative controls dictated by Appendix R may be eliminated. We would prefer to see an evaluation of such issues in the context of a fire PRA.

We are concerned that neither the NRC nor NEI has any plans for conducting fire tests for refining the probabilistic analysis of time-to-suppression. We also have concerns about weakening the requirement for automatic fire detection systems, the lack of a methodology for treating the potentially damaging effects of smoke, the use of a limited fire initiation database, and the neglect of consideration of fire during shutdown. We will address these concerns should the rulemaking process advance.

Even though we support the use of PRA in the development of a performance-based rule, we note that, given the uncertainties in the state of the art, fire PRAs cannot be the sole basis for regulatory requirements. Developing the right mix of criteria based on PRA and criteria based on good engineering practice is a challenge and a necessary requirement for a well-written rule.

We believe it will take some time and resources to develop and institute performance-based fire regulation. We also believe doing so is an important step in the agency's move in this direction.

Additional comments by ACRS Members George Apostolakis, James C. Carroll, and Ivan Catton are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Additional Comments by ACRS Members George Apostolakis, James C. Carroll, and Ivan Catton

We support the Committee letter but have further comments for your consideration. The use of performance-based rules for fire protection is frustrated by conventional attitudes. The desire of regulators to have simple rules and tests for administrative convenience contrasts with the need of plant operators to have flexibility to arrive at optimal solutions. Unfortunately, the prescriptive characteristics embodied in regulations are accepted without proof, while any engineering solution supporting a performance requirement is subjected to a disproportionately higher standard of proof.

References:

1. Letter dated February 2, 1995, from W. Rasin, Nuclear Energy Institute, to John C. Hoyle, Acting Secretary, NRC, Subject: Petition for Rulemaking to Amend 10 CFR 50.48
2. SECY-94-090 dated March 31, 1994, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Institutionalization of Continuing Program for Regulatory Improvement
3. SECY-95-034 dated February 13, 1995, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Status of Recommendations Resulting from the Reassessment of the NRC Fire Protection Program
4. Memorandum dated December 30, 1994, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Eighth Quarterly Report on the Status of the Thermo-Lag Action Plan

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

October 30, 1995

Dr. Thomas S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: THE NUCLEAR ENERGY INSTITUTE PETITION FOR RULEMAKING TO AMEND 10 CFR 50.48, "FIRE PROTECTION"

Dear Dr. Kress:

The staff has received your letter, dated September 15, 1995, and appreciates your comments on the petition and generally on performance-based fire protection regulation.

As indicated in SECY-94-090, the staff intends to revise NRC's fire protection regulations to make them more performance-based and risk informed. As you indicated, we also recognize the difficulty of developing performance-based regulation from risk considerations in the area of fire protection. We agree with the Committee that developing the right mix of criteria based on PRA and good engineering practice is a challenge.

In the near term, the staff plans to review public comments on NEI's petition, consider your comments, and develop a plan of action for the rulemaking. We anticipate that additional research may be necessary to support certain options. We will forward this rulemaking plan to you for your information and will be prepared to brief the Committee, if it so desires.

In November 1995, the staff also plans to conduct interactions with the public and industry through Rulenet (electronic "forum" through the information highway) on the petition for rulemaking and other possible approaches for rulemaking in this area. We will inform you of the details of this "forum."

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: Chairman Jackson
Commissioner Rogers
SECY
OGC

12

Item B.3:

Development of Improved Nondestructive Examination (NDE) Techniques

(Dr. Seale)

ITEM B.3: DEVELOPMENT OF IMPROVED NONDESTRUCTIVE EXAMINATION (NDE) TECHNIQUES

In the June 16, 1995 Staff Requirements Memorandum, the Commission requested that the ACRS assist 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.

On September 7, 1995, the ACRS heard presentations by and held discussions with representatives of the NRC staff, the Electric Power Research Institute (EPRI) Technical Advisory Group on NDE, ZETEC, Babcock & Wilcox Nuclear Technologies, ABB-Combustion Engineering, and Westinghouse Electric Corporation regarding ongoing and planned industry activities to improve NDE techniques. The presentations indicated that substantial progress had been made toward resolving circumferential cracking problems and toward developing innovative NDE techniques for detecting steam generator tube defects.

In a report to the Commission dated September 15, 1995, related to development of improved NDE techniques, the Committee noted that adoption of a new steam generator rule with realistic requirements for demonstrating tube integrity could provide the industry with a strong economic incentive to develop more effective NDE techniques. The Executive Director for Operations (EDO) responded to the ACRS report in a letter dated October 16, 1995, pointing out that the ACRS comments on this issue are consistent with the staff perspective and direction; however, a new rule, in and of itself, will not ensure that industry continues to develop and implement improved NDE techniques. The EDO also noted that development of methodologies and databases for other defect mechanisms will require a major effort and significant resources on the part of the industry.

Attachments:

- Staff Requirements Memorandum dated June 16, 1995, from A. Bates, Acting Secretary, Office of the Secretary, to the File, Subject: Meeting with Advisory Committee on Reactor Safeguards, 10:00 a.m., Thursday, June, 8, 1995, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to the Public) (p. 15)

- Report dated September 15, 1995, from T. S. Kress, Chairman, ACRS, to Shirley A. Jackson, Chairman, NRC, Subject: Development of Improved Nondestructive Examination Techniques (pp. 16-17)

- Letter dated October 16, 1995, from J. M. Taylor, Executive Director for Operations, to T. S. Kress, ACRS, Chairman, Subject: Development of Improved Nondestructive Examination Techniques (pp. 18-19)

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.

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

September 15, 1995

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

Dear Chairman Jackson:

SUBJECT: DEVELOPMENT OF IMPROVED NONDESTRUCTIVE EXAMINATION (NDE) TECHNIQUES

During the 424th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 1995, we heard presentations from representatives of the Electric Power Research Institute (EPRI), the EPRI Technical Advisory Group on NDE, Zetec, Babcock & Wilcox Nuclear Technologies, ABB-Combustion Engineering, and Westinghouse Electric Corporation regarding activities to improve NDE techniques for more accurately detecting and assessing steam generator tube defects. The status of staff activities on the development of a new steam generator rule and a supporting research program was also discussed. We had the benefit of the documents referenced.

In the June 16, 1995 Staff Requirements Memorandum, the Commission asked the ACRS to assist the staff in encouraging the industry to develop improved NDE techniques for steam generator tube inspections. The industry presentations at our meeting indicated that substantial progress is being made on the development of techniques that will provide significantly improved capabilities for detecting and sizing circumferential flaws. Not surprisingly, industry efforts are focused on a rapid resolution of the circumferential cracking problem using evolutionary improvements in eddy current technology. In addition, development is proceeding on innovative techniques such as ultrasonic guided (Lamb) waves, in situ fluorescent dye-penetrant inspections, in situ tube burst pressure testing, and combined ultrasonic and eddy current probes. Improved methods of signal processing and display are being developed to aid interpretation of NDE results. We believe modern, real time, signal processing technologies could provide great improvements in signal interpretation, defect detection, and defect sizing.

The staff and industry both recognize that the current regulatory approach to steam generator inspections discourages the development and adoption of improved NDE techniques. In the current framework an increased detection capability leads to more plugging or repair, without necessarily improving safety. We believe that adoption of a new steam generator rule with realistic requirements for demonstrating tube integrity could provide the industry with a strong economic incentive to develop more effective NDE techniques. Careful thought must be given to the requirements for adequate "performance demonstrations" of the NDE techniques essential for implementing a new rule. The steam generator mockup being developed by Westinghouse Electric Corporation under the Office of Nuclear Regulatory Research sponsorship may provide a useful independent regulatory check on the adequacy of NDE inspection techniques.

Dr. William J. Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Staff Requirements Memorandum dated June 16, 1995, from Andrew L. Bates, Acting Secretary of the Commission, Subject: Meeting with ACRS, June 8, 1995
2. NRC Information Notice 94-88, "Inservice Inspection Deficiencies Result in Severely Degraded Steam Generator Tubes," dated December 23, 1994
3. NRC Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes," dated April 28, 1995

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

October 16, 1995

Thomas S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: DEVELOPMENT OF IMPROVED NONDESTRUCTIVE EXAMINATION TECHNIQUES

Dear Mr. Kress:

In your letter to Chairman Shirley A. Jackson, dated September 15, 1995, you provided the Advisory Committee on Reactor Safeguards' (ACRS) assessment of the industry's ongoing efforts to develop improved nondestructive examination (NDE) techniques for tubes in pressurized-water-reactor (PWR) steam generators and the staff's ongoing activities to develop a new rule addressing steam generator tube integrity. In summary, your letter indicated that the ACRS believes the industry is making substantial progress on development of NDE techniques that will significantly improve capabilities for detecting and sizing circumferential flaws. The ACRS believes, as does the staff, that a new steam generator rule with realistic requirements for demonstrating tube integrity could provide the industry with a strong economic incentive to develop (and implement) more effective NDE techniques. However, the ACRS noted that careful thought must be given to the requirement for adequate performance demonstration of the NDE techniques essential for implementing the new rule.

The staff would like to thank the ACRS for the opportunity to present its views on steam generator tube NDE and its ongoing activities relating to rulemaking. We appreciate the comments made by the ACRS and believe that they are consistent with the staff's perspective and direction on these issues. As a point of clarification, however, we would like to point out that a new rule, in and of itself, will not be a panacea for ensuring that industry continues to develop and implement improved NDE techniques. The new rule will provide a regulatory framework within which licensees will have the flexibility to apply the most appropriate NDE techniques for the defect mechanisms of concern without being driven to perform tube repairs and/or plugging beyond that needed to ensure adequate tube integrity. To take advantage of this flexibility for a given flaw mechanism, industry will have to develop the necessary implementing methodology and databases. At present, such methodologies and databases have only been developed for a few mechanisms, most notably outer-diameter stress corrosion cracking at the tube support plates. Development of these methodologies and databases for other defect mechanisms will require a major effort and significant resources on the part of the industry.

We plan to continue with the rulemaking activity by issuing a draft rule in June 1996 for public comment. We hope to again have the opportunity to brief you on this effort before June 1996.

Sincerely,

James M. Taylor
Executive Director for Operations

cc: Chairman Jackson
Commissioner Rogers
SECY
OGC
OCA
OPA

Item B.4:

National Academy of Sciences/National Research Council Study on "Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues" - Phase 1

(Dr. Miller)

ITEM B.4: NATIONAL ACADEMY OF SCIENCES/NATIONAL RESEARCH COUNCIL STUDY ON "DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS IN NUCLEAR POWER PLANTS, SAFETY AND RELIABILITY ISSUES" PHASE 1

During the June 8, 1995 meeting with the Commissioners, the ACRS discussed the issues associated with the use of digital instrumentation and control systems in nuclear power plants and ongoing National Academy of Sciences/National Research Council (NAS/NRC) effort to conduct a workshop and perform a study to assist the staff in developing regulatory requirements for computers and digital instrumentation and controls in nuclear power plant safety systems. In September 1995, the NAS/NRC Committee issued its Phase 1 report which defined the important safety and reliability issues concerning hardware, software, and man-machine interfaces.

On October 5, 1995, the NAS/NRC Committee Chairman briefed the ACRS on the results of the Phase 1 study which identified the following eight key issues - six technical and two strategic:

Technical:

- software quality assurance
- common-mode software failure potential
- system aspects of digital I&C technology
- human factors and human-machine interfaces
- safety and reliability assessment methods
- dedication of commercial off-the-shelf hardware and software

Strategic

- case-by-case licensing process
- adequacy of technical infrastructure

The NAS/NRC report notes that these issues are common to other industries where software is required for dependable operation of systems.

On October 13, 1995, the ACRS provided a report to the NRC Chairman commenting on the NAS/NRC Phase 1 report and staff efforts to develop regulatory guidance. The ACRS noted that the issues identified in the Phase 1 report will be important considerations as digital technology is used more extensively in nuclear power plants. The ACRS highlighted the importance of the staff evaluating the effects of environmental stressors for digital instrumentation and control systems. The ACRS expressed concern regarding a potential conflict between the Phase 2 completion schedule and the staff's schedule for issuing the Standard Review Plan (SRP) and associated regulatory guides. The Committee stated that it is important that the SRP, regulatory guides, and other regulatory guidance benefit from the insights in the Phase 2 report. On October 31, 1995, the Executive Director for Operations sent a letter to the ACRS Chairman expressing agreement with the ACRS comments.

On October 17, 1995, the NRC staff briefed the NAS/NRC Committee on staff efforts in developing programs and guidance for digital instrumentation and control systems. During this meeting, the NAS/NRC Committee and NRC staff held discussions on several issues including, regulatory processes for reviewing safety evaluations performed by the licensees per 10 CFR 50.59, evaluating potential unreviewed safety questions, reviewing case-by-case license amendments, and evaluating the effects of environmental stressors.

The ACRS plans to review and provide comments on the NAS/NRC Phase 2 study report, proposed SRP, regulatory guides, and other regulatory guidance subsequent to receiving these documents.

<u>Attachments</u>:

- Report dated October 13, 1995, from T. S. Kress, Chairman, ACRS, to Shirley A. Jackson, Chairman, NRC, Subject: National Academy of Sciences/National Research Council Study on "Digital Instrumentation and Control Systems, Safety and Reliability Issues" - Phase 1 (pp. 22-23)

- Letter dated October 31, 1995, from J. M. Taylor, Executive Director for Operations, to T. S. Kress, Chairman, ACRS, Subject: The National Academy of Sciences' Report on Digital Instrumentation and Control, Safety and Reliability Issues (pp. 24)

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

October 13, 1995

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

Dear Chairman Jackson:

SUBJECT: NATIONAL ACADEMY OF SCIENCES/NATIONAL RESEARCH COUNCIL STUDY ON "DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS IN NUCLEAR POWER PLANTS, SAFETY AND RELIABILITY ISSUES" - PHASE 1

During the 425th meeting of the Advisory Committee on Reactor Safeguards, October 5-7, 1995, we reviewed the National Academy of Sciences/National Research Council (NAS/NRC) Phase 1 report on Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues. The NAS/NRC Committee Chairman described the results of the Phase 1 report. We also had the benefit of the documents referenced.

The objective of the Phase 1 study was to define the important safety and reliability issues concerning hardware, software, and human-machine interfaces that arise from the use of digital instrumentation and control technology in nuclear power plant operations. The report identifies eight key issues: six technical and two strategic. It notes that these issues are common to other industries where software is required for dependable operation of systems. The report succinctly presents the issues that the NAS/NRC Committee found to be important.

We agree that the issues identified in the Phase 1 report will be important considerations as digital technology is used more extensively in nuclear power plants. In the past, we have called attention to the effects of environmental stressors. The NAS/NRC Chairman stated that the NAS/NRC Committee considered, but decided not to raise this issue to the level of a "key technical issue." We continue to believe this is an important issue that the staff must address as it develops its regulatory guidance for digital systems. However, this is part of the broader issue of environmental qualification of safety-related equipment and does not need to be a key issue of the Phase 2 study.

We have concerns regarding a potential conflict between the Phase 2 completion schedule and the staff's schedule for issuing the Standard Review Plan (SRP) and associated regulatory guides. We believe it is important that the SRP and other regulatory guidance benefit from the insights in the Phase 2 report.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Report dated 1995, from the Committee on Application of Digital Instrumentation and Control Systems to Nuclear Power Plant Operations and Safety, Board on Energy and Environmental Systems, Commission on Engineering and Technical Systems, National Research Council, Subject: Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues - Phase 1
2. Memorandum dated December 2, 1993, from Ivan Selin, Chairman, NRC, to NRC Commissioners, Subject: Computers in Nuclear Power Plant Operations
3. Letter dated July 14, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Proposed National Academy of Sciences/National Research Council Study and Workshop on Digital Instrumentation and Control Systems
4. Letter dated August 23, 1994, from Ivan Selin, Chairman, NRC, to T. S. Kress, Chairman, ACRS, regarding ACRS letter of July 14, 1994 on National Academy of Sciences/National Research Council Proposal for a Study and Workshop on the "Application of Digital Instrumentation and Control Technology to Nuclear Power Plant Operations and Safety"

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

October 31, 1995

Dr. Thomas S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: THE NATIONAL ACADEMY OF SCIENCES' REPORT ON DIGITAL INSTRUMENTATION AND CONTROL, SAFETY AND RELIABILITY ISSUES

Dear Dr. Kress:

I am responding to your letter on this subject, dated October 13, 1995, in which the Advisory Committee on Reactor Safeguards (ACRS) commented on the National Academy of Sciences/National Research Council (NAS/NRC) Phase 1 Report, "Digital Instrumentation And Control Systems In Nuclear Power Plants, Safety And Reliability Issues."

We agree that the issue of environmental stressors is key to the qualification of safety-related digital instrumentation and control systems. Environmental stressors are defined as an issue in the NAS/NRC study Phase 1 report, but not as a "key technical issue." As you know from past briefings to the ACRS, the staff is conducting confirmatory research to investigate and characterize the failure modes and degradation mechanisms of digital technologies proposed for use in nuclear power plants. Furthermore, this research is assessing the impact of smoke on advanced instrumentation and control hardware in nuclear power plants. The goal of this research is to provide the technical basis for a regulatory guide on the environmental qualification of digital instrumentation and control systems. We informed the NAS/NRC Committee about our activities on this key issue in an October 17, 1995, meeting with them.

We share your concerns regarding a potential conflict between the Phase 2 completion schedule and the staff's schedule for issuing the Standard Review Plan (SRP) and associated regulatory guides. Our contract with NAS/NRC calls for the completion of the study by September 30, 1996, which includes the delivery of the Phase 2 report. The staff has expressed its concern and will continue to encourage a timely completion of the NAS/NRC study. We agree that it is important that the SRP and other regulatory guidance benefit from the insights expected from the Phase 2 report.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: Chairman Jackson
Commissioner Rogers
SECY

Item B.5:

Proposed Final Revision 1 to Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants"

(Dr. Miller)

ITEM B.5: PROPOSED FINAL REVISION 1 TO REGULATORY GUIDE 1.152, "CRITERIA FOR DIGITAL COMPUTERS IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS"

On October 5, 1995, the Committee reviewed the NRC staff's proposed final Revision 1 to Regulatory Guide 1.152 which provides a method acceptable to the NRC for complying with the Commission's regulations for promoting high functional reliability and design quality for the use of digital computers in safety systems of nuclear power plants. The revised Regulatory Guide endorses IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations (IEEE Standard 7-4.3.2-1993), with the exception of quantitative reliability goals (Section 5.15).

On October 13, 1995, the ACRS provided a letter to the Executive Director for Operations (EDO), stating that the ACRS had no objection to the issuance of Revision 1 to Regulatory Guide 1.152, subject to modification of the language in the Regulatory Guide regarding the use of quantitative reliability goals to be consistent with the language in the staff's response to one of the public comments. Several Committee Members provided additional comments expressing the view that the staff could have made its point regarding the adequacy of quantitative reliability goals for software without taking exception to the IEEE Standard.

On November 15, 1995, the EDO responded to the October 13, 1995 ACRS letter, stating that the staff had modified the language in the Regulatory Guide as recommended by the ACRS and made changes to address the Committee member's additional comments.

Attachments:

- Letter dated October 13, 1995, from T. S. Kress, Chairman, ACRS, to J. M. Taylor, Executive Director for Operations, Subject: Proposed Final Revision 1 to Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants" (pp. 26-27)

- Letter dated November 15, 1995, from James M. Taylor, Executive Director for Operations, to T. S. Kress, Chairman, ACRS, Subject: Revision 1 to Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants" (pp. 28-36)

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

October 13, 1995

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL REVISION 1 TO REGULATORY GUIDE 1.152, "CRITERIA FOR DIGITAL COMPUTERS IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS"

During the 425th meeting of the Advisory Committee on Reactor Safeguards, October 5-7, 1995, we reviewed the proposed final Revision 1 to Regulatory Guide 1.152. The revised Regulatory Guide endorses IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations (IEEE Standard 7-4.3.2-1993), "with the exception of quantitative reliability goals (Section 5.15)." During this meeting, we had the benefit of discussions with the NRC staff. We also had the benefit of the documents referenced.

Based on our review, we concur with the Regulatory Position of Revision 1 to Regulatory Guide 1.152. However, we offer the following comment.

In the proposed Regulatory Guide, the staff declines to endorse the use of quantitative reliability goals as the sole means of meeting the Commission regulations for reliability of digital computers in safety systems. This position is consistent with our previously expressed views as provided in our report of March 18, 1993 to Chairman Selin. The language used in the staff response to Public Comment 1 on this issue provides a clearer expression of the staff position on quantitative reliability goals than does the language used in the Regulatory Guide. During our discussion, the staff agreed to modify the language in the Regulatory Guide to be consistent with its response to the public comment.

Subject to the staff's planned modification, we have no objection to the issuance of Regulatory Guide 1.152, Revision 1.

Additional comments by ACRS Members George Apostolakis, Ivan Catton, Mario H. Fontana, William J. Lindblad, and Charles J. Wylie are presented below.

Sincerely,

J. S. Kress

T. S. Kress
Chairman

Additional Comments by ACRS Members George Apostolakis, Ivan Catton, Mario H. Fontana, William J. Lindblad, and Charles J. Wylie

We believe that in taking exception to IEEE 7-4.3.2-1993, Section 5.15, the staff is tilting at windmills. We would endorse the Standard in its entirety. The staff could make its point regarding the adequacy of quantitative reliability goals for software without taking exception to this Section.

References:

1. Regulatory Guide 1.152, Revision 1, dated September 1995, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," transmitted by memorandum dated September 1, 1995, from David L. Morrison, NRC Office of Nuclear Regulatory Research, to John T. Larkins, ACRS
2. Institute of Electrical and Electronics Engineers, Standard 7-4.3.2-1993, "Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," September 15, 1993
3. Letter dated July 31, 1995, from C. L. Terry, Group Vice President, Nuclear, TUELECTRIC, to U.S. NRC, Subject: TU Electric Comments on Draft Regulatory Guide DG-1039, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants"
4. Report dated March 18, 1993, from Paul Shewmon, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Computers in Nuclear Power Plant Operations

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

November 15, 1995

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

Dear Mr. Kress:

SUBJECT: REVISION 1 TO REGULATORY GUIDE 1.152 "CRITERIA FOR DIGITAL COMPUTERS IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS"

With reference to your letter dated October 13, 1995, on Regulatory Guide 1.152, the staff has modified the language in the regulatory guide on page 1.152-3 (Attachment 1 of the Enclosure) as recommended by the Advisory Committee on Reactor Safeguards. Further, the regulatory guide was revised on page 1.152-4 to address the additional comments provided by several ACRS members. The revised guide has been sent to the Committee to Review Generic Requirements for their review.

Should you have any questions with regard to this guide, please contact Satish Aggarwal at 415-6005.

Sincerely,

James M. Taylor
Executive Director
for Operations

Enclosure: Memorandum dated October 27, 1995 from D. Morrison to E. Jordan

cc: Chairman Jackson
Commissioner Rogers
SECY

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

October 27, 1995

MEMORANDUM TO: Edward L. Jordan, Chairman
Committee to Review Generic Requirements (CRGR)

FROM: David L. Morrison, Director
Office of Nuclear Regulatory Research

SUBJECT: REGULATORY GUIDE 1.152, REVISION 1, "CRITERIA FOR DIGITAL COMPUTERS IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS," (EFFECTIVE GUIDE)

Attached is a copy of the Revision 1 of Regulatory Guide 1.152 (Attachment 1). This revision of the guide endorses IEEE Std. 7-4.3.2-1993, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," with a minor exception. The proposed revision was issued for public comment in May 1995. The comment period expired on July 31, 1995. One comment letter was received (Attachment 2). The staff responses to public comments are contained in Attachment 3, "Resolution of Public Comments."

A draft Value/Impact Statement was published with the draft of this guide, Task DG-1039, when it was published for public comment in May 1995. No substantial changes were necessary, but a few editorial changes were made for clarity and consistency as a result of public comment. Attachment 4 is the revised Value/Impact Statement.

During its October 1995 meeting, the Advisory Committee on Reactor Safeguards concurred in the regulatory position. However, in their letter dated October 13, 1995 to Mr. James M. Taylor, Executive Director for Operations, the ACRS observed that the staff does not endorse the use of quantitative reliability goals as the sole means of meeting the Commission regulations for reliability of digital computers in safety systems. This staff position is consistent with the ACRS views, as provided by the ACRS to Chairman Selin in their report dated March 18, 1993. In their October 1995 meeting, the ACRS concluded that the language used in the staff response to Public Comment 1 (see Attachment 3) on this issue provides a clearer expression of the staff position on quantitative reliability goals than does the language used in the regulatory guide. Subsequent to the ACRS meeting, the staff modified the language in the regulatory guide to be consistent with its response to the public comment, as recommended by the ACRS.

This guide has no backfit implications. There are no new requirements in Revision 1. The staff simply endorses the advances in technology as reflected in the latest national consensus standards. Compliance with the guidance in this guide is voluntary.

Because the staff does not believe that the proposed action involves backfitting, we have not included the CRGR Enclosure (Staff Response to

Section IV.B of the CRGR Charter). Please note that the language for the implementation section to indicate that there are no backfit requirements was coordinated with the CRGR staff.

If CRGR wishes to review Revision 1 of Regulatory Guide 1.152, please let me know. Your response is requested by November 30, 1995.

Attachments:
1. Revision 1 to RG 1.152
2. Public Comment Letter
3. Resolution of Public Comments
4. Revised Valve/Impact Statement

Revision 1
October 1995

REGULATORY GUIDE 1.152
(Draft was issued as DG-1039)

CRITERIA FOR DIGITAL COMPUTERS IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS

A. INTRODUCTION

Criterion 21, "Protection System Reliability and Testability," of Appendix A, "General Design Criteria for Nuclear Power Plants," in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, among other things, that protection systems be designed for high functional reliability commensurate with the safety functions to be performed. Criterion III, "Design Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," of 10 CFR Part 50 requires, among other things, that quality standards be specified and that design control measures be provided for verifying or checking the adequacy of design.

This regulatory guide describes a method acceptable to the NRC staff for complying with the Commission's regulations for promoting high functional reliability and design quality for the use of digital computers in safety systems of nuclear power plants. The term "computer" is a system that includes computer hardware, software, firmware, and interfaces.

The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 50, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 50 have been approved by the Office of Management and Budget, Approval No. 3150-0011.

B. DISCUSSION

Instrumentation and Control (I&C) systems that use digital computers in safety systems make extensive use of advanced technology, i.e., equipment and design practices that are expected to be significantly and functionally different from current designs. These designs include, but are not limited to, the use of microprocessors, digital systems and displays, fiber optics, multiplexing, and different isolation techniques to achieve the needed independence and redundancy.

IEEE Std 7-4.3.2-1993, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,"[1] was jointly prepared by the Nuclear Power Engineering Committee of the Institute of Electrical and Electronics Engineers (IEEE) and the Nuclear Power Plant Standards Committee of the American Nuclear Society (ANS). The NRC staff has worked with IEEE and ANS in developing IEEE Std 7-4.3.2-1993 to ensure that the guidance provided by the consensus standard is consistent with the Commission's regulations. IEEE Std 7-4.3.2-1993 has evolved from ANSI/IEEE-ANS-7-4.3.2-1982, "Applications Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations." IEEE Std 7-4.3.2-1993 is a significant improvement over its 1982 version. The 1993 version was approved by the IEEE Standards Board on September 15, 1993. This standard identifies guidelines for digital computers (including hardware, software, firmware, and interfaces) to supplement IEEE Std 603-1991, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations."[1] The NRC staff recognizes that development processes for computer systems continue to evolve.

Digital I&C systems share data transmissions, functions, and process equipment to a greater degree than analog systems. Although this sharing forms the bases for many of the advantages of digital systems, it also raises a key concern with respect to its vulnerability to a different type of failure. The concern is that a design using shared data bases and process equipment has the potential to propagate a common cause failure of redundant equipment. Another concern is that software programming errors can defeat the redundancy achieved by the hardware architectural structure. Because of these concerns, the NRC

[1]IEEE publications may be purchased from the IEEE Service Center, 445 Hoes Lane, Piscataway, NJ 08854.

staff has placed significant emphasis on defense-in-depth against propagation of common cause failures within and between functions.

The principle of defense-in-depth is to provide several levels or echelons of defense to challenges to plant safety, such that failures in equipment and human errors will not result in an undue threat to public safety. A detailed defense-in-depth study and failure mode and effect analysis or an analysis of abnormal conditions or events should be made to address common cause failures. The Commission's position for providing defense against common cause failures in digital I&C systems for future light-water reactors is given in the Staff Requirements Memorandum of July 21, 1993, on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs"[2] (specifically in point 18: II Q, "Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems").

Section 5.15, "Reliability," of IEEE Std 7-4.3.2-1993 states, "When qualitative or quantitative reliability goals are required, the proof of meeting the goals shall include software used with the hardware." The staff does not endorse the concept of quantitative reliability goals as a sole means of meeting the Commission's regulations for reliability of the digital computers used in safety systems. The NRC staff's acceptance of the reliability of the computer system is based on deterministic criteria for both the hardware and software rather than on quantitative reliability goals.

Software failures that are not the consequence of hardware failures are caused by design errors and, therefore, do not follow the random failure behavior used for hardware reliability. The NRC staff believes that quantitative reliability determination, using a combination of analysis, testing, and operating experience, provides information regarding the safety importance of the computer system and also provides an added level of confidence in its reliable performance. If quantitative software reliability goals are used, the staff believes that the amount of testing of the safety system instrumentation and control equipment will increase. The staff recognizes that the commercial dedication of "commercially" available digital

[2]Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

systems in nuclear applications relies a great deal on quantitative methods because of the operating experience data (such as number of hours of successful operation) accumulated over the years. The staff does not intend to preclude operating experience data from the justification of a successful commercial dedication.

Section 6, "Sense and Command Features--Functional and Design Requirements," of IEEE Std 7-4.3.2-1993 indicates that no requirements beyond IEEE Std 603-1991 are necessary. IEEE Std 603-1991 specifies the need to ensure acceptable response time for the instrumentation and control system in order to accomplish necessary safety functions. Consideration of the sampling rate of plant variables is an important aspect of the design of a digital system when satisfying this criterion.

IEEE Std 7-4.3.2-1993 includes 8 annexes. This standard states that these informative annexes are not part of IEEE Std 7-4.3.2-1993. The NRC staff believes these annexes contain information that may be useful. However, the information in these annexes should not be viewed as the only possible solution or method. Since a consensus has not been reached in the nuclear industry, these annexes are not endorsed by the NRC staff.

C. REGULATORY POSITION

Conformance with the requirements of IEEE Std 7-4.3.2-1993, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," with the exception of relying solely on quantitative reliability goals (Section 5.15), is a method acceptable to the NRC staff for satisfying the Commission's regulations with respect to high functional reliability and design quality requirements for computers used as components of a safety system.

Section 2 of IEEE Std 7-4.3.2-1993 references several industry codes and standards. If a referenced standard has been separately incorporated into the Commission's regulations, licensees and applicants must comply with the standard as set forth in the regulation. If the referenced standard has been endorsed by the NRC staff in a regulatory guide, the standard constitutes an acceptable method of meeting a regulatory requirement as described in the regulatory guide. If a referenced standard has been neither incorporated into the Commission's regulations nor endorsed in a regulatory guide, licensees and applicants may consider and use the information in the referenced standard if appropriately justified, consistent with current regulatory practice.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this guide.

Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described in this guide will be used in the evaluation of submittals in connection with applications for construction permits and operating licenses. It will also be used to evaluate submittals from operating reactor licensees that propose system modifications voluntarily initiated by the licensee if there is a clear nexus between the proposed modifications and this guidance.

VALUE/IMPACT STATEMENT

A draft Value/Impact Statement was published with the draft of this guide, Task DG-1039, when it was published for public comment in May 1995. No substantive changes were necessary, but a few editorial changes were made for clarity and consistency. A copy of the revised Value/Impact Statement for Revision 1 of Regulatory Guide 1.152 is available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

Item B.6:

Status Report on Individual Plant Examinations (IPEs)

(Dr. Apostolakis)

ITEM B.6: STATUS REPORT ON INDIVIDUAL PLANT EXAMINATIONS (IPEs)

As a result of a meeting between the ACRS and the Commission on June 8, 1995, a staff requirement memorandum (SRM) was issued on June 16, 1995, in which "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 NRC has begun a transition from prescriptive regulations to a more risk-informed performance approach. The NRC established an implementation plan for probabilistic risk assessment (PRA) to achieve improved regulatory decision making and more efficient use of licensee and NRC resources. The increased use of PRA in the regulatory process is dependent upon the state-of-the-art of PRA methods and the data available to support PRAs.

The ACRS has a long history of advocating the appropriate use of risk-informed performance regulation and rulemaking. The move toward risk-informed performance regulations should reduce the number of deterministic regulatory requirements and allow licensees greater flexibility in plant design and operation. The engineering models, probabilistic risk assessments, and regulatory structure needed to support risk-informed performance regulations need to be developed and reviewed.

On October 26-27, 1995, the ACRS Subcommittees on Probabilistic Risk Assessment and on Individual Plant Examination held a joint meeting to establish a common understanding between the NRC staff, industry, and licensees regarding the concept of risk-informed performance regulation. The technical goals of the meeting were to review and identify the regulatory issues that are amenable to a risk-based approach. These Subcommittees plan to hold another joint meeting in January 1996 to assess the current models and to identify the analytical needs and tools for such a regulatory approach, to review the engineering models, defense-in-depth concept, and the NRC risk-based regulatory framework. In addition, the ACRS will assess whether the IPEs are sufficiently robust to support risk-based regulatory applications.

The Committee will continue its review of issues related to risk-informed performance regulation during future meetings.

Attachments:

- Staff Requirements Memorandum (SRM) dated June 16, 1995, from A. Bates, Acting Secretary, Office of the Secretary, to the File, Subject: Meeting with Advisory Committee on Reactor Safeguards, 10:00 a.m., Thursday, June, 8, 1995, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to the Public) (p. 39)

- Letter dated November 16, 1992, from P. Shewmon, Chairman, ACRS, to I. Selin, Chairman, NRC, Subject: Risk-Based Regulation (pp. 40-41)

- Letter dated March 17, 1995, from T. S. Kress, Chairman, ACRS, to I. Selin, Chairman, NRC, Subject: Proposed Rulemaking – Revision to 10 CFR Parts 2, 50, and 51 Related to Decommissioning of Nuclear Power Reactors (pp. 42-43)

- Letter dated September 15, 1995, from T. S. Kress, Chairman, ACRS, to J. M. Taylor, Executive Director for Operations, Subject: The Nuclear Energy Institute Petition for Rulemaking to Amend 10 CFR 50.48, "Fire Protection" (pp. 44-46)

- Letter dated October 30, 1995, from J. M. Taylor, Executive Director for Operations, to T. S. Kress, Chairman, ACRS, Subject: The Nuclear Energy Institute Petition for Rulemaking to Amend 10 CFR 50.48, "Fire Protection" (p. 47)

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.

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

November 16, 1992

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

Dear Chairman Selin:

SUBJECT: RISK-BASED REGULATION

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

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

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

Far more important to us is the issue of coherence of the various efforts now in progress in various parts of the staff to develop and implement activities that could be collected under the name of risk-based regulation. We have commented earlier about the Maintenance Rule, Regulations Marginal to Safety, and other initiatives involving the use of risk analysis, and have at this meeting heard about Risk-Based Regulation, revision of the Regulatory Analysis Guidelines, and the Prioritization of Generic Safety Issues. Each of these requires informed use of quantitative risk information and appropriate attention to the Commission's safety goals, yet each is being analyzed by an independent group, with an independent perspective on the NRC's needs. In addition to this, there is the PRA Working Group, whose progress we have been following closely. We are unable to find any focal point for all these efforts, except at the level of the EDO.

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

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

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

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

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

March 17, 1995

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Selin:

SUBJECT: PROPOSED RULEMAKING - REVISION TO 10 CFR PARTS 2, 50, AND 51 RELATED TO DECOMMISSIONING OF NUCLEAR POWER REACTORS

During the 419th meeting of the Advisory Committee on Reactor Safeguards, March 9-10, 1995, we reviewed the proposed rule on decommissioning of nuclear power reactors. During our review, we had discussions with representatives of the NRC staff and the Nuclear Energy Institute. We had the benefit of the document referenced.

The proposed revision to the decommissioning rule appears to allow significant flexibility for different possible circumstances under which a nuclear plant may cease operation and transition into the decommissioning mode. The proposed revision to the rule reduces unnecessary burdens on both the licensees and NRC staff.

We believe that the proposed rule should be issued for public comment. We are concerned, however, that the proposed rule has not been founded on a risk basis. Realistic risk analyses for decommissioning nuclear power reactors have not been done. Consequently, there is no clear relationship between the requirements being retained in the revised rule and the realistic risks to the public health and safety and the environment posed by decommissioning. The revised rule may still impose unnecessary burdens on licensees and may make excessive demands on NRC resources. We hope that steps can be taken in the near future to establish a risk basis for reformulating 10 CFR Parts 2, 50, and 51. We believe this is an issue on which comment from the industry and the public should be sought.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Reference:
Memorandum dated January 27, 1995, from Bill Morris, Director, Division of Regulatory Applications, RES, to John Larkins, Executive Director ACRS, forwarding Proposed Rule to Amend 10 CFR Parts 2, 50, and 51

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

September 15, 1995

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Taylor:

SUBJECT: THE NUCLEAR ENERGY INSTITUTE PETITION FOR RULEMAKING TO AMEND 10 CFR 50.48, "FIRE PROTECTION"

During the 424th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 1995, we completed our discussion regarding the subject rulemaking petition. Our Auxiliary and Secondary Systems Subcommittee met on June 7, 1995, to begin the review of this matter. During these meetings, we had the benefit of discussions with representatives of the staff, the Nuclear Energy Institute (NEI), and the Electric Power Research Institute (EPRI). We also had the benefit of the documents referenced.

The NEI petition for rulemaking proposes to amend 10 CFR 50.48, "Fire Protection," by adding an Appendix S, which is described as a "performance-based" alternative to the existing prescriptive Appendix R. NEI believes that the recommended addition to 10 CFR 50.48 will be "safety neutral" and that considerable cost savings will result.

We support risk-based regulations. It is not clear, however, how performance-based regulations should be developed from risk consideration. It is our perception that such regulations should include the following elements:

- Clearly stated objectives with demonstrable performance requirements, expressed either in deterministic or probabilistic terms.
- Flexibility in the methods that the licensee is permitted to use to meet the performance goals or criteria. These methods should be supported by operational experience and experimental results.

- The regulatory body must have a valid means to establish that the performance criteria have been met.

Unfortunately, the proposed rule in the NEI petition is deficient in all these elements.

The objective of the proposed rule is to assure "that the safety functions required to safely shut a plant down and maintain it in a safe condition are maintained during and following a fire." It is further stated that fire modeling, as well as PRAs, may be used to identify the pertinent performance criteria. The proposed rule, however, avoids setting probabilistic requirements and uses non-quantitative language. Thus, there are references to "credible" fires and "credible" scenarios, as well as to "adequate" time for completing safety functions. These concepts need to be defined in quantitative, probabilistic terms. For example, we would expect a quantitative performance requirement for the probability that fire will compromise safe shutdown equipment and lead to core damage.

Some of the issues that the proposed rule raises could be naturally resolved in a PRA context. Examples are the inadvertent actuation of automatic suppression systems and the relevance of the current requirements regarding the concurrent occurrence of a fire and loss of offsite power. In addition, the proposed rule does not address the issue of transient fuels. PRAs have shown that, in some cases, transient fuels are required to produce fires of severity sufficient to damage redundant safety systems. Such transient fuels have been found in controlled areas in the past. Not only are transient fuels not addressed, the proposed rule suggests that some administrative controls dictated by Appendix R may be eliminated. We would prefer to see an evaluation of such issues in the context of a fire PRA.

We are concerned that neither the NRC nor NEI has any plans for conducting fire tests for refining the probabilistic analysis of time-to-suppression. We also have concerns about weakening the requirement for automatic fire detection systems, the lack of a methodology for treating the potentially damaging effects of smoke, the use of a limited fire initiation database, and the neglect of consideration of fire during shutdown. We will address these concerns should the rulemaking process advance.

Even though we support the use of PRA in the development of a performance-based rule, we note that, given the uncertainties in the state of the art, fire PRAs cannot be the sole basis for regulatory requirements. Developing the right mix of criteria based on PRA and criteria based on good engineering practice is a challenge and a necessary requirement for a well-written rule.

We believe it will take some time and resources to develop and institute performance-based fire regulation. We also believe doing so is an important step in the agency's move in this direction.

Additional comments by ACRS Members George Apostolakis, James C. Carroll, and Ivan Catton are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Additional Comments by ACRS Members George Apostolakis, James C. Carroll, and Ivan Catton

We support the Committee letter but have further comments for your consideration. The use of performance-based rules for fire protection is frustrated by conventional attitudes. The desire of regulators to have simple rules and tests for administrative convenience contrasts with the need of plant operators to have flexibility to arrive at optimal solutions. Unfortunately, the prescriptive characteristics embodied in regulations are accepted without proof, while any engineering solution supporting a performance requirement is subjected to a disproportionately higher standard of proof.

References:
1. Letter dated February 2, 1995, from W. Rasin, Nuclear Energy Institute, to John C. Hoyle, Acting Secretary, NRC, Subject: Petition for Rulemaking to Amend 10 CFR 50.48
2. SECY-94-090 dated March 31, 1994, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Institutionalization of Continuing Program for Regulatory Improvement
3. SECY-95-034 dated February 13, 1995, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Status of Recommendations Resulting from the Reassessment of the NRC Fire Protection Program
4. Memorandum dated December 30, 1994, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Eighth Quarterly Report on the Status of the Thermo-Lag Action Plan

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

October 30, 1995

Dr. Thomas S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: THE NUCLEAR ENERGY INSTITUTE PETITION FOR RULEMAKING TO AMEND 10 CFR 50.48, "FIRE PROTECTION"

Dear Dr. Kress:

The staff has received your letter, dated September 15, 1995, and appreciates your comments on the petition and generally on performance-based fire protection regulation.

As indicated in SECY-94-090, the staff intends to revise NRC's fire protection regulations to make them more performance-based and risk informed. As you indicated, we also recognize the difficulty of developing performance-based regulation from risk considerations in the area of fire protection. We agree with the Committee that developing the right mix of criteria based on PRA and good engineering practice is a challenge.

In the near term, the staff plans to review public comments on NEI's petition, consider your comments, and develop a plan of action for the rulemaking. We anticipate that additional research may be necessary to support certain options. We will forward this rulemaking plan to you for your information and will be prepared to brief the Committee, if it so desires.

In November 1995, the staff also plans to conduct interactions with the public and industry through Rulenet (electronic "forum" through the information highway) on the petition for rulemaking and other possible approaches for rulemaking in this area. We will inform you of the details of this "forum."

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: Chairman Jackson
Commissioner Rogers
SECY
OGC