UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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

Location: Rockville, Maryland

Date: Thursday, June 8, 1995

Pages: 1 - 63

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6 ON REACTOR SAFEGUARDS (ACRS) - PUBLIC MEETING

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10 Nuclear Regulatory Commission

11 One White Flint North

12 Rockville, Maryland

13

14 Thursday, June 8, 1995

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17 The Commission met in open session, pursuant to

18 notice, at 9:30 a.m., Ivan Selin, Chairman, presiding.

19

20 COMMISSIONERS PRESENT:

21 IVAN SELIN, Chairman of the Commission

22 KENNETH C. ROGERS, Commissioner

23 E. GAIL de PLANQUE, Commissioner

24 SHIRLEY A. JACKSON, Commissioner

25

1 STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

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3 JOHN HOYLE, Secretary of the Commission

4 MARTIN MALSCH, Deputy General Counsel

5 DR. THOMAS S. KRESS, CHAIRMAN, ACRS

6 DR. ROBERT SEALE, ACRS

7 DR. IVAN CATTON, ACRS

8 DR. WILLIAM SHACK, ACRS

9 DR. DANA POWERS, ACRS

10 MR. JAMES CARROLL, ACRS

11 DR. DON MILLER, ACRS

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

2 [9:30 a.m.]

3 CHAIRMAN SELIN: Good morning, ladies and

4 gentlemen.

5 The Commission is very pleased to welcome Dr.

6 Kress and the Members of the Advisory Committee on Reactor

7 Safeguards which we will call ACRS for the rest of the

8 morning, and they have the opportunity to discuss a rather

9 large number of issues that are of importance to the

10 industry today.

11 Since this will be the last meeting I will Chair

12 of the ACRS, I would like to express my personal

13 appreciation for the very valuable service on a large number

14 of issues that you have rendered over the years. The

15 service has always been very high, and I particularly

16 appreciate how much more flexible the working conditions

17 between the Committee and the Staff, in particular, have

18 become, so that I do believe that your services to the

19 Commission have remained at a very high level, but the value

20 to the Agency as a whole have been increased by your

21 willingness to work with whoever is at the appropriate level

22 and set of people to work with.

23 In particular, I would like to commend the

24 extraordinary amount of work that was done in a very short

25 time to help with the certifications of the two advanced

1 reactors. I also would like to recognize three newly
2 appointed members of the ACRS and to present their
3 certificates to them.
4 We have Dr. George Apostolakis, who brings to the
5 Committee some great experience in PRA and human factors
6 expertise; we have Dr. Mario Fontana, who brings some
7 digital instrumentation and controls expertise; and Dr. Don
8 Miller, who brings us a nuclear plant system and reactor
9 safety background. We would like to welcome you to the
10 committee.
11 We will start with George.
12 [Certificates were presented and pictures were
13 taken.]
14 CHAIRMAN SELIN: Dr. Kress, would you kindly
15 proceed.
16 MR. KRESS: Yes. Thank you for those kind words
17 earlier, and on behalf of the Committee, I would like to say
18 it has been a pleasure working with you and Dr. de Planque,
19 and we are going to miss you when you are gone, and we wish
20 you all the best.
21 CHAIRMAN SELIN: Thank you.
22 MR. KRESS: With that, we will turn to the first
23 item on our agenda which is the thermal hydraulic issues
24 related to the passive plant designs. That is Ivan Catton's
25 subject matter.

1 We are going to get started.

2 MR. CATTON: There are several things I would like

3 to just briefly mention.

4 First, the NRC test and analysis programs, and

5 that is mainly RES. Our most recent review is documented in

6 our letter of 12 April. In general, we believe there has

7 been a dramatic improvement in the research program. I

8 believe this is due to the new management and the existence

9 of a Thermal Hydraulic Advisory Committee that is made up of

10 probably the best in the business.

11 There are still some tough issues that remain to

12 be dealt with, and I will just briefly mention them. The

13 RELAP5 code still needs to be augmented so that it can treat

14 condensation and thermal stratification both, which are

15 important issues to the AP600. With our meeting it became

16 clear to us that there is a potential for water hammer in

17 the AP600, and this needs some attention.

18 We are quite concerned about the ability of codes

19 like RELAP5, which includes TRAC and others, to do the kinds

20 of computations that are needed for AP600, and that is

21 because the accident goes on for such a long time and the

22 numerics are probably not the best for that kind of

23 circumstance. Whether or not they can do a complete

24 calculation, in my mind, is kind of up in the air.

25 RES has responded to our letter, and their

1 response indicates that they are addressing all three of
2 these areas. I just mention them in passing.
3 The Westinghouse test and analysis programs, there
4 are several parts to it, the most recent first, we reviewed
5 the AP600 passive containment and analysis program. We had
6 a two-day meeting at the end of March, and we are left with
7 several issues that could be difficult to deal with.
8 The first, you have to do computations in order to
9 predict the behavior of the containment, and in order to do
10 it, Westinghouse assumes that the containment volume, and
11 this is a million cubic feet, is well mixed in spite of data
12 and analysis showing stratification. Why that is important
13 is because stratification reduces the heat transfer, and it
14 is getting the heat out that maintains the pressure.
15 The second is the main steamline break in the
16 containment. They operate under the assumption that it is
17 forced convection, and there is no way I know of to find the
18 velocity to stuff into the correlations that have to be used
19 in the code.
20 Third, the containment annulus in their test
21 facility I don't believe was scaled appropriately, and the
22 scaling they used leads to higher heat transfer per unit
23 volume in the containment. I think this is going to have to
24 be dealt with.
25 There are several questions about external surface

1 film cooling. You know how this containment works, it is
2 just a huge sphere and you pour water on the top of it.
3 There are questions. First, is the impact of preheating
4 this surface, because the water doesn't get turned on
5 straight away, so it is actually hot when they pour the
6 water on it. There are inconsistencies between the models
7 they use, this is the physical models in their computer
8 code, and observations. There are concerns about how one
9 should scale the amount of water to be sure that their
10 correlations are valid that are being used in the code.
11 Finally, the code they use is called GOTHIC and
12 this was developed by EPRI. It just runs too slow to do a
13 full calculation of the containment behavior. I forget the
14 numbers, but maybe one of my colleagues remembers, weeks is
15 what comes to mind to do a single complete calculation.
16 Something is not right, and I think the problem is the use
17 of canned codes. You take GOTHIC, you try to make your
18 system fit into it. You really ought to do a little
19 thinking of how you do your calculation. That is going to
20 be a problem.
21 Another area that we have reviewed, and we just
22 recently wrote a letter was on the COBRA/TRAC for a large-
23 break LOCA and NOTRUMP if a small-break. These are separate
24 codes. NOTRUMP is a code developed by Westinghouse.
25 COBRA/TRAC actually is a spinoff of NRC work that

1 Westinghouse has adapted to their own needs.

2 We know very little about NOTRUMP, which will be
3 used for small-break which is a higher probability event.
4 We are in the middle of discussions with the Staff and
5 Westinghouse about how COBRA/TRAC should be dealt with. The
6 problem here is, it is the first review of a best-estimate
7 code as a best-estimate tool. The first one under the rule
8 that was promulgated, I think, in '88 or '89, and where the
9 pinchpoint is coming is how you deal with words that are in
10 the rule that say high probability of non-exceedence, and it
11 is a definition of "high probability," what is it?

12 We believe its meaning should be stated somehow in
13 terms of both probability and confidence. The relevant reg.
14 guide says the probability of non-exceedence should be 95
15 percent and is silent on confidence. Our view is that the
16 present path with lead to low confidence, and this is mainly
17 because of the limited number of data points that are
18 available for Westinghouse to use in coming to this
19 conclusion. There are other ways they could have addressed
20 it. They chose a particular path. At present, the Staff
21 agrees with that particular path and we disagree.

22 There are three other submittals of best-estimate
23 codes that are about to go under review, and in our letter
24 to them we ask that they not start that review until we have
25 somehow ironed this process out.

1 MR. CARROLL: Ivan, you mentioned that they have
2 chosen a particular path. It might be helpful to elaborate
3 on what path we believe would lead to the right answer.
4 MR. CATTON: Well, we think that they should state
5 both probability and confidence, and there is the CSAU
6 process that was developed prior to promulgation of the
7 rule.
8 See, in 1983, there was a SECY that gave an
9 interim method for trying to bring some of what we knew to
10 bear on the process. The rules was not promulgated until a
11 method was developed and they could demonstrate that,
12 indeed, you could come up with both a probability and
13 confidence. The method was called the CSAU. After it was
14 in place, the rule was promulgated. Westinghouse could well
15 have chosen that path. There are probably others.
16 I am not sure that is what you were getting at?
17 MR. CARROLL: Yes, that was what I was getting at.
18 MR. CATTON: The reason that I feel that this is
19 important is because there are a large number of existing
20 plants that could take advantage of being able to use a
21 best-estimate calculation. To me, that is probably more
22 important than the AP600 or anything else because those
23 plants are there and they are there now, and we are trying
24 to push this forward as quickly as we can, mainly because of
25 those plants.

1 COMMISSIONER ROGERS: How difficult do think it is
2 to do the confidence? Is that a lot more difficult
3 responsibility? It usually is.
4 MR. CATTON: It just depends on the amount of data
5 you have.
6 COMMISSIONER ROGERS: Well, yes, but is there
7 adequate data out there?
8 MR. CATTON: Not to do it they way they are doing
9 it. The main reason is, this is large-break LOCA, so there
10 is a whole series of things that happen. First, you have
11 the blowdown phase and you have a peak temperature in the
12 blowdown phase. This is followed by reflood where you catch
13 it on the rise from the decay heat. There is very little
14 data that covers the whole span.
15 So what happens to you is, you can find enough
16 data for the blowdown phase, and there is lots of data for
17 the reflood phase, but there is very little that has both.
18 As a result, it is awful hard to argue probability and
19 confidence, and that's where the problem is.
20 COMMISSIONER ROGERS: You think that more
21 experimental work has to be done to get that?
22 MR. CATTON: No. It is a matter of how you
23 approach it. See, these big codes are really built from a
24 bunch of submodels, and a lot of empiricism. Actually, it
25 is just one big empirical tool is what it is with a lot of

1 bad numerics and bad other things. So the tuning is
2 important.
3 You take the small parts, and there is enough data
4 for each of the pieces for you to come to some conclusions,
5 and you can actually come up with distributions on the heat
6 transfer coefficient, for example, and all sorts of things,
7 and then you can put these together, and you propagate them
8 through the tool that you have and come to some conclusion.
9 This was the CSAU process. And it was the only way they
10 felt you could ever get at it because of the limited data.
11 Well, there is no more data today than there was then.
12 Now you could get around this by making the
13 argument that the large-break LOCA is a very low probability
14 event, so why are we spending all this time on it, and you
15 could then argue that you should be able to accept lower
16 probability of non-exceedence and lower confidence in your
17 result. But somebody should state what those are, then go
18 for whatever this goal is; not this process of just sort of
19 working back and forth until everybody agrees, well, it
20 looks pretty good and I am confident, let's go. I don't
21 think that will wash.
22 The rule 10 CFR 50.46 says "high probability."
23 The reg guide defines high probability. But there is no
24 statement on confidence. As a result, the number of data
25 points that Westinghouse is going to use, and I believe it

1 is 13 or 14 or something, one of my colleagues made a
2 calculation and comes to the conclusion that it is 13
3 percent confidence. Well, if that is what it is and that is
4 what is acceptable, then it should be stated, not just left
5 silent. At least, I don't believe it should be.
6 That is sort of where we are at.
7 As far as their experimental test programs, of
8 which there are a large number, we have the final SPES
9 report in hand, we will soon have the AP600 simulated test
10 from Oregon State. I don't expect much from the SPES, but
11 the OSU data should go a long way to demonstrating the good
12 or the bad about the AP600.
13 CHAIRMAN SELIN: Why don't you expect much from
14 SPES?
15 MR. CATTON: I just don't like the facility. I
16 mean, first --
17 COMMISSIONER ROGERS: Is this a scaling question?
18 MR. CATTON: Well, it was more than that. I mean,
19 there are lots of things about it that bothered me. The
20 actual AP600 has two cold leg pumps and there can be
21 recirculating flow, and the SPES had the two pipes come
22 together and they actually put a divider plate between them
23 so that they could keep things going in the same direction.
24 There are lots of little things like that that are kind of
25 squirrely about the facility. Lots of heat losses.

1 You could live with some of the scaling questions
2 because the codes can smooth through that for you somewhat.
3 I was never really convinced the facility was needed,
4 although my colleagues were. I think that, in my own mind,
5 I am not too caught up in who got the data from where. A
6 lot of what is coming out of the ROSA facility has addressed
7 the questions that SPES would address. So you sort of
8 already have the answers, and it is a big headache to go
9 through a report that must be 18 inches high.
10 CHAIRMAN SELIN: I guess I should ask the question
11 differently. If you take a look at all the facilities and
12 all the data that are necessary, are you comfortable that
13 most of the data that would be needed will be available
14 regardless of where they come from?
15 MR. CATTON: Yes. It may take a lot of
16 interpretation and hammering around, but I think when all is
17 said and done and the reports are on the table, yes, I think
18 there is enough data.
19 CHAIRMAN SELIN: That is really our only
20 responsibility to be sure that we have the available data we
21 need.
22 MR. CATTON: I think our more serious problems in
23 this whole process are the codes themselves, but that is a
24 separate issue from the data.
25 CHAIRMAN SELIN: I thought there was a lot of work

1 going on at INEL and a couple of other places to see whether
2 the codes could be extrapolated to cover the AP600
3 situation. Are you not comfortable that --
4 MR. CATTON: They are having trouble with the --
5 see, the process very quickly becomes very slow, almost
6 quasi-steady. These codes just take forever to do those
7 kinds of computations and there are other things.
8 CHAIRMAN SELIN: It is a programming and structure
9 problem, not a computational problem?
10 MR. CATTON: Well, if you have enough time and
11 enough machine horsepower, you could probably get it done.
12 What that does to you is, you don't do enough calculations
13 to be sure you understand the behavior, so you are sort of
14 caught. I don't know anything about the Westinghouse code
15 NOTRUMP, so I can't address whether it will be able to do
16 these things well or not yet.
17 CHAIRMAN SELIN: Okay.
18 MR. CATTON: Part of it is because these codes
19 came from an interest in the large-break LOCA. These were
20 very rapid transients. You could do a lot of things sloppy
21 and still get results that were pretty good because certain
22 things just didn't matter.
23 CHAIRMAN SELIN: There is not enough time to do
24 all that there is.
25 MR. CATTON: The strength of the expansion process

1 that was going on just swamped a lot of the numerical
2 difficulties you were having. In order to make them run
3 right, you put excessive damping into them, but that didn't
4 matter because your forcing was so strong.
5 CHAIRMAN SELIN: So it is not just a problem with
6 the fact that the codes take too much computer time, but you
7 do have serious reservations about whether the situation
8 they described is appropriate for the AP600?
9 MR. CATTON: They are not the right code for the
10 problem. Now maybe you can make them work, maybe you can't,
11 but they are having difficulties in trying, and this is
12 coming up again with the containment with the use of the
13 GOTHIC. It is just that the computations take just too
14 long. As a result, you wind up not doing a sufficient
15 number of the computations in order to cover all
16 possibilities.
17 COMMISSIONER ROGERS: I wanted to ask you about
18 what you thought about contingencies because your letter
19 said that you thought that the Staff ought to provide some
20 kind of contingency plan.
21 MR. CATTON: The reason for that being in there
22 was this concern.
23 COMMISSIONER ROGERS: Yes. The question is, what
24 do you have in mind in the way of a contingency plan, what
25 would that involve? If new codes are developed, those codes

1 have to be validated in some way.

2 MR. CATTON: Well, you have the data. Certainly

3 it is a problem. The large-break LOCA should have been put

4 on a shelf a long time ago. This is not a new concern of

5 the ACRS. We have made this recommendation off and on over

6 the past ten years, that the kind of code and the kind of

7 numerics are not appropriate for the long quasi-steady kinds

8 of transients. You should do something else.

9 CHAIRMAN SELIN: Let me see if I understand. If

10 the models were better, we wouldn't need so many runs to

11 describe --

12 MR. CATTON: No. It has nothing to do with the

13 quality of the models.

14 CHAIRMAN SELIN: Well, you said that there were

15 large-break LOCA models that swamp a lot of settled cases

16 that we need to understand for the AP600.

17 MR. CATTON: That's right.

18 CHAIRMAN SELIN: If we had more appropriate

19 models, we might not need so many runs. We need a huge

20 number of runs, in large part, because the codes don't quite

21 fit the situation, and we can't afford to get them because

22 it takes so long.

23 MR. CATTON: No, I think 'you need the large

24 number of runs so that you can exercise over parameter space

25 that you may have to deal with as an accident. That is what

1 I meant.

2 CHAIRMAN SELIN: Even if the models had been

3 written for small-break alone?

4 MR. CATTON: It doesn't matter. I think you still

5 need to make a lot of runs if you want to fully understand

6 the behavior of your system. What happens is, you will

7 figure out how to make this code run, but you might only run

8 it once or twice because you can't afford six months on a

9 Cray. That is where the impact is. This was the reason

10 that, in the past, we have recommended that a better way of

11 doing the problems should be sought, and a better way is to

12 build in more thinking and less straight numerics into the

13 kind of code you write.

14 We put that in the letter, in part, because we

15 have discussed this research somewhat, and they are in

16 agreement that they had better give this some thought, and

17 they are. They have a rather strong advisory committee.

18 The people who are on it are some of the best. So I think

19 that if there is any way through this, they are going to

20 find it, and we will help them as much as we can.

21 I am finished.

22 CHAIRMAN SELIN: Thank you very much, Dr. Catton.

23 We should -- obviously we will, but you should as

24 questions of the members as they finish their pieces. There

25 is no sense in waiting until to the end because there is not

1 that much synergy.

2 MR. CATTON: I prefer the questions as we go

3 because if you ask them when I am done, I won't remember

4 what I said.

5 [Laughter.]

6 CHAIRMAN SELIN: Dr. Kress?

7 MR. KRESS: We are ready to move on to the next

8 subject which is also Dr. Catton's.

9 CHAIRMAN SELIN: I will give you the same advice,

10 just keep going. Don't wait for questions, they will come.

11 MR. CATTON: I will just get started. In this

12 case, I am addressing the topic for Bill Lindblad. One of

13 his many sons is graduating from MIT and he is attending the

14 graduation. So I hope you don't have very many questions,

15 and I don't plan to say a whole lot.

16 Our review, really, has just begun on the whole

17 system. Our thermal hydraulic review has been going on

18 quite a while because of anticipated problems. The review

19 of the plant has really just begun. Our first meeting was

20 in January, and the second was a week ago. Our first

21 meeting really was just more planning and Westinghouse

22 briefing us on their design philosophy and some of the plant

23 systems.

24 For the most part, there really were no issues

25 that arose. At the recent meeting, which we plan to

1 document in a letter to you, I think, during this meeting --
2 is that right, Tom?
3 MR. KRESS: You are right.
4 MR. CATTON: There are a number of issues, some
5 which I will just briefly talk about here. During
6 discussions on leak before break, it appeared to us that the
7 Staff should create a more rational set of regulations.
8 This, it seemed to us, was a good place to put in risk-
9 based regulation rather than being so prescriptive.
10 I am not going to be able to answer many questions
11 on this, and you look like you are going to ask one. We
12 discussed a little bit the passive containment system, but I
13 have already talked about that. There was some discussion
14 of regulatory treatment of non-safety systems, and here, if
15 you have any questions, I will defer it to Tom.
16 We were pleased to hear that the vessel will be
17 cooled externally in the case of a core melt accident. I
18 think this is a sensible thing to do. There are some things
19 that need to be done, sort of proof of concept or some heat
20 transfer questions that need to be answered, but I think
21 these are on the way.
22 Again, we are preparing a letter, you should have
23 it next week.
24 Tom, back to you.
25 CHAIRMAN SELIN: That's a very effective

1 technique.

2 MR. KRESS: Squelching of questions.

3 [Laughter.]

4 MR. KRESS: I wonder if I could use that.

5 MR. CARROLL: I would have added one other thing

6 that jumped out at me, and that is the rather innovative

7 security system that Westinghouse is proposing for the

8 AP600.

9 CHAIRMAN SELIN: You approve of it?

10 MR. CARROLL: I don't know. I am still thinking

11 about it. It has some very clear benefits. I think there

12 are also some problems that they haven't thought about quite

13 yet. We did point out to them that it looked like they were

14 going to have an awful time getting the numbers of people

15 through the security system that would be required during

16 outages. It is basically a two portal entrance to the vital

17 area of the plant.

18 MR. KRESS: The next item is regulatory analysis

19 guidelines. As you know, over the past several years, the

20 ACRS has been reviewing the Staff's efforts to update this,

21 and revise it. We believe this to be a very important

22 activity. We think this is a policy document that is very

23 important, and so we are very pleased to be able to review

24 it, and we have written two reports on the subject. One in

25 1992 November, and another one recently in September of

1994.

In our first report, we generally agreed with the overall approach and most of the things they were doing, but we did have some very specific disagreements and complaints, I guess we would call them. Without going into much detail as to what they were, we thought that there was a need for an overall policy statement on how the regulatory process works as a preface to that thing.

We complained about the fact that implementation of the safety goal part of it, at that time, dealt only with changes in core melt frequency and didn't seem to accommodate changes in the conditional containment failure probability.

We complained about the use of a discount rate that was different than what OMB had been recommending to other agencies, and we didn't specifically think they ought to use the OMB, but they ought to justify the use of the one they had.

We also thought their treatment of averted onsite cost could lead to difficulties. They were subtracting those from cost as opposed to treating them as a benefit. You only have a problem with that if you use the ratio of cost to benefit, it will skew that measure. We thought it ought to be treated, like it is, as a benefit.

We complained about the use of $1,000 per man rem

1 as the monetary value for averted dose. We thought that was
2 outdated and needed updating.
3 COMMISSIONER de PLANQUE: I notice Staff in their
4 February 7th response to you indicated that would come mid-
5 '95. Do you know the status of that?
6 MR. KRESS: We have been told that that is
7 pending, and that we will hear about it either next meeting
8 or the one after that.
9 COMMISSIONER de PLANQUE: So you haven't seen
10 anything.
11 MR. KRESS: We haven't seen the new revised
12 dollars manual. We are looking forward to that because of
13 all these recommendations we made, the Staff agreed to them
14 and changed their document, with the exception of two
15 things, and that is the averted onsite cost and the $1,000
16 per man rem which we hadn't heard about. They went ahead
17 with that one as an interim. So we are anxious to hear what
18 the new value is.
19 COMMISSIONER de PLANQUE: Do you think, since this
20 is on our plate right now, do you think if that is imminent,
21 something on the $1,000 per man rem, that it is better to
22 wait until we have that before proceeding with approving the
23 guidelines as they are?
24 MR. KRESS: We made that as a recommendation. We
25 thought it was better, but if you went ahead with it, then

1 you have to go through the process of making a settlement
2 and redoing it, and we didn't see any real need for urgency
3 for the document at the time, so we thought it was better to
4 wait until you had it complete and had the new values.
5 CHAIRMAN SELIN: Are you going to talk about the
6 containment failure definition also?
7 MR. KRESS: I didn't intend to, but I will.
8 CHAIRMAN SELIN: Please.
9 MR. KRESS: Yes, absolutely.
10 CHAIRMAN SELIN: The reason I raise that is, you
11 and the Staff both agree that the man rem figure should be
12 changed. You have a management disagreement on how to
13 implement it. I would more likely go along with the Staff
14 on that in the sense of whether their judgment is that we
15 should do it interim or not, unless there were a technical
16 problem.
17 On the other hand, you have a substantive
18 disagreement on the containment failure definition, and so
19 that I really would like to hear what you have to say about
20 it.
21 MR. KRESS: Well, the definition they had was one
22 that did not really -- was not the definition one would use
23 to develop the risk value. So, since the main use of the
24 guidelines is to make risk calculations so that you can do a
25 risk/benefit thing, one ought to use the containment failure

1 definition that is most appropriate to determine the risk.
2 We thought it was the one that was used in NUREG-1150.
3 Dana, you may want to comment on that.
4 CHAIRMAN SELIN: But the Staff's position is, it
5 is unusual, but the Staff's position is really quite a bit
6 more nuance than that. They are not saying that there is
7 anything wrong with the 1150 definition per se, they just
8 say there is a wider range of things that we do with this
9 definition. We find this definition too narrow for our
10 purposes. I would like you to comment on that. It leads to
11 counterintuitive results when applied beyond just the PRA.
12 MR. KRESS: That may be true, but we thought the
13 major use of the definition really had to do with
14 determining the risk when one did a risk benefit analysis.
15 We thought that is where it would be mostly applied and
16 would have the most impact. Therefore, we thought the
17 definition ought to focus on that part of it more. That is
18 where we had the departure. There may be other definitions
19 of containment failure that you ought to use for other
20 parts, but we thought the risk was the most important part
21 of the calculation.
22 CHAIRMAN SELIN: The reason I keep pushing on this
23 is that you have two different situations. One is where you
24 just need a clear definition because you are going to do a
25 calculation, and if the definition is a little different

1 then that leads to what amounts to a different scaling of
2 the figures, but the relative amounts would be the same. If
3 you had a more rigorous definition, you would have a more
4 generous threshold; if you had a less rigorous definition,
5 you would have a more rigid threshold. But it doesn't much
6 matter how you define things as long as you have a
7 consistent definition so you can look at different scenarios
8 within a plant or different plants.
9 The second has to do with getting the kind of
10 definition that is useful for absolute regulatory questions
11 where if the definition is too rigid we will end up not
12 accepting performance which we would all think is a fact
13 acceptable performance. I am trying to understand where you
14 think this definition comes in at this point.
15 MR. KRESS: Well, the way we viewed it is, when
16 one calculates the risk, one, it is dominated by early
17 containment failure of rather substantial.
18 CHAIRMAN SELIN: Right.
19 MR. KRESS: And that the calculation of the doses
20 and the conversion to monetary value of the impacts, when
21 you make a value impact, the thing that you have to do to
22 determine whether, for example, a backfit is justified, one
23 needs to calculate that number in a rather absolute sense
24 instead of relative, and then one ought to use the values
25 and the definitions for the things that goes into that

1 calculation that really lead to a risk number. It is an
2 absolute value. It is one of those places where you have to
3 rely on the bottom line number of a PRA-like analysis.
4 There is hardly any way to get out of it because you are
5 trying to make a judgment as to whether it is worth the
6 effort to do it. You have the justification. That was our
7 thinking.
8 COMMISSIONER JACKSON: Dr. Kress, given that you
9 say that, and my understanding is that when you are talking
10 about early containment failures in BWRs, there is some
11 issue in terms of a number of hours that you would expect
12 there to be some failure, some timeframe within which you
13 would expect there to be some failure. My understanding is
14 that there was a difference of opinion within the Staff
15 between those in research and those in NRR, and there seemed
16 to be quite a divergence in terms of the number of hours.
17 At this time, or at least my understanding is, there is sort
18 of a compromise perspective on that.
19 The question is, given what you just said, are you
20 comfortable with the analysis that went into that, given
21 that what you want in the end is some calculation of risk?
22 MR. KRESS: Yes, I am comfortable with the fact
23 that they decided to go with the NUREG-1150 definition for
24 the risk part of it. I don't recall the nature of the
25 division of the Staff. When we heard the presentation, we

1 were given the one definition.

2 COMMISSIONER JACKSON: Okay.

3 CHAIRMAN SELIN: Can I go back to the $1,000 per

4 man hour?

5 MR. KRESS: Yes.

6 CHAIRMAN SELIN: As I understand the argument,

7 Staff is saying, we have made some significant improvements.

8 We still have to work on this figure, but we would like to

9 start using the improvements because, even though they are

10 partial, the guidance is still better today than to just

11 freeze it at the previous level, and they made a management

12 judgment that they would rather go through with two-phase

13 implementation of the new guidance.

14 I think it is appropriate for the Committee, not

15 so much to question the management judgment as to say

16 whether using the wrong figure in the new guidance would be

17 more harmful than using the wrong figure in the old guidance

18 if they had to do something in the future.

19 MR. KRESS: I agree with you on that completely.

20 It is not our job to too much question the management

21 decisions. We would rather be questioning the technical

22 aspects.

23 CHAIRMAN SELIN: But if there were some egregious

24 inconsistencies between using a wrong man rem figure.

25 MR. KRESS: We haven't seen the new figures or

1 their basis yet, so I have difficulty commenting on it. Our
2 feeling is that the old figures are outdated and aren't well
3 justified.
4 CHAIRMAN SELIN: And the Staff agrees.
5 MR. KRESS: We haven't seen the new ones yet. I
6 am looking forward to reviewing it and see what their basis
7 are, and what the new numbers are.
8 COMMISSIONER de PLANQUE: How far off do you think
9 these numbers are in terms of leading to realistic
10 assessments?
11 MR. KRESS: I see numbers that are as high as ten
12 times higher than that. I don't think they are off that
13 far, personally. I think they may be off by a factor of
14 five. But this is all just a guess.
15 The other problem with the $1,000 per man rem, it
16 is generally used for both health effects and for land
17 contamination, and that doesn't seem right to me. It is
18 also used as a surrogate for not discounting future effects,
19 and that looks to me like a problem also.
20 I am anxious to see whether they fix both of
21 those, whether it is two different values for land
22 contamination versus health effects, and whether or not they
23 intend to have a value that is discounting for future
24 effects.
25 COMMISSIONER de PLANQUE: In any event, you think

1 this is a topic that should be treated fairly expeditiously?
2 MR. KRESS: Oh, yes. We are anxious to hear it.
3 As soon as the Staff is ready to come with the new figures,
4 we want to take it on our agenda and look at it. I think it
5 is an important subject.
6 CHAIRMAN SELIN: Absolutely.
7 MR. KRESS: And it permeates the whole
8 cost/benefit analysis.
9 CHAIRMAN SELIN: Okay.
10 MR. KRESS: The next item is application of risk
11 analyses and rulemaking, and Dr. Powers is going to discuss
12 that.
13 MR. POWERS: This, of course, touches upon what I
14 think is one of the most exciting things that the NRC is
15 doing now, this innovation in its regulations to adopt a
16 risk-based or a performance-based basis for its body of
17 regulations. In my opinion, this is a revolution whose time
18 has come. It has captured the imagination, and I think we
19 will see many in the regulatory bodies looking to the NRC
20 for the leadership it provides in this area.
21 The changes, of course, are being made
22 progressively as we encounter opportunities to go to a risk-
23 based, performance-based. We see this happening, but we
24 also see that there have been missed opportunities for
25 making some of these changes. Certain we have written to

1 you recently about a missed opportunity in connection with
2 the decommissioning of plants.
3 We have seen a second example of what we think is
4 a missed opportunity to go to a risk basis or a performance
5 basis in the extension of leak-before-break regulations to
6 small diameter pipes in AP600.
7 What we think is, there is inherently a
8 flexibility and a discretion being allowed to both the
9 applicant and to the regulator when you go to a performance-
10 base or risk-base, and people get very uncomfortable with
11 this discretion, this call to judgment that is in inherent
12 in this, and they become uncomfortable with this
13 flexibility, we think, perhaps because there is not enough
14 technical support as we broaden the applications of risk-
15 based regulation beyond that core associated with just the
16 reactor performance to new areas.
17 What we are concerned about is that the research
18 programs that NRC is supporting may not be sufficiently
19 broad to support the widening role that we are seeing risk
20 and opportunities for risk-based regulation go into new
21 areas like the decontamination and decommissioning of
22 nuclear plants. Certainly in the high-level waste and
23 medical uses the PRA is not as well developed as it is for
24 the regulation of nuclear reactors, and this is an area that
25 we think we would like to take on our agenda to see what

1 type of research the NRC should be supporting if we are
2 going to go to a truly risk-based regulatory body.
3 CHAIRMAN SELIN: So you are basically laying out a
4 marker for future work, correct?
5 MR. POWERS: That's right.
6 CHAIRMAN SELIN: Could I just stress the
7 importance of -- I guess it is time to be a little skeptical
8 about certain aspects of risk-based calculation, and it is
9 not on the probabilistic side. There are two problems that
10 arise, and as you look at that I hope you bear these in
11 mind. One is, there is a certain amount of certainty to say
12 that you take in a meter, you measure something. If the
13 number is greater than a level, you are okay; and if it is
14 not, you are not. When you go risk-based, you lose that.
15 Now that is fine for reactors when the payoff is
16 huge and we understand the analysis. As we apply it to
17 other areas, we just have to be careful when you have small
18 licensees who can't do these kinds of analyses, that we are
19 not asking them -- they almost want certainty, even if it is
20 inconvenient, because then they can live with that.
21 MR. POWERS: Certainly. Getting a prescription is
22 an easy thing to design to and to operate to.
23 CHAIRMAN SELIN: If you tell a doctor who does the
24 right thing, he is okay. If you tell a doctor that it
25 depends on how the patient comes out. It is one thing to

1 use risk in developing the limits, but in others to apply it
2 in a rolling way for each licensee -- I just lay this out,
3 as you look at these things, to bear this in mind.
4 The second is, we just can't afford the situation
5 where a perfectly adequate program just becomes inadequate
6 overnight because somebody has come up with a different
7 model. There has to be a continuity of the process as seen
8 by the regulated community. We don't know how to answer
9 this, and we do really welcome your getting involved in
10 these questions. I have said it, so I will stop.
11 MR. POWERS: I think there are two things. Just
12 to respond to you, I think we are aware of both issues that
13 you bring up. Some of us are a little more passionate about
14 risk and performance-based regulation and need to be
15 reminded that for some small things it is easier to have a
16 number. But I think we are careful in saying there is risk-
17 based analysis and there is performance-based, and we do see
18 a distinction from them. They sometimes can be closely
19 related.
20 In the area of where the risks are inherently low,
21 necessarily low, we think that the performance-base that is
22 not so tightly coupled to risk is the way to go.
23 The second common idea is that I think you see
24 universally that as we come with these performance and risk-
25 based regulations, we are always allowing the licensee the

1 opportunity to stay with the old and the familiar.

2 CHAIRMAN SELIN: That's very good.

3 MR. POWERS: These things come to the fore much

4 more often on the new and the innovative where the industry

5 itself is trying to make a change in the technology. I

6 think you will see in our comments that we make a stronger

7 emphasis when it is associated with something new as opposed

8 to changing the basis and allowing more flexibility for

9 those applicants that want that flexibility and can use that

10 flexibility.

11 MR. KRESS: The other point is that risk-based

12 regulations can become very prescriptive also as long as

13 this prescription has a basis in risk, and you can stay with

14 the certainty of that sort of stuff and still have it risk-

15 based.

16 CHAIRMAN SELIN: The big problem is that -- I

17 mean, they are really two quite different things. One is to

18 use risk more explicitly in arriving at regulations, but the

19 form of the regulations be quite deterministic. I don't

20 want to say prescriptive, but deterministic.

21 MR. KRESS: Deterministic.

22 CHAIRMAN SELIN: The second is, you will have a

23 risk meter that you take with you and you try to do a

24 continuous calculation, and those are quite different pieces

25 of knowledge.

1 MR. KRESS: They are quite different.

2 CHAIRMAN SELIN: Generally the form of our

3 regulations are more like the former, but we always allow

4 people to come in and do a site-specific risk calculation

5 and say that the regulations are unreasonable in our case

6 because we can show that we should get an exception based on

7 a risk. As long as there are optional pieces, we don't have

8 a regulatory problem, but when we really move to risk-based

9 regulation, it is just hard for us. That is the only point

10 I would make, and one has to take some action.

11 MR. KRESS: I don't think we should scrap the

12 system that has served us well for many years.

13 CHAIRMAN SELIN: No. You drive a car, you run

14 into the back of somebody else's car, you are wrong. You

15 don't have to get out and do an analysis of it.

16 MR. KRESS: Regardless of whether he backed into

17 you or not, that's correct.

18 CHAIRMAN SELIN: It has avoided a lot of analysis

19 and made things very simple.

20 MR. KRESS: There is a lot to be said for that.

21 CHAIRMAN SELIN: Thank you very much, doctor.

22 MR. KRESS: The next item is the proposed final

23 rule on technical specs. Jay Carroll has the lead on that.

24 MR. CARROLL: Okay. The final policy statement on

25 tech spec improvements for nuclear power reactors dated July

1 23rd, 1993, established four criteria to define requirements
2 that should be controlled by tech specs. In June of '93, we
3 reviewed the situation and agreed with what was being done
4 in regard to tech spec improvements with the exception of a
5 concern that there was a need to better define the
6 terminology "significant to public health and safety" as
7 used in the fourth criteria of the policy statement.
8 The Staff following all that has gone out for
9 public comment and has responded to -- I guess we reviewed
10 this during our April meeting and issued a letter on April
11 13th in which we argued that there continued to be a need
12 for better guidance on the meeting of "significant to public
13 health and safety."
14 The Staff believes that their several years of
15 experience using the criteria under the interim and final
16 policy statements demonstrates that the criteria that can be
17 used in a consistent and controlled manner, and they also
18 believe that there will be additional guidance coming up in
19 the form of guidance developed under the PRA Implementation
20 Plan.
21 So we say more guidance is needed, and they say,
22 well, we are okay now and more guidance will be forthcoming.
23 As we understand it, you have this as an issue you need to
24 resolve.
25 CHAIRMAN SELIN: And we are helpful that a word

1 which has a clear meaning but not a precise one can be made
2 more specific through a combination of reg guides or, if
3 necessary, the first couple of times it is applied to the
4 situation, but I certainly agree with your concern.
5 MR. CARROLL: In our original letter, we did make
6 the point that this was not unique to tech specs. This
7 language is used in a lot of other contexts, and we just
8 used tech specs as a vehicle to --
9 CHAIRMAN SELIN: Make a point.
10 COMMISSIONER ROGERS: Did you have anything in
11 mind specifically as to how to sharpen up that definition?
12 MR. KRESS: I told you he would ask that.
13 MR. CARROLL: I know you did. I said I was going
14 to defer to you.
15 MR. KRESS: We were bouncing things around, and
16 the things that automatically come to mind are "delta core
17 melt frequencies," the things like that, but we don't really
18 have a recommendation yet that we have discussed among the
19 Committee and agreed upon.
20 COMMISSIONER ROGERS: Well, just a general
21 approach, even saying that is a suggestive way to look at
22 it.
23 MR. KRESS: Yes. If it is significant to the
24 health and safety of the public, it has to be a risk type
25 number and it has to be a delta-something, a core melt

1 frequency or a containment failure criteria. We haven't
2 bounced it around enough to come to a firm recommendation
3 yet.
4 COMMISSIONER ROGERS: Well, it also has to be one
5 that is consistent with our decisions of the past, unless we
6 think they were wrong.
7 MR. KRESS: Yes.
8 COMMISSIONER ROGERS: That may be part of the
9 problem. In covering the full-scope of wherever we have
10 used that terminology to make sure that a more mechanistic
11 definition such as you are suggesting doesn't somehow or
12 other conflict with some earlier decisions.
13 MR. KRESS: Yes, we agree.
14 MR. CARROLL: I don't think ten to the minus five
15 does that, conflicts with other things.
16 MR. KRESS: Yes, that's the problem.
17 MR. CARROLL: It needs to be looked at.
18 MR. KRESS: The next item, cracking and fatigue
19 issues associated with nuclear reactor components, our
20 expert on that is Dr. Shack, but Dr. Seale will carry the
21 lead for reasons he will explain.
22 MR. SEALE: Okay. These issues, a variety of
23 issues that are related to metal cracking and fatigue in
24 components comes under the general purview of the Materials
25 and Metallurgy Subcommittee. Dr. Shack is our resident

1 blacksmith, but it turns out that he has a conflict on
2 several of the issues, and where those have existed, I have
3 presided at the subcommittee meetings and so on.
4 But we have picked his brain as appropriate, and
5 he participates, except that he is not involved in any
6 decision votes, or anything like that. So when we have
7 questions, he is likely to field them for us today, too, but
8 I will go through the discussion.
9 The issues that are involved include the BWR core
10 shroud problem; the general question of reactor vessel
11 integrity; steam generator tube degradation; reactor vessel
12 head cracking, and that is penetrations and that is for
13 PWRs; and then the adequacy of fatigue life evaluations for
14 metal components.
15 There are a few comments worth making there with
16 regard to the BWR shroud issue. That came up I guess almost
17 a couple of years ago now. That effort has been
18 characterized by a very strong cooperation between the Staff
19 and the industry. The industry really grabbed the problem,
20 developed some very thoughtful and good ways of trying to
21 evaluate, locate the cracks, identify where the problems
22 were, and then actually effect repairs, and there have been
23 repairs effected in some cases, and they are proceeding on
24 through the inventory of those kinds of plants. It is a
25 very worthwhile example of how the Staff and the industry

1 can cooperate in an effective way to the benefit of
2 everyone.
3 The second issue has to do with reactor vessel
4 integrity. Over the last couple of years, the Staff has
5 been working to improve its -- let's say the science of its
6 position with regard to charpy upper-shelf definitions and
7 so on. Then recently we have a heightened concern about
8 pressurized thermal shock that has come up as the result of
9 the analysis of the data from a retired steam generator
10 which is presumed to be a qualifying sibling, if you will,
11 of a reactor vessel for that same plant where the copper
12 content was found to be higher than had originally been
13 expected.
14 We are following that. There are a couple of
15 comments we have. One is, we are kind of intrigued that a
16 single point like that would suddenly change the statistics
17 of that whole evaluation so dramatically as to require
18 expeditious reevaluation. Certainly the reevaluation is
19 appropriate, and the idea of getting a better pedigree on
20 some of the vessels that are out there is probably a good
21 idea. But we are kind of intrigued by the fact that one
22 data point suddenly popped all of this concern up out of the
23 background, and we want to look into that a little bit, I
24 think.
25 The steam generator tube degradation is another

1 issue, of course, that has had a lot of publicity here
2 lately. It is noted by its diversity in terms of the kinds
3 of cracking that takes place, why it happens, who it happens
4 to, who the culprit is, the whole thing. If it says
5 anything at all, it says water chemistry is more important
6 than almost anybody else on the plant, at least if you are a
7 water chemist.
8 Seriously, I think we feel that there has been a
9 lot of progress made. The interesting thing, though, now is
10 that in examining the latest results, we find that the past
11 attempts to detect circumferential cracks haven't been as
12 good as we thought they were. This is looking at the Yankee
13 steam generator and --
14 COMMISSIONER ROGERS: In Maine Yankee.
15 MR. SEALE: Yes. We are really interested in
16 that. I think we will be following that as we go along
17 because that is another issue.
18 COMMISSIONER ROGERS: On that whole question of
19 instrumentation for detection of these kinds of
20 degradations, have you any thoughts as to whether really
21 there could be better means than what is now the eddycurrent
22 technique, basically, that is used?
23 I always have the feeling that we tend to turn to
24 this really primitive detector and try to lay on it all the
25 possible improvements in dealing with whatever signals come

1 out of these things, but we don't look at what are the
2 alternative basic phenomena that might reveal these besides
3 eddycurrents.
4 I know there are issues of being able to do this
5 rapidly at low-cost in an existing plant when you have
6 thousands and thousands of tubes to look at, but somehow I
7 have the feeling that high-tech hasn't caught up with this
8 business.
9 MR. SEALE: I can't argue with that, and clearly
10 there is an emotional commitment now to the eddycurrent
11 technique. I mean, after all, everyone has invested in
12 training the people on how to do it, and you have come to
13 grips with the health physics problems associated with
14 carrying out these measurements in the way that they are
15 done, the whole bevy of problems and, yes, there probably is
16 a better science there, but we haven't found the genius yet
17 that knows where that science is.
18 COMMISSIONER ROGERS: Well, is anybody trying?
19 MR. SEALE: That is an interesting question.
20 MR. CARROLL: There is some development work on
21 UT, isn't there?
22 MR. SHACK: There is development. The Belgians,
23 for example, have gone to UT to essentially quantify
24 circumferential cracking because they were uncomfortable
25 with the capability of the rotating pancake coil to do it.

1 Again, though, these changes are difficult to
2 make. There is a great deal of experience. To quantify the
3 performance of one of these techniques is not a trivial
4 thing. We are still trying to understand the performance of
5 the eddycurrent system. I think that is one of the
6 surprising things about the Maine Yankee situation is that
7 there was a reasonable degree of confidence that the
8 rotating pancake coil, everybody understood that it was
9 difficult to quantify circumferential cracking, but there
10 was a reasonable degree of confidence that you could detect
11 it. I think that is the surprising thing about Maine
12 Yankee, not that the crack depth was 29 percent when the RPV
13 said it was 35 percent.
14 It is missing cracks that is the crucial item here
15 because, again, I think the Staff has done a good job with
16 the voltage-based repair criterion to deal with the axial
17 cracking in the ODSCC. One has a certain degree of comfort
18 with that because if you are wrong, the cracks are
19 surrounded by the tube sheet, and there is a certain degree
20 of defense-in-depth from that. With the circumferential
21 cracks, the cracks are more naked, and there is a bigger
22 problem if you are wrong. So we do have to have a high
23 degree of confidence in our ability to detect those cracks.
24 Again, there are research programs in the NRC.
25 The Europeans have looked at these other techniques, but it

1 is a difficult and complex thing to introduce a new --

2 COMMISSIONER ROGERS: You can keep going down,

3 riding your buggy everyday forever, or you can start

4 thinking about alternative modes. I mean, it just seems to

5 me that all the advances in material science that have taken

6 place around the world, that there isn't another technique

7 that could be developed that would be at least as good and

8 as fast, and maybe even better.

9 I just have a feeling that there is nobody putting

10 any money into this. I don't know, in the university

11 research programs where people are looking at things like

12 this, and if it is not being done there, where is it being

13 done? Unless somebody is really gung ho on it, it is not

14 going to get done. It seems to me it is a glaring example

15 of a lack of tapping into modern material science knowledge

16 when that is what the problem is.

17 There is a strong vested interest in using the

18 equipment that already exists and the techniques that

19 already exist, and that is okay, but I don't know, it seems

20 to me that this Maine Yankee situation ought to be a wake-

21 up call to us that we ought to be looking for better ways to

22 do these things. We shouldn't be surprised by something

23 like that. We shouldn't have surprises like the Maine

24 Yankee steam generator circumferential cracking that, lo and

25 behold, the detection techniques that were being used in

1 good faith by everybody just didn't turn out.

2 MR. CARROLL: What success are the Belgians having

3 with UT?

4 MR. SHACK: They are fairly positive on it. They

5 have been using it more to quantify cracking. That is, the

6 usual response, I believe, to circumferential cracking in

7 this country is essentially to plug or to repair because

8 your confidence in being able to quantify the thing is not

9 very high.

10 The Belgians were looking at UT not so much to

11 improve their detection capability but to improve their

12 characterization capability so that they wouldn't have to

13 plug or repair as many indications.

14 Again, as in all of these techniques, you know, it

15 is like looking at Vitamin C. You want to find a study that

16 shows that Vitamin C is a good thing, I can find you a

17 study. If you want to find a study that says Vitamin C

18 doesn't work, I can find you a study. The results are

19 always mixed.

20 MR. CARROLL: Are the Belgians using eddycurrent

21 as their detection technique and then following it up with

22 UT to characterize, is that their approach?

23 MR. SHACK: I am a blacksmith, not a mind reader

24 of the NDE types. But I believe the Belgian position is

25 that UT is their main tool for looking at circumferential

1 cracking.

2 MR. CARROLL: Both detection and characterization,

3 okay.

4 MR. SEALE: Perhaps it would be worthwhile to

5 offer anybody with a new toy a shot.

6 MR. SHACK: I think the Staff has at least asked

7 everybody to. The generic letter has asked them to

8 reevaluate this problem with circumferential cracking, and

9 we will be seeing responses from that generic letter.

10 COMMISSIONER ROGERS: That goes to the licensees,

11 they have a problem they have to solve fast. If they detect

12 something, they have to get on with it. But what I am

13 looking at is a longer-term approach here that isn't being

14 driven by the immediacy of, we have to get a plant back

15 online, but we are going to come up with a better way to do

16 this, even if it takes five years, and a plant isn't going

17 to wait five years for something like that to develop. It

18 has to be on a different track. I just don't see anything

19 moving in that direction.

20 MR. CATTON: Maybe research ought to be encouraged

21 to have some universities take an interest in it through the

22 grant program.

23 COMMISSIONER ROGERS: Well, yes, but is that

24 really a responsibility for NRC's budget?

25 MR. CATTON: Well, that would be cheap.

1 COMMISSIONER ROGERS: Well, we have to ask that
2 question very hard of everything we spend today, and yet
3 this is of enormous interest to the industry. I am really
4 quite disappointed I haven't seen more ingenuity.
5 MR. CATTON: Universities typically just don't
6 have research or just don't do research anymore.
7 COMMISSIONER ROGERS: Universities like dollars.
8 MR. CATTON: You bet they do.
9 COMMISSIONER ROGERS: They like dollars very much.
10 MR. CATTON: You bet they do.
11 COMMISSIONER ROGERS: Just waive enough dollars,
12 and I think you will find some takers.
13 MR. CATTON: Well, you need a waiver of the
14 dollars, you need somebody to do it, to give it lots of
15 attention.
16 MR. CARROLL: The other possible candidate, of
17 course, to do work in this area is the EPRI NDE center which
18 has done a lot in support of the eddycurrent technique.
19 MR. KRESS: And you might think that DOE would be
20 interested in doing some research along those lines at the
21 national labs.
22 COMMISSIONER ROGERS: Yes, but I think what we are
23 finding is, everybody is pointing to somebody else.
24 MR. CARROLL: I think until Maine Yankee happened,
25 we were about done and happy with rotating pancake coils.

1 MR. SHACK: That certainly was the conventional
2 wisdom, that it was reasonably good. Again, the performance
3 demonstration for the adequacy of all these things is an
4 interesting problem just in itself. It is a nontrivial
5 because, again, to do the performance demonstration you need
6 to have some tubes. To do that you have to characterize the
7 tubes. You have to characterize the tubes by cutting them
8 up. They are not real useful for much of anything else. So
9 it is kind of circular kind of argument, very difficult.
10 COMMISSIONER ROGERS: We see that this is a very
11 serious question for any PWRs, and we are talking about
12 billions of dollars now. We are not talking about tens of
13 millions, we are talking about billions of dollars at stake
14 whether old plants will have their licenses renewed or not.
15 I mean those economic decisions are very big decisions that
16 could be affected by being able to do this a little bit
17 better. There is a lot of leverage involved here.
18 CHAIRMAN SELIN: We hear your interest and we
19 appreciate your report.
20 I think we might go on to the next item.
21 MR. SEALE: Well, there are just a couple of other
22 things. Again, we would like to compliment the Staff in
23 working with the industry people on the vessel head
24 penetration cracking problem which I guess first showed up
25 in France, and they are working through that problem in this

1 country as well.

2 Then the fatigue life problem is going on. We are

3 going to hear something about the Staff position on that

4 sometime this summer. It has kind of been abeyance for a

5 while, but it is an active issue.

6 The Committee has also expressed some interest in

7 looking into the pressurized thermal shock issue in a little

8 more detail. It is a problem that is both materials and

9 thermal hydraulics, and we have been trying to stir up some

10 interest in that, and hopefully that activity will get a

11 little bit more attention in the near future.

12 That is all I have.

13 MR. KRESS: All right. Thank you, Dr. Seale.

14 The next item is the digital I&C, and we are going

15 to baptize our new member who is our expert on this subject,

16 Don Miller from Ohio State.

17 Dr. Miller.

18 MR. MILLER: Yes. I have outlined my comments on

19 pages 82 of your package there. I guess I am still using my

20 professorial type approach, so I have an outline.

21 As kind of the new person here, I did take the

22 advantage of going back and looking at a number of ACRS

23 Staff letters, or letters to the Commission which kind of

24 set the stage for what I think is the opportunity for the

25 NRC to become a leader in digital I&C.

1 I asked the question, why digital I&C in reactors,
2 and give you the normal answers right there. Probably the
3 major driving force in current plants is to replace the
4 degrading analog systems in which there is really, in many
5 cases, no replacements otherwise.
6 From a bigger picture, it might be to capture the
7 computer revolution which has put the PCs on everybody's
8 desk and so forth, and it is a way to put into our
9 technology the revolution of computers over the last 10 or
10 20 years.
11 But it has not only opportunities, it also has
12 problems which are captured quite well in the March '93
13 letter of the ACRS to the Commission which I encourage all
14 of you to review again. I thought that captured some of the
15 real upsides and downsides.
16 The opportunities are to improve plant
17 performance, add additional functions, and really to
18 maintain the increasing trend of capacity factors we have
19 seen over the last 10 years. Our capacity factors, as you
20 know, have gone up dramatically in the last ten years.
21 Digital may be a way of maintaining that momentum, so to
22 speak.
23 CHAIRMAN SELIN: Would you elaborate on that a
24 little bit.
25 MR. MILLER: In several ways, first of all, as you

1 know, digital systems will introduce the possibility of
2 self-testing, reduced operation and maintenance, that will
3 lead to, hopefully, reduced outage times and so forth. You
4 can predict problems that are going on in your plant where
5 you might currently have to look for them. Maybe it will
6 reduce surveillance requirements, and so forth. All of
7 those things are possibilities we have to look for.
8 As you saw on the certifications of the
9 evolutionary plants, probably at least a major difference in
10 the evolutionary plants and the current plants are certainly
11 the digital I&C built into those plants. Of course, the
12 AP600 will be in the same vein.
13 I summarized on the next page a number of
14 activities of which NRC has been a participant in many. The
15 first organized, so to speak, use of digital began with EPRI
16 in 1990 on what is called the demonstration or I&C
17 Demonstration Program which they have about four of five
18 plants involved in that representing all the different types
19 of plants.
20 In '92, the NRC proposed a generic letter which
21 basically said, all digital could be unreviewed safety
22 items, and that did not -- that stimulated, I guess is the
23 way to put it, a fairly and I think positive collaborative
24 effort between NRC, industry, EPRI which provided some of
25 the leadership which resulted in the guideline which was

1 published in '93, and with a couple of clarifications was
2 endorsed by the NRC here in April of '95.
3 I think the guidelines set the stage for digital
4 I&C upgrades in current plants, and it is up to the NRC to
5 implement the guideline, but it sets the stage.
6 In addition to that, the ACRS stimulated a
7 National Academy of Science study which is now ongoing to
8 give an independent review of digital I&C. That study group
9 is comprised of industry experts outside of the industry as
10 well as experts within the industry.
11 I think it is possibly grappling a little bit with
12 its mission which may have changed since it was originally
13 instituted, but I think it will give us a very good
14 perspective of knowing what the other industries are doing
15 and how we might incorporate some of those things into our
16 industry.
17 A couple of other things happening, we have now --
18 we, the NRC, has put out for public comment a revision of
19 the Reg Guide 1.115, which is the industrial standard on
20 programmable software and programmable computers. That is
21 out for comment, and will be back to us in how long those
22 comment periods take. I will learn sooner or later, I
23 guess.
24 MR. CARROLL: Long.
25 MR. MILLER: In addition, there is a program

1 starting with DOE, EPRI and industry to provide a
2 coordinated program to pull all of that together into a
3 overall I&C upgrade of all the plants here in the United
4 States.
5 Finally, a major effort is undergoing with NRC
6 leadership in upgrading the standard review plan, basically
7 Section 7 which is I&C. That is ongoing now. Primarily
8 being done by Lawrence Livermore under guidance of research.
9 To kind of summarize everything, as those of you
10 who have been around a while know, the ACRS was, how do I
11 put it, critical in these sense of NRC's incorporation of
12 this new revolutionary technology and stimulated a number of
13 things that are going on now, and I encourage you once again
14 to read these letters, particularly the March '93 letter.
15 But what is happening now, I see, is the NRC
16 Staff, the industry and NEI and everybody is working
17 together to bring this to pass. I think it is now kind of
18 the ball more or less is in the NRC's court to take the
19 guideline and use the guideline, not only in current plants,
20 on new plants. They can actually use the guideline for new
21 plants, and it will be doing this through a series of
22 procedural changes and training sessions. Basically, the
23 NRC Staff has to learn how to implement things in
24 concurrence with what industry is going to be doing.
25 So we are going to review the first Staff training

1 programs in July of this year, and hopefully it will
2 incorporate some of the things that ISA is doing and also
3 INPO has done over the years for the I&C training program.
4 A final statement, ACRS, and I will guarantee
5 this, we will continue to look at this very closely and make
6 certain this technology will be implemented in an
7 appropriate way. I think it is exciting. To me, it is
8 equally exciting as risk-based approaches. I think both are
9 concurrent, actually.
10 I welcome any questions here.
11 COMMISSIONER JACKSON: Actually, you mentioned
12 risk-based approaches. In some earlier correspondence, the
13 Committee had been concerned about the application of PRA
14 methodology to computer systems. Will any of the studies or
15 activities that you have outlined address specifically the
16 application of PRA methods to computer or microprocessor-
17 based systems?
18 MR. MILLER: That's a good question. In a way, it
19 is difficult to apply it because of the statistical nature
20 of software failure not being well-defined. I might defer
21 that to Dana.
22 I don't know if you want to make any comments on
23 that or not? We have had comments in the past on that.
24 MR. POWERS: One of the biggest motivations for
25 the Academy of Science studies was, in fact, to address this

1 question. It is not that PRA techniques can't be applied,
2 it is the application is different than it would be of an
3 analog system. What we were interested in is what is the
4 available technology for understanding both the common mode
5 issue, that is software gets programmed wrong -- software
6 systems don't fail, they are just in error -- and what we
7 are finding as these software systems get bigger is that
8 they are all wrong. They all have bugs in them, and what do
9 you do about that.
10 Perhaps the biggest example is, in fact, the FAA's
11 flight controller software, which is proving to be a major
12 difficulty for them in the application of digital
13 technologies for their purposes.
14 We wanted to find out what was available within
15 the industrial technology and also in the DOD side that
16 could form a basis for application of risk and performance-
17 based technologies to these digital I&C technologies.
18 Specifically, what do you do on the subject of
19 redundancies and diversity in these areas.
20 COMMISSIONER JACKSON: So you are telling me that,
21 yes, the studies will address these issues?
22 MR. POWERS: That's right. That's right. It is
23 specifically intended to.
24 I might also comment that the Staff has an
25 independent program looking at exactly that issue. It is a

1 multi-laboratory effort going on. Certainly Oak Ridge is
2 involved, but there are several other places that are
3 involved.
4 MR. CATTON: Our new member would like to comment
5 on this same subject.
6 CHAIRMAN SELIN: Dr. Apostolakis, the lectern
7 there is probably the easiest thing to do.
8 MR. APOSTOLAKIS: Just a point of clarification.
9 I think the objections have been raised against the use of
10 traditional reliability methods, and there is vast
11 literature where people have really tried to force
12 reliability models on software. It is not so much against
13 PRA, per se, which goes much beyond the traditional
14 reliability method. So it is just a point of clarification
15 that they are two different things.
16 MR. MILLER: For software you can't define a
17 failure rate.
18 COMMISSIONER JACKSON: No, I understood all that.
19 My point was that you, in fact, had seen that your earlier
20 correspondence was, in fact, criticizing the Staff's
21 attempts to use traditional methodologies.
22 MR. CARROLL: Indeed it was.
23 COMMISSIONER JACKSON: Given that that seemed to
24 show up in succeeding reports and correspondence, then I
25 wanted to be sure that that was an issue that, in fact, was

1 being followed up on.

2 MR. MILLER: Definitely it will be followed up. I

3 think we are in a position to be much more positive and, in

4 a sense, be a leader in this in the future.

5 MR. CARROLL: I can't help but note that we had a

6 somewhat different presentation on digital I&C than we have

7 had in the past.

8 CHAIRMAN SELIN: That was pretty clear in your

9 comments. Your whole attitude is quite different on both

10 sides.

11 MR. KRESS: The last item is also mine, and it is

12 the question of operating reactors conformance to the safety

13 goals which I will cover very quickly.

14 You recall that it came about because I shot my

15 mouth off earlier and said I thought that the adequate

16 protection level was already below, in terms of risk, the

17 safety goals, and that that had implications and it would be

18 nice to do something to validate that concept or that

19 feeling I had.

20 I thought that it would be possible to do that

21 using the IPEs and IPEEEs. Now, of course, we are well

22 aware of the recent concerns you have expressed on IPEs and

23 the use of bottom lines, but --

24 CHAIRMAN SELIN: I hope you realize that as far as

25 our remarks went, they were judicious remarks. That deep

1 down we feel even more concerned about the possibilities of
2 misusing the IPEs, or overstating the quality of the work.
3 MR. KRESS: And we agree with that.
4 CHAIRMAN SELIN: There was an article in the
5 Christian Science Monitor that took us to task for something
6 that we didn't deserve to be taken to task for, but we do
7 have to be sensitive to that.
8 MR. KRESS: I think the IPEs have served a good
9 purpose.
10 CHAIRMAN SELIN: Absolutely.
11 MR. KRESS: If they are used correctly, they are a
12 very good thing. Once again, we are trying to use them in a
13 different mode, and it may be very problematic to use them
14 for comparing with safety goals.
15 CHAIRMAN SELIN: Well, IPEs can be used for those,
16 but not the particular IPEs that we have today in the
17 groundrules.
18 MR. KRESS: If they have an ideal IPE.
19 CHAIRMAN SELIN: Right, given the groundrules
20 under which they were performed.
21 MR. KRESS: So, this is more of a status report.
22 We did proceed with setting up an ad hoc subcommittee
23 consisting of three of the members, one of the technical
24 staff and one of our fellows, and we outlined an approach,
25 and we have been following that approach.

1 The first thing that one notices is that there are
2 shortcomings to the IPEs that have to be overcome if we are
3 going to do this, and these involve things like, none of
4 them did an offsite consequence. If you are going to really
5 compare to safety goals, you have to have that. The CDF
6 values did not include shutdown conditions or seismic or
7 fires, those were done by the FIVE methodology and the
8 margins. Many of them did not go through a containment
9 conditional failure probability, so that if you are going to
10 do a risk you have to have that. There was really no
11 assessment of uncertainties.
12 So our ad hoc committee focused on, how do we deal
13 with those things if we are going to do a risk number, and
14 we did make some progress. We devised a way to extrapolate
15 the plant-specific consequences if you have a conditional
16 containment failure probability in a CDF, and that is fairly
17 innovative. We have developed ways to estimate the CDF
18 contribution that would be made by the seismic and by the
19 fires, given you have a margins calculation and given you
20 have a FIVE.
21 So we are making some progress. We still haven't
22 addressed the question of how you come about with a
23 conditional containment failure probability if you don't
24 have that, or how to deal with the uncertainties that are
25 present, but we are making progress, and we are still

1 hopeful that we can come up with some sort of a number.

2 CHAIRMAN SELIN: I think your observations are

3 timely, but I suggest for your future considerations that we

4 don't worry so much about a methodology for extending the

5 IPEs to containment failures or to offsite consequences so

6 much as we worry about how to have them redone on a more

7 standardized, replicable basis, no further than they have

8 gone today.

9 In other words, just get the core damage

10 frequency, but talk about standard factors, et cetera,

11 because the industry has shown at the expense -- and this is

12 really for the record, not for the Committee -- but it is

13 quite difficult for the Commission to encourage industry to

14 do the IPEs that were done, which had a pretty limited

15 application.

16 Now, if we are to move towards not just a risk-

17 based but more of a probability regulatory environment, good

18 IPEs, replicable, auditable, uniform IPEs will have to be

19 the basis for this. Industry seems to understand this.

20 There has been some conversation that encourages me that

21 they will be prepared to come forward and say, okay, now

22 let's talk about an agreed program for doing just the first

23 base of the IPEs, but doing them in a way that could be used

24 for risk-based.

25 The Committee could be enormously useful in

1 helping the Commission talk about how should we specify
2 these, what should be the standard factors, what should be
3 the components of standard IPEs, not the voluntary, this is
4 what we are willing to do and it is better than we were
5 before, but a basis for probability regulatory. You can be
6 enormously beneficial, but we need to get a much better base
7 before we talk about extending them to the Phase II, or
8 what-have-you.
9 MR. KRESS: Personally, I am very pleased to hear
10 you say that because this has been an underlying theme of
11 this whole -- we were trying to come up with this bottom
12 line question of the safety goals versus the risk status of
13 the plant, but permeating that whole discussion is this
14 question of the IPEs, how do you make them better, how do
15 you get them consistent, and what are the problems with
16 them. We had intended to actually come up with some
17 recommendations along that line. So I am very glad to hear
18 you say that.
19 COMMISSIONER ROGERS: That would be very helpful.
20 CHAIRMAN SELIN: I would strongly encourage you to
21 do that.
22 MR. KRESS: That would be an outcome of this
23 activity.
24 CHAIRMAN SELIN: I mean, the problem is to say,
25 given where we are today, what we have done is essentially

1 some anecdotal IPEs, is really the best way to call them,
2 and assuming an agreement between the industry and the
3 regulators that it would be useful to have these replicable
4 fashion, both from plant to plant and within plant from
5 scenario to scenario, just the sort of Phase I(a) IPE, you
6 might be very helpful not just waiting for the Staff to come
7 through, but thinking about what are the parameters and what
8 kind of guidelines ought to be set for doing such an IPE,
9 which is just very different from what we have today.
10 MR. KRESS: An excellent suggestion, and I think
11 we will definitely take it to heart and act on it.
12 CHAIRMAN SELIN: Did you have anything else?
13 MR. KRESS: No. That is basically just it, a
14 status report.
15 CHAIRMAN SELIN: Well, first of all, before you
16 get off it, I found -- I mean, your documents are always
17 useful, but I would like to congratulate Dr. Larkins and the
18 Committee Staff. I thought that this presentation was
19 really just extraordinarily useful. It was very well set
20 out. It is really very helpful.
21 And then I would like to repeat my remarks about
22 how you have kept your independence and integrity, but have
23 been able to adjust the interaction with both the Commission
24 and its Staff to be much more productive, and I benefit
25 myself from the intellectual stimulation of these meetings.

1 So I want to thank you.

2 MR. KRESS: We really appreciate that because,

3 basically we have made a conscious effort to do that, and we

4 appreciate the good word.

5 CHAIRMAN SELIN: That is greatly appreciated.

6 Commissioner Rogers?

7 COMMISSIONER ROGERS: No. I just wanted to add

8 that I certainly share the Chairman's view on the successful

9 of this meeting, and the way we are working together. It is

10 very good.

11 COMMISSIONER de PLANQUE: I would just ditto that

12 as well.

13 COMMISSIONER JACKSON: I would ditto it and add

14 that I particularly would be interested in following up on

15 the IPE issue and the computer and I&C issues.

16 MR. KRESS: If I might say, Dr. Jackson, we would

17 welcome the opportunity to have you come down and give us

18 the benefit of your philosophy about regulations and what

19 your priorities are. We took advantage of that with Dr.

20 Selin when he first came on, and we found it most useful to

21 us.

22 CHAIRMAN SELIN: I have to warn you, Commissioner

23 Jackson, they actually listened to what I had to say.

24 [Laughter.]

25 COMMISSIONER JACKSON: Unlike most others.

1 CHAIRMAN SELIN: It is a real responsibility.

2 MR. KRESS: We are looking forward to working with

3 you.

4 COMMISSIONER JACKSON: I am looking forward to it

5 also. Thank you very much.

6 CHAIRMAN SELIN: Thank you very much, Dr. Kress.

7 [Whereupon, at 10:59 a.m., the meeting was

8 concluded.]

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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, June 8, 1995

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

Transcriber: Tessa Minson

Reporter: Tessa Minson

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

June 5, 1995

MEMORANDUM TO: John C. Hoyle
Secretary of the Commission

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

R. Savio for

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

The ACRS is scheduled to meet with the NRC Commissioners between 9:30 and 11:00 a.m. on Thursday, June 8, 1995, to discuss the following items. Background materials related to these items are attached:

1. Introduction (NRC Chairman) — 9:30 - 9:35 A.M.
   a. Introduction of new Members
   b. Presentation of appointment certificates
2. Thermal Hydraulic Issues Related to Passive Plant Designs (Catton) (pp. 1-28) — 9:35 - 9:50 A.M.
   a. NRC test and analysis programs (TAPs) related to the AP600 and SBWR passive plant designs
   b. Status of issues associated with Westinghouse TAPs being conducted in support of the AP600 passive plant design certification
3. Status of Westinghouse AP600 Design Review (Catton) (pp. 29-31) — 9:50 - 9:55 A.M.
4. Regulatory Analysis Guidelines (SECY-95-028) (Kress) (pp. 32-45) — 9:55 - 10:05 A.M.
5. Application of Risk Analysis in Rulemaking (Kress/Powers) (pp.46-58) — 10:05 - 10:15 A.M.

6. Proposed Final Rule on Technical Specifications (SECY-95-128) (Carroll) (pp. 59-68) — 10:15 - 10:25 A.M.

7. Cracking and Fatigue Issues Associated with Nuclear Reactor Components (Seale/Shack) (pp. 69-79) — 10:25 - 10:35 A.M.

   a. BWR Core Shroud Cracking

   b. Reactor Vessel Integrity

   c. Steam Generator Tube Degradation

   d. Reactor Vessel Head Cracking

   e. Metal Fatigue of Primary Pressure Boundaries

8. Digital Instrumentation and Control (Miller) (pp. 80-97) — 10:35 - 10:45 A.M.

   a. National Academy of Sciences/National Research Council study of digital systems in nuclear power plants - Status Report

   b. Development, issuance, and NRC staff endorsement of EPRI TR-102348 "Guideline on Licensing Digital Upgrades"

   c. Ongoing staff and industry initiatives in the digital I&C area

9. Operating Reactors Conformance to the Safety Goals - Status Report (Kress) (pp. 98-103) — 10:45 - 10:55 A.M.

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

Attachments:
As stated

cc: ACRS Members
ACRS Technical Staff

ITEM(2): THERMAL HYDRAULIC ISSUES RELATED TO PASSIVE PLANT DESIGNS

The following thermal hydraulic issues are to be discussed:

- NRC Test and Analysis Programs (TAPs) related to the AP600 and Simplified Boiling Water Reactor (SBWR) Passive Plant Designs

- Status of Issues Associated with Westinghouse Test and Analysis Programs Being Conducted in Support of AP600 Passive Plant Design Certification

a. NRC Test and Analysis Programs (TAPs) Related to the AP600 and SBWR Passive Plant Designs

AP600 - The Committee has provided comments on the NRC-RES AP600 TAPs in its letters of April 12, 1995 and November 10, 1994. In general, the ACRS believes that the RES TAPs have dramatically improved over the last 6-8 months and that RES is now performing thorough analyses of the data from tests conducted by Westinghouse at the Oregon State University (OSU) and SPES facilities as well as the data from the tests conducted at the ROSA facility. Some issues of potential concern at this time are:

- Potential for Water Hammer - Attention needs to be given to identification of where and under what circumstances water hammer could occur in the AP600 design. The safety significance of this matter should be assessed soon in order to avoid potential impact of the design certification schedule. RES thermal hydraulic consultants should be helpful in this matter.

- Applicability of TRAC & RELAP5 Codes - The need for these codes to model thermal hydraulic behavior for which they have been shown to be weak is of concern for modeling of the AP600 (and SBWR) designs. The ACRS has recommended that RES consider development of some sort of "fall-back" position, should the codes fail to adequately predict these key phenomena.

- Scaling Analysis - RES is developing scaling criteria to be applied to the ROSA, SPES, and OSU facility test results. The Committee has recommended that the scaling analysis performed at OSU and sponsored by Westinghouse be used as a starting point by RES for its analysis.

  In its May 8, 1995 letter of response, the NRC staff indicated that it is responding to the above concerns.

/

SBWR - The Committee last commented on the RES SBWR TAP in its letter of November 10. 1994. The centerpiece of the RES test program is the Purdue University Multi-Dimensional Integral Test Assembly (PUMA) facility. In its November 10, 1994 letter, the ACRS made comments relative to the scaling analysis, associated code development work, and details associated with the PUMA facility and test program. In its December 28, 1995 letter of response (pp.), the staff indicated that all of the ACRS comments have been or will be addressed.

The Committee plans to meet with RES representatives to continue its review of this matter in the near future.

b. Status of Issues Associated With Westinghouse TAPs Being Conducted in Support of the AP600 Passive Plant Design Certification

The ACRS Subcommittee on Thermal Hydraulic Phenomena has continued its review of the Westinghouse TAPs. Meetings have been held on the test and analysis programs supporting the development of the Passive Containment System, and the Westinghouse computer code to be used for realistic (best-estimate) LOCA analysis (COBRA/TRAC). The Subcommittee review of these matters is still under way.

During its May 1995 meeting, the Committee discussed NRR's review methodology for use of COBRA/TRAC codes. In a May 17, 1995 letter to the EDO, the Committee expressed concerns regarding the staff's review methods. The central concern was that NRR did not have in place a comprehensive review procedure to ensure that the provisions of the revised ECCS rule will be met. Concern was also expressed over a lack of adequate documentation associated with the staff's review of the Westinghouse best-estimate code.

The Committee's review of the Westinghouse TAPs will continue, consistent with the progress of both Westinghouse and the staff.

Attachments:

- Letter dated April 12, 1995, from T. S. Kress, Chairman ACRS, to J. Taylor, Executive Director for Operations, Subject: NRC Test and Analysis Program in Support of AP600 Advanced Light Water Passive Plant Design Review (pp. 4-6)

- Letter dated November 10, 1994, from T. S. Kress, Chairman ACRS, to J. Taylor, Executive Director for Operations, Subject: NRC Test and Analysis Programs In Support of AP600 and SBWR Advanced Light Water Reactor Passive Plant Design Certification Reviews (pp. 7-11)

- Letter dated May 8, 1995, from J. Taylor, Executive Director for Operations, to T. S. Kress, Chairman ACRS, Subject: Staff Response to ACRS Letter dated April 12, 1995, on NRC Test and Analysis Program in Support of AP600 Advanced Light Water Passive Plant Design Reviews (pp. 12-17)

- Letter dated December 28, 1994, from J. Taylor, Executive Director for Operations, to T. S. Kress, Chairman ACRS, Subject: Staff Response to ACRS Letter dated November 10, 1994, on NRC Test and Analysis Programs in Support of AP600 and SBWR Advanced Light Water Reactor Passive Plant Design Certification Reviews (pp. 18-25)

- Letter dated May 17, 1995, from T. S. Kress, Chairman ACRS, to J. Taylor, Executive Director for Operations, Subject: Review of Best-Estimate Models for Evaluation of Emergency Core Cooling System Performance (pp. 26-28)

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

April 12, 1995

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

Dear Mr. Taylor:

SUBJECT: NRC TEST AND ANALYSIS PROGRAM IN SUPPORT OF AP600 ADVANCED LIGHT WATER PASSIVE PLANT DESIGN REVIEW

During the 420th meeting of the Advisory Committee on Reactor Safeguards, April 6-7, 1995, we discussed the confirmatory test and analysis program being conducted by the Office of Nuclear Regulatory Research (RES) in support of the design certification review for the Westinghouse AP600 advanced light water reactor. During this meeting, we had the benefit of discussions with representatives of RES and its contractor, the Idaho National Engineering Laboratory. Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on March 27-28, 1995, to discuss this matter. The Committee previously reviewed this matter during its October and November 1994 meetings and provided formal comments in its November 10, 1994 letter. We also had the benefit of the documents listed.

During the past year, the RES thermal-hydraulic program has undergone a dramatic change for the better. The presentations made to the Thermal Hydraulic Phenomena Subcommittee and the Committee were clear, well-organized, and demonstrated good technical thinking. We compliment the management, the staff, and the contractors for the improvement. We also note that RES is making good use of a cadre of high-quality thermal-hydraulic consultants.

Completion of the Phenomena Identification and Ranking Table (PIRT) for the AP600 remains an important task. It was much easier to develop the PIRT for the current operating plants because a great deal of relevant test data were available. This is not the case for the AP600 and SBWR passive plants. Development of the PIRT should be concurrent with a scaling analysis and review of test results to provide quantitative support for the engineering judgments that must be made. The RES approach appears to be systematic and well organized. We recommend, however, that RES fully document the development of the PIRT.

The RES analysis of test data from ROSA and Oregon State University (OSU) was very thorough. We encourage the staff to continue such efforts, while drawing on the insights from the ongoing scaling analysis. RES should strive to provide complete documentation of the test analysis effort and should also document the phenomena that are not important.

The ongoing RES scaling analysis for the test facilities is an important effort. This analysis can be used to assess the impact of scaling distortions and atypicalities of the different facilities to support the conclusions of PIRT as well as to understand the physical phenomena important to AP600 thermal-hydraulic

behavior. For the current operating plants, the PIRT was developed for existing systems whose thermal-hydraulic behavior was demonstrated over a 20-year period. For the AP600 design certification review, however, comparable understanding must be gained quickly. We believe that a rigorous analysis of test data based on the use of a good scaling analysis and the PIRT should permit this to be done. We recommend that the OSU scaling effort performed in support of the Westinghouse test program be a starting point for the development of a consistent set of AP600 scaling criteria for application to the ROSA, OSU, and SPES test facilities.

Several issues were discussed during our meetings with RES. The first is the potential for water hammer in the AP600 design during LOCAs. Attention should be given to identifying where and under what circumstances water hammer could occur. A second is the potential for thermal stratification in the Core Makeup Tank, the Incontainment Refueling Water Storage Tank, and in the horizontal pipe runs of the reactor coolant system. The occurrence of thermal stratification in the cold leg combined with the possibility of steam injection could be a precursor to a significant water hammer. We recommend that the potential safety problems caused by these phenomena be identified and their significance to safety be assessed soon in order to avoid questions at the time of certification. The RES thermal-hydraulic consultants could be very helpful in this regard.

We are concerned about the applicability of the present thermal-hydraulic codes (TRAC, RELAP5) for analysis of plants like the AP600. These codes have to predict types of thermal-hydraulic behavior for which they have been shown to be weak; i.e., prediction of condensation, thermal stratification, and water level. We recommend that RES consider developing a contingency plan in the event that the codes cannot adequately predict these key phenomena.

Although the focus of our meetings with RES was on the development of the PIRT, some reference was made to determination of computational uncertainty. The uncertainty parameter of choice is peak clad temperature for the large-break LOCA while reactor vessel primary system inventory is the choice for the small-break LOCA. With resources being reduced, we recommend that RES focus its attention on the more safety-significant small-break LOCA.

Overall, much progress in the RES thermal-hydraulic program is evident. It is well structured and will yield a great deal of valuable insight into the behavior of passive plants.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Memorandum dated February 14, 1995, from M. Wayne Hodges, NRC Office of Nuclear Regulatory Research, to John T. Larkins, ACRS Executive Director, transmitting INEL draft report, "Interim Phenomena Identification and

Ranking Tables for Westinghouse AP600 Small Break Loss-of-Coolant Accident, Main Steam Line Break, and Steam Generator Tube Rupture Scenarios," INEL-94/0061

2. Memorandum dated February 14, 1995, from M. Wayne Hodges, NRC Office of Nuclear Regulatory Research, to John T. Larkins, ACRS Executive Director, transmitting LANL draft report by B. Boyack, "AP600 Large-Break Loss-of-Coolant Accident Phenomena Identification and Ranking Tabulation"
3. Letter dated February 15, 1995, from Gary E. Wilson, INEL, to Tim Lee, NRC, Subject: Transmittal of AP600 T/H Consultants Meeting Minutes
4. ACRS report dated November 10, 1994, from T. S. Kress, ACRS Chairman, to James M. Taylor, Executive Director for Operations, Subject: NRC Test and Analysis Programs in Support of AP600 and SBWR Advanced LWR Passive Plant Design Certification Reviews

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

November 10, 1994

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

Dear Mr. Taylor:

SUBJECT: NRC TEST AND ANALYSIS PROGRAMS IN SUPPORT OF AP600 AND SBWR ADVANCED LIGHT WATER REACTOR PASSIVE PLANT DESIGN CERTIFICATION REVIEWS

During the 414th and 415th meetings of the Advisory Committee on Reactor Safeguards, October 6-7 and November 3-4, 1994, we discussed the confirmatory test and analysis programs being conducted by the Office of Nuclear Regulatory Research (RES) in support of the design certification reviews for the Westinghouse AP600 and GE Nuclear Energy (GENE) Simplified Boiling Water Reactor (SBWR) advanced light water reactors. During these meetings, we had the benefit of discussions with representatives of RES. Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on August 25-26, 1994, to discuss this matter. We also had the benefit of the documents referenced.

In the absence of a full-scale test facility, an understanding of the thermal hydraulic behavior of a passive plant design will depend on the use of computer codes. The NRC staff has decided to modify RELAP5/MOD3 for its confirmatory thermal hydraulic analysis of the AP600 and SBWR designs. The important phenomena the code must simulate should be delineated in the Phenomena Identification and Ranking Table (PIRT), thus allowing one to formulate integral and separate effects experiments that will yield appropriate data for code validation. Code validation should be an integrated process involving code development, experimentation, and an understanding of the physics of two-phase flow and heat transfer.

The major objective of the thermal hydraulic code development effort should be to produce a code capable of predicting the behavior of a full-scale nuclear power plant with acceptable uncertainties. For existing nuclear plant designs, we have had the benefit of many integral and separate effects experiments at a wide variety of scales to help arrive at an estimate of the uncertainties in the code predictions. We are now dealing with two passive plant designs which evidence more complex thermal hydraulic system

dynamics, and for which there is a paucity of relevant experimental data. There are several causes for this more complex dynamic behavior: (1) steam condensation at low pressure, (2) use of gravity-driven coolant injection, and (3) the existence of many components and complex hydraulic paths that give the system many degrees of freedom. Understanding this dynamic behavior requires evaluation of scale distortion effects and dynamic characteristics in the various test facilities. In this regard, two questions should be addressed and resolved: (1) is the evolution of a particular transient influenced by configurational and/or scale distortions, and (2) do configurational and/or scale distortions in the various test facilities preclude simulation of some important dynamic effects while introducing other dynamic effects that may not be important in a full-scale plant design? To address these questions, a top-down scaling analysis must be performed.

The NRC staff has test and analysis programs under way to address issues arising during its evaluation of the AP600 and the SBWR designs. The AP600 evaluation will be supported by testing at the Japan Atomic Energy Research Institute ROSA-V facility and the use of RELAP5/MOD3. The SBWR evaluation will be supported by testing at the Purdue University PUMA facility and the use of RELAP5/MOD3. We believe that the use of RELAP5/MOD3 for both AP600 and SBWR simulations will lead to the development of a more robust computational tool. Both programs are discussed below and some comments about the technical direction of these programs are provided.

AP600 Program

The PIRT in support of the AP600 analysis has not yet been completed. There is no indication that a PIRT was utilized for allocating resources, for assigning test objectives, or for developing the test matrices. It is necessary to complete the PIRT and confirm it on the basis of relevant scaling groups. To ensure that RELAP5/MOD3 can simulate the high ranking phenomena, specific tests in the test matrix should be associated with the high ranking phenomena in the PIRT. By doing this, all important phenomena will be addressed.

The PIRT and a proper scaling analysis for the AP600 would cover all test facilities for AP600. Unfortunately, the scaling efforts conducted for the OSU, SPES, and ROSA-V test facilities were not coordinated. The global scaling of the AP600 design, including consideration of the dynamic interactions between the major system components (pressure vessel, core makeup tank, pressurizer, steam generators, passive residual heat removal system, and accumulators), was omitted. Depressurization is not scaled, even though the methods for doing so are known. The scaling analysis for OSU, while still incomplete, could serve as a model for ROSA and SPES.

Direct counterpart tests in ROSA, OSU, and SPES are not possible. This makes it difficult to extrapolate the observed thermal hydraulic behavior to full scale. A well-planned effort to integrate experiments with code improvement and assessment is needed to quantify uncertainties. At present, RELAP5/MOD3 predicts strong oscillations both when they are observed in tests and when they are not. Consequently, the calculated behavior can neither be attributed conclusively to numerical nor physical effects. The mechanisms by which the various observed modes of oscillation are initiated and maintained need to be understood so that their potential influence on the thermal hydraulic behavior of the AP600 can be evaluated. The judicious selection of test conditions for the facilities, together with the conduct of a careful data analysis and scaling, should provide a satisfactory solution.

The demonstrated propensity for condensation oscillation events in the AP600 points to a need to identify both the likelihood and damage potential of water hammer events. Furthermore, the influence of thermal stratification on the thermal hydraulic behavior of the AP600 also remains to be evaluated.

SBWR Program

The objective of the PUMA test program is to obtain data for assessing computer code simulation of important SBWR-specific phenomena. The focus of this test program is on the operability of the passive cooling systems and their interactions with the reactor vessel.

Again, a PIRT has not been completed. The PIRT effort should be brought to a close so that a proper evaluation of PUMA and the GENE test facilities (GIST, GIRAFFE, and PANDA) can be made.

Scaling of phenomena identified in the Purdue University preliminary PIRT has been a major part of the PUMA test program. At present, the scaling effort has primarily focused on the details of local phenomena whereas global scaling appears to be incomplete. To preclude atypicalities in the interactions of the various systems and to help determine an appropriate set of initial and operating conditions for the PUMA system, the scaling of the global dynamic component interactions (among the reactor vessel, drywell, wetwell, PCCS, ICS, and GDCS) should be completed before the facility design is frozen.

We are pleased to see that one of the PUMA program principal investigators is a code developer. Input from a code developer on the selection of instrument type, number, and location will yield a much more useful set of data for code assessment.

The PUMA facility will allow testing that both overlaps and extends the accident period covered by the GENE test facilities (GIST, GIRAFFE, and PANDA), while allowing the simulation of a broad spectrum of postulated accidents. This should be helpful in confirming the validity of the results obtained at the GENE facilities.

The following comments are specific to the PUMA program:

- The current plan is to measure the heat transfer characteristics and infer the noncondensible gas concentration. We would like to point out that knowledge of the noncondensible gas distribution is fundamental and necessary if one is to avoid compensating errors in the computational process. We recommend that the noncondensible gas concentration be measured directly at several locations.

- The test matrix does not include a long-duration test. We believe it should because the SBWR containment performance requirement is 72 hours, which scales to 144 hours of PUMA test time.

- Since the interface temperature of the suppression pool is directly coupled to the containment pressure, an evaluation of thermal stratification in the pool is needed.

- Some tests should be conducted with initial nitrogen concentrations in the drywell to evaluate the impact of steam line breaks outside containment.

- The planning of the PUMA experiments should include consideration of phenomena arising as a consequence of failures of active mitigating systems.

- Data analysis and evaluation are not part of the contract with Purdue University. This is unfortunate because in this case the principal investigators at Purdue University are highly qualified for such a task. Further, those conducting the testing can bring valuable insights to the process. We recommend that the contract with Purdue University be modified to include a data analysis and evaluation task.

Technical Oversight

The RES staff now plans technical oversight of thermal hydraulic research for the AP600 and the SBWR through the Advanced Light Water Reactor Thermal Hydraulic Research Integration Group (ATRIG). This unwieldy ATRIG is not the technical oversight recommended by the ACRS in the past and subsequently approved by the Commission. Lessons learned from the CSAU program should be remembered. A small (5 or 6 members) cohesive group with well-qualified leader-

ship is needed to integrate the technical issues of scaling, data collection, data analysis, and code development.

Sincerely,

T. S. Kress
Chairman

References:

1. "Summary of the LSTF Characterization Tests Performed in Conjunction with the ROSA/AP600 Experiments," R. A. Shaw, et al., Draft report dated August 1, 1994, transmitted by memorandum dated August 5, 1994, from G. S. Rhee, Office of Nuclear Regulatory Research
2. U. S. Nuclear Regulatory Commission, Draft NUREG/CR, PU-NE 94/1, Subject: Scientific Design of Purdue University Multi-dimensional Integral Test Assembly (PUMA) for GE SBWR, July 1994, transmitted by memorandum dated August 4, 1994, from J. T. Han, Office of Nuclear Regulatory Research
3. Memorandum dated August 8, 1994, from M. Ishii, Purdue University, to J. Han, U. S. Nuclear Regulatory Commission, transmitting replacement pages for the report, "Preliminary Scientific Design of Purdue University Multi-dimensional Integral Test Assembly (PUMA) for GE SBWR"
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6066, EGG-2705, "Scaling and Design of LSTF Modifications for AP600 Testing," T. J. Boucher, et al., August 1994
5. SECY-94-138, memorandum dated May 20, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Confirmatory High Pressure Integral System Testing of the Westinghouse AP600 Safety Systems
6. "Quick Look Report for ROSA/AP600 Experiment AP-CL-03," R. A. Shaw, et al., undated rough draft, transmitted by memorandum dated August 5, 1994, from G. S. Rhee, Office of Nuclear Regulatory Research
7. Advisory Committee on Reactor Safeguards Report, dated November 18, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: NRC Confirmatory Test Program in Support of the AP600 Design Certification

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

May 8, 1995

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

SUBJECT: STAFF RESPONSE TO ACRS LETTER DATED APRIL 12, 1995, ON NRC TEST AND ANALYSIS PROGRAM IN SUPPORT OF AP600 ADVANCED LIGHT WATER PASSIVE PLANT DESIGN REVIEWS

Dear Dr. Kress:

The staff met with the ACRS Subcommittee on Thermal Hydraulic Phenomena on March 27-28, 1995, and the Full Committee April 6-7, 1995, to discuss the confirmatory test and analysis programs being conducted by the Office of Nuclear Regulatory Research (RES) in support of the certification review for the Westinghouse AP600 advanced light water reactor. The ultimate objective of these programs is to produce a version of RELAP5 system computer code that is assessed against test data and which can then be used with reasonable confidence to analyze the behavior of the AP600 plant under a wide variety of transient and accident conditions. The results of the ACRS review are reported in a letter dated April 12, 1995, from you to me, "NRC Test and Analysis Programs in Support of AP600 Advanced Light Water Passive Plant Design Review." We are encouraged by the constructive interactions between the staff and the ACRS Thermal Hydraulic Phenomena Subcommittee and are appreciative of the ACRS comments on these programs.

We note for the most part the comments were favorable to the staff efforts on AP600 test and analysis programs. However, your letter also expressed a concern about the potential that the present thermal hydraulic codes (TRAC, RELAP5) might not be applicable to analyzing the AP600 and recommended that RES consider developing a contingency plan in the event that the codes cannot adequately predict some key phenomena important for AP600. While it is the responsibility of the vendor to perform experiments and analyses necessary to support design certification, we consider this issue to be very critical to the success of the NRC independent confirmatory calculations of the AP600. The staff believes the RELAP5 code, which is being used to analyze the more safety-significant small break LOCA, will contain appropriate models to address issues associated with AP600. However, to ensure that your concern is addressed further, we have requested the thermal hydraulic consultants to review the issue and assess the adequacy of the RELAP5 code to address the safety issues important to the staff's independent assessment of AP600. We also requested the thermal hydraulic consultants to identify any reliable alternative tool or method that may be used by the staff to quickly perform extensive sensitivity analyses to ensure that the results obtained by RELAP5 are adequate. The staff will work closely with the consultants and the ACRS Thermal Hydraulic Phenomena Subcommittee to reach an early resolution to this issue.

The staff response to the above issue and other ACRS comments is provided in Enclosure 1 to this letter.

Finally, we believe that the computation of uncertainties associated with the peak clad temperature for the large-break LOCA is not necessary because of the large margin provided by the AP600 design. Therefore, we agree with the ACRS that with resources being reduced, it is better to focus our effort on the more safety-significant small-break LOCA.

Sincerely,

James M. Taylor
Executive Director
for Operations

Enclosure: As stated

cc: The Chairman
Commissioner Rogers
Commissioner de Planque
Commissioner Jackson
SECY

ENCLOSURE 1
Response to ACRS Comments
Letter to James M. Taylor dated April 12, 1995,
"Subject: NRC Test and Analysis Program in Support of
AP600 Advanced Light Water Passive Plant Design Review."

The ACRS comments are grouped into 4 categories: 1) documentation of the development of the Phenomena Identification and Ranking Table (PIRT), and of the test analysis effort, 2) use of the Westinghouse OSU scaling as starting point for the development of consistent set of AP600 scaling criteria, 3) the potential for water hammer, and for thermal stratification in the Core Makeup Tank, the Incontainment Refueling Water Storage Tank, and in horizontal pipe runs, and 4) code inadequacies. The staff response to each of these categories is provided below.

1. Documentation of: the PIRT Development, and Test and Analysis Effort

The ACRS noted that development of the PIRT should be concurrent with a scaling analysis and review of test results to provide quantitative support for the engineering judgements that must be made. The ACRS recommended that RES fully document the development of the PIRT. The ACRS also noted that the RES analysis of test data from ROSA and OSU was very thorough and recommended that RES should strive to provide complete documentation of the test analysis effort and also document the phenomena that are not important.

We agree with the ACRS that PIRT development should be concurrent with a scaling analysis and review of test results. Our contractor at INEL is tasked to integrate the results of data evaluation, scaling analysis and code assessment using the PIRT as guide. The thermal hydraulic consultants, charged with ensuring that the above integration is systematic, auditable, and traceable, are doing research in a more individual fashion in the areas of their expertise to support the development of scaling analysis, the evaluation of the experimental data, verifying the validity of the AP600 PIRT, and in assessing the RELAP5 development plans.

Integration of the above efforts will be centered initially around four tests: 1-inch cold leg break (1" CLB), 200% direct vessel injection line break, 2-inch cold leg pressure balance line break, and inadvertent ADS-1 opening. These tests were conducted in the three integral test facilities (OSU, ROSA, and SPES) and represent scenarios that challenge the passive safety features of the AP600: CMT, Accumulators, PRHR, ADS, and IRWST. Additional tests may be included if anomalous behavior is observed. The overall objective of the integration is to assess and enhance the capabilities of the RELAP5 code to predict AP600 behavior. Other specific objectives include: understand the phenomena that determine the behavior of the plant, identify and understand the effects of configuration and scaling distortions, assess the code's ability to predict and reproduce the relevant phenomena observed in the tests, and verify the PIRT. The results from the first scenario (1" CLB) will be presented to the thermal hydraulic consultants on June 13-15, 1995. We will present the result of the 1" CLB to the ACRS Thermal Hydraulic Phenomena Subcommittee shortly thereafter, after implementing the Consultants comments.

We also fully concur with the ACRS that documentation is essential for an efficient and successful use of the data and to communicate the staff's sound understanding of the important physical phenomena that occur during transients. More importantly, documentation will provide the rationale why certain phenomena are not considered important. The staff and its contractors are working arduously on developing this documentation. We will continue to provide the ACRS Thermal Hydraulic Phenomena Subcommittee with these documents as they become available.

2. Use of The Westinghouse OSU Scaling Effort

The ACRS recommended that the OSU scaling performed in support of the Westinghouse test program be a starting point for the development of a consistent set of AP600 scaling criteria for application to the ROSA, OSU, and SPES test facilities.

As indicated in response to item one above, selected members of the thermal hydraulic consultants are working with the staff's contractor at INEL to develop global scaling methodology that incorporates the dynamic response and interactions of AP600. One of those consultants, Professor Jose Reyes of Oregon State University, is the author of the OSU scaling methodology. It is our intention, to the maximum extent practicable, to use the OSU scaling so as not to repeat that effort. A Bond Graph model of the system configuration is being used to define a complete set of governing equations for the behavior of the different tanks, pipes, and heat transfer mechanisms in the AP600. The global non-dimensional equations and their coefficients for each phenomena, for the different components in all facilities can be compared directly and the source of distortion can be identified. A demonstration of the methodology using the results from the 1" CLB will be presented to the thermal hydraulic consultants in June 13-15, 1995. We will present the global scaling methodology results when we present the result of the 1" CLB to the ACRS Thermal Hydraulic Phenomena Subcommittee.

In addition to the above global scaling approach, we are developing a bottom-up or component scaling approach which will evaluate all available data to address the mechanism associated with key physical phenomena that control the overall behavior of AP600; e.g., CMT draining, ADS depressurization, IRWST injection. The method also will look at different oscillatory behavior observed in the test facilities to ascertain if the behavior is due to facility distortion or is expected to occur in the AP600. This effort will include what caused the oscillations to start and stop and determine the applicability of these phenomena to the AP600.

3. Water Hammer and Thermal Stratification

The ACRS indicated that potential safety problems caused by water hammer and thermal stratification should be identified and their significance to safety assessed soon in order to avoid questions at the time of certification. The ACRS also indicated that thermal hydraulic consultants could be very helpful in this regard.

As stated in the meetings on March 27-28, 1995, with the ACRS Thermal Hydraulic Phenomena Subcommittee, NRR has requested Westinghouse to address the issue of water hammer for AP600. Nonetheless, we concur with the ACRS that the thermal hydraulic consultants can, and indeed are providing excellent help in this regard. We plan to present to the full thermal hydraulic consultants group during the June 13-15, 1995, meeting, the results of Professor Griffith's examination of the conditions that are conducive to, and the magnitude of load associated with, water hammer events in AP600. Again we will present the result of this work to the ACRS Thermal Hydraulic Phenomena Subcommittee upon implementing the consultants comments.

4. Code Inadequacies

The ACRS expressed a concern about the applicability of the present thermal hydraulic codes (TRAC, RELAP5) for analysis of plants like AP600. The ACRS believes that these codes cannot adequately predict key thermal hydraulic behavior such as condensation, thermal stratifications, and two-phase level tracking.

In order to gain confidence in the code's ability to predict the full scale behavior, it is necessary to show that RELAP5 reproduces phenomena believed to be important to the overall AP600 system behavior during transient and accident conditions. It is also necessary to show either that the distortions caused by differences of scale and geometry in the different test facilities are not important to the processes under investigation, or that the distortions are characterized, understood, and accounted for when using the RELAP5 code to analyze the full size plant. Toward this end, we tasked INEL to use the PIRTs to determine if the RELAP5 code has adequate models to represent the highly ranked phenomena and whether these models are assessed against data that cover the range of interest to the AP600. The above report will be forwarded, by May 31, 1995, to selected members of the Thermal Hydraulic Consultants for their review, comments, and presentation at the full thermal hydraulic consultant meeting that will be held June 13-15, 1995. Should the Subgroup and the full thermal hydraulic consultant group conclude that the RELAP5 has limitations for the AP600 application, we will move expeditiously to assess the need to develop additional capabilities to address that concern. We are not proposing an alternative code to replace the RELAP5 code, since such an alternative code, if it exists, will require substantial preparation and assessment that will not be feasible to accomplish within the time frame available to certify the AP600. However, since our understanding of the key phenomena during the different phases of AP600 transients is vastly improving through our thorough evaluation of the different AP600 test data, performing scaling analysis, and help from the thermal hydraulic consultants, we believe that special phenomenological models can be developed to characterize and bound the key phenomena of interest.

Even though the AP600 RELAP5 calculation is running very slowly and is exhibiting some anomalies that will be corrected, we believe that the RELAP5 code is suitable for AP600 application. INEL is tasked: 1) to upgrade the two phase-level tracking model based on work developed by Professor Mahaffy at Penn State, and 2) to upgrade the interfacial heat transfer package for handling direct contact condensation occurring in the AP600 steam generator

outlet plenum, PRHR return, and the RPV upper plenum. With regard to thermal stratification in the cold leg and the CMT, a one dimensional code like RELAP5 cannot handle this situation properly. Therefore, instead of spending tremendous resources on developing such capabilities in RELAP5, it is our strategy to perform sensitivity studies to bound the behavior as it affects key phenomena such as break flow and CMT draining. As stated earlier, our plan to address other effects of thermal stratification; i.e, water hammer, is being developed by Professor Griffith. The approach is to develop a method 1) to identify the conditions in the AP600 that are conducive to water hammer, and 2) calculate the magnitude and duration of these events. We have requested the views and recommendations of the thermal hydraulic consultants on the above strategy.

It is the staff view that the activities discussed above will address all the concerns raised by the ACRS in its letter.

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

December 28, 1994

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

SUBJECT: STAFF RESPONSE TO ACRS LETTER DATED NOVEMBER 10, 1994, ON NRC TEST AND ANALYSIS PROGRAMS IN SUPPORT OF AP600 AND SBWR ADVANCED LIGHT WATER REACTOR PASSIVE PLANT DESIGN CERTIFICATION REVIEWS

Dear Dr. Kress:

The staff met with the ACRS Subcommittee on Thermal Hydraulic Phenomena on August 25-26, 1994, and the Full Committee on October 6, 1994, to discuss the confirmatory test and analysis programs being conducted by the Office of Nuclear Regulatory Research (RES) in support of the certification reviews for the Westinghouse AP600 and General Electric Nuclear Energy Simplified Boiling Water Reactor (SBWR). Because of concerns raised by the ACRS Subcommittee on Thermal Hydraulic Phenomena regarding the need to integrate the technical issues of scaling, data collection, data analysis, and code development, the staff presentation to the Full Committee focused on the RES plan to integrate the different elements of these programs. The ultimate objective of these programs is to produce a version of the RELAP5 system computer code that is assessed against test data which can then be used with reasonable confidence to analyze the behavior of the AP600 and SBWR plants under a wide variety of transient and accident conditions. The results of the ACRS review are reported in your letter to me dated November 10, 1994, "NRC Test and Analysis Programs in Support of AP600 and SBWR Light Water Reactor Passive Plant Design Certification Review." We are appreciative of the ACRS comments on these programs. The staff has reviewed the ACRS comments, and its response is provided in Attachment 1 to this letter.

Sincerely,

Hugh L. Thompson Jr.

James M. Taylor
Executive Director
for Operations

Attachment:
As stated

cc: The Chairman
Commissioner Rogers
Commissioner de Planque
SECY

ATTACHMENT 1

Response to ACRS Comments

Letter to James M. Taylor dated November 10, 1994, "Subject: NRC Test and Analysis Programs in Support of AP600 and SBWR Light Water Reactor Passive Plant Design Certification Review."

The ACRS comments are grouped into six categories: 1) the need for top-down scaling analysis, 2) use of Phenomena Identification and Ranking Table (PIRT), 3) counter part tests in OSU, SPES, and ROSA-V, 4) condensation oscillation in AP600 and damage potential of water hammer events, 5) comments on NRC's plans for testing at Purdue University's PUMA facility, and 6) technical oversight and integration efforts. The staff response to each of these categories is provided below.

1. The Need for Top-Down Scaling Analysis

a) The ACRS indicated that a top-down scaling analysis must be performed in order to address the following two questions:

1. Is the evolution of a particular transient influenced by configurational and/or scale distortions?

2. Do configurational and/or scale distortions in the various test facilities preclude simulation of some important dynamic effects while introducing other dynamic effects that may not be important in a full-scale plant design?

We fully concur with the ACRS that the above two questions must be addressed as part of our integration effort to use the data from the different test facilities to verify the RELAP5 system code. In order to gain confidence in the code's ability to predict the full scale behavior, it is necessary to show that the RELAP5 sufficiently models phenomena important to the overall AP600 and SBWR system behavior during transient and accident conditions. It is also necessary to show either that the distortions caused by differences of scale and geometry in the different test facilities are not important to the processes under investigation, or that the distortions are characterized, understood, and accounted for when using the RELAP5 code to analyze the full size plant. We have formed a scaling subgroup to address these issues. The review of the experimental data, code development and assessment, and scaling issues will be done in a thorough and timely fashion and will address the above questions. To assist us in this regard, we are soliciting input from a range of well known experts in the thermal hydraulic field (more detail about the thermal hydraulic experts is provided later, Item 6). The results of our reviews will be documented and presented to the ACRS, and the ACRS is invited to participate in the reviews.

Development of scaling groups to describe the components and hydraulic paths that give the AP600 system behavior many degrees of freedom requires extensive manipulation of equations which describe the different interactions that can occur between the controlling phenomena for a wide spectrum of design basis accident scenarios. This process can be best accomplished initially, as the staff has done, by using a system level code like RELAP5. Additionally, we are developing an in-depth understanding of the major physical phenomena that are taking place. Since the code embodies models that approximate system behavior, its use in conjunction with independent analyses of the test data is appropriate. This approach has been discussed with the thermal hydraulic expert group to define the additional steps needed to ensure the robustness of the approach. Therefore, while the staff did not perform the analysis using the same method used by OSU, as suggested by the ACRS, our approach is still appropriate, and it accounts for all dynamic interactions listed in the ACRS letter. With data becoming available from the three AP600 test facilities (OSU, SPES, ROSA-V), the staff, its contractors, and the thermal hydraulic consultants are analyzing the data in detail, guided by the PIRT (discussed later), to address known configurational distortions, and local scaling distortions, to determine if similar system behavior is observed in all test facilities. If this is the case, a non-dimensional representation of the phenomena will be developed to describe the system behavior. This analysis will be performed for the three phases that accidents in AP600 go through, i.e., high pressure phase, ADS actuation phase, and IRWST injection phase. If the observed behavior is different among the different facilities, additional data analysis will be conducted to determine the cause of the discrepancy and describe the processes that are taking place. Non-dimensional groups will be developed to describe the system behavior. If additional tests are needed to resolve a particular concern, both the ROSA-V and the NRC-sponsored OSU facilities can be used to provide the needed information.

b) With regard to the ACRS statement that the SBWR scaling effort has primarily focused on the details of the local phenomena whereas global scaling appears to be incomplete, it should be noted that global scaling was completed "before" the PUMA facility design was finalized. We believe that the confusion might be caused by the terminology used in the draft PUMA design report (a copy was sent to ACRS in August 1994). The report will be revised to indicate that the PUMA scaling consists of (1) global scaling or top-down approach (namely, integral system scaling and mass and energy inventory and boundary flow scaling), and (2) local phenomena scaling or bottom-up approach. Together, the two approaches provide a scaling methodology that is practical and yields technically justifiable results. More detailed derivation of the integral system scaling is given in NUREG/CR-3267, -3420, and -4584 reports.

2. Use of Phenomena Identification and Ranking Table (PIRT)

The ACRS noted that the PIRT in support of the AP600 analysis has not yet been completed and there is no indication that a PIRT was used for allocating resources, for assigning test objectives, or for developing the test matrices.

While we agree with the ACRS that the AP600 PIRT is not complete, it is worth mentioning that the nature of an ongoing development and assessment program is that the PIRT will not be complete until the assessment steps are complete and it can be shown that the important phenomena have indeed been identified and ranked appropriately. A preliminary PIRT was developed based upon the results of calculations and our judgement as to the relative importance of the thermal hydraulic phenomena expected. This PIRT was used to determine the applicability and limitations of the ROSA facility for AP600 testing. This PIRT was also used to formulate a development plan for RELAP5 MOD3 which led to the development of Version 3.2. Although not formally documented, this preliminary PIRT was discussed with the ACRS Subcommittee on Thermal Hydraulic Phenomena. Subsequent to the preliminary PIRT, we have been working on an "interim" PIRT which reflects the results of the latest AP600 design changes and analyses. Now that experimental data are becoming available, the PIRT is being revised to reflect our understanding of the relative importance of the thermal hydraulic phenomena observed.

We have instructed our contractor at INEL to document the "interim" PIRT. We will submit the "interim" PIRT to the ACRS in early March 1995. We then plan to meet with the ACRS and brief them about the changes resulting from our review of the now available test data. It should be noted that after only a few months of analyzing results, it is estimated that about 70% of the "interim" PIRT would need to be revised to bring it to our current level of understanding. It should also be noted that some of the ACRS thermal hydraulic consultants observed the formal review of the "interim" PIRT which was carried out between October 1993 and September 1994.

It should also be noted that the development of an SBWR PIRT is in the interim phase. However, it is being used by Purdue University as a basis for scaling and facility design, and for outlining a test matrix to cover the range of phenomena and processes identified.

3. Counter Part Tests in OSU, SPES, and ROSA-V

The ACRS indicated that "direct counterpart tests in OSU, SPES, and ROSA-V are not possible. This makes it difficult to extrapolate the observed thermal hydraulic behavior to full scale. A well-planned effort to integrate experiments with code improvement and assessment is needed to quantify uncertainties."

While we agree with the ACRS that exact counterpart tests among all facilities are not possible, we are confident that our methodical approach of: data analysis, test calculation and analysis, PIRT, and scaling analysis will be able to identify the AP600 dominant phenomena, identify code improvements, and quantify uncertainties. The main objective of the data analysis is to identify and understand the phenomena, their ranges and interactions. The main objective of the test calculation and analysis is to determine the code capability to predict the phenomena observed in the tests and their ranges.

4. Condensation Oscillation in AP600, Damage Potential of Water Hammer Events

a) The ACRS stated that "The demonstrated propensity for condensation oscillation event in the AP600 points to a need to identify both the likelihood and damage potential of water hammer events. Furthermore, the influence of thermal stratification on the thermal hydraulic behavior of the AP600 also remains to be evaluated."

We agree with the ACRS. We identified both concerns to Westinghouse, and specifically recommended that Westinghouse run additional tests to look for condensation events in SPES; one such test with scaled PRHR capacity equivalent (approximately) to ROSA was performed. The results are currently under evaluation.

Both the thermal stratification and water hammer issues are being carried in the AP600 DSER as Open Items for Westinghouse to resolve. We plan to examine the condensation behavior observed in all test facilities to ensure that the RELAP5 code can predict the period when severe condensation occurs. The staff will use other tools to understand and properly account for any three-dimensional behavior that contributes to or exacerbates the condensation events to determine the potential for and severity of water hammer. It should be noted that while condensation oscillations were observed in all ROSA and OSU tests, there was no damage to either despite repeated testing.

b) The ACRS stated that "RELAP5/MOD3 predicts strong oscillations both when they are observed in tests and when they are not. Consequently, the calculated behavior can neither be attributed conclusively to numerical nor physical effects. The mechanisms by which the various observed modes of oscillation are initiated and maintained need to be understood so that their potential influence on the thermal hydraulic behavior of the AP600 can be evaluated."

We agree that the mechanisms underlying the oscillations observed in both the tests and the RELAP5 analyses need to be understood. Our thermal hydraulic consultants, in concert with staff and contractor personnel, will consider the oscillations observed in the tests and in the RELAP5 calculations. The preliminary analysis by NRC staff of the ROSA AP-CL-03 test was presented to the ACRS and is indicative of the detailed effort that will be made in this area. Additionally, the impact of these oscillations upon the thermal hydraulic behavior of the AP600 will be investigated.

5. Comments on T/H Testing at Purdue University

ACRS comment:

We recommend that the noncondensible gas concentration be measured directly at several locations.

Staff response:

PUMA instrumentation does provide noncondensible gas concentration (measuring oxygen concentration, the total noncondensible gas concentration is derived from the air/oxygen mass ratio) at 12 locations.

ACRS comment:

The test matrix does not include a long-duration test. We believe it should because the SBWR containment performance requirement is 72 hours, which scales to 144 hours of PUMA test time.

Staff response:

Long-duration tests are currently included in the PUMA test matrix. A duration of 72 hours of SBWR time is equivalent to 36 hours, not 144 hours, of PUMA test time. However, the duration of these tests will depend on how long it takes for the PUMA facility to reach a quasi-steady state with decreasing containment pressure and temperature.

ACRS comment:

Since the interface temperature of the suppression pool is directly coupled to the containment pressure, an evaluation of thermal stratification in the pool is needed.

Staff response:

PUMA has several thermocouples to measure thermal stratification in the suppression pool. There are also electric heaters in PUMA that can be turned on to raise the surface temperature of the pool for separate-effects tests on thermal stratification, if needed.

ACRS comment:

Some tests should be conducted with initial nitrogen concentrations in the drywell to evaluate the impact to steam line breaks outside containment.

Staff response:

Some tests will be conducted in PUMA with initial nitrogen concentrations in the drywell.

ACRS comment:

The planning of the PUMA experiments should include consideration of phenomena arising as a consequence of failures of active mitigating systems.

Staff response:

We agree and this is included.

ACRS comment:

Data Analysis and evaluation are not part of the contract with Purdue University. This is unfortunate because in this case the principal investigators at Purdue University are highly qualified for such a task. Further, those conducting the testing can bring valuable insights to the process. We recommend that the contract with Purdue University be modified to include a data analysis and evaluation task.

Staff response:

We agree. The Purdue contract is being modified to include a data analysis and evaluation task.

6. Technical Oversight and Integration Efforts

The ACRS indicated that the Advanced Light Water Reactor Thermal Hydraulic Research Integration Group is not the technical oversight recommended by them in the past and subsequently approved by the Commission. The ACRS indicated that a small (5 or 6 members) cohesive group with well-qualified leadership is needed to integrate the technical issue of scaling, data collection, data analysis, and code development.

As discussed with the Full Committee on October 6, 1994, expert thermal hydraulic consultants are assisting the program in the review of the experimental data, code development and assessment and scaling. Some of these consultants are under contract with INEL and the staff's existing thermal hydraulic consultants are also being utilized. The list of consultants and their primary areas of assistance is given below.

Data and Analysis Expert Consultants:

P. Griffith (MIT), N. Todreas (MIT), G. Wallis (Dartmouth), B. Jones (U. Illinois), G. Lellouche (Independent)

Scaling Methodology Expert Consultants:

M. Ishii (Purdue), J. Reyes (OSU), G. Kojasoy (U. Wisconsin), S. Banerjee (UCSB), M. Podowski (RPI)

Code Development and Assessment Expert Consultants:

V. Ransom (Purdue), J. Mahaffy (Penn State), Y. Hassan (Texas A&M)

NRC staff and INEL personnel will participate in the above activities as needed. Coordination and integration of all activities and decisions regarding program objectives are the responsibility of the staff.

The plan to use additional expert consultants was discussed with the ACRS on October 6, 1994. Given the large amount of information that must be reviewed and the range of expertise required, we do not think that the ACRS recommendation to utilize only a small group is practical. We believe that

with proper organization of the experts and management oversight by INEL and the staff, the review of the experimental data, code development and scaling issues will be done in a thorough and timely fashion and will benefit from input from a range of well known experts in the thermal hydraulic field. This is consistent with the way we have intended to use the expert consultants. We envision the thermal hydraulic expert consultants will meet approximately once per month over the next 3-4 months (first meeting was held on November 29-December 2, 1994) and then once every two months over the next 1-2 years. One of the ACRS thermal hydraulic consultants attended the first meeting.

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

May 17, 1995

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

Dear Mr. Taylor:

SUBJECT: REVIEW OF BEST-ESTIMATE MODELS FOR EVALUATION OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE

During the 421st meeting of the Advisory Committee on Reactor Safeguards, May 4-6, 1995, we discussed the methodology being applied by NRR for reviewing the acceptability of best-estimate calculations of emergency core cooling system (ECCS) performance in accordance with the revisions made to 10 CFR 50.46 (ECCS Rule). Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on May 2, 1995, to discuss this matter. During these meetings, we had the benefit of discussions with representatives of NRR and the Westinghouse Electric Corporation. We also had the benefit of the documents referenced.

A historical impediment to the use of best-estimate predictions of plant behavior following a large-break LOCA was the lack of a method for determining the accuracy of the predicted peak cladding temperature. In a September 16, 1986 report, the ACRS made the following comment:

> "The acceptability of realistic evaluation models rests on the development of a satisfactory methodology for determination of the code overall uncertainty. . . . We recommend that the methodology used to evaluate uncertainty be subjected to peer review."

This was done and the ACRS reviewed and endorsed the resulting Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology. It is our view that the CSAU methodology provides a well-structured, traceable, and practical technical basis for quantifying best-estimate code uncertainty. It was the development and demonstration of the CSAU methodology that allowed the successful promulgation of the revision to the ECCS Rule.

Westinghouse Electric Corporation is presenting an alternative approach to the CSAU methodology for determining the uncertainty in

its best-estimate computer code predictions for both existing plants and the AP600 passive plant design. This best-estimate code is intended to meet the requirement of the ECCS Rule that to a "high level of probability," the ECCS criteria will not be exceeded. Although the ECCS Rule allows alternative approaches, none has been reviewed to date nor have review criteria been developed. If Westinghouse persists in following its present path, it is unclear if the intent of 10 CFR 50.46 will be met. Based on the staff presentations, it appears that adoption of the alternative approach would require a weakening of the acceptance criterion for evaluating uncertainty. We believe the staff should be able to confirm that the Westinghouse uncertainty evaluation conforms to the applicable requirements of 10 CFR 50.46 Paragraph (a)(1)(i) in terms of both high probability and high confidence.

During our meeting, we learned that at least two more applicants are requesting approval of best-estimate computer codes. We do not know how they plan to address the nonexceedance requirement of 10 CFR 50.46. A clear statement is needed from the staff as to what constitutes an acceptable demonstration that the ECCS nonexceedance criterion has been met. We would like to see such a statement before the staff begins its review of these other best-estimate codes.

Several aspects of the current review process that were discussed during our meeting should be noted. The review of the Westinghouse best-estimate code has been under way since 1992. We were told that during this period, there has been no formal documentation of this review. Key elements of the alternative approach proposed by Westinghouse for uncertainty have not been addressed. The material submitted by Westinghouse in support of its best-estimate code application is confusing and difficult to follow.

The staff waits for Westinghouse to present its arguments and then reacts as best it can, using some of the provisions of Regulatory Guide 1.157 to guide the review. This reactive approach is a risky procedure for both Westinghouse and the staff. Furthermore, it is much more resource intensive to both because of the iterative nature of "wait-and-see," followed by rounds of questions and answers. This process is time consuming, unstructured, and difficult to trace.

We recommend prompt attention to these matters.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. 10 CFR 50.46(a), as amended through August 31, 1992, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors

2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance" May 1989

3. Westinghouse Electric Corporation Report Addressing Compliance of the Westinghouse Best-Estimate LBLOCA Code and Methodology described in WCAP-12945-P with NRC Regulatory Positions Described in Regulatory Guide 1.157 (**Westinghouse Proprietary**), transmitted by telecopy from Westinghouse Electric Corporation dated March 31, 1995

4. Table 2.1.2-1, Comparison of Regulatory Guide 1.157 Requirements and Westinghouse's Best-Estimate Large-Break LOCA Model (Draft), transmitted by telecopy from INEL dated March 23, 1995

5. Westinghouse Response to Requests for Additional Information on WCAP-12945-P, Volume 5, COBRA/TRAC Code Qualification Document, transmitted by telecopy from INEL dated April 12, 1995, [Westinghouse Proprietary]

6. U.S. Nuclear Regulatory Commission Report, "Quantifying Reactor Safety Margins - Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Los-of-Coolant Accident," NUREG/CR-5249, December 1989

7. Letter dated April 24, 1995, from L. W. Ward, INEL, to F. Orr, Office of Nuclear Reactor Regulation, NRC, transmitting Draft Westinghouse Report, "Review and Evaluation, Westinghouse Code Qualification for Best Estimate LOCA Analysis," dated April 24, 1995

8. ACRS Report dated September 16, 1986, from D. A. Ward, Chairman, ACRS, to L. Zech, Jr., Chairman, NRC, Subject: ACRS Comments on the Proposed Revision to the ECCS Rule in 10 CFR 50.46, "Acceptance Criteria for ECCS for Light Water Nuclear Power Reactors," and Appendix K, "ECCS Evaluation Models"

9. ACRS Report dated July 20, 1988, from W. Kerr, Chairman, ACRS, to V. Stello, Jr., Executive Director for Operations, NRC, Subject: Comments on the Staff's Draft Safety Evaluation of the Westinghouse Topical Report, WCAP-10924, "Westinghouse Large-Break LOCA Best-Estimate Methodology"

ITEM(3): STATUS OF WESTINGHOUSE AP600 DESIGN REVIEW

Westinghouse briefed the Committee on the Westinghouse AP600 passive nuclear power plant design on June 6, 1991. Westinghouse issued the AP600 Standard Safety Analysis Report (SSAR) and the AP600 Probabilistic Risk Assessment (PRA) on June 26, 1992. Westinghouse revised the SSAR on January 13, 1994, and the PRA on January 31, 1995. Based on the SSAR revision and the Westinghouse responses to requests for additional information, the staff issued the Draft Safety Evaluation Report (DSER) on November 30, 1994. Items in the DSER were not sufficiently developed to support a Committee review of the technical issues.

On January 11, 1995, the Westinghouse Standard Plant Designs Subcommittee heard a briefing on the status of the AP600 design review, from representatives of the staff and Westinghouse. On May 31, 1995, the Subcommittee discussed a proposed Commission paper on design and policy issues, and status of the AP600 design review with representatives of the NRC staff and Westinghouse.

Since September 1994, the Thermal Hydraulic Phenomena Subcommittee has held six days of meetings to review the AP600 test program.

- Areas of Interest to the Commissioners

ITAAC: Westinghouse plans to submit a sample set of pilot Inspections, Tests, Analyses, and Acceptance Criteria (ITAACs) in early June, discuss NRC review comments on the pilot ITAACs, and issue proposed ITAACs in August or September. The extent to which these ITAACs are different from the evolutionary plant ITAACs will be reviewed after Westinghouse completes its submittal.

ASME Piping Code: Westinghouse has committed to use the NRC-endorsed 1989 Addenda of the ASME Code for piping support design. The Committee plans to consider the 1994 Addenda on piping support after the staff has completed its review in late 1995.

The Committee plans to consider the above matters in the future.

Attachment:

- Proposed Schedule for ACRS Review of the AP600 Design (Note: ACRS is waiting for the staff and Westinghouse to suggest a schedule for ACRS reviews.) (pp. 30-31)

May 15, 1995

PROPOSED SCHEDULE FOR ACRS REVIEW OF THE AP600 DESIGN

The following schedule is for planning purposes only. The quality and timely submittal of the Draft Safety Evaluation Report (DSER), the Final Safety Evaluation Report (FSER), and the test analysis reports will affect the final review schedule.

8/25-26/94 Thermal Hydraulic Phenomena: AP600 and SWBR Confirmatory Test Program

DSER ISSUED ON 11/30/94

1/11/95 Westinghouse Standard Plant Designs:
Overview [5.0 hrs]
Chap. 1: Introduction and General Description of the Plant [1.5]

1/30/95 Westinghouse submitted Rev 2 to PRA

2/15+16/95 Thermal Hydraulic Phenomena: Review COBRA/TRAC codes for W AP600

3/27+28/95 Thermal Hydraulic Phenomena: Review NRC research program to modify the RELAP5/MOD3 code for W AP600

3/29+30/95 Thermal Hydraulic Phenomena: Review test and analysis programs for AP600 Passive Containment Cooling System

4/05/95 Senior Management Meeting between Westinghouse and NRC staffs

?/??/95 Thermal Hydraulic Phenomena: Staff review of W Code Qualification Documentation for the WCOBRA/TRAC code

?/??/95 Thermal Hydraulic Phenomena: two or more subcommittee meetings to review analysis of testing program data; to be scheduled when analysis reports become available

5/31/95 Westinghouse Standard Plant Designs: Review staff commission paper on status of ten design review issues

6/95 Staff issues a consolidated AP600 issues report to the Commission

9/8/95 Westinghouse Standard Plants Designs:
Chap. 7: Instrumentation and Controls
Chap. 19: Severe Accidents/Probabilistic Risk Assessment

10/31/95 STAFF - ISSUE DSER SUPPLEMENT; Testing and analysis only

12/06/95 Westinghouse Standard Plant Designs:
Chap. 2: Site Characteristics [0.5]
Chap. 3: Design of Structures, Comp., Equip., and Systems [2.0]
Chap. 4: Reactor [2.0]
Chap. 5: Reactor Coolant System and Connected Systems [2.0]
Chap. 11: Radioactive Waste Management [1.0]

May 15, 1995

3/06/96	Westinghouse Standard Plant Designs: Chap. 6: Engineered Safety Features [3.0] Chap. 7: Instrumentation and Controls [2.0] Chap. 8: Electrical Power [1.0] Chap. 9: Auxiliary Systems [2.0] Chap. 10: Steam and Power Conversion System [1.5]
5/07/96	Westinghouse Standard Plant Designs: Chap. 12: Radiation Protection [1.5] Chap. 13: Conduct of Operations [1.5] Chap. 16: Technical Specifications [1.0] Chap. 17: Quality Assurance [1.5] Chap. 18: Human Factors Engineering [2.0]
5/08/96	Chap. 14: Initial Test Program [1.0] Chap. 15: Accident Analyses [3.0] Probabilistic Risk Assessment [3.0]
5/15/96	**STAFF - ISSUE FSER**

The following meetings will be held as needed to compete the design review.

6/06/96	Westinghouse Standard Plant Designs: Review FSER Chapters 1-5
6/07/96	Westinghouse Standard Plant Designs: Review FSER Chapters 6-10
7/10/96	Westinghouse Standard Plant Designs: Review FSER Chapters 11-15
7/11/96	Westinghouse Standard Plant Designs: Review FSER Chapters 16-18 and Probabilistic Risk Assessment
7/12/96	Full Committee: Review FSER and write letter
8/09/96	Full Committee: Issue letter
8/96	Commission approval
9/96	Staff issues Final Design Approval and Federal Register Notice
10/96	Westinghouse submits Design Certification Document
9/97	Staff issues Design Certification

ITEM(4): REGULATORY ANALYSIS GUIDELINES

Over the past few years, the ACRS has reviewed the staff's continuing effort to develop the revised Regulatory Analysis Guidelines (RAGs). These RAGS will be used in the evaluation of proposed actions by the NRC that may be needed to protect the public health and safety. These evaluations are intended to aid the staff and the Commission to determine the need for and to provide adequate justification for the proposed actions, and to provide a clear and well-documented explanation of why a particular action was recommended.

The Committee has provided comments on the proposed RAGs in two reports, one dated November 12, 1992 and the other dated September 14, 1994. In the 1994 report, the Committee indicated that the staff had satisfactorily addressed most of the concerns expressed in the 1992 report, specifically, the staff's action to address these concerns included the following:

- Guidance for addressing safety goal considerations in the RAGs is consistent with SECY-91-270 and the Commission's SRM dated June 15, 1990.
- Analyses and information necessary to satisfy the backfit rule and/or CRGR review are in conformance with the RAGs.
- Values and impacts are redefined to be more consistent with OMB guidance and the net present-value approach.
- Averted on-site costs are specifically called out as value attributes to be assessed when appropriate.
- The treatment of licensees' voluntary actions in regulatory analyses is explicitly defined, i.e., no credit given.
- Present-worth calculations are now based on OMB discount rate guidance.

However, in the 1994 report, the Committee stated that the issuance of the new RAGs should be delayed until the following issues are reconsidered:

- New guidance should be developed on the appropriate monetary value to apply to adverse health and land contamination effects to replace the undiscounted $1000/man-rem value, currently in use.
- The definition of containment failure should be consistent with the definition used in NUREG-1150, Severe Accident Risks: An Assessment For Five U.S. Nuclear Power Plants.

Attachments:

- Letter dated November 12, 1992, from P. Shewmon, Chairman ACRS, to J. Taylor, Executive Director for Operations, Subject: Revised Regulatory Analysis Guidelines (pp.34-36)

- Letter dated September 14, 1994, from T. S. Kress, Chairman ACRS, to I. Selin, Chairman NRC, Subject: Revised Regulatory Analysis Guidelines (pp.37-38)

- Letter dated February 7, 1995, from J. Taylor, Executive Director for Operations, to T. S. Kress, Chairman ACRS, Subject: Revised Regulatory Analysis Guidelines (pp.39-45)

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

November 12, 1992

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

Dear Mr. Taylor:

SUBJECT: REVISED REGULATORY ANALYSIS GUIDELINES

During the 391st meeting of the Advisory Committee on Reactor Safeguards, November 5-7, 1992, we reviewed a draft of NUREG/BR-0058, Revision 2, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission." Our Subcommittee on Safety Philosophy, Technology, and Criteria considered this matter during a meeting on October 28, 1992. During these meetings, we had the benefit of discussions with representatives of the NRC staff, and of the document referenced.

This brochure will be NRC's policy-setting document with respect to regulatory analyses. As such, it deals with a number of very important issues that bear directly on the overall NRC regulatory philosophy and approach. Some of the positions taken in the proposed guidelines represent departures from current practice, have never been formalized before, or differ from the industry and the Office of Management and Budget (OMB) positions.

We believe this to be such an important document that even a draft version to be issued for public comment should reflect high levels of intellectual and technical content, coherence, and clarity of thought and presentation. Although the draft document does have much to commend it, we believe the subject deserves better. We recommend that substantial additional effort be put into rethinking and redeveloping some of the regulatory positions and into developing a "showcase" document with respect to content, style, and quality of prose. We do not see any urgent need for, and recommend against, issuing the draft document at this time. We expect to review the revised document before it is issued for public comment.

In its presentations to us, the staff identified some specific issues for particular attention. Although we agree with some of the positions taken on these in the document, we have fundamental differences with several of them. We provide you with our comments below.

Safety Goal Implementation

This document suffers from the absence of a clear statement of the means by which the Commission's overall regulatory philosophy will be implemented through the concepts of adequate protection, safety goals, the backfit rule, ALARA principles, etc. Whether here or elsewhere, such a statement is urgently needed.

The safety goal decision chart only deals with issues that result in changing the core-damage frequency. We believe it should also consider issues that could change the conditional containment failure probability.

Quantification of Benefits

Figure 3.1 of the proposed guidelines should include a step in which a determination is made on whether the proposed enhancement is something that can be evaluated by quantitative risk estimates. If so, we believe that PRAs must be used to quantify the benefits. If not, the analysis would go to a different decisionmaking scheme (e.g., expert opinion, engineering/regulatory judgment).

Treatment of Voluntary Actions

We agree with the position taken on voluntary actions in the proposed guidelines. However, we are concerned that this will tend to discourage voluntary actions. Some means, outside the regulatory analysis process, should be sought to promote and encourage such actions.

Discount Rate

While the OMB directive of 1981 (which has never been rescinded) applied specifically to executive agencies, NRC ought to have good reasons for ignoring it. The fact that others do so is not a good reason. We were told that efforts had not been made to better understand OMB's rationale. We recommend that this be done.

Simultaneously Satisfying the Requirements of the Backfit Rule and/or the Committee to Review Generic Requirements

We agree that regulatory analyses should be made in such a manner that they also meet these other needs.

Treatment of Averted Onsite Costs

The staff intends to treat averted onsite costs (AOSC) as an offset to the costs incurred by the utilities in implementing the associated requirement. We believe AOSC should be included in the benefits column and not the costs column. We are concerned, however, that the methods and assumptions used for computing AOSC

are highly uncertain and can dominate the final answer. Accordingly, we recommend that further effort be given to establishing definitive guidance for AOSC evaluations.

In the draft document, the staff recommends that the results be presented in terms of net value (value minus impact) rather than as a ratio (value/impact). This should not be an issue because these are entirely different measures and both should be part of the decision process.

Discounting of Health and Safety Effects

We are unconvinced by the arguments presented for the staff's position that health and safety effects not be discounted in the value/impact analyses. Appropriate balancing of costs and benefits require discounting of each.

Monetary Value of a Person-Rem Averted

There is, in principle, no problem with the staff's proposed interim position, "continuing to use the value of $1000/person-rem until a final recommendation can be made after further review and analysis," except that such a position has existed for about 15 years, and can persist indefinitely. We recommend that an appropriate treatment of the monetary values to be associated with onsite and offsite health effects (both early and latent) and land contamination be developed promptly.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

Reference:
Letter dated September 11, 1992, from C. J. Heltemes, Jr., Office of Nuclear Regulatory Research, to Raymond F. Fraley, Advisory Committee on Reactor Safeguards, transmitting:

(a) Draft SECY paper (undated) for the Commissioners from James M. Taylor, Executive Director for Operations, NRC, Subject: Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission (Predecisional)

(b) Draft NUREG/BR-0058, Revision 2 (undated), "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" (Predecisional)

(c) Separate Enclosures (undated) on Averted Onsite Costs and Discounting of Health and Safety (Predecisional)

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

September 14, 1994

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

Dear Chairman Selin:

SUBJECT: REVISED REGULATORY ANALYSIS GUIDELINES

During the 413th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 1994, we discussed the proposed final "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission." During this meeting, we had the benefit of discussions with representatives of the NRC staff. We note that the industry did not have the opportunity to review the staff response to public comments. We had provided comments on a preliminary version of these Guidelines to the Executive Director for Operations in a letter dated November 12, 1992. We also had the benefit of the documents referenced.

In our November 12, 1992 letter, we made a number of substantive comments on areas in which we disagreed with the staff proposals. In the revised version, the staff has satisfactorily addressed most of our earlier concerns. In addition, we believe the staff response to the public comments has been balanced and appropriate.

We believe these Guidelines will be valuable to the NRC staff in its various decision-making functions. At this time, we still have concerns in two areas:

1. Until new guidance has been developed on the appropriate monetary values to apply to adverse health and land contamination effects, the staff proposes the continued use of an undiscounted $1000/man-rem. as a surrogate for the actual discounted values.

   We do not support this proposal. The correct treatment requires separate, realistic values for each effect and these should be discounted for present-worth evaluation. The Guidelines should not be issued until a technically correct approach with the appropriate values is developed.

2. The revised Guidelines now propose a definition for containment failure that is "... consistent with the performance goal used in the review of evolutionary ALWRs and documented in SECY-93-087." This is a change from the definition used in a prior version of the Guidelines which was taken from NUREG-1150.

   The definition in NUREG-1150, which addresses the risk dominant sequences, is the appropriate one for use in these Guidelines.

The issuance of the new Regulatory Analysis Guidelines should be delayed until these issues are reconsidered.

Sincerely,

T. S. Kress
Chairman

References:

1. Letter dated June 29, 1994, from C. J. Heltemes, Jr., NRC Office of Nuclear Regulatory Research, to T. S. Kress, ACRS Chairman, transmitting draft SECY Paper: Regulatory Analysis Guidelines of the U.S. NRC (Draft Predecisional)
2. Letter dated November 12, 1992, from Paul Shewmon, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Revised Regulatory Analysis Guidelines

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

February 7, 1995

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

Dear Chairman Kress:

SUBJECT: REVISED REGULATORY ANALYSIS GUIDELINES

This letter is in response to your September 14, 1994, letter to Chairman Selin on the referenced subject. In your letter, the Advisory Committee on Reactor Safeguards (ACRS) identified two areas of concern with respect to the Guidelines and called for a delay in its issuance until the two issues are reconsidered.

With respect to the first ACRS concern, the staff fully agrees that new guidance is needed on the appropriate monetary values to apply to health and land contamination effects and that these attributes should be subject to present worth considerations. Development of this guidance is on-going, and the staff currently plans to appear before the ACRS on this matter before or by mid-1995. However, the staff sees no practical value in delaying the issuance of the new Guidelines until this new guidance is in place because such a delay will have no effect on the staff's treatment on these issues during the interim period. If the new Guidelines are delayed, NRC analyses would continue to be subject to the guidance in the existing Guidelines (Rev. 1, 1984) whose monetary conversion factor and present worth treatment are fully consistent with the interim guidance proposed in the new Guidelines. Furthermore, by adopting ACRS's recommendation during this interim period, the NRC would forego the many other areas of improved guidance provided by the new Guidelines. Therefore, the staff believes that the publication of the Guidelines should go forward. The staff plans to publish the revised dollar value of person-rem averted in a revision to the Regulatory Analysis Technical Evaluation Handbook, NUREG/BR-0184, once the revised value is reviewed by ACRS and approved by the Commission.

The second area of concern relates to the definition of containment failure. In the version of the Guidelines presented to the ACRS, the definition used was one that is consistent with the performance goal used in the review of evolutionary ALWRs and documented in SECY-93-087. This was a change from the definition used in the version that was published in September 1993 for public comment. In that document, the definition of containment failure was taken from NUREG-1150. The ACRS has expressed a view that the definition in NUREG-1150, which addresses the risk dominant sequences, is the appropriate one for use in the Guidelines. The staff agrees with ACRS that the NUREG-1150 definition of containment failure would provide an acceptable basis for the safety goal portion of regulatory analyses. However, the staff recognizes the need for flexibility and has modified the Guidelines to allow an analyst to make the case that if the screening criteria do not apply for particular

issues where the uncertainties or risk contributions may not be insignificant, the decision to pursue the issue should be subject to further management decision. Please find attached for your information the modified pages of both the Commission paper and Section 3 in the Guidelines.

Based on the above discussion, the staff believes there is no need to further delay the publication of the Guidelines.

Sincerely,

James M. Taylor
Executive Director
for Operations

Attachments: As stated

cc: The Chairman
Commissioner Rogers
Commissioner de Planque
SECY
OGC
OPM

R. Other Comments Related to Safety Goal Evaluation.

Comment: NUBARG requested that Section 3 of the Guidelines be supplemented with a discussion of the cumulative summation of risk reduction. NUBARG argued that the regulatory analysis should focus not just on the relative change in safety, but on an absolute evaluation of safety based on the residual risk in view of the safety enhancements already implemented. NUBARG further suggested that, because probabilistic risk assessments have become more refined, it should now be possible to determine the "baseline risk" by maintaining a cumulative summation of the core damage frequency reduction from backfits over time (i.e., a "master" risk assessment).

An NRC employee argued that the NUREG-1150 definition of early containment failure is rather arbitrary and should not be used as part of the regulatory analysis process without further justification. In the comment, the employee noted that SECY-93-138 (Recommendation on Large Release Definition) suggests that a possible definition would be a release within the first 24 hours following the onset of core damage. The commenter further suggested that whatever definition is used, it should involve the potential for serious offsite health consequences. The commenter believes that a release occurring 2 hours after reactor vessel breach could have serious offsite health consequences, including the possibility of early fatalities. As a result, the commenter did not feel that the NUREG-1150 definition of early containment failure is appropriate.

Response: The draft Guidelines provided no guidance on how to treat the cumulative summation of risk reduction. However, the draft Handbook recognizes and discusses this issue in Section A.2 of Appendix A. The Handbook states that the analyst should be aware of previous or ongoing safety improvements which have already impacted or have the potential to impact the status quo for the issue being addressed. Without a formal process for incorporating this issue into the regulatory analysis framework, the Handbook states that the analyst is to use a "best effort" approach in accounting for preexisting or concurrent regulatory actions. The final Guidelines have been supplemented to include a brief discussion of the risk reduction issues based on the discussion in Section A.2 of the draft Handbook.

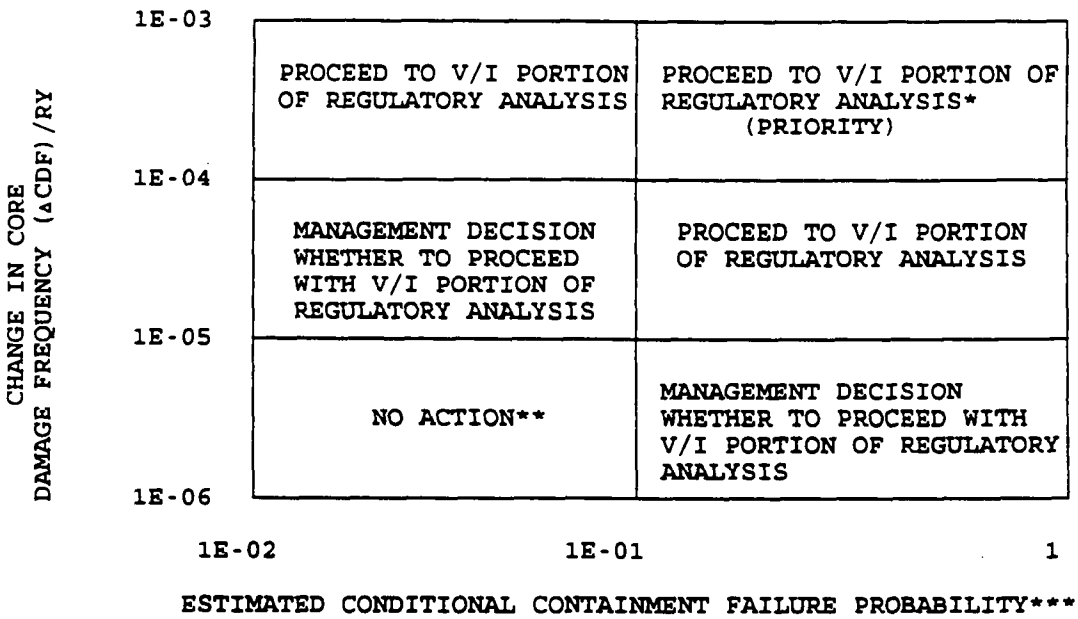
The draft Guidelines stated that the NUREG-1150 definition of early containment failure should be used for evaluating safety goal considerations (footnote in Section 3.3.2). The NUREG-1150 definition states that early containment failures are "those containment failures occurring before or within a few minutes of reactor vessel breach for pressurized water reactors and those failures occurring before or within 2 hours of vessel breach for boiling water reactors." The definition recognizes the impacts of early failure and uses that as a baseline from which to assess containment performance, e.g., CPCFB changes. In applying these screening criteria, the CPCFB definition may be extended, if appropriate, to up to four hours after vessel breach, to permit initiation of accident management and emergency preparedness actions. The NUREG-1150 definition of early containment failure implicitly includes consideration of the potential for offsite health effects.

Furthermore, many scenarios of early containment failure were analyzed in the Containment Performance Improvement (CPI) Program, which studied the results of a variety of challenges to containments. These analyses did not find any generic improvements for containments that could be supported on a cost-beneficial basis. (Cf. Generic Letter No. 88-20, Supplement No. 3, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities," July 6, 1990.) The CPI results and their use are described more fully in the Handbook. However, the staff acknowledges that in certain instances, even though they might be rare, the screening criteria may not adequately address certain accident scenarios of unique safety or risk interest. An example is one in which certain challenges could lead to containment failure after the time period adopted in the safety goal screening criteria, yet early enough that the contribution of these challenges to total risk would be non-negligible, particularly if the failure occurs prior to effective implementation of accident management measures. The Guidelines have been revised such that in these circumstances, the analyst should make the case that the screening criteria do not apply and the decision to pursue the issue should be subject to further management decision.

GUIDELINES

Estimated Reduction In CDF	Staff Action
> 10^{-4}/reactor year	• Proceed with the regulatory analysis on a high priority basis.
10^{-4}-10^{-5}/reactor year	• The decision whether to proceed with the regulatory analysis is to be made by the responsible Division Director (see Figure 3.2).
< 10^{-5}/reactor year	• Terminate further analysis unless the Office Director directs otherwise based upon strong engineering or qualitative justification (see Figure 3.2).

FIGURE 3.2 SAFETY GOAL SCREENING CRITERIA



* A determination is needed regarding adequate protection or compliance; as a result, a value/impact analysis may not be appropriate.

** Unless Office Director decides that the screening criteria do not apply (see Section 3.3.2).

***Conditional upon core damage accident that releases radionuclides into the containment (see Section 3.3.2).

3.3.2 Additional Consideration of Containment Performance

The previous section focuses on accident prevention, that is, on issues intended to reduce core damage frequency. To achieve a measure of balance between prevention and mitigation, the safety goal screening criteria established for safety goal evaluations include a mechanism for having greater consideration of issues, and associated accident sequences, with relatively poor containment performance. The measure of containment performance to be used in safety goal evaluations is the conditional probability of early containment failure or bypass (CPCFB).[1] The safety goal screening criteria shown in Figure 3.2 are subdivided to require greater staff emphasis on the higher valued (i.e., >0.1) CPCFBs. A CPCFB value of 0.1 is consistent with Commission guidance on containment performance for evolutionary designs. In effect, the use of the CPCFB reduces the priority of or eliminates the additional study of issues with associated CPCFBs of less than 0.1.

The safety goal screening criteria provided here are based upon the recognition that the severe accident risk to the individual is dominated by the overall frequency of the following kinds of scenarios:

1. Those involving core damage and release into an intact containment with early containment failure occurring.
2. Those involving core damage and for which the containment system is breached as a result of accident phenomena either prior to or early in the core damage or melt progression.
3. Those involving pre-existing conditions that cause loss of containment integrity prior to core damage (e.g., large openings).
4. Those for which containment is bypassed entirely and which have high probability of causing core damage to occur (e.g., intersystem loss-of-coolant accident).

It is acknowledged that in certain instances, the screening criteria may not adequately address certain accident scenarios of unique safety or risk interest. An example is one in which certain challenges could lead to containment failure after the time period adopted in the safety goal screening criteria, yet early enough that the contribution of these challenges to total risk would be non-negligible, particularly if the failure occurs prior to

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[1]CPCFB in this context is the conditional probability of early containment failure or bypass given a core melt. In NUREG-1150, early containment failure is defined as "Those containment failures occurring before or within a few minutes of reactor vessel breach for PWRs and those failures occurring before or within 2 hours of vessel breach for BWRs. Containment bypass failures (e.g., interfacing-system loss-of-coolant accidents) are categorized separately from early failures." The definition recognizes the impacts of early failure and uses that as a baseline from which to assess containment performance, e.g., CPCFB changes. In applying these screening criteria, the CPCFB definition may be extended, if appropriate, to up to four hours after vessel breach, to permit initiation of accident management and emergency preparedness actions. It is not a goal being sought since the staff recognizes the benefits of prolonging containment failure where such scenarios risk early failure.

effective implementation of accident management measures. In these circumstances, the analyst should make the case that the screening criteria do not apply and the decision to pursue the issue should be subject to further management decision.

Furthermore, it should be noted that the safety goal screening criteria described here do not address issues that deal only with containment performance. Consequently, issues which have no impact on core damage frequency (delta CDF of zero) cannot be addressed with the safety goal screening criteria. However, because mitigative initiatives have been relatively few and infrequent compared with accident preventive initiatives, mitigative initiatives will be assessed on a case-by-case basis with regard to the safety goals. Given the very few proposed regulatory initiatives that involve mitigation, this should have little overall impact from a practical perspective on the usefulness of the safety goal screening criteria.

3.3.3 Summary of Safety Goal Screening Criteria Guidance

The safety goal screening criteria discussed in Chapter 3 are summarized in Figure 3.2, which graphically illustrates the criteria and provides guidance as to when staff should proceed to the estimation and evaluation of the values and impacts portion of the regulatory analysis and when a management decision is needed.

Management with responsibility for preparation of a safety goal evaluation should review the results of the evaluation and the overall uncertainty and sensitivity of associated estimates. A judgment should be made whether substantial additional protection would be achievable and whether continuation of the regulatory analysis process is therefore warranted.

3.3.4 Regulatory Analysis

If the safety goal evaluation of the proposed regulatory action results in a favorable determination (i.e., any decision except "no action"), the analyst may presume that the substantial additional protection standard of 10 CFR 50.109(a)(3) is achievable. The initiative should then be assessed in accordance with Section 4.3 of these Guidelines (see Figure 3.1). If the net value calculation required by Section 4.4 is not positive, further activities and analyses should be terminated unless there is a qualitative justification for proceeding further.

ITEM(5): APPLICATION OF RISK ANALYSIS IN RULEMAKING

The ACRS has a long history of advocating the appropriate use of risk-based regulation and rulemaking. The last Committee report on this subject was issued on November 16, 1992. This report refers to earlier ACRS comments regarding risk implications associated with the Maintenance Rule, Regulation Marginal to Safety, and with other ACRS reports regarding the staff's use of PRA. Examples of these are also attached.

In a report dated March 17, 1995, on proposed rulemaking for decommissioning of nuclear power plants, the ACRS expressed a concern that the proposed rule has not been founded on a risk basis. It further stated that realistic risk analyses for decommissioning had not been done, and expressed a desire that steps be taken to establish a risk basis for reformulating 10 CFR Parts 2, 50, and 51.

The Committee plans to continue its review of issues related to risk-based regulation during future meetings.

Attachments:

- Letter dated April 11, 1989, from Forrest J. Remick, Chairman ACRS, to L. W. Zeck, Jr, Chairman NRC, Subject: Proposed Final Rulemaking Related to Maintenance of Nuclear Power Plants (pp. 47-48)
- Letter dated July 19, 1991, from David A. Ward, Chairman ACRS, to I. Selin, Chairman NRC, Subject: The Consistent Use of Probabilistic Risk Assessment (pp. 49-52)
- Letter dated August 11, 1992, from David A. Ward, Chairman ACRS, to I. Selin, Chairman NRC, Subject: Elimination of Requirements Marginal to Safety (pp. 53-54)
- Letter dated November 16, 1992, from P. Shewmon, Chairman ACRS, to I. Selin, Chairman NRC, Subject: Risk-Based Regulation (pp. 55-56)
- Letter dated March 17, 1995, from T. S. Kress, Chairman ACRS, to I. Selin, Chairman NRC, Subject: Proposed Rulemaking-Revision to 10 CFR Parts 2, 50, and 51 Related to Decommissioning of Nuclear Power Reactors (pp. 57-58)

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

April 11, 1989

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: PROPOSED FINAL RULEMAKING RELATED TO MAINTENANCE OF NUCLEAR POWER PLANTS

During the 348th meeting of the Advisory Committee on Reactor Safeguards, April 6-8, 1989, we discussed the draft Commission paper related to final rulemaking for maintenance of nuclear power plants, including a draft regulatory guide. Our Subcommittee on Maintenance Practices and Procedures discussed this matter with representatives of the NRC staff and the Nuclear Management and Resources Council during a meeting held on March 30, 1989. We also had the benefit of the document referenced. We previously commented on the proposed rulemaking in a report dated September 13, 1988.

In our September 13, 1988 report we did not endorse the staff's proposal to establish a maintenance rule. After review of the proposed final rule, including the public comments and a related draft regulatory guide, our position remains essentially the same. We still believe that good maintenance is a necessary ingredient in any operational program that seeks to ensure reliable and safe plant operation, but that is not the issue. The issue is how to achieve good maintenance.

We were told by the industry that its aggressive emphasis on the development of effective maintenance programs over the past several years has resulted in a marked improvement in the maintenance programs themselves, and in significant progress toward reaching its objectives. The staff members with whom we conferred agreed that this is the case. Further, we were told that a staff evaluation of a sample of maintenance programs, which included about one quarter of those plants now operating, indicated that only a few percent of the operating plants may have poor maintenance programs. Given an environment in which there is already a scarcity of industry and NRC resources, we believe that it is more cost effective to seek improvements applicable to the few plants with "poor" maintenance programs by means of existing regulations rather than burdening all plants with a costly program of unproven efficacy.

The scope of the proposed final rule is also of concern. The Commission has the responsibility to regulate the operation of nuclear power plants in a way that ensures protection of the public health and safety, but

does not have the responsibility for managing plant operation. The proposed final maintenance rule strains severely and may violate the boundary between regulating and managing. The scope of the proposed final rule and its accompanying regulatory guide is so broad that almost every facet of plant operation would be under the continuing scrutiny of the NRC on the basis of its effect on maintenance. This would be counterproductive.

Because everyone involved believes that maintenance programs are improving, and because the industry is committed to additional improvements, we recommend that the staff continue to monitor the industry's progress and not intervene at this time.

The proposed final rule would introduce a major policy change extending NRC responsibility far beyond identified safety systems. We do not believe such a significant change in policy should occur in the guise of a maintenance rule which deals only with maintenance and provides no guidance on which systems deserve special attention. The ACRS has in the past recommended more emphasis on the performance of some nonsafety systems. For example, the Committee recommended an evaluation of the contributions to risk from failures of nonsafety-grade control systems. More recently, the Committee has recommended a reevaluation of the current set of regulations in the light of additional insights provided by risk-based evaluations of plant performance and the adoption of safety goals. We would endorse a well-conceived reevaluation of current regulations which would undoubtedly suggest that more regulatory emphasis should be placed on some systems that in the past have been treated as balance-of-plant, and less on others. However, this evaluation should be done in an integrated manner which would, on the basis of what has been learned about risk contributions, identify some systems for special attention.

Sincerely,

Forrest J. Remick
Chairman

Reference:
Memorandum dated April 6, 1989 from Bill M. Morris, Office of Nuclear Regulatory Research, for Raymond F. Fraley, ACRS, Subject: Draft Commission Paper for Notice of Final Rulemaking For Maintenance of Nuclear Power Plants," w/enclosures (Predecisional)

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

July 19, 1991

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

Dear Chairman Selin:

SUBJECT: THE CONSISTENT USE OF PROBABILISTIC RISK ASSESSMENT

During the 375th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 1991, and in earlier meetings, we discussed the unevenness and inconsistency in the use of probabilistic risk assessment (PRA) in NRC. PRA can be a valuable tool for judging the quality of regulation, and for helping to ensure the optimal use of regulatory and industry resources, so we would have liked to see a deeper and more deliberate integration of the methodology into the NRC activities. Our recommendations to this end are directed at problems that took time to develop, and are likely to take a long time to solve.

PRA is not a simple subject, so there are wide variations in the sophistication with which it is used by the various elements of NRC. There are only a few staff members expert in some of the unfamiliar disciplines -- especially statistics -- that go into a PRA, so it is not surprising that there are inconsistencies in the application of the methodology to regulatory problems.

To illustrate the problems, let us just list a few of the fundamental aspects of the use of PRA, in which different elements of the staff seem to go their own ways. These are just illustrations, but each can lead to an erroneous regulatory decision.

1. The proper use of significant figures is in principle a trivial matter, but it does provide a measure of a person's understanding of the limitations of an analysis. Yet we often hear from members of the staff who quote core-damage probabilities to three significant figures, and who appear to believe that the numbers are meaningful. It is a rare PRA in which even the first significant figure should be regarded as sufficiently accurate to play an important role in a regulatory decision, but there is something mesmerizing about numbers, which imbues them with misleading verisimilitude.

They deserve respect, but not too much, and it is wrong to err in either direction.

2. Closely related is uncertainty. There is no way to know how seriously to take the results of a PRA without some estimate of the uncertainty, yet we often hear thoroughly unsatisfactory answers (some perhaps invented on the spot) when we ask about uncertainty. One of the advantages of PRA is that it provides a mechanism for estimating uncertainty, uncertainty which is equally present, but not quantified, in deterministic analyses.

3. Conservatism. A PRA should be done realistically. The proper time to add an appropriate measure of conservatism is when its results are used in the regulatory process. If the PRA itself is done with conservative assumptions (more the rule than the exception at NRC), and is then used in a conservative regulatory decision-making process, self-deception can result, or resources can be squandered.

   The inconsistent use of conservatism was illustrated by a pair of briefings at our April 1991 meeting, which included updates on proposed rules on license renewal and on maintenance. In the former case, we were told that a licensee could use PRA to add an item for later review, but never to remove one -- a one-way sieve. In the latter case we were told that PRA could be used to justify either enhancement or relaxation of maintenance requirements. Foolish consistency may be a hobgoblin, as Emerson said, but there is nothing foolish in seeking consistency in regulation.

4. The bottom line. It has been widely recognized since WASH-1400 that the bottom-line probabilities (of either core melt or immediate or delayed fatalities) are among the weakest results of a PRA, subject to the greatest uncertainties. (That doesn't mean they are useless, only that they should be used with caution and sophistication.) Yet we find staff members unaware of these subtleties, often dealing with small problems, justifying their actions in terms of the bottom-line probabilities. This is only in part due to the Backfit Rule, which almost requires such behavior; it is also inexperience and lack of sensitivity to the limitations of the methodology.

   A number of staff actions and proposals use bottom-line results of a PRA as thresholds for decision making, often with the standard litany about the uncertainty in the reliability of these results. In fact, the quantified uncertainty in the bottom-line results of a PRA is just as important a number as the probability itself. It would be straightforward to employ a decision-making algorithm that <u>prescribes</u> a confidence level

for the decision, and uses both the bottom-line probability and the uncertainty to achieve this. A further improvement would be to incorporate the consequences of erroneous decisions, what statisticians would call the loss function, into the decision-making process. The Commission has come close to this approach in its recent instructions to the staff on the diesel generator reliability question.

These are just a few examples of problems with the use of PRA in NRC, all common enough to be disturbing, and increasing in frequency as the use of PRA increases. It has been more than fifteen years since the publication of WASH-1400, a pioneering study which, despite known shortcomings, established the NRC at the forefront of quantitative risk assessment. One could have hoped that by now a coherent policy on the appropriate use of PRA within the agency, on both large and small problems, could have evolved.

We recommend that:

A. A mechanism be found (perhaps a retreat) through which the few PRA and statistical experts now scattered throughout the agency (and generally ignored) can be brought together with the appropriate senior managers and outside experts, to work toward a consistent position on the use of PRA at NRC. It could be worth the time expended. (Among other long-term benefits, such an interaction would add an element of horizontal structure to the NRC's predominantly vertical organization.)

B. The Commission then find a way to give credence and force to that position.

C. The Commission emphasize recruitment of larger numbers of professionals expert in PRA and statistics.

D. The Commission consider some kind of mandate that any letter, order, issue resolution, etc., that contains or depends on a statistical analysis or PRA, be reviewed by one of the expert PRA or statistical groups.

We do not pretend that this is an easy problem. The solution involves not only a cultural shift, so that those few experts already at NRC have some impact, but also substantial enhancement of the staff capabilities. That will require incentives that only the Commission can supply. It is interesting that the Commission's Severe Accident Policy Statement, dated August 1985, stated that "within 18 months of the publication of this severe accident statement, the staff will issue guidance on the form, purpose and role that PRAs are to play in severe accident analysis and decision making for both existing and future plant designs...."

Additional comments by ACRS Members Harold W. Lewis and J. Ernest Wilkins are presented below.

Sincerely,

David A. Ward
Chairman

Additional Comments by ACRS Members Harold W. Lewis and J. Ernest Wilkins

We thoroughly endorse this letter, and regret only that the Committee chose to ignore the parallels between the PRA problems and those in a number of other newer technologies significant to nuclear safety. Recommendation C should have included mention of some of these -- electronics and computers, for example -- which are of increasing importance. Weaknesses in those areas also need correction. Computerized protection and control systems, in particular, require the kind of sophisticated review that NRC is in no position to provide.

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

August 11, 1992

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

Dear Chairman Selin:

SUBJECT: ELIMINATION OF REQUIREMENTS MARGINAL TO SAFETY

During the 388th meeting of the Advisory Committee on Reactor Safeguards, August 6-8, 1992, we reviewed SECY-92-263, "Staff Plans For Elimination of Requirements Marginal to Safety." We had the benefit of discussions with the NRC staff and NUMARC, and of the documents referenced.

The staff is not very far along in this enterprise, and proposes a workshop (or more than one), interactions with the public, the industry, and ACRS, to elicit input. It has also produced a list of candidate regulations for further study. We have no problem with any of this, and both hope and expect to remain involved. We do have a few observations at this stage, as follows, in no special order.

The list of candidate regulations so far seems a bit thin, and doesn't really represent the fruits of the kind of comprehensive review we have often recommended. That may come later.

Even this list, however, can serve to test the methods, and highlight the very difficult questions of implementation which have yet to be considered. Resolution of those issues (to which we turn in a moment) can provide paradigms for later developments.

The presentation we heard described the objective as removal of regulations when this can be done "without adverse safety impact." Since such a determination requires that the regulation be judged as having had no safety significance at all, it is hard to imagine the staff acquiescing in such a judgment. The original term was "marginal to safety," and that has yet to be defined. It will not be easy.

In many cases (if not most), it is not the plain wording of a regulation that causes the problems, but the implementation and interpretation by the staff. In short, it is the body of regulatory practice, not just the regulations, that is at issue.

The staff did not mention that as a problem, so the effort is so far very incomplete.

We have given many examples in recent years of incoherence among regulations and practices, and some important examples of overkill only appear in that context. These are more difficult to find by searching each regulation independently. Yet this program is an important vehicle for the study of coherence in regulation.

In the whole enterprise, it will be necessary to make informed and sophisticated use of PRA to separate the marginal from the essential requirements. Everything we have recently said on that subject should be regarded as repeated here.

Finally, it is worth noting that the health of the organism is not only a function of the health of its parts. The reasons for seeking to remove marginal regulations are not only to reduce wasted effort and burden, but also to improve the signal-to-noise ratio of the regulatory process. Regulations that do no harm in and of themselves nonetheless can do harm to the focus and effectiveness of the enterprise.

Sincerely,

David A. Ward
Chairman

References:
1. SECY-92-263, dated July 24, 1992, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Staff Plans for Elimination of Requirements Marginal to Safety (Draft Predecisional).
2. Memorandum dated July 13, 1992, from Byron Lee, Jr., NUMARC, for NUMARC Board of Directors, Subject: Industrywide Initiative.

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

November 16, 1992

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

Dear Chairman Selin:

SUBJECT: RISK-BASED REGULATION

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

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

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

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

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

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

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

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

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

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

March 17, 1995

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

Dear Chairman Selin:

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

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

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

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

ITEM(6): PROPOSED FINAL RULE ON TECHNICAL SPECIFICATIONS (SECY-95-128)

In May 1993, the Commission directed the staff to publish the draft Policy Statement on Technical Specifications (TS) Improvements as final (without public comment), and to codify in 10 CFR 50.36 the four criteria for the scope of items which must be included in TS. Such a rule change would allow licensees to relocate TS items which are outside of this scope to licensee-controlled documents, with predictable staff review and approval. The Final Policy Statement on TS Improvements was issued in July 1993. The public comment period on the draft rule ended in December 1994.

The ACRS reviewed the proposed Final Policy Statement on TS Improvements in June 1993 and expressed concern that more detailed guidance was needed on the definition of "significant to public health and safety" as it is used in Criterion 4 (the risk-based criterion), and suggested that this guidance appear in the implementing regulatory guide(s).

The staff has not developed a Regulatory Guide for this rule, preferring instead to rely on the Final Policy Statement on TS Improvements, the Standard Technical Specifications, and the Backfit Rule as its implementing guidance.

In April 1995, the ACRS reviewed the proposed final rule change and again reiterated its concern over Criterion 4 and recommended that the rule itself include a more explicit definition of "significant to public health and safety."

The staff has stated that concerns over the implementation of Criterion 4 would be addressed by the following additional implementing guidance:

- NRC's PRA Implementation Plan,
- The Commission's "Proposed Policy Statement on the Use of PRA Methods in Nuclear Regulatory Activities," issued for public comment in December 1994,
- Draft "Regulatory Analysis Guidelines" Revision 2, issued for public comment in August 1993, and
- NUREG/CR-6141 "Handbook for Risk-Based Analyses of Technical Specifications," December 1994.

Attachments:

- Letter dated June 18, 1993, from J. Ernest Wilkins, Jr., to I. Selin, Chairman NRC, Subject: Policy Statement on Technical Specifications Improvements for Nuclear Power Plants (pp. 61-63)

- Letter dated July 22, 1993, from J. Taylor, Executive Director for Operations, to J. Ernest Wilkins, Jr., Chairman ACRS, Subject: Policy Statement on Technical Specifications Improvements (pp. 64)

- Letter dated April 13, 1995, from T. S. Kress, Chairman ACRS, to I. Selin, Chairman NRC, Subject: Proposed Final Rule Change to 10 CFR 50.36, Technical Specifications (pp. 65-66)

- Letter dated May 31, 1995, from J. Taylor, Executive Director for Operations, to T. S. Kress, Chairman ACRS, Subject: Proposed Final Rule Change to 10 CFR 50.36, Technical Specifications (pp. 67-68)

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

June 18, 1993

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

Dear Chairman Selin:

SUBJECT: POLICY STATEMENT ON TECHNICAL SPECIFICATIONS IMPROVEMENTS FOR NUCLEAR POWER PLANTS

During the 398th meeting of the Advisory Committee on Reactor Safeguards, June 10-11, 1993, we reviewed the NRC staff's draft Final Policy Statement on Technical Specifications Improvements for Nuclear Power Plants, as originally presented in SECY-93-067. We also reviewed a revised draft of this Policy Statement which is responsive to the Commission's comments included in the Staff Requirements Memorandum dated May 25, 1993. During our meeting, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

In SECY-93-067, the staff recommended that the draft Final Policy Statement be published for a 90-day public comment period. However, the Commission approved publication of the Policy Statement in final form, subject to the following comments:

1. The Commission directed the staff to prepare a rulemaking package that would codify the four criteria contained in SECY-93-067, delineating those aspects of nuclear power plant design and operation that should be included in Technical Specifications. (We note that the staff has proposed the use of these same criteria for establishing plant systems and components requiring an "effective program" under the license renewal rule.) The Commission also directed the staff, in developing the proposed rule, to ensure that the voluntary nature of the improved Standard Technical Specification program be preserved and that the Federal Register notice indicate that public comments on the proposed rule will be welcomed, considered, and addressed during preparation of the final rule. The staff was also directed to prepare any regulatory guides needed to implement this rule.

   We agree with the above actions by the Commission and believe that the staff has appropriately modified the Policy Statement in response to the Commission's comments. The staff, of

course, needs to proceed with the other matters covered by these comments.

2. The Commission also directed the staff to modify the Policy Statement to clarify how it intends to utilize probabilistic risk assessments (PRAs) in its review of Technical Specification change requests involving Criterion 4 - "A structure, system, or component which operating experience or probabilistic safety assessment [PRA] has shown to be significant to public health and safety." The Commission apparently has no problem with this criterion, but believes that if the results of a PRA indicate that Technical Specifications can be relaxed or removed, a deterministic review should be performed. If the results of the deterministic review also support relaxing or removing the Technical Specifications, the staff should not preclude such action.

   We agree with the view expressed by the Commission on this issue. The staff believes that it has responded to the Commission's comment in the modified Policy Statement by clarifying how it intends to utilize PRA in its review of Technical Specification change requests. We believe that the staff needs to provide more detailed guidance on the definition of "significant to public health and safety." This additional guidance should probably appear in the implementing regulatory guide(s).

This problem with Criterion 4 also exists in a number of recent staff initiatives (obvious examples are structures, systems, and components to be covered by the Maintenance Rule and the staff's reluctance to define "vulnerabilities" with respect to the Individual Plant Examination program).

Many problems related to the use of PRA by the NRC staff were described in our May 20, 1993 letter concerning the "Draft Report of the PRA Working Group." The issue raised in the present report is in the same class.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

<u>References</u>:

1. SECY-93-067 dated March 17, 1993, for the Commissioners from James M. Taylor, Executive Director for Operations, NRC, Subject: Final Policy Statement on Technical Specifications Improvements

2. Memorandum dated June 3, 1993, from Brian K. Grimes, Office of Nuclear Reactor Regulation, for John T. Larkins, ACRS, Subject: Request for ACRS Review of Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors
3. Staff Requirements Memorandum dated May 25, 1993, from Samuel J. Chilk, Secretary, for James M. Taylor, Executive Director for Operations, NRC, Subject: SECY-93-067 - Final Policy Statement on Technical Specifications Improvements

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

July 22, 1993

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

Dear Dr. Wilkins:

Subject: Policy Statement on Technical Specifications Improvements

I am responding to the letter you sent to the Chairman on June 18, 1993, particularly with regard to your comments regarding the need for more detailed guidance on the definition of "significant to public health and safety," as it relates to the application of "Criterion 4," and the more general concerns regarding the use of probabilistic analyses by the NRC staff.

As you noted, the Commission has directed the staff to prepare a rulemaking package that would codify the four criteria in SECY-93-067, and any regulatory guides needed to implement the rule. During the rulemaking effort, the staff expects to receive public comments on the application of Criterion 4, including the interpretation of "significant to public health and safety." The staff will address those comments, and clarify the appropriate use of probabilistic analyses relative to the application of Criterion 4, in a manner that is consistent with my July 6, 1993 response to Dr. Shewmon's May 20, 1993 letter, regarding the results of the PRA Working Group and the staff's use of probabilistic analyses.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
SECY

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

April 13, 1995

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

Dear Chairman Selin:

SUBJECT: PROPOSED FINAL RULE CHANGE TO 10 CFR 50.36, TECHNICAL SPECIFICATIONS

During the 420th meeting of the Advisory Committee on Reactor Safeguards, April 6-7, 1995, we discussed with representatives of the NRC staff and the Nuclear Energy Institute the subject proposed final rule change to technical specifications. We had the benefit of the documents listed.

The "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 23, 1993, established four criteria to define requirements that should be controlled by technical specifications. The Commission concluded that it was appropriate to codify these criteria in a rule that would be consistent with the Policy Statement and preserve the voluntary nature of adopting the improved Standard Technical Specifications for previously licensed plants.

In our June 18, 1993 report, we stated our agreement with the views expressed by the Commission on this matter and concluded that the staff had appropriately modified the Policy Statement in response to the Commission's comments. We did express a concern that there was a need for more detailed guidance on the definition of "significant to public health and safety" as it is used in Criterion 4 of the final Policy Statement.

The staff proposes to implement Criterion 4 in a manner consistent with the Commission's policies on the use of probabilistic risk assessment methods and the staff's PRA Implementation Plan.

The staff maintains that the improved Standard Technical Specifications, the final Policy Statement, the Backfit Rule, and the statement of consideration for this proposed final rule change contain sufficient guidance for implementing Criterion 4. We do not agree with this position.

We have previously objected to regulations that are subject to a variety of interpretations which rely solely on the judgment of the

regulator. In the interest of coherence in regulation and predictability of the regulatory process, we recommend that codification of the rule include more explicit definition and guidance on the implementation of the "significant to public health and safety" provision of Criterion 4. We believe a rule that omits this is not complete and will not meet the pressing need for a rule on Technical Specifications Improvements. We recommend delaying issuance of the rule until it is complete.

Sincerely,

T. S. Kress
Chairman

References:

1. Draft Commission Paper, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Final Rulemaking Package for 10 CFR 50.36, "Technical Specifications," (Predecisional) transmitted by Memorandum dated March 27, 1995, from B. K. Grimes to John T. Larkins
2. Staff Requirements Memorandum dated May 25, 1993, from Samuel J. Chilk, Secretary, for James M. Taylor, Executive Director for Operations, Subject: SECY-93-067 - Final Policy Statement on Technical Specifications Improvements
3. ACRS letter dated June 18, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Policy Statement on Technical Specifications Improvements for Nuclear Power Plants
4. Nuclear Regulatory Commission, 10 CFR Part 50, Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, July 23, 1993
5. SECY-94-219 dated August 19, 1994, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Proposed Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA)
6. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities: Proposed Policy Statement," issued for public comment on December 1, 1994

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

May 31, 1995

Mr. T. S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: PROPOSED FINAL RULE CHANGE TO 10 CFR 50.36, TECHNICAL SPECIFICATIONS

Dear Mr. Kress:

In your letter to the Chairman on April 13, 1995, you commented on the ACRS discussions with representatives of the NRC staff and the Nuclear Energy Institute on the subject proposed final rule change.

In your report of June 18, 1993, you expressed a concern that there was a need for more detailed guidance on the definition of "significant to public health and safety" as it is used in Criterion 4 of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors."

To address these concerns, the staff, in the Federal Register notice for the subject proposed final rule change, proposes to implement Criterion 4 in a manner consistent with the Commission's policies on the use of probabilistic risk assessment (PRA) methods and the staff's PRA Implementation Plan. The staff believes that the improved standard technical specifications, the backfit rule, and the statement of consideration for this proposed final rule change contain sufficient guidance for implementing Criterion 4.

Your letter indicated that the ACRS does not agree with this position and has recommended that the staff include more explicit definition and guidance on the implementation of the "significant to public health and safety" provision of Criterion 4 in the Statement of Consideration for the rule. The letter stated that the ACRS believes a rule that omits such definition and guidance is not complete and will not meet the pressing need for a rule on technical specifications improvements. Therefore, your letter stated, the ACRS has recommended delaying issuance of the rule until this additional definition and guidance is incorporated.

The staff understands the Committee's concern and has carefully considered the advantages and disadvantages of publishing the final rule change in its current form versus delaying publication until the staff can develop more definitive guidance on the implementation of Criterion 4 under the PRA Implementation Plan. As discussed with the ACRS, the staff believes that the past eight years of experience using the criteria under the interim and final policy statements demonstrates that the staff can apply Criterion 4 in a consistent and controlled manner. In addition, the backfit rule has and will continue to provide adequate controls on the staff for the imposition of new regulatory requirements.

The staff has concluded that the benefit of codifying the criteria now, thus providing a clear framework for technical specifications, justifies proceeding with publication of the final rule change in lieu of waiting until additional guidance is developed under the PRA Implementation Plan, a process which will be developing over a year or more. Therefore, the staff has forwarded the final rule package to the Commission with a recommendation that the Commission proceed with publication of the final rule change. The final rule package includes a discussion of the comments made by both the ACRS and the Committee to Review Generic Requirements.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: The Chairman
Commissioner Rogers
Commissioner de Planque
Commissioner Jackson
SECY

ITEM(7): CRACKING AND FATIGUE ISSUES ASSOCIATED WITH NUCLEAR REACTOR COMPONENTS

In December 1994, the Committee heard an information briefing on cracking in nuclear reactor components. The staff presented the types of cracking experienced at nuclear power plants and discussed research on environmentally assisted cracking. During subsequent Committee discussions, the Committee noted pressurized thermal shock (PTS) as a possible area for new research. Issues supported by an industry owners group, such as the BWR core shroud cracking issue, appear to be well managed by the staff. Other issues (SG tube degradation, RPV integrity) appear to be event driven.

a. BWR Core Shroud Cracking

The Committee heard information briefings on the status of core shroud cracking and licensee repairs in September and December 1994. The staff has issued two NRC Information Notices and Generic Letter 94-03 that address core shroud cracking. Eleven facilities identified cracks in their core shrouds and have planned or completed repairs. The industry-sponsored BWR Vessel and Internals Project developed the industry response to the problem, and is reviewing the ability to detect cracking of other reactor vessel internal components.

b. Reactor Vessel Integrity

The Committee heard two briefings on NUREG-1511, "Reactor Pressure Vessel Status Report," in October and December 1994. The Committee reviewed and commented on the proposed final regulatory guide, "Evaluation of Reactor Pressure Vessels With Charpy Upper-Shelf Energy Less Than 50 FT-LB," in December 1994. The Committee plans to review proposed rulemaking on reactor vessel annealing during the June 1995 ACRS meeting.

NUREG-1511 identified two reactor vessels that would exceed the PTS screening limit before the end of their license (EOL). One of these facilities, Palisades, developed additional data by analyzing retired steam generator material. The data indicated an earlier date for exceeding the limit specified in the PTS screening criteria. The staff issued a supplement to Generic Letter 92-01 "REACTOR VESSEL STRUCTURAL INTEGRITY, 10 CFR 50.54(f)" on May 19, 1995, requesting all licensees to develop additional fabrication data and to reanalyze the structural integrity of their reactor vessels.

c. Steam Generator Tube Degradation

The Committee provided comments on the draft and proposed final Generic Letter, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes," on September 12, 1994, and May 12, 1995, respectively. The generic letter allows relaxation of the repair

criteria for a specific type of cracking in specific types of Westinghouse steam generators. In cooperation with the Nuclear Energy Institute (NEI), the staff is developing proposed rulemaking for comprehensive steam generator surveillance and maintenance. Advances in non-destructive examination (NDE) hardware and NDE techniques are improving the detection and sizing of defects in steam generator tubes. Recent areas of concern are circumferential cracking at the tube sheet, cracking of sleeves and plugs, and the qualification of NDE techniques.

d. Reactor Vessel Head Cracking

The Committee heard an information briefing on the status of reactor vessel head penetration cracking in September 1994. Cracks in reactor vessel head penetrations were first identified in Europe. Small cracks were identified at two of the three facilities inspected in the United States. NEI concluded no further actions were necessary. The staff has requested that the industry address the cracking mechanism. European countries are continuing to conduct research on the issue.

e. Metal Fatigue of Primary Pressure Boundaries

In a May 21, 1993 Staff Requirements Memorandum, the Commission requested that the ACRS review and comment on the Branch Technical Position dealing with metal fatigue. The staff developed a staff action plan in July 1993 and updated the plan in June 1994. The Committee is waiting for a draft of the staff position paper to begin its review.

Attachments:

- Letter dated September 12, 1994, from T. Kress, Chairman ACRS, to I. Selin, Chairman NRC, Subject: Proposed Generic Letter 94-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes" (pp.72-74)

- Letter dated October 19, 1994, from J. Taylor, Executive Director for Operations, to T. S. Kress, Chairman ACRS, Subject: Response to ACRS comments Regarding Proposed Generic Letter 94-XX and the Advanced Notice For Proposed Rulemaking on Steam Generator Tube Integrity (pp.75-)

- Letter dated May 15, 1995, from T. S. Kress, Chairman ACRS, to J. Taylor, Executive Directors for Operations, Subject: Proposed Final Generic Letter 95-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes" (pp.76-77)

- Letter dated December 20, 1994, from T. S. Kress, Chairman ACRS, to J. Taylor, Executive Director for Operations, Subject: Proposed Final Draft Regulatory Guide, DG-1023, "Evaluation of Reactor Pressure Vessels With Charpy Upper-Shelf Energy Less Than 50 FT-LB" (pp. 78)

- Letter dated February 17, 1995, from J. Taylor, Executive Director for Operations, to T. S. Kress, Chairman ACRS, Subject: Proposed Final Regulatory Guide, DG-1023, "Evaluation of Reactor Pressure Vessels With Charpy Upper-Shelf Energy Less Than 50 FT-LB" (pp. 79)

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

September 12, 1994

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

Dear Chairman Selin:

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

During the 412th meeting of the Advisory Committee on Reactor Safeguards, August 4-5, 1994, we reviewed the subject generic letter (GL), an associated differing professional opinion (DPO), and a draft of an Advance Notice of Proposed Rulemaking on Steam Generator Tube Integrity. During the 413th meeting, September 8-10, 1994, we discussed the NRC staff's revised calculations for radiological consequences of a main steamline break associated with a degraded steam generator. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI), as well as the author of the DPO. We also had the benefit of the documents referenced. In part, this report is in response to a request made by the Executive Director for Operations in a July 15, 1994, memorandum to the Executive Director of the Advisory Committee on Reactor Safeguards.

Although existing mechanics-based design criteria and evaluation methods have served to ensure adequate steam generator tube integrity, they appear to be overly conservative for some types of degradation, and result in unnecessary tube plugging or repair. The proposed GL provides an alternate approach applicable solely to axially oriented outside diameter stress corrosion cracking (ODSCC) of tubes at the tube-support-plate intersections in Westinghouse steam generators with drilled-hole support plates.

We support the issuance of the proposed GL for public comment. We have reviewed the DPO and do not believe that it identifies any fundamental shortcomings in the approach proposed in the GL.

The DPO cites a high core damage frequency (CDF) of 3.4×10^{-4}/RY. This value was based on a preliminary scoping analysis performed by the Office of Nuclear Regulatory Research (RES). Subsequent analyses performed by RES in support of the application of the interim plugging criteria for the Trojan Nuclear Plant and for NUREG-1477 give CDFs of less than 2×10^{-6}/RY. These values are based on conservative estimates of leakage from degraded tubes. Except perhaps for steamline breaks, the structural restraint provided by the tube-support plate provides a high degree of assurance against tube bursts.

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The criticism in the DPO of the approach used in the proposed GL and in the Standard Review Plan to compute radiological releases during a main steamline break appears to warrant further consideration. The basis for the definition of the iodine spike during a rapid depressurization transient as 500 times the equilibrium release rate is not clear. However, an alternate calculation of the release based on the gap inventory of iodine in leaking fuel elements appears to give comparable releases. In both approaches there appears to be margin in meeting the 10 CFR Part 100 limits. The staff should review the spiking data or consider other approaches to estimate the iodine release to provide a more satisfactory basis for the radiological dose estimates. In particular, we encourage the staff to quantify the level of conservatism in its analyses.

While the proposed GL appears to provide a useful interim approach for assessing steam generator tube integrity, the database for the present empirical correlations for burst pressure and leakage with the bobbin coil voltage, appears to be only marginally adequate, and more data need to be developed.

The use of such empirical correlations as the basis for assuring the integrity of steam generator tubing would also seem to require an ongoing tube-pull program with associated burst and leak testing and metallurgical examinations as outlined in the proposed GL to ensure that the correlations remain valid as degradation continues. In the longer term, it would be worthwhile to reconsider a fracture-mechanics-based approach utilizing improved non-destructive examination techniques that provide more accurate detection and characterization of degradation. Ongoing efforts in RES and in industry to develop and implement such an approach should be continued and encouraged.

We agree with the staff position that rulemaking is the preferred regulatory approach to the problem of steam generator tube degradation, although we are skeptical that a new rule can be developed as expeditiously as the proposed schedule suggests. The overall objective and attributes of the new rule, as described by the staff, pay proper obeisance to performance-based regulation. We would like to be kept informed of the progress by the staff in the implementation of a performance-based approach.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Memorandum dated July 8, 1994, from F. J. Miraglia, Deputy Director, Office of Nuclear Reactor Regulation, for E. L. Jordan, Chairman, Committee to Review Generic Requirements, Subject: CRGR Review of Generic Letter 94-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes"
2. Memorandum dated July 15, 1994, from J. M. Taylor, NRC Executive Director for Operations, for J. T. Larkins, ACRS Executive Director, Subject: ACRS Review of Proposed Generic Letter 94-XX, Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes
3. U.S. Nuclear Regulatory Commission, 10 CFR Part 50, RIN 3150-, Steam Generator Tube Integrity (7590-01), Draft Advance Notice of Proposed Rulemaking, received July 20, 1994
4. Memorandum dated August 17, 1994, from J. A. Calvo, NRC Office of Nuclear Reactor Regulation, for J. T. Larkins, ACRS Executive Director, Subject: Revisions to Slides Used by Staff During August 3, 1994, Subcommittee Briefing on Steam Generator Alternate Repair Criteria
5. U.S. Nuclear Regulatory Commission, NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," Draft Report for Comment, June 1993
6. Memorandum dated January 15, 1993, from E. S. Beckjord, Director, Office of Nuclear Regulatory Research, to T. E. Murley, Director, Office of Nuclear Reactor Regulation, Subject: Interim Plugging Criteria for Trojan Nuclear Plant

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

October 19, 1994

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

Dear Dr. Kress:

SUBJECT: RESPONSE TO ACRS COMMENTS REGARDING PROPOSED GENERIC LETTER 94-XX AND THE ADVANCED NOTICE FOR PROPOSED RULEMAKING ON STEAM GENERATOR TUBE INTEGRITY

I received your letter of September 12, 1994, in which you forwarded the comments of the Advisory Committee on Reactor Safeguards (ACRS) on both the proposed generic letter on voltage-based repair criteria (Generic Letter 94-XX, "Voltage-Based Repair Criteria for the Repair of Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking") and the advanced notice for proposed rulemaking on steam generator tube integrity. The staff will address these comments as part of its continuing efforts in the steam generator tube integrity area. The staff will keep ACRS informed of progress made on both the generic letter and the steam generator tube integrity rule.

Your input on this important subject is appreciated.

Sincerely,

James M. Taylor
Executive Director
for Operation

cc: The Chairman
Commissioner Rogers
Commissioner de Planque
SECY

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

May 15, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL GENERIC LETTER 95-XX, "VOLTAGE-BASED REPAIR CRITERIA FOR WESTINGHOUSE STEAM GENERATOR TUBES"

During the 421st meeting of the Advisory Committee on Reactor Safeguards, May 4-6, 1995, we discussed the subject generic letter. During this meeting, we had the benefit of discussions with representatives of the NRC staff, the Nuclear Energy Institute, and the Southern Nuclear Operating Company. We also had the benefit of the documents referenced.

We provided comments on a draft version of the generic letter in our report dated September 12, 1994. A number of changes have been made in the generic letter as a result of public comments. These changes do not affect our technical assessment that the generic letter provides an acceptable approach to ensure the integrity of tubing subject to axially oriented outside diameter stress corrosion cracking in Westinghouse steam generators with drilled-hole support plates.

In our September 12, 1994 report, we noted that the database for the present empirical correlations of burst pressure, leakage, and bobbin coil voltage appears to be only marginally adequate. Because of this, we believe the staff decision to retain the conservative lower voltage limits of 2 volts for 7/8-inch diameter tubing and 1 volt for 3/4-inch diameter tubing until more experience is gained with the application of the criteria is prudent and appropriate.

In our previous report, we noted that the concern raised in the differing professional opinion on the calculation of the radiological releases during a main steamline break appeared to warrant further consideration. This issue has not yet been resolved, but we believe that timely implementation of the generic letter should proceed to prevent unnecessary tube repairs and reduce staff resources associated with plant-specific reviews. However, the radiological release issue should be addressed in the proposed rule on steam generator tube maintenance and surveillance.

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Memorandum dated April 6, 1995, from Brian Sheron, Director, Division of Engineering, NRR, to John Larkins, Executive Director, ACRS, Subject: ACRS Review of Generic Letter (GL) 95-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes"
2. ACRS Report dated September 12, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Proposed Generic Letter 94-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes"

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

December 20, 1994

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL DRAFT REGULATORY GUIDE, DG-1023, "EVALUATION OF REACTOR PRESSURE VESSELS WITH CHARPY UPPER-SHELF ENERGY LESS THAN 50 FT-LB"

During the 416th meeting of the Advisory Committee on Reactor Safeguards, December 8-10, 1994, we discussed the subject regulatory guide with representatives of the NRC staff. We provided comments on an earlier version of this regulatory guide in a letter dated July 15, 1993. We also had the benefit of the documents referenced.

The data from ongoing reactor surveillance programs suggest that the Charpy upper-shelf energy will decrease to less than the present regulatory limit of 50 ft-lb for a number of operating reactor pressure vessels. The need for a guide for the evaluation of the integrity of such pressure vessels was highlighted during the discussion of the Yankee Rowe reactor pressure vessel. We believe that this guide will prove useful to the licensees and the staff, and endorse its adoption.

Additional comments by ACRS members Ivan Catton, Thomas S. Kress, Dana A. Powers, and Robert L. Seale are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

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

February 17, 1995

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

Dear Dr. Kress:

SUBJECT: PROPOSED FINAL REGULATORY GUIDE, DG-1023, "EVALUATION OF REACTOR PRESSURE VESSELS WITH CHARPY UPPER-SHELF ENERGY LESS THAN 50 FT-LB"

I am writing in response to your December 20, 1994, letter concerning the subject proposed final regulatory guide. We appreciate your review and endorsement of the proposed guide and we are proceeding with the review process.

With regard to the additional comments provided by ACRS members, the staff is preparing a response to those comments and will forward it to you in the near future.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: The Chairman
Commissioner Rogers
Commissioner de Planque
SECY

ITEM(8): DIGITAL INSTRUMENTATION AND CONTROL

a. National Academy of Sciences/National Research Council study of digital systems in nuclear power plants - Status Report

Following an extended ACRS review (2+ years) of the state of digital technology and its use in nuclear power plants, ACRS wrote a March 18, 1993 letter to the Commission recommending that the NRC seek the advice of the National Academies of Sciences and Engineering/National Research Council (NAS) regarding appropriate regulatory requirements needed for the use of digital technology in nuclear power plants. The staff chose to conduct its own digital reliability workshop with selected digital and nuclear industry participants. ACRS was not satisfied with the results of this workshop and again repeated its earlier recommendation in a letter dated November 16, 1993. On December 16, 1993, the Commission directed the staff to engage NAS to conduct a workshop to assist in establishing the appropriate approach to regulation of digital systems in nuclear safety and control systems.

Following an ACRS briefing to the Commission in March 1994, the Commission requested the ACRS to review and comment on the NAS proposal for a study and workshop. Subsequent to a presentation by NAS representatives, the Committee issued a letter dated July 14, 1994, generally supporting the NAS proposal. The NAS is expected to produce the first of two reports by the end of July 1995. The ACRS will meet with NAS representatives in August to discuss this "phase one" report which is expected to identify the significant issues to be studied further in "phase two." ACRS is particularly interested in how the NAS report addresses software specification/verification/validation, hardware environmental qualification, and system reliability (including common-mode failure).

b. Development, issuance, and NRC staff endorsement of EPRI TR-102348 "Guideline on Licensing Digital Upgrades"

In September 1994 the ACRS commented on the staff's proposed generic letter on the use of EPRI report TR-102348, "Guideline on Licensing Digital Upgrades," indicating general agreement and recommending that the staff emphasize in the generic letter the need to address the full range of potential environmental stressors in implementing digital upgrades. The staff incorporated the ACRS comments and issued this generic letter during April 1995.

c. Ongoing staff and industry initiatives in the digital I&C area

The ACRS intends to closely follow the staff revisions to the Standard Review Plan and development of Branch Technical Positions and Regulatory Guides intended to create a coherent framework for staff review of digital system upgrades. This effort is expected to continue through most of 1996.

Attachments:

- Talking points for ACRS briefing to Commission (pp. 82-84)

- Letter dated March 18, 1993, from P. Shewmon, Chairman ACRS, to I. Selin, Chairman NRC, Subject: Computers in Nuclear Power Plant Operations (pp. 85-90)

- Letter dated November 16, 1993, from J. Ernest Wilkins, Jr., Chairman ACRS, to I. Selin, Chairman NRC, Subject: Computers in Nuclear Power Plant Operations (pp. 91-92)

- Letter dated July 14, 1994, from T. S. Kress, Chairman ACRS, to I. Selin, Chairman NRC, Subject: Proposed National Academy of Sciences/National Research Council Study and Workshop on Digital Instrumentation and Control Systems (pp. 93-94)

- Letter dated September 14, 1994, from T. S. Kress, Chairman ACRS, to J. Taylor, Executive Director for Operations, Subject: Proposed Generic Letter on the Use of NUMARC/EPRI Report TR-102348, "Guideline on Licensing Digital Upgrades" (pp. 95-96)

  Letter dated September 30, 1994, from J. Taylor, Executive Director for Operations, to T. S. Kress, Chairman ACRS, Subject: Proposed Generic Letter on the Use of NUMARC/EPRI Report TR-102348, "Guideline on Licensing Digital Upgrades" (pg. 97)

TALKING POINTS FOR ACRS DIGITAL I&C DISCUSSION WITH NRC COMMISSIONERS

Why Digital Instrumentation and Control (I&C) Systems in Nuclear Power Plants?

- Replace Obsolete Analog Instrumentation
- Maintain Safety
- Improve System Functions and Plant Reliability
- Increase Plant Performance
- Reduce Operation and Maintenance (O&M)

There has been a major increase in Nuclear Power Plant capacity factors over the past ten years.

Digital I&C Systems will:

- Contribute to trend of increasing capacity factors,
- Provide opportunity for increased power output,
- Reduce costs.

Summary of Key Activities

- EPRI/Industry Demonstration NPP I&C System Upgrade Program, 1990 - present (ongoing)

- EPRI/NUMARC* Guideline on Licensing Digital Upgrades

  - Stimulated by staff's 1992 draft Generic Letter stating that all digital upgrades would be unreviewed safety questions.

  - Following intensive effort by EPRI, NUMARC, Industry, and NRC staff, Guideline was published December 1993.

  - NRC Generic Letter, issued April 1995, endorses Guideline with two clarifications.

- National Academy of Science Study, 1995 - present (ongoing)

  - ACRS recommendations prompted NRC to initiate NAS study and workshop.

  - Independent review of digital I&C nuclear regulatory issues by experts from different industries.

- Update of NRC Standard Review Plan (NUREG-0800), Section 7 (I&C), 1995 - present (ongoing)

- Regulatory Guide 1.152 revision 1 endorses IEEE Standard 7-4.3.2 "Criteria for Programmable Digital Computer Software in Safety Systems in Nuclear Power Plants" (issued for public comment in May 1995).

- DOE/EPRI/Industry Integrated I&C System Upgrade Program

  - High priority partnership agreement.

* Note: Now known as Nuclear Energy Institute (NEI).

Summary

- Through a series of critical reviews and recommendations, ACRS has encouraged the NRC staff to more clearly define regulatory requirements for digital I&C systems in NPPs.
- The NRC staff and the nuclear industry have worked together to establish and define regulatory guidance for the installation of digital I&C systems in NPPs.
- The NRC staff is developing inspection procedures and training programs to implement regulatory guidance.
- The ACRS will continue to monitor staff and industry activities in this important area.

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

March 18, 1993

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

Dear Chairman Selin:

SUBJECT: COMPUTERS IN NUCLEAR POWER PLANT OPERATIONS

During the 395th meeting of the Advisory Committee on Reactor Safeguards, March 11-12, 1993, we discussed the staff's progress in defining the regulatory requirements for digital instrumentation and control systems. During this meeting, we had the benefit of discussions with members of the NRC staff.

We have now had a long series of meetings, and have heard from many relevant people, but by no means all. To some extent our input has been biased in the direction of people, groups, and organizations who have experienced problems, and we have not heard from the legions of organizations who have successfully made the move into the computer world. It is important not to develop a tabloid mentality about new technology, i.e., aberrations from the norm treated as if they were the norm.

A first observation is that many of the anecdotes about catastrophic failures of major computer systems refer to systems far larger than those of interest here. Even the software systems on the C-17 aircraft, written in nearly a dozen languages for nearly a dozen machines, are far larger than any of relevance to the nuclear business. The Strategic Defense Initiative dispute is even less relevant. So we have to maintain perspective about scale.

A second observation is that computerization provides an opportunity, not a threat. The extraordinary reliability of electronic systems (unless abused), their potential for continuous and extensive self-testing in real time, their potential for relatively painless upgrades as experience accumulates, their ability to cover an enormous function space and to accommodate unseemly amounts of input data, their remarkable immunity to wear

(few, if any, moving parts)—all these provide the potential for safety enhancement. Much of our input from the staff has been devoted to the negative aspects of computerization, as if it were a disease to be kept in check.

A related observation is that the transition to computerized operation, control, instrumentation, support, recordkeeping, and maintenance procedures and records, is inevitable. The job of the NRC is not to manage or resist the transition, but to maintain a reasonable level of assurance that it is accomplished with proper accounting for the impact on safety. With any reasonable use of the technology the impact is expected to be large and positive.

The regulatory issues we have isolated in our series of subcommittee meetings fall broadly into two categories. One is a consequence of lack of nuclear regulatory experience with modern electronics, especially computers, leading to both extraordinary conservatism relative to unfamiliar accident sequences, and the application to a new technology of review methods and nomenclature derived from old habit and experience. The second is a collection of genuinely new problems associated both with the complexity of the new technology and with the consequent difficulty of assessing (as distinguished from assuring) its level of safety. We deal with these in order.

Failures of computerized systems (excluding fans, hard disks, and other mechanical components) do not follow the traditional bathtub curve of infant mortality, stable performance, and then wearout. Electrons don't wear out. Both in electronic hardware and software there tends to be a period of infant and young adult mortality (to which we will return), with performance and reliability gradually improving with time simply through natural selection—bugs are ironed out through experience and through extensive testing. There is no later period of wear, so there is no place for the regulatory and maintenance procedures associated with that part of the reliability pattern. Further, self-testing can provide constant assurance of full functionality of the electronics.

As a consequence, however, there has been little progress in applying the methods of probabilistic risk analysis, on which we have become so heavily dependent for mechanical, hydraulic, and electromechanical systems, to computer systems. Indeed the semiconductor components of the computerized systems are inherently so reliable that high-temperature life-testing is the only means available, in most cases, for generating any failures at all. Whereas one can generate probabilities for the existence of perinatal defects, there is no such thing as a probability per unit

time for the development of disease. Nor does in-service inspection play the same role.

These are important points, because the concepts of reliability and reproducibility differ, and the testing and verification procedures used depend on which is to be assured. A mechanical component with a presumed reliability of 10^{-3} failures per demand can be tested a few thousand times to assure that level of reliability, but a software-based system with a hidden bug that will be revealed in the event of an unlikely input configuration can be tested without failure until the cows come home, but will still always fail with that particular input. Interest has therefore to be directed at the probability that there is such a hidden bug, and the probability that some other circumstance may generate the unfortunate input. Neither of these probabilities will be discovered by repetitive testing under normal conditions. Randomized input testing can tell one something about the former probability, but not the latter. It is therefore misleading to bandy failure probabilities around, as if they had the same meaning as they do for familiar mechanical and electrical components. It also makes the direct comparison of computerized system reliability with the reliability of older technology more difficult.

These and other considerations mandate a format adjustment for the regulatory system, and such changes tend to be painful. What we have seen here is an unfortunate effort to cling to the old ways, to the point of asking that all digital systems have analog backups—not because the latter are better or more reliable, but because they are more familiar to the regulator and therefore easier to regulate. That alone could place an unwarranted burden on those seeking to improve safety by updating technology.

The second category of issues follows from the undoubted fact that computerized systems do indeed introduce unfamiliar failure modes, which require both recognition and palliative measures. Too much attention appears to have been concentrated on a microcosm of the more recognizable of these matters, specifically vulnerability of digital systems to electromagnetic interference (a subject on which there is enormous military expertise, largely untapped by the NRC staff), and the fact that replicated defective software (like replicated defective hardware) can be the source of common-mode failures. Both of these are real issues, but, in our judgment, not the central ones.

Let us first consider software issues. The literature is full of examples of cases in which carefully written and tested software still contains errors. Indeed it is doubtless true, though in

principle unprovable, that any large program that has not undergone a formal verification and validation (V&V) contains yet undiscovered errors. Lest there be confusion, it is well to be quantitative about the problem of implementing a function in software.

The simplest of all digital programs might generate a logic function, a mapping that accepts a number of binary inputs (say n) and generates a single binary output—a signal that might, in turn, activate a pump or a valve or some other sequence of events. Such a logic function has 2^n possible input states, over a thousand for n=10 and over a million for n=20. These are not unreasonable numbers of input states, because the input of a single number to one percent accuracy requires seven (usually more) binary inputs. Since each input state can have either output state (on/off), that means that even a modest eight-input binary converter of this sort can represent 2^{256} or 10^{77} different logic functions. A defect (either hardware or software) can change the desired function into any of the others. It is therefore reasonable to expect to test the system to make sure that it performs as designed, but not reasonable to expect to explore, by brute force, all consequences of all possible defects. The point is only strengthened if one has more complex outputs than just a single bit.

If, therefore, the requirements specified for the system describe the full mapping of the input space to the output space, special methods will be required to verify that this has been accomplished correctly. Such methods exist, and are applicable to relatively simple software packages. When formal V&V is possible, it provides assurance that the code, as written, correctly implements the formal specifications laid upon the design. When it is not possible (because the code is too long or too complex), there are many alternatives, but none of them provides the kind of assurance of code fidelity that is provided by formal V&V.

There appears to be a consensus among the experts we have consulted that the safety-related software in nuclear power plants is within reach of formal V&V methods, and that the potential for serious error lies more in incorrect expression of the specifications than in incorrect programming. Formal V&V can assure that the code correctly expresses the specifications, but not that the specifications are correct. In either case, it would appear that the staff emphasis on the possibility of common-mode errors in code segments used in different parts of the instrumentation and control system is misdirected. We continue to see an urgent need for staff augmentation with people experienced in thinking in the terms outlined above.

We believe that the experience of other industries that have accepted the progress has been characterized, almost without exception, by increases in efficiency and reliability, and by concomitant decreases in cost. (While the latter is not the NRC's business, it remains true that resources and attention released from unproductive safety concerns may, at least in part, find their way to better use.) There are genuine safety issues in this transition, of which one unfamiliar one is surely the requirement, in order to generate verifiable software, for precise no-nonsense attention to the specification of the functions to be implemented by the software.

The gist of our concerns is that the regulatory procedures developed during the decades preceding the full flowering of the electronic revolution (which may not yet have occurred) are inappropriate to the regulation of computerized functions in nuclear power plants. (This is true for both hardware and software—too much emphasis on the distinction is not helpful.) As a consequence, the staff has been dealing with the problems that have shown up so far on an ad hoc basis, applying methods created for each problem, with little underlying methodology. That has resulted in such distractions as the analog-to-digital conversion problem, the overemphasis on electromagnetic interference problems, the singling out of software common-mode failure as a central issue, etc., all without a framework into which the broad issues of regulatory emphasis and consistency can be fitted. We can cavil about the specific staff approaches to each of these, but the central issue is that neither the staff nor the Commission has established what could be described as a standard review plan or even a regulatory guide that could help both the staff and the industry know what is expected of them. A statement of the applicable standards ought to precede, not follow, their application. Without such a definition of objectives, coherence is an inevitable victim.

What, then, do we recommend? We frankly doubt that a coherent and effective review plan for computerized applications in nuclear power plants will be produced by the staff, the Commission (whose job is at a higher policy level), or the Committee (which is limited in both resources and expertise). Still, if one believes (as we do) that it needs to be done, it will be necessary to bring in outside help. It was in that context that we initiated our long series of subcommittee meetings on the subject. Our recommendation is that a workshop and study (with a charter to produce such a plan) be commissioned to be done by the National Academies of Sciences and Engineering. To derive maximum benefit from such a

study, there should be appropriate participation by key senior members of the staff.

Additional comments by ACRS Members James C. Carroll and Carlyle Michelson are presented below.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

Additional Comments by ACRS Members James C. Carroll and Carlyle Michelson

We agree with most of the technical observations made in this report. However, we disagree with the report's recommendation that a workshop and study be undertaken by the National Academies of Sciences and Engineering for the purpose of developing a review plan for computerized applications in nuclear power plants. Contrary to the view of our colleagues, we believe that the staff and its consultants are making satisfactory progress toward developing a "coherent and effective" review plan. Ideally, such a plan should have been developed in advance of the receipt of applications for the use of this rapidly changing technology. As a practical matter, it has been necessary for the staff to interact with the first group of applicants proposing computerized systems in order to gain an understanding of these systems. This has been a necessary first step before a generic review plan can be developed. Our view is that the proposed National Academies of Sciences and Engineering workshop and study would add little to the process of developing a staff review plan at this point in time.

We note that the staff has attended the series of ACRS subcommittee meetings on computerized applications in nuclear power plants that form the basis for this Committee report. In addition, the staff is planning to sponsor a workshop this fall and plans to obtain ACRS feedback on speakers and topics to be covered.

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

November 16, 1993

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

Dear Chairman Selin:

SUBJECT: COMPUTERS IN NUCLEAR POWER PLANT OPERATIONS

On March 18, 1993, we wrote you a report on the NRC staff approach to regulation of computers in nuclear power plant operations and upgrades. While there were many specific observations and suggestions in that letter, it ended by concluding that a fresh start was called for in developing an effective approach to this new and difficult subject, and recommended that you ask the National Academies of Sciences and Engineering to conduct a workshop directed at this end.

In the interim the staff has conducted its own workshop on digital systems, with the help of the National Institute of Standards and Technology, on September 14-15, 1993. Some of us attended that workshop, and our Chairman gave introductory remarks. It is therefore appropriate to ask whether that workshop served as a reasonable substitute for our earlier recommendation. We have concluded that it did not.

To begin, it was not a workshop, in the usual sense of the word. It was organized much as a technical session of a learned society, with a variety of relatively disconnected speeches by experts, limited opportunity for questions from the audience, and only a little opportunity for the experts to discuss the issues with each other.

The recommendation in our earlier letter was based on the belief that an open-minded approach, using the wealth of expertise in the outside world, might help to supply the framework on which a coherent regulatory structure might be hung. Wrangling over specific details of the staff position, like the requirement for hard-wired redundancy, or concentration on electromagnetic interference, could lead to a compromise animal, half fish and half cat, with little underlying rationale.

Based on our observation of the staff workshop, and discussions with our foreign colleagues during the recent Quadripartite Meeting of Advisory Committees, we have concluded that our recommendation to seek help outside, with a different format, remains appropriate.

The NRC can muddle through the next few years on current momentum, but lack of an underlying rationale will ultimately exact a price, perhaps a high one. There are deep issues of regulatory philosophy here, and a case-by-case approach will continue to ignore them.

We repeat our original recommendation.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

Reference:
Report dated March 18, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations

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

July 14, 1994

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

Dear Chairman Selin:

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

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

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

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

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

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

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

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

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

September 14, 1994

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

Dear Mr. Taylor:

SUBJECT: PROPOSED GENERIC LETTER ON THE USE OF NUMARC/EPRI REPORT TR-102348, "GUIDELINE ON LICENSING DIGITAL UPGRADES"

During the 413th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 1994, we reviewed the subject proposed generic letter. During our 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.

The proposed generic letter endorses, with two clarifications, Nuclear Management and Resources Council/Electric Power Research Institute (NUMARC/EPRI) Report TR-102348 as useful guidance for effectively implementing digital upgrades and for determining when these can be performed without prior NRC staff approval under the requirements of 10 CFR 50.59.

We basically concur with the proposed generic letter and have no objection to issuing it for public comment. However, we believe that additional clarification should be provided regarding equipment environmental compatibility. Specifically, it should be made clear in the generic letter that the environmental requirements as defined in Subsection 5.3, "Compatibility With the Environment," of the NUMARC/EPRI report include all environmental conditions resulting from internal and external events to which the equipment may be subjected. This subsection currently focuses on the need to address electromagnetic interference. We believe that any guideline which purports to cover environmental compatibility issues for replacement equipment must require that other environmental stressors such as temperature, humidity, radiation, vibration/seismic, and smoke be addressed. We note that the need to prioritize these and to verify the appropriateness of current research programs was identified in our letter of November 12,

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1992, and that you agreed. We anticipate a briefing on the results of this effort.

Sincerely,

T. S. Kress
Chairman

References:

1. Memorandum dated August 30, 1994, from E. Doolittle, NRC Office of Nuclear Reactor Regulation, to J. Larkins, ACRS Executive Director, forwarding Proposed NRC Generic Letter on the Use of NUMARC/EPRI Report TR-102348, "Guideline on Licensing Digital Upgrades"
2. Letter dated December 22, 1993, from W. Rasin, NUMARC, to W. Russell, NRC Office of Nuclear Reactor Regulation, forwarding EPRI Report TR-102348
3. ACRS letter dated November 12, 1992, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Environmental Qualification for Digital Instrumentation and Control Systems
4. Letter dated December 10, 1992, from James M. Taylor, NRC Executive Director for Operations, to Paul Shewmon, ACRS Chairman, Subject: Environmental Qualification for Digital Instrumentation and Control Systems

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

September 30, 1994

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

SUBJECT: PROPOSED GENERIC LETTER ON THE USE OF NUMARC/EPRI REPORT TR-102348, "GUIDELINE ON LICENSING DIGITAL UPGRADES"

Dear Dr. Kress:

I am responding to your letter of September 14, 1994, which provided comments on the subject proposed generic letter. The staff agrees with your comment that a further clarification regarding equipment environmental compatibility for digital upgrades is appropriate. Therefore, the staff will clarify the proposed generic letter to state that digital upgrades are to be qualified for operability against those environmental stressors and for those events specified in the plant specific licensing basis. The staff will note that temperature, humidity, and radiation are cited in EPRI Report TR-102348 as examples, but that other environmental stressors may require consideration. The staff expects to issue the proposed generic letter for public comment within a month.

In addition, you noted in your letter that the staff is continuing its efforts to prioritize and develop appropriate criteria for addressing environmental stressors on digital systems. The staff will work with the ACRS staff to establish a time for a briefing on this effort as important milestones are achieved.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: Chairman Selin
Commissioner Rogers
Commissioner de Planque
SECY

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ITEM(9): OPERATING REACTORS CONFORMANCE TO THE SAFETY GOALS - STATUS REPORT

Following the September 8, 1994 meeting between the Committee and the Commission, the Commission issued a Staff Requirements Memorandum (SRM) dated September 20, 1994, in which the Commission requested the ACRS for further guidance and insight on determining where the current population of operating plants, both individually and collectively, fall in relation to the Safety Goals. The Committee has undertaken an effort to develop a methodology which would use the results from NUREG-1150, the Lasalle Risk Methods Integration and Evaluation Program (RMIEP), and Individual Plant Examinations (IPEs) for this purpose. An ad hoc Working Group (WG) was established to explore this matter and develop comments and recommendations for consideration by the full Committee. The WG discussed this matter on January 26, 1995, and the full Committee was briefed during its February and March 1995 meetings. The observations and suggestions provided below are preliminary in nature and are yet to be reviewed and concurred in by the full Committee.

By way of background, the "complete" PRA analyses for comparison with the safety goals would include Level 1, Level 2 and Level 3 PRA analyses for sequences initiated by internal and external events during full power operation and sequences initiated during low power and shutdown modes of operation. In addition, a risk integration analysis is required to construct the overall risk parameters for all operating modes and all sequence initiators. Absent the performance of "complete" risk assessment for each plant, the comparison of the risk from the population of U.S. commercial plants with the safety goals will require extrapolation from existing analysis with the guidance and insights available from detailed PRAs (e.g., NUREG-1150, Lasalle RMIEP). However, the primary resource base for this extrapolation is the IPE/IPEEE (IPE/Individual Plant Examination of External Events) analyses.

The Commission issued an SRM dated April 28, 1995, in which they cautioned the staff regarding the use of the IPE results without full understanding of the assumptions and basis of each report. The Committee is cognizant of these concerns.

There are several major risk assessment areas where the IPE/IPEEE information is unavailable or not in a form readily useful for assessing risk:

- Extrapolation of Level 2 PRA Results to Offsite Consequences
- Core Damage Frequency (CDF) for plants using Seismic Margin methodology
- CDF for plants using Fire Induced Vulnerability Evaluation (FIVE) methodology

- Containment Accident Progression
- Assessment of Uncertainties
- Shutdown Risk
- Other External Events (external flooding, high winds, etc)

These areas are discussed below including the status of the ACRS consideration of these areas and preliminary insights and suggestions for extrapolation of IPE/IPEEE results. For comparison with risk results obtained from the simplified methods discussed below, there may be a need to develop "complete" risk profiles for several of the NUREG-1150 plants and Lasalle. The attached Table summarizes the PRA analyses performed for the NUREG-1150 plants and Lasalle.

Areas Addressed to Date

Extrapolation of Level 2 Results to Offsite Consequences

The WG has expanded on a methodology proposed by ORNL for the Generic Environmental Impact Statement on License Renewal. Our proposed approach correlates offsite risk with wind direction frequency, population demographics, volatile fission product release fraction, and the warning time for protective actions.

CDF for Seismic Margin Plants

The WG identified an approach to estimating seismic core damage frequency from the plant specific High Confidence of Low Probability of Failure (HCLPF) value determined in a seismic margins analysis, generic information on the characteristics of uncertainty distributions associated with plant seismic fragility curves from seismic PRAs, and from published seismic hazard curves (1994 LLNL Seismic Hazards Study - NUREG-1488).

CDF for Fire Induced Vulnerability Evaluation (FIVE) Plants

It is difficult to determine the conservatism in each individual analysis performed using the FIVE methodology. Consequently, we believe any risk-significant sequences identified by FIVE analysis that exceed a selected threshold (such as 10 percent of the plant total CDF) need to be evaluated further using PRA methods prior to including these fire sequence results in the plant risk profile.

Areas to be Addressed

Containment Accident Progression

Discussion is continuing on methods for estimating the conditional early containment failure probability and the radionuclide source term characteristics for classes of accident sequences for which a detailed Level 2 containment analysis may not have been performed

as part of the IPE/IPEEE program (e.g., fire and seismic sequences).

Assessment of Uncertainties

The Working Group is exploring how to utilize the point estimate values provided in the IPEs/IPEEEs and how to estimate uncertainty distribution parameters (such as means and 95th percentiles) for important risk results.

Other Areas

We are evaluating the need for providing insights and guidance in other areas including shutdown risk and other external events (external flooding, high winds, etc).

Attachments:

- Staff Requirements Memorandum dated September 20, 1994. Subject: Staff Requirements - Periodic Meeting With Advisory Committee On Reactor Safeguards (ACRS), 1:30 P.M., Thursday, September 8, 1994, Commissioners' Conference Room, One White Flint North, Rockville, Maryland. (pg. 102)
- Staff Requirements Memorandum dated April 28, 1995. Subject: Staff Requirements - Briefing On IPE Program and Severe Accident Research Program, 10:00 A.M., Wednesday, April 28, 1995, Commissioners' Conference Room, One White Flint North, Rockville, Maryland. (pg. 103)

Ten Elements of Required PRA Analysis for Safety Goal Comparison and Summary of Completed Analysis for the NUREG-1150 Plants and Lasalle

	Level 1 Core Damage Frequency Analysis			Level 2 Containment Accident Progression Analysis			Level 3 Offsite Consequence Analysis			Risk Integration Analysis
PLANT	Internal Events	External Events	Shut-down	Internal Events	External Events	Shut-down	Internal Events	External Events	Shut-down	
Surry	C	C	C	C	C	C	C		P	
Sequoyah	C			C			C			
Zion	C			C			C			
Peach Bottom	C	C		C	C		C			
Grand Gulf	C		C	C		C	C		P	
Lasalle	C	C		C	C		C	C		

C - Completed and Reported Analysis
P - Planned (or Ongoing) Analysis

ATTACHMENT 3

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

IN RESPONSE, PLEASE
REFER TO: M940908A

September 20, 1994

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

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

SUBJECT: STAFF REQUIREMENTS - PERIODIC MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), 1:30 P.M., THURSDAY, SEPTEMBER 8, 1994, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the Advisory Committee on Reactor Safeguards.

The Commission requested further guidance and insight on determining where the current population of operating plants, both individually and collectively, fall in relation to the safety goals.

The ACRS committed to provide the Commission with a copy of the trip report which discusses the French move toward performance-based fire regulations. (Subsequently, on September 12, 1994 the ACRS staff forwarded a copy of the report to the Acting Secretary for distribution to the Commission.)

The Commission requested that the ACRS continue to monitor the NRC's actions and ensure there are no areas being ignored or overlooked through an error of omission.

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

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IN RESPONSE, PLEASE
REFER TO: M950419A

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

April 28, 1995

OFFICE OF THE
SECRETARY

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John C. Hoyle, Secretary /s/

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON IPE PROGRAM AND SEVERE ACCIDENT RESEARCH PROGRAM, 10:00 A.M., WEDNESDAY, APRIL 19, 1995, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on the IPE program and severe accident research program. The Commission was pleased with the work accomplished and supported the continued efforts of the IPE program. However, the Commission notes that industry IPE results do not provide a complete basis for supporting risk-based regulatory decision-making. As stated by the staff, the staff's IPE reviews have not been of sufficient depth to allow the staff to indicate approval of, or concurrence with, the absolute values and conclusions stated in the IPE reports, and do not serve as validation of the results.

As an additional caution, the staff should not attempt to draw insights and conclusions from detailed comparisons of different IPE reports without full understanding of the assumptions and basis used in developing each report.

The Commission suggested that if there is to be further use of PRAs as a basis for risk-based regulatory changes, then the industry should, in coordination with the staff, initiate the actions necessary to develop PRAs that are acceptable for risk-based regulatory use (i.e., standardized methods, assumptions, level of detail).

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