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PUBLIC MEETING

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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MEETING WITH ADVISORY COMMITTEE

ON REACTOR SAFEGUARDS (ACRS)

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PUBLIC MEETING

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Nuclear Regulatory Commission

One White Flint North, Room 1F-16

11555 Rockville Pike

Rockville, Maryland

Thursday, April 2, 1998

The Commission met in open session, pursuant to notice, at 1:08 p.m., the Honorable SHIRLEY A. JACKSON, Chairman of the Commission, presiding.

COMMISSIONERS PRESENT:

SHIRLEY A. JACKSON, Chairman of the Commission

NILS J. DIAZ, Member of the Commission

EDWARD McGAFFIGAN, JR., Member of the Commission

GRETA J. DICUS, Member of the Commission

STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

DR. ROBERT L. SEALE, Chairman, ACRS

DR. DANA POWERS, Vice-Chairman, ACRS

DR. GRAHAM B. WALLIS, Member, ACRS

DR. GEORGE APOSTOLAKIS, Member, ACRS

MR. JOHN BARTON, Member, ACRS

DR. THOMAS S. KRESS, Member, ACRS

DR. MARIO H. FONTANA, Member, ACRS

P R O C E E D I N G S

[1:08 p.m.]

CHAIRMAN JACKSON: Good afternoon, ladies and gentlemen. It's a pleasure to meet again with Dr. Seale and the members of the NRC Advisory Committee on Reactor Safeguards, who plan to discuss a number of topics of interest to the Commission at today's session.

But first, I would like to welcome, if he is here, Dr. Graham B. Wallis to the Commission's Advisory Committee on Reactor Safeguards. We're pleased to have you on board.

The Commission is fortunate to be able to draw upon views and experiences of this selected group of experts as we try to solve and address various technical concerns in licensing and regulation.

During today's briefing, the Commission -- I'm sorry -- the Committee will discuss the following topics.

First, improvements to the Senior Management Meeting process; next proposed revisions to 10 CFR 50.59 and related issues; third, risk-informed and performance-based regulation, including the use of PRA in the regulatory decision-making process; fourth, status of the AP600 review; fifth, shut-down and low-power operations; sixth, NRC safety research programs; seventh, license renewal; and eighth, fire protection rule-making.

Commissioner McGaffigan has already made note of

the fact that our two Commission meetings this afternoon have been scheduled for three hours but probably each involve about five hours.

So, Dr. Seale, my colleagues and I welcome you to this meeting and anticipate another candid and informative session with the Committee, and I understand that copies of the briefing material are available at the entrances to the room.

Unless anyone has any opening comments, I think we had better proceed.

DR. SEALE: Very good.

Well, good afternoon, Chairman Jackson, Commissioner Dicus, Commissioner Diaz, and Commissioner McGaffigan.

As always, the ACRS is pleased to have the opportunity to meet with the Commission and exchange information and for us to provide our views on items of interest to you.

We have a very ambitious agenda today and would not be offended if most or all of the discussion time were consumed in the first four items or so, because --

CHAIRMAN JACKSON: It may come to that.

DR. SEALE: It may come to that. And as the last four items are all work in progress and the view-graphs summarize these items fairly succinctly, I don't think

there's a lot of pressure to necessarily pound the program into the time.

Occasionally -- or additionally, I'd like to mention that we have submitted copies of the ACRS operating plan, and this contains planned activities, priorities, and metrics for assessing ACRS performance. Any comments you may have on that plan we would very much appreciate. We expect to update it quarterly -- that is, July being our first update.

I think we'll get right into the program, and John Barton, Plant Operations Subcommittee Chairman, will begin with a discussion of the ACRS deliberations on the Senior Management Meeting process.

John?

MR. BARTON: Thank you, Dr. Seale.

ACRS has been actively involved in the review of the proposed improvements to the SMM process. In March 1997, the Committee reviewed the prepared Arthur Anderson report and, since then, has had several meetings with the staff and prepared two reports to the Commission.

In the September report to the Commission -- some highlights of that report,

The Committee supported the goal of codifying the SMM information-gathering and review process. However, the basis for the top-level criteria contained in the template

was not clear to the Committee.

Furthermore, the process by which the template led to formation -- formulation of decisions also was not apparent to the Committee.

The Committee preferred to see a top-down structure that starts with a point of decision, identified the objectives of the decision, and then proceeded to define the informational needs to support the decisions.

In a memorandum subsequent to the Committee report -- it was a memo from the ACRS Executive Director -- forwarded comments from an ACRS member, Dr. Apostolakis, which laid out for the staff an approach to the top-down decision-making approach.

Also, another item in the September report, we talked about the assurance of the needs of the new performance standards to be objective and reduce reliance on event-driven assessments, and we made the point that, although progress had been made improving information basis of the senior management process, considerable work remained in areas such as developing tools for assessing management and organizational effectiveness and testing their implementation before being included in the SMM process.

Also, in our September report, with regards to staff's integrated review of the assessment process, we noted the staff had not defined requirements, preferably

quantitative requirements, for an adequate program to assess license performance.

It was not apparent to the Committee at that time how well-designed recommendations could be formulated without explicit definition of the requirements for an assessment program that met the agency's needs.

It was also not clear how preferred opinions -- options could be selected absent these requirements, and we recommend the NRC staff develop these requirements for an adequate licensing performance assessment program.

Subsequent to that report, we had additional meetings with the staff and issued a second report on the subject in March of this year, and in that report, we reviewed the draft Commission paper.

We looked at the overall objectives. We felt that they were not sufficiently specific to allow evaluation of the proposed assessment process. We recommended at that time the development of specific objectives and performance measures that could be applied directly to the process.

The assessment decision model, logic model, we felt should show how the selected decision options noted in the draft paper would utilize the performance measures.

CHAIRMAN JACKSON: Dr. Barton, I think Commissioner Dicus has a question.

COMMISSIONER DICUS: Yes. About the objectives

and the performance measures, could you be a little more specific on what sort of measures you think would be useful to provide the clarity?

MR. BARTON: George?

Dr. Apostolakis led this thought, and I'd like him to expand on that.

DR. APOSTOLAKIS: Well, the overall objective of a process like the Senior Management Meeting is usually something that is general, noble, but not operational. So, as I recall, it says something to the effect that we want to make sure that the plants are safe.

Now, that doesn't mean anything. You have to tell me what safe means.

For example, if you want this to be risk-informed, would you like to prevent the occurrence of initiating events?

Now, that's something specific, that's something I understand, and that certain contributes to safety.

Would you like to make sure that the safety functions have a certain reliability? Again, that's operational.

Now, operational -- well, maybe that's an exaggeration, but -- so, the second level, the second tier would be objectives of this type that elaborate on the top level.

Then you might ask yourself, well, what does it mean to assure the safety function, reliability? You go down one further level.

Now you become more specific. Maybe you will say I don't want such-and-such an event to happen, and you may have to go down two or three or four levels until you reach a point where you say, well, now, this I can measure, this I can track, and then you have this hierarchy construction that shows the rest of us why you selected certain things to monitor and why you left certain other things out.

Right now, we have the top objective, and then we jump way down to the six categories, what is called a template, and the connection is not clear. I mean it's not that there is no logic. I'm sure there is some logic someplace, but it's not evident from reading the document why, for example, I have to worry about human error, I mean besides the general feeling that human error is important.

So, that was really the idea of requiring that.

COMMISSIONER DICUS: Okay. Thank you.

MR. BARTON: Also in our March report we made the comment and recommendation regarding that the staff should work through at least one example that uses the actual inspection reports and demonstrate the implementation of the new assessment decision logic.

We wanted to be sure that the new engineered

approach, taken to an actual case and worked through, would lead you to the same decision that was arrived without this approach. It was kind of a test of the new approach.

CHAIRMAN JACKSON: Now, my understanding is that there has been some piloting of the process since the time you had the discussions with the staff. Do you have any updated commentary?

MR. BARTON: No, we do not, not at this time. We know that they were going to try that process, but we haven't had feedback as to how well that process worked.

We also recommended at that time that the categories in the proposed templates -- the six categories of the template be evaluated and see if they were at the appropriate level and whether there was any unnecessary overlap.

We recommended the assessment process contain provisions to ensure consistent results are obtained among the regions. The new process really drives back to the regions most of the work; decision-making is done at the region level.

We wanted to assure that there would be consistency, that in the new process would be enough built into it that we could assure consistency among the regions without having to rely on headquarters people down at the regions looking for the consistency.

The process itself should have built in reasonable assurance of consistency among the regions. That was a concern we had, and we didn't see how -- weren't sure how that was in the model.

CHAIRMAN JACKSON: Commissioner Dicus.

COMMISSIONER DICUS: I want to dwell a little bit just briefly on the consistency issue, because I think it is a problem.

Are you talking about consistency of implementation of the process, or is there a greater problem or another problem with regard to the consistency of plant performance from a regional or a national basis?

MR. BARTON: We were concerned with consistency in the process. You know, no process is perfect. That's probably the reason we're changing the current process, to improve it, make it more scrutable, more objective.

We wanted to ensure that, in designing that new process, that the same performance indicators that you were measuring in one region, you measured in another region and gave you the same result. That's what we were looking at.

We also made a recommendation that the measured -- plant performance be measured at a more global level.

We had some discussions with industry at one of the Committee meetings, and we felt that the input to the new process that the staff was proposing was set a real low

compliance enforcement level and saw some opportunity in what the industry was proposing as performance indicators that maybe the staff and industry might get together and raise the performance indicators and the input into the process.

CHAIRMAN JACKSON: Have you had discussions among yourselves about the connectivity between the suggested performance indicators from the industry to the kinds of issues that Dr. Apostolakis raised?

I mean is there a migratory path? Are those -- have you looked at whether those would be the appropriate performance indicators to achieve what he wants? Have you agreed as a committee that you agree with what was in its memo?

MR. BARTON: We have discussed this amongst ourselves, and I think there is an agreement that -- based on what Dr. Apostolakis mentioned before and where the industry was coming from, I believe there's agreement in the Committee -- if I'm not right in that, please, any member speak up -- that there should be more attention paid at the higher level.

Anybody want to comment on it?

DR. APOSTOLAKIS: I don't believe, Chairman Jackson, that, as a Committee, we looked at that specific aspect of the NEI presentation, but that should be

relatively easy to check, because it's higher level, higher level requirements.

But that is an issue that will keep coming back. Where do you set the performance measures? We had a presentation this morning on the new performance-based initiative. Where do you do that? Do you use risk information? Do you use something else?

Because ideally -- not ideally -- you would like them to be as high as possible where the highest level is, of course, the QHOs. Practically, you can't do that.

So, where is the optimum so that we will satisfy that third feature, I believe, of performance-based regulation, namely giving flexibility to the licensees. The lower you go, the less flexibility they will have.

CHAIRMAN JACKSON: Right. But I'm really actually turning back on you something that you said the staff needs to ensure, and that has to do with consistency.

If you're going to talk, on the one hand, about the need to agree on performance indicators starting with some that may have already been developed by the industry or somewhere else and if you're going to make that recommendation, then there has to be a connectivity between that recommendation at whatever level these performance indicators would come in, with a judgement as to (a) is that the right level, (b) if it is, you know, what the connection

has to be to what the staff would, quote/unquote, actually measure or look at, because in the end, it doesn't do -- you have recommendations that go like this or like this, but at any rate, you have to ensure that, if you're going to make the recommendations on the one hand, in one area, that they are consistent with the recommendations you make in the other.

DR. SEALE: If I may make a comment, it strikes me that, realistically, what you have to do is to erect this connective tissue between -- or lines between the low level and the high level indicators, and once you've done that, then the kind of gradation that occurs is deciding how you tune to get blips on your radar screen.

One of the things you have to have is a scheme or a system that gives you data that tells you what's going on in the plant.

CHAIRMAN JACKSON: Well, I think that's where we all want to get, obviously, and the issue is that, if you're starting at the -- if you want a hierarchical scheme, right, you have to have the connectivity all the way down.

However, what I'm saying is something slightly different. I'm saying that, if you're talking about imposing a set of performance indicators, that you've got have a fundamental decision made as to whether they are the right performance indicators for regulatory agency.

DR. SEALE: Yes.

CHAIRMAN JACKSON: Okay. And then you're dealing with in the context of this hierarchical or connected approach.

Yes, Commissioner Diaz.

COMMISSIONER DIAZ: If I might build up on that, I think, essentially, what we should be asking, also, is is there a process of convergence between the different opinions and if that convergence is naturally happening or does it need to be a function, you know, that will make it happen?

DR. POWERS: It strikes me that you need to be careful not to misinterpret what the Committee was saying when it made its recommendations.

It was saying that we feel there should be a hierarchical structure, and in that hierarchical structure, you will arrive at high-level performance indicators, higher level than perhaps what the staff is proposing, like what the industry was saying.

We did not espouse the industry's indicators per se but, rather, suggested that, when they created this structure, they would encounter these higher level and those might be better to use than the lower-level indicators.

I don't think the Committee was saying adopt these that the industry has proposed.

DR. SEALE: No, we did not come to that conclusion.

DR. POWERS: Rather, these industry proposed indicators looked to be higher and you will arrive at them in the course of your hierarchy.

CHAIRMAN JACKSON: But I would also argue that, from an implementation point of view, you've got to ask who uses what when, and I assume this is implicit in what Dr. Apostolakis is talking about, because you can talk about having your higher-level indicators, but the issue is who's making use of them and to what end?

Are they being used as a consistency check? Are they being used in decision-making? Are they best used at a very high senior management level? That may be different than what the guy does in the field, and so, we have to be very clear in that.

DR. POWERS: In a moment or two, Mr. Barton, we'll get to the issue of requirements -- agency requirements for the assessment process, and that will come up in spades.

CHAIRMAN JACKSON: All right. Well, then I better let Mr. Barton proceed, then.

DR. APOSTOLAKIS: A forcing function in the form of a delta function will be very welcome, by the way.

CHAIRMAN JACKSON: A delta function -- a forcing function to you or a forcing function to the staff? Let's

be clear on who we're forcing to do what.

COMMISSIONER DIAZ: If I may amend the record, the Chairman who uses what when, also for what, and that goes back to your performance measures.

DR. KRESS: That would call for different sets of performance measures, one for the inspector and another one for the senior management and even a different one for you guys.

CHAIRMAN JACKSON: Let Mr. Barton continue.

MR. BARTON: Dana, in our September report -- I mentioned earlier -- this was a comment that we had made. We had noted that we had not -- staff had not yet defined the requirements for the program to assess licensee performance. Would you like to expand on that? It was also in our September report.

DR. POWERS: Staff is now attempting to develop an integrated assessment program, and what we saw was what I would characterize as an assumed solution to that assessment, to integrate together assessment that currently takes place in three different areas into a single assessment.

I call it assumed, because there did not appear to us to have been an attempt to define what the agency needs for its own purposes as an assessment of plant performance, what are the requirements that you had.

Then, once you had those requirements, one could presumably define a number of strategies for obtaining those assessments and compare them on the basis of some ranking system, some preferred alternatives, preferences that you had, how you would compare various strategies, all of which met the requirements the agency had but some of which may be preferred because they're less costly, less manpower-intensive, more transparent to the public.

We had not seen that kind of structure in developing this integrated assessment and found it very difficult, then, to look at this integrated assessment and say does it, in fact, meet all the agency needs, as you said, from the front line inspector, the eyes and the ears of the agencies at the plant itself, to the top level sitting at this table.

You need to have an assessment that meets all those needs. It's difficult to judge if we don't know what all those needs are.

COMMISSIONER McGAFFIGAN: My concern comes at it from a slightly different direction.

The staff is going to talk to us -- I don't want to spend a lot of time on this, but they're going to talk to us in an hour-and-a-half about this stuff, and they have a slide of boundary conditions, which boundary conditions are sort of like requirements, and I'm not sure I agree with all

of them. I probably don't. And I've heard additional requirements coming from you all this morning that aren't among their boundary conditions, that this should be risk-informed. That's not something that they're aspiring to at the moment. They do aspire to line up better with enforcement, which I'm hearing some criticism of and I have concerns.

But I think there's a real danger in over-constraining this problem so that there is zero solutions. In fact, it may already been well past that point, and when you try to design a single process to meet, you know, a multiplicity of requirements and the requirements keep growing, you know, if we aren't at the point where there's zero solution, we'll certainly get there rapidly.

DR. POWERS: The one thing you have to have in any kind of design-making is to have an agreed-upon set of requirements, and I forgot to say agreed.

COMMISSIONER McGAFFIGAN: Agreed-upon, right.

DR. POWERS: That's an essential step, and it is not beyond the bounds of credulity to say that I can create enough requirements that there is no solution, and then you have to have an agreement upon reduction in those requirements.

I think it is better to do that, to follow that

tact, than to have a set of requirements created after you have assumed the solution, and I think that's all we were trying to communicate.

COMMISSIONER McGAFFIGAN: You think the staff has this rock, as some people call it, and has the following characteristics which they then say are the boundary conditions for the rock.

DR. POWERS: I think there is a strong component of that. I think that they, indeed, did see criticism of having three or four, depending on how you count them, different approaches to doing plant assessments, and they said my requirement for this is to have one, and they took that.

CHAIRMAN JACKSON: I'm not here to be the defender of the staff, but in fact, I think we all have to take ownership, because I think, in fact, the staff was trying to be responsive to what it thought it was hearing from the Commission.

DR. POWERS: I have no doubt.

CHAIRMAN JACKSON: So, that defined at least part of the rock.

DR. POWERS: I have no doubt that's true. You have an excellent staff that's very responsive, and in this particular case, you have a particularly ambitious fellow leading this product that's anxious to produce a product

that everybody likes.

I mean he really is trying very hard, and we're simply trying to hone his strategy a little bit here in our comments.

COMMISSIONER McGAFFIGAN: He may have produced something that nobody likes.

DR. POWERS: And he won't be the first.

COMMISSIONER McGAFFIGAN: Right.

CHAIRMAN JACKSON: Well, the real question I really have in terms of an over-arching way, since I think this is the last view-graph on this subject -- it's a question but embodied in it is a comment, and that is how much did you treat this as a work in progress and an opportunity to help shape where it's going as opposed to assuming that it is the product that needs to be accepted or rejected?

DR. POWERS: I think we recognized exactly that it was very much a work in progress. That's how it was presented to us, if I can characterize it.

DR. FONTANA: Yes.

MR. BARTON: Yes. And tried to help the staff develop the process as they went along.

COMMISSIONER McGAFFIGAN: Can I ask one other question? One item you slipped over on the previous view-graph was perform additional research prior to use of

economic indicators. I don't know whether that is a kind way of putting this off to the third millennium or later.

Is there any prospect that we're going to be able to come up with something that's useful in economic indicators if we throw research dollars at it, or is that something that we should just --

MR. BARTON: I'm not sure we were talking about throwing a lot of research dollars at it. I think we were coming at it from the perspective of can you really gain -- what can you really gain from some of the economic indicators?

There's changes in how plants spend money that go on for years before you see some performance changes.

So, I think what we're really saying is be careful how you use economic indicators. It may be a data point, but we're not sure at this point that it should be a decision point. I think that's where we are on the economic indicators.

DR. SEALE: But it's certainly an input to the product, and so, you should keep track of the economic activity supporting the plant.

CHAIRMAN JACKSON: That's interesting. I mean the comments that the two of you have made actually have raised a point of another clarification that perhaps needs to be in the process and that is making distinctions between what is

input and knowing how that input is to be used versus the decision point.

DR. SEALE: Yes.

CHAIRMAN JACKSON: Okay.

DR. SEALE: Well, you're back in the barrel again, John, along with Tom on proposed revision to 50.59.

CHAIRMAN JACKSON: Proposed.

DR. SEALE: Proposed. We try to be careful with some of these words.

CHAIRMAN JACKSON: Okay.

MR. BARTON: Again, just some background.

We provided the reports in April, October, and December on the proposed 50.59 process change.

The first slide, which is the April report -- I won't go a lot into that. That's kind of -- it's history. We proposed something and it went out for public comment.

So, skipping ahead till our October report, we proposed that the NRC should issue revision 1 to Generic Letter 91-18. We felt that it did clarify the applicability of 50.59 evaluations to address the degraded and non-conforming conditions. Also, it addressed completeness and some inconsistency.

Also in that report, we recommended that there be work continued to continue to develop the plan for a 50.59 process that's consistent with the risk-informed

performance-based regulation.

This is where Dr. Kress was driving the Committee to focus on the risk-informed piece of the regulation.

Tom, would you like to expand on that?

DR. KRESS: Certainly.

I guess it would be easier to tell you what we didn't mean by that bullet rather than what we did mean.

We did not mean that the 50.59 process ought to be done by means of a PRA looking at delta-CDF and delta-LERF like the Reg. Guide 1.174, and in fact, we don't think that's even possible.

The consistency part meant that any changes that are proposed that have a direction of risk increase, even though it's small or minimal, should not be inconsistent with the values that are in here. They should be very small.

CHAIRMAN JACKSON: But to the extent that there could be a direct comparison or could be cast --

DR. KRESS: If they could be.

Now, the other part of this is we take those levels of risk change or outside the purview of PRA, that PRA is just not good enough to quantify at those levels, so that the challenge is going to be, for the staff, to quantify both this word "minimal" or "small," as well as to develop ways at which one could -- criteria or attributes

that one could use for a licensee to be guided on what qualifies for that kind of change.

Now, that's going to be a real challenge, and my personal view is that you don't set up a set of criteria that says, if the change meets these criteria, that it qualifies. I think that's almost an infinite set.

I think what you do is set up criteria that, if the change meets these things, then it does not qualify, and clearly, one of these would be, if it's a decrease in risk, it automatically qualifies.

But some of the other things for increases in risk are going to be much more difficult to come by, and they are performance in nature because we have already said you can't quantify them with a PRA, so you have to use intuition, judgement, and I think there would be things like do they impact defense-in-depth, is the change on some system or component that's safety-important or safety-related.

I don't claim to know what these rules ought to be, but I think that's where the challenge lies, and that's, I think, how you make it risk-informed and consistent. That was the intent of that bullet.

CHAIRMAN JACKSON: Can you look at it in terms of how it might affect design basis or FSAR accident frequency?

DR. KRESS: Yes, I think that would be one of the criteria, if it affects the design basis.

Another one would be, if you can -- if it's obvious that you can use a PRA to quantify the change in risk, then I don't think it's 50.59. I think that automatically puts it in 1.174.

COMMISSIONER McGAFFIGAN: Can I ask a question?

CHAIRMAN JACKSON: Please.

COMMISSIONER McGAFFIGAN: The staff has shown you a view-graph that isn't quite the one that's in Reg. Guide 1.174 at the moment where 10 to the minus 7 core damage frequency is described as negligible in terms of risk-informed regulations, and presumably, things are going to get handled very rapidly if somebody can convince the staff that they're in that range, and there was at one point a claim that 10 to the minus 7 was the limit of resolution of PRA technology, and then that was clarified to say no, there are lower levels of resolution that you all can deal with, as low as 10 to the 10th, 10 minus 10, 10 minus 12.

DR. KRESS: I think the Committee disagrees.

COMMISSIONER McGAFFIGAN: Disagrees with that. Okay.

That gets us maybe back to where we originally were. If it's 10 to the minus 7 or below in core damage frequency, is that a -- I know you're going to talk about severe accident space versus design basis accident space, but if it's that level, should it be a 50.59 issue or should

it be an issue that comes to the Commission staff for review and approval?

DR. KRESS: I think the feeling of the Committee was we're not quite certain yet what that level ought to be, because we're talking about cumulative risk over -- there may be hundreds or even thousands at a given plant.

COMMISSIONER McGAFFIGAN: Right.

DR. KRESS: So, we're not quite sure that 10 to the minus 7 is the correct level, but assuming there is some level down there that's about there or even lower, we just do not think that there is a good way to quantify that, and you'll have to come up with a set of rules that you feel qualifies a change to be in that level even though you can't quantify it, and that's going to be a real challenge. That's where we think the challenge is going to be.

CHAIRMAN JACKSON: But don't you think a point that one has to keep in mind -- and that's the difference between the intended use of the reg guide and the Standard Review Plan, is that, in fact, the kinds of changes -- let's leave aside the issue of whether you can put the kinds of changes to the plant that would occur under 50.59 into this space, but those levels are determined within a context that, by definition, the staff is going to be reviewing those, whereas 50.59 is meant to be a screening rule that relates to screening in terms of things that can happen

without coming to the staff, coming to NRC, so that one has to keep in mind, if you're talking about numbers, that the one has a set of numbers that's being used together with other things but being used in the context of changes to the licensing basis that, by definition, are being reviewed by the staff.

The other is a screening set of criteria, and that's a very different kind of thing.

DR. KRESS: Yes, I think that captures the essence of it.

CHAIRMAN JACKSON: Let me just follow on for a minute. If one wanted to do to risk-informed -- and I think that's what you're really talking about, as opposed to having performance-based per se approaches -- is it possible to do something within design basis accident space, where one can talk about a comparable kind of thing, like design basis accident, frequency of probability in a quantifiable way.

DR. KRESS: We have not discussed that, but I personally don't think so. In fact, I don't think there is a good connection now between risk and design basis space. There is a connection. I don't think we have it well quantified or well thought out.

CHAIRMAN JACKSON: Well, let's talk about it for a second, because I'm trying to understand something. Isn't

what you would call a design basis accident something that, at least for certain things, really what would be an initiator in a PRA calculation?

DR. KRESS: Yes. It's generally an initiator, and then there's stylized --

CHAIRMAN JACKSON: -- stylized sequences that would lead to having you determine whether Part 100 limits would be exceeded, right?

DR. KRESS: Yes.

CHAIRMAN JACKSON: So, is there a possibility of starting with a design basis accident, as laid out within --

DR. KRESS: Well, certainly, because those were selected --

CHAIRMAN JACKSON: Right. And then taking those and going through -- is it possible to arrive at, going through a sequence of things that could lead you to exceed Part 100, if you then were able to assign the same kinds of probabilities --

DR. KRESS: You certainly could do it that way --

CHAIRMAN JACKSON: -- and then arrive at some probability of exceeding Part 100?

DR. KRESS: I think you could certainly do it that way. I would not recommend that.

CHAIRMAN JACKSON: Okay.

DR. KRESS: Because I don't think that's true in

risk-informed.

CHAIRMAN JACKSON: Purity of risk-informed means tied to severe accident analyses, but risk-informed, in many people's mind, has come to mean tied to severe accident consequences.

One could argue that you could have a risk-informed process that examines the probabilities of some other consequence, of coming to some other consequence.

DR. KRESS: Oh, certainly.

CHAIRMAN JACKSON: And in that sense, I disagree with your statement that you can risk-inform an analysis to a different consequence.

DR. FONTANA: I can understand what you're saying.

I think, in the best of all worlds, there would be a seamless spectrum from a severe accident all the way down to --

CHAIRMAN JACKSON: Absolutely.

DR. FONTANA: -- and design basis would be a set in those accidents. So, one ought to be able to do a risk analysis with the lowest spectrum of accidents.

CHAIRMAN JACKSON: Right.

DR. FONTANA: We're not there yet.

CHAIRMAN JACKSON: Well, all I'm saying is that my understanding is that, essentially, what you would call a design basis accident, in many ways, is an initiator when

you do your typical PRA calculation, and so, you could have a way, it strikes me, if it is an initiator, to put it into the kind of methodology that 1.174 envisions, and you come out with an answer, which in that case would be expressed in terms of something like a core damage frequency or large early release frequency, and that's one part of a screen if there were some level set.

DR. KRESS: You could certainly put that on the initiating frequency itself.

CHAIRMAN JACKSON: Right, exactly.

DR. KRESS: But once again, you're going to have a great deal of difficult quantifying these types of changes that will propagate through and end up at 10 to the minus 8-like levels.

CHAIRMAN JACKSON: All I'm trying to say is --

DR. KRESS: There certainly would be a way to do it.

CHAIRMAN JACKSON: There are two pieces, because I said that's one part of a screen. Okay? The other part of a screen may be one that's rooted in, you know, the defense-in-depth concepts, etcetera.

DR. SEALE: Yes.

DR. KRESS: Yes.

CHAIRMAN JACKSON: And so, since we're talking screens, we're talking gates.

DR. KRESS: yes.

CHAIRMAN JACKSON: Okay. And so, maybe you have an "and" gate that you have "and, and," that you have a screen or a gate that's related to your defense-in-depth pieces but you also do a consistency check.

DR. KRESS: That is, in fact, what I meant by these sets of rules.

CHAIRMAN JACKSON: Right.

DR. KRESS: They would be that sort of "and" gate.

CHAIRMAN JACKSON: Commissioner McGaffigan.

COMMISSIONER McGAFFIGAN: My understanding was -- and you can correct me, because I haven't looked at the documents, but I thought the staff, in the follow-on reg guides for in-service testing, in-service inspection, etcetera -- that they were struggling with exactly these issues, because some of -- they're going to be looking at license amendments in the context of design basis evaluations and yet have to make risk-informed judgements.

So, I hope they're ahead of us in this discussion, but you all probably have looked at these later reg guides, and how are they doing in the more issue-specific reg guides in trying to make this translation from severe accident space to design basis accident space and back?

DR. SEALE: Of course, they're change tech specs, so there's no doubt they have to go through a 1.174.

CHAIRMAN JACKSON: I think his point --

DR. SEALE: I agree.

CHAIRMAN JACKSON: -- that you're talking about changes the things that fall within design basis.

DR. SEALE: That's an interesting template, if you will, or connection.

COMMISSIONER McGAFFIGAN: We just know they're struggling. I don't know whether they're succeeding, but I know that they're working on it.

CHAIRMAN JACKSON: Is there struggling, Gary?

MR. HOLOHAN: Gary Holohan, Staff.

I'd like to think the staff is succeeding.

CHAIRMAN JACKSON: Thank you so much.

All right. Let's go on.

MR. BARTON: The other recommendations in our December report have been overcome by events. You've issued directions to the staff, and essentially we agree.

CHAIRMAN JACKSON: Well, in fact, I think the direction agrees with -- I mean it resolves essentially all of the kinds of issues --

MR. BARTON: Yes, it does.

CHAIRMAN JACKSON: -- that you had raised.

MR. BARTON: Yes.

DR. SEALE: Okay. Are we through with that one now?

MR. BARTON: Yes.

DR. SEALE: Okay. Fine.

The next one is on risk-informed performance-based regulation, including use of PRA in the regulatory decision-making process, and if this sounds like deja vu all over again, it's because it is.

George?

DR. APOSTOLAKIS: Thank you, Bob.

The first slide is just some of the activities of the Committee the last several months, so we can skip that.

The next one, on ISI, we are, in fact, meeting with the staff tomorrow morning to discuss the new version of the guide, so I don't have anything to say right now. What we said last July still stands, but I think, in the next few weeks, you will see a letter from us on this guide.

The next one is the major recommendations that the Committee made on Regulatory Guide 1.174 and associated Standard Review Plan. Obviously, we agree with what the staff did there. We think they are succeeding. There's no reason to read what's here.

We have a figure later which will give me an opportunity to talk about some of these things.

Now, the other guides on IST, GQA, and technical specifications -- we also recommended that they be approved.

We were not too excited by the GQA guide, 1.176,

as you probably have guessed already from the letter.

We felt that this version of a guide was a significant improvement over the first one that we had seen, which I believe we had called timid, but still, it doesn't go far enough, even if one accepts the fact, which is true, that the lack of a model for assessing the quantitative impact of QA requirements is really a major problem here.

CHAIRMAN JACKSON: Does that imply that you think we have a difficulty or no way of assessing the benefits of our QA program, period?

DR. APOSTOLAKIS: I think the benefits of the QA requirements are grossly exaggerated.

CHAIRMAN JACKSON: This is a Committee point of view?

MR. BARTON: There are some members that agree with Dr. Apostolakis.

CHAIRMAN JACKSON: Let's take a poll.

DR. KRESS: I agree.

CHAIRMAN JACKSON: Do you agree?

DR. SEALE: I think so.

CHAIRMAN JACKSON: Do you agree?

DR. POWERS: I think we have to be very careful about saying we have no way of assessing the benefits of our QA program, period. I think we definitely do have ways of assessing the benefits of our QA program. Are the QA

benefits grossly exaggerated? In the minds of whom?

What I think the more pertinent issue here is, do we have a way to quantitatively describe those benefits and to translate them into a reduction in risk? We do not now, and so, when you ask us to do a risk-informed gradation of QA, we quickly get very handicapped.

What we can do is a risk-informed gradation of systems and components and structures in this system, and then we can assert that surely there must be some gradation in the QA associated with them accordingly.

The problem is how do you judge that?

CHAIRMAN JACKSON: So, it has to do with quantitative modeling.

DR. POWERS: It's the quantitative modeling here. I don't think we ought to get into the subjective and sometimes pejorative statements concerning the QA and QC programs that exist.

There's no question that there's a benefit, and there's no question in people's mind that, even without quantification, for those items that deal with very risk-significant systems, I think everyone, licensee and regulator alike, would just as soon err on the conservative side to assure we have QA.

It is in the lower regions that I think that we worry that too much work is expended, too much work and cost

is expended on assuring the QA of particularly procurement on items that are probably adequately reliable off the shelf rather than having a QA back to the mine in which the metal came from.

Our concern as a Committee, a Committee position, has been the first steps here were timid, that it was possible to take bolder steps.

Our view on the current version of this is a bolder step has been taken, and we understand the inhibitions to going yet farther, and that's why we caveat our endorsement of this by suggesting it be revisited both after experience and additional research.

CHAIRMAN JACKSON: Okay.

DR. POWERS: I think there's room for more here.

COMMISSIONER DIAZ: If I try to extrapolate from what you said, will it be fair to say that a graded QA focus resources on a matter that there are safety.

DR. POWERS: That's right.

Now, a licensee might well find it in his own interest to grade his QA on reliability and economic impact and loss of time and things like that, but as a regulatory institution, we would want to focus on safety.

CHAIRMAN JACKSON: But nonetheless, you're saying that, in the graded QA area, that the reg guides and the associated SRP sections ought to be issued for use because

you think that out of that will come --

DR. POWERS: We think that experience and comfort -- and in fact, if one looks at this whole business of the quantification of risk since 1974 -- I think that was when it first became very apparent to the community at large -- you find that there is a substantial component of becoming comfortable, to see that it does not immediately result in the madmen running wild on the plants, that in fact this is not a license to kill, it's a license to focus, and so, it takes some comfort, especially as you move in these non-traditional areas.

My own experience within the application of PRA within the Department of Energy was that, before it became at all tolerable to people in maintenance, the PRA people had to learn to speak maintenance-ese instead of PRA-ese, and I think that's -- the graded QA may be a classic example of where we need to develop that language out of the quantification of PRA that the QA/QC professionals in the organization can understand in their context, and then we can take these bolder steps with comfort and assurance.

CHAIRMAN JACKSON: Okay.

DR. FONTANA: I take it we don't have to answer.

CHAIRMAN JACKSON: I'm letting you off the hook, let the record show.

DR. APOSTOLAKIS: Well, when I say they were

grossly exaggerated, I didn't mean that -- we have to be precise here. I'm not saying that we should throw out of the window all the requirements.

What has been grossly exaggerated is the significance of the difference between the current requirements and some form of relaxation.

CHAIRMAN JACKSON: I think we understood that.

DR. APOSTOLAKIS: Okay.

CHAIRMAN JACKSON: And to the extent that your recommendation relates to that, then that's the point you want to make to us. Is that correct?

DR. APOSTOLAKIS: Yes.

CHAIRMAN JACKSON: I think you should go on.

DR. APOSTOLAKIS: Risk-informed regulation -- this was an attempt to -- which I thought was successful -- to show that PRA -- that this is an evolutionary process. We are not about to drop defense-in-depth and safety margins. We do want to proceed in a cautious way. Therefore, changes should be small, and of course, they should be monitored using some strategy.

So, I think these five principles -- the formulation of these principles was a significant step forward.

The next slide shows one of the figures -- one refers to CDF, the other to LERF. This is on CDF, and I

think I should make a few comments here.

First of all, the lines between Region I and the other regions should not have been so bright, but I think it's a problem of software. It should have been a smoother transition to send a message that there are uncertainties in PRA, there are imprecisions.

We are not going to make a decision based on whether a number is 10 to the minus 5 or 1.1 10 to the minus 5. So, the transition should have been smoother.

I think the text makes it very clear, but I think it's worth mentioning that.

Second, the issue of -- well, it doesn't show very well there, but as you see in the actual figures in the guide, we have this shade of gray that becomes darker and darker as we approach areas that we don't like, and it's explained in the footnote that this means we'll pay more attention, we'll scrutinize what you're doing more, and I think that's very important because recognizing explicitly again that there are some issues with PRA, but we are aware of them, we're willing to spend the appropriate time to understand what you're proposing if you are in that region.

So, I think that there is an adequate message that's being sent by these two figures, and of course, the text elaborates on these.

Sometimes, you know, trying, again, to be as

complete as we can, maybe we turn people off, because somebody who does not intend to do a complete PRA picks it up and sees all this discussion on model uncertainty and parameter uncertainty and say, my God, I can't do this. But again, it's trying to satisfy many requirements in one document.

But I think it was the right thing to do.

CHAIRMAN JACKSON: Yes, Commissioner McGaffigan.

COMMISSIONER McGAFFIGAN: When you all saw this view-graph last fall, it had that 10 to the minus 7 and negligible category in it. Should it have been retained? It basically had one ore -- it had Region IV, I guess.

DR. APOSTOLAKIS: I don't remember that.

COMMISSIONER McGAFFIGAN: You don't remember that.

DR. APOSTOLAKIS: I remember that Region III was not going to the right as far as it goes now. No, Region III did not exist at all. That's why I'm confused.

COMMISSIONER McGAFFIGAN: Region III didn't exist? I have seen a view-graph where there is a 10 to the minus 7 and below -- it would imply that the degree of review would be quite modest for things down in that category, and I was wondering whether you had any views on retaining that category or not.

DR. APOSTOLAKIS: My personal view is that it would not really serve any purpose to add it there, but

that's personal. The Committee hasn't discussed this.

COMMISSIONER McGAFFIGAN: You talked about the words, you think, make up for the fact that the lines look sort of bright on the view-graph. I'm not absolutely convinced of that. I think proof will be when somebody comes in at the margins of one of these bright lines and asks something where no changes are allowed.

If I'm not .9 times 10 to the minus 5 today and I propose something that's going to be 1.1 times 10 to the minus 5 and, therefore, is in the region where no changes are allowed, then I'd still be a 2, which is a factor of 5 better than this goal that we don't have of 10 to the minus 4. Should I not be considered at that point, or should I be considered?

I take your remarks to mean that maybe I should get considered even though -- if there's a good reason for it. If I'm going to save large amounts of money and I'm still well within any regulatory requirement, maybe I should be considered.

I'm not sure the words in the reg guide reflect that, but you all are saying put it out and let's get some practice and maybe we'll get some hard cases at that point.

DR. POWERS: I definitely think practice is essential here, but you raised the question of review, how much review is required, a very minimal amount of review.

I think we ought not forget there is a big tough nut to crack when you come into this risk-informed regulation, and that is the review on your PRA that you're basing this on.

That is a non-trivial review that the staff is going to have to undertake, and it's compounded by the fact that, in many cases, the total quantification of risk is going to involve some estimations.

Those estimations become more pandemic once you go to any kind of WARF number. This is a non-trivial burden for a licensee to approach even if he's coming in with one of his 10 to the minus 7th sort of things.

Now, I think he gets over that once -- once he's done one, it becomes a lot easier after that, because staff's not going to go back to ground zero on every review for every licensee, I'm sure, but there is a tough issue we face here for -- in thinking about where your resources -- your manpower resources are going to go in regards to this risk-based regulation.

You've got a front-end cost on this that's non-trivial, and I assure you, the licensees are concerned about that cost. They are not interested in getting involved in something where they will, to quote them, be run ragged chasing thousands of our requests for additional information.

They need some confidence and some standardization here to approach -- whenever we get to talk about fire protection, we'll get into that issue more realistically, because it is a barrier there.

DR. SEALE: I would add, I think the prompt attention to Reg. Guide 1.174-type requests and pilot studies and so forth is probably the single most important aspect of encouraging licensees to be responsive to the offer of risk-informed regulation.

DR. APOSTOLAKIS: Okay.

The next topic is the report we sent in December on uncertainties versus point values, and again, this summarizes the recommendations.

I would like to say a few words about the first bullet, which sounds like a trivial thought, you know, to what degrees are confidence of the PRA results and insights will improve on the existing regulatory system.

I submit to you that is a question that is never asked. The question that is always asked is, is PRA perfect to be applied to this new area and not whether PRA can contribute to doing things better.

So, we thought it was important to put that there even though it doesn't really relate to uncertainties and point values.

COMMISSIONER DIAZ: What is the answer to the

question?

DR. APOSTOLAKIS: What question?

CHAIRMAN JACKSON: The question posed here.

DR. APOSTOLAKIS: It's situation-specific. We had the presentation of higher perfection the other day, and the discussion was all on the limitations of higher PRA. Nobody told me anything about the limitations of the existing regulations regarding fires.

I would like to see two columns. The existing regulation has these problems and it does certain things well. PRA has these problems, but it also does certain things well, and when you put the two together, you have a better system.

CHAIRMAN JACKSON: I think that I would warn against statements that go too far to the pejorative, because I think, in fact, the kinds of questions the Commission was asking in the fire protection briefing, in fact, were exploring just that issue in terms of what the limitations are of the current situation vice where we might go in a risk-informed approach, and the Commission has not made a decision on that yet, and so, I think we should leave it at that.

DR. APOSTOLAKIS: I was not referring to that.

DR. SEALE: We get the language from other places, as well.

DR. APOSTOLAKIS: It was a theme that was coming back when we were discussing the regulatory guides and so on. It was always how good is PRA, PRA doesn't do this, PRA doesn't do that, and what we're saying here that's only one part of the question.

CHAIRMAN JACKSON: I think what you're doing is -- I think we're moving down this track, so let's keep moving down the track.

DR. APOSTOLAKIS: Now, plant-specific application of safety goals -- Dr. Kress will say a few words about that.

CHAIRMAN JACKSON: Slide 23.

DR. APOSTOLAKIS: Twenty-three.

DR. KRESS: The question arose, of course, because the safety goal policy statement specifically says not to do this, and then we come up against what's here called DG-1061, which is now Reg. Guide 1.174, which goes right ahead and does that in the context of requests for changes to licensing basis, and it came to us as a question as to whether that was appropriate or not, and we came down on the side that it certainly was; in fact, there was no other way to do 1.174.

Then the question broadened itself to the whole subject of risk-informed regulations in general, not just in the context of changes to the licensing basis, and it was

our feeling that, in order to have a coherent system like that, you have to do it on a plant-specific basis, and that if you're going to use the safety goals as your top-level criteria, that they have to be applied on a plant-specific basis. It was just apparent to us. So, there was nothing very deep there.

The question then got down to the surrogates, the LERF and the CDF, to the QHOs, and is it possible to use those on a plant-specific basis when the QHOs actually involve site characteristics and population and so forth, and our final conclusion was, yes, there's not that much variability in the effects of the site, that you can actually use those and they will focus your attention on the things that we can best deal with in a regulatory agency, and that's the meaning of the other two bullets.

CHAIRMAN JACKSON: Okay.

DR. KRESS: We also did note on this last bullet that there probably ought to be more attention to developing -- if we revisit the safety goal policy statement, there ought to be more attention given to developing a societal risk measure, because the ones we have now intend to do that, but in practice, they focus on individual risk, and we felt one risk -- societal risk was total early fatalities as opposed to individual, was a rather robust one.

It's not the only one. One should think about

land interdiction and other things, but we think that would be a good listing to the safety goals if, indeed, they are revisited.

CHAIRMAN JACKSON: Okay.

DR. APOSTOLAKIS: The final subject is elevation of CDF for fundamental safety goal and possible revision of safety goal policy.

As you see here, we have very carefully listed only facts. We're still debating the issue. There is a meeting tomorrow with the staff to discuss certain things, and we felt it was important to schedule a subcommittee meeting two weeks from today to go more deeply into these issues. So, maybe we should leave it at that today.

COMMISSIONER DIAZ: I just wanted to look at the entire presentation, and like Chairman Jackson said, we already engaged the staff on this.

If you look at your presentation, the presentation was really on risk-informed regulation. Yet, the title says risk-informed performance-based, and I think we are trying to make the point that these issues should be separated, and when they are together, that's fine. They're together, they mean something different, because the process is much more complex than if you look at each one of them by themselves.

And if I might go as bold as going to when I asked what is the answer, I think it would be important if the

Commission would get some sense from the Committee in that, when applied properly, cases in which PRA have definitely improved the regulatory system, because asking a question is great, but if we could have at least some specific answers, like you said, that are area-specific, then that will certainly help us to get a better idea.

DR. APOSTOLAKIS: Did you ask where PRA can or has? I didn't catch the verb.

COMMISSIONER DIAZ: I think both.

DR. APOSTOLAKIS: Okay.

COMMISSIONER DIAZ: It will be an important contribution to our body of knowledge.

DR. APOSTOLAKIS: Regarding the title, I think we sort of routinely, since day one, have been using risk-informed performance-based regulation, you are right, this was on risk-informed part only. From now on we should be more careful.

We did have a discussion today on performance-based regulation, by the way, so we are following that, but you're absolutely right, this was not part of it.

CHAIRMAN JACKSON: That is not a statement, because you have performance-based regulation without risk-informed.

DR. APOSTOLAKIS: Exactly.

CHAIRMAN JACKSON: And vice versa or both.

DR. APOSTOLAKIS: That's right.

CHAIRMAN JACKSON: That's the point.

DR. APOSTOLAKIS: But this presentation did not address performance-based regulation at all.

CHAIRMAN JACKSON: Well, in some ways one could argue that this presentation was PRA regulation.

DR. APOSTOLAKIS: As it should be.

DR. SEALE: Is that all, George?

CHAIRMAN JACKSON: I think so.

DR. APOSTOLAKIS: Yes.

DR. SEALE: Next we'll discuss --

CHAIRMAN JACKSON: Because I'm chairing this meeting, that's all.

DR. SEALE: We'll discuss the AP600 review.

MR. BARTON: AP600 -- it seems that the meetings have been going on forever, since 1991, the Subcommittee first met with Westinghouse and the staff. We seem to be able to see the light at the end of the tunnel. There have been no recent contentious issues such as in-containment spray system, but I think the process is moving. We've had meetings with Westinghouse this week. Six more chapters were reviewed -- SAR plus draft SERs -- and questions are getting closed out raised by the staff and also by the Subcommittee and the full Committee.

The major hard spots between -- we see between now and the schedule and issuance of the final report are the issues that the thermal hydraulics subcommittee has had with the test analysis program, and Dr. Kress has a few comments on those issues and where he sees their resolution.

DR. KRESS: I don't know that most of these issues arise, thermal hydraulics, because thermal hydraulics is so important or because of personalities. I get different views from the Committee on that.

It does seem that most of the bones of contention have been in that area.

I would like to say that the test analysis program that Westinghouse has done to demonstrate that their plant meets the requirements and that their codes are valid has been very impressive and, I think, a very good set of programs, and we think, as a Committee, that the -- we've listed a number of issues that have come up in the thermal hydraulics subcommittee. We put them, in I think, in our interim AP600 letter -- I forget the date. They were divided between the RCS and the containment in terms of issues.

I don't really see any show-stoppers in either of those. These have been -- the staff has been very responsive in putting these together as requests for additional information from Westinghouse. We are looking

for responses back to those.

I think there are legitimate good answers to all of them, particularly with the RCS.

The one area that I see may still be a problem has to do with the containment, and the problem is hard to put into words, because I think, if you look at the codes they use, which, in particular, GOTHIC is one of them, it's a lump-parameter code, and in order for the thermal hydraulics part and the fission product behavior part of those to be appropriate for AP600, you have to demonstrate that AP600 is a well-mixed containment, and they have not come forth with an appropriate demonstration to us to convince us that they do, sure enough, have a well-mixed and handle the stratification problem well.

CHAIRMAN JACKSON: Was this the first time these issues had been raised?

DR. KRESS: I think we raised them -- it's a question of how much emphasis is actually put on them, because sometimes you raise an issue in a meeting, a subcommittee meeting, and it gets on the minutes and not much more gets done about it sometimes. But they have been raised.

DR. POWERS: These issues have been focuses of attention -- foci of attention since the AP600 design was first advanced as a passive plant with natural circulation.

COMMISSIONER DIAZ: If I might go to that first bullet, I think this is a matter that even I am confused at times. Lack of adequate justification for level of conservatism. I understand lack of adequate justification, but as to level of conservatism, is it too high, adequate, or too low? It doesn't tell me there which way you're pointing.

And then, in relation to the Chairman's question, there's an enormous laundry list of issues that came very late.

DR. KRESS: Those didn't come very late. They were just consolidated from various lists that existed up to then. We wanted to get them all on one plate.

This one bullet -- number one, I don't think there is a regulatory requirement for level of conservatism. We're talking about peak clad temperature here in design basis space. This is the RCS.

The regulatory requirement says that, when making the analysis to determine what your peak clad temperature is for the various design basis accidents, that you use a conservative analysis.

They haven't demonstrated yet to us that the conservatisms they have claimed for the analysis are really conservatisms that add up to a conservatism that one would be comfortable with.

But I have to say, personally, I think the RCS is not a problem, that they have good ECCS systems. The analysis codes, why they have a lot of difficulties dealing with these low-pressure flows and stuff -- the test and analysis program is very robust and has demonstrated to me that they really do not have a problem. They're a much better system than standard plants.

COMMISSIONER DIAZ: So you would say the level of conservatism in the proposed design based on the calculation is adequate.

DR. KRESS: Not based on the calculation, based on the test and analysis program. But the calculations still need to be -- some issues still need to be -- I do not think they will -- when their issues are finally ironed and the questions are answered, I don't think the answer will be yes, we are in bad shape and the conservatisms aren't there. I think the answer will be it's okay, we've proven it for the RCS.

It's a little different with the containment. The containment -- what I see there is a code that is a lump-parameter. It has known errors in it that we pointed out. The calculations -- the conservatisms they claim in the calculations haven't been demonstrated at all and are, indeed, somewhat small.

The calculated peak pressure with respect to

design pressure requires you to take credit for all the heat transfer mechanisms, to the thermo-dynamics of mixing with the atmosphere, heat transfer to the walls, heat transfer to the structures, plus the passive containment cooling system, and then you barely peak at the peak pressure, and this is coupled with the fact that they haven't demonstrated it's well-mixed, and if it's not well-mixed, this is not conservative.

Plus they have an aerosol calculation that involves using the lambda, the decay factor, that invokes diffusiophoresis, diffusion, sedimentation, agglomeration, as well as thermophoresis, and basically that's unprecedented in our regulations, we have never allowed that before, and to me, they haven't demonstrated that they've conservatively chosen those values, and with this combination, you end up just barely meeting 10 CFR 100 guidelines, just barely, and what we have is a containment that's basically a volume like a standard plant.

It's relatively weak in pressure, like 45 psi design pressure. That's pretty strong, but -- compared to a BWR, but compared to a large dry -- and it's a thin shell, which we've had little experience with, and think shells tend to fail catastrophically as opposed to leaking like a containment, and you barely meet the design basis criteria and you don't have a spray.

The aerosols stay in there a long time, the pressures stay in there a long time, and although you meet what appears to be all the regulatory requirements, it doesn't leave us with a warm feeling.

COMMISSIONER DIAZ: It might be worthwhile if you would bound your real concerns in this area so the staff will have an area which they can point and focus on.

DR. KRESS: I think we have, and I think it involves looking at the answers to the requests for additional information and seeing what the revised scaling analysis, what the revised code results give us, and then we could make a better assessment.

CHAIRMAN JACKSON: Let me ask you two questions.

You know, the staff has stated that ITAAC will be open still on May 1st on their FSER submittal to you, but they hope to close it out shortly thereafter. Does that pose a problem for you?

MR. BARTON: The information they gave us at the Subcommittee, if they meet the commitment, that will not be a problem. The Final SER by May 1 is the only question-mark at this point, whether they can support that date.

COMMISSIONER McGAFFIGAN: Sort of following on Commissioner Diaz, as I understand this issue --

CHAIRMAN JACKSON: Actually, I wasn't done.

COMMISSIONER McGAFFIGAN: I'm sorry.

CHAIRMAN JACKSON: The second question -- the staff reduced the open items from about 500 to 7 over the last couple of months, and I understand that you were briefed on one of these open items, fire protection, this week. Do you have some initial assessment of the staff's position in this area?

DR. POWERS: We have an initial assessment that we're going to look at it more carefully. We've asked for that through a fire protection subcommittee activity.

My assessment is that we will find the staff position in their SER and the Westinghouse position in their application supportable, that it's essentially taking an Appendix R position.

We just want to look at it a little more closely, and we have some concerns about feedwater supply and things of detail like that that we just need to look at a little more closely than we were able to do in our grander subcommittee meeting.

MR. BARTON: We will re-look at those in the May subcommittee meeting.

DR. POWERS: We are committed to close that out for Mr. Barton and his work for the may subcommittee meeting, and I would not want to leave you feeling that we have identified some red-flag issue. We just want to walk through the details fairly carefully on this.

This is one of those lovely prescriptive regulations that you can go through check-lists, and we're going through the check-list.

CHAIRMAN JACKSON: Commissioner McGaffigan.

COMMISSIONER McGAFFIGAN: I just want to understand the issue that you're talking about with containment.

The staff isn't here, but I understand the staff doesn't share the same concerns that you all have with the use of the codes, and I'm just trying to understand how we are going to -- whether that is a resolvable matter in the next month.

DR. KRESS: I think it's resolvable. I think the staff has asked for requests for additional information that reflect the concerns that we have on containment, and we're awaiting these answers to come back, and so is the staff. I don't know whether they actually --

COMMISSIONER McGAFFIGAN: When I listen to you, just to try to -- theoretically, one could construe you as saying they have to come up with a new code --

DR. KRESS: Oh, no.

COMMISSIONER McGAFFIGAN: -- invent it as they go along.

DR. KRESS: No.

COMMISSIONER McGAFFIGAN: No?

DR. KRESS: No. In fact, a demonstration by other means that the AP600 is well mixed would certainly go a long way in my mind to saying that the GOTHIC code is an appropriate way to treat the analysis for AP600. No, there's definitely not a need for a new code.

DR. POWERS: I think that, when the examination of AP600 began, it certainly became clear that it would sure be nice to have a code that solved the momentum equation instead of lump-parameter codes, but a stride that has been made over the last few years has been to recognize, indeed, with appropriate calibration against experiments, it is possible to justify the use of a lump-parameter code.

There's no question in our mind that, if we'd had a fast-running CDF-type code -- competition fluid dynamics code, I'm sorry -- that could apply to this containment, things might have gone more smoothly, but we don't, and we have to rely on a lump-parameter code.

That means you have to have an excellent calibration against experiments and scale properly to the actual plant, and it's those details that you go through, and it's a grinding sort of thing to go through, because you are doing an approximation to the Navier-Stokes equation, and those approximations need to be justified, and there's a rigorous, precise science associated with that. That's all we're doing.

MR. BARTON: That's it for AP600.

CHAIRMAN JACKSON: I actually think I'm going to allow the Commissioners to ask any final questions. We're actually going to end the meeting on this subject.

COMMISSIONER DICUS: To reiterate, you said there's no red flags so far. You should know by now if there are. You don't anticipate any?

DR. SEALE: Well, certainly, if we can get a satisfactory word on this mixing problem in the containment, that's the one area where I see an issue that could give all of us pause.

COMMISSIONER DICUS: Okay. But given that, you think this September date is meetable?

DR. SEALE: Yes. We certainly plan to meet our schedule.

MR. BARTON: Which is a July report to the Commission.

DR. SEALE: That's right. Yes.

CHAIRMAN JACKSON: Very good. Thank you.

I think this has been a very healthy discussion --

DR. SEALE: Thank you.

CHAIRMAN JACKSON: -- and your views are critical in our evaluation of a number of difficulty and, frankly, I think very forward-looking stances and issues that the Commission is dealing with, and I, therefore, encourage you

to continue to be forward-looking in bringing issues to our attention, and we'll cull through the remaining list and see which ones might be appropriate for our next discussion.

DR. SEALE: Let me make one statement.

CHAIRMAN JACKSON: Please.

DR. SEALE: It's a real pleasure for us to get again a demonstration that, when we make our recommendations, they are not recommendations that are -- well, they receive scrutiny --

CHAIRMAN JACKSON: Yes.

DR. SEALE: -- receive critical thought on your part, and that's the only way we can possibly have an impact, is if they do, and we appreciate it very much.

CHAIRMAN JACKSON: Well, that's the game in town. We're adjourned.

[Whereupon, at 2:35 p.m., the public 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: Thursday, April 2, 1998

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: Tamarra Shaffer

Reporter: Mark Mahoney

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

March 26, 1998

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, APRIL 2, 1998—SCHEDULE/BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners between 1:00 and 2:30 p.m. on Thursday, April 2, 1998, to discuss the items listed below. Background materials related to these items are attached.

A.1	Introduction - NRC Chairman	1:00 - 1:05 p.m.
2	Opening Comments - ACRS Chairman	1:05 - 1:10 p.m.
B.1	Senior Management Meeting Process - Mr. Barton (slides 2-6)	1:10 - 1:30 p.m.
2	Proposed Revision to 10 CFR 50.59 Process - Mr. Barton/Dr. Kress (slides 7-12)	1:30 - 1:40 p.m.
3	Risk-Informed, Performance-Based Regulation, Including Use of PRA in the Regulatory Decision Making Process - Dr. Apostolakis (slides 13-24)	1:40 - 2:00 p.m.
4	Status of AP600 Review - Mr. Barton/Dr. Kress (slides 25-28)	2:00 - 2:10 p.m.
5	Shutdown and Low Power Operation - Dr. Powers (slides 29-31)	2:10 - 2:15 p.m.
6	NRC Safety Research Program - Dr. Seale/Dr. Powers (slides 32-35)	2:15 - 2:20 p.m.

7	License Renewal - Dr. Fontana (slides 36-40)	2:20 - 2:25 p.m.
8	Fire Protection Rulemaking - Dr. Powers (slides 41-43)	2:25 - 2:30 p.m.

Attachment: As stated

cc: ACRS Members
ACRS Technical Staff

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

MEETING WITH
U.S. NUCLEAR REGULATORY COMMISSION

APRIL 2, 1998

SENIOR MANAGEMENT MEETING PROCESS

MR. JOHN BARTON
ACRS

BACKGROUND

- March 1997: ACRS review of Arthur Andersen recommendations to improve the Senior Management Meeting process

- September 1997: ACRS review of NRC Action Plan to improve Senior Management Meeting process - Committee report dated September 10, 1997

- March 1998: ACRS Review of Integrated Assessment Process - Committee report dated March 13, 1998

COMMENTS AND RECOMMENDATIONS:
ACRS REPORT - SEPTEMBER 10, 1997

- o Use of a hierarchial structure to define SMM information needs
- o Need to determine if significant plant events foreshadowed prior plant performance
- o Assessment of the pros and cons on the use of economic indicators
- o Ensure objective plant performance standards/reduce reliance on event-driven assessments
- o Development of tools for assessing management/organizational effectiveness
- o Lack of bases for criteria used in the Plant Performance Template

ACRS REPORT - MARCH 13, 1998

- o Develop specific objectives and performance measures
- o Demonstrate, by example, that Decision Logic Model is understood and workable
- o Evaluate Template Categories in conjunction with Assessment Logic Model
- o Develop and test tools for assessing management and operational effectiveness
- o Perform additional research prior to use of economic indicators

ACRS REPORT - MARCH 13, 1998 (CONT'D)

- o Ensure consistent assessment results among Regional Offices

- o Work more closely with industry to agree on a set of Performance Indicators

- o ACRS requests to review prior to issuance for public comment:

  - Requirements for Integrated Assessment Program
  - Decision Logic Model including model demonstration and associated Template evaluation

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 10, 1997

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

Dear Chairman Jackson:

SUBJECT: STAFF ACTION PLAN TO IMPROVE THE SENIOR MANAGEMENT MEETING PROCESS

During the 444th meeting of the Advisory Committee on Reactor Safeguards, September 3-5, 1997, we met with representatives of the NRC staff to discuss its Action Plan to improve the Senior Management Meeting (SMM) process. Our Subcommittees on Probabilistic Risk Assessment, Plant Operations, and Fire Protection also discussed this matter during a joint meeting on August 28-29, 1997. We also had the benefit of the documents referenced.

The SMM process is being revised in response to Commission direction. A report prepared by Arthur Andersen contained recommendations for improving the SMM process in two areas: the SMM information base and the SMM evaluation process. The first area for improvement involves inputs to the SMM decisionmaking process, including performance indicators and the decision criteria used by the senior managers. The second area involves the role of SMM participants, the method of reaching consensus, the presentation of information, and the documentation of meeting results.

The objectives of the revised SMM process are: to provide more structure to the performance evaluations, increase participation of senior managers, improve consistency among the Regions, and enhance the scrutability of the process and decisions to both the Commission and the public. In addition, the Commission directed the staff to make further improvements to the SMM process by developing better performance indicators that can provide a more objective basis for judging whether a nuclear power plant licensee can be placed on or be removed from the NRC Watch List. These improved indicators and objective measures are expected to enhance the staff's ability to take appropriate regulatory actions, including additional enforcement measures — some of which have not been effective in the past.

The staff presented to us its plan to improve the SMM information flow, in order to obtain objective data for use in the assessment process. A key element in this improved process is the development of a Performance Template which is designed to coordinate all relevant data to improve decisionmaking.

We support the goal of codifying the SMM information gathering and review process, however, the basis for the top-level criteria contained in the Template is not clear. Furthermore, the process by which the Template leads to the formulation of decisions is also not apparent.

We would prefer to see a top-down structure that starts with the point of decision, identifies the objectives of the decision, and proceeds to define the informational needs. For example, in a risk-informed approach, one could envision as an objective (one of several) the prevention of the occurrence of initiating events and the degradation of safety functions. To satisfy this objective, one would look for precursors to these undesirable events and then would proceed to identify relevant performance indicators, and address the issue of how these would be measured. In this way, the logic behind the Template would be transparent and easy to communicate to the various stakeholders. A similar systematic approach would be taken for the other objectives. The staff told us that such a hierarchical structure will be developed. We recommend that its development be accelerated, and we would like to be kept informed.

In addition to the concern with the Template discussed above, we recommend that the staff address the following items regarding its efforts to improve the SMM information base:

- Examination of a sample of significant operational events is needed to determine if they were foreshadowed by prior plant performance.

- Careful assessment of the pros and cons on the use of economic indicators is needed, as the relationship between economic indicators and safety performance is not clearly understood.

- Evaluation of how the revised process will focus on the competency of plant management and culture is needed.

- Assurance is needed that the new performance standards are objective and reduce reliance on event-driven assessments.

We note that although progress has been made in improving the information base of the SMM, considerable work remains in such areas as the development of tools for assessing management/organizational effectiveness and testing their implementation.

The staff also discussed the status of its plans to perform an integrated review of the NRC assessment process. The Office of Nuclear Reactor Regulation (NRR) has a number of programs in place to assess licensee performance. Among these are the Systematic Assessment of Licensee Performance, the SMM process, the Plant Performance Review, and the Plant Issues Matrix. Each of these programs provides insight on some aspects of licensee performance. Currently, there is no integrated assessment of licensee performance. The NRR staff is undertaking a review and examination of its current programs with the intention of identifying improvements that will provide a better, integrated, and more comprehensive assessment of licensee performance. Development of a hierarchical structure similar to the one recommended above would be useful here.

The staff plans to complete its integrated review of the assessment process by March 1998, and provide recommendations to the Commission by June 1998. The staff has not yet defined the requirements (preferably quantitative) for an adequate program to assess licensee performance. It is not apparent to us how well-designed recommendations can be formulated without the explicit definition of the requirements for the assessment program to meet Agency needs. It is not clear how preferred options can be selected absent explicit requirements. We strongly recommend that NRR develop requirements for an adequate licensee performance assessment program.

We plan to meet with the NRC staff as it continues its integrated review of the NRC assessment process for operating commercial nuclear power plants.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

<u>References</u>:

1. SECY-97-072, Memorandum dated April 2, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, for the Commissioners, Subject: Staff Action Plan to Improve the Senior Management Meeting Process.
2. SECY-97-192, Memorandum dated August 21, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, for the Commissioners, Subject: Peer Review of the Arthur Andersen Methodology and Use of Trending Letters.
3. U.S. Nuclear Regulatory Commission, Senior Management Meeting (SMM), Directive 8.14, Volume 8: Licensee Oversight Programs, Approved March 19, 1997.
4. Memorandum dated March 14, 1997, from John C. Hoyle, Secretary, NRC, to L. Joseph Callan, Executive Director for

Operations, NRC, Subject: Staff Requirements - Briefing on Analysis of Quantifying Plant Watch List Indicators (Arthur Andersen Study), Commissioners Conference Room, February 18, 1997.

5. Memorandum dated June 24, 1997, from John C. Hoyle, Secretary, NRC, to L. Joseph Callan, Executive Director for Operations, NRC, and John T. Larkins, ACRS/ACNW, Subject: Staff Requirements - Briefing on Staff Response to Arthur Andersen Study Recommendations, April 24, 1997, Commissioners Conference Room.
6. Memorandum dated June 30, 1997, from John C. Hoyle, Secretary, NRC, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements - Briefing on Operating Reactors and Fuel Facilities, June 25, 1997, Commissioners Conference Room.
7. Memorandum dated August 19, 1997 from John C. Hoyle, Secretary, NRC, to L. Joseph Callan, Executive Director for Operations, NRC, and John T. Larkins, ACRS/ACNW, Subject: Staff Requirements - SECY-97-122 - Integrated Review of the NRC Assessment Process for Operating Commercial Nuclear Reactors.

September 11, 1997

MEMORANDUM TO: Chairman Jackson
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan

FROM: John T. Larkins John T. Larkins
Executive Director, ACRS/ACNW

SUBJECT: ACRS LETTER ON THE SENIOR MANAGEMENT MEETING PROCESS, SEPTEMBER 11, 1997

You recently received a letter from the ACRS (dated September 10, 1997) related to NRC staff initiatives to improve the Senior Management Meeting process. ACRS Member Dr. George Apostolakis has provided some additional comments (attached) related to the staff's initiative that further elaborate his views on this matter. These comments were not available during Committee deliberations and, therefore, are provided separately from the report on this subject.

Attachment:
Letter dated September 8, 1997, from G. E. Apostolakis, MIT, to J. T. Larkins, ACRS, regarding ACRS Letter on the Senior Management Meeting

cc: J. Callan, OEDO
T. Martin, AEOD
S. Collins, NRR
M. Knapp, RES
ACRS Members

MASSACHUSETTS INSTITUTE OF TECHNOLOGY

77 Massachusetts Avenue Cambridge, Massachusetts 02139-4307

George Apostolakis
Professor
Room 24-221

tel: (617) 252-1570
fax: (617) 258-8863
e-mail: apostola@mit.edu

September 8, 1997

Dr. J.T. Larkins
ACRS Executive Director
Nuclear Regulatory Commission
Washington, DC 20555-0001

RE: ACRS Letter on the Senior Management Meeting

Dear John:

I fully agree with the letter that the Committee has prepared. I offer the following comments in order to elaborate on my views regarding the top-down approach and to offer constructive suggestions to the staff.

My research group at MIT participated recently in a project sponsored by the Department of Energy entitled "Risk Communication, Assessment, and Management at Hazardous Waste Sites." One of the goals was to bring all the stakeholders into the decision-making process (the selection of a remedial action alternative, RAA) and to improve communication between the technical community and the stakeholders.

An important part of the adopted approach was the development of a "value tree." This allowed the stakeholders to structure their concerns and the analysts to understand better what these concerns were. The top four levels of the attached diagram show this tree.

The tree starts from the very general *objective categories* and proceeds to more specific items. The categories cover a wide variety of concerns, from programmatic to human health & safety. Note that these categories must refer to *fundamental* (as opposed to *means*) objectives, i.e., they should not contain objectives that are means of achieving other objectives belonging to a different category.

The next level shows the *objectives*. A higher degree of specificity is, thus, achieved. The objectives themselves are not measurable. The *performance measures* (shown at the next level of the tree) are measurable (or can be calculated).

Since the objective of that project was to develop a decision-making methodology, the available decision options are represented at the bottom of the tree ("select RAA"). The impact of each alternative on the Performance Measures is calculated using available models (the "analysis and assessment" level of the tree; the developed influence diagram is not shown). Note that several of these objectives are dependent in a probabilistic sense. For example, "groundwater contamination," an event that will appear at the "analysis and assessment" level, will influence several Performance Measures.

We found this approach to contribute to communication significantly. For example, several stakeholders (in fact, the majority) did not wish to see the public risks under the "human health & safety category" but, rather, under the "environment" category. This came as a surprise to the analysts, yet the stakeholders insisted. I am convinced that, without the benefit of the diagram, such an input from the stakeholders would have been very difficult to obtain.

At this point in the restructuring of the SMM process, I think that the most valuable part of this approach is the top structure (the value tree). It would add a significant amount of scrutability to the process, if such a tree were to be developed to show explicitly why a particular structure of the Template is adopted. The staff would send a clear message to the industry regarding its objectives and would make a convincing case as to why certain information is needed. The logic of the Template would be transparent and its elements defensible. For example, as I stated during our meeting with the staff, I don't understand the current place of "culture," which is a very broad concept, in the Template. Furthermore, I don't know what the fundamental objectives of the SMM process are and why the information listed in the current Template helps the staff achieve these objectives.

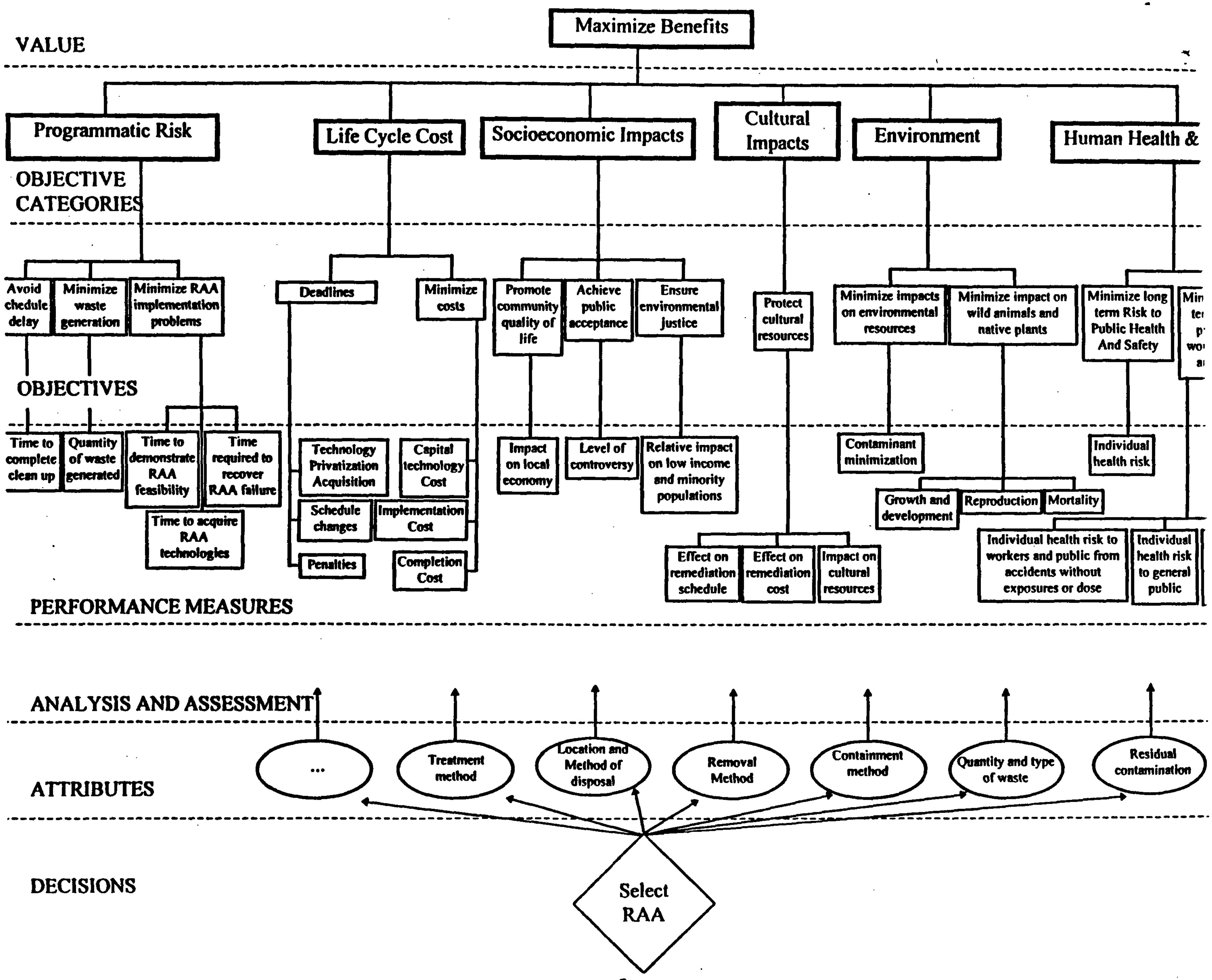
The value tree will also be very useful later, when the decision-making model is constructed. For example, the various categories and performance measures can be prioritized using standard analytical tools, such as the Analytic Hierarchy Process. But this will have to wait until a good value tree is developed.

I hope that these comments are helpful and constructive. I realize that the problem described above and the SMM process are not identical. I do believe, however, that the concepts and the analytical tools are transferable.

Please forward copies of this letter to the staff, as appropriate.

Sincerely,

G.E. Apostolakis
Professor


VALUE
Maximize Benefits
OBJECTIVE CATEGORIES
Programmatic Risk
Life Cycle Cost
Socioeconomic Impacts
Cultural Impacts
Environment
Human Health &
OBJECTIVES
Avoid chedule delay
Minimize waste generation
Minimize RAA implementation problems
Deadlines
Minimize costs
Promote community quality of life
Achieve public acceptance
Ensure environmental justice
Protect cultural resources
Minimize impacts on environmental resources
Minimize impact on wild animals and native plants
Minimize long term Risk to Public Health And Safety
PERFORMANCE MEASURES
Time to complete clean up
Quantity of waste generated
Time to demonstrate RAA feasibility
Time required to recover RAA failure
Time to acquire RAA technologies
Technology Privatization Acquisition
Capital technology Cost
Schedule changes
Implementation Cost
Penalties
Completion Cost
Impact on local economy
Level of controversy
Relative impact on low income and minority populations
Effect on remediation schedule
Effect on remediation cost
Impact on cultural resources
Contaminant minimization
Growth and development
Reproduction
Mortality
Individual health risk
Individual health risk to workers and public from accidents without exposures or dose
Individual health risk to general public
ANALYSIS AND ASSESSMENT
ATTRIBUTES
...
Treatment method
Location and Method of disposal
Removal Method
Containment method
Quantity and type of waste
Residual contamination
DECISIONS
Select RAA

October 28, 1997

Dr. Robert L. Seale, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: STAFF ACTION PLAN TO IMPROVE THE SENIOR MANAGEMENT MEETING PROCESS

Dear Dr. Seale:

On September 3-5, 1997, representatives of the NRC staff met with the members of the Advisory Committee on Reactor Safeguards (ACRS) to discuss the staff's Action Plan to improve the Senior Management Meeting (SMM) process. In addition, representatives from the Office of NRR discussed the status of its plans to perform an integrated review of the NRC assessment process. These plans include a review and examination of the current programs with the intention of identifying improvements that will provide a better, more integrated, and more comprehensive assessment of licensee performance.

On September 10, 1997, you issued a letter to Chairman Jackson recommending that the staff address certain items regarding the staff's efforts to improve the SMM process.

The purpose of this letter is to address the ACRS Committee's concerns regarding the following items:

ITEM 1: The process by which the plant performance template leads to the formulation of decisions is not apparent. ACRS recommends a top-down structure that starts with the point of decision, identifies the objectives of the decision, and proceeds to define the informational needs. In addition, the basis for the top-level criteria contained in the template is not clear.

Response: The staff agrees with this suggestion. A cooperative effort between the Offices of AEOD, RES, and NRR is underway to develop a decision model and associated criteria.

ITEM 2: Examination of a sample of significant operational events is needed to determine if they were foreshadowed by prior plant performance.

Response: The Office of AEOD will perform an examination of a sample of significant operational events to determine if they were foreshadowed by prior plant performance. The status of this study will be provided at the next scheduled ACRS briefing to discuss improvements to the SMM.

ITEM 3: Careful assessment of the pros and cons on the use of economic indicators is needed, as the relationship between economic indicators and safety performance is not clearly understood.

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Response: The staff received similar comments from the Commissioners during the September 19, 1997 Commission briefing and is currently assessing the pros and cons regarding the use of economic indicators, as well as the relationship between economic indicators and safety performance. The status of this assessment will be provided at the next scheduled ACRS briefing to discuss improvements to the SMM.

ITEM 4: Evaluation of how the revised process will focus on the competency of plant management and culture is needed.

Response: The Offices of AEOD and RES have initiated development of a process to assess leading indicators of management and operational effectiveness on an ongoing basis. A key milestone in this effort included a week-long workshop which took place in Idaho on August 18-22, 1997. The Arthur Andersen Co. has also been contracted to assist in the development of methodologies for measuring management and operational effectiveness. The staff will develop characteristics, measures, and indicators based on insights from existing NRC inspection programs, ongoing NRC research, and industry evaluation techniques.

After identifying and validating appropriate measures, the staff will determine how to best integrate the assessment of the measures into the agency's processes. In addition, the staff will evaluate the need for staff training in this area and develop any required inspection guidance.

ITEM 5: Assurance is needed that the new performance standards are objective and reduce reliance on event-driven assessments.

Response: The staff's principal goal during the development of the plant performance template is to include all aspects of plant operation. The plant performance template will also ensure objectivity and reduce reliance on event-driven assessments by focusing on operational and organizational influences. In addition, to the extent possible, the indicators chosen for use in the revised trend plots will be related to nuclear safety and regulatory performance; will be based on information readily available to NRC; should not be subject to manipulation; should be comparable among licensees; should reflect a range of performance, should be independent of each other; and should be leading. These include the AEOD Performance Indicators.

ITEM 6: Development of a hierarchical structure for the assessment process (during the integrated review) similar to the process described in item 1 would be useful.

Response: The staff agrees with this suggestion. The integrated review is being implemented using a structured approach that begins with desired decisions and then works down to what is needed to support the decision in each area. In addition, input needed to support the process is being identified. Finally, a cooperative effort between the offices of AEOD, RES, and NRR is underway to develop a decision model and associated criteria.

ITEM 7: Develop requirements for an adequate licensee performance assessment program.

Response: The staff agrees. As one of its first activities, the integrated review team developed criteria (qualitative and quantitative) that the final product process must achieve.

The staff looks forward to continued interaction with the ACRS as we progress toward conclusion of these efforts. We plan to provide the Committee a formal status update during the ACRS meeting scheduled for February 1998. However, the staff will provide the Committee with an earlier status update, as appropriate.

L. Joseph Callan
Executive Director
for Operations

cc: Chairman Jackson
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
SECY
OCA
OPA
OGC
OIG
CFO
CIO
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March 13, 1998

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

Dear Chairman Jackson:

SUBJECT: PROPOSED IMPROVEMENTS TO THE SENIOR MANAGEMENT MEETING PROCESS

During the 448th and 449th meetings of the Advisory Committee on Reactor Safeguards, February 5-7 and March 2-4, 1998, respectively, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss proposed improvements to the Senior Management Meeting (SMM) process and the efforts of the Integrated Review of Assessment (IRA) Team. Our Subcommittee on Plant Operations discussed these matters during a meeting on February 3, 1998. We also had the benefit of the documents referenced.

The proposed combined single process is an improvement over the existing three separate but related assessment processes. The new assessment process proposed by the IRA Team will retain many of the positive attributes of the current process. Also, the new process will not preclude taking appropriate regulatory action in a timely manner and will be closely aligned with the NRC's enforcement policy.

Recommendations

- We recommend that the documents being developed from the IRA effort not be released for public comment until the staff develops a set of explicit program requirements, quantitative if possible, for the plant performance assessment, completes its work on the Assessment Decision Logic Model, and presents both to the Committee for its review.

- The overall objectives stated in Attachment 1 of the draft Commission paper, which was received on February 18, 1998, are not sufficiently specific to allow evaluation of the proposed assessment process. We recommend the development of specific objectives and performance measures that can be applied directly to this process. The Assessment Decision

Logic Model should show how the selected decision options noted in the draft paper will utilize these performance measures.

- We recommend that the staff work through at least one example that uses actual inspection reports to demonstrate that the implementation of the Assessment Decision Logic Model is fully understood and workable. This example should include the conversion of the report findings to numerical scores, the processing of these scores through the model, and the decision reached. We would like to review the example before public comments are solicited.

- We recommend that the six categories of the proposed template be evaluated to determine that they are at the appropriate level and whether they overlap unnecessarily. This evaluation must be done in the context of the Assessment Decision Logic Model.

- We recommend that the staff complete the development and testing of the tools for assessing management and operational effectiveness. The Committee is interested in discussing the results of this effort with the staff when they have completed their work.

- We recommend that economic indicators in their present form not be used in the decisionmaking process at this time and that additional research be performed.

- Indicators that measure plant performance at a more global level, such as those discussed by the industry, would be more useful. We would like to see the staff and NEI agree on a set of performance indicators.

- We recommend that the assessment process contain strong provisions to ensure that consistent results are obtained among the Regions.

Discussion

The Committee has had discussions with the staff and NEI on the status of the NRC Integrated Review of Assessment (IRA) process for operating nuclear power plants. Although the staff has acted upon some previous Committee recommendations, additional work remains to be done. As discussed in our September 10, 1997 report to the Commission, the development of a hierarchical structure of program requirements and decision logic for the assessment process is important to the design of the new process.

In transitioning from a process that had three separate assessments -- systematic assessment of licensee performance (SALP), plant performance review (PPR) and the senior management meeting (SMM) -- to a single assessment process, it is essential to ensure that the requirements of the agency will still be met. These requirements for the single process should be expressed in explicit terms, quantitative if at all possible. A list of these requirements would be useful for evaluating alternate approaches to the assessment process.

The staff is assessing the inputs to the Plant Issues Matrix that include most of the licensee performance indicators from the existing assessment process. We believe that these indicators measure performance at such a low level that the nexus between this performance level and overall plant safety is not evident. We believe that the use of indicators that measure performance at a more global level (such as those discussed by the industry) would be more useful. We would like to see the staff and NEI agree to a set of performance indicators. This work could be accomplished during the workshops planned by the NRC staff.

At present, the staff has found that economic indicators alone are not useful plant performance indicators. They may have value when used in conjunction with technical plant performance indicators but in their present form are not essential for decisions that have to be made. Because economic pressures arising from deregulation may have a significant effect on long-term safety performance, additional research on economic indicators is needed.

The new assessment process moves the evaluation and decisionmaking back to the Regional Offices, where it was before the Senior Management Meeting process began. A key requirement for the new process is that the tools employed, i.e., the Plant Issues Matrix and Assessment Decision Logic Model, contain provisions to ensure that consistent results are obtained among the Regions.

The staff has not completed its work on the Integrated Assessment Process and has not developed an agreed-upon set of requirements for the new process. The process by which the plant performance template leads to the formulation of decisions is not apparent. Development of a hierarchical structure begins with the desired outcome, considers alternate ways to achieve it, and then works down to the most effective means to ensure this outcome. The Committee has yet to see such a design process applied to this issue. We do not believe the staff will receive useful public comment on the proposed IRA documents as they now exist. We recommend that the documents not be released for public comment until the staff develops a set of requirements for the plant performance assessment

program, describes the Assessment Decision Logic Model in sufficient detail, and presents both to the Committee for its review.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

References:

1. Draft Commission paper from L. Joseph Callan, Executive Director for Operations, NRC, to the Commissioners, Subject: Update on the Status of the Integrated Review of the NRC Assessment Process for Operating Commercial Nuclear Reactors, received February 18, 1998, (Predecisional)
2. Draft report (LA-UR-97-4911) dated December 17, 1997, Prepared by Los Alamos National Laboratory for Office of Nuclear Reactor Regulation, "Integrated Review of the Nuclear Regulatory Commission Assessment Process for Operating Commercial Nuclear Reactors," Working Report 3: Conceptual Design of the Revised Assessment Process. (Predecisional)
3. Note dated February 27, 1998, from Jack E. Rosenthal, Office for Analysis and Evaluation of Operational Data, NRC, to Michael T. Markley, ACRS, transmitting Draft Report AEOD/S98-xx, Prepared by William S. Raughley, AEOD, "Special Study Identifying Financial Indicators," dated February 27, 1998. (Predecisional)
4. Memorandum dated January 20, 1998, from Richard J. Barrett, AEOD, to John T. Larkins, ACRS, transmitting AEOD draft report, "Interim Report on the Development of the Plant Performance Template," dated January 22, 1998. (Predecisional)
5. Memorandum dated November 6, 1997, from C. E. Rossi, AEOD, to Addressees, Subject: Request for Review of Interim Report - Development and Findings of the Performance Trending Methodology. (Predecisional)
6. Memorandum dated February 10, 1998, from John C. Hoyle, Secretary of the Commission, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements - Briefing on Operating Reactors and Fuel Facilities, January 21, 1998
7. Memorandum dated October 24, 1997, from John C. Hoyle, Secretary of the Commission, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements - Briefing on Improvements in Senior Management Assessment Process for Operating Reactors, September 19, 1997.
8. Report dated September 10, 1997, from R. L. Seale, Chairman, NRC, to Shirley Ann Jackson, Chairman, NRC, Subject: Staff Action Plan to Improve the Senior Management Meeting Process

9. Memorandum dated September 11, 1997, from John T. Larkins, ACRS, to the Commissioners, Subject: ACRS Letter on the Senior Management Meeting Process, September 11, 1997.

PROPOSED REVISIONS TO 10 CFR 50.59
(Changes, Tests and Experiments)

MR. JOHN BARTON
DR. THOMAS KRESS
ACRS

BACKGROUND

- April 1997: ACRS review of SECY-97-035 - Committee report dated April 8, 1997

- October 1997: ACRS review of SECY-97-205 - Committee report dated October 9, 1997

- December 1997: ACRS review of proposed rulemaking for 10 CFR 50.59 - Committee report dated December 12, 1997

- December 1997: ACRS review of interim guidance for updating FSARs - Office memorandum to EDO dated December 10, 1997.

- February 1998: ACRS update briefing on status on Commission action on SECY-97-205.

ACRS REPORT - APRIL 8, 1997

- o Proposed guidance in SECY-97-035 should not be issued for public comment. The staff should work with industry to consider a possible version of NSAC-125 or draft guideline NEI 96-07 that may be sufficient to address concerns over 10 CFR 50.59 implementation.

- o EDO letter dated May 5, 1997, informed Committee that the Commission had approved issuing SECY-97-035 for comment.

ACRS REPORT - OCTOBER 9, 1997

- Issue proposed Revision 1 to Generic Letter 91-18, since it clarifies the applicability of the 10 CFR 50.59 evaluation process to address degraded and nonconforming conditions.

- Because the current legal interpretation of 10 CFR 50.59 is at variance with past staff and industry practices, rulemaking appears to be necessary.

- Continue to work with the NEI to reconcile NEI 96-07 with the staff's position rather than developing separate guidance for implementing the 10 CFR 50.59 process.

- Continue development of a plan for the 10 CFR 50.59 process that is consistent with risk-informed, performance-based regulation.

ACRS REPORT - DECEMBER 12, 1997

- Two-step process is appropriate because of need for stabilization of the 10 CFR 50.59 process. However, the constraint of "zero increase" will exacerbate excessive staff resources being required to review a large number of changes that are risk insignificant.

- Development of a risk-informed rule should be continued on an expedited basis in the second phase. Rule should eliminate the "zero increase" criteria and take the position that qualifying changes have effects on risk that are too small to require quantification of either the magnitude or direction of change.

- Because PRAs will be insensitive to changes made under 10 CFR 50.59, it will be challenging to develop performance criteria for guidance to licensees. Performance criteria should be rooted in the concepts of very small risk effects and compatible with the proposed Regulatory Guide 1.174.

ACRS PLANS

The Committee plans to review:

- o The revised version of proposed 10 CFR 50.59 rulemaking when available.
- o Proposed final Generic Letter for updating FSARs after reconciliation of public comments.
- o Reconciliation of guidance proposed by the industry and the staff position related to these matters.

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

December 12, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED REVISIONS TO 10 CFR 50.59 (CHANGES, TESTS AND EXPERIMENTS)

During the 447th meeting of the Advisory Committee on Reactor Safeguards, December 3-6, 1997, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss proposed revisions to 10 CFR 50.59 (Changes, Tests and Experiments). We also discussed the proposed alternate rulemaking language proposed by NEI and guidance contained in NEI 96-07, Revision 0A, "Guidelines for 10 CFR 50.59 Safety Evaluations." We had the benefit of the documents referenced.

As a first step, the staff proposes a set of revisions to 10 CFR 50.59 that would clarify the current rule with respect to: (1) the criteria to be used to determine what constitutes an "unreviewed safety question" and (2) the language that requires "zero increase" in probability and consequences. The staff's stated intent is to continue developing a second-phase rule that would make the 10 CFR 50.59 process more risk-informed.

We support this two-step process because we agree with the staff and industry that there is an urgent need for stabilization of the 10 CFR 50.59 process. The proposed phase one revisions to the rule can provide interim stabilization. However, we believe the constraint of "zero increase" in the proposed revisions will serve to exacerbate the problem of excessive staff resources being required to review a large number of changes that are risk insignificant.

Therefore, in the second phase, we urge that the development of a new risk-informed rule be continued on an expeditious schedule. This rule should eliminate the "zero-increase" criteria and, instead, take as a starting point the

position that qualifying changes have effects on risk that are considered too small to require quantification of either the magnitude or the direction of change. Because probabilistic risk assessments (PRAs) will be insensitive to the types of changes made under 10 CFR 50.59, it will be challenging for the staff to develop a set of performance criteria for guidance to licensees for determining what changes qualify for consideration within the revised 10 CFR 50.59 process. It is essential that such performance criteria, rooted in the concepts of very small risk effects and compatibility with the proposed Regulatory Guide 1.174 (formerly DG-1061) process, be developed as guidance for implementing the risk-informed rule.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

References:

1. Memorandum dated November 24, 1997, from Jack W. Roe, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, transmitting summary information on 10 CFR 50.59 Rulemaking for December 4, 1997, meeting with ACRS.
2. U. S. Nuclear Regulatory Commission, Proposed Rule, 10 CFR Part 50 (and Part 60, 72, and 76), Changes, Tests and Experiments, dated November 6, 1997.
3. Nuclear Energy Institute, NEI 96-07, Draft Revision 0A, "Guidelines for 10 CFR 50.59 Safety Evaluations," July 1997.
4. Report dated October 9, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: "Proposed Changes to 10 CFR 50.59 and Proposed Revision 1 to Generic Letter 91-18."
5. Report dated April 8, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: "Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments)."

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

December 10, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: PROPOSED GENERIC LETTER ON INTERIM GUIDANCE FOR UPDATING FINAL SAFETY ANALYSIS REPORTS

During the 447th meeting of the Advisory Committee on Reactor Safeguards, December 3-6, 1997, the Committee considered the proposed NRC Generic Letter 97-XX, "Interim Guidance Regarding Updating Final Safety Analysis Reporting in Accordance with 10 CFR 50.71(e)." The Committee decided to continue its review of this matter following the reconciliation of public comments. The Committee looks forward to working with the staff in evaluating proposed industry guidance and in developing proposed final NRC guidance.

References:
1. Proposed NRC Generic Letter 97-XX, "Interim Guidance Regarding Updating Final Safety Analysis Reporting in Accordance with 10 CFR 50.71(e)," received November 6, 1997.
2. Letter dated November 14, 1997, from Anthony R. Pietrangelo, NEI, to Jack W. Roe, NRR, Subject: "Draft Industry Update Guidelines for Final Safety Analysis Reports."

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
J. Roe, NRR
T. Martin, AEOD
M. Knapp, RES

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

October 9, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED CHANGES TO 10 CFR 50.59 AND PROPOSED REVISION 1 TO GENERIC LETTER 91-18

During the 445th meeting of the Advisory Committee on Reactor Safeguards, October 2-3, 1997, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss SECY-97-205, "Integration and Evaluation of Results From Recent Lessons-Learned Reviews," which includes proposed changes to 10 CFR 50.59 (Changes, Tests and Experiments) and Revision 1 to Generic Letter 91-18, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions." We also discussed the proposed industry guidance document NEI 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations." We had the benefit of the d--uments referenced.

Conclusions and Recommendations

1. We recommend that the NRC issue Revision 1 to Generic Letter 91-18, since it explicitly clarifies the applicability of 10 CFR 50.59 evaluation process to address degraded and nonconforming conditions.

2. Because the current legal interpretation of 10 CFR 50.59 is at variance with past staff and industry practices, rulemaking appears to be necessary.

3. The staff should continue to work with NEI to reconcile NEI 96-07 with the staff's position rather than developing

separate guidance for implementing the 10 CFR 50.59 process. We recommend that the NRC endorse this industry approach with appropriate exceptions and clarifications.

4. We encourage the continued development of a plan for a 10 CFR 50.59 process that is consistent with risk-informed, performance-based regulation.

Discussion

In our April 8, 1997 report to the Commission, we recommended that the proposed guidance related to implementation of 10 CFR 50.59, as described in SECY-97-035, not be issued for public comment. Instead, we recommended that the NRC work with the industry to build on the guidance contained in NSAC-125. Our recommendation was based on consideration of over 30 years of industry experience, during which the staff identified problems in only a very small number of situations evaluated under 10 CFR 50.59.

Because the legal interpretation of 10 CFR 50.59 is at variance with past staff and industry practices, rulemaking appears to be necessary. However, rather than developing new regulatory guidance to support the current rule, the staff should issue a safety evaluation report or regulatory guide endorsing the guidance in the revised NEI 96-07 document. Any provisions in NEI 96-07 that the staff finds unacceptable could be identified as exceptions to NRC's acceptance of the industry guidance. This would be similar to past NRC practices of endorsing industrial standards subject to certain exceptions and clarifications.

The debate spawned by the proposed changes to 10 CFR 50.59 is indicative of the need to accelerate the move to risk-informed, performance-based regulation. The current 10 CFR 50.59 requirements already implement a form of this regulatory philosophy but at a very detailed level and in a manner that is inconsistent with current risk-management technology. Ideally, the performance requirements would be identified at a system or function level, and the licensees would have flexibility to manage the plants so long as these performance requirements are met (i.e., they stay within the defined envelope). Defining such performance requirements in advance would eliminate the present disagreements over whether "small" or "zero" risk increases are allowed.

The staff outlined a plan designed to enhance NRC oversight of licensee activities and to improve the existing regulatory process during the transition period to a more risk-informed, performance-based regulatory framework. In the interim, the industry needs to know whether it has a method acceptable to the NRC for performing proper safety evaluations per 10 CFR 50.59. We were informed by representatives of NEI that the industry is currently reviewing NEI 96-07, Revision 0, and that it is expected licensees will uniformly accept this guidance for performing safety evaluations.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

References:

1. SECY-97-205, Memorandum dated September 10, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, for the Commissioners, Subject: Integration and Evaluation of Results from Recent Lessons-Learned Reviews.
2. Draft NRC Generic Letter 91-18, Revision 1, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," September 1997.
3. SECY-97-035, Memorandum dated February 12, 1997, from Hugh L. Thompson, Jr., Acting Executive Director for Operations, NRC, for the Commissioners, Subject: Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments).
4. U.S. Nuclear Regulatory Commission, Draft NUREG-1606, "Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments)," April 1997.
5. Letter dated July 21, 1997, from Ralph E. Beedle, Nuclear Energy Institute, to Frank J. Miraglia, Jr., NRC, regarding NEI 96-07, Final Draft, Subject: Guidelines for 10 CFR 50.59 Safety Evaluations.
6. Report dated April 8, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed

Regulatory Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments).

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

April 8, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED REGULATORY GUIDANCE RELATED TO IMPLEMENTATION OF 10 CFR 50.59 (CHANGES, TESTS AND EXPERIMENTS)

During the 440th meeting of the Advisory Committee on Reactor Safeguards, April 3-4, 1997, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding SECY-97-035, "Proposed Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments)." We discussed the staff's approach to clarifying guidance for implementing 10 CFR 50.59 and proposed options for resolving policy issues.

Conclusions and Recommendations

We recommend that SECY-97-035, as now formulated, not be issued for public comment. We recommend, instead, additional NRC and industry interaction regarding this matter before the proposed guidance in SECY-97-035 is issued for public comment.

Discussion

The industry and staff have over 30 years of experience in implementing 10 CFR 50.59. Over this time, the staff has identified concerns in only a small subset of situations evaluated under 10 CFR 50.59. In SECY-97-035, the staff stated the following with regard to the current process and industry implementation of NSAC-125:

> Although the staff has not endorsed NSAC-125, it has concluded, as discussed in the April 15, 1996, memorandum from James M. Taylor to Chairman Jackson, that NSAC-125 has given the nuclear power industry a reasonable foundation to establish a process that will, in most instances, produce effective evaluations related to changes to plant design or procedures. Changes of significance are highly likely to be identified by the licensee through implementation of the NSAC-125 guidance.

Inspection results have confirmed that the quality of the evaluations of changes has improved since licensees began implementing the NSAC-125 guidance. However, the NSAC-125 guidance is not a requirement for any licensee, and each licensee develops its own program for performing the required evaluations under 10 CFR 50.59.

The staff also found that difficulties arise in the licensee's day-to-day use of the 10 CFR 50.59 process when the staff and licensee have a different understanding and different expectations for implementation of the rule. The staff, therefore, is proposing additional regulatory guidance in SECY-97-035 to reduce the potential for deficiencies in implementing 10 CFR 50.59. Since the staff appears to agree that when the NSAC-125 guidance has been implemented properly it has generally resulted in satisfactory safety evaluations, it would seem more effective to work with the industry to build on NSAC-125. The goal would be for the staff to endorse an appropriate version of NSAC-125 with exceptions, as needed. It is our understanding that the industry has attempted to improve on NSAC-125 through the development of draft guideline NEI 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations." These improvements may well address many of the present concerns.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

References:
1. SECY-97-035, Memorandum dated February 12, 1997, from H. L. Thompson, Jr., Acting Executive Director for Operations, NRC, for the Commissioners, Subject: Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments).
2. Memorandum dated April 15, 1996, from James M. Taylor, Executive Director for Operations, NRC, to Shirley Ann Jackson, Chairman, NRC, Subject: Action Plan for Improvements to 10 CFR 50.59 Implementation and Oversight.
3. Electric Power Research Institute, Nuclear Safety Analysis Center, NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations," June 1989.
4. Nuclear Energy Institute, NEI 96-07, draft Revision A, "Guidelines for 10 CFR 50.59 Safety Evaluations," July 1996.

RISK-INFORMED, PERFORMANCE-BASED REGULATION, INCLUDING USE OF PRA IN THE REGULATORY DECISIONMAKING PROCESS

DR. GEORGE APOSTOLAKIS
DR. THOMAS KRESS
ACRS

BACKGROUND

Standard Review Plan and Regulatory Guides

- o Committee report dated July 14, 1997
- o Committee report dated December 11, 1997
- o Committee report dated March 12, 1998

Treatment of Uncertainties Versus Point Values

- o Committee report dated December 16, 1997

Plant-Specific Application of Safety Goals

- o Committee report dated September 19, 1997

Elevation of CDF to a fundamental safety goal and possible revision to the Commission's Safety Goal Policy Statement

- o Review Continuing

DRAFT SRP & REGULATORY GUIDE FOR ISI
ACRS REPORT - JULY 14, 1997

- The approach described in the subject documents will lead to substantial improvements in ISI of piping and should be issued for public comment.

- ACRS review of the proposed final version of the Regulatory Guide (RG) and associated Standard Review Plan (SRP) documents is scheduled for April 2-4, 1998.

PROPOSED FINAL RG 1.174 & SRP CH. 19
ACRS REPORT - DECEMBER 11, 1997

- Reg. Guide 1.174 and associated SRP Chapter 19 be approved and issued for use by the industry and staff. Modification of acceptance guidelines to allow consideration of very small increases in CDF and LERF for a broader range of the total CDF and LERF values is appropriate.

- Decisionmaking process in RG 1.174 and its treatment of uncertainties is sound. Staff has correctly focused on identifying the sources of uncertainty and determining impact on decisions, rather than using final distributions as the sole basis for decisionmaking.

- Discussion on PRA quality in these documents is appropriate. Assessment of the scope/quality of probabilistic analyses should focus on whether they are adequate for the purpose intended.

SRP & RGs FOR IST, GQA, AND TS
ACRS REPORT - MARCH 12, 1998

- o Reg. Guides 1.175 (IST), 1.176 (GQA), and 1.177 (TS) and associated SRP sections should be issued for use.

- o Reg. Guide 1.176 does not take full advantage of the information that PRA provides. The lack of a model for assessing the quantitative impact of QA requirements on PRA parameters makes this a difficult document to write. RES should consider a research project to assess the impact of QA requirements on PRA parameters.

- o The staff should prepare a plan for improving RG 1.176 after experience with its application and related studies and brief the Committee in the next two years. Urged expeditious closure on the risk-informed pilots for changes to the CLB.

PRINCIPLES OF RISK-INFORMED REGULATION

- o The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.
- o The proposed change is consistent with the defense-in-depth philosophy.
- o The proposed change maintains sufficient safety margins.
- o When proposed changes result in an increase in core damage frequency and/or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
- o The impact of the proposed change should be monitored using performance measurement strategies.

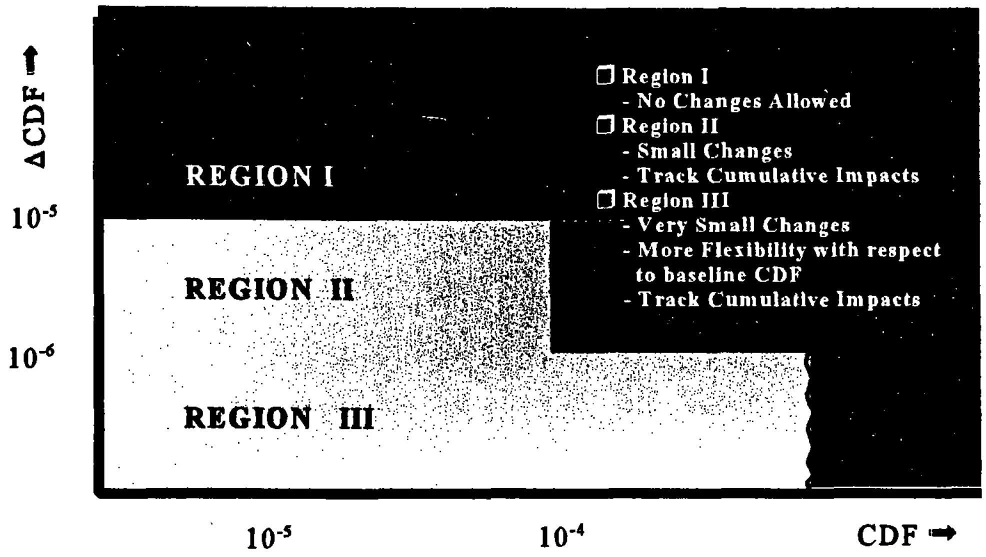

Figure 3 - Acceptance Guidelines* for Core Damage Frequency (CDF)

* The analysis will be subject to increased technical review and management attention as indicated by the darkness of the shading of the figure. In the context of the integrated decision making, the boundaries between regions should not be interpreted as being definitive; the numerical values associated with defining the regions in the figure are to be interpreted as indicative values only.

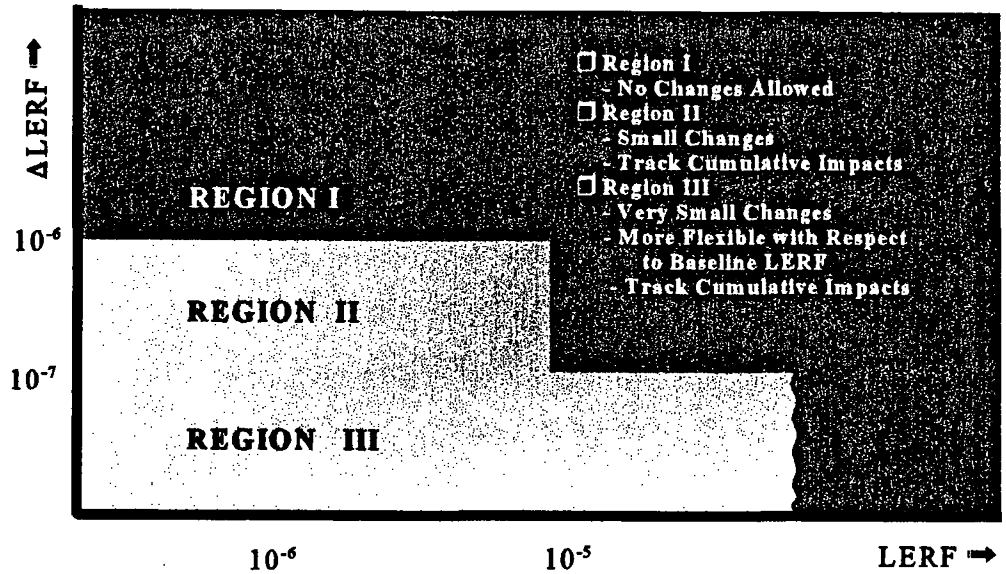

Figure 4 - Acceptance Guidelines* for Large Early Release Frequency (LERF)

* The analysis will be subject to increased technical review and management attention as indicated by the darkness of the shading of the figure. In the context of the integrated decision making, the boundaries between regions should not be interpreted as being definitive; the numerical values associated with defining the regions in the figure are to be interpreted as indicative values only.

UNCERTAINTIES VS. POINT VALUES
ACRS REPORT - DECEMBER 16, 1997

- o When PRA results and insights are proposed to be used in the regulatory process, the question should be: To what degree is there confidence that the PRA results and insights will improve on the existing regulatory system for the problem of interest?

- o The Bayesian interpretation of probability provides the appropriate framework for PRA. Probability distributions for the parameters of PRA models, e.g., failure rates, should be developed using all available evidence and propagated to produce the probability distribution of the quantity of interest, e.g., CDF and LERF.

- o The only "point estimates" that are unambiguously defined are those that are summary measures of a probability distribution; e.g., the mean value, the median value, and various percentile values.

- o Regulatory decisions must be made in light of all relevant uncertainties. These include uncertainties quantified in PRAs, as well as significant unquantified uncertainties. Although "point" values, defined above, can be useful for screening purposes, they are summary measures of the probability distributions and should not be the sole basis for decisionmaking.

- o The dominant scenarios should be an integral part of the deliberation on uncertainties.

- o The unquantified uncertainties associated with a proposed change to the CLB should include the possible beneficial impact of the proposed change on plant safety.

- o The decisionmaking process described in RG 1.174 treats uncertainties and point values in a manner consistent with the ACRS recommendations included in the December 11, 1997 report.

PLANT-SPECIFIC APPLICATION OF SAFETY GOALS

ACRS REPORT - SEPTEMBER 19, 1997

- o Consideration of siting factors, discussed in DG-1061 (now RG 1.174), should be given much greater visibility and prominence as part of the decisionmaking process.

- o There is insufficient site-to-site variability in the factors that influence individual early fatality risk to warrant site-specific differences in the LERF subsidiary criterion.

- o Large site-to-site variations in population density result in large variations in total early fatality risk. This robust indicator of societal risk should be made more explicit and prominent in the criteria to be used in assessing plant specific changes to the CLB.

ELEVATION OF CDF TO A FUNDAMENTAL SAFETY GOAL AND POSSIBLE REVISION TO SAFETY GOALS

- August 15, 1996: ACRS recommended elevation of CDF to a fundamental safety goal.
- July 2, 1997: Chairman Jackson memorandum to the EDO regarding the use of CDF as a fundamental safety goal.
- July 23, 1997: NEI letter objecting to elevation of CDF.
- October 8, 1997: Commissioner Diaz request for ACRS review.
- October 16, 1997: SRM requesting staff recommendations.
- March 2-4, 1998: ACRS review; April 2-4 review draft response to SRM; RPRA Subcommittee April 16 to consider matters in detail.

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

July 14, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED REGULATORY GUIDE AND STANDARD REVIEW PLAN CHAPTER FOR RISK-INFORMED, PERFORMANCE-BASED INSERVICE INSPECTION

During the 443rd meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 1997, we met with representatives of the NRC staff to review the proposed Regulatory Guide DG-1063 and Standard Review Plan (SRP) Chapter 3.9.8 for risk-informed, performance-based inservice inspection. Our Subcommittee on Probabilistic Risk Assessment also met on July 8, 1997 with the staff, industry representatives, and other interested parties to discuss these documents and industry initiatives. We also had the benefit of the documents referenced.

We believe that the approach described in proposed SRP Chapter 3.9.8 and Regulatory Guide DG-1063 will lead to substantial improvements in inservice inspection for piping. In response to our comments, the staff identified changes it plans to make to these documents before they are issued for public comment. We recommend that these documents be issued for public comment subject to incorporation of those changes. The staff also proposed a list of questions regarding issues that arose during our meetings, which it plans to include in the Federal Register notice to solicit public comments. We agree with these questions.

Dr. Dana Powers did not participate in the Committee's deliberations regarding draft Regulatory Guide DG-1063.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

References:

Memorandum dated June 3, 1997, from B. Sheron, NRR, M. W. Hodges, RES, L. C. Shao, RES, G. Holahan, NRR to J. Larkins, ACRS, Subject: Transmittal of Pre-decisional Draft Regulatory Guide DG-1063: "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Inspection of Pipes," and Draft Standard Review Plan Chapter 3.9.8, "Standard Review Plan for the Review of Risk-Informed Inservice Inspection Applications."

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

December 11, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED FINAL REGULATORY GUIDE 1.174 AND STANDARD REVIEW PLAN CHAPTER 19 FOR RISK-INFORMED, PERFORMANCE-BASED REGULATION

During the 446th and 447th meetings of the Advisory Committee on Reactor Safeguards, November 6-7 and December 3-6, 1997, respectively, we met with representatives of the NRC staff to review proposed final Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," and Standard Review Plan (SRP) Chapter 19 (General Guidance) for risk-informed, performance-based regulation. We discussed the staff's reconciliation of public comments on the subject documents, including proposed changes to address policy issues under consideration by the Commission. Our Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) met with the staff and industry representatives on October 21-22 and November 12-13, 1997, to discuss these matters. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. We recommend that Regulatory Guide 1.174 and associated Standard Review Plan Chapter 19 be approved and issued for use by the industry and staff.

2. The modification of the acceptance guidelines to allow consideration of very small increases in CDF (core damage frequency) and LERF (large, early release frequency) for a broader range of the total CDF or LERF values is appropriate.

3. The decisionmaking process described in Regulatory Guide 1.174 and, in particular, its treatment of quantified and unquantified uncertainties, is sound. The staff has correctly focused on identifying the important sources of uncertainty and determining their impact on decisions, rather

than simply using the final distribution as the sole basis for decisionmaking.

4. The staff discussion of PRA quality in these documents is appropriate. We agree with the staff position that the assessment of the scope and quality of the probabilistic analyses should focus on whether they are adequate for the purpose intended.

As we stated in our report dated March 17, 1997, we believe that this new process and these documents are a significant achievement that will contribute to the safe and efficient use of nuclear power.

We believe that these documents will evolve as experience is gained. We again urge the staff to seek innovative applications of the risk-informed approach to regulation so that this Regulatory Guide and the associated Standard Review Plan Chapter will be tested and improved upon in practice. We request the staff to brief the Committee periodically on this regulatory activity.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

References:

1. Memorandum dated November 24, 1997, from M. Wayne Hodges, Office of Nuclear Regulatory Research, NRC, and Gary M. Holahan, Office of Nuclear Reactor Regulation, NRC, to John Larkins, ACRS, Subject: "General Regulatory Guide (DG-1061) and Standard Review Plan (SRP-Chapter 19) for Risk Informed Regulatory Decisionmaking for Plant Specific CLB Changes," with attachments, as follows:

   - Proposed Final Regulatory Guide 1.174 (Draft Guide DG-1061) dated November 25, 1997, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis."

   - Proposed Final Standard Review Plan Chapter 19, Revision N, dated November 25, 1997, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance."

2. Draft SECY dated November 7, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, for the Commissioners, "Final Regulatory Guidance on Risk-Informed Regulation: Policy Issues."

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

March 12, 1998

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

Dear Chairman Jackson:

SUBJECT: PROPOSED FINAL STANDARD REVIEW PLAN SECTIONS AND REGULATORY GUIDES FOR RISK-INFORMED, PERFORMANCE-BASED REGULATION FOR INSERVICE TESTING, GRADED QUALITY ASSURANCE, AND TECHNICAL SPECIFICATIONS

During the 449th meeting of the Advisory Committee on Reactor Safeguards, March 2-4, 1998, we met with representatives of the NRC staff to review proposed final Standard Review Plan (SRP) sections and regulatory guides for risk-informed, performance-based regulation including individual applications for inservice testing, graded quality assurance, and technical specifications. We discussed the staff's reconciliation of public comments on the subject documents. Our Subcommittee on Reliability and Probabilistic Risk Assessment met with the staff and industry representatives on February 19, 1998, to discuss these matters. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. We recommend that Regulatory Guides 1.175 (Inservice Testing), 1.176 (Graded Quality Assurance), and 1.177 (Technical Specifications) and associated SRP sections be approved and issued for use.

2. We do not believe that Regulatory Guide 1.176 takes full advantage of the information that probabilistic risk assessment (PRA) provides. We recognize, however, that the lack of a model for assessing the quantitative impact of quality assurance requirements on PRA parameters makes this a particularly difficult document to write.

3. We recommend that the Office of Nuclear Regulatory Research consider a research project to assess the impact of quality assurance requirements on PRA parameters.

4. We recommend that the staff prepare a plan for improvements to Regulatory Guide 1.176 after experience with its application and related studies and brief the Committee sometime in the next two years.

As stated in our previous reports, we believe that the next major step in the process will be the use of these documents in practice. We urge the staff to move expeditiously to reach closure on the pilot risk-informed requests for changes to the current licensing basis that are currently under review. We were pleased to hear a presentation from the Nuclear Energy Institute on the new risk-informed initiative that it is sponsoring. We plan to follow developments in these activities with great interest.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

References:

1. U.S. Nuclear Regulatory Commission, proposed final SRP Section 3.9.7, "Risk-Informed Inservice Testing," draft dated March 2, 1998 (Predecisional).
2. U.S. Nuclear Regulatory Commission, proposed final Regulatory Guide 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," draft dated March 2, 1998. (Predecisional)
3. U.S. Nuclear Regulatory Commission, proposed final SRP Chapter 16.1, "Risk-Informed Decisionmaking: Technical Specifications," draft dated March 2, 1998 (Predecisional).
4. U.S. Nuclear Regulatory Commission, proposed final Regulatory Guide 1.176, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance," draft dated March 2, 1998 (Predecisional).
5. U.S. Nuclear Regulatory Commission, proposed final Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," draft dated March 2, 1998 (Predecisional).
6. Report dated March 17, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Standard Review Plan Sections and Regulatory Guides for Risk-Informed, Performance-Based Regulation.
7. Report dated December 11, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Final Regulatory Guide 1.174 and Standard Review Plan Chapter 19 for Risk-Informed, Performance-Based Regulation.

8. Memorandum dated October 30, 1997, from John C. Hoyle, Secretary of the Commission, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements Memorandum - SECY-97-229, "Graded Quality Assurance/Probabilistic Risk Assessment Implementation Plan for the South Texas Project Electric Generating Station."
9. Memorandum dated May 28, 1997, from John C. Hoyle, Secretary of the Commission, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements Memorandum - SECY-97-095, "Probabilistic Risk Assessment Implementation Plan Pilot Application for Risk-Informed, Performance-Based Regulation."

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

December 16. 1997

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

Dear Chairman Jackson:

SUBJECT: TREATMENT OF UNCERTAINTIES VERSUS POINT VALUES IN THE PRA-RELATED DECISIONMAKING PROCESS

During the 443rd. 444th. 446th. and 447th meetings of the Advisory Committee on Reactor Safeguards. July 9-11. September 3-5. November 6-7. and December 3-6. 1997. respectively. we met with representatives of the NRC staff to discuss issues included in the Staff Requirements Memorandum dated May 27. 1997. regarding the use of uncertainty versus point values in the PRA-related decisionmaking process (Reference 1). Our Subcommittee on Reliability and Probabilistic Risk Assessment (RPRA) met with the staff and industry representatives to discuss these matters on July 7. August 28. October 21-22. and November 12-13. 1997.

Background

Uncertainty has always been of concern to nuclear power regulators. As early as 1956. Willard F. Libby. Acting Chairman of the Atomic Energy Commission (AEC). wrote to the Congressional Joint Committee on Atomic Energy that "it is incumbent upon the new industry and the Government to make every effort to recognize every possible event or series of events which could result in the release of unsafe amounts of radioactive material to the surroundings and to take all steps necessary to reduce to a reasonable minimum the probability that such events will occur in a manner causing serious overexposure to the public." (Reference 2)

Even though Dr. Libby used the word "probability." about 20 years would pass before systematic calculations of probabilities would be produced for the "possible event or series of events" to which he referred. The "reasonable minimum" of the unquantified probability that was achieved at that time was attained through the development and application of the concepts of defense in depth and safety margins.

Defense in depth is advocated in numerous documents as the principal means of controlling the (still unquantified) probability of accidents. For example, during the 1971 hearings on emergency core cooling, the AEC staff stated: "The safety goal, therefore, is the prevention of exposure of people to this radioactivity. This goal can be achieved with a high degree of assurance, *although not perfectly* [emphasis added], by use of the concept of defense in depth...The three separate lines of the defense in depth provided for power reactors are considered appropriate to reduce to an acceptable value the probability and potential consequences of radioactive releases." (Reference 3)

Although the approaches of defense in depth and safety margins have served the industry well from the safety perspective, they were intended to be conservative and, as implemented today, they impose a heavy regulatory burden. The level of safety was not quantified. The first call for a more rational approach to regulation based on improved understanding of risk came in 1967 from F. Reginald Farmer (Reference 4) of the United Kingdom Atomic Energy Authority. The Reactor Safety Study (WASH-1400) (Reference 5) soon followed in 1975. Not surprisingly, the WASH-1400 study itself proved to be conservative in some areas, e.g., the analysis of the containment, and nonconservative in others, e.g., the analysis of earthquakes and fires. There has been tremendous progress in our understanding of the risks from nuclear power plants since that study (a history of PRA developments since WASH-1400 is given in Reference 6).

Realizing that the availability of risk numbers made it possible to reexamine the question of how safe is safe enough, the Commission issued the safety goal policy in 1986 (Reference 7). The recognition that uncertainties had to be dealt with is reflected in the following three statements from the policy statement:

<u>Statement I</u>: "It is the Commission's intent that the risks from all the various initiating mechanisms be taken into account to the best of the capability of current evaluation techniques."

<u>Statement II</u>: "To the extent practicable, the Commission intends to ensure that the quantitative techniques used for regulatory decisionmaking take into account the potential uncertainties that exist so that an estimate can be made on the confidence level to be ascribed to the quantitative results."

<u>Statement III</u>: "The Commission has adopted the use of mean estimates for the purposes of implementing the quantitative objectives of this safety goal policy...."

The Commission's safety goals were derived from societal considerations, i.e., independent of the PRA state of the art. Even though they were expressed both

qualitatively and quantitatively, it was clear that the Commission did not intend to simply compare a PRA "point estimate" (however it was defined) with the numerical goals.

The Issue

As noted above, the numerical estimates that PRAs produce have been scrutinized to an extraordinary degree since the early days of WASH-1400. Sometimes the debate regarding the accuracy of these numbers detracts from the intended use of PRA.

It is not the intent to regulate on the basis of risk estimates alone (thus, "risk-informed" regulation). The objective is to gain enough confidence in the numerical probabilities of a set of accident scenarios so that the traditional approaches (defense in depth and safety margins) that have already been applied to this set can be better managed. This means either relaxing some existing requirements, if proven burdensome and non-contributing to risk reduction, or adding new requirements, if the traditional approaches have not covered some detrimental events.

The preceding discussion suggests that the question regarding the quality of PRA results ought not to be an absolute one, but, rather, a comparative one. Therefore, we offer the following observation:

> Observation 1:
>
> When PRA results and insights are proposed to be used in the regulatory process, the question to be asked should be: To what degree is there confidence that the use of PRA results and insights will improve on the existing regulatory system for the problem of interest?

The words "PRA results and insights" include the set of dominant scenarios to risk (or core damage, as the case may be), as well as an assessment of the uncertainties regarding the frequencies of these scenarios. The utilization of PRA results and insights depends on our confidence that their use will improve the regulations in accordance with the Commission's vision. It is definitely not a case of PRA versus the traditional approach.

In Observation 1, the key words are "will improve." There is improvement when the regulations contribute to the safe and efficient use of nuclear materials, as per the recently articulated vision of the Commission: "In implementation of its mission, Nuclear Regulatory Commission actions enable the Nation to safely and efficiently use nuclear materials." (Reference 8)

Uncertainties

As our brief historical review has demonstrated, the uncertainties regarding off-normal events and incidents in nuclear power plants have been of concern since the early days of reactor regulation. In the early seventies, quantifying the uncertainties was synonymous with developing probability distributions for the failure rates and the frequencies of accident initiators. This explicit quantification of uncertainties posed a new problem to safety analysts. They soon discovered that the interpretation of the concept of probability was controversial among mathematicians. Several schools of thought were available, of which the frequentist and the Bayesian schools were dominant. When the nuclear debate was heating up in the mid-seventies, the analysts were reluctant to get involved in an additional controversy.

This attitude, although understandable in the context of the times, was unfortunate, because it led to confusion and the perception that uncertainty analysis was controversial and to be avoided. It also led to some circumlocutions. For example, the WASH-1400 treatment of failure rates is purely Bayesian, yet that voluminous report does not acknowledge this fact explicitly. Similarly, the NUREG-1150 studies (Reference 9) claimed to elicit "weighting factors" from the experts, rather than admit that they were eliciting probabilities. Although "officially," both frequentist and Bayesian viewpoints were equally valid, no PRA had been done using frequentist methods because it cannot be done. Industry-sponsored PRAs, however, have readily acknowledged using Bayesian methods in an explicit way (Reference 10).

It is now known that uncertainties in failure rates and other parameters appearing in PRA can be quantified via probability distributions using available generic and plant-specific data and appropriate Bayesian methods. The propagation of these distributions through the PRA logic diagrams is straightforward using standard computer packages. We believe that there is no excuse for failing to do an uncertainty analysis on the parameters of the PRA models. Therefore, we offer the following observation:

Observation 2:

The Bayesian interpretation of probability provides the appropriate framework for PRA. Probability distributions for the parameters of PRA models, e.g., failure rates, should be developed using all available evidence and propagated to produce the probability distribution of the quantity of interest, e.g., core damage frequency (CDF) and large, early release frequency (LERF).

Since regulators must confront uncertainties, it is evident that, if PRA is to be used as in our Observation 1, the probability distributions of Observation 2 must be derived. Anything less does not represent what is actually known about these failure rates. This brings up the issue of "point estimates," for which we offer the following observation:

> Observation 3:
>
> The only "point estimates" that are unambiguously defined are those that are summary measures of a probability distribution; e.g., the mean value, the median value, and various percentile values.

Ill-defined "point estimates," such as "best estimates," have limited utility. Point estimates are valuable for screening purposes after a convincing case has been made that the uncertainties have been handled appropriately, e.g., they are either negligible or have been bounded. In fact, the use of such point values is an important tool in screening the thousands of minimal cut sets that a PRA produces. Such use, however, should be followed by a rigorous uncertainty analysis of the dominant sequences.

The uncertainties of interest in reactor regulation have been termed "state-of-knowledge" uncertainties (Reference 11) or, more recently, "epistemic" uncertainties (References 12, 13). The parameter uncertainties that are referred to in Observation 2 are only a part of the total epistemic uncertainties. Uncertainties resulting from model assumptions and approximations are also epistemic and more difficult to quantify. Examples would include models used for evaluating severe accident phenomena in Level II PRAs.

Model uncertainty is the key to any use of PRA results. When events or processes are modeled poorly or not at all, there is uncertainty that has not been quantified, in the sense that it is not part of the probability distributions produced by propagating parameter uncertainties. The fact that uncertainty is not quantified does not mean, however, that nothing is known about it. The PRA structure provides a good framework within which these uncertainties can be assessed qualitatively through sensitivity analyses or other means (see, for example, Reference 14). These uncertainties exist independently of whether or not they are quantified in PRAs. Recalling Observation 1, use of PRA insights must include a qualitative description of unquantified uncertainties, in addition to those that have been quantified. Any PRA-based argument for easing the regulatory requirements of the traditional approach is weakened when the unquantified uncertainties are very large and pertinent to the application. Therefore, we offer the following observation:

Observation 4:

Regulatory decisions must be made in the light of all the relevant uncertainties. These include the uncertainties quantified in PRAs, as well as significant unquantified uncertainties. Although "point" values, defined as in Observation 3, can be useful for screening purposes, they are summary measures of the probability distributions and should not be the sole basis for decisionmaking.

The deliberation on uncertainties that we are recommending is best accomplished by considering the scenarios that dominate the event of interest. The set of dominant scenarios is one of the most important results of PRA and has been proven to be very useful in risk management (Reference 15). A discussion of the overall uncertainties without a discussion of the sources of uncertainties is of limited value. Thus, we offer the following observation:

Observation 5:

The dominant scenarios should be an integral part of the deliberation on uncertainties.

The regulatory decisions of immediate interest are those related to requests for changes in the current licensing basis (CLB). In discussing uncertainties, it is important to consider possible benefits of the proposed change. For example, a change that reduces the regulatory burden in certain areas could allow the reallocation of resources to more risk significant issues and activities. Therefore, we offer the following observation:

Observation 6:

The unquantified uncertainties associated with a proposed change in the CLB should include the possible beneficial impact of the proposed change on plant safety.

Finally, we note that the decisionmaking process described in Regulatory Guide 1.174 treats uncertainties and point values in a manner consistent with our

recommendations as discussed in our report dated December 11, 1997. (Reference 16)

Sincerely,

R. L. Seale
Chairman

References:

1. Staff Requirements Memorandum dated May 27, 1997, from John C. Hoyle, Secretary of the Commission, to John T. Larkins, ACRS, Subject: Meeting with Advisory Committee on Reactor Safeguards on Friday, May 2, 1997.

2. Letter dated March 14, 1956, from Willard F. Libby, Acting Chairman of the Atomic Energy Commission, to Senator B. Hickenlooper, Joint Committee on Atomic Energy, reproduced in: D. Okrent, *Nuclear Reactor Safety*, The University of Wisconsin Press, Madison, 1981.

3. Testimony of the AEC Regulatory Staff at a Public Rulemaking Hearing on Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Reactors on January 27, 1972, Issued December 28, 1971, U. S. Atomic Energy Commission.

4. F. Reginald Farmer, "Reactor Safety and Siting: A Proposed Risk Criterion," *Nuclear Safety*, Vol. 8, No. 6, pp. 539-548, Nov.-Dec. 1967.

5. U. S. Nuclear Regulatory Commission, NUREG-75/014, "Reactor Safety Study, An Assessment -- Accident Risks in U.S. Nuclear Power Plants, WASH-1400," October 1975.

6. Eric S. Beckjord, Mark C. Cunningham, and Joseph A. Murphy, "Probabilistic Safety Assessment Development in the United States 1972-1990," *Reliability Engineering and System Safety*, Vol. 39, pp. 159-170, 1993.

7. U. S. Nuclear Regulatory Commission, "Safety Goals for the Operation of Nuclear Power Plants: Policy Statement," *Federal Register*, Vol. 51, p. 30028, August 21, 1986.

8. U.S. Nuclear Regulatory Commission, *"Strategic Plan: Fiscal Year 1997 - Fiscal Year 2000,"* September 1997.

9. U. S. Nuclear Regulatory Commission, NUREG-1150, *"Severe Accident Risks: An Assessment for Five US Nuclear Power Plants,"* December 1990.

10. Pickard, Lowe, and Garrick, Inc., Westinghouse Electric Corporation, and Fauske & Associates, Inc., *"Zion Probabilistic Safety Study,"* Report prepared for Commonwealth Edison Company, Chicago, 1981.

11. Stanley Kaplan and B. John Garrick, "On the Quantitative Definition of Risk," *Risk Analysis*, Vol. 1, No. 1, pp. 11-28, March 1981.

12. George E. Apostolakis, "A Commentary on Model Uncertainty," in U. S. Nuclear Regulatory Commission, NUREG/CP-0138, *Proceedings of Workshop I in Advanced Topics in Risk and Reliability Analysis: Model Uncertainty, Its Characterization and Quantification*, October 20-22, 1993.

13. Gareth W. Parry, "The Characterization of Uncertainty in Probabilistic Risk Assessments of Complex Systems," *Reliability Engineering and System Safety*, Vol. 54, pp. 119-126, 1996.

14. Dennis Bley, Stanley Kaplan, and David Johnson, "The Strengths and Limitations of PSA: Where We Stand," *Reliability Engineering and System Safety*, Vol. 38, pp. 3-26, 1992.

15. Mardyros Kazarians, Nathan Siu, and George Apostolakis, "Risk Management Application of Fire Risk Analysis," *Proceedings of the First International Symposium on Fire Safety Science*, pp. 1029-1038, C.E. Grant and P.J. Pagni, Editors, Hemisphere Publishing Corporation, New York, 1986.

16. Report dated December 11, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: "Proposed Final Regulatory Guide 1.174 and Standard Review Plan Chapter 19 for Risk-Informed Performance-Based Regulation"

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

September 19, 1997

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

Dear Chairman Jackson:

SUBJECT: SITE-TO-SITE VARIATION IN RISK-BASED REGULATORY ACCEPTANCE CRITERIA FOR PLANT-SPECIFIC APPLICATION OF SAFETY GOALS

In the Staff Requirements Memorandum dated May 27, 1997, the Commission requested that the ACRS determine the change in core damage frequency (CDF) and large, early release frequency (LERF) from site-to-site when these lower-tier criteria are derived from the individual early fatality quantitative health objective (QHO). In response to this Commission request, during the 443rd and 444th meetings of the Advisory Committee on Reactor Safeguards, July 9-11 and September 3-5, 1997, we discussed the plant-specific application of NRC Safety Goals and derivation of subsidiary criteria. These criteria would be used in determining the acceptability of proposed changes to the licensing basis. During the discussions, we had the benefit of the documents referenced.

This report discusses the site variability in LERF as a risk-acceptance criterion derived from the individual early fatality QHO. The bases for the conclusions and recommendations in this report are provided in the attached studies. We addressed the CDF criterion in our April 11, 1997 report.

<u>Variability in LERF Criteria Derived from the Safety Goal Individual Early Fatality QHO</u>

In support of preparing our response to the Commission's request, an ACRS Senior Fellow performed a study (Attachment 1) to answer the following questions:

- Is there sufficient site-to-site variability in the site characteristics important to individual early fatality risk to warrant site-specific determination of lower level acceptance criteria - e.g., LERF?

- Can this range of variability be evaluated and bounded?

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- Can generic criteria or site-specific criteria be determined using simplified approximate methods?

The range of variability in individual early fatality risk due to the site-to-site variations in the parameters important to individual early fatality risk, such as site-to-site population distribution, wind direction frequency distribution, exclusion zone size, and meteorology record, was evaluated for all U.S. plant sites and was found to be relatively small (a variation of a factor of 4).

This study has been independently reviewed, and although the reviewers had different opinions on some of the details of the analysis, all of the reviewers concurred with the overall conclusion on the magnitude of the variability. Since this variability is much less than the magnitude of uncertainties associated with the probabilistic risk assessment (PRA) calculation of the LERF, this study concluded that the site-to-site variability in individual early fatality risk is insufficient to warrant development of site-specific LERF criteria. Hence, a single LERF criterion can be determined on a generic basis. This is consistent with the approach used by the staff in the draft Regulatory Guide DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis."

We believe that the information provided in the study can be used to bound the variability of site-specific LERFs.

Adequacy of Individual Risk Metric

In addition to the individual risk metric, DG-1061 contains deterministic considerations that include other risk parameters — one of which is "siting factors." A second study, which was performed by an ACRS Senior Fellow (Attachment 2), noted that one such siting factor, site population density, is a robust indicator of total (societal) early fatality risk. Consequently, we recommend that the consideration of siting factors, mentioned in DG-1061 only in passing, be given much greater visibility and prominence as part of the decision making process.

Conclusions and Recommendations

We have determined that there is insufficient site-to-site variability in the factors that influence individual early fatality risk to warrant site-specific differences in the LERF subsidiary criterion.

Large site-to-site variations in the population density result in large variations in total early fatality risk. We recommend that this robust indicator of societal risk be made more explicit and

prominent in the criteria to be used in assessing plant-specific changes to the current licensing basis.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

References:

1. Memorandum dated May 27, 1997, from John C. Hoyle, Secretary, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Staff Requirements - Meeting with the ACRS, May 2, 1997, Commissioners' Conference Room.
2. Report dated November 18, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Plant-Specific Application of Safety Goals.
3. Report dated April 11, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Risk-Based Regulatory Acceptance Criteria for Plant Specific Application of Safety Goals.
4. U. S. Nuclear Regulatory Commission, NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," Prepared by Sandia National Laboratories, December 1982.

Attachments:

1. Memorandum dated June 27, 1997, from R. Sherry, Senior ACRS Fellow to ACRS Members, Subject: Considerations for Plant-Specific, Site-Specific Application of Safety Goals and Definition of Subsidiary Criteria.
2. Memorandum dated June 11, 1997, from R. Sherry, Senior ACRS Fellow to ACRS Members, Subject: Consideration of Societal Risk in Plant-Specific, Site-Specific Application of Safety Goals and Definition of Subsidiary Criteria.

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

August 15, 1996

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

Dear Chairman Jackson:

SUBJECT: RISK-INFORMED, PERFORMANCE-BASED REGULATION AND RELATED MATTERS

During the 433rd meeting of the Advisory Committee on Reactor Safeguards, August 8-10, 1996, we discussed the issues identified in the Staff Requirements Memorandum dated May 15, 1996. We also discussed the pilot applications for risk-informed, performance-based regulation. Our Subcommittee on Probabilistic Risk Assessment (PRA) met with representatives of the NRC staff and the nuclear industry on July 18 and August 7, 1996. We also had the benefit of the documents referenced.

The staff presentations dealt only with the development of guidelines from the Commission's safety goals to be used as an element of the evaluation of licensee-initiated changes to licensing commitments. All of our comments address the application of risk-informed regulation in that context. At a later time, we will discuss the larger question of the application of the safety goals on a plant-specific basis.

CONCLUSIONS

Issue 1: *Should the Commission's safety goals and subsidiary objectives be referenced or used to derive guidelines for plant-specific applications and, if so, how?*

We believe the safety goals and subsidiary objectives can and should be used to derive guidelines for plant-specific applications. It is, however, impractical to rely exclusively on the Quantitative Health Objectives (QHOs) for routine use on an individual plant basis. Criteria based on core damage frequency (CDF) and large, early release frequency (LERF) focus more sharply on safety issues and can provide assurance that the QHOs are met. They should be used in developing detailed guidelines.

<u>Issue 2</u>: *How are uncertainties to be accounted for?*

This is a difficult issue. There are models and formal methods to account explicitly for a large number of uncertainties. However, other uncertainties are unquantifiable. The staff proposes to explore a number of options, such as establishing margins in the acceptance guidelines, placing more importance on defense-in-depth, and others, to deal with such uncertainties. Such approaches seem appropriate, although much work remains to be done.

<u>Issue 3</u>: *Should requested changes to the current licensing basis be risk-neutral or should increases be permitted?*

We agree with the staff and industry that increases in risk should be permitted in some situations. Acceptance guidelines expressed in terms of the proposed change in risk and the current risk estimates should have three regions: a region in which some increase in risk is acceptable, one in which it is unacceptable, and one in which further analysis and evaluation would be required.

<u>Issue 4</u>: *How should performance-based regulation be implemented in the context of risk-informed regulation?*

We agree with the staff that, where practical, performance-based strategies should be included in the implementation and monitoring step of the risk-informed decision-making process. The pilot programs may provide an opportunity for a more concrete definition and development of performance-based strategies.

DISCUSSION

<u>Issue 1</u>

Even though a CDF could be derived from the QHOs that could be greater than 10^{-3} per reactor-year, the current subsidiary goal of 10^{-4} per reactor-year should be maintained and should be stated as a fundamental safety goal, along with the QHO. Accident sequences that have a high probability of leading to severe consequences could be controlled by the QHOs, but a more workable measure would be a subsidiary goal on the LERF. The definition of the latter needs to be improved. Whether the LERF should be a fixed value or derived from the QHOs, which would allow the LERF goal to include site-specific characteristics, needs to be investigated.

We recommend that the staff develop guidance for handling situations in which high values of the CDF occur for short periods of time (for example, 10^{-2} per reactor-year for a day).

Issue 2

In accounting for uncertainties, it is important to distinguish between those plant characteristics or phenomena that are modeled in the PRA and those that are not modeled (e.g., the actual layout of components and organizational factors). For those that are modeled, parameter and model uncertainties should be explicitly quantified and propagated through the PRA. The resulting distributions should be an input to the decision-making process along with other qualitative input.

Mean values of distributions should, in general, be used for comparison with goals or criteria, although the sensitivity of the mean value to the high tail of a distribution should not be overlooked. For very broad distributions, such as those that typically result when significant model uncertainty is present, reliance on the mean values may not be appropriate and a more detailed investigation of the reasons for this large uncertainty should be undertaken. This could possibly lead to decisions to conduct additional research or to take other measures.

Accounting for uncertainty in the case of plant characteristics or phenomena that are not currently modeled at all is much more difficult. The staff proposes to explore a number of options, such as establishing margins in the acceptance guidelines, placing more importance on defense-in-depth, and others. We agree and encourage the staff to actively pursue the resolution of this issue.

Issue 3

The concept of a "three-region" approach is consistent with the Electric Power Research Institute's PSA Applications Guide (PSAAG), although the boundaries of the regions used in the PSAAG are not necessarily the ones that the staff will adopt.

The staff has raised the issue of how "packaged" requests a. ɜ to be handled. Packaging is the process by which risk trade-offs can be accomplished. It is a significant benefit of risk-informed regulation. We believe that it is the overall impact on plant risk that is important, and related changes should be handled as a package. Such changes should be consistent with the current philosophy of risk management; i.e., that the "bottom-line" numbers should not be the only input to the decision-making process, and other concepts such as defense-in-depth must be maintained.

We will continue to monitor the progress of the staff on these issues.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Staff Requirements Memorandum dated May 15, 1996, from John C. Hoyle, Secretary, NRC, to James M. Taylor, Executive Director for Operations, NRC, regarding Briefing on PRA Implementation Plan on April 4, 1996
2. Memorandum dated June 20, 1996, from James M. Taylor, Executive Director for Operations, NRC, to the Commission, Subject: Status Update of the Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA) (from March 1, 1996 to May 31, 1996)
3. Electric Power Research Institute, EPRI TR-105396, Final Report dated August 1995, "PSA Applications Guide"

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

CHAIRMAN

2-sided

July 2, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

FROM: Shirley Ann Jackson

SUBJECT: THE STATEMENT OF CORE DAMAGE FREQUENCY OF 10^{-4} AS A FUNDAMENTAL COMMISSION GOAL

In a Staff Requirements Memorandum, dated June 15, 1990, the Commission stated that, "Implementation of the safety goal may require development and use of 'partitioned' objectives." The Commission further stated that, "A core damage probability of less than 1 in 10,000 per year of reactor operation appears to be a very useful subsidiary benchmark in making judgements about that portion of our regulations which are directed toward accident prevention."

In a letter, dated August 15, 1996, the ACRS stated:

> We believe the safety goals and subsidiary objectives should be used to derive guidelines for plant-specific applications. It is, however, impractical to rely exclusively on the Quantitative Health Objectives (QHOs) for routine use on an individual plant basis. Criteria based on core damage frequency (CDF) and large, early release frequency (LERF) focus more sharply on safety issues and can provide assurance that the QHOs are met. They should be used in developing detailed guidelines.

In the same ACRS letter, the committee also stated, "...the current subsidiary goal of 10^{-4} per reactor-year should be maintained and should be stated as a fundamental safety goal, along with the QHO."

There appears to be both pros and cons for using CDF as a fundamental Commission goal. The pros include: (1) the CDF of 10^{-4} is by de facto already used as a fundamental Commission goal; (2) the derivation of a CDF from the QHOs may yield unacceptably large CDFs; and (3) a core damage frequency goal would constitute a fundamental expression of our defense-in-depth philosophy.

The cons include: (1) several operating plants do not meet the CDF of 10^{-4} as measured by their IPEs, and (2) the CDF goal is difficult to justify on a societal basis (i.e., the QHOs follow directly from societal considerations).

I request that you send a policy paper to the Commission with your views on the merits of the ACRS recommendation to elevate the subsidiary CDF objective to a fundamental safety goal. The paper should clearly articulate the rationale and the pros and cons for your recommendation and should also propose a mechanism for stating CDF as a fundamental safety goal.

cc: Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
SECY
OGC
CIO
CFO
ACRS

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

ONER

October 8, 1997

MEMORANDUM TO: John Larkins, ACRS

FROM: COMMISSIONER DIAZ

SUBJECT: SAFETY GOAL

In a letter to Chairman Jackson dated August 15, 1996, the ACRS recommended that "...the current subsidiary goal of 10E-4 per reactor-year should be maintained and should be stated as a fundamental safety goal, along with the QHO." Chairman Jackson requested in her July 2, 1997, memorandum that the staff assess the merits of this ACRS recommendation. In response, the staff forwarded SECY-97-208 to the Commission recommending that the decision be deferred to allow for further study and discussion with the ACRS. The September 23, 1997, letter from the EDO to Dr. R. L. Seale of the ACRS reiterated the staff's desire to have further discussions with the ACRS on this subject.

I hereby request that the ACRS consider inclusion of the topic of elevating the 10E-4/RY CDF to the NRC's safety goal in their near term meeting agenda. I am also interested in meeting with the available ACRS members to hear their views on this topic and exchange ideas on the implementation of risk-informed regulation. Your effort to arrange this meeting would be appreciated.

cc: Chairman Jackson
Commissioner Dicus
Commissioner McGaffigan
SECY

NEI

NUCLEAR ENERGY INSTITUTE

Joe F. Colvin
PRESIDENT AND
CHIEF EXECUTIVE OFFICER

July 23, 1997

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Mail Stop O-16 G15
Washington, DC 20555-0001

Dear Chairman Jackson:

We read with interest your letter of July 2, 1997, to Mr. Joseph Callan regarding elevation of the core damage frequency subsidiary objective to a fundamental safety goal, as well as your letter of June 26, 1997, that expressed support for a structured, broad-based, risk-informed, performance-based pilot project. We look forward to working with Mr. Ashok Thadani during the pilot project.

As you noted, the pilot project will explore possible ways to improve the efficiency and effectiveness of the regulatory framework by considering the integrated effects of plant activities on overall plant safety. This integrated approach differs significantly from the previous regulation-specific pilot projects that were used to develop Draft Regulatory Guide (DG) 1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis." For this reason, we believe it is inappropriate to constrain the project with the principles and expectations associated with the much narrower risk applications guidance in DG-1061. Since a regulatory guide represents but one way to meet regulatory requirements, the pilot project will be the opportunity to develop alternative approaches.

We wish to reaffirm that the pilot project is not intended to diminish the defense-in-depth philosophy. In fact, the objective of the project is to identify ways to maintain or improve safety margins by improving the effectiveness and efficiency of the regulatory process. We fully intend the pilot project to be consistent with the Commission's Probabilistic Risk Assessment (PRA) policy.

Your letter also indicated the NRC staff's preference for using the subsidiary objectives on core damage frequency and large early release frequency for regulatory decisionmaking. We agree that the frequency of undesirable severe accident events should be kept sufficiently low. The proposed project will trend core damage frequency to ensure that adverse trends are not developing.

We are concerned, however, that the guidance in DG-1061 could cause the NRC staff to treat core damage frequency and large early release frequency values as absolute

limits. We believe such a move would be too restrictive and would preclude regulatory enhancements. Focus on these values would divert both NRC and industry resources to matters that have insignificant core damage frequency impact for plants that were at or slightly above the subsidiary objective.

We also note in your memorandum to Mr. Joseph Callan dated July 2, 1997, that consideration is being given to elevating the core damage frequency subsidiary objective of 10^{-4} per reactor-year to a fundamental safety goal. We believe the safety goals should retain their current, direct focus on public health and safety for the following reasons:

- While core damage frequency is a useful value in managing plant risk, a "one size fits all" value does not relate directly to public health and safety effectively for all plant sites.

- It is inappropriate to raise the 10^{-4} per reactor-year core damage frequency subsidiary objective to a fundamental safety goal for existing plants. The subsidiary objective has been historically characterized as being a mean industry value and not applied on a plant-specific basis. It is unfair to backfit a specific core damage frequency criteria to a generation of plants that were designed without this criteria. Given that the safety goals are widely viewed as the Commission's expression of "how safe is safe enough," using core damage frequency as a fundamental safety goal now would send a message to the public that plants that exceed the core damage frequency objective are unsafe, even though they may be well below the safety goal quantitative health objectives.

- In addition, establishing a 10^{-4} per reactor-year core damage frequency value as a fundamental goal could have a chilling effect on licensee willingness to develop more complete core damage frequency risk analyses that could put their plants close to or above an established goal.

We believe a better course of action would be to assess a realistic range of core damage frequency values, based on complete core damage frequency risk analyses, performed by both the industry and the NRC. These more accurate, and more certain, values could then be used to determine the margins to the quantitative health objectives of the safety goal policy. Information being developed in conjunction with the proposed pilot project supports this approach.

Sincerely,

Joe F. Colvin

Joe F. Colvin

STATUS OF AP600 REVIEW

MR. JOHN BARTON
DR. THOMAS KRESS
ACRS

STATUS OF AP600 REVIEW

- o 22 ACRS Subcommittee meetings since December 17, 1991
- o 5 ACRS full Committee meetings since June 9, 1995
- o ACRS reports and letters
- o 2/19/98: Concerns related to the Test And Analysis Program
- o 6/17/97: Supported requirement for nonsafety-related spray system
- o 8/15/96: Endorsed staff position on policy and key technical issues
- o 6/15/95: Recommended quantifying uncertainties in passive system reliability and separating radioactive leakage from containment design criteria

ACRS CONCERNS RELATED TO THE TEST AND ANALYSIS PROGRAM

- Lack of adequate justification for level of conservatism used in modeling reactor coolant system behavior during design-basis accidents

- Insufficient demonstration of adequate design-basis margin relative to passive containment system behavior

- Lack of quality documentation

ACRS PLANS

3/31-4/1	Subcommittee: SSAR and draft FSER Chapters
4/2-4	Full Committee [prepare interim letter]
4/28-29	Subcommittee: Test and Analysis Program
5/12-14	Subcommittee: SSAR and draft FSER Chapters, PRA, and ITAAC
6/3-5	Full Committee [prepare interim letter]
6/17-18	Subcommittee: Draft FSER Chapters
7/6-7	Subcommittee: Outstanding Issues
7/8-10	Full Committee [prepare final report]

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

February 19, 1998

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: INTERIM LETTER ON THE SAFETY ASPECTS OF THE WESTINGHOUSE ELECTRIC COMPANY APPLICATION FOR CERTIFICATION OF THE AP600 PLANT DESIGN

During the 448th meeting of the Advisory Committee on Reactor Safeguards, February 5-7, 1998, we reviewed the AP600 test and analysis program and various chapters of the AP600 Standard Safety Analysis Report. Our Subcommittees on Advanced Reactor Designs and on Thermal Hydraulic and Severe Accident Phenomena have reviewed these matters previously, as listed in Attachment 1. During these reviews, we had the benefit of discussions with representatives of the NRC staff and the Westinghouse Electric Company (Westinghouse) and of the documents referenced.

TEST AND ANALYSIS PROGRAM

The central goals of the Westinghouse Test and Analysis Program (TAP) are to confirm the design basis for the nuclear power plant components and systems unique to the AP600 design and to provide test data to support validation of relevant plant system codes. Westinghouse has concluded its testing programs, and its current focus is on the verification of the pertinent analytical tools.

Our Thermal Hydraulic and Severe Accident Phenomena Subcommittee began its review of the Westinghouse TAP in December 1991 and several meetings of the Subcommittee have been held in the interim. The Subcommittee last met to review the status of the key elements of the Westinghouse TAP on December 9-12, 1997.

Over the course of these reviews, the Thermal Hydraulic and Severe Accident Phenomena Subcommittee has raised a number of issues that have been documented only in Subcommittee minutes and transcripts, Subcommittee Chairman's reports, and ACRS consultants' reports. In the interest of documenting these issues in a single report, a listing is provided below. We recommend that Westinghouse

provide responses on these issues to the NRC staff for review. Those issues that we consider to be of higher priority are marked with an asterisk.

Reactor Coolant System Issues:

* The basis for not including the momentum fluxes in the NOTRUMP code, particularly during the blowdown phases of the accident analyses

* Explanation of the applicability of Equation 3-63 of Reference 3 to the critical flow of a single component two-phase fluid

* Validation basis for the drift-flux modeling of horizontal flow

* Explanation of why the blowdown flows out of the automatic depressurization system (ADS) valves 1, 2, and 3 and out of the break itself are not predicted well by the NOTRUMP code and what will be done to assure a conservative prediction of AP600 behavior (we are particularly concerned about using modeling deficiencies as compensating effects)

* A more complete demonstration that the proposed penalty on fluid level in the in-containment refueling water storage tank (IRWST) provides sufficient conservatism to offset the uncertainties in the calculated pressurizer level holdup and resulting minimum core level

* The basis for validation for the liquid entrainment model used for the ADS-4 line

* Justification for the absence of (or completion of) a "multi-loop" scaling analysis during the IRWST cooling phase when the vessel inventory approaches a minimum

• Description of the pressurizer flooding model and its validation basis (treatment of the surge line from the hot leg to the pressurizer)

• Explanation of how upstream flow effects were treated in reducing the data in the ADS separate effects tests and in the NOTRUMP code

• The basis for the inconsistencies between the NOTRUMP code noding used for the integral system test configurations and that used for the AP600 plant model

Containment Issues:

* Justification for the use of an incorrect expression in the rate-of-pressure-change equation (Equation 34 in Reference 6)

* Justification for the inappropriate cancellation of the partial derivative of internal energy at constant pressure by the partial derivative of internal energy at constant volume to arrive at Equation 34 of Reference 6

* Re-evaluation of the derivation and quantification of the scaling pi groups resulting from a correction of Equation 34 of Reference 6

* Justification for using the WGOTHIC lumped parameter model well-mixed assumption for calculating the AP600 containment behavior

- Justification for the use of steady-state testing in the Passive Containment System Large Scale Test facility to validate transient heat transfer correlations in the WGOTHIC code

- Justification for the normalization of the rate-of-pressure-change term in Equation 34 in Reference 6

- Technical basis for the treatment of the cooled containment boundary laminar sublayer in the WGOTHIC code

- Validation basis for assuming a low elevation for the main steam line break

- Justification that the calculated peak containment pressure has appropriate margin in view of the observation that all three of the containment cooling system mechanisms (i.e., the passive cooling water system, heat transfer to the containment shell, and heat transfer to the internal structures) are required to turn the pressure over just as it reaches the design value

- Quantification of the impact of incorrect (with respect to AP600) relative magnitudes of energy and mass addition and energy removal during the Large Scale Tests on the usefulness of the data for WGOTHIC code validation for use on AP600

In addition to the above, we are disturbed by the poor status of documentation related to information needed to certify the AP600 design. We believe that any certification should be contingent upon documentation of sufficient quality to provide a traceable and well-archived licensing basis.

SAFETY ANALYSIS REPORT

Our Advanced Reactor Designs Subcommittee began its review of the AP600 design in January 1995. Since then, we have issued two reports to the Commission: one report concerned policy and key technical issues, and the second supported the requirement for a containment spray system.

We have reviewed the following Standard Safety Analysis Report chapters and have no comments at this time:

- Chapter 1 - Introduction
- Chapter 4 - Reactor
- Chapter 5 - Reactor Coolant and Connected Systems
- Chapter 7 - Instrumentation and Controls
- Chapter 8 - Electrical Power
- Chapter 11 - Radioactive Waste Management
- Chapter 13 - Plant Operations (excluding security)
- Chapter 18 - Human Factors Engineering

SUMMARY

We have identified a number of issues associated with the Westinghouse Test and Analysis Program that should be resolved during the staff review. Our assessment of the adequacy of the Standard Safety Analysis Report chapters discussed to date is incomplete. Completion of our review is contingent on the timely receipt of draft Final Safety Evaluation Report chapters.

Sincerely,

R. L. Seale

Robert L. Seale
Chairman

References:

1. Westinghouse Electric Corporation. "AP600 Standard Safety Analysis Report," updated through Revision 16 dated September 2, 1997.
2. Letter dated January 16, 1998, from William Huffman, NRC, to Nicholas Liparulo, Westinghouse Electric Corporation, Subject: Open Items Associated with the AP600 Safety Evaluation Report on the AP600 Containment Design and Accident Analyses.
3. Westinghouse Electric Corporation, WCAP-14727, Revision 1, "AP600 Scaling and PIRT Closure Report," July 1997 (Proprietary).
4. Westinghouse Electric Corporation, WCAP-10079-P-A, "NOTRUMP - A Nodal Transient Small Break and General Network Code," August 1985 (Proprietary).
5. Westinghouse Electric Corporation, WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (Proprietary).
6. Westinghouse Electric Corporation, WCAP-14845, Revision 2: "Scaling Analysis for AP600 Containment Pressure During Design Basis Accidents," June 1997 (Proprietary).
7. Westinghouse Electric Corporation, WCAP-14407, Revision 1, "WGOTHIC Application to AP600," July 1997 (Proprietary).
8. Westinghouse Electric Corporation, WCAP-14326, Revision 1, "Experimental Basis for the AP600 Containment Vessel Heat and Mass Transfer Correlations," May 1997 (Proprietary).
9. Westinghouse Electric Corporation, WCAP-14807, Revision 2, "NOTRUMP Final Validation Report for AP600," June 1997 (Proprietary).
10. Westinghouse Electric Corporation, WCAP-14967, Revision 0, "Assessment of Effects of WGOTHIC Solver Upgrade From Version 1.2 to 4.1," September 1997 (Proprietary).
11. Westinghouse Electric Corporation, WCAP-14135, "Final Data Report for PCS Large-Scale Tests, Phase 2 and Phase 3," July 1994 (Proprietary).

Attachment:

1. Chronology of the ACRS Review of the Westinghouse Application for AP600 Standard Design Certification

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

June 17, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED STAFF POSITION REGARDING INCLUSION OF A CONTAINMENT SPRAY SYSTEM IN THE AP600 DESIGN

During the 442nd meeting of the Advisory Committee on Reactor Safeguards, June 11-14, 1997, we met with representatives of the NRC staff and the Westinghouse Electric Corporation to discuss the proposed staff position that the AP600 design should include a containment spray system or equivalent for accident management following a severe accident. We also had the benefit of the documents referenced.

The staff position is that the addition of a nonsafety-related containment spray system in the AP600 design would achieve an appropriate balance between prevention and mitigation of severe accidents. The staff stated that such a system would compensate for the uncertainties associated with natural removal mechanisms for aerosols during severe accidents and provide for accident mitigation and operator intervention capability as part of a long-term accident management strategy. The staff believes that a containment spray system or equivalent is consistent with the AP600 passive design philosophy and the Commission's defense-in-depth philosophy.

The Westinghouse position is that the AP600 design meets existing regulatory prevention and mitigation criteria, including the Safety Goals. This may well be the case; however, we have not yet completed our review. Westinghouse also contends that a requirement for additional systems is neither justified nor warranted. The information presented to us by Westinghouse did not address the relevant uncertainties associated with the AP600 probabilistic risk assessment.

Ideally, the determination of the need for a containment spray system should be based on a judgment as to the levels of

uncertainties associated with aerosol depletion and overall risk, as well as on the value of additional accident management capability. The first question of interest is, what are the nature and extent of the uncertainties of concern. If all uncertainties were quantifiable, it would be fairly straightforward to determine whether sufficient defense-in-depth is built into the system by assessing the risk status with respect to the subsidiary Safety Goals (core damage frequency and large, early release frequency). At present, however, a large component of uncertainties remain unquantified. The identification of these uncertainties and the qualitative judgments regarding their impact on regulatory decisions would make the debate more specific and would enhance communication among the stakeholders.

In judging the usefulness of a containment spray system in compensating for these uncertainties, both positive and negative impacts of this system should be evaluated in a quantitative and qualitative way. A judgment based on such an evaluation would help make the decision more acceptable to stakeholders because the basis for the decision would be explicit and transparent. Furthermore, such an evaluation process would be a good first step towards the integration of risk and traditional concepts such as defense-in-depth.

Although we prefer to have the information from the evaluation outlined above, based on our current state of knowledge, we support the staff's contention that the addition of a severe accident mitigation system is appropriate. The addition of a spray system to the AP600 containment would significantly increase its effectiveness in fission product control and provide the ability to intervene and control the course of an accident. We believe, however, that the spray design concept suggested by the staff is marginally adequate.

The debate associated with this issue and the difficulty of making a decision highlight our belief that the NRC needs to develop a new policy statement that would provide more guidance on the extent and nature of defense-in-depth expected by the Commission.

Dr. Dana A. Powers did not participate in the Committee's deliberations regarding this matter.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

<u>References</u>:

1. ACRS letter dated June 15, 1995, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Commission Paper on Staff Positions on Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design.
2. ACRS report dated August 15, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design."
3. Memorandum dated November 12, 1996, from James M. Taylor, Executive Director for Operations, NRC, to the NRC Commissioners, Subject: Clarification of Staff Position in SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standard Pressurized Reactor Design."
4. Memorandum dated January 15, 1997, from John C. Hoyle, Secretary, NRC, to Hugh L. Thompson, Jr., Acting Executive Director for Operations, NRC, and Karen D. Cyr, General Counsel, NRC, Subject: Staff Requirements - SECY-96-128 - Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design.
5. Memorandum dated February 19, 1997, for the Commissioners, from Hugh L. Thompson, Jr., Acting Executive Director for Operations, NRC, Subject: SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design."
6. Memorandum dated March 18, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to Chairman Jackson, Subject: Use of Non-Safety-Related Equipment to Address Safety Concerns on Nuclear Power Plants.
7. Letter dated March 13, 1997, from Brian A. McIntyre, Westinghouse Electric Corporation, to John Hoyle, Secretary, NRC, Subject: Westinghouse Comments on SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standard Pressurized Reactor Design."
8. Memorandum dated May 16, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to the NRC Commissioners, Subject: Westinghouse Comments on SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standard Pressurized Reactor Design."

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

March 17, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: PUBLICATION OF PROPOSED JOURNAL ARTICLE CONTAINING ROSA TEST DATA

In a December 13, 1996 letter, Mr. James M. Taylor (then Executive Director for Operations) requested that the ACRS provide an independent review of the technical merits of a Westinghouse Electric Corporation concern that the data from some of the ROSA-V tests included in a proposed journal article jointly authored by NRC and Japan Atomic Energy Research Institute personnel when combined with other publicly available information could permit a competitor to deduce proprietary design details through "reverse engineering."

During our 439th meeting, March 6-8, 1997, we reviewed the merits of this concern. Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this issue during a meeting on February 19, 1997. During this review, we had the benefit of discussions with representatives of the NRC staff and Westinghouse Electric Corporation. We also had the benefit of the documents referenced.

During the February 19, 1997 Subcommittee meeting, representatives of Westinghouse stated that their concern was somewhat broader than that posed to the Committee by Mr. Taylor. Their broader concern was that a third party could use information as presented in the proposed article along with other available information to provide credibility to a competing design and thereby undermine the competitive international marketing position of Westinghouse.

We recognize that there are elements of this issue that will not be captured by a strictly technical review. These could include, for example, NRC needs for (1) sufficient public disclosure of its safety case, (2) independence from the vendors, and (3) uninhibited future publication of data from other test facilities. Nevertheless, in our review, we have chosen to focus on two technical questions associated with the broader concern:

(1) Is there sufficient information available in the proposed article (in its current form) that could be used along with

other information in the public domain by a knowledgeable third party to develop a technically defensible thermal-hydraulic computer mockup of the ROSA-V facility?

(2) If so, can the data in the form currently proposed to be published in the article be used along with a computer model to provide credibility to a competing design for which the performance of the passive systems is demonstrated via the computer model?

It is our opinion that a technically defensible computer model of the ROSA-V facility could be developed by a third party from the information in the proposed article when combined with other available information in the public domain, and that such a model could be an important factor in providing credibility to a competing passive design. A claim could be made that the design is based on an NRC-approved design. In our opinion, there is significant merit to the concerns of Westinghouse and these concerns ought to be given appropriate consideration in any decision regarding the form of data presentation in the proposed article.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

<u>References</u>

1. Letter dated December 13, 1996, from James M. Taylor, Executive Director for Operations, NRC, to Thomas S. Kress, Chairman, ACRS, Subject: ACRS Review of ROSA and APEX Data Prior to Publication.
2. Proposed journal article, "Implications of the Rig of Safety Assessment/AP600 High- and Intermediate-Pressure Test Results," by Louis M. Shotkin, Nuclear Regulatory Commission, and Yutaka Kukita, Japan Atomic Energy Research Institute **(Review for Public Release Pending)**.
3. Letter dated February 4, 1997, from Brian A. McIntyre, Westinghouse Electric Corporation, to Robert Seale, Chairman, ACRS, Subject: ACRS Review of the Proposed Publication of a Journal Article Concerning AP600 Test Results.
4. Proposed journal article, "Core Markup Tank Behavior Observed During the ROSA-AP600 Experiments," by T. Yotomot, Japan Atomic Energy Research Institute, et al. **(Review for Public Release Pending)**.
5. Proposed paper, "ROSA-AP600 Experiment Simulating Steam Generator Tube Rupture Transient," by H. Nakamura and Y. Kukita, Japan Atomic Energy Research Institute, for the 2nd

International Topical Meeting on Advance Reactors Safety, Orlando, Florida, June 1-4, 1997.

6. Proposed paper, "Analysis of Wall Heat Capacity Effect on Core Makeup Tank Drain-Down Behavior in ROSA/AP600 Experiment," by Masaya Kondo, Japan Atomic Energy Research Institute, et al., for the 2nd International Topical Meeting on Advance Reactors Safety, Orlando, Florida, June 1-4, 1997.
7. American Nuclear Society proceedings paper, "Passive Residual Heat Removal System Heat Exchanger Characterization During Simulated Station Blackout," by Owen Stevens and Jose N. Reyes, Jr., Oregon State University, from the 1996 National Heat Transfer Conference, Houston, Texas, August 3-6, 1996.
8. American Nuclear Society proceedings paper, "Evaluation of the APEX Break Flow Measurement System During Subcooled Depressurization," by David A. Pimentel and Jose N. Reyes, Jr., Oregon State University, from the 1996 National Heat Transfer Conference, Houston, Texas, August 3-6, 1996.
9. International Atomic Energy Agency proceedings paper, "SPES-2. AP600 Integral Systems Test Results," by L. E. Conway and R. Hundal, Westinghouse Electric Corporation, Italy, May 1995.
10. Proceedings paper, "Comparison of the SPES-2 Pre-Test Predictions and AP600 Plant Calculations Using RELAP5/MOD3," by A. Alemberti, C. Frepoli, and G. Graziosi, ANSALDO Nuclear Division (Copies available from the NRC Office of Nuclear Reactor Regulation).
11. Proceedings paper, "SPES-2 Cold Leg Break Experiments: Scaling Approach for Decay Power, Heat Losses Compensation and Metal Heat Release," by A. Alemberti, C. Frepoli, and G. Graziosi, ANSALDO Nuclear Division (Copies available from the NRC Office of Nuclear Reactor Regulation).
12. Proceedings paper, "SPES-2, The Full-Height, Full-Pressure, Integral System AP600 Test Facility," by M. Bacchiani, SIET S.P.A., et al., Twenty-Second Water Reactor Safety Information Meeting, Bethesda, Maryland, October 24-26, 1994.
13. Proceedings paper, "SPES-2 RELAP5/MOD3 Noding and 1 Inch Cold Leg Break Test S00401," by A. Alemberti, C. Frepoli, and G. Graziosi, ANSALDO Nuclear Division, Twenty-Second Water Reactor Safety Information Meeting, Bethesda, Maryland, October 24-26, 1994.
14. Abstract of thesis, "Characterization of the Advanced Plant Experiment (APEX) Passive Residual Heat Removal System Heat Exchanger," by Owen Stevens, Oregon State University, presented June 7, 1996.
15. Abstract of thesis, "Two-Phase Fluid Break Flow Measurements and Scaling in the Advanced Plant Experiment (APEX)," by David Alan Pimentel, Oregon State University, presented on May 9, 1996.

August 15, 1996

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

Dear Chairman Jackson:

SUBJECT: SECY-96-128, "POLICY AND KEY TECHNICAL ISSUES PERTAINING TO THE WESTINGHOUSE AP600 STANDARDIZED PASSIVE REACTOR DESIGN"

During the 433rd meeting of the Advisory Committee on Reactor Safeguards, August 8-10, 1996, we reviewed the subject document. Our Subcommittee on Westinghouse Standard Plant Designs met on July 19, 1996 to review this matter. During this review, we had the benefit of discussions with representatives of the staff and of the Westinghouse Electric Corporation. We also had the benefit of the documents referenced.

Conclusion

We endorse the positions recommended by the staff in addressing the following three policy issues pertaining to the Westinghouse AP600 standardized passive reactor design.

Policy Issues

- Prevention and Mitigation of Severe Accidents

  The staff is seeking Commission approval to consider the use of non-safety systems in the AP600 design to address the uncertainties associated with the passive fission product removal mechanisms for design-basis analysis and for balance between prevention and mitigation of severe accidents. Westinghouse has no objection to the staff's crediting of non-safety equipment that is already a part of the AP600 design, but objects to a requirement for adding a non-safety-grade containment spray system.

  The applicant's submittals provide some support for demonstrating fission product removal using only passive removal mechanisms. Nonetheless, we are persuaded by the staff position that systems beyond the passive removal mechanisms should be evaluated to provide greater confidence in the performance of the plant design in mitigating design-basis and severe accidents. We recommend Commission approval.

- External Reactor Vessel Cooling

  The staff is seeking Commission approval for requiring that the applicant provide limited analytical evaluation of postulated ex-vessel phenomena, notwithstanding that the AP600 design is intended to prevent reactor vessel melt-through. We recommend Commission approval.

- Post-72-hour Actions

  The staff is seeking Commission approval for requiring that the AP600 design be capable of sustaining all design-basis events with onsite equipment and supplies for the long term. We recommend Commission approval.

Technical Issues

The staff added spent fuel pool cooling to its list of technical issues being tracked in the review. At present, the applicant will be required to provide additional onsite capability to remove decay heat from the spent fuel pool over an extended period of time. We believe this requirement may be found unnecessary after considering the low risk associated with the current design.

Dr. Dana A. Powers did not participate in the Committee's deliberations regarding the severe accident source term. Dr. T. S. Kress did not participate in the Committee's deliberations regarding external reactor vessel cooling.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. U. S. Nuclear Regulatory Commission, SECY-96-128, dated June 12, 1996, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design
2. Letter dated June 15, 1995, from T.S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Commission Paper on Staff Positions on Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design
3. Letter dated August 8, 1995, from James M. Taylor, Executive Director for Operations, NRC, to T.S. Kress, Chairman, ACRS, Subject: Response to ACRS Comments on Commission Paper on Technical Issues Pertaining to the Westinghouse AP600 Design

CORRECTED COPY: 2/8/96
(see Page 2)

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

June 15, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED COMMISSION PAPER ON STAFF POSITIONS ON TECHNICAL ISSUES PERTAINING TO THE WESTINGHOUSE AP600 STANDARDIZED PASSIVE REACTOR DESIGN

During the 422nd meeting of the Advisory Committee on Reactor Safeguards, June 8-10, 1995, we discussed the subject Commission paper. Our Subcommittee on Westinghouse Standard Plant Designs met on May 31, 1995, to review this matter. During these meetings, we had the benefit of discussions with representatives of the staff and Westinghouse. We also had the benefit of the documents referenced.

The intent of the proposed Commission paper is to record the staff positions on ten separate issues. In some cases, however, the reviews have not progressed to the point that the staff can recommend a position. In such cases, the paper describes the approach that Westinghouse is proposing in its application with little staff comment. The staff is continuing its review of these matters.

Our comments follow the same organization found in the attachment to the paper.

I. <u>Leak-Before-Break Approach</u>

Westinghouse proposes that any dynamic effects associated with postulated pipe ruptures in a broad range of pipe sizes can safely be excluded from the AP600 piping design basis by virtue of the current understanding of leakage and flaw sizes, and the proposed leakage rate limit of 0.5 gpm. The range of pipe sizes (4 inch diameter and greater) that would be covered by the leak-before-break (LBB) approach is broader than that allowed in currently operating pressurized water reactors for which the usual plant leakage rate limit is set at 1.0 gpm.

The staff agreed that the leakage rate limit of 0.5 gpm is achievable in the AP600 design but wishes to add conservatism in applying the LBB approach at the design certification stage by requiring that all loads used in the piping design be multiplied by a factor of 1.4. The staff considers this prudent because the detailed design of piping configuration

ENCLOSURE 7

and the as-built stress levels will not be available for review at the certification stage. Westinghouse argued that this added conservatism is not needed and will act to limit the gains in plant arrangement, economy, and safety that application of the LBB approach could provide.

We believe that the staff is hard pressed to justify adding conservatism on all the piping loads above that which has been applied to other plants. Although it is true that the details of the piping design are some years away, the staff and Westinghouse should now be able to combine the standard piping design protocols with what is known about the performance of flawed pipes into a design criterion without excessive conservatism.

II. Security Design

The proposed AP600 plant arrangement includes a vehicle barrier at a "stand-off distance," but the personnel access control will be located within the nuclear island of the plant. The vital areas of the plant are coterminous. This feature is not specific to the passive nature of the plant design and might be offered in other plant designs as well. The staff continues to review the proposed design, but seems receptive to the idea. The staff believes that inspections, tests, analyses, and acceptance criteria (ITAAC) may be required for this security design.

We believe the proposed security design could meet the safety and security requirements when implemented, and we are interested in the continuing staff review of the proposed design. We also noted that the design seems to offer less flexibility for the many work access points that operating plants need during outage periods.

III. Technical Specifications

Westinghouse proposes that hot shutdown, rather than cold shutdown, be considered the safe shutdown end state. The staff evaluation has not progressed to the point where the staff could make substantial comment. We also will withhold comment at this time. We expect that review of the probabilistic risk assessment regarding this issue will be instructive.

IV. Initial Test Program

Westinghouse and the staff have been discussing the content of the initial test program to be performed by the first plant built under the design certification, and test programs to be performed by subsequent plants. We believe that the staff is approaching the matter appropriately. When the discussions have resulted in new submittals from Westinghouse, we may have more information on which to comment.

V. Passive System Thermal-Hydraulic Performance Reliability

The staff believes that the magnitude of the natural forces relied on for the passive safety systems leads to large uncertainties in the thermal-hydraulic performance. It stated that one could quantify these

uncertainties, but only with "a prohibitively large number of computations." The staff proposed instead that a surrogate conservative risk-based margins approach be developed to eliminate the need to quantify thermal-hydraulic uncertainty for most, if not all, accident sequences.

This approach may be expedient, but we believe efforts should continue on the quantification of the uncertainty for use in probabilistic risk assessments.

VI. Regulatory Treatment of Non-Safety Systems

Westinghouse and the staff have been meeting to review the need for some level of regulatory treatment for systems and components that are not safety grade, but that have important support and backup functions. A key issue identified by the staff in this regard is the reliance that Westinghouse places on equipment or materials that may be required beyond 72 hours following an accident but which are not to be stored onsite. The staff review of this issue is currently under way, and the staff has not stated a position beyond identifying concerns.

Accident scenarios for existing plants reach a point when reliance must be placed on offsite materials. We expect that the staff will need to be satisfied that the AP600 design can be brought to a stable condition using onsite equipment, and that any additional needed resources are reasonably available.

VII. Containment Performance

The staff intends to use both deterministic and probabilistic containment performance goals in reviewing the AP600. This is consistent with the Commission direction given in the July 21, 1993 Staff Requirements Memorandum related to SECY-93-087. We believe that the staff position is appropriate.

VIII. External Reactor Vessel Cooling

Westinghouse proposes a severe accident mitigation strategy for the AP600 that includes the ability to flood the cavity under the reactor to a level that is effective in cooling the lower reactor vessel shell and preventing reactor vessel melt-through following core melt. The staff stated that this would be a desirable feature if the technical issues can be resolved. The staff is pursuing those issues with Westinghouse. We believe that the staff is following an appropriate path, but we will closely follow the resolution of the technical issues.

IX. Passive Hydrogen Control Measures

The proposed AP600 design includes unpowered catalytic recombiners to control hydrogen generated in a design-basis accident (DBA). This is consistent with the overall concept of controlling design-basis accidents with passive measures. (The plan is to use igniters to control severe

accident hydrogen.) There are technical questions involving the qualification and effectiveness of catalytic recombiners in an accident environment. The staff proposes to approve the use of passive recombiners contingent on the resolution of these issues. We believe that the staff position is appropriate.

X. DBA and Long-Term Severe Accident Radiological Consequences

While the passive nature of the AP600 safety features is very attractive, the design has some downside characteristics. Post-accident pressure in the containment will remain positive longer than a plant designed with active cooling. Further, following severe accidents, the removal of radioactive species from the containment atmosphere is expected to be less efficient with passive means than it would be using active sprays or filters. Thus, there is the potential for radioactive leakage for an extended period, compared to that of the existing plants. The staff believes that this situation calls for consideration of additional means, such as a nonsafety-grade containment spray, to reduce containment pressure and suspended radionuclides following a severe accident. The staff has asked Westinghouse to reconsider its proposed position in this regard.

In addition, Westinghouse proposes a source term somewhat different from what the staff would use with respect to both timing and release fractions. The staff indicates that the technical differences here would not be of much concern if the staff can be satisfied that there would be an active system available to reduce the containment leakage potential.

We believe that the issues associated with the potential for radioactive leakage and the source term should be treated separately. We believe that the staff position on the source term is appropriate. The radioactive leakage from the proposed containment design, however, should be considered with respect to public risk and the safety goals.

In the course of this review, it has occurred to us that the certification of advanced light-water reactors provides an important opportunity to continue the evolution toward performance-based regulation. Current plans, unfortunately, do not take complete advantage of this opportunity, perhaps because of schedule constraints. The debate over the procedure to impose unquantified levels of conservatism on analyses of leak-before-break for small-diameter piping reflects a continuation of past practice. The aspirations of both the industry and the NRC would be better served by a performance-based criterion. Similarly, arguments on the time frame for analyses of radionuclide concentrations in containment would be unnecessary if a performance-based criterion were derived. In general, such performance-based criteria would be more consistent with the state-of-the-art engineering being employed in the design of advanced light-water reactors than the continued use of traditional criteria developed in the past when there was a poorer understanding of safety-related processes and phenomena.

Dr. Dana A. Powers did not participate in the Committee's deliberations regarding the severe accident source term. Dr. Thomas S. Kress did not participate in the Committee's deliberations regarding external reactor vessel cooling.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>References</u>:

1. Memorandum dated May 15, 1995, from J. Taylor, NRC Executive Director for Operations, to the Commissioners, Subject: Advance Information Copy of Forthcoming Commission Paper - Staff Positions on Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design
2. SECY-93-087 dated April 2, 1993, from J. Taylor, NRC Executive Director for Operations, to the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs
3. SRM dated July 21, 1993, from S. Chilk, Secretary of the Commission, to J. Taylor, NRC Executive Director for Operations, Subject: SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs

SHUTDOWN AND LOW POWER OPERATION

DR. DANA A. POWERS
ACRS

BACKGROUND

- o The Committee discussed need for baseline study with the Commission during the May 2, 1997 meeting

- o During the 444th ACRS meeting, the Committee reviewed SECY-97-168, "Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation"

- o The Committee provided a report to the Commission on September 10, 1997, recommending that the proposed Rule not be issued for public comment

CURRENT STATUS

- The Commission issued an SRM on December 11, 1997, directing the staff not to proceed further with the Rulemaking Package
- The Commission approved the staff's recommendation in an SRM dated December 17, 1997, to revise the Maintenance Rule to include provision that the Rule will be applicable to normal shutdown operations

ACRS VIEWS

- ACRS has communicated with the Commission on shutdown and low power operations risk in the past
- ACRS feels staff must develop an understanding of shutdown risk for RG 1.174 if nothing else

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

January 30, 1998

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

SUBJECT: PROPOSED RULEMAKING PACKAGE FOR SHUTDOWN AND FUEL STORAGE POOL OPERATIONS AT NUCLEAR POWER PLANTS

Dear Dr. Seale:

Thank you for your letter of September 10, 1997, commenting on the proposed rulemaking package for shutdown and fuel storage pool operations. After receiving your letter, the Commission issued a staff requirements memorandum (SRM) on December 11, 1997, directing the staff not to proceed further with this rulemaking package.

In response to the Commission SRM, the staff will continue to monitor licensee performance, through inspections and other means, in the area of shutdown operations. If the Commission revises the maintenance rule and the staff determines that further codification of requirements relating to shutdown operations is warranted, the staff will contact the ACRS to discuss your comments.

Sincerely,

L. Joseph Callan
Executive Director for Operations

cc: Chairman Jackson
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
SECY
OGC
OCA
OPA
CFO
CIO

December 17, 1997

MEMORANDUM FOR: L. Joseph Callan
Executive Director for Operations

Karen D. Cyr
General Counsel

FROM: John C. Hoyle, Secretary /s/

SUBJECT: STAFF REQUIREMENTS: SECY-97-173 - POTENTIAL REVISION TO 10 CFR 50.65(a)(3) OF THE MAINTENANCE RULE TO REQUIRE LICENSEES TO PERFORM SAFETY ASSESSMENTS

The Commission approved the staff's recommendation to develop a proposed rulemaking to revise the maintenance rule to require that safety assessments be taken into account prior to performing maintenance activities, subject to the following comments:

1. Although all three alternatives, including not changing the rule, should be considered as part of the regulatory analysis for proposed rulemaking, extended or protracted regulatory analysis of Alternative 1 is unnecessary.

2. In addition to the change from "should" to "shall" in section 50.65(a)(3) as proposed by the staff in Alternative 2, the proposed rule should also incorporate the following changes that are consistent with NRC Regulatory Guide 1.160, Revision 2, and NUMARC 93-01, Revision 2. The staff may suggest alternative wording for Commission consideration as part of the proposed rulemaking package, if the following rule language is problematic:

   a. Since the requirements of the maintenance rule, including the assessment of structures, systems, and components proposed to be removed from service, are applicable during all modes of plant operation, the following clarification should be added as a preamble to the maintenance rule:

      The requirements of this section are applicable during all conditions of plant operation, including normal shutdown operations.

   b. Revise the third sentence of (a)(3) to read as follows:
      Adjustments shall be made where necessary to ensure that the objective of **preventing** failures of structures, systems, and components through maintenance is appropriately balanced against the objective of minimizing unavailability of structures, systems, and components due to monitoring or **preventive** maintenance.

   c. The final sentence of section 50.65 (a)(3) should be redesignated as (a)(4) and revised as follows:

Prior to performing maintenance activities on SSCs within the scope of this section (including, but not limited to, surveillance testing, post-maintenance testing, corrective maintenance, performance/condition monitoring, and preventive maintenance), an assessment of the current plant configuration as well as expected changes to plant configuration that will result from the proposed maintenance activities shall be conducted to determine the overall effect on performance of safety functions. The results of this assessment shall be used to ensure that the plant is not placed in risk-significant configurations.

3. Since the changes to the maintenance rule are part of a larger set of initiatives, including, but not limited to, changes to 10 CFR 50.59 and the integrated review of the NRC assessment process for commercial reactors, the staff should ensure consistency among these efforts.

(EDO) (SECY suspense: 4/30/98)

4. In the limited regulatory analysis discussion of Alternative 3, staff should briefly consider how this alternative might be pursued. One disadvantage of Alternative 2 is that licensees could theoretically use technically inferior methods for conducting safety assessments and could theoretically perform maintenance in configurations involving risk levels that may be imprudent, yet still argue that they are in compliance with the requirements of the revised maintenance rule to take into account safety assessments prior to performing maintenance.

(EDO) (SECY Suspense: 4/30/98)

To address this issue, the Commission would consider, as part of a future separate rulemaking, a staff proposal to incorporate by reference updates to NUMARC 93-01, Revision 2, and NRC Regulatory Guide 1.160, Revision 2, which emerge from the activities described in item 5.

5. As part of the regulatory guidance for this proposed rulemaking, the staff should supplement and expand on the discussion that was provided in the statements of consideration for the original maintenance rule with regard to (1) variations in the rigor and sophistication of the assessments depending on the number and safety significance of SSCs out-of-service and (2) NRC's general expectations with regard to risk levels that the assessment should take into account to ensure a plant is not placed in risk-significant configurations during maintenance activities. This discussion should acknowledge that there can be several inputs to the determination of risk significance of plant configurations, including PRA, deterministic analysis, considerations of defense in depth, and qualitative measures. This discussion would be short of Alternative 3's comprehensive treatment of these issues in the rule itself and would not constitute binding regulatory requirements. In developing this guidance, the staff should also consider whether the Guidelines for Industry Actions to Assess Shutdown Management, NUMARC 91-06, as referenced in Section 11.2 of NUMARC 93-01, Revision 2, could be endorsed by the NRC. Consistent with the Commission's decision on DSI-13, the staff should interact with stakeholders in developing regulatory guidance. Development of regulatory guidance should not delay issuance of the proposed rule, and the Commission expects the final rule to be issued by December 15, 1998.

(EDO) (SECY Suspense: 12/15/98)

cc: Chairman Jackson
Commissioner Dicus
Commissioner Diaz

Commissioner McGaffigan
OGC
CIO
CFO
OCA
OIG
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR
DCS

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

September 10, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED RULEMAKING FOR SHUTDOWN AND FUEL STORAGE POOL OPERATIONS AT NUCLEAR POWER PLANTS

During the 444th meeting of the Advisory Committee on Reactor Safeguards, September 3-5, 1997, we reviewed SECY-97-168, "Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation." During this review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced.

In our April 18, 1997 report to the Commission, we expressed concerns about the existing understanding of risk during shutdown operations. The NRC staff has recognized that shutdown operations may pose substantial risks. The staff sees a need for enforceable rules to ensure safety during shutdown operations. We note that, on numerous occasions in the past, the staff has found enforcement means when deficiencies in shutdown operations have been encountered. It is true, however, that there is no coherent set of requirements now available for the staff to inspect and monitor licensee activities during shutdown operations. This situation may not be relieved by a contrived interpretation of existing rules or unusual interpretations of conventional terms.

The nuclear industry has also recognized the importance of both safety and efficiency during shutdown operations. The industry has established cost-effective practices to achieve levels of safety during shutdown operations. These practices would be acceptable to the staff if it could enforce, inspect, and monitor these practices. Indeed, the proposed rule is a codification of a minimum subset of practices with aims similar to current industry practices.

The staff has been handicapped in the formulation of a rule for shutdown operations by an incomplete understanding of risk during this operating mode. Consequently, risk estimates made by the

staff in its regulatory analysis in support of the proposed rule are largely judgmental and lack the strong technical justifications that are available to estimate risks during power operations. The proposed rule would be vulnerable to a hostile or unsympathetic review.

The regulatory analysis guidelines constrain the staff from giving any credit for benefits of voluntary actions by the industry. As a result, the estimated risks of shutdown accidents for the purposes of a regulatory analysis should not be taken as an indicator of the real risks, which are likely to be considerably lower.

Many of the benefits attributed by the staff to elements of the proposed rule are of the intangible variety associated with enforceability and cannot be quantified in a meaningful way. The requirements in the proposed rule for fuel storage pool operations have benefits that are almost entirely of this intangible variety and provide no real safety benefit. We see little reason to burden an already difficult rulemaking effort with these requirements for fuel storage pool operations that provide so little safety benefit.

The staff has included requirements for fire protection in the proposed rule. There is, indeed, widespread agreement that fire protection is an essential safety issue during shutdown operations. It is not evident to us that existing fire protection rules are not applicable to shutdown operations and there is a need to augment these rules. We do recognize that the staff is committed to revise the fire protection regulations. It may not be opportune to add another element to the existing fire protection requirements only to modify these requirements in the near future.

We do believe shutdown operations of nuclear power plants deserve regulatory attention. Such regulatory attention is, in fact, overdue in light of the number of incidents of risk significance that have occurred in recent years and the heightened regulatory activity devoted to shutdown events. Coherent, risk-informed, enforceable requirements for shutdown operations that could be inspected and monitored in a consistent predictable manner would serve the interests of both the industry and the public. Seldom, however, has there been a clearer case for the cooperative development of such requirements by the industry and the staff. It is clear that it is possible to achieve a level of safety acceptable to the NRC staff at a cost palatable to the industry. Indeed, this situation may already exist absent only the elements of enforcement capability and bases for inspection and monitoring.

We find no merit in the suggestion that the Maintenance Rule (10 CFR 50.65) should be the basis for regulation of shutdown operations. We find compelling the view expressed by a representative of the Office of General Counsel that the

Maintenance Rule serves to assure that systems, components, and structures are capable of performing their safety functions. The rule cannot be construed to specify these safety functions.

We do not recommend issuance of the proposed rule on shutdown and fuel storage pool operations for public comment. We do recommend that the NRC staff explore the flexibilities of its policies and practices so it can work with industry to find ways to achieve enforceability and support for inspecting and monitoring the voluntary and apparently effective industry practices without impugning or penalizing such practices. We reiterate our belief that the staff needs to develop a more quantitative understanding of risk during all phases of low-power and shutdown operations. We further recommend that fuel storage pool operations not be included in the reexamination of the regulation of shutdown operations. Similarly, fire protection requirements for shutdown operations may be deferred to the more comprehensive reexamination of the existing fire protection regulations.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

<u>References</u>:

1. SECY-97-168, Memorandum dated July 30, 1997, for the Commissioners, from L. Joseph Callan, Executive Director for Operations, NRC, Subject: Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation.
2. U. S. Nuclear Regulatory Commission, "Regulatory Analysis for the Proposed Regulation § 50.67, 'Shutdown and Fuel Storage Pool Operations at Nuclear Power Plants,'" July 24, 1997.
3. U. S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1066, "Shutdown and Fuel Storage Pool Operations at Nuclear Power Plants," July 24, 1997.
4. Report dated April 18, 1997 from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Establishing a Benchmark on Risk During Low-Power and Shutdown Operations.

NRC SAFETY RESEARCH PROGRAM

DR. ROBERT SEALE
DR. DANA POWERS
ACRS

- o Staff Requirements Memorandum (September 9, 1997) -- Transfer of the Research Advisory Function to the ACRS

- o ACRS to take an active role in reviewing ongoing research program initiatives

- o The ACRS to examine how the research program is positioned for the changing environment such as:
  - Economic
  - Deregulation
  - Aging
  - Premature Retirement of Plants
  - License Renewal
  - Maturation of PRA
  - Congressional Mandate For Rational Regulation
  - Improved Licensee Performance
  - New Technologies
  - Public Law For Endorsing Consensus Standards
  - Declining NRC Resources For Inspection and Monitoring

- February 1998 - ACRS Report To Congress
- Examples of areas of Research that will be impacted as a result of declining NRC resources
- March 1998 - ACRS reviewed pertinent Commission directives and strategic plans including:
  - Research program should identify and focus on the most risk-significant issues.
  - The program should include both confirmatory and anticipatory elements.
  - Each research activity should establish a clear nexus between the outputs of the activity and the agency goals it supports.

- The research program should respond to line organization needs.

- The research program should maintain sufficient expertise and capability to respond to future needs.

- The research program should anticipate and explore problems proactively rather than reactively.

- Consideration should be given to revision or redefinition of research goals.

- Consideration should be given to intermediate goals closer to NRC operating regimes.

- ACRS report to the Commission in May 1998

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

February 24, 1998

The Honorable Albert Gore, Jr.
President of the United States Senate
Washington, DC 20510

Dear Mr. President:

I am pleased to transmit to the Congress the 1997 report of the Advisory Committee on Reactor Safeguards on the U.S. Nuclear Regulatory Commission's Safety Research Program. This report is required by Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209. This report concludes that severe budget reductions are causing substantial deterioration of the internationally respected capability of the U.S. Nuclear Regulatory Commission to conduct a forward-looking, effective safety research program.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

Enclosure:
Nuclear Safety Research, A Report to the U.S. House of Representatives and the U.S. Senate, by the Advisory Committee on Reactor Safeguards of the U.S. Nuclear Regulatory Commission, dated February 1998

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

February 24, 1998

The Honorable Newt Gingrich
Speaker of the United States
House of Representatives
Washington, DC 20515

Dear Mr. Speaker:

I am pleased to transmit to the Congress the 1997 report of the Advisory Committee on Reactor Safeguards on the U.S. Nuclear Regulatory Commission's Safety Research Program. This report is required by Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209. This report concludes that severe budget reductions are causing substantial deterioration of the internationally respected capability of the U.S. Nuclear Regulatory Commission to conduct a forward-looking, effective safety research program.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

Enclosure:
Nuclear Safety Research, A Report to the U.S. House of Representatives and the U.S. Senate, by the Advisory Committee on Reactor Safeguards of the U.S. Nuclear Regulatory Commission, dated February 1998

NUCLEAR SAFETY RESEARCH

A Report to the U.S. House of Representatives and the U.S. Senate

by the

Advisory Committee on Reactor Safeguards

of the

U.S. Nuclear Regulatory Commission

February 1998

The Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209, requires that the Advisory Committee on Reactor Safeguards of the U.S. Nuclear Regulatory Commission report annually to Congress on the status of nuclear reactor safety research. This is the 1997 report of the Advisory Committee on Reactor Safeguards.

This report concludes that severe budget reductions are causing substantial deterioration of the internationally respected capability of the U.S. Nuclear Regulatory Commission to conduct a forward-looking, effective safety research program. As we described in the report of 1996, this deterioration is occurring at a time when the U.S. nuclear power industry is undergoing substantial changes in response to economic deregulation made possible by the Energy Policy Act of 1992. These changes may have safety implications that must be addressed by the Commission. Research is needed to ensure that the agency effectively addresses these changes. The deterioration in research capabilities is also inhibiting the ability of the Commission to continue the evolution of nuclear reactor regulation to a risk-informed, performance-based structure. Finally, the Commission's core capability in nuclear waste research has been dramatically reduced. Further reductions could inhibit the Commission staff's effectiveness and timeliness in conducting reviews of the nuclear waste repository program and cause delays and additional expenditure of National resources.

Background

The use of nuclear energy to provide electricity to the civilian population was pioneered in the United States. This technology has now spread among the developed nations of the world and all indications are that it will also be adopted by developing nations in the future. Today, the majority of the 450 operating nuclear power plants and plants under construction throughout the world are based on U.S. technology.

It was, of course, well recognized in the initial applications of nuclear energy for civilian purposes that the health and safety of the public must be adequately protected. Because there was at the time so little experience with such a new technology, very conservative, prescriptive regulations emphasizing a defense-in-depth approach to safety were established to control the civilian use of

nuclear power and the management of nuclear waste. Overly conservative regulations that do not have safety significance serve only to inhibit the fruitful application of the technology. Congress recognized, however, that even the most stringent regulations might not anticipate all the safety issues of a new technology. Congress, therefore, encouraged safety research to further develop and refine the regulation of nuclear power. Recently, Congress has encouraged all regulatory agencies, including the Nuclear Regulatory Commission, to assess and refine regulatory actions to ensure that the costs and burdens imposed by regulatory actions are commensurate with the derived societal benefit.

Since the early days, nuclear power has become an essential, reliable contributor to the Nation's energy supplies. Today, nuclear energy provides about 20 percent of the overall electrical energy in the country, and it does so with very low emissions of particulate and gaseous pollutants. There are regions of the country where nuclear power is the dominant source of electrical energy. In some countries, nuclear power is an even more important source of electrical energy. Along with its role in the development and dissemination of this technology, the U.S. has become the world leader in nuclear safety. This leadership is due in no small part to the thorough safety research that the U.S. Nuclear Regulatory Commission has been able to perform in the past. This safety technology contributes significantly to the acceptance and purchase of U.S. nuclear technology by other nations. Well researched, well maintained standards and regulations for the nuclear fuel cycle, such as those developed by the Commission, reduce the potential for the proliferation of nuclear weapons materials as the worldwide use of nuclear power expands.

The Situation Today

While use of nuclear energy in the United States is not growing, the U.S. nuclear industry is by no means static. The industry is, in fact, undergoing substantial change. Changes due to modern technical developments such as the "digital revolution" in the instrumentation and control of nuclear reactors are to be expected and will improve both safety and efficiency if properly implemented. The changes that occur as nuclear power plants age must be addressed to ensure continued safety and reliability. Of more importance, and a definite source of greater uncertainty, is the change in the nuclear industry caused by economic deregulation. The pressures of increased competition will produce changes that could well have safety implications. Certainly, steps taken

by the industry to reduce manpower and to enhance the productivity of the remaining personnel need to be scrutinized closely and researched for safety significance. Similar comments can be made about steps being taken by the nuclear industry to extend the lifetime of nuclear fuel and to diversify the suppliers of nuclear fuel for individual plants.

The nuclear industry also faces the challenge in the future of a growing volume of spent nuclear fuel. Repositories for the disposal of nuclear wastes are prerequisites for the sustained use of nuclear power. Radioactive disposal facilities will also be crucial for continued use of nuclear materials in medicine, other industries, and scientific research.

If regulation is not to stifle economic and technical improvements in the U.S. nuclear industry, the U.S. Nuclear Regulatory Commission must be in a position to modernize its safety regulations. Modernization will also be essential if the United States is to maintain its world leadership as a supplier of both nuclear technology and nuclear safety technology. There are now about 3,000 reactor-years of operational experience in the commercial use of nuclear power. The Commission has fostered through research the development and refinement of systematic methods to collect operational safety data, to assess these data, and to combine the data sets into integrated evaluations of the safety of nuclear power plants. On the basis of the data and analyses, the Commission is now undertaking an important evolution of its regulations to a risk-informed, performance-based structure. The Commission is the leader, in fact, among this country's regulatory agencies and within the world's nuclear regulatory agencies at rational regulation that focuses efforts on topics of the greatest safety significance and assures that regulations are commensurate with the derived societal safety benefits.

Over the last year, the U.S. Nuclear Regulatory Commission has initiated changes to its regulations and practices that encourage risk-informed considerations to be taken in the development of technical specifications for nuclear power plants, in-service inspection and testing of safety systems, and graded quality assurance programs for safety systems. Pilot applications of these efforts to improve regulations are being conducted and the results are now being assessed. These moves toward risk-informed regulation are expected to improve safety and regulatory efficiency. They are also expected to reduce costs to the nuclear industry and to the American public. For example, the recent move to performance-based containment leak rate testing is

expected to produce cost savings approaching a billion dollars over the projected lifetimes of existing plants.

Innovations being made today by the U.S. Nuclear Regulatory Commission in its regulations are made possible by the research that has been done in the past. The Commission has, in fact, a good record regarding the prudent identification of important safety research issues and the effective conduct of research. During the last year, for example, past results from the research program have enabled the Commission to assess industry arguments concerning required inspections of reactor vessel welds. Potential problems identified by the research program have led to requirements for additional attention to the qualification of motor-operated valves in existing nuclear power plants. Past research has also made possible the certification of two new nuclear power plant designs: the General Electric Advanced Boiling Water Reactor and the ABB-CE System 80+ pressurized water reactor. The research program is contributing to the evaluation of the advanced light water reactor design now being proposed by Westinghouse for certification by the Commission.

The Crisis in Nuclear Safety Research

Despite the substantial changes the nuclear industry is undergoing, the budget available for the conduct of regulatory activities by the Nuclear Regulatory Commission is decreasing. In the face of declining resources, priority, of course, must be given to operational activities such as effective monitoring and inspection of licensees and the disposition of current licensing actions. Recent, well-publicized events at particular nuclear facilities have underscored the priority that needs to be given to such continuing vigilance. Consequently, many of the longer term benefits that could come from research have had to be deferred. The resources available for research have decreased disproportionately in the last several years. The research program has sustained reductions of 23 percent in 1996, 19 percent in 1997, and 16 percent in 1998. The declining resources available for needed research are having impacts now. Examples include:

- A program to monitor industry research and to anticipate initiatives that may require revisions of regulations in the future has not been undertaken. The Nuclear Regulatory Commission is being forced into a position where it must wait and react to industry

proposals and thereby delay implementation of innovations even if these initiatives improve safety. Delays have already been encountered in the implementation of revised accident source terms and new dosimetry methods because the Commission cannot afford to complete needed research. Delays caused by deferred research on risk-informed pilot projects have distressed the nuclear industry whose hopes for improved regulations in the near future have begun to dwindle.

- Research needed to evaluate the potential for safety-significant human errors, especially as the nuclear industry "downsizes" staff in response to economic deregulation, remains in the planning stages despite continuing evidence from plant operations that human errors are important contributors to off-normal events at nuclear power plants.

- The technology has not been developed to extend systematic evaluations of risk from normal power plant operations to shutdown and low power operations despite evidence that these are modes of operation that pose risk to the public comparable to that from power operations.

- Research needed to evaluate licensee proposals to extend the lifetime of reactor fuel, which will also reduce the societal burden of spent nuclear fuel, remains to be performed.

- Safety research that will be needed to regulate the use of mixed oxide fuels as a means for the disposal of the Nation's excess weapon grade plutonium has not been initiated.

- The program to develop a technical understanding of public health and safety risks posed by severe reactor accidents may have to be terminated prematurely. Research on the safety and risk significance of fires has been deferred. The ability of the Commission to leverage dwindling research resources by collaboration in initiatives by other countries with more ambitious research programs may be jeopardized.

- Validation of industrial standards to use in place of Government-formulated regulations will be slowed.

- Key elements of a well-designed research program to assist in the licensing of a high-level nuclear waste repository are being adversely impacted by Congressional funding reductions. Without the research results that reduce uncertainties, it may be necessary to add conservatism, and thus raise costs for the design of the waste repositories to ensure adequate protection of the public health and safety.

- Fifteen of the generic safety issues identified since the 1979 amendments to the Energy Reorganization Act of 1974 have still not been resolved.

Deficiencies in the research program that the U.S. Nuclear Regulatory Commission can afford to maintain will affect the performance of line organizations responsible for ongoing regulatory activities with licensees. Even today, requests or "user needs" for research by line organizations are being withheld because it is known that the reduced research program cannot respond to such requests. Of concern now are limitations developing in the ability of the research program to conduct systematic examinations of the effectiveness of existing regulations and to identify additional areas for risk-informed, performance-based improvements. There are also concerns about the availability of financial resources to sustain safety research on emerging digital technologies. Without advanced safety research, application of these superior technologies to the instrumentation and control of nuclear power plants will be delayed, along with attendant improvements in safety and plant performance.

Conclusions

The U.S. Nuclear Regulatory Commission and the safe regulation of nuclear power plants have benefited from research done in the past. Reductions in the Commission budget have forced serious cutbacks in the research program and deterioration of the research capability. The Commission still needs a research program. It certainly needs a viable program to be able to evaluate proposals independently and to assess safety arguments advanced by the industry. It needs a stronger research program to continue the evolution of its safety regulations. The Commission also needs a research program to meet new obligations it is undertaking. Notable among the new obligations is the implementation of safety regulations for a geologic repository for

spent nuclear fuel. The agency is also conducting a pilot program to assess the viability of undertaking the safety regulation of certain Department of Energy nuclear facilities.

The Nuclear Regulatory Commission capacity for research is no longer commensurate with the agency's regulatory obligations. It will not be possible to maintain core competencies in all the areas that have historically proven to be of recurring importance in safety and regulatory actions by the agency. Modernization of regulations will be delayed because research cannot be performed to ensure that appropriately high levels of safety are maintained. Responses to industrial initiatives taken as a result of competitive pressures will be slowed without a broader research program. Delay in the implementation of cost competitive innovations may well force the nuclear industry to retire more plants prematurely, and the Nation will incur all the societal costs such unnecessary retirements entail. The development of a high-level nuclear waste repository is facilitated by the availability of well-researched safety regulations and analytical tools for licensing. Uncertainties left when research cannot be done because of funding constraints may delay the development of the repository or force the addition of costly conservatism.

In summary, there are benefits to the entire society that may be delayed or even lost as the research capability of the U.S. Nuclear Regulatory Commission deteriorates in response to declining financial resources.

LICENSE RENEWAL

DR. MARIO FONTANA
ACRS

LICENSE RENEWAL

- NRC amended 10 CFR PART 54, "License Renewal Rule" in May 1995
- Rule focuses on the effects of aging on Long-Lived Passive Structures and Components (SC)
- The License Renewal application requires Time-Limited Aging Analyses (TLAAs), and Integrated Plant Assessment (IPA)
- Reverification of current licensing basis (CLB) compliance is not required. CLB carries forward into extended period.
- Focus on managing the effects of aging during the period of extended operation

NRC STAFF ACTIVITIES

- o Review industry and Owners' Group Topical Reports
- o Standard Review Plan (SRP)
  - CHAPTER 1: Administrative Review
  - CHAPTER 2: Screening Methodology for Identifying SC Subject to Aging
  - CHAPTER 3: IPAs - Aging Management Review (AMR)
  - CHAPTER 4: TLAAs
- o Regulatory Guide (RG) [based on NEI-95-10 document]
- o Inspection program for license renewal (LR)
- o Environmental review of 10 CFR PART 51, that includes SAMDA, ESRP, Generic Environmental Impact Statement (GEIS)

INDUSTRY ACTIVITIES

- o NEI-95-10 document to provide acceptable approach to implement the requirements of 10 CFR Part 54

- o Expected applications
  - BG&E — CALVERT CLIFFS
  - DUKE POWER — OCONEE
  - SOUTHERN NUCLEAR — HATCH
  - OTHERS

- o Topical Reports

- o FSAR Supplement

- o Technical Specification Changes

- o Environmental Report Supplement

ACRS PLANS

ACRS Plans to Review:

- o SAMDA and source-term issues
- o Selected industry Topical Reports
- o Environmental issues, transportation, impacts on biota other than man
- o Research activities regarding aging
- o Updated SRP and regulatory guides
- o NRC staff's evaluation of license renewal applications

X

POLICY ISSUE
(Information)

September 25, 1997 SECY-97-216

FOR: The Commissioners

FROM: L. Joseph Callan
Executive Director for Operations

SUBJECT: STANDARD REVIEW PLAN FOR LICENSE RENEWAL

PURPOSE:

To inform the Commission of the status of the development of a standard review plan for license renewal.

BACKGROUND:

The Commission amended the license renewal rule (10 CFR Part 54) in 1995. As discussed in SECY-97-118, dated June 5, 1997, the staff and industry are engaged in a number of activities associated with implementing this rule. One of these activities is the development of a standard review plan for license renewal (SRP-LR).

The staff placed a revised working draft of the SRP-LR in the NRC's Public Document Room (PDR) and will publish the draft SRP-LR after it reviews several applications for license renewal.

DISCUSSION:

The SRP-LR offers guidance to the staff for reviewing applications for license renewal. A working draft of the SRP-LR, reflecting the proposed rule amendment, was made publicly available in the PDR on December 13, 1995.

Contact: S. Lee, PDLR/NRR
415-3109

SECY NOTE: To be made publicly available in 5 working days from the date of this paper.

As directed by the Commission in a June 28, 1993, staff requirements memorandum, the staff incorporated the technical information and agreements from its earlier review of Nuclear Management and Resources Council (NUMARC, now the Nuclear Energy Institute (NEI)) industry reports addressing license renewal in the working draft of the SRP-LR.

Since then, the staff and industry have gained additional experience with the implementation of the amended rule. The staff has revised the working draft of the SRP-LR to incorporate the following information:

(1) experience gained to date from the staff's review of plant-specific technical reports and owners group topical reports addressing license renewal

(2) experience gained from the staff's development of a draft regulatory guide for license renewal which proposes to endorse NEI 95-10, Revision 0, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule"

(3) the final amended rule, including administrative requirements

(4) the staff's preliminary assessment of the recent 10 CFR 50.55a rule amendment on containment inservice inspections with respect to license renewal requirements

(5) the staff's evaluation of the generic safety issue (GSI) on metal fatigue

(6) the staff's evaluation of its Structural Action Plan

(7) industry comments on NUREG-1557, "Summary of Technical Information and Agreements fro Nuclear Management and Resources Council Industry Reports Addressing License Renewal," which documents technical information from the NUMARC industry reports

(8) development of national codes and standards addressing aging management of safety-related electric equipment.

The staff plans to use the working draft of the SRP-LR in its future reviews of plant-specific technical reports and owners group topical reports, as appropriate. By focusing on the review of aging management of actual plant structures and components, the staff and the industry would gain a better understanding of the implementation issues involved in the license renewal rule. Lessons learned from the trial use of the working draft of the SRP-LR and draft regulatory guide for license renewal would provide feedback to further improve the working draft. As such, the working draft of the SRP-LR is a living document.

Some examples of areas in which staff review guidance would be further developed based on experience with the working draft of the SRP-LR are:

(1) docketing of a timely and sufficient renewal application

(2) crediting existing programs for managing aging of long-lived passive structures and components

(3) establishing review criteria necessary to make the finding in 10 CFR 54.29.

The staff plans to focus on plant-specific and owners group reviews of actual plant structures and components to gain needed experience with implementation of the rule and the working draft of the SRP-LR. As discussed in SECY-97-118, the staff intends to gain additional experience before proceeding with formal approval of the SRP-LR and regulatory guide for license renewal. During the review of technical reports and initial renewal applications, the staff, licensees, owners groups, and other interested parties will identify technical and procedural issues resulting from the trial application of the working draft of the SRP-LR. The staff will develop revisions to the working draft in conjunction with the resolution of issues related to the staff's evaluation of the particular affected technical reports or renewal applications. At appropriate times, the staff will correspond with NEI and will arrange public meetings to address (1) the generic aspects of both technical and procedural issues and (2) possible changes to NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule." Policy issues will be referred to the Commission, for resolution. The resolution of all of these issues will be documented in the supporting justification for the SRP-LR and the regulatory guide for license renewal.

As discussed in SECY-97-118, the staff plans to publish the draft SRP-LR after it reviews several applications for license renewal. The draft SRP-LR is tentatively scheduled to be published for public comment in early 2001.

<u>COORDINATION</u>:

The Office of the General Counsel has reviewed this paper and has no legal objection to it.

<u>RESOURCES</u>:

This paper presents the status of the development of the SRP-LR and does not involve changes in resource requirements. The Office of the Chief Financial Officer has reviewed this Commission paper for resource implications and has no objections.

INFORMATION TECHNOLOGY:

No impact on information management or information technology is anticipated.

L. Joseph Callan
Executive Director
for Operations

DISTRIBUTION:
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EDO
SECY

FIRE PROTECTION RULEMAKING

DR. DANA A. POWERS
ACRS

ACRS VIEWS

- o Views not completely developed (e.g., fire protection requirements for decommissioning plants and shutdown operations, etc.)
- o Premature to devise an alternative or substitute for existing regulations on fire protection
- o Useful to collect and refine guidance on existing regulations
  - Not a high priority
- o Need strategy for CLB changes to fire protection under RG 1.174
- o Encourage NFPA to augment fire protection standard development with risk information
- o Develop strategies to review IPEEE findings:
  - Is Appendix R Adequate, or
  - Are IPEEE Methods Adequate

ACRS PLANS

- o Review fire protection status at Brown's Ferry and in Region II (Comanche Peake, & Region IV done previously)
- o Review draft IPEEE Insights Report
- o Review Quad Cities and the SISBO response at other plants
- o Review draft NFPA fire protection standard
- o Review results of Functional Fire Protection Pilot Inspections
- o Review RES plans for fire protection risk analysis
- o Consider draft White Paper on ACRS views regarding fire protection

September 11, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

FROM: John C. Hoyle, Secretary /s/

SUBJECT: STAFF REQUIREMENTS - SECY-97-127 - DEVELOPMENT OF A RISK-INFORMED, PERFORMANCE-BASED REGULATION FOR FIRE PROTECTION AT NUCLEAR POWER PLANTS

This is to advise you that the Commission has reviewed the subject paper and agreed on the following approach. The staff should finalize the current research and study by the end of this year, as noted in the paper. The staff should then obtain OGC feedback on the backfit implications and industry feedback on interest in a rule and present this information in a briefing to the Commission. The briefing should incorporate all findings, observations, and conclusions to that point, including, but not limited to, PRA and fire modeling results, fire protection functional inspection(s) results, IPEEE (fire) results, backfit determinations, industry interaction and comments, and other relevant information.

(EDO) (SECY Suspense: 2/13/98)

The staff should provide the Commission a schedule for expedited rulemaking.

(EDO) (SECY Suspense: 10/10/97)

The staff should expedite the resolution of issues necessary to formulate a proposed rule which will eliminate the need for most of the 850 exemptions granted under current rules and which takes a more risk-informed (as opposed to deterministic) and a more performance-based (as opposed to prescriptive) approach where that is appropriate and justifiable. However, the staff should not force-fit risk-informed, performance-based elements into areas that are not amenable to such approaches. In the development of the a fire protection rule and performance objectives the staff should fully consider and develop an approach consistent with the current state of fire modeling and PRA usage in fire protection programs.

The responsibility for this rulemaking effort should be shifted from Research to NRR in accordance with the guidance in DSI-22. The staff should continue to coordinate additional research (performed cooperatively with industry, if possible) as necessary to complete any longer term items, or improvements to regulatory guidance in support of further risk-informed efforts. The staff should assess the current regulatory requirements so as not to eliminate current requirements that continue to be appropriate during the transition to more risk-informed fire protection

requirements.

The Commission should be informed of significant policy and technical issues that arise as a result of staff efforts that impact the schedule.

cc: Chairman Jackson
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
OGC
CIO
CFO
OCA
OIG
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR
DCS

NFPA
1896-1996

National Fire Protection Association

1 Batterymarch Park, Quincy, Massachusetts 02269-9101 USA
Telephone (617) 770-3000 Fax (617) 770-0700

George D. Miller
President

November 7, 1997

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

Dear Chairman Jackson:

We have recently learned that the Nuclear Regulatory Commission (NRC) plans to develop a new fire protection rule under a very rapid schedule. This is to advise you that the National Fire Protection Association (NFPA) is developing a standard (NFPA 805, "*Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants*") which covers the same subject. Members of the NRC staff, Messrs. Patrick Madden and Edward Connell are participating in this effort. This NFPA effort is due to be completed in May 2000. Accordingly, it appears that there may be an opportunity to combine these two efforts.

U.S. Government Circular OMB A-119 (*Federal Register* Volume 58, No. 205/10-26-93, page 57648) and last year's Public Law 104-113, "*Technology Transfer Act*" encourages U.S. Government adoption of national consensus standards and participation in voluntary standards development. Section 12, Subsection (d)(1) of P.L. 104-113 states: "*In General. Except as provided in paragraph (3) of this subsection, all federal agencies and departments shall use technical standards that are developed or adopted by voluntary consensus standards bodies, using technical standards as a means to carry out policy objectives or activities determined by the agencies and departments.*" It is our opinion that the intent of this new law and OMB A-119 is for NRC to ultimately use the NFPA standard as NRC criteria (provided the Commission determines the standard meets the intent of the law).

The NFPA is an International, non-profit Association which develops consensus fire codes and standards that are accredited by the American National Standards Institute (ANSI) as American National Standards. We feel the proposed NFPA 805 standard will be consistent with OMB A-119 and the new Public Law and will serve the NCR's future needs.

Therefore, we would like to initiate discussion with NRC to promote a high degree of cooperation with the NRC to develop any new rule and guidance in a timely manner. NFPA would be willing to meet with you or any of your advisory groups such as ACRS to discuss the subject or make a presentation.

Thank you for this opportunity to bring this issue to your attention and we look forward to hearing from you soon.

Sincerely,

George D. Miller

George D. Miller
President & CEO

GDM:wh

c: Commissioner Greta Joy Dicus
U.S. Nuclear Regulatory Commission

Commissioner Nils J. Diaz
U.S. Nuclear Regulatory Commission

Commissioner Edward McGaffigan, Jr.
U.S. Nuclear Regulatory Commission

NUCLEAR ENERGY INSTITUTE

Ralph E. Beedle

December 11, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Project Number: 689

Dear Mr. Callan:

A significant element of the approach to performance-based rulemaking for fire protection outlined in the Staff Requirements Memorandum (SRM) to SECY 97-127 is a determination of industry interest in fire protection rulemaking. We conducted a survey of the industry to establish a position on this issue and identify specific concerns. This letter summarizes the industry position based on responses from all Chief Nuclear Officers.

First, a new fire protection rule is not desired, nor is one considered necessary to assure or improve safety. The industry acknowledges that the current fire protection regulations have been prescriptive, generally inflexible, and difficult to implement, especially in the case of Appendix R. Having expended tremendous resources over the past 15 years to comply with those rules, and provide processes to assure continued compliance, the safety benefit of a new regulation is not obvious nor is the cost justified. Licensees are almost certain to incur additional cost with any reexamination of a new fire protection rule.

We understand some in the NRC perceive the numerous exemptions to Appendix R as an indication that the regulation is flawed. Yet, the exemption process was a key consideration of the Commission and the Federal Courts in determining that the rule was acceptable for promulgation. There was a conscious recognition that the rule would result in the installation of large amounts of piping, valves and circuitry in plants that were already built. The exemption process was viewed to be absolutely necessary for those instances when plant-specific configurations made it unduly burdensome to conform with the physical requirements of the rule. In such cases, the exemption process allowed licensees to seek NRC approval of alternate means of establishing the requisite level of safety. An exemption did not result in licensees waiving the applicability of the specific elements of the rule.

Most of the exemptions were obtained early in the process of complying with the regulation, and these exemptions have been integrated into plant licensing bases. A great deal of experience in using this process has accumulated over the years. Elimination of currently approved exemptions without due consideration would have a significant impact.

Another factor that made the initial implementation of Appendix R difficult and costly was the evolving NRC guidance that established the basis for compliance. Despite best of intentions, implementing a new rule would undoubtedly result in a similar resource intensive iteration process between licensees and NRC staff as new expectations and clarifications evolve. A mature industry in a time of increasing competition, finite resources, and reduced cost margins can ill afford a churning regulatory process with so little apparent gain.

NRC staff have stated that a high core damage frequency from fire sequences at a plant that is in compliance with Appendix R is further indication of a flawed rule. Industry believes that such results, and the plant-specific efforts to address them, are instead evidence of the success of the Individual Plant Examinations of Internal and External Events (IPEEE). The studies were designed to find such vulnerabilities. These insights are most often plant-specific, not generic, and are further evidence that compliance with existing regulations has generally led to effective fire protection and safe shutdown programs.

We recognize the industry position on fire protection rulemaking is different from that previously communicated to the NRC when NEI petitioned for fire protection rulemaking (proposed Appendix S to 10 CFR Part 50) in 1995. There are two primary reasons for this change. The SRM to SECY 97-127 implies a very strong desire to have a new rule that is mandatory; not optional. This is in contrast to the NRC position in 1994. An understanding that a new fire protection rule would be optional for current licensees was a basic assumption in proposing Appendix S.

The other reason for a changed position on rulemaking is the recent experience in applying risk-informed, performance-based methods in other technical areas, and the promulgation of draft risk-informed regulatory guidance. Coupled with NRC staff concerns about the maturity of risk and performance tools for supporting a risk-informed, performance-based fire protection rule, it appears such techniques offer even less opportunity for a cost effective outcome.

Any consideration for rulemaking should be based on the three key activities currently in progress. NRC staff review of fire IPEEEs, NRC staff conduct of Fire Protection Functional Inspections, and the National Fire Protection Association's (NFPA's) development of a performance-based standard are essential elements in determining the need for a new rule.

The pilot Fire Protection Functional Inspections and fire IPEEEs should be completed and evaluated to determine if there are generic safety issues that dictate a need for a new regulation or modifying existing regulations or guidance.

Similarly, the NFPA's effort to develop a performance-based, consensus nuclear plant fire protection standard should be factored into any new rule development. The current schedule for NRC rulemaking activity outlined in SECY 97-127 does not permit this.

Another key point from the survey is that while the use of risk and performance techniques have merit and should be pursued, a new rule is not required to do so. It makes sense to focus regulations on issues of safety significance. For fire protection, the opportunity to apply these concepts on a case-by-case basis within the context of the exemption process for the existing regulation appears more cost-effective than embarking on the development of a new regulation. Most plants are at a mature stage in their operating lifetimes and have considerable experience with the current regulations; they generally do not see great gain at this stage from a new risk-informed, performance-based rule.

Finally, more meaningful alternatives to rulemaking exist that could improve the fire protection regulatory guidance and practices. For instance, recent NRC staff interpretations of fire protection requirements relative to spurious actuation and reactor coolant pump lube oil collection systems that differ from that of licensees indicate a need for better understanding between industry and NRC. A useful task would be for NRC staff and industry to identify and then clarify those items in the current rules and regulatory guidance where NRC expectations and utility implementation are not well aligned.

We look forward to continued dialog between NRC and the industry as this important issue moves forward. Please call me, or have your staff contact Fred Emerson at 202-739-8086, with any questions about this information.

Sincerely,

Ralph E. Beedle

RB/FAE/djm

c: Mr. Sam Collins, NRC
Dr. Brian Sheron, NRC

Mr. Gary M. Holahan, NRC
Dr. Robert L. Seale, ACRS