

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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REACTOR SAFEGUARDS

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PERIODIC BRIEFING BY THE ADVISORY COMMITTEE
ON REACTOR SAFEGUARDS

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PUBLIC MEETING

Nuclear Regulatory Commission
One White Flint North
Rockville, Maryland

Friday, May 14, 1993

The Commission met in open session,
pursuant to notice, at 10:30 a.m., Ivan Selin,
Chairman, presiding.

COMMISSIONERS PRESENT:

IVAN SELIN, Chairman of the Commission
KENNETH C. ROGERS, Commissioner
JAMES R. CURTISS, Commissioner
FORREST J. REMICK, Commissioner
E. GAIL de PLANQUE, Commissioner

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STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

SAMUEL J. CHILK, Secretary

LAWRENCE CHANDLER, Assistant General Counsel for
Hearings and Enforcement

PAUL SHEWMON, Chairman, ACRS

J. ERNEST WILKINS, Vice Chairman, ACRS

CARLYLE MICHELSON, ACRS

JAMES CARROLL, ACRS

HAROLD LEWIS, ACRS

CHARLES WYLIE, ACRS

WILLIAM LINDBLAD, ACRS

IVAN CATTON, ACRS

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

10:30 a.m.

CHAIRMAN SELIN: Good morning, ladies and gentlemen.

We're pleased to welcome members of the Advisory Committee on Reactor Safeguards for the periodic meeting with the Commission. But before we begin, I would like to note, in fact I am very sad to note that this is Doctor Shewmon's last meeting with the Commission, as he will retire this month after 16 years of service on the ACRS.

Doctor Shewmon, if you would please come forward, we'd like to present to you this plaque in appreciation for all the work that you've done for us. Thank you very much, Paul.

DOCTOR SHEWMON: Thank you.

CHAIRMAN SELIN: The Commission has unanimously -- we'll miss you and by a vote of four to one, we think your work was terrific.

Today we'll hear from the ACRS on three issues. First, the status of the Committee's review of the evolutionary and passive light water reactor designs, including the EPRI utility requirement documents and the policy issues associated with these designs.

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1 Next we're to hear about the use of
2 computers and digital instrumentation and control
3 systems in nuclear power plants, and personally I just
4 think the Committee's contribution in this area has
5 been absolutely terrific. I think you've really kept
6 the Commission on the right path in this area.

7 Finally, the ACRS will provide their views
8 on the current issues in license renewal.

9 We look forward to this meeting and to the
10 periodic meetings that we have with ACRS to solicit
11 their views on issues affecting the NRC.

12 Copies of the handouts are available.

13 Do any of my colleagues care to make any
14 remarks?

15 Doctor Shewmon?

16 DOCTOR SHEWMON: The status of review,
17 we've seen the proposed schedules. They're
18 aggressive. We understand them. We have sent letters
19 on the ABWR and raised questions that haven't yet been
20 answered, though there's a meeting in San Jose in June
21 in which hopefully we will get more answers. We have
22 individuals here who are responsible for each of
23 these. I don't really have any other presentation.
24 We would be pleased to respond to questions on the
25 status of the review.

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1 COMMISSIONER ROGERS: Well, I just have a
2 few questions that maybe we could hear a little bit
3 more on. First on the fire protection issue. You've
4 indicated that you still have some concerns about the
5 HVAC system. Really, do you have some specific
6 criteria in mind to deal with your concerns there that
7 there might not be suitable isolation in the event of
8 a fire?

9 MR. MICHELSON: Yes, there are specific
10 criteria that must be met. First of all, you have to
11 determine what the challenge is to the system. The
12 challenge that is of greatest interest is actual
13 rupture of a pipe which creates far higher pressures
14 in its fire rate. Having decided though what those
15 pressures are, then you simply put in the equipment
16 that can withstand the effects or show the effects are
17 acceptable. That's the criteria.

18 COMMISSIONER ROGERS: Are you talking
19 about the HVAC system now?

20 MR. MICHELSON: Yes. Pipe breaks will
21 blow out ventilation damper if they aren't designed
22 for the peak pressures. It may run from five to 11
23 pounds depending on which particular compartment
24 you're talking about.

25 COMMISSIONER ROGERS: So this is a

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1 question of redundancy rather than isolation, is that
2 it?

3 MR. MICHELSON: No. Redundancy helps not
4 at all. They're all exposed to the same peak
5 pressure, all of them in the compartment. So, it's
6 the ability of the valves to isolate under those
7 conditions that is at issue.

8 MR. CARROLL: I guess, Carl, my way of
9 looking at it is that the design has some very
10 desirable features in it by having trains separated by
11 division and the weakness in the whole thing is that
12 Carl's pointing out a break in division one can impact
13 equipment at division two if not designed to withstand
14 the kind of pressure.

15 MR. MICHELSON: And this is because
16 there's one ventilation system and that's not separate
17 trains?

18 DOCTOR SHEWMON: That's right. It's the
19 common normal ventilation system which is not used
20 during the emergency situation but is already there
21 and is already connecting the divisions and it must be
22 isolated at the time of the event, and the event we're
23 talking about is a pressurization of several pounds
24 and it's a tough application for a heating and
25 ventilating system, so we're going to pursue with GE

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1 further. They understand the problem.

2 The same problem exists on the doors and
3 electrical penetrations, the pipe penetrations, the
4 whole bit. It's not designed for these kinds of
5 pressures necessarily.

6 COMMISSIONER ROGERS: So it's pretty
7 clear, though, exactly what your concern is there?

8 MR. MICHELSON: Yes. I think everyone
9 appreciates what the concern is.

10 COMMISSIONER REMICK: Can I follow-up on
11 that?

12 COMMISSIONER ROGERS: Sure.

13 COMMISSIONER REMICK: Carl, one thing that
14 wasn't clear to me is the case that they're not
15 designed for that or you're not sure that they're
16 designed for it.

17 MR. MICHELSON: The case right now is we
18 have no commitment to design for it and I assume that
19 in June I'll find that they now have a commitment.

20 COMMISSIONER REMICK: Okay.

21 MR. MICHELSON: And then you have to deal
22 with the adequacy of the commitment, the timing of the
23 closure, what kind of test requirements will there be
24 to verify that such a device can work under such
25 conditions, things of that sort. It's all

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1 straightforward. It's not exotic.

2 Now you may show that it's a non-problem
3 because you can stand to have them go out. It's a
4 little difficult, takes a lot of analysis and you have
5 to deal with things like projectiles created by doors
6 coming off and things of that sort. It's easier to do
7 it by design for the conditions.

8 COMMISSIONER ROGERS: Okay. On the
9 hydrogen control --

10 DOCTOR CARROLL: It is worth noting,
11 though, that this is a vulnerability that existing
12 plants have.

13 COMMISSIONER ROGERS: Yes.

14 DOCTOR CARROLL: We're simply saying,
15 "Hey, for this certified design we think you ought to
16 do better."

17 MR. MICHELSON: Well, you ought to do it
18 right.

19 COMMISSIONER ROGERS: Yes. Right

20 All right. Hydrogen control, you
21 suggested or you said that the staff doesn't have a
22 sufficient basis to establish a final hydrogen control
23 position. Do you have some ideas or comments about
24 specific data that you think have to be developed? I
25 know there are research programs underway. I was up

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1 at Brookhaven a week or so ago and listened to a
2 presentation on their studies. But how closely can
3 you pinpoint your concerns here as to what further
4 data need to be developed and any additional criteria
5 that have to be established regarding the location of
6 equipments?

7 DOCTOR SHEWMON: That's Doctor Catton's
8 item.

9 COMMISSIONER ROGERS: Yes.

10 Doctor Catton?

11 DOCTOR CATTON: I'll just try to walk
12 through what some of these things are.

13 The hydrogen rule talks about uniform
14 distribution and, if you have uniform distribution,
15 placement of igniters is no problem. But the results
16 from the Germany tests that the HCR containment shows
17 significant stratification, so one of the areas that
18 I think there's not a sufficient database deals with
19 how do you place these igniters so that you can be
20 sure they work, that they're going to be in the right
21 place and that somehow the hydrogen isn't going to get
22 by them and be somewhere else and then you condense
23 out the steam, wind up with near stoichiometric
24 mixtures and have a detonation. This is really what
25 the concern is.

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1 Now we plan to meet with some of the NRR
2 people at noon today to work out more of the details
3 of this. I think that's where the problems are.

4 COMMISSIONER ROGERS: So you think that
5 this is going to require an additional research
6 program of some sort or could one introduce some
7 requirements that would take care of it without having
8 to do model studies or something of that sort?

9 DOCTOR CATTON: I don't believe a research
10 program is needed, but I do believe somebody has to
11 sit down and do the kinds of calculations that tell
12 you where your problems are going to be. Where is the
13 hydrogen going to move through an AP-600 containment
14 building. I think many of the tools are available.
15 They just haven't been exercised.

16 DOCTOR SHEWMON: And where could a large
17 enough volume of it accumulate so that a detonation is
18 of concern to you.

19 DOCTOR CATTON: That's right.

20 DOCTOR SHEWMON: The small ones really
21 don't matter.

22 DOCTOR CATTON: That's right, and there
23 have been several arguments against igniters both by
24 Tony Oppenheim in the Academy report that was some
25 time ago and also Karwot in Germany has argued against

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1 use of igniters. And part of it has to do with not
2 knowing quite how to place them.

3 And the other thing is, I don't really
4 see--

5 COMMISSIONER ROGERS: What would they
6 substitute for igniters? Recombiners or nothing?

7 DOCTOR CATTON: I think that you -- well,
8 the Siemens-Framatome containment was much like AP-600
9 with the passive cooling and the thin steel shell.
10 What they did is they somehow incorporated concrete to
11 take a detonation, so the problem goes away. It's the
12 thin shell that gets you into trouble.

13 There are other things you could do too.
14 You could design your containment so that you would
15 direct where the hydrogen is going to go so you know
16 where it is.

17 Now at the outset EPRI said that they had
18 in their requirements document some sort of statements
19 about this, but I've not been able to find them and
20 the EPRI people said that they would get me the part
21 of their document that addresses this. But this means
22 sloped ceilings, high point vents and all of this sort
23 of thing so that the hydrogen gets to a place where
24 you put your igniter. You don't just place igniters.

25 COMMISSIONER ROGERS: What's your opinion

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1 as to how this is being dealt with? I guess that's
2 really going to take place today in your meeting with
3 the staff.

4 DOCTOR CATTON: We're going to do that
5 today.

6 COMMISSIONER ROGERS: All right. Very
7 good. Good luck.

8 DOCTOR CATTON: We'll get there.

9 COMMISSIONER ROGERS: The core debris
10 coolability concern that you mentioned, there's a
11 cavity floor area greater than 200th of a square meter
12 per megawatt thermal. There's a SECY that says that
13 the staff neither supports nor disputes that
14 criterion. Do you have any suggestions as to how to
15 close out this matter?

16 DOCTOR CATTON: Well, a long time ago we
17 did some simple calculations I guess to help us put
18 together the containment report that we wrote for you
19 and at that time we did calculations and came to the
20 conclusion that a number like .04 you could be sure
21 that you could quench it. I just feel that backing
22 away from requiring that it be quenched is in the
23 wrong direction. If it's not quenched, you've still
24 got to deal with it.

25 DOCTOR CARROLL: Why don't you explain

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1 what "backing away" means?

2 DOCTOR CATTON: Oh. In 90-016, the
3 statement in there is that it be quenched, and
4 quenched means I now have particulate not molten core.

5 Presently the position is that it just be
6 coolable or, I'm not quite sure of the wording, but
7 it's a significant step back from a requirement that
8 you deal with it. Coolable, without stating time,
9 could be days, months, weeks. Who knows? I think it
10 should be a little bit more specific.

11 COMMISSIONER REMICK: Is it a question of
12 coolable or quenchable?

13 DOCTOR CATTON: Well, quenching has a
14 particular meaning.

15 DOCTOR SHEWMON: Well, tell us what it is.
16 To a metallurgist they sound very similar.

17 DOCTOR CATTON: Well, to me quenching
18 means that it's solid.

19 DOCTOR SHEWMON: Yes. And coolable means?

20 DOCTOR CATTON: Coolable means you're
21 slowly taking the energy out of it. It may be days
22 before it's solidified. It's liquid.

23 DOCTOR WILKINS: It could even be boiling.

24 DOCTOR CATTON: Well, I hope it won't be.

25 DOCTOR WILKINS: Well, in the general

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1 sense of coolability.

2 DOCTOR CATTON: Yes. I just don't think
3 enough attention has been paid to it, at least not
4 what I've seen. We ought to take a shot at trying to
5 do some calculations to figure out just what we do
6 need.

7 COMMISSIONER ROGERS: I'm not familiar
8 with the history of this. I know it's got a long
9 history, but I'm sure some calculations have been done
10 of some sort. Is it a major problem that needs to be
11 addressed?

12 DOCTOR CATTON: Well, I'm not sure. Sure
13 calculations have been done. On the one side
14 assumptions of very high heat transfer coefficients
15 are made and you arrive at the .02, but it turns out
16 the calculations that led to .02 don't stand a test of
17 scrutiny. I think the approach that was taken to
18 arrive at the .02 was a sensible one, but they ought
19 to use more reasonable numbers in their calculations
20 and just see what you get and at least I haven't seen
21 that that's been done.

22 Now there was the experimental program
23 that was at Argonne that was supposed to address these
24 questions, but they had a lot of difficulty with the
25 experimental program. My feeling is everybody kind of

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1 stood back to wait and see what would happen, so now
2 we're at the same position we were when they started
3 the test program which was several years ago.

4 I think somewhere we have to engineer.

5 COMMISSIONER REMICK: I had a question in
6 this area too. I think what you're referring to on
7 087, the staff says "provide a means to flood the
8 reactor cavity to assist in the cooling process" and
9 that's what you think is not strong enough.

10 DOCTOR CATTON: That's right.

11 COMMISSIONER REMICK: You think that's
12 backing down?

13 DOCTOR CATTON: Yes, from the word
14 "quench."

15 DOCTOR CARROLL: I guess the perspective
16 on all of this, however, is that people have proceeded
17 with design of their plants on the basis --

18 DOCTOR CATTON: Of the .02.

19 DOCTOR CARROLL: It's late in the day to
20 make any changes.

21 COMMISSIONER ROGERS: Yes. Well, for some
22 designs, yes, very difficult.

23 COMMISSIONER REMICK: But you're not
24 necessarily hanging your hat on .04?

25 DOCTOR CATTON: No. No, it's just that

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1 doing a sort of conservative analysis you can get to
2 a number of .04, of that order rather than the .02.
3 It's probably somewhere in between.

4 DOCTOR SHEWMON: One thing that bothers me
5 about any of those calculations is that they assume
6 the stuff is going to stay liquid when it drops out
7 and cools and runs out in a nice uniform distribution,
8 which I think is utterly baseless. I'm waiting to --

9 COMMISSIONER ROGERS: You want it to stack
10 up like that. You think it will stack up?

11 DOCTOR SHEWMON: It will stack up like
12 that.

13 DOCTOR CATTON: If you don't put in water,
14 it might. But if there's no water there, it probably
15 won't.

16 DOCTOR SHEWMON: There's a big difference
17 in temperature and radiation and it's not even all
18 liquid to begin with.

19 DOCTOR CATTON: But its thermal
20 diffusivity is low. This is probably not an argument
21 that they want to hear.

22 COMMISSIONER ROGERS: I can see why this
23 matter has not been settled yet.

24 DOCTOR CATTON: Not even among ourselves.

25 CHAIRMAN SELIN: We could just come back

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1 when you're finished.

2 COMMISSIONER ROGERS: Well, it looks like
3 it's something that's going to require continuing
4 attention and whether it ever will get settled is a
5 question in my mind.

6 There is one issue in your letter of
7 August 12th, 1992, that is of concern to me. You
8 stated that there appears to be no means by which the
9 ABWR ITAAC tier 1 requirements will ensure that the
10 ABWR components and systems can be expected to have
11 reliabilities which are consistent with those assumed
12 in the PRA for the ABWR and that in your opinion
13 appropriate reliability values for the PRA deserve
14 more study.

15 Is this still an open issue with you
16 folks, the reliabilities that are being assumed in the
17 PRAs?

18 MR. MICHELSON: Of course there were
19 several people involved in the response, in the
20 particular components.

21 A good case in point, though, are the
22 isolation valves on the containment where they have to
23 isolate potential ruptures in piping outside of
24 containment. The reliability numbers you use are not
25 those you experience on a day to day closure basis,

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1 since under that closure there's no real high
2 velocities and large differential pressures and things
3 of that sort challenging the valve.

4 However, under the pipe break case now
5 when you want to know the reliability, you have to go
6 an entirely different data set. That data set doesn't
7 exist. We haven't broken enough pipes yet and got
8 enough reliability data on valves closing under those
9 kinds of conditions to predict.

10 I think the staff is trying to work with
11 what tests we've done, and there's been a handful, to
12 see if we can come up with a guesstimate even of how
13 does reliability change. This is true of a lot of
14 components. They see a different condition under the
15 accident than you do under normal operation. You know
16 that the reliability is, intuitively at least, going
17 to be reduced. How much we don't know unless you do
18 more tests, so the PRA is put in question under those
19 circumstances.

20 That's why we missed the reactor water
21 coolant line breaks, because the way the PRA was done
22 they used normal closure reliabilities, 10^{-3} , 10^{-4} , and
23 of course two of those in series becomes a non-problem
24 if that's the way they really do it.

25 COMMISSIONER ROGERS: Well, is there any

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1 way of dealing with some of these questions short of
2 waiting for a lot of new data that may take forever to
3 get?

4 MR. MICHELSON: Well, conservatism of
5 design will help, but there's no sure way short of
6 having reliability numbers. The main thing is be sure
7 you remember that this is the real world and that you
8 use some judgement in looking at the PRA for those
9 particular kinds of problems.

10 COMMISSIONER ROGERS: Well, have you folks
11 been able to flag the classes of problems that you're
12 concerned about here?

13 MR. MICHELSON: I don't know if we've made
14 a -- well, we've made an attempt to identify all the
15 places where reliability will change because of the
16 event.

17 COMMISSIONER ROGERS: But in your
18 judgement the most serious deficiencies in the data--

19 MR. MICHELSON: Well, clearly the most
20 serious one is the case of the reactor water clean-
21 out. Closely associated is the HPCIS and RCICS and
22 other isolation arrangements from the standpoint that
23 they all represent a LOCA if you don't isolate,
24 because they are carrying reactor fluid under normal
25 operating circumstance and if they fail to isolate

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1 then you've got a LOCA outside of containment. The
2 PRA has to be analyzed for that case and you don't use
3 the reliability numbers under normal operation. It's
4 a totally different situation.

5 DOCTOR CARROLL: If my memory is any good,
6 when we first broached this with the GE PRA people in
7 the issue of motor-operated isolation valves. They
8 went to the standard tables and got 10^{-4} . We asked,
9 "Have you heard about the problems that are going on
10 trying to make motor-operated valves work under break
11 conditions?" The answer was, "No, it comes out of the
12 tables." So, that was what this --

13 MR. MICHELSON: That's what their whole
14 valve testing program under 89-10 is about.

15 DOCTOR CATTON: You should be fair to GE,
16 though. I think this is a problem with all PRA
17 practitioners. They use some sort of standard values
18 in spite of the fact that in that use the valve is 20
19 times less reliable.

20 MR. MICHELSON: So when you start putting
21 in perhaps more realistic predictions of reliability
22 you begin to identify new accidents which have
23 previously gone away and that's why we don't inspect,
24 for instance, stress corrosion cracking for reactor
25 water clean-out, because it doesn't show up in the

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1 PRA. So the inspectors tell us, "We don't have to
2 look at those systems because they don't pop out of
3 your PRA. You go to your big contributors that you do
4 get in the PRA," and this is a potential contributor
5 that gets overlooked because of the way the PRA is
6 done.

7 COMMISSIONER ROGERS: Yes. It seems that
8 this is a very serious kind of problem of over-
9 reliance on a PRA, I mean, if we're simply -- my
10 impression is that the staff is not all that convinced
11 of PRAs and that other experiential judgement has to
12 be introduced and not simply a bottom line number from
13 a PRA.

14 MR. MICHELSON: I believe the staff
15 understands this problem and I believe that they're
16 taking a look at it. We haven't yet heard what
17 they've come up with, at least I haven't.

18 COMMISSIONER ROGERS: I think I'll let
19 somebody else --

20 CHAIRMAN SELIN: Commissioner Curtiss?

21 COMMISSIONER CURTISS: I just have two
22 questions of a more general nature.

23 Since we last heard from you there's been
24 quite a bit of work or progress, depending on how you
25 look at it, on the ITAAC, working in particular with

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1 GE on those ITAAC. It's not a subject that you've
2 addressed in any of your recent letters in terms of
3 the general question of ITAAC and how the process is
4 evolving, but, if you could speak to where you think
5 we are with respect to ITAAC at this point, that would
6 be a helpful update.

7 DOCTOR CARROLL: Okay. I chair the ad hoc
8 group on ITAAC and DAC. I see very significant
9 progress being made. I think it's coming to closure.
10 It's been sort of a painful process, but I think we'll
11 have a decent set of DACs and ITAACs when the dust
12 finally settles here shortly.

13 MR. MICHELSON: But we actually haven't
14 seen them.

15 DOCTOR CARROLL: We haven't seen them.

16 MR. MICHELSON: So until we see them, it's
17 difficult to --

18 DOCTOR CARROLL: We do plan one more round
19 of looking at the final product.

20 COMMISSIONER CURTISS: Are you familiar
21 with the discussions that have taken place between GE
22 and the staff and you're up to date, say, for having
23 seen the actual ITAAC themselves?

24 DOCTOR CARROLL: I keep getting big stacks
25 of paper which I --

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1 COMMISSIONER CURTISS: Okay.

2 DOCTOR CARROLL: -- look through and I
3 assume I'm pretty much up to date. It looks like good
4 progress.

5 COMMISSIONER CURTISS: But at this point,
6 and based upon what you know, recognizing that you'll
7 need to see the ITAAC, you don't see any significant
8 potential hard spots on the horizon?

9 DOCTOR CARROLL: I do not.

10 COMMISSIONER CURTISS: Okay.

11 The second question is really a schedule
12 question. Somebody mentioned that the schedule is an
13 aggressive one. In view of the technical issues that
14 you've identified and the posture of those issues and
15 having seen the staff's most recent schedules and the
16 time allowed for ACRS review and the approach, is that
17 a schedule, albeit aggressive, that the ACRS can
18 support?

19 MR. MICHELSON: I'll address that for the
20 ABWR. I had a meeting with Murley last full committee
21 to discuss this and they show us, of course, that
22 about six weeks between the receipt of the final
23 safety evaluation requirement that we issue a report.
24 This would be, I think, keeping in mind that in that
25 period of time we must have full committee discussions

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1 and perhaps subcommittee discussions very likely,
2 that's a nonrealistic schedule if you were to see the
3 report for the first time on the first day of that six
4 week period. It simply is unrealistic.

5 The agreement is that we will see these as
6 soon as possible before that time. I have already
7 scheduled meetings for October and November of this
8 year just to look at the preliminary material.
9 Hopefully it's most of it. If it isn't, then it's
10 going to be difficult to meet the schedule. If it is
11 most of it and it's in reasonably final form, I think
12 we can meet the schedule.

13 COMMISSIONER CURTISS: Okay. Are there
14 things that you believe the staff ought to be doing in
15 addition to what they're doing now, providing early
16 copies and so forth, that would facilitate your
17 meeting the schedule?

18 MR. MICHELSON: There are a number of
19 areas we have not reviewed even for the first time.
20 We've pointed these out in our previous letters that
21 we haven't done that review yet. The staff says we're
22 going to start getting to those this summer.

23 COMMISSIONER CURTISS: Okay.

24 MR. MICHELSON: And presumably we'll get
25 them out of the way.

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1 COMMISSIONER CURTISS: Okay. Thank you.

2 DOCTOR WILKINS: Let me say, Commissioner,
3 that the Committee will do its damndest. But unless
4 we have the material, we can't review it. And that's
5 a truism. And what's more, the material has to be in
6 reasonably good shape and in reasonably final form.

7 Carl's subcommittee has had a number of
8 meetings in which he's addressed much earlier versions
9 of various drafts of things and in retrospect, but
10 only in retrospect, that has turned out to be a rather
11 inefficient way for the ACRS and its subcommittee to
12 operate.

13 CHAIRMAN SELIN: If I might just add to
14 that point, the issue isn't a question of willingness
15 to work hard -- I mean, that's taken for granted and
16 in fact it's shown in your previous efforts -- or even
17 willingness to look at incomplete documents. The
18 issue is to be able to schedule below the committee
19 level at the staff level between your supporting staff
20 and the Agency staff, what questions will be asked,
21 what kinds of things are needed, so that I mean
22 there's a lot of paper that has to flow. Things have
23 to be distributed and it's a big project. It's not
24 just a question of a few smart people listening to a
25 few other people and saying yes and no.

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1 We've been talking to Doctor Larkins with
2 considerable positive feedback about this is as big a
3 project as many of the other projects. There has to
4 be a project management system set up, documents in,
5 what kind of questions are needed, what kind of
6 supporting guys, what kind of test results you'll need
7 at different points so that you don't get hung-up
8 being pressed to give fast answers when in fact the
9 staffs haven't done the homework that you need to
10 support you.

11 Commissioner Curtiss?

12 COMMISSIONER CURTISS: No, I don't have
13 anything else.

14 MR. WYLIE: I might point out, though,
15 that we've principally been talking here about ABWR,
16 but the CE System 80 and passive plants, those
17 schedules -- you've seen the schedules -- they seem to
18 stack up on top of each other and all that material is
19 coming in about the same time, and so it's going to be
20 a lot of work. Doctor Wilkins just mentioned that
21 it's going to be important that we get that material
22 on time because many of us are on the same
23 subcommittees and need to look at all of these things.

24 CHAIRMAN SELIN: Of course.

25 Commissioner Remick?

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1 COMMISSIONER REMICK: Commissioner Rogers
2 covered the areas that I had questions on.

3 CHAIRMAN SELIN: Thank you.

4 Commissioner de Planque?

5 COMMISSIONER de PLANQUE: Nothing on this.

6 CHAIRMAN SELIN: I had sort of just a
7 general question. The Committee has been extremely
8 helpful over the last three years. You've actually
9 sent six different reports on the various policy
10 issues that have been involved in these reviews and
11 they've been very good general reports, but they have
12 mentioned that there are some areas of concern.

13 We are to hear from our staff in the not
14 too distant future on the policy issues for the
15 evolutionary and the passive plants and, if there's
16 anything that you care to add to your comments to
17 date, that would be helpful. I don't know that there
18 are any gaps in these points, but we've gotten really
19 quite a long and thorough report from the staff on
20 these some 14 policy issues. We're to be briefed on
21 them. You've followed them individually and
22 collectively very closely and so, if there are any
23 updates or any more specific points on the concerns
24 either now or after the meeting, I mean in the next
25 several weeks, that would be useful.

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1 DOCTOR SHEWMON: Fine. We'll take it
2 under advisement. I don't think of any, but we'll
3 check.

4 CHAIRMAN SELIN: Okay. Thank you.

5 Doctor Shewmon?

6 DOCTOR SHEWMON: Okay. Well, we all agree
7 that the use of computers holds exceptional promise
8 for instrumentation control. We also have found that
9 the review and certification involves exceptional
10 challenges and requires new approaches.

11 Doctor Lewis will handle this
12 presentation.

13 DOCTOR LEWIS: Thanks, Paul.

14 I'll try to carry the ball a little bit on
15 this one. Let me just remind you of the recent
16 history and bring us up to date and I'll try, to the
17 extent that I'm able, to represent the Committee's
18 views, but I will try to let you know when I'm
19 deviating from that track.

20 As you recall, we wrote you a letter back
21 last September, I believe, and we were somewhat
22 exercised about what appeared to be an inflexible
23 staff policy that required that whenever an analog
24 system was replaced by a digital system it be backed
25 up by a hard-wired system all the way front to end,

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1 and we were told by staff that what they really meant
2 was an analog system. That sort of annoyed a number
3 of us because in most respects digital systems can be
4 made much more reliable than analog systems. They can
5 do self-testing, all those good things. We've gone
6 through them.

7 And we wrote you a letter last September
8 in which we said that there was an over-emphasis on
9 the issue of common mode failures in connection with
10 digital systems, although the common mode failures can
11 occur with both digital and analog systems and indeed
12 with hardware, valves, you name it. Wherever you have
13 anything that is common to several trains it can lead
14 to common mode failure, so we recommended last
15 September that this position of the staff be relaxed
16 but we had some additional recommendations which I
17 just looked up.

18 We recommended that the staff augment its
19 own capability to deal with this kind of question
20 because they were treating -- the term that was used
21 was they were treating the digital systems as a
22 disease rather than a cure, and that they broaden
23 their interaction with the rest of the world that's
24 been involved with this business and not be confused
25 about the difference between the kind of relatively

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1 simple digital systems we're dealing with in nuclear
2 power plants and the scare stories we hear from SDI or
3 these other really far larger systems.

4 The staff did then relax its position on
5 the hard wiring connection with this, but only in a
6 way that remains to be seen in practice. That is,
7 they said, yes, they won't be doctrinaire about it.
8 They'll consider alternatives, even the possibility of
9 digital systems which are backed up by other digital
10 systems. How serious this was we simply have to see.
11 We have good will about it.

12 Then the next episode was not a Committee
13 activity, but I can mention it because it was
14 subsequently taken up by the Committee. We wrote a
15 letter. A few of us signed a letter to you saying,
16 "Gee, this is a big subject. It's really quite new to
17 the Agency and it would be good to get a broad outside
18 look at the general question of how one does regulate
19 and license digital systems," and three of us
20 recommended that you hire the Academy, National
21 Academy of Sciences and Engineering, to do this for
22 you because some of us have had experience with such
23 things. They tend to be broad-based. They will not
24 tell you how to do your job tomorrow, which is what
25 you really want to know, but they can give you some

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1 perspective about what experts -- who serve unpaid,
2 incidentally -- from the outside world would tell you
3 about it. We recommended that.

4 The staff countered by setting up a
5 workshop, which I'll get to in a moment, and, Chairman
6 Selin, you wrote the three of us saying, "Well, thank
7 you for your suggestion and the staff is moving along
8 those lines through their workshop." I can mention
9 that because in the later letter to which I'm now
10 coming the Committee, to my surprise I have to say,
11 picked up on the Academy proposal and made it a
12 Committee position.

13 In the later letter, then, we dealt with
14 a lot of the specific issues that have to do with
15 digital systems. They are complicated. They are
16 different. But they don't wear out, they can self-
17 test. I won't go through all those things for you
18 because we've been through them before and they're all
19 in the letter, but we concluded that a fundamental
20 problem was that the staff was applying to the
21 regulation of these digital systems, which do have
22 different accident modes and different
23 vulnerabilities, applying methods that were developed
24 earlier in connection with the analog systems and,
25 even worse, with the mechanical systems.

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1 The reactors we're dealing with in the
2 current group are reactors that were designed in the
3 '60s and '70s. They're really rather old designs.
4 That was a long time ago, even by old folks'
5 standards, so it was improper the Committee said to
6 apply the old habits which have to do with redundancy,
7 diversity blindly with respect -- obviously they have
8 a role, but it was improper to apply them blindly to
9 the digital systems. And again, the Committee picked
10 up on the proposal that the Academy be hired to take
11 a broad look at the subject and ask -- I hate the word
12 "paradigm," but I've used it the last couple of days
13 a couple of times -- try to think through what
14 paradigm should be used in the regulation of these
15 really different systems with which neither the
16 industry nor the staff is really familiar

17 Just as the reactor systems we're looking
18 at now were designed in the '60s and '70s, much of
19 your staff was designed in the '60s and '70s. I ran
20 into a distinguished physicist, really very
21 distinguished, who was an old friend of mine at an
22 airport the other day and he said, "You know, Hal, I'm
23 thinking maybe I ought to get a computer." Such
24 people still exist even among my friends and there's
25 a certain amount of that in your staff.

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1 In any case, we did write this letter. It
2 was full of recommendations but it included this
3 observation that you can't use the old methods. In
4 particular, you cannot use even the PRA ideas of a
5 failure rate per unit time or probability of failure
6 per application for the digital systems, especially
7 when you deal with software.

8 Software systems do have a tendency to
9 work the same each time if they're given the same
10 input and they may have -- there is a certain
11 probability that they will have a defect in them to
12 begin with, no question about that, a bug probability,
13 but once they're in operation the probability is not
14 so much that they will suddenly malfunction like a
15 valve.

16 The probability you're really dealing with
17 is what was the chance that there is a bug in there
18 and what is the chance that the bug will be discovered
19 by some unexpected input into the system that
20 exercises an element of the logic that has not
21 previously been exercised. Those are different things
22 and they're unfamiliar things, so the Committee wrote
23 a letter which really ended up saying, "Gee, you ought
24 to get some people, real computer scientists" -- I
25 wouldn't turn the regulation of nuclear power over to

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1 computer scientists, of course, but -- "an infusion of
2 that kind of thinking which is fairly common out in
3 the world because it applies to all large process
4 control systems, large safety-related systems."

5 NRC is not at the bottom of the heap, you
6 know. NASA doesn't do all that well either on these
7 things. The regulators of aviation don't do all that
8 well either, because they have the same problem that
9 all regulatory agencies have that you build up a lore
10 based on earlier things and it's kind of hard to
11 change. So, we made that recommendation and I have
12 lots of details I can give you.

13 In any case, two weeks ago we received an
14 answer from the EDO to that March letter and I have
15 the answer in front of me and I was pleased to hear
16 you, Doctor Selin, say that we've had a major impact
17 on the staff in this regard. I don't find that in the
18 letter.

19 Let me read you what his letter says. His
20 letter says, "You raise the question" -- I'll
21 paraphrase one paragraph -- "of whether the staff has
22 developed a coherent and effective plan. Moreover,
23 you express concern the staff may not have sufficient
24 expertise to accomplish this task in view of the
25 nature of the issues involved."

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1 Then I'll read you verbatim. "In response
2 to these questions, I continue to believe that the
3 staff has developed an appropriate regulatory approach
4 to digital systems as explained in numerous meetings
5 and correspondence with the ACRS. Further, the staff
6 believes that its approach is based on expert input
7 from the staff itself, its consultants in both the
8 computer and nuclear power industry including
9 international regulatory and utility experience."

10 So, you know, that's in a sense a rather
11 unambiguous rejection of the letter. He has a right
12 to do it, of course, but we have a right to differ
13 with that judgement. The Committee has not had a
14 chance to develop a collegial position with respect to
15 his response to our response, you know, the next step.
16 That has not happened among the Committee members so
17 I can't speak communally on that one, but I can say I
18 am impressed with it.

19 So why don't we turn to the next subject,
20 which is the famous workshop, because in a certain
21 sense we have had an impact in the sense that the
22 staff probably would not have organized this workshop
23 without a certain amount of pressure from you and from
24 us. But they have indeed organized it and it's
25 important to ask whether this workshop is in some

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1 sense a surrogate for the Academy study that others
2 have recommended, whether it's going to do the kind of
3 same thing.

4 And now I'll give you my own views because
5 the Committee doesn't have a position on that subject.

6 My own view is that it's not going to for
7 a couple of reasons. One is it's done the way,
8 forgive me, NRC often does things. That is to say,
9 it's organized by NRC for NRC to help NRC do its job.
10 That's what matters to the NRC staff and rightly so.
11 They have a real job to do.

12 They tend to organize it, and this is
13 organized that way, in such a way as to say "What
14 we're doing is really what we need to do and we need
15 help in doing it better." That is, it doesn't
16 contemplate another possible philosophical approach
17 and, in fact, it's a little bit worse than that in
18 this case because not only is the workshop organized
19 with a day and a half of shotgun briefings by a wide
20 variety of people, many of them quite good -- I can
21 provide an enormous list of people who are just as
22 good who were not invited, but you can't invite
23 everybody so that doesn't offend me at all -- and then
24 a half day of a panel discussion devoted to "what else
25 should we be doing?"

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1 That isn't my definition of a workshop.
2 That is to say, it's peer input followed by panel
3 discussion, chaired by the NRC incidentally, which
4 will ask for input in doing the job. I don't see any
5 room in there for introduction of other ways of
6 organizing the regulation of digital systems in
7 nuclear power plants.

8 Now there's been one recent change and
9 that is that we were asked, with a very short time
10 frame, to comment on the agenda. And I blush to admit
11 that I typed a response into my computer at home,
12 suffered a computer failure while I was doing it and
13 so it was erased and I was so frustrated and angry
14 that I pounded my fist and sulked. Now, it wasn't
15 really computer failure. It was that it was such a
16 long response that I ran out of my allocated time on
17 our internal bulletin board system and was not able to
18 do it.

19 In any case, we did send a response.

20 DOCTOR CARROLL: There is a rule that
21 says, "save often."

22 DOCTOR LEWIS: Yes, and I know the rule.
23 I know it.

24 COMMISSIONER de PLANQUE: Human failure
25 again.

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1 DOCTOR LEWIS: In fact, I keep thinking of
2 implementing that rule for my word processor but I
3 haven't gotten around to it yet. I'm as bad as the
4 next guy.

5 In any case, we did make a number of
6 recommendations. Mostly they had to do with getting
7 more outside input, if that's not a contradiction in
8 terms, and, "gee, at least invite somebody from our
9 crowd," so they have indeed invited Ernest to come and
10 give a talk at their workshop. I have a conflict, but
11 I will try to come and just listen to what he says to
12 exercise some quality control on him.

13 But if I were to make a bet, that is if
14 you were to lay a buck on the table and say "Is this
15 going to do what you guys wanted in your letter?" I
16 would say "No, it's not organized to do that."

17 So the Committee has not -- let me put it
18 as precisely as I can -- the Committee has not
19 retracted its recommendation that you go for an
20 Academy workshop and I personally think it's probably
21 the best way to go. It's not Heaven, you know. It
22 sometimes doesn't work, but I have known it to work
23 and you need a fresh infusion.

24 Now let me talk about fresh infusion and
25 then I'll shut up and let you attack me. The staff

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1 claims it has interacted with many people and indeed
2 it has interacted with many people, but in the format
3 that the workshop has. That is to say, they've had
4 talks from many people. I've spoken to some of the
5 people who've interacted with the staff. It says it's
6 had a lot of international experience. If you look at
7 what really happened, there is a staff member who has
8 attended most international meetings. What his impact
9 on the rest of the staff has been, I do not know.
10 He's not a bad guy, but that's still not extensive
11 international interaction.

12 Now the staff also is going to hedge the
13 workshop in another way. They've just announced that
14 they're going to try to draft a standard review plan
15 for the regulation of digital systems and use that as
16 a focus for the workshop, and that's further down the
17 track of "Here's what we're going to do. Tell us how
18 to sharpen it up." And it's not, again, a format
19 which admits of new thinking.

20 I try to tell people that, you know, there
21 are some things you can't do. You can't turn country
22 music into Bach one note at a time. And my friends
23 tell me I don't know enough about country music to
24 make that kind of comment, but it is true. There are
25 some things you can't pick at, and yet, for people who

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1 are trying to do a job, they don't want in a sense to
2 change the way they're doing it. They just want help
3 doing it. They want people to pick at it, so there's
4 a diversity of wishes and expectations and that's why
5 I personally think you ought to go to the outside and
6 get a fresh view. It may come to nothing. It doesn't
7 cost much anyway.

8 Now I'll shut-up.

9 CHAIRMAN SELIN: Doctor Wilkins, would
10 you-- Doctor Lewis pointed out that after a certain
11 point he was speaking on his own and it's always been
12 useful for us to have --

13 DOCTOR LEWIS: Yes, that's right.

14 CHAIRMAN SELIN: -- a range of views and,
15 since you've accepted the invitation, perhaps you
16 might contribute something.

17 DOCTOR WILKINS: Yes. I would again
18 repeat that this is not a collegial statement at all.
19 It's a statement of my own views.

20 I think the Academy workshop would be
21 better than what we've got. I don't see it as a
22 panacea. In the final analysis, you have to change
23 the hearts and minds of the staff and that is going to
24 be a very difficult, very difficult operation.

25 What we're doing here today and what we

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1 have done in our letters is to start working on you
2 folks, trying to convince the Commission itself that
3 there is an issue, that there is a problem which
4 requires attention either directly from you or through
5 the EDO or through the office directors.

6 I'm going to appear before this group. It
7 isn't quite fair to say that I'm going to give them a
8 talk.

9 DOCTOR LEWIS: Oh, I thought you were.

10 DOCTOR WILKINS: It's more a ceremonial
11 kind of thing. One of you is supposed to address --
12 I've forgotten. Brian Sheron told me which one, but
13 I've now forgotten. One of you is supposed to -- I
14 guess it won't be you, Commissioner Curtiss.

15 COMMISSIONER CURTISS: That's for sure.

16 DOCTOR LEWIS: You may seize the occasion
17 to give a talk.

18 DOCTOR WILKINS: And then Eric Beckjord
19 will welcome the group and sort of give a keynote, and
20 then I'm supposed to have the opportunity to say what
21 I want to about the ACRS position and I intend to take
22 advantage of that opportunity to do precisely that.
23 But it's not a technical talk. It will not be a talk
24 which says "this is what you ought to do" or "this is
25 what I think you ought to do."

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1 CHAIRMAN SELIN: Doctor Shewmon?

2 I'm sorry --

3 DOCTOR WILKINS: Yes, I think that's
4 really about all I have.

5 DOCTOR LEWIS: I do want to add one thing
6 I should have said. In the context of changing the
7 hearts and minds, we all know the slogans that went
8 with that in some other circumstances.

9 The staff says that we recommended that
10 they augment their capabilities. The staff does say
11 that it's trying vigorously to hire people with the
12 kinds of skills involved.

13 I had to talk at some conference in
14 Florida a few months ago where there was an NRC
15 recruiting brochure lying on a table. I picked it up
16 and looked through it. There was no mention of the
17 subject in the very long list of requirements. Partly
18 that happens because these things are written up by
19 the office directors and, again, when the office
20 directors when they lose a mechanical engineer they
21 want a mechanical engineer to replace them. They
22 don't want to change their work force balance.

23 So, even though they say they're trying
24 hard, I don't think they are.

25 DOCTOR SHEWMON: We had a meeting with Tom

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1 Murley yesterday and this was one of the topics which
2 we discussed. One of the things which he brought up
3 was that he had a problem he wanted some solutions to
4 now or as soon as he could get them and, if he gave it
5 to the Academy, his experience was this was a hiatus
6 of a year or two. I think one could say in his
7 defense or as a comment that possibly both could be
8 done in parallel because there's what are you going to
9 do this year and then what are you going to do
10 different next year and --

11 CHAIRMAN SELIN: I'm glad you said that
12 because I'd like to respond to a couple of things that
13 you said, Doctor Lewis.

14 First of all, I have the unenviable task
15 of trying to convince the Committee it's been a lot
16 more effective than it gives itself credit for. When
17 we started on this it was analog versus digital, not
18 one mode of computation versus a second or whether it
19 was computing or instrumentation. I mean, there
20 really has been a considerable amount of
21 sophistication that's come to the reviews.

22 Second, substantively the staff positions
23 have changed enormously in what they --

24 DOCTOR LEWIS: Oh, yes.

25 CHAIRMAN SELIN: -- have required of the ,

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1 bidder and that's that they want to take a look at the
2 standard review plan and discuss this with some
3 experts. I personally consider that a sign of
4 strength and not weakness, because hearts and minds
5 are fine but it has to get down to how we're going to
6 do business and if --

7 DOCTOR LEWIS: The devil is in the
8 details, they say.

9 CHAIRMAN SELIN: Yes, sir. And when you
10 pick up a recruiting document that doesn't show what
11 we say we're doing, then that's a real sign that we're
12 not doing what we say. Similarly, the actual things
13 that have been required of GE have changed enormously
14 since you've taken a strong interest in this point.

15 DOCTOR LEWIS: I know that.

16 CHAIRMAN SELIN: The third point is sort
17 of a general point, but that is we don't design the
18 safety systems. We don't design the computer systems.
19 We, the Agency, not the Commission, react to what
20 comes in. Now we can very much affect what will come
21 in by giving indications of what will be considered
22 welcome or not welcome, but we do have to basically be
23 following the designs and a lot of the GE design is
24 based on their experience overseas so there is a
25 specific set of designs to react to.

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1 But of the more important points, other
2 than the fact that I sort of -- what did Voltaire say?
3 He didn't exactly say this, but I agree with what
4 you're saying but I disagree with your right to say
5 it. Or more precisely, I agree with the bottom line
6 of what you're --

7 DOCTOR LEWIS: I like that. I'll use that
8 some day.

9 CHAIRMAN SELIN: Well, Voltaire can be
10 turned around also.

11 But the key thing is that I don't consider
12 the staff has an affliction for analog systems or many
13 of the other things you said, but I do agree with the
14 conclusions even though I might have reached them
15 through a different approach.

16 And myself, I'm particularly sympathetic
17 to the idea that if we always ask short-term questions
18 we'll always get short-term answers and we never will
19 get the chance to do the longer-term pieces. It's
20 clear in retrospect we should have done something like
21 the Academy study a couple of years ago so that some
22 of the results would be in hand now to handle the
23 first set of applications, but it's not as if they're
24 going to come and go and then we're going to be gone.

25 I think Doctor Shewmon's remark is really

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1 well taken. We do need to look at the influence on
2 the standard review plans, on the way we work with the
3 applications in hand, but we need to be in a better
4 position than we were a couple years ago for that
5 which will follow.

6 So I really don't think you should be so
7 pessimistic either about the impact you've had on the
8 staff or where they're going. The changes have been,
9 by our glacial standards, pretty spectacular in the
10 last couple years in this particular area and I think
11 you should -- you might not like the results, but you
12 ought to take a lot of credit for the direction in
13 which they're moving.

14 Commissioner Rogers?

15 DOCTOR LEWIS: I'm not sure I would use
16 the word "pessimistic." I'm not sure I should have
17 any impact on the staff.

18 CHAIRMAN SELIN: Well, Doctor Lewis, I
19 assure you that you do. You don't have to worry about
20 that.

21 COMMISSIONER ROGERS: Well, I just think
22 that you have had a great impact on not only the staff
23 but our own thinking on things. I think that some of
24 the points you just made, Hal, I think are extremely
25 important with respect to a totally different view of

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1 how one sets standards for digital instrumentation and
2 control versus mechanical systems, and I haven't heard
3 that point of view expressed before in writing
4 anywhere and I think that that's very important.

5 I think that your suggestion that perhaps
6 one has to take a new look at the front end rather
7 than try to fix things up as you go along is probably
8 not so different from a point of view that I think the
9 Commission has been adopting in several ways in
10 totally different other contexts of looking before we
11 put our thoughts down on paper to what the problems
12 are as the rest of the world sees them. In some cases
13 this has nothing to do with technical matters, really,
14 public perceptions, but here's one that's a technical
15 matter that perhaps we really ought to be looking at
16 the front end before we start to try to fix things up
17 as we go along.

18 So I personally am very sympathetic to
19 your point of view, however the staff does have -- as
20 the Chairman has said, the staff really has to respond
21 to what comes to us from the vendors. Those are the
22 products that we have to look at. We don't design
23 things. We don't suggest to them how they ought to
24 design. We have to look at them from the standpoint
25 of their safety, but how we do that perhaps should be

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1 very different from how we've done it before and I
2 think that is an important, very important observation
3 that I personally am quite receptive to.

4 So I think that not only have you had an
5 impact, but I think the articulation of these ideas
6 has sharpened up as you have expressed your
7 frustrations with the lack of what you see to be
8 progress. There's very good reason for that. I think
9 things haven't been so clear as they might have been
10 as to how to proceed and I would say, had you made the
11 statement that you just made now with respect to
12 suggestions say maybe four years ago, it would have
13 been really very, very, very important. It's still
14 important, but I think your perception of what has to
15 be said has probably changed in the last four years
16 too.

17 DOCTOR LEWIS: Of course.

18 COMMISSIONER ROGERS: So I think that
19 we're working on something here. Our concern in many
20 ways I think is that we not be disheartened by a lack
21 of obvious progress. I personally feel that the staff
22 has come a long way in its thinking about computers,
23 the use of computers in this Agency and how to deal
24 with digital systems. They've come a very long way.
25 Now it may not be where we have to be, but I've seen

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1 enormous progress in the last four or five years here.

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And it has been somewhat frustrating sometimes because some of us feel that it should have gone faster, but we have I think a very properly controlled response to situations that maybe is a little bit slow but in the long run may not be too bad. I hope that we can speed it up a little bit, so I think what you've said today in my mind is very important.

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I do think that our staff has improved its skills and knowledge and certainly commitment to learning and doing more. They've been somewhat stung, I think, by criticism. They're proud people and they should be proud and they want to do a better job.

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I do know that there have been very strenuous efforts to hire some top-notch people in this area and we have struck out. It is not -- it may not have appeared in that recruiting brochure, but I do know that we have tried very hard and it has not been an easy task to complete successfully. We would like very much to be able to bring in some top-notch people maybe for a year or two to really help us and that has been very difficult to achieve. Any help that ACRS can give us I think would be welcomed here.

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1 DOCTOR LEWIS: Don't misunderstand me.
2 There are some very bright young people on the staff.
3 There are. That isn't where the problem is.

4 COMMISSIONER ROGERS: Yes.

5 Well, let me ask just one little nitty-
6 gritty question rather than these rather broad-range
7 ones, and that has to do with this question of your
8 views on the necessity of safety grade displays and
9 controls in the -- independent set of safety grade
10 displays and controls in the control room. You've
11 commented on this a number of times and you've
12 indicated that other arrangements than the independent
13 displays and controls in the control room might be
14 acceptable and I wonder if you could say a little bit
15 more about what you might have in mind there.

16 DOCTOR LEWIS: Well, you understand that's
17 a Committee letter you're reading from, so I'm
18 forbidden to interpret Committee letters.

19 DOCTOR CARROLL: Which letter do you mean,
20 by the way?

21 COMMISSIONER ROGERS: Let's see. It's on
22 page 81 of the materials that I have here.

23 DOCTOR LEWIS: You understand that was a
24 response to a very hard and fast staff position which
25 has been relaxed or allegedly relaxed.

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1 DOCTOR SHEWMON: Do you think your 81 has
2 any resemblance to our 81?

3 COMMISSIONER ROGERS: Yes, I think it
4 does. If it's written in hand, it probably -- at the
5 bottom. It's a September 16th, 1992 letter.

6 DOCTOR WILKINS: I see the language that
7 you were reading.

8 COMMISSIONER ROGERS: Yes. "The staff
9 proposed an independent set of safety grade displays
10 and controls in the main control room. We believe
11 that other arrangements might be shown to be
12 acceptable." And I'm just curious as to what you
13 think those other arrangements might be. Anybody can
14 answer. You don't have to interpret.

15 DOCTOR LEWIS: I can tell you what I
16 think. I don't remember that particular sentence.
17 It's a rather weak sentence and that's unusual for an
18 ACRS letter.

19 But the staff had a particular thing in
20 mind which was an analog system, hard-wired, no
21 multiplexing, no radial links, no nothing. In many,
22 many cases radial links are more reliable than wires.
23 Infrared links are more reliable than radial links in
24 some cases. Multiplex systems which contain error
25 correction can be more reliable than hard wires.

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1 People sometimes joke about computers backing up
2 computers, but sometimes it's the best way to go.

3 When Tom Murley was with us yesterday I
4 quoted an unnamed admiral I once knew who chaired a
5 committee I was on who was a very anti-technology
6 admiral, unlike of course most admirals, and he came
7 away saying about some particular system, he said,
8 "You know, it's amazing these things never sleep."
9 They don't and they don't suffer from human error once
10 they're working.

11 There are lots of ways to skin and cat and
12 the staff was on a particular route and I think all
13 that sentence really meant to say was that there are
14 other ways of doing it.

15 COMMISSIONER ROGERS: You think that the
16 thinking since 1992 has changed, the staff's thinking
17 on this?

18 DOCTOR LEWIS: That the staff thinking
19 has? Oh, yes.

20 DOCTOR CARROLL: This really was the hard-
21 wired backup issue.

22 DOCTOR LEWIS: That's right. Their
23 thinking has changed.

24 COMMISSIONER ROGERS: Fine. That's fine.

25 DOCTOR LEWIS: I don't care about their

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1 thinking. Their position has changed.

2 COMMISSIONER REMICK: In your discussion
3 how the workshop and possibility of a National Academy
4 study, no mention was made at the workshop that ACRS
5 itself held last fall, I believe, and to what extent
6 does that complement, supplement staff's workshop that
7 is proposed? You had a large number of people come in
8 from different countries and --

9 DOCTOR LEWIS: I'm sorry, I'm missing the
10 question.

11 COMMISSIONER REMICK: Well, I want to know
12 to what extent the workshop that you folks conducted
13 complements what the staff would be doing.

14 DOCTOR LEWIS: Oh, I think it would be
15 quite different. The staff is asking for answers to
16 short-term questions. They're important. I'll tell
17 you one anecdote. One of our distinguished
18 consultants, a guy named Bill Kerr, whom you may
19 remember, went to a meeting in Florida a couple of
20 weeks ago, I wasn't able to go to it, and came back
21 with a story of somebody who wanted to install a
22 particular computerized system on his reactor that had
23 been installed on a half dozen other reactors
24 successfully before this. It was not an innovative
25 activity. It was Eagle-21. The NRC made him redo the

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1 electromagnetic interference study he had previously
2 done because that happened to be the preoccupation of
3 the week, electromagnetic interference, without any
4 guidance or philosophy. We've been complaining about
5 that for a long time. They did. It only cost them
6 \$300,000.00, but it's down the drain and it was
7 unnecessary. That's today's problems.

8 COMMISSIONER REMICK: But the point I was
9 trying to get at, and I apparently didn't make it
10 clear, but your workshop was a broader based --

11 MR. CARROLL: Well, it wasn't a workshop
12 specifically, it was just one of a series of
13 subcommittee meetings that happened to call in a lot
14 of international expertise.

15 COMMISSIONER REMICK: Okay. But that was
16 a much broader look, not just necessarily at today's
17 problems. Is that correct? Wouldn't the staff
18 benefit from that information that you accumulated at
19 that time?

20 DOCTOR LEWIS: Are we talking about the
21 proposed Academy workshop?

22 COMMISSIONER REMICK: I'm talking about a
23 workshop six months ago, the meetings, the
24 subcommittee meeting that the ACRS held back last
25 fall.

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1 DOCTOR LEWIS: Oh, we've had a dozen
2 subcommittee meetings on that --

3 COMMISSIONER REMICK: No, this is one that
4 you invited in a lot of people from foreign countries.

5 DOCTOR LEWIS: Oh, yes.

6 COMMISSIONER REMICK: And that was a
7 broader look at some of these issues. Would not the
8 staff have benefitted from that information to give
9 them a broader perspective?

10 DOCTOR LEWIS: Oh, they were there. They
11 were there.

12 COMMISSIONER REMICK: Yes. So, I'm not
13 sure that their workshop should be of that type since
14 they've already participated in one that you know.

15 DOCTOR LEWIS: No, it should not be the
16 type of our subcommittee meeting. No, no, no. The
17 thing that I would like, and I guess the Committee
18 since we approved it, would be a fairly open gauged
19 workshop. You know how things are done with the
20 Academy. I've spoken to the person who was current
21 chairman of the -- oh, God -- computers and
22 telecommunications board at the Academy. He's a nice
23 guy, Professor of Computer Science at Virginia. His
24 wife has just been appointed to a high -- DDR&E at the
25 Pentagon, as a matter of fact. He would love to run

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1 a workshop for NRC. He knows all the good computer
2 science. He knows what the issues are. He could
3 organize it with a viewpoint that is different from
4 the staff viewpoint.

5 I agree with Doctor Selin that you really
6 need answers to both short-term and long-term
7 questions and if you never address long-term questions
8 because it's too late, you never will.

9 COMMISSIONER REMICK: No, I agree. My
10 point is I thought that your subcommittee meeting was
11 about the longer term problems.

12 DOCTOR LEWIS: Well, the main things that
13 came out of our subcommittee meetings in brief were
14 that the difference between the smaller safety-related
15 systems in nuclear power plants and the large systems
16 on which people have scare stories, the fact that the
17 smaller systems or programs that one uses in nuclear
18 power plants are just on the verge of being amenable
19 to formal validation and verification, although that
20 hasn't been considered in the nuclear business. If
21 you do that, you really can prove in a mathematical
22 sense that the software is bug-free in the sense that
23 it reflects what the requirements laid on it at the
24 beginning were. But the problems are as often in the
25 requirements as they are in anything else.

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1 Nancy Levinson, who is one of the revered
2 consultants to the staff, has written an extremely
3 interesting report, which you might enjoy reading
4 about the Thorac 25 disasters. These are radiation
5 machines made by Atomic Energy of Canada. I have to
6 admit -- well, you know, Gail -- it killed a fair
7 number, a half dozen people or something like that.
8 I think it took three before they admitted that there
9 could be a problem with the software. The problems
10 they had were partly the classical problems, partly
11 new ones. The classical ones were a register that
12 recycled to zero and made problems when it recycled to
13 zero. Another one, because there a timing loop in the
14 thing which never shows up as a problem when people
15 were learning to run the thing, but when they became
16 skillful and typed faster, that it produced a problem.
17 Little things like that in the end are what make
18 trouble. But you can do V&V on those small systems if
19 you have to. That came out of the things. There were
20 a lot of things.

21 DOCTOR SHEWMON: There was no report.

22 Shall we get on to the last topic?

23 COMMISSIONER REMICK: Well, I have a
24 couple questions yet on this.

25 In your April 23rd, '93 letter, you

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1 addressed the control enunciator alarm reliability and
2 point out where you have alarms for manually activated
3 systems that are not backed up by automatic control
4 systems. The staff is proposing Class 1E equipment
5 and your final bottom line is we believe that the
6 staff needs to provide clarification and additional
7 justification for this position. It's not clear to me
8 that you're differing with the staff or you're just
9 arguing that they haven't made their case clearly.
10 It's a --

11 MR. CARROLL: Those were my words you were
12 reading.

13 DOCTOR LEWIS: This is not my baby.

14 MR. CARROLL: I guess I couldn't figure
15 out what they wanted people to do and what this system
16 was. I don't know that there are any of them in a
17 power plant. So, at a break in the meeting, I asked
18 the EPRI representative if they had any problems with
19 these words and he said, "Well, I don't think we ever
20 noticed them because we don't think we have any
21 systems that fall into that category, but I'm glad you
22 raised the issue because we want to find out more
23 about it."

24 COMMISSIONER REMICK: Okay. So, it is a
25 lack of --

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1 MR. CARROLL: To me, I don't understand
2 what this issue is about or what they think -- what
3 problem they're trying to solve.

4 COMMISSIONER REMICK: Okay. Well, we have
5 a meeting with the staff this afternoon on this
6 subject. We'll explore it with them at that time. I
7 wasn't quite clear what your final comment meant.
8 Okay.

9 One other point that I'd like to make.
10 I'm not addressing this to the Committee, but to my
11 colleagues and the comment made that it would have
12 been nice to have that National Academy study several
13 years ago, four years ago. I certainly agree. But we
14 have to be careful. Right now we are considering what
15 resources to put in some of the advanced designs and
16 I'm afraid we tend to say when we get an application
17 we'll begin the review and that's always four years
18 too late. I think we've learned that many, many times
19 already on reviewing the current set of reactors and
20 we have to be careful that we don't cut off our
21 resources for looking ahead to the future so that four
22 years from now we'll be saying the same things, I wish
23 four years ago we would have continued research or we
24 would have had this study and so forth.

25 That's not addressed to the Committee,

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1 it's addressed to my colleagues.

2 DOCTOR WILKINS: You have a letter from
3 AECL, for example, that urges you to continue study.

4 COMMISSIONER REMICK: Yes.

5 DOCTOR LEWIS: I'm fond these days of
6 quoting something I just recently learned from Yogi
7 Berra, which is if you come to a fork in the road,
8 take it.

9 COMMISSIONER REMICK: That's all I have.

10 COMMISSIONER de PLANQUE: I just wanted to
11 say I thought your March 18th letter was a very good
12 exposition of all the problems in this area.

13 DOCTOR LEWIS: Thank you.

14 DOCTOR WILLIAMS: You have a little

15 COMMISSIONER de PLANQUE: I have one
16 specific question and then a general one.

17 On the electromagnetic interference
18 problem, we've been kicking that around for awhile.

19 DOCTOR LEWIS: Yes.

20 COMMISSIONER de PLANQUE: And I realize
21 you don't consider this one of the central issues, but
22 again I see the comment that there's a lot of
23 expertise in here in the military area and still
24 untapped by the staff. Is that true?

25 DOCTOR LEWIS: Not entirely untapped, but

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1 the military experience where you know you're planning
2 to put systems into an environment in which people
3 will deliberately expose it to electromagnetic
4 interference, they've learned a lot. The keys are
5 shielding, protection of wires, limiting the signals
6 and, I think I said this to Tom Murley yesterday,
7 people have learned that the digital systems are more
8 immune to electromagnetic interference than the analog
9 systems. So, if you're really worried about it, you
10 should push for moving to CMOS digital systems.

11 No, I think the staff has spoken to
12 people. I think, in fact, this is a case in which
13 there has been an impact. They've been sort of pushed
14 to do it and they've done it.

15 COMMISSIONER de PLANQUE: Okay.

16 DOCTOR LEWIS: But whether they've
17 learned, I don't know.

18 COMMISSIONER de PLANQUE: Okay. But
19 they've done what you think should be done in this
20 area?

21 DOCTOR LEWIS: Yes. That's right, yes.

22 MR. CARROLL: Well, the issue started with
23 them coming in in response to our saying, "What
24 research are you doing?"

25 COMMISSIONER de PLANQUE: Right.

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1 MR. CARROLL: And the only research they
2 were doing was on TMI.

3 COMMISSIONER de PLANQUE: Right.

4 MR. CARROLL: So, I think that they have
5 now shifted gears and they have a pretty credible
6 broad-range research program going on.

7 COMMISSIONER de PLANQUE: Okay.

8 DOCTOR LEWIS: It was the wallet under the
9 lamp post.

10 COMMISSIONER de PLANQUE: Yes. Okay.

11 Back to the workshop and the Academy
12 study. A lot of the letters on this I see could be
13 characterized as a he said-she said, or he said-he
14 said problem and it's hard to figure out where the
15 truth is in between.

16 On the workshop situation, you talked
17 about a problem to some extent with the format because
18 it's not amenable to coming up with newer ideas, not
19 country music, not Bach, but something else. And the
20 content to some extent. Now, you said you haven't
21 come up with a Committee opinion on this yet. Are you
22 going to do this and are you going to come back with
23 this --

24 DOCTOR LEWIS: The Committee has
25 recommended at the workshop the Academy study. We

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1 haven't provided an agenda or a format for it. That's
2 usually done by talking to the people at the Academy.

3 COMMISSIONER de PLANQUE: What I have in
4 mind more is are you going to come back with some
5 comments to us on the workshop as you see it now and
6 any way in which you think the workshop that's
7 scheduled for the fall --

8 DOCTOR LEWIS: Oh, you mean the September
9 one?

10 COMMISSIONER de PLANQUE: Yes. Any way
11 that should be changed?

12 DOCTOR WILKINS: It's almost too early to
13 answer that question.

14 COMMISSIONER de PLANQUE: Okay.

15 DOCTOR WILKINS: I'll give you an offhand
16 answer, which is that I would propose to make a trip
17 report, so to speak, of that meeting to the Committee
18 and then ask the Committee what it wants to do. Among
19 other things, the Committee might decide to write you
20 another letter, or it might decide that things are
21 going pretty nicely and just let it slide. It's too
22 early to predict how that will turn out, Commissioner.

23 COMMISSIONER de PLANQUE: Okay.

24 DOCTOR LEWIS: I think if we're motivated
25 to say something, we will.

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1 MR. CARROLL: Just for the record, I would
2 note that two members signed additional comments
3 opposing --

4 COMMISSIONER de PLANQUE: Yes, we saw.
5 Yes.

6 MR. CARROLL: At this time.

7 COMMISSIONER de PLANQUE: Yes.

8 MR. WYLIE: Let me ask a question. You
9 were speaking about whether the Committee would
10 comment on the agenda, were you not?

11 COMMISSIONER de PLANQUE: On the what?

12 MR. WYLIE: On the agenda.

13 DOCTOR LEWIS: On the agenda or the
14 product?

15 DOCTOR WILKINS: On the agenda or what
16 happened?

17 COMMISSIONER de PLANQUE: The workshop
18 that's planned for the fall.

19 MR. WYLIE: Yes.

20 COMMISSIONER de PLANQUE: Are you going to
21 comment on the agenda? Are you going to make
22 suggestions about changes in format or the agenda per
23 se?

24 DOCTOR WILKINS: I thought we had already
25 had an opportunity to comment on the agenda.

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1 DOCTOR LEWIS: We were asked to comment on
2 the agenda. There was a fairly short time given. We
3 did give them some comments. We suggested only minor
4 changes, get more outside involvement. We also said
5 that two days are really not enough to do this job and
6 two crowded days are certainly not enough, things like
7 that. As a result, Ernest has been invited to be our
8 distinguished -- several of us will probably go to it.

9 COMMISSIONER de PLANQUE: Okay.

10 MR. LINDBLAD: Mr. Chairman, could I --

11 CHAIRMAN SELIN: Yes, Mr. Lindblad.

12 MR. LINDBLAD: As we discuss how we might
13 achieve a certain end, the regulation of digital
14 computers, I think at the end I really want to mention
15 that there's a number of us on the Committee, and I
16 think all members of the Committee think that our
17 motivation is that the proper application of computers
18 and digital technology has a great potential for
19 making safe plants, and that it is not a matter of
20 efficiency or cost or not getting replacement parts
21 for analog. It's the thought that digital technology
22 really opens up many opportunities for improving
23 safety in our nuclear plants.

24 DOCTOR LEWIS: And on top of that,
25 Chairman Selin, you mentioned that NRC doesn't design

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1 reactors and that's certainly right. But the people
2 who do design them are very conscious of what it's
3 easy to get through you people.

4 CHAIRMAN SELIN: I did give sort of an
5 offhand concession that that's true that we very much
6 affect -- we give signals, inadvertent or otherwise,
7 about what we will receive. One thing I'd like to
8 receive now is Doctor Shewmon's introduction to the
9 third topic.

10 DOCTOR SHEWMON: License renewal, we think
11 there's been movement in the right direction. We look
12 forward perhaps to even more and Bill Lindblad will
13 handle the presentation here.

14 MR. LINDBLAD: Just last month we sent you
15 a letter with a Committee position on the recent SECY-
16 93-049, which we reviewed in March and April. I'll
17 quote from a part of that to state what the
18 Committee's views are.

19 In a number of our letter reports, we
20 provide comments and recommendations on various
21 aspects of the license renewal rule, including our
22 recommendation that this rule and the maintenance rule
23 be better integrated in the interest of long-term
24 coherence of the regulatory process.

25 Additionally, we've provided reports to

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1 Mr. Taylor expressing our opposition to staff
2 proposals to use license renewal as a means of dealing
3 with problem issues, such as electrical cable
4 qualification and mechanical component fatigue life
5 when, in fact, these issues potentially affect
6 presently operating plants that may or may not seek
7 license renewal. Now, we continue to support these
8 views.

9 Since that time, the SECY-93-113 was
10 released on April 30th and the Committee has not had
11 an opportunity to review and discuss the staff
12 document. But some of us individually have quickly
13 scanned through the document and we find great efforts
14 by the staff to link requirements in the license
15 renewal effort to existing programs that operating
16 plants have today such as recordkeeping and quality
17 assurance and even maintenance procedures and
18 adherence to a new maintenance rule. This seems to us
19 individually as being very desirable. We encourage
20 that as we look at other opportunities to change,
21 whether it's in a rulemaking or in further
22 implementation by the staff, that similar things be
23 done. I really believe that the license renewal and
24 the 20 years beyond the 40 year licenses should be a
25 continuation of the safety programs that we have today

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1 rather than be a divergence of what we've done in the
2 past. I have the greatest confidence in what we know
3 how to do and what we do regularly. To the extent
4 that a license renewal would generate different
5 requirements or a divergence of requirements or
6 conflicts in requirements, I think that's --
7 personally I think that's not desirable.

8 I do recognize that in the issuance of a
9 renewal that there are things to be demonstrated
10 perhaps in hearings that assure the public that this
11 has been maintained, that safety is being maintained
12 in the extended plant. I think the best case can be
13 made that the extended life will be a continuation of
14 the safety programs that are currently running. So,
15 to the extent that license renewal requirements
16 reflect the existing requirements, I think that's to
17 be encouraged.

18 It was stated earlier that in the digital
19 context that sometimes we wait until the license
20 application comes in before we understand how it's to
21 be implemented or find the faults and the like. I
22 personally compliment the staff and you the Commission
23 for demonstrating in this most recent SECY
24 implementation examples so that applicants can see
25 whether it makes sense to plan for license renewal or

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1 not. I think that in the recent history we've seen
2 some people who have been discouraged by license
3 renewal requirements, along with other economic issues
4 that the utilities might have on their plate. I
5 believe in the safety of the operating plants and I
6 think that they will continue to operate safely with
7 license renewal and with the oversight that the NRC
8 has on these plants.

9 Any questions?

10 CHAIRMAN SELIN: I'd just like to point
11 out three things. First of all, the general
12 principles that Mr. Lindblad mentioned are, in fact,
13 and have always been the Commission's principles.
14 They're in the statement of consideration and the
15 rule, namely that not only is the current licensing
16 basis the basis for the extension of lifetime, but we
17 should not impose on people requirements in the first
18 40 years just because they're coming in for an
19 addition extension.

20 To follow-up on that, the staff has
21 followed your advice both on the questions of fatigue
22 and of environmental qualifications and therefore,
23 although there is a question I'd like to ask you, I'd
24 like not to do it now. That is whether you believe
25 the staff has responded appropriately to whether the

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1 questions on metal fatigue are overly done or not.
2 But they're not any longer in the context of license
3 renewal since they have taken the basic idea that says
4 if fatigue is a problem, it's a problem now and we
5 should take a look at it.

6 Now, perhaps you might revisit that in the
7 not too distant future since your advice was both very
8 general and very specific. I know that the staff has
9 responded to the general advice and I'd be interested
10 in whether they've, in your opinion, responded to the
11 specific advice. In other words, that the branch
12 technical position on metal fatigue was not justified
13 by any safety -- sufficiently justified by safety
14 consideration and safety --

15 DOCTOR SHEWMON: I have not seen the new
16 position, but we'd certainly be pleased to respond to
17 it.

18 CHAIRMAN SELIN: Okay. And then the third
19 point, just in passing, is that clearly in large part
20 because of Commissioner Curtiss' leadership, which for
21 the record I would like to say will be very, very
22 sorely missed on this and many other topics. Unlike
23 Commissioner Remick, I don't always acknowledge when
24 I'm speaking to my colleagues instead of to the
25 audience, but in this case I am. The importance of

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1 wrapping the two together, the maintenance program and
2 the license plant life extension program have been
3 recognized, I hope on a timely basis, and there is a
4 lot of work going on in this area.

5 Commissioner Rogers?

6 I'm sorry, Doctor Shewmon.

7 DOCTOR SHEWMON: Before you finish, I have
8 one other item I'd like to bring up.

9 We've been seeing a variety of new faces
10 around the ACRS table and I'd like to introduce two
11 people who --

12 CHAIRMAN SELIN: We'd like to have the
13 introductions, but that's not going to get you out of
14 listening to the questions from the other -- I'm going
15 to have to step out, so please introduce them now.
16 Would you?

17 DOCTOR SHEWMON: This is Bob Seale

18 CHAIRMAN SELIN: Doctor Seale, how are
19 you?

20 DOCTOR SHEWMON: And Pete Davis.

21 CHAIRMAN SELIN: Nice to see you.
22 Welcome.

23 DOCTOR SHEWMON: Okay. Pardon me. Go
24 ahead.

25 CHAIRMAN SELIN: I'm going to give you

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1 both the floor.

2 COMMISSIONER ROGERS: That's fine. I
3 don't have any questions.

4 CHAIRMAN SELIN: I need to excuse myself.

5 COMMISSIONER ROGERS: All right. Fine.

6 CHAIRMAN SELIN: Thank you.

7 COMMISSIONER ROGERS: I don't have any
8 specific questions.

9 Commissioner Curtiss?

10 COMMISSIONER CURTISS: I don't have any
11 questions, but I do want to make a couple of comments
12 and maybe a suggestion. I read the Committee's
13 letters on this subject with great interest. The
14 tenor of your comments, I think in a general way, is
15 very harmonious with the views that I have come to
16 hold as to how we ought to approach license renewal
17 and in particular how we might best bring about a
18 coherent integration of the license renewal rule with
19 what we now have in the maintenance rule.

20 Your most recent letter of April 23rd, as
21 you've sought to frame what you understand are the
22 Commission's interests in this area, and here I have
23 a suggestion to make, the question that I think we
24 have now before the Commission in SECY-93-049 and
25 SECY-93-113 is a fairly simple one. It has a lot of

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1 complexities, but boiled down to its essence it is
2 what from a technical standpoint is a reasonable
3 approach to take to the renewal of licenses for up to
4 an additional 20 year period.

5 In view of the programs and processes that
6 were currently have in place for operating reactors,
7 including but not limited to the maintenance rule, I
8 think you probably constrain your focus unduly in the
9 way you frame the interest of the Commission there at
10 the bottom of that page when you ask the question
11 whether the present license renewal rule can be
12 legally construed to accommodate the staff's proposal
13 in SECY-93-049.

14 I'd offer you, I guess, a gratuitous
15 suggestion and that is as you look at the
16 recommendations contained in the two recent SECY
17 papers, and I think as you're particularly well
18 equipped to do, it would be useful to hear from you as
19 to whether you believe the technical approach set
20 forth therein to the many various issues that the
21 staff has focused on over the past several months is,
22 in fact, a reasonable one.

23 There are some complexities in that paper
24 that I don't fully appreciate at this point and since
25 it is such a complex issue, I'm not going to ask you

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1 for your views at this point on that matter.

2 The question of whether that approach then
3 can be squared with the current license renewal rule,
4 in my view, is a secondary consideration and not a
5 primary one. If we believe a reasonable technical
6 approach in fact can be taken, needs to be taken that
7 is not consistent with what we're currently doing, we
8 ought to change that. If it's not consistent with the
9 current license renewal rule, I think the course of
10 action suggests itself. But I wouldn't want you to
11 limit your focus in this regard to whether you believe
12 the approach that's been suggested is, in fact,
13 something that one can square with the license renewal
14 rule.

15 MR. CARROLL: That isn't what we were
16 really attempting to say. I think the point we were
17 trying to make is on the next page. When all is said
18 and done, we ought to leave good tracks because the
19 present Commission isn't going to be around when the
20 first application is finally acted on.

21 COMMISSIONER CURTISS: Yes.

22 MR. CARROLL: We need to leave a very good
23 history.

24 COMMISSIONER CURTISS: Yes. I think that
25 point is extremely well taken. I look forward to your

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1 views on this and thank you for your effort.

2 COMMISSIONER REMICK: I have no question
3 on licensing renewal. One question though on your
4 April 26th letter on SECY-93-087. That's a very
5 weighty document. Can I assume that that is basically
6 your final letter on those issues that are up for
7 Commission decision?

8 DOCTOR SHEWMON: What's the title of 93-
9 087?

10 COMMISSIONER REMICK: It's the policy
11 technical and licensing issues pertaining to
12 evolutionary and advanced light water reactors.

13 MR. WYLIE: Well, I understand that
14 there's a new set of these coming out shortly that
15 we'll be looking at.

16 COMMISSIONER REMICK: A new set?

17 DOCTOR SHEWMON: A remaining set.

18 MR. WYLIE: Well, remaining set, yes.

19 COMMISSIONER REMICK: Oh, there will be
20 additional follow-up. But I'm thinking about the
21 items where the staff has made specific
22 recommendations for Commission decision. This is your
23 final letter on those?

24 MR. WYLIE: That's correct.

25 COMMISSIONER REMICK: All right.

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1 COMMISSIONER de PLANQUE: I have no
2 questions on this.

3 COMMISSIONER ROGERS: Well, we thank you
4 all very much. It's been, I think, as usual, a very
5 illuminating meeting. I think these meetings with the
6 ACRS, in my view, over the last few years have become
7 more and more enjoyable and more and more open ended,
8 less formalized, less constrained by time in some
9 ways. Witness the Chairman's leaving before we're
10 finished. And I personally feel that we have gained
11 a great deal through the work of the ACRS and we
12 appreciate your fine thoughts and help for the
13 Commission in its activities.

14 Thank you very much.

15 DOCTOR SHEWMON: Thank you.

16 (Whereupon, at 12:04 p.m., the above-
17 entitled matter was concluded.)
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of the United States Nuclear Regulatory Commission entitled:

TITLE OF MEETING: PERIODIC BRIEFING BY THE ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS

PLACE OF MEETING: ROCKVILLE, MARYLAND

DATE OF MEETING: MAY 14, 1993

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

Carol Lynch

Reporter's name: Peter Lynch

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

May 6, 1993

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

R. Savin for

FROM: John T. Larkins, Executive Director, ACRS

SUBJECT: ACRS MEETING WITH THE NRC COMMISSIONERS ON
MAY 14, 1993 - BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners on Friday, May 14, 1993, between 10:00 and 11:30 A.M. to discuss items of mutual interest, including the following. Background material related to these matters is attached:

1. Status of ACRS Review of Evolutionary and Advanced Light Water Reactor Designs - P. Shewmon, C. Michelson, J. Carroll and C. Wylie (PP.2-63)
2. Digital Instrumentation and Control Systems - H. Lewis (PP.64-83)
3. License Renewal - W. Lindblad (PP.84-100)

Attachments: As Stated

cc: ACRS Members
ACRS Technical Staff

1

ITEM 1: STATUS OF ACRS REVIEW OF EVOLUTIONARY AND
ADVANCED LIGHT WATER REACTOR DESIGNS

The Committee previously discussed the status of the advanced reactor reviews with the Commission on March 5, September 11 and December 11, 1992. Since then, a number of Committee and Subcommittee meetings have been held to discuss various aspects of the advanced reactor reviews, mainly, the GE ABWR, EPRI Requirements Document, Testing for the W AP600 and the GE SBWR, Design Acceptance Criteria (DAC), and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). The following is a brief summary of the status of ACRS reviews of these matters.

EVOLUTIONARY PLANTS

• General Electric Advanced Boiling Water Reactor (GE ABWR)

The ACRS Subcommittee on Advanced Boiling Water Reactors and other subcommittees have held 23 meetings beginning in October, 1989, to discuss the NRC staff's Draft Safety Evaluation Reports (DSERs), the GE Standard Safety Analysis Report for the ABWR, and related matters. The Committee has provided three letters to the EDO and two reports to the Commission on matters related to this review. The Committee anticipates issuing its final report on the final design approval (FDA) for the ABWR, shortly after the FSER is provided for review. This schedule is based on the assumption that GE and the NRC staff will provide the ACRS with reasonably complete documentation (SSAR Chapters and corresponding draft FSER prior to the final FSER). This will enable the Subcommittees on Advanced Boiling Water Reactors, Severe Accidents and Ad Hoc on DAC to hold meetings as appropriate and gather information for the Committee to use in the discussions of the FSER to complete a final ACRS report on the FDA for the ABWR.

The following documents are attached:

- ACRS report to the Commission dated March 18, 1993. Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule (PP.7-8)
- ACRS report to the Commission dated October 16, 1992. Subject: Second Interim Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design (PP.9-12)
- ACRS report to the Commission dated August 12, 1992. Subject: Inspections, Tests, Analyses, and Acceptance Criteria Program for the GE ABWR Design (PP.13-16)

- ACRS letter to James M. Taylor (EDO) dated April 13, 1992. Subject: Review of the Draft Safety Evaluation Reports on the GE Advanced Boiling Water Reactor Design (PP.17-25)

- Westinghouse RESAR SP/90 Design

The ACRS review of the Westinghouse's application for the Preliminary Design Approval (PDA) for the RESAR SP/90 design has been completed. The Committee provided a report to the Commission dated December 12, 1990 on this matter.

The following document is attached:

- ACRS Report to the Commission dated December 12, 1990. Subject: Westinghouse's Application for Preliminary Design Approval for the RESAR SP/90 Design (PP.26-31)

- ABB-CE System 80+ Design

The ACRS Subcommittee on Advanced Pressurized Water Reactors has held six meetings beginning in April 1990 to discuss the ABB-CE Systems 80+ design features and related issues such as the Licensing Review Basis (LRB) document. The Committee provided a report to the Commission dated November 14, 1990 in regard to the LRB. The staff's Draft SER on the Systems 80+ design was provided to the ACRS on October 1, 1992, and a Subcommittee meeting was held on February 10, 1993 to discuss this document. During the Subcommittee meeting, two major issues were discussed. These were human factors engineering and diversity of instrumentation and control. As these issues are not yet resolved, the Committee did not comment on them. Future Subcommittee meetings will be scheduled to continue discussion of this matter as additional information becomes available.

On April 13, 1993, some members of the ACRS Subcommittee on Advanced Pressurized Water Reactors visited the dynamic mockup of the control room at the ABB-CE facility in Windsor, Conn.

The following document is attached:

- ACRS report to the Commission dated November 14, 1990. Subject: SECY-90-353, Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor (PP.32-33)

3

- EPRI Utility Requirements Document for Evolutionary Plants

The ACRS Subcommittee on Improved Light Water Reactors has held seven meetings to discuss the staff's draft SERs related to various chapters of the EPRI Requirements Document for Evolutionary Light Water Reactor Designs. The Committee provided reports to the Commission dated April 23, 1991 and August 18, 1992 on this matter. The staff plans to issue a supplement to the FSER after all evolutionary policy issues have reached final resolution. The ACRS expects to review the supplement to the FSER.

The following document is attached:

- ACRS Report to the Commission dated August 18, 1992. Subject: Electric Power Research Institute Advanced Light Water Reactor Utility Requirements Document -- Volume II, Evolutionary Plants (PP.34-37)

PASSIVE PLANTS

- Westinghouse AP600

The Committee and the Subcommittees on Advanced Pressurized Water Reactors/Thermal Hydraulic Phenomena have heard presentations regarding design details for the Westinghouse AP600 passive plant and the test programs proposed by both Westinghouse and the staff in support of the AP600 passive plant design certification. The Committee provided reports dated November 14, 1991 and March 10, April 6, and July 17, 1992 to the Commission in regard to the test programs. The Committee will continue its review of the ongoing experimental and analyses programs related to the certification of the AP-600 design.

The Standard Safety Analysis Report for the AP600 was issued on June 26, 1992. The Committee will continue its discussion of this matter on a schedule consistent with the development of the staff's SER.

The following document is attached:

- ACRS report to the Commission dated July 17, 1992. Subject: Integral System and Separate Effects Testing in Support of the Westinghouse AP600 Plant Design Certification (PP.38-42)

4

- General Electric SBWR

The Committee and the Subcommittees on Advanced Boiling Water Reactors/Thermal Hydraulic Phenomena have heard presentations regarding design details and test programs for the General Electric SBWR passive plant. The Committee provided a report to the Commission dated June 10, 1992 regarding the proposed test programs in support of the SBWR design certification. The Committee will continue its review of the ongoing experimental and analyses programs related to the certification of the SBWR design.

The Standard Safety Analyses Report for the SBWR was received on August 26, 1992. The Committee will continue its discussion of this matter on a schedule consistent with the development of the staff's SER.

The following document is attached:

- ACRS report to the Commission dated June 10, 1992. Subject: Testing and Analysis Programs in Support of the Simplified Boiling Water Reactor Design Certification (PP.43-46)

- EPRI Requirements Document for Passive Plants

The Committee and the Subcommittee on Improved Light Water Reactors have been briefed on the EPRI Requirements Document for Passive Plant Designs. The ACRS plans to continue its review of this matter after receiving the staff's final SER.

POLICY ISSUES FOR EVOLUTIONARY AND PASSIVE PLANTS

The Committee has discussed the resolution of technical and policy issues at various meetings beginning in early 1990. The Committee has provided six reports to the Commission or the EDO on this matter. Subject to the availability of all necessary information from the staff and industry groups and satisfactory completion of the reviews, the Committee expects to continue providing reports to the Commission on these issues. The Committee will continue its review of additional issues as they are identified by the staff and/or the Committee.

5

The following documents are attached:

- ACRS report to the Commission dated April 26, 1993. Subject: SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." (PP.47-50)
- ACRS letter to James M. Taylor (EDO) dated September 16, 1992. Subject: Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs" (PP.51-54)
- ACRS letter to James M. Taylor (EDO) dated August 17, 1992. Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements (PP.55-59)
- ACRS letter to James M. Taylor (EDO) dated May 13, 1992. Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements (PP.60-63)

6



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

March 18, 1993

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

Dear Chairman Selin:

SUBJECT: ADVANCED BOILING WATER REACTOR (ABWR) REVIEW SCHEDULE

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

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

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

- design acceptance criteria/inspections, tests, analyses and acceptance criteria, digital control systems, control room and human factor provisions, and severe accident/probabilistic risk assessment considerations
- interface requirements and representative conceptual designs for uncertified portions of the design
- technical resolution of Unresolved Safety Issues and Generic Safety Issues as required by 10 CFR 52.47
- closure of open and confirmatory items in the October 1992 draft of the FSER

7

March 18, 1993

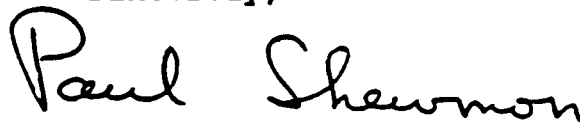
- closure of open items and concerns from the ACRS Advanced Boiling Water Reactors Subcommittee meetings of August 19, October 21, and November 18-19, 1992

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

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

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

Sincerely,



Paul Shewmon
Chairman

References:

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





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

October 16, 1992

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

Dear Chairman Selin:

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

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

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

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

9

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

Control Room Design DAC

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

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

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

Instrumentation and Controls (I&C) ITAAC

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

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October 16, 1992

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

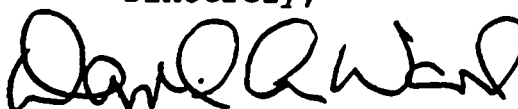
We learned at our meetings that GE had submitted a new revision of ITAAC 3.5 since the issuance of SECY-92-299. It was this new material, that had not been reviewed by the staff, that we reviewed. We had a number of suggested clarifications to the language of this ITAAC. In addition, there are certain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

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

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

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


Sincerely,



David A. Ward
Chairman

Additional Comments by ACRS Member Harold W. Lewis

I have a reservation about the Committee letter, for the specific issue of software certification. I have already taken (Reference 4) a more relaxed position than the Committee in the general area of DACs. That position reflects my view that we are dealing with a mature industry, not at all inexperienced in the design of modern



reactors, and therefore requiring a different style of regulation than may have been the case in an earlier period. The most effective role of NRC is through oversight of the safety of the industry product, rather than on certification of each detail. The DAC process lends itself to this kind of regulation, but only in areas in which the staff itself has the experience and expertise necessary to assume this more global role. I hope that the staff will not inhibit the application of modern technology through excessive specificity, as exemplified by the analog backup controversy, on which the Committee has previously commented (Reference 6).

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

References:

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

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

August 12, 1992

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

Dear Chairman Selin:

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

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

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

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

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

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

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

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

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

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

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August 12, 1992

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

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

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

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

Sincerely,



David A. Ward
Chairman

References:

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

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- Standby Liquid Control System (2.2.4)
 - Residual Heat Removal System (2.4.1)
 - Reactor Building Cooling Water System (2.11.3)
 - Emergency Diesel Generator System (Standby ac Power Supply - 2.12.13)
 - Control Building (2.15.12)
4. Report dated September 10, 1991, from David A. Ward, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications.



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

April 13, 1992

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

Dear Mr. Taylor:

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

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

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

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

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

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

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

1. Control Building Flooding

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

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

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

2. Adequacy of Physical Separation

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

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

3. Protection of Environmentally Sensitive Equipment

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

4. Review of Chilled-Water Systems

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

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

5. Use of Leak-Before-Break Methodology

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

6. Use of Integral Low-Pressure Turbine Rotors

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

7. Cavity Floor Area Beneath Reactor Vessel

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

8. Adequacy of the ABWR PRA

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

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

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

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

9. Containment Hydrodynamic Loads

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

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

The early work to address problems arising from analyses of the Mark I, II, and III containments is not sufficient to address similar processes that will occur following a LOCA in an ABWR containment. The ABWR is different for two reasons: (a) the volume of the wetwell air space in the ABWR is approximately that of a Mark II, and (b) the impact of the

air-clearing loads will be alleviated somewhat because the expected blowdown flows are much smaller than those expected in a Mark I or Mark II. Nevertheless, the combination of a much smaller wetwell and the lower mass flow from the break have not received sufficient attention to be written off by the staff or GE without further analysis or experimental investigation. We are not aware of any testing of the ABWR type geometry. We believe there are sufficient differences in both geometry and LOCA characteristics to require further evaluation of the air-clearing phase of the LOCA by more extensive analysis and/or experimental investigation.

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

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

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

- There is little useful information presented in the SSAR that describes how the Japanese codes and standards used for the RWCU System design can be converted to domestic design standards. The Quality Group classifications for certain portions of the RWCU System are inconsistent with the Japanese code-related classifications shown on the Piping and Instrumentation Diagrams. The Safety Class/Quality Group transition between the piping inside primary containment and that outside primary containment is not in accordance with ANSI/ANS safety class standards for BWR fluid systems.
- The questionable ability of system isolation valves to close under large-break-LOCA conditions has been the subject of extensive NRC testing and a Generic Letter (GL 89-10). However, the SSAR specifies no special performance requirements for these valves.
- The safety-grade leak detection and isolation system which actuates the system isolation valves was not described in detail sufficient to support an assessment of its adequacy.
- The ABWR PRA did not evaluate as initiating events RWCU System line breaks (or other LOCAs) outside the primary

containment. The exclusion of these breaks was based erroneously on an analysis of the effects of suppression pool bypass events on overall risk. However, the analysis failed to take into account that the bypass path (e.g., RWCU System pipe break) could be the initiator for the core-damage event.

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

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

11. Plant Design Life and Aging Management

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

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

12. Station Grounding and Surge Protection

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

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

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

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

13. Corrosion Control for Structures

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

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

Sincerely,



David A. Ward
Chairman

References:

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



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

December 12, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: WESTINGHOUSE'S APPLICATION FOR PRELIMINARY DESIGN
APPROVAL FOR THE RESAR SP/90 DESIGN

During the 367th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 1990, we completed our review of Westinghouse's application for Preliminary Design Approval (PDA) for the Westinghouse Reference Safety Analysis Report (RESAR SP/90) nuclear power block (NPB). We heard presentations from the NRC staff and the applicant concerning the staff's draft Safety Evaluation Report (SER) (NUREG-1413) for this PDA during our meeting. Representatives of the staff and of the Office of the General Counsel (OGC) discussed the related draft PDA document. Our Subcommittee on the Advanced Pressurized Water Reactors has held a series of meetings with the staff and representatives of the applicant regarding this matter over the past two and a half years. We also had the benefit of the documents referenced.

1.0 Scope and History of RESAR SP/90 Application

The RESAR SP/90 is an evolutionary (as contrasted with passive) Advanced Light-Water Reactor (ALWR) design for a single-unit NPB, rated at a reactor power of 3800 Mwt. Although many basic design decisions were made by Westinghouse prior to completion of the EPRI ALWR Utility Requirements Document, the design of this four-loop pressurized water reactor generally conforms to the EPRI requirements for such designs.

RESAR SP/90 NPB contains preliminary design information for the portion of the design that encompasses NPB buildings, structures, systems, and components. Specifically excluded from the scope are the turbine building, the waste disposal building, the service building, the administration building, the service water/cooling water structure, and the ultimate heat sink. These features will be the design responsibility of an applicant proposing to build a facility referencing the RESAR SP/90 design. Interface information addressing the pertinent safety-related design requirements necessary to ensure the compatibility of the referenced system with

the plant-specific portion of the facility has been included in the RESAR SP/90 application.

On October 24, 1983, Westinghouse submitted an application for a PDA for RESAR SP/90 NPB design in accordance with 10 CFR Part 50, Appendix O, "Standardization of Design: Staff Review of Standard Designs," which was the then existing regulatory basis for this type of application. The application was docketed on November 30, 1983 (Docket No. 50-601). The RESAR SP/90 application describing the design of the NPB was submitted in modular form during the period from October 23, 1983 to March 9, 1987. In addition, the information in RESAR SP/90 has been supplemented by 47 amendments to these modules.

2.0 Regulatory Background

Before the promulgation of 10 CFR Part 52 in May of 1989, the review of RESAR SP/90 had been performed by the staff pursuant to Appendix O to 10 CFR Part 50, using a procedure similar to that used for custom plant reviews for which guidance to staff reviewers is provided in the Standard Review Plan. This evaluation was analogous to a construction permit (CP) licensing review for a specific facility and conducted with the intent that, following satisfactory completion of the reviews performed by the staff and the ACRS, a PDA could be issued by the staff. The promulgation of 10 CFR Part 52 resulted in the transfer of Appendix O to 10 CFR Part 52; hence a PDA can now be issued for this application pursuant to 10 CFR Part 52. A PDA is optional for a Final Design Approval (FDA) and/or Design Certification under the provisions of 10 CFR Part 52.

3.0 The Staff's SER and the PDA

The SER and PDA represent the first stage of the staff's review of the design, construction, and operation of the RESAR SP/90 design. During our meetings, we learned that there is no prospective CP applicant nor does Westinghouse intend to apply for an FDA and/or Design Certification of the RESAR SP/90 design until there is a proven interest on the part of a domestic or foreign utility. The staff's SER summarizes the results of the staff's radiological safety review of the RESAR SP/90 NPB design and delineates the scope of the technical details considered in evaluating the proposed design. This review took place over the period of October 1983 to October 1989 (the date on which the staff decided to close its review). Environmental aspects were not considered in the staff review of RESAR SP/90, but would be addressed in a utility's plant-specific application.

3.1 Comments on the Staff's SER

There are 170 open items that will require resolution during the review of a plant-specific application for an Operating License (OL). Most of these appear to be the kind of open issues expected at this stage of the design. Of the 170 open items, 17 are site specific, 110 involve information in the scope of an OL or FDA and/or Design Certification application, and 43 had not been resolved by the staff when it closed its review in October 1989. (Westinghouse submittals on many of these 43 open items, including its proposed resolution of Generic Safety Issues, Unresolved Safety Issues, post-TMI regulatory requirements, and outstanding PRA issues are yet to be reviewed by the staff.) In view of these open items and our concerns regarding the SER and the many unresolved severe accident issues, we indicated to the staff that its conclusions on page 25-1 of the draft SER were stated too strongly. The staff agreed to revise this language.

The Committee is not of one mind regarding the issuance of a PDA for the RESAR SP/90. On the one hand, there is merit to the argument that Westinghouse's application for the RESAR SP/90 PDA was made in good faith in 1983 under a different set of regulations and that it is now appropriate to document the reviews that have taken place to date and issue the PDA for potential future use as a reference design for an individual plant CP application or as the starting point for an FDA and/or Design Certification application. Both Westinghouse and the staff advocate this approach; neither believes that it can devote further resources to this effort.

On the other hand, we view the RESAR SP/90 SER as a mixed bag of staff evaluations that were performed over the seven-year period since the application was filed. Some are current and well done; others are poorly done and/or were performed years ago and do not meet the standards that we believe should be applied to a current SER. A major contributor to this problem appears to be the staff's reliance on the July 1981 Standard Review Plan (SRP) (NUREG-0800) in performing this review. This SRP needs updating to reflect the current situation for the licensing of ALWRs.

Some examples of our concerns with the staff's SER are:

- 3.1.1 SER Chapter 7, Instrumentation and Controls, references a staff review that was performed in 1979 for the Westinghouse RESAR 414 design. The staff concluded that the computer based integrated reactor protection system design for RESAR SP/90 is acceptable for a PDA on the basis of the "similarity" of the RESAR 414 design to that proposed for RESAR SP/90. It is our view that the staff should have developed improved standards for the review of such systems during this 11-year period. We are

particularly concerned about the verification and validation of the software employed with computer based reactor protection systems. It appears that there is a need to augment existing staff resources with expertise in the computer science area so that appropriate standards can be developed for the review of computer based reactor protection systems. All of the proposed evolutionary and passive ALWRs employ such systems.

- 3.1.2 For materials used in the fabrication of pressure boundary components, Westinghouse has committed to follow applicable codes, standards, and regulatory guides. Many of these are not representative of current industry practice for such materials. We learned that Westinghouse has developed internal specifications for pressure boundary materials that presumably do reflect current industry practice. These were not submitted for the staff's review.
- 3.1.3 The proposed design employs water displacer control rods and associated control rod drive mechanisms, which is a new feature for Westinghouse plants. The SER describes the function of and strategy for use of these control rods. The SER, however, does not discuss the pressure boundary integrity of these new control rod drive mechanisms or the potential for reactivity insertion accidents that could result from misoperation of these control rods. Although Westinghouse submitted information on these subjects, the staff has not completed its review of this information. In general, we believe that new features of this kind should be thoroughly reviewed at an early stage of review.
- 3.1.4 Our review, which represents only a sampling effort, revealed a number of factual errors and inconsistencies in the SER; the staff has agreed to correct these errors. We believe that a review of the draft SER by Westinghouse, which has not yet had access to this predecisional document, would reveal additional errors that should be corrected. We recommend that this be done.

3.2 Comments on the PDA Document

The PDA states that the preliminary design information contained in RESAR SP/90 "complies with the requirements of 10 CFR Part 52, Appendix O . . . and is acceptable for incorporation by reference in applications for individual construction permits" The PDA does not describe how this preliminary design information would be used in a future FDA and/or Design Certification application.

We were told by OGC that this results from the fact that Westinghouse has not made an application under 10 CFR Part 52.

Given the quality of the SER for this PDA, we are concerned with the language of the PDA that requires the staff and ACRS to utilize and rely on the "approved preliminary design" in their reviews of any individual facility construction permit application " . . . unless significant information which substantially affects the determination set forth in this PDA, or other good cause, is present." OGC advised us that this requirement would apply only to the staff and ACRS reviews of a CP application and that both entities would be able to revisit any issue in their review of any type of application that would lead to an OL. This is satisfactory to us but could present problems for the staff in dealing with a contested CP application.

4.0 Comments on the SP/90 Design

We have two concerns regarding SP/90 design features:

- 4.1 Our review of the NPB layout indicates that Westinghouse has provided many desirable features from the standpoint of separation of equipment trains for protection against fires and industrial sabotage. However, we are concerned about the location of the emergency diesel generators (EDGs) on the same floor and corridor from the control room. We believe that another location for the EDG room should be specified in view of the potential for fire and/or explosions associated with the operation of large diesel generators.
- 4.2 The proposed RESAR SP/90 design employs a spherical containment. To deal with core/concrete interaction, the layout of the containment employs a cavity floor area beneath the reactor vessel that is based on the EPRI requirement of 0.02 m² per MWt. If a larger area is required, major changes to the containment sizing and layout may be needed. Timely development of a Commission position on this issue is important not only to this design but also to the design of all of the ALWRs.

5.0 ACRS Recommendations on the Issuance of a PDA

We believe, subject to the above comments, that the proposed design of the RESAR SP/90 NPB can be successfully completed and used in an application for an individual plant CP. Accordingly, we recommend that a PDA be issued for the proposed Westinghouse RESAR SP/90 NPB.

6.0 Concluding Remarks

Finally, we wish to commend the Westinghouse Electric Corporation, the Japanese APWR program participants, the EPRI ALWR Utility Steering Committee, and the EPRI staff for the effort they have expended in the development of this evolutionary design. The RESAR SP/90 design represents an important step forward in providing improved LWR designs that incorporate many of the lessons related to safety, performance, and reliability that have been learned by the nuclear power industry over the past 30 years.

Sincerely,



Carlyle Michelson
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Draft NUREG-1413, "Safety Evaluation Report Related to the Preliminary Design of the Standard Nuclear Steam Supply Reference System, RESAR SP/90" (Predecisional)
2. Draft Westinghouse Electric Corporation, Docket No. 50-601, Reference Safety Analysis Report (RESAR SP/90 Nuclear Power Block Standard Design), Preliminary Design Approval (PDA) (Predecisional) (Discussed during the November 8-10, 1990 ACRS full Committee meeting)
3. Letter NS-EPR-2675 dated November 1, 1982 from E. P. Rahe, Jr., Westinghouse Electric Corporation, to F. Miraglia, U.S. Nuclear Regulatory Commission, Subject: Westinghouse Advanced Pressurized Water Reactor Licensing Control Document



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

November 14, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: SECY-90-353, LICENSING REVIEW BASIS DOCUMENT FOR THE
COMBUSTION ENGINEERING, INC. SYSTEM 80+ EVOLUTIONARY
LIGHT WATER REACTOR

During the 367th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 1990, we reviewed the staff's SECY-90-353, "Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor," dated October 12, 1990. Our Subcommittee on Advanced Pressurized Water Reactors also considered this matter during a subcommittee meeting on November 1, 1990. During this review, we had the benefit of discussions with representatives of the NRC staff and of Asea Brown Boveri Combustion Engineering. We also had the benefit of the documents referenced.

The staff has recommended that the Licensing Review Basis (LRB) effort for the Combustion Engineering (CE) System 80+ design, which is well advanced, be continued to completion. There does not appear to be any substantive disagreement between the staff and CE on issues addressed in the LRB document.

The only approved LRB document was proposed by the General Electric Company (GE) as a way of obtaining early agreement with the staff on major process and technical issues for the review of its advanced boiling water reactor design certification application. It was approved by the Director of NRR in a letter to Mr. R. Artigas, GE, on August 7, 1987. This letter contains the qualification that the LRB represented the approach in "certain key areas" that GE was committed to follow ". . . until final Commission positions and staff requirements are defined and implemented." At that time, neither 10 CFR Part 52 nor Commission-approved staff positions relating to the certification of advanced light water reactors such as SECY-90-016 (referenced) were available. We note that 10 CFR Part 52 does not discuss the use of LRB documents as a part of the final design approval or certification process. These regulatory requirements and others under development have preempted the need for and diminished the usefulness of an LRB document for the CE System 80+ design. We recommend that no further effort be devoted to the proposed LRB document for the CE System 80+ design.

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November 14, 1990

Additional comments by ACRS members Ivan Catton, Paul G. Shewmon, and J. Ernest Wilkins, Jr., are presented below.

Sincerely,



Carlyle Michelson
Chairman

Additional Comments by ACRS Members Ivan Catton, Paul G. Shewmon, and J. Ernest Wilkins, Jr.

We understand that this LRB document can be completed and issued with relatively little additional effort. If so, we would prefer to see an orderly disposition of this LRB document in accordance with the staff recommendation in SECY-90-362 (referenced). We would agree with our colleagues that the CE System 80+ LRB effort be terminated now if the Commission, the staff, and the ACRS need to invest any significant additional effort.

References:

1. SECY-90-353, "Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor," dated October 12, 1990.
2. SECY-90-362, "Staff Comments on the Continuing Need for a License Review Basis Document for Each Passive Design," dated October 24, 1990.
3. SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements," dated January 12, 1990.
4. Letter LD-90-005 dated January 22, 1990 from A. E. Scherer, Combustion Engineering, to R. Singh, Subject: System 80+ Licensing Review Basis Document.
5. Letter LD-90-060 dated August 28, 1990, from E. H. Kennedy, Combustion Engineering, to Thomas V. Wambach, NRC, Subject: Licensing Review Basis for the System 80+ Standard Design.



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

August 18, 1992

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

Dear Chairman Selin:

SUBJECT: ELECTRIC POWER RESEARCH INSTITUTE ADVANCED LIGHT WATER
REACTOR UTILITY REQUIREMENTS DOCUMENT -- VOLUME II,
EVOLUTIONARY PLANTS

During the 387th and 388th meetings of the Advisory Committee on Reactor Safeguards, July 9-11 and August 6-8, 1992, we reviewed the NRC staff's Final Safety Evaluation Report (FSER) for Volume II of the Electric Power Research Institute's (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) for Evolutionary Plants. Our Subcommittee on Improved Light Water Reactors held meetings on June 17-18 and July 27, 1992, to review this subject. During these meetings, we had the benefit of discussions with representatives of the NRC staff and EPRI. We also had the benefit of the documents referenced.

In the early 1980s, EPRI established the ALWR program to support the United States utility industry efforts to ensure a viable nuclear power generation option for the 1990s and beyond. The objective of the program was to ensure that future nuclear power plants would be safer, simpler, more robust with greater margins, more easily operated and maintained, and more certain of being constructed and licensed without delays. This was accomplished using utility experience, by establishing design philosophy, producing design criteria and guidance to achieve the objective, and addressing the policies and regulations of the NRC.

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

- Volume I, "ALWR Policy and Summary of Top-Tier Requirements," is a management-level synopsis of the URD, including the design objectives and philosophy, the overall physical configuration and features of a future nuclear plant design, and the steps necessary to take the proposed ALWR design criteria beyond the conceptual design state to a completed, functioning power plant.

- Volume II, "ALWR Evolutionary Plant," consists of 13 chapters and contains utility design requirements for an evolutionary nuclear power plant (approximately 1350 Mwe).
- Volume III, "ALWR Passive Plant," contains utility design requirements for passive nuclear power plants (approximately 600 Mwe).

We have followed the development of the EPRI ALWR program from its inception and offered suggestions regarding safety improvements on several occasions. We also held numerous subcommittee and Committee meetings to consider and discuss the development of the EPRI URD program and the NRC staff's reviews.

The staff's review of the URD was conducted as described in NUREG-1197. As noted therein, the staff used NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," for review guidance. In addition, the staff's review reflects the requirements of 10 CFR 52, the Commission's policy statements on severe accidents, and the safety goals.

Although the SRP was used by the staff as guidance, the level of detail in the EPRI submittal did not permit a review of its completeness. (The SRP was written to support the review of safety analysis reports on specific plant designs for which a significant amount of design and construction information was available.) The staff conducted its review with the understanding that EPRI design criteria would meet all current regulations, except where deviations were identified. The staff's review of the URD focused primarily on determining whether the EPRI criteria did or did not conflict with current regulatory requirements.

In its review of Volume II of the URD, the staff identified a number of issues that will require additional information before the staff can reach a final conclusion. Initially, the staff divided the outstanding issues into three categories: (1) open policy issues on which the staff has proposed a position, but for which the Commission has not yet provided guidance, (2) open issues that must be satisfactorily resolved before the staff can complete its review of the URD, or (3) confirmatory issues for which the staff will ensure follow up of commitments in the URD.

At this date the staff has identified 21 open policy issues that are included in a draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements" that was issued on February 27, 1992. We provided our recommendations on the open policy issues pertaining to evolutionary plants in our letters which addressed SECY-90-016, SECY-91-078, and the draft SECY paper of February 7, 1992.

August 18, 1992

The staff has handled the remaining 410 open issues which were identified in the FSER for Volume II by classifying them as "Vendor or Utility Specific Items" which must be satisfactorily addressed during the staff's review of a vendor- or utility-specific application. The staff plans to issue a supplement to the FSER after all evolutionary policy issues have reached final resolution. The staff indicated that they plan to interact with EPRI in an attempt to resolve significant open issues which may be resolved generically, and to include in a supplement any which are resolved.

We recommend generic resolution of as many of these issues as possible.

We commend EPRI for developing a comprehensive set of requirements. These will aid in the design of nuclear plants which will be safer, simpler, more robust, and more easily operated and maintained.

We commend the NRC staff for a very thorough review of the EPRI ALWR Evolutionary URD, and its work with EPRI to identify and resolve many issues relevant to licensing future LWRs. We recognize the NRC staff's position that its review necessarily is incomplete.

Sincerely,



David A. Ward
Chairman

References:

1. SECY-92-172, dated May 12, 1992, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Final Safety Evaluation Report for Volume II of the Electric Power Research Institute's Advanced Light Water Reactor Requirements Document, including the following enclosures:
 - Draft Safety Evaluation Report for Volume I, "Program Summary of the NRC Review of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992
 - Safety Evaluation Report for Volume II, "NRC Review of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document for Evolutionary Plant Designs," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992

2. Advanced Light Water Reactor Utility Requirements Document, Volume II, "ALWR Evolutionary Plant," Chapters 1-13, through Revision 4, dated April 1992, Prepared for Electric Power Research Institute



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

July 17, 1992

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

Dear Chairman Selin:

SUBJECT: INTEGRAL SYSTEM AND SEPARATE EFFECTS TESTING IN SUPPORT
OF THE WESTINGHOUSE AP600 PLANT DESIGN CERTIFICATION

During the 387th meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 1992, we discussed the programs of integral system and separate effects testing being planned by both Westinghouse and NRC to support the certification effort for the Westinghouse Electric Corporation's AP600 passive plant design. We held discussions on this matter during our 381st through 384th (January-April 1992) meetings, inclusive. Our Subcommittee on Thermal Hydraulic Phenomena held meetings on December 17, 1991, March 3, 1992, and June 23-24, 1992 to review this issue. During these meetings, we had the benefit of discussions with representatives of the Westinghouse Electric Corporation and the NRC staff. We also had benefit of the referenced documents. We have previously reported to you on this matter in our letters of March 10 and April 6, 1992.

BACKGROUND

Appropriately validated thermal hydraulic computer models must be relied on to support the safety assessments required for certification of the AP600. Westinghouse has indicated that it plans to use its more mechanistic assessment code, WCOBRA/TRAC, only for large-break LOCA analyses, and will rely on its evaluation model, NOTRUMP, for analyses of all other design-basis events. The NRC plans to use RELAP5/MOD3 to support its assessments.

The NOTRUMP code is an evaluation model code that is based on 10 CFR Part 50, Appendix K, requirements. The other two codes, WCOBRA/TRAC and RELAP5/MOD3, are more mechanistic codes that have been qualified as best-estimate tools only for large-break LOCAs. All of these analysis tools will be required to simulate the AP600 behavior in regimes where the codes are known to be weak. These regimes include phenomena such as horizontal (perhaps countercurrent stratified) flows, interface movements, thermal

stratification, rapid "shock" condensation, boron mixing, and low-pressure gravity-driven flows.

To develop the necessary data for improvement and validation of these models for AP600 assessment, Westinghouse now has plans for conducting a number of separate effects tests at several different facilities, and integral system tests. The integral system test programs are to be conducted in a low-pressure facility now nearing final design at the Oregon State University (OSU) and in an existing high-pressure facility, SPES (in Italy), to be modified to better simulate AP600.

The NRC has proposed to conduct high-pressure confirmatory testing by modifying and using the existing ROSA-IV facility at JAERI in Japan. The modified facility will be referred to as ROSA-V. The NRC has no specific plans for additional separate effects testing. The staff does plan to conduct low-pressure integral system testing in the OSU facility after the Westinghouse program has been completed.

At this time, we have the following comments and recommendations regarding various aspects of these planned and proposed efforts.

WESTINGHOUSE PROGRAM

We believe that, with certain enhancements, the Westinghouse program will be adequate for the certification process. We have the following specific comments and recommendations:

- We are concerned that Westinghouse plans to rely primarily on its NOTRUMP evaluation model (EM) code. It is a step backwards to use computer codes of only EM sophistication and capabilities to evaluate the thermal hydraulic behavior of new nuclear power plants.
- The Westinghouse separate effects tests of most importance to the certification of AP600 are the Core Make-up Tank (CMT) tests and the Automatic Depressurization System (ADS) tests. The test matrices for these do not cover ranges of conditions that are broad enough to yield an adequate data base for the required model development. We recommend that pressure disturbances of the types that would be caused by either ADS valve actuation or by rapid steam condensation when cold CMT fluid is injected into the downcomer region be part of the test program.
- An additional separate effects test facility is needed to investigate the asymmetric effects associated with the downcomer and with the cold-side plenum of the steam generator.

- SPES is generally a good choice for conducting full-height, full-pressure integral system tests. However, in addition to the scaling problems associated with a high ratio of surface area to fluid volume that plague small-scale simulations of this kind (and must be dealt with), the proposed modified version, SPES-II, has two important scaling defects that should be eliminated: (a) the aspect ratio (height to diameter) of the simulated pressurizer is different from that of the AP600 and (b) the cold leg configuration is not geometrically similar to that of AP600.

We recommend that Westinghouse be required to preserve the scaling of the pressurizer and the geometrical configuration of the cold legs, to better simulate AP600 behavior (this would include simulation of a reactor coolant pump in each leg).

- The method proposed for simulating steam generator tube ruptures in SPES-II is flawed in that it does not appear to allow the break flow from the primary system to be from both the hot and cold sides of the tube. We recommend that Westinghouse develop a better simulation method.
- The OSU low-pressure integral system testing facility is well conceived. We commend Westinghouse for its efforts with respect to this facility. Our evaluation of the scaling rationale for the facility design (discussed during the subcommittee meeting of June 23-24, 1992) is that it is soundly based. Further, the 400 psia design capability should allow considerable simulation of high-pressure effects, while providing the more important low-pressure behavior.

NRC PROGRAM

Our understanding of the justification provided by the NRC staff for its proposed confirmatory high-pressure integral system testing in the ROSA-V facility is as follows:

- Because ROSA-V is considerably larger than SPES-II, such confirmatory testing would provide an additional check on the adequacy of the scaling capabilities of the codes, and would help confirm that important effects have not been overlooked.
- The confirmatory test program would provide the opportunity to maintain the staff's thermal hydraulic expertise and up-to-date knowledge in this field.

While we agree that the above considerations have some merit, we have not been persuaded that confirmatory high-pressure testing by the staff is needed before the AP600 design certification and, even if this were the case, we have significant reservations about the

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adequacy of the ROSA-V facility for this purpose. These positions are based on the following observations:

- The NRC staff has not presented convincing arguments supporting its needs for confirmatory testing, particularly at high pressures.
- The SPES-II facility appears to be sufficient to meet all the high-pressure integral system testing needs. The NRC will be able to use the SPES-II facility for its confirmatory testing needs just as it plans to use the OSU facility.
- The desired staff experience will come from pre-test and post-test evaluations of the various tests using the RELAP5/MOD3 code. This experience can just as easily be obtained by evaluating the SPES-II and OSU tests and results.
- The ROSA-V facility contains several atypicalities that will manifest themselves in difficult-to-explain behavior relative to that expected for AP600 (the sensitivity of the ROSA-V thermal hydraulic behavior is well documented in the INEL report, NUREG/CR-5853).
- The tests would be in a distant location. There would be a very limited number of tests, because of the expense involved. In addition, we are concerned that the adequacy of instrumentation (for example) might have to be compromised in order to reduce overall program costs.

For the above reasons, we believe that NRC resources would be better used by focusing on three areas: (a) possible additional separate effects testing to support the modeling needs for RELAP5/MOD3, (b) participation in the pre-test and post-test analyses efforts associated with the SPES-II and the OSU test programs, and (c) consideration of utilizing the SPES-II facility for high-pressure confirmatory testing needs in the same way the staff plans to use the OSU facility for its confirmatory low-pressure testing needs.

To accomplish the above objectives, we believe that the staff should consider the establishment of a task force of experts in related fields to assist it in the development of the analytical and experimental programs necessary for timely certification of the AP600 passive plant design.

Sincerely,

Paul Shewmon

Paul Shewmon
Acting Chairman

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References:

1. U.S. Nuclear Regulatory Commission, NUREG/CR-5853, "Investigation of the Applicability and Limitations of the ROSA-IV Large Scale Test Facility for AP600 Safety Assessment (Draft)," dated May 1992
2. T. Boucher, Idaho National Engineering Laboratory, et al., "Scaling Issues for a Thermal-Hydraulic Integral Test Facility," Paper transmitted via a memorandum from L. Shotkin, NRC-RES, for P. Boehnert, ACRS, dated June 29, 1992
3. Oregon State University Report, OSU-NE-9204 (Draft), "Scaling Analysis for the OSU AP600 Integral System and Long Term Cooling Test Facility," J. Reyes, Jr., dated June 1992 (W Proprietary Report)
4. Letter dated January 22, 1992, from G. Saporano, ENEA, Italy, to E. S. Beckjord, NRC, transmitting documentation on SPES test facility
5. Memorandum dated June 13, 1991 from S. Modro, INEL, for L. Shotkin, NRC-RES, transmitting draft report, "Evaluation of Scaled Integral Test Facility Concepts for the AP600" by Modro, et al.
6. U.S. Nuclear Regulatory Commission, SECY-92-219, "NRC-Sponsored Confirmatory Testing of the Westinghouse AP600 Design," dated June 16, 1992 (Predecisional)
7. U.S. Nuclear Regulatory Commission, SECY-92-219A, "Addendum to SECY-92-219 - Providing Additional Information to Justify Sole Source Procurement," dated July 9, 1992 (Predecisional)
8. Memorandum dated April 21, 1992, from S. Chilk, Secretary, for J. M. Taylor, EDO, and W. Parler, General Counsel, Subject: SECY-92-037 - Need for NRC-Sponsored Confirmatory Integral System Testing of the Westinghouse AP600 Design
9. Westinghouse Topical Report, WCAP-13277, "Scaling, Design and Verification of the SPES-2, the Italian Experimental Facility Simulator of the AP600 Plant," dated April 1992 (W Proprietary Report)



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

June 10, 1992

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

Dear Chairman Selin:

SUBJECT: TESTING AND ANALYSIS PROGRAMS IN SUPPORT OF THE
SIMPLIFIED BOILING WATER REACTOR DESIGN CERTIFICATION

During the 385th and 386th meetings of the Advisory Committee on Reactor Safeguards, May 6-9 and June 4-5, 1992, we reviewed the testing and analysis programs in progress and proposed by GE Nuclear Energy (GE) in support of the certification effort for the Simplified Boiling Water Reactor (SBWR) passive plant design. Our Subcommittee on Thermal Hydraulic Phenomena held meetings to discuss this topic on April 23 and June 2, 1992. During these meetings, we had the benefit of discussions with representatives of GE and the NRC staff. We also had the benefit of the documents referenced.

GE will use its best-estimate code, TRACG, to evaluate the SBWR thermal hydraulic behavior under accident conditions ranging from ATWS with instabilities to long-term behavior of the Passive Containment Cooling System (PCCS). GE representatives presented a very good analysis of processes and phenomena important to accident scenarios postulated for the SBWR. The results were summarized in tables which are to be used by GE to validate the TRACG computer code. However, these same tables appear not to have been used to guide the design and operation of the experimental facilities that are to support the code validation process.

The GE experimental program consists of three elements:

- 1) Laboratory scale experiments to obtain fundamental heat transfer data,
- 2) Separate effects tests to obtain data for parts of the total system and full-scale components where necessary, and
- 3) Integral system tests to obtain system data.

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Although we were shown some comparisons of TRACG predictions with data from GE's integral system tests (GIST and GIRAFFE facilities), the question of whether or not the facilities can scale the important phenomena was not addressed in either GE's presentation or in the documents supplied to the ACRS by GE. A rigorous scaling analysis is needed if integral system test data alone are to be used to demonstrate that a TRACG calculation is meaningful.

We have some comments about the elements of the GE test plan. The initial conditions for the integral system tests are based on conditions assumed to exist some time after vessel depressurization. These conditions include an initial drywell and PCCS nitrogen mass fraction of 15 percent. The nitrogen concentration could be much higher. GE should develop a basis for its choices of initial conditions or broaden its test matrix to include some tests at much higher values of the nitrogen concentration, both in the drywell and in the PCCS.

Separate effects tests to be conducted in the PANTHERS facility will yield the data needed to characterize heat exchanger behavior under a variety of expected conditions. In particular, GE has agreed to add instrumentation to the individual heat exchanger tubes to obtain local heat transfer data. This will make the GIRAFFE integral system experiments more meaningful. We believe GE has been very responsive to issues raised by both the ACRS and the NRC staff in this regard.

The oscillatory behavior observed in the GIRAFFE integral system tests needs more detailed study to ensure that the suppression pool does not overheat due to steam bypass of the PCCS through the suppression pool top horizontal vents. The steam flow rate will be low which could lead to a stratified condition. The suppression pool is not a very effective heat sink when this process occurs. This may well require a separate effects study to obtain data for development of a low steam flow model for the horizontal vent. Further, review of the GIRAFFE facility instrumentation is needed to ensure that the resulting data will support TRACG model validation.

The SBWR has full pressure isolation condensers (IC) capable of removing 4.5 percent of full power decay heat at full system pressure. The behavior of isolation condensers is well understood and introduces no new processes. GE has indicated that it will collect relevant IC operating data for staff review. The SBWR is automatically depressurized when the vessel water level drops to some prescribed value by a staged opening of squib-type valves. Further, GE has had a great deal of experience with automatic depressurization and only the squib-type valve itself is of a new design. As a result, we do not believe that full-height, full-

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June 10, 1992

pressure integral system testing is required for certification of the SBWR design.

The GE program includes conduct of integral system testing at the PANDA facility located in Switzerland. The NRC staff would like GE to obtain data from this facility in time to support its design certification review of the SBWR. To do so, GE would have to accelerate its schedule by six months. We agree with the NRC staff that further integral system testing of the PCCS is needed prior to the final design approval. It has not been demonstrated by GE that existing data obtained from GIRAFFE or GIST testing are sufficient for validation of the TRACG code, nor that the PANDA test facility will yield the needed data. A more definitive assessment by GE is needed; this assessment should include both the scaling rationale for the GIRAFFE, GIST, and PANDA facilities, and a demonstration of how the effects of test facility scaling distortion impact the important processes and phenomena outlined by GE in its evaluation of TRACG. As a part of such an effort, it may be possible to show that one can obtain the needed data by some combination of additional separate effects tests and judicious use of the GIRAFFE and GIST facilities.

To summarize, we agree with the NRC staff views that full-height, full-pressure integral system testing is not needed to support the SBWR design certification. Further, we agree that early integral system testing of the PCCS is essential to meet the present design certification schedule. We have not, however, seen evidence that the PANDA facility is adequate to obtain the needed data.

Sincerely,



David A. Ward
Chairman

References:

1. Memorandum dated February 26, 1992, for the Commissioners from James M. Taylor, Executive Director for Operations, transmitting Advance Copy of proposed Commission paper, "Evaluation of the General Electric Company's (GE's) Test Program to Support Design Certification for the Simplified Boiling Water Reactor (SBWR)"
2. Letter dated February 3, 1992, from R. C. Mitchell, GE Nuclear Energy, to U.S. Nuclear Regulatory Commission, Subject: GE Response to Request for Information on SBWR Testing Program

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3. Joint Study Report, "Feature Technology of Simplified BWR (Phase I) GIRAFFE (Final Report)," dated November 1990, The Japan Atomic Power Company, et al. (GE Proprietary Information)
4. GE Nuclear Energy, GEFR-00850, "Simplified Boiling Water Reactor (SBWR) Program Gravity-Driven Cooling System (GDCS) Integrated Systems Test - Final Report," A.F. Billig, dated October 1989 (Applied Technology Restriction)
5. "ALPHA - The Long Term Passive Decay Heat Removal and Aerosol Retention Program at the Paul Scherrer Institute, Switzerland," by P. Coddington, et al., Paul Scherrer Institute, undated
6. Paper from the Proceedings of The International Conference on Multiphase Flows '91 - Tsukuba, Japan, September 24-27, "Condensation in a Natural Circulation Loop with Noncondensable Gases Part 1 - Heat Transfer," K. M. Vierow, GE Nuclear Energy, and V. Schrock, University of California
7. GE Draft Report: "Test Specification for IC & PCC Tests," undated (GE Proprietary Information)
8. Paper submitted to the Department of Energy, "The Effect of Noncondensable Gases on Steam Condensation Under Forced Convection Conditions," M. Siddique, Ph.D. Thesis - Massachusetts Institute of Technology, dated January 1992



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

April 26, 1993

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

Dear Chairman Selin:

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

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

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

I. SECY-90-016 ISSUES

E. Fire Protection

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

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

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

G. Hydrogen Control

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

H. Core Debris Coolability

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

J. Containment Performance

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

L. Equipment Survivability

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

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II. OTHER EVOLUTIONARY AND PASSIVE DESIGN ISSUES

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

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

T. Control Room Annunciator (Alarm) Reliability

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

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

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

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

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



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

September 16, 1992

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

Dear Mr. Taylor:

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

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

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

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

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

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

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

B. Analyses of External Events Beyond the Design Basis

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

C. Elimination of the Operating Basis Earthquake from Seismic Design

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

D. Multiple Steam Generator Tube Ruptures (MSGTRs)

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

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

E. Probabilistic Risk Assessment (PRA) Beyond Design Certification

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

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

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

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

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

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

G. Control Room Annunciator (Alarm) Reliability

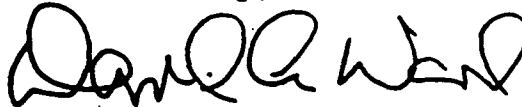
We agree with the staff's position that the alarm system for ALWRs should meet the requirements of the EPRI Utility Requirements Document.

September 16, 1992

H. Regulatory Treatment of Nonsafety Systems

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

Sincerely,



David A. Ward
Chairman

Reference:

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



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

August 17, 1992

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

Dear Mr. Taylor:

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

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

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

E. Fire Protection

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

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

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

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

F. Intersystem Loss-of-Coolant-Accident

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

G. Hydrogen Control

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

The staff also recommends that the system be capable of precluding uniform containment concentrations of hydrogen greater than 10 percent. We are aware of analytical work in

support of the resolution of Generic Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas," that suggests the possibility of transition to detonation at average concentrations as low as 12 percent. We recommend that the staff do a similar analysis of the impact of hydrogen combustion, and possible detonation including stratification, before establishing a limit for the average hydrogen concentration. This is of particular importance to steel-shell containments.

I. High Pressure Core Melt Ejection

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

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

J. Containment Performance

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

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

the applicant without any real guidance from the Commission on this important topic.

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

K. Dedicated Containment Vent Penetration

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

L. Equipment Survivability

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

N. In-Service Testing of Pumps and Valves

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

III.E - Control Room Habitability

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

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August 17, 1992

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

Sincerely,



David A. Ward
Chairman

References:

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



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

May 13, 1992

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

Dear Mr. Taylor:

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

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

I. SECY-90-016 Issues

Item M. Elimination of Operating Basis Earthquake

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

II. Other Evolutionary and Passive Design Issues

Item A. Industry Codes and Standards

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

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

Item D. Leak Before Break

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

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

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

Item F. Tornado Design Basis

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

Item H. Containment Leakage Rate Testing

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

Item I. Post-Accident Sampling System (PASS)

The staff is requesting approval of changes in requirements for the PASS currently found in 10 CFR 50.35(f)(2)(viii). These requirements, and the

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guidance contained in Regulatory Guide 1.79 and in NUREG-0737, resulted from consideration of the TMI-2 accident.

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

1. The requirements as contained in the above referenced regulation refer to "the reactor coolant system and containment that may contain TID-14844 source term radioactive materials" and to measurement of these and other materials. In light of source terms now considered in severe accident analysis, it is advisable to revise this obsolete description.
2. The proposal for "Elimination of the Hydrogen Analysis of Containment Atmosphere Samples" is appropriate, given that safety grade hydrogen monitoring instrumentation will be installed.
3. The Electric Power Research Institute (EPRI) proposed elimination of an existing requirement for the capability to sample the reactor coolant at operating pressure in order to measure the dissolved gas and chloride in the coolant. EPRI claims that maintaining the systems on existing plants produces significant exposure of operating personnel, and that given a severe accident, no useful information, not otherwise available, is provided by this capability. The staff proposes to retain the requirement, but to change the time after accident onset at which the capability must be available from 8 to 24 hours. During our discussion with the staff, we were unable to elicit any reason for this requirement other than that it was established following the TMI-2 accident. We cannot endorse continuation of the requirement for high pressure sampling on the basis of information available to us.
4. The staff proposes approval of a position that "would require the capability to take samples for boron and for activity measurements 8 hours and 24 hours, respectively, after the end of power operation." The intent appears appropriate, however, we suggest that it might be better to specify a time at which the information from measurements becomes avail-

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

Item N. Site-Specific Probabilistic Risk Assessment

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

Sincerely,



David A. Ward
Chairman

References:

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

ITEM 2: DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

The Committee provided two reports in September 1992 regarding the staff's recommended approach toward the design certification policy issue associated with digital instrumentation and control systems. In a letter to the EDO dated September 16, 1992, the Committee generally agreed with the staff's recommendations, with the exception that independent "hardwired" analog safety-grade displays and controls might not be the only solution to concerns over common-mode failures of digital systems. In a separate report to the Commission, the Committee expanded upon its concern that the staff's recommended position may reject a digital-based I&C backup system as a means of overcoming common-mode failure, without having assessed the reliability of such a system. In response, the EDO provided a letter dated October 23, 1992, in which he indicated that, based on comments received from the ACRS, EPRI and industry, the staff will allow flexibility in implementing the required independent set of displays and controls, dependent upon an evaluation of the specific design features provided by the vendor.

The Committee was briefed by the staff at the January 1993 meeting on the status of the draft generic letter on analog-to-digital I&C conversions for operating plants. The staff also recently transmitted the NUMARC draft guidelines for these conversions. The Committee remains much interested in this issue and plans to consider it prior to the issuance of a final generic letter.

The final ACRS Subcommittee meeting of the two-year series on these issues was held in February 1993. During the March 1993 meeting, the Committee discussed the staff's progress in defining the regulatory requirements for digital instrumentation and control systems. The Committee plans to follow-up on this matter,

The following documents are attached:

- ACRS report to the Commission dated March 18, 1993. Subject: Computers in Nuclear Power Plant Operations (PP.66-71)
- Chairman Selin's letter to Drs. Lewis, Kress and Wilkins dated February 23, 1993. Subject: Response to Members Letter of December 11, 1992 (PP.72-73)
- Individual ACRS Members letter to Chairman Selin dated December 11, 1992. Subject: Comments on Digital Systems (PP.74-75)

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- ACRS report to the Commission dated September 16, 1992. Subject: Digital Instrumentation and Control System Reliability (PP.76-79)
- ACRS letter to James M. Taylor (EDO) dated September 16, 1992. Subject: Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs" (PP.80-83)

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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. Electronics 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,

A handwritten signature in cursive script that reads "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.

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

February 23, 1993

Dr. Harold W. Lewis
Dr. Thomas S. Kress
Dr. J. Ernest Wilkins
Members, Advisory Committee
on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Drs. Lewis, Kress, and Wilkins:

On behalf of the Commission, I am responding to your letter of December 11, 1992, in which you commented on the staff's regulatory program plan for digital instrumentation and control (I&C) systems in nuclear plants. The Commission shares the view that the staff should take advantage of a broad range of expert opinion in formulating policy on the safety of digital I&C systems. The staff has informed the Commission that they will sponsor a workshop this year involving both domestic and international experts from a number of industries, including nuclear, aerospace, defense, and telecommunications. The workshop will focus on digital systems and software reliability related to nuclear safety, including issues such as verification and validation techniques, fault avoidance techniques, and functionally diverse designs. As planning for the workshop proceeds, the staff is obtaining suggestions for the program from internationally recognized people in this field.

The staff has benefited from the advice from industry experts involved with computer system reliability and will continue to interact with experts in both the nuclear and other industries. In July 1992, the staff and the contractor held a workshop with an expert panel consisting of Bev Littlewood, City University, London; Nancy Leveson, University of California; John Rushby, Stanford Research Institute; and Ricky Butler, National Aeronautics and Space Administration (NASA), to discuss the development of highly reliable software.

In addition to the planned workshop, the staff plans to conduct a session on the regulatory aspects of digital systems during the May 1993 Regulatory Information Conference. The staff will also hold further discussions

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with NASA and the Department of Defense to resolve digital system design issues, particularly issues related to electromagnetic and radio-frequency interference (EMI/RFI).

We appreciate your comments and welcome your continued advice in this area.

Sincerely,

A handwritten signature in black ink, appearing to read "Ivan Selin". The signature is fluid and cursive, with the first name "Ivan" and last name "Selin" clearly distinguishable.

Ivan Selin



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

December 11, 1992

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

Dear Chairman Selin:

Though you know us as members of the ACRS, this is a personal letter, not approved by the Committee. It reflects only our own views.

Over the last 22 months, the ACRS Subcommittee on Computers in Nuclear Power Plant Operations has been running a series of meetings to explore the regulatory and safety implications of the trend toward digital technology in both existing and proposed nuclear reactors. In this effort it has sampled the views of the industry, the staff, the vendors, and a number of other elements of the community, both in and outside of the nuclear world. It has also heard, less expansively of course, from both regulatory and user groups outside the United States.

While the ACRS has not yet formulated a focused set of recommendations, the picture has become sufficiently clear to us to permit one early proposal. Given the uncontested fact that the potential for harm from a control-system failure, digital or otherwise, is large, we wish to pass this one along independently. Of course, the potential for safety benefit is also large, as emphasized in the ACRS letter of September 16, 1992, on Digital Instrumentation and Control System Reliability.

The impression we have received from the subcommittee meetings is that the NRC lags its counterparts in other countries, and probably even the industry it regulates, in the depth with which it addresses both the risks and benefits generated by computer systems. The ACRS letter on the proposed requirement to back up digital systems with analog systems addressed one small symptom of the problem; there are many others. It is of course impossible to deal with a general problem piecemeal, just as one can't turn country music into Bach one note at a time. We think the NRC needs a program plan to provide a stronger sense of direction, aimed at realistic but clear objectives. In particular we know that there is a large, accessible, and competent community of people concerned with and expert in these matters—the problems pervade industry and business—and the staff has had only limited contact with them.

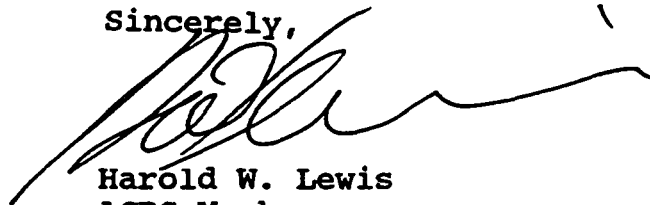
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We know no magic solution to the problem of inadequate resources, but think it would be helpful if the staff (and the Commission) were exposed in some depth to the current thinking of the computer reliability experts, who have for years been hard at work for the aerospace community, the telephone company, the bankers and money-handlers, the Defense Department, etc. The problems are in each case unique to the community involved, but with features common to all.

We recommend that you consider funding a short workshop on the relationship between digital systems reliability and nuclear safety by the National Academies of Sciences and Engineering, which are able to direct the attention of the best technical experts available toward the specific problems of an agency, at an acceptable cost, while preserving both the appearance and the substance of independence. We do not recommend a full study by the Academies—such efforts can proceed with glacial speed—but the kind of two- or three-day workshop, followed by a report, that has worked well on other problems. One would not be looking here for a detailed charter for the future, but rather for an orientation, focus, and state-of-the-art document, from which real planning can proceed. The Academies are uniquely suited to play such a role, with no problem of conflict of interest.

We have eschewed the temptation to provide in this letter a proposed set of Terms of Reference for such a workshop, but are convinced that a reasonable set can be formulated. It would be best if these terms not reflect only the NRC's own perception of its needs; breadth of perspective is essential to a fresh view. Each of us would of course be happy to assist in drawing up an acceptably focused suite of objectives and deliverables, which should be done in the usual way, through negotiation with the Academies.

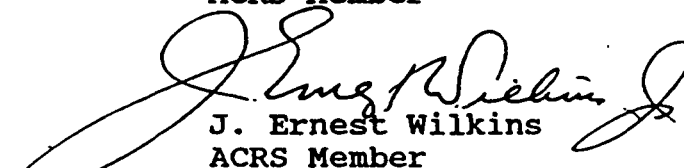
Sincerely,



Harold W. Lewis
ACRS Member



Thomas S. Kress
ACRS Member



J. Ernest Wilkins
ACRS Member

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

September 16, 1992

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

Dear Chairman Selin:

SUBJECT: DIGITAL INSTRUMENTATION AND CONTROL SYSTEM RELIABILITY

During the 389th meeting of the Advisory Committee on Reactor Safeguards, September 10-12, 1992, we reviewed the staff's proposed approach with respect to defense against common-mode failure of digital I&C systems, as discussed in policy issue "A" of the draft Commission paper entitled, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs," forwarded to the Commission on June 25, 1992. Specific comments on policy issue "A" are contained in a letter to Mr. Taylor dated September 16, 1992. The concerns we raise here are, however, more generally applicable, e.g., in connection with the staff's proposed generic letter on analog-to-digital replacements.

The trend in most industries over the last few decades has been toward the replacement of analog instrumentation and control systems with digital alternatives, and the nuclear industry has been no exception. This has been true for both functional replacements within existing nuclear facilities and for new designs, so it has been necessary for the staff to develop regulatory practices to deal with both the novel opportunities and the novel threats posed by these systems.

Experience, both military and industrial, has generally shown the digital systems to be more reliable and versatile than their analog counterparts. There are, however, some caveats and some regulatory conundrums. An advantage is that the digital systems are capable of more complex functions, so it is possible to build in self-testing capabilities that provide continuous assurance of operability with negligible system stress. In addition, the digital systems don't wear out; a billion activations of a CMOS gate are no more damaging than a thousand. While much has been made of the vulnerabilities of multiplexed data transmission systems, some of which are doubtless real, such systems generally provide greater fidelity and reliability of data transfer, along with greater fault

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tolerance through error-correcting coding. (If an analog signal is corrupted, it is often not possible to know it has happened.) Indeed, error detection and error correction can be carried to arbitrary lengths for digitized data. There are many other advantages, and the future clearly belongs to digital systems, where they can be used.

On the negative side, the available complexity of function afforded by digital systems invites the creation of complex software, which can be difficult to validate and can be subject to surprising error modes. Such systems are also hard to regulate, because only the simplest programs are amenable to formal validation and verification (V&V), in the sense of a complete analysis of the mapping of the input space to the output space. For more complex programs (relevant to nuclear control systems, but not necessarily to instrumentation or safety actuation systems), there are many analytical techniques in use, none perfect. That is also true of analog systems. Solid-state systems, whether digital or analog, are also peculiarly vulnerable to environmental damage, e.g., from overheating. Finally, programmable digital systems have their own special vulnerabilities to human error.

The staff has concentrated its attention on one of these many issues, the vulnerability of digital systems to certain kinds of common-mode failures, principally through programming errors introduced into the software, and therefore common to all channels.

To deal with this supposedly special susceptibility to common-mode failure, the staff has proposed a set of regulatory requirements. The set includes some unarguable items, like the provision of adequate diversity to cope with common-mode failures that can affect safety systems, and analysis of the appropriate accident sequences. The set also includes some items whose desirability is less clear, and we now turn to these. Since each of these would require an extensive discussion to develop the point completely, and since our recommendation is that the staff revisit all these points, we will be brief. There is no special order.

The lack of explicit and quantifiable safety standards for instrumentation and control systems is particularly troublesome here. The staff speaks of reliability for digital systems in the same terms (failures per demand) that it uses for items which do wear out, like relays and switches. The entirely different failure mechanisms make this an inappropriate transfer of terminology. Indeed, a simple software-based system, in which the hardware is kept within its environmental constraints, and whose software is simple enough to have been subjected to a full validation and verification (in the sense used above) can be expected to never fail. (Never is only a slight exaggeration.) The failure anecdotes we all know are typically in systems that are too complex for formal V&V, leaving the door open to software errors, or have

been mistreated, opening the door to hardware failures. The latter problem is not unique to digital systems.

In view of the lack of explicit standards for the reliability of the digital systems, the staff seems to have drifted to what has been called the "bring me a rock" posture, in which the industry is asked to analyze its own vulnerabilities, after which the staff will make its ruling about the adequacy of the design. The spirit of the safety-goal initiative was presumably to help make regulation more predictable, and this approach is clearly in the other direction.

The focus on common-mode failures is troublesome. Software errors in single systems can lead to accidents just as serious as those due to common-mode failures in redundant systems, and the entire question of software reliability greatly transcends the issues raised here. We have been conducting a coordinated series of meetings on the safety issues involved in the inevitable computerization of the industry, already in progress. When we report on these, we will doubtless raise the question of whether sufficient talent, both in quantity and in experience, is being directed at these issues by NRC. That question is also an underlying issue here.

For the specific issue of protection against common-mode failures, whether for digital systems or such devices as diesel generators, there is a set of standard prophylaxes like diversity and defense in depth, which are useful when applied sensibly. (Slogans can be overplayed. It makes no sense to insist that multi-engine aircraft have a suitable mix of turbine and piston engines.)

The most controversial specific position taken by the staff is that there must be a safety-grade set of displays and controls located in the control room, independent of the computer systems, and "conventionally hardwired" to the lowest level practicable. Though the intent of the words in quotations is unclear, we were assured that it was to require analog backup systems. We do not concur in this proposed requirement. We think that the staff is unnecessarily mixing up the issues of digital/analog, hard wire/multiplex, and software/hardware.

Each instrumentation and control system that is important to the safety of a plant ought to meet some identifiable standard of reliability and fault tolerance, regardless of the hardware/software basis used in designing and fabricating the system. It is not necessary that any given element of the system be perfect, but that the system as a whole meet some recognized standard, presumably in the form of a relevant surrogate for the Commission's safety goals. Both the identification of that standard and the evaluation of conformance for the system in question pose problems, but each should somehow be completed

before, not after, a regulatory position is established. For example, the staff proposes to require that a backup system provide protection equivalent to that of the primary system, whereas the need is for sufficient protection to assure the adequate safety of the plant. It is not at all uncommon for backup systems to be designed to lower standards than the primaries, taking into account the fact that they will be called upon less often. (Consider spare tires.)

It is entirely possible that a digital system may turn out to be a better backup than an analog system. (The proposed position does accommodate this idea, but the staff briefings did not.) For some situations a light beam is a more reliable means of communication than a hard wire. A general-purpose microprocessor that is in widespread commercial use may be more reliable (and more thoroughly tested) than a special-purpose analog switch. And so forth.

In each case it is necessary to make a specific reliability analysis, measured against a reasonable standard, and the staff gave no evidence of having done so for any case. Instead, it has adopted a general requirement for an analog backup for all cases, and we were not convinced by the justification provided.

We recommend that the staff revisit these issues, augment its own capabilities, and broaden its interaction with those elements of the outside world who have previously dealt with such problems. It would be unwise, however, to read too literally into the nuclear arena the considerations that are relevant to far more complex systems. We are dealing here with the relatively simple safety-centered parts of the computerized instrumentation and control system, and an architecture that exploits this fact may be more robust.

Sincerely,



David A. Ward
Chairman

References:

1. Memorandum dated June 25, 1992, from James M. Taylor, Executive Director for Operations, NRC, for The Commissioners, Subject: Review of the Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs"
2. 57 Federal Register, 36680, August 14, 1992, Proposed Generic Communication; Analog-to-Digital Replacements Under the 10 CFR 50.59 Rule



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

September 16, 1992

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

Dear Mr. Taylor:

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

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

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

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

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

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

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

B. Analyses of External Events Beyond the Design Basis

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

C. Elimination of the Operating Basis Earthquake from Seismic Design

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

D. Multiple Steam Generator Tube Ruptures (MSGTRs)

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

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

E. Probabilistic Risk Assessment (PRA) Beyond Design Certification

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

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

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

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

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

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

G. Control Room Annunciator (Alarm) Reliability

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

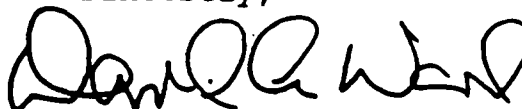
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September 16, 1992

H. Regulatory Treatment of Nonsafety Systems

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

Sincerely,

A handwritten signature in black ink, appearing to read 'David A. Ward', with a stylized, cursive script.

David A. Ward
Chairman

Reference:

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

ITEM 3: LICENSE RENEWAL

The Committee has provided six reports to the Commission or the EDO regarding the staff's approach toward the proposed rule on plant license renewal and its implementation. The Committee was briefed by the staff at the April 1993 meeting on the outcome of the staff's senior management review of key licensing renewal issues and proposals for implementing the provisions of 10 CFR 54. The Committee remains much interested in this matter and plans to review it again as additional information becomes available.

The following documents are attached:

- ACRS report to the Commission dated April 23, 1993. Subject: SECY-93-049, Implementation of 10 CFR Part 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants (PP.85-86)
- ACRS letter to James M. Taylor (EDO) dated October 22, 1992. Subject: Proposed Branch Technical Position on Environmental Qualification of Electrical Equipment for License Renewal (PP.87-89)
- ACRS letter to James M. Taylor (EDO) dated August 17, 1992. Subject: Proposed Regulatory Guide and Interim Standard Review Plan for License Renewal and a Related Branch Technical Position on Fatigue Evaluation Procedures (PP.90-93)
- ACRS report to the Commission dated April 17, 1991. Subject: Draft Final Rule on Nuclear Power Plant License Renewal (PP.94-95)
- ACRS letter to James M. Taylor (EDO) dated October 11, 1990. Subject: Draft Implementation Documents for the Proposed License Renewal Rule (PP.96-98)
- ACRS report to the Commission dated April 11, 1990. Subject: Proposed Rule on Nuclear Power Plant License Renewal (PP.99-100)



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

April 23, 1993

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

Dear Chairman Selin:

SUBJECT: SECY-93-049, IMPLEMENTATION OF 10 CFR PART 54,
REQUIREMENTS FOR RENEWAL OF OPERATING LICENSES FOR
NUCLEAR POWER PLANTS

During the 395th and 396th meetings of the Advisory Committee on Reactor Safeguards, March 11-12 and April 15-17, 1993, we discussed with the NRC staff its proposal in SECY-93-049 for implementing the License Renewal Rule. During our April meeting, we also heard from representatives of NUMARC on this matter. We had the benefit of the documents referenced.

In a number of our past reports, we provided comments and recommendations on various aspects of the License Renewal Rule including our recommendation that this rule and the Maintenance Rule be better integrated in the interest of long-term coherence of the regulatory process. We have also commented that the operational phase reliability assurance program required of applicants licensed under 10 CFR Part 52 needs to be integrated with these two rules. Additionally, we have strongly opposed the staff's proposals to use license renewal as a means of dealing with such issues as electrical cable qualification and mechanical component fatigue life on the occasion of a plant's 40th birthday, when, in fact, these issues potentially affect presently operating plants that may or may not seek license renewal. We continue to support these views.

The staff's recent efforts to develop an approach to facilitate a more effective and efficient license renewal process that would rely heavily on use of the requirements of the Maintenance Rule appear to have the potential for making significant improvements in this process. However, we believe further dialogue between the staff and the industry on this matter is needed so that the many subtle issues involved in this approach are fully explored.

As we understand the present situation, the Commission's major concern is whether (1) the present License Renewal Rule can be legally construed to accommodate the staff's proposal in SECY-93-049 or (2) there is a need to revise or formally interpret the existing rule and its accompanying statements of consideration to

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accurately reflect the course of action proposed by the staff. From a policy point of view, we believe that this Commission's thinking on the formulation of the ultimate implementation of the License Renewal Rule needs to be documented by a policy statement, interpretive rulemaking, or a revision to the rule and its statements of consideration. Based on our discussions with the staff and NUMARC, it appears that the needed time is available to revise the rule without significantly impacting licensees' schedules for making decisions regarding license renewal.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. SECY-93-049, dated March 1, 1993, for the Commissioners, from James M. Taylor, Executive Director for Operations, NRC, Subject: Implementation of 10 CFR Part 54, Requirements for Renewal of Operating License for Nuclear Power Plants
2. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Guidance for Implementation of the Maintenance Rule, 10 CFR 50.56, October 15, 1992
3. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Regulatory Guide and Interim Standard Review Plan for License Renewal and a Related Branch Technical Position on Fatigue Evaluation Procedures, August 17, 1992



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

October 22, 1992

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

Dear Mr. Taylor:

**SUBJECT: PROPOSED BRANCH TECHNICAL POSITION ON ENVIRONMENTAL
QUALIFICATION OF ELECTRICAL EQUIPMENT FOR LICENSE
RENEWAL**

During the 390th meeting of the Advisory Committee on Reactor Safeguards, October 8-10, 1992, we reviewed a proposed Branch Technical Position (BTP) on Environmental Qualification of Electrical Equipment for License Renewal. Our Subcommittees on Plant License Renewal and Reliability and Quality reviewed this matter during a joint meeting on September 16, 1992. The staff proposes that the BTP be issued for public comment. During these meetings, we had the benefit of discussions with members of the NRC staff, its consultants, and representatives of industry. We also had the benefit of the documents referenced.

Under the License Renewal Rule, 10 CFR Part 54, applicants will be required to develop a comprehensive program to identify in their plants all structures, systems, and components (SSCs) which may be subject to age-related degradation unique to the license renewal period. A further program to manage these components to ensure continued safe operation of the plant is also required. The staff is now proposing an additional program, by means of a BTP, which singles out environmental qualification of electrical equipment for special treatment in the license renewal period. The particular concern of the staff seems to be that the qualification standards for insulation used on electrical cables prior to 1984 (representing 87 of 111 licensed nuclear power plant units) may not ensure adequate performance of cables for extended plant life. That, of course, is the issue for all SSCs in a plant, and it is not clear to us why the more general treatment of SSCs called for under 10 CFR Part 54 is not adequate for electrical cables as well.

Industry representatives expressed objection to the staff proposal for a BTP. They believe that while older plant cables were qualified to a lesser standard than has been in use since 1984, these cables have been approved for continued use in the plants (as

has much other equipment where standards have evolved) and are part of the Current Licensing Basis (CLB) for each of these plants. Their interpretation of 10 CFR Part 54 is that the CLB is to be preserved with the exception that those SSCs subject to age-related degradation unique to the license renewal period should be subjected to specific management programs. They see no need for the BTP and believe it will result in unnecessary cable replacements and add significantly to plant costs for license renewal.

We are not convinced that the proposed BTP has been shown to be necessary or appropriate. It should not be issued for public comment until the matters discussed below have been addressed.

Neither the staff nor the industry presented any risk perspective on this issue. In simple terms, the risk is as follows: During the license renewal period the electrical cable in a key system might degrade in a way that the degradation would remain undetected during normal operation and by normal maintenance, testing, and surveillance practices. Then, during an accident, i.e., a LOCA, the insulation would fail and the key system would not perform its design function to mitigate effects of the accident. Present licensing practice assumes, and experience seems to confirm, that the probability of this sequence during the initial license period is acceptably low. At issue is whether the probability during the license renewal period is significantly greater. No evidence has been presented either way. Analysis of the risk importance of this issue should be made before the BTP is finally accepted or rejected. Such an analysis should include estimates of downside risks inherent in major projects intended to improve nuclear power plant safety.

Many electrical cables are covered with fire retardant materials. These coatings could have important effects on the aging of the cable insulation. Apparently, these effects have not been considered by the staff in development of this BTP. We do not know whether they have yet been explicitly considered in the selection and evaluation of important SSCs in license renewal programs. They should be.

Dr. Thomas Kress did not participate in the Committee's deliberations regarding this matter.

Sincerely,



David A. Ward
Chairman

References:

1. Memorandum dated July 10, 1992, from John W. Craig, Office of Nuclear Reactor Regulation, NRC, for Raymond F. Fraley, Advisory Committee on Reactor Safeguards, Subject: Request for Review of Branch Technical Position on Environmental Qualification of Electrical Equipment for License Renewal, with enclosures
2. Letter dated October 7, 1992, from M. H. Philips, Jr., and W. A. Horin, Counsel to the Nuclear Utility Group on Equipment Qualification, to D. A. Ward, Advisory Committee on Reactor Safeguards, Subject: NRC Staff Proposed License Renewal BTP Regarding Environmental Qualification of Electric Equipment, with enclosures



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

August 17, 1992

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

Dear Mr. Taylor:

SUBJECT: PROPOSED REGULATORY GUIDE AND INTERIM STANDARD REVIEW
PLAN FOR LICENSE RENEWAL AND A RELATED BRANCH TECHNICAL
POSITION ON FATIGUE EVALUATION PROCEDURES

During the 387th and 388th meetings of the Advisory Committee on Reactor Safeguards, July 9-11 and August 6-8, 1992, we reviewed a proposed Regulatory Guide (DG-1024) and an interim Standard Review Plan (SRP) (NUREG-1299) to be used in plant license renewal. We also considered a proposed Branch Technical Position (BTP) (an appendix to the SRP) on fatigue evaluation procedures which would provide a basis for license renewal reviews. These matters were also considered during a joint meeting of our Subcommittees on Plant License Renewal and Materials and Metallurgy on July 7, 1992. During these meetings, we had the benefit of presentations by the NRC staff, its consultants, and representatives of industry. We also had the benefit of the documents referenced.

We commented on the earlier version of the Regulatory Guide and on the interim SRP in our report of October 11, 1990, and on an early version of the License Renewal Rule in our report of April 11, 1990. Since these reports were issued, the NRC staff has issued a final rule, which incorporates some significant changes from the earlier version, and has also received and evaluated public comments on the 1990 Regulatory Guide and the SRP proposals. The now-proposed Regulatory Guide and SRP are intended to reflect the final rule, the public comments, and ACRS comments. Because there have been significant changes in these documents, the staff proposes to publish them for another round of public comments.

In addition, in its development of the license renewal process, the staff has identified a concern about the adequacy for extended service of some existing plant components in accommodating metal fatigue. Generally, plants were designed with a 40-year life expectation. Operation for 60 years could mean that some fatigue limits would be exceeded. A BTP has been proposed as a basis for staff evaluation of fatigue status in the license renewal process.

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In general, the new versions of the Regulatory Guide and SRP seem to be improvements and appropriately reflect changes made in the rule. We have no objection to their being published for public comment. We have, however, three comments at this time:

1. The major concern expressed in our October 11, 1990 report was about control of the process for selection of Structures, Systems and Components (SSCs) "important to license renewal." Without adequate constraints, reviewers are likely to expand the list of SSCs beyond that needed to provide reasonable assurance that aging of important plant systems will be adequately controlled. This could be carried to a point of being unnecessarily burdensome on licensees, thereby discouraging plant owners from seeking license renewals. We were told that this concern was being addressed by improving the definition of SSCs. We have reservations about whether this is adequate and retain our concern.

We were told that estimates by management of the lead plant involved in the license renewal effort indicate that 65 percent of the components in the plant will be on the SSC list and that the cost of developing the required program could be as much as \$25 million. The length of the SSC list is obviously a substantive issue. We believe some better mechanism for control should be established. Creation of a review function for each plant's SSC list at a senior level within the agency, perhaps something similar to the CRGR, should be considered.

2. Requirements imposed on licensees with the Maintenance Rule have much commonality with requirements under the License Renewal Rule. We asked the staff presenters whether consideration had been given to combining or at least coordinating the two rules. Apparently none has been. We believe this is a mistake. Requirements for the two rules have been developed in different branches of the Office of Nuclear Reactor Regulation. The two sets of requirements have somewhat, but not greatly, different scopes and purposes. We were told that a licensee who decides to apply for license renewal and meets the requirements of the rule will not automatically meet the Maintenance Rule because of some scope differences. Nor will the opposite be true. Thus, a licensee will have to meet both sets of requirements over the life of a plant.

One interesting difference in the proposed implementation of these two rules is in how PRA would be used. With the Maintenance Rule, risk arguments will be accepted by the staff for either excluding or including specific items. In the case of the License Renewal Rule, we were told that risk arguments

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would be accepted by the staff for inclusion but not for exclusion of specific items in the plant from the SSC list.

We note that a reliability assurance program is being developed for the ABWR and understand that such programs will be required for all ALWRs. These programs should be coordinated with the requirements of the License Renewal and Maintenance Rules.

We have often commented on the need for greater coherence among the many parts of the overall fabric of NRC regulatory policies and practices. The Commission has recently spoken on the need to reduce unnecessary regulatory burden on licensees. We recommend that before the Maintenance Rule and License Renewal Rule are implemented, a comprehensive study be carried out to determine if combining the rules would foster the aims of increased regulatory coherence and reduced regulatory burden.

3. The BTP on metal fatigue appears to require more of licensees than is justified. The BTP would require evaluation of the fatigue life (cumulative usage factor, CUF) in certain components of the reactor coolant pressure boundary based on 60 years of service. The actual transient history could be used, resulting in a lower CUF than that assumed in the original design, but it would require the replacement of any component which exceeded one-third of the ASME Section III design life. Calculating the CUF can be time-consuming, and using it in the way suggested by the BTP will usually require the replacement of components which would otherwise perform satisfactorily for the remaining life of the plant.

The industry position is that the calculated CUF should not be regarded as an absolute service limit. Industry spokesmen suggested that equally appropriate and more economical approaches are available and should be used. We believe the staff proposal is unreasonable. A better approach would be to use the procedures of ASME Section XI for inspection and repair. Consideration might be given to requesting a clarification from the appropriate ASME Code Committee on what it believes should be done if CUF approaches 1.0 for a component. This would be time-consuming, but time does seem to be available in this instance.

August 17, 1992

Additional comments by ACRS Members William Kerr, Thomas S. Kress, Harold W. Lewis, and Charles J. Wylie are presented below.

Sincerely,



David A. Ward
Chairman

Additional comments by ACRS Members William Kerr, Thomas S. Kress, Harold W. Lewis, and Charles J. Wylie.

We do not wish to address the details of the elaborate regulatory structure the staff proposes to erect to support its review of license extension applications for currently operating nuclear plants, but only to provide some perspective. We are impressed by the contrast between the licensing of nuclear plants and of other complex systems, like aircraft. Both nuclear plants and aircraft are complex systems, each synthesized from components of a wide variety of effective lifetimes, rates and modes of degradation, and importance to the safety of the system. In the nuclear case, a term license is issued for the system—in the aviation case the aircraft airworthiness certificate is permanent, provided the maintenance and replacement of the aging components are managed in a timely and effective manner. That seems to us to be far more effective, and is consistent with the Committee's recommendation to coordinate this plan with the Maintenance Rule. The purpose of licensing is to ensure and maintain the protection of the public health and safety—it is not an end in itself.

Of course we recognize that initiatives in this matter are constrained by the terms of the Atomic Energy Act, but it is not unthinkable that laws can be adjusted if it is in the public interest to do so. This was not an important matter forty years ago—it is now.

References:

1. Memorandum dated June 10, 1992, from John W. Craig, Office of Nuclear Reactor Regulation, NRC, for Raymond F. Fraley, Advisory Committee on Reactor Safeguards, Subject: Request for Review of Branch Technical Position on Fatigue for License Renewal, with enclosures
2. Memorandum dated June 5, 1992, from John W. Craig, Office of Nuclear Reactor Regulation, NRC, for Raymond F. Fraley, Advisory Committee on Reactor Safeguards, Subject: Request for Review of Interim Regulatory Guide and Standard Review Plan for License Renewal, with enclosures



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

April 17, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: DRAFT FINAL RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL

During the 372nd meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1991, we reviewed the draft of the final rule on nuclear power plant license renewal (10 CFR Part 54). Our Subcommittee on Plant License Renewal discussed this matter during its April 8, 1991 meeting. During our consideration of this matter, we had the benefit of discussions with representatives of the NRC staff, NUMARC, and Northern States Power Company. The latter is the licensee for the Monticello Nuclear Generating Plant, which is a lead plant in the license renewal program. We also had the benefit of the document referenced.

The ACRS reported to you on the proposed license renewal rule in its report of April 11, 1990. Since that time, the proposed rule was published for public comment. The staff received 197 comments. It has assimilated information from these comments and information received in a number of interactions with industry and has prepared a draft final rule. The schedule calls for the final rule to be published by June 28, 1991, and for other parts of the rulemaking package, a regulatory guide and a standard review plan, to be published about one year later.

As stated in our April 11, 1990 report, we concur with the approach being taken by the staff in this rulemaking. However, there are two areas of disagreement between the staff and NUMARC that we would like to bring to your attention. The first might require a modification in the draft final rule. The second is related to implementation of the rule.

The first matter is an issue on which we do not have a recommendation except that it should receive your consideration. The draft final rule requires that each applicant for license renewal develop a "compilation" of its Current Licensing Basis. Although it is not precisely clear what this means, it was agreed that it would, at a minimum, include a list of all licensing commitments agreed to by the applicant over the history of its plant. Industry representatives believe this is unnecessary.

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April 17, 1991

The second issue is how implementation of the rule will be limited in scope to concentrate resources for aging management where needed. The rule would require that each applicant develop a list of Systems, Structures, and Components Important to License Renewal (SSCITLR) and then implement an aging management program appropriate for items on that list. The staff's position is that the original SSCITLR list should include all those items in the plant that play a role in meeting any docketed commitment the licensee has made. This would include the original license; commitments related to new rules as they came into being; and commitments made in response to Safety Evaluation Reports, Information Notices, Bulletins, Generic Letters, and Orders.

The industry representatives told us that such a definition of SSCITLR would result in a list that includes 85 to 90 percent of all equipment in the plant. They believe that application of a special aging program to all of these items would be unnecessary and onerous. The process of reducing the initial SSCITLR list to just those items to be covered by a special aging program is critical. Items important to implement other commitments would not thereby be ignored. They would be maintained through the new license period just as they are now.

We believe that selection of those items to be subjected to a special aging program should be based on technical rather than legal argument. Our understanding is that a program of this nature can be developed with the rule as presently drafted. However, implementation will require careful crafting of the regulatory guide and the standard review plan. We would like the opportunity to review these documents before they are issued.

Sincerely,



David A. Ward
Chairman

Reference:

Memorandum dated March 6, 1991 from Warren Minners, Office of Nuclear Regulatory Research, to Raymond F. Fraley, ACRS, Subject: Final Rule on Nuclear Power Plant License Renewal, with enclosures (Predecisional)



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

October 11, 1990

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

Dear Mr. Taylor:

SUBJECT: DRAFT IMPLEMENTATION DOCUMENTS FOR THE PROPOSED LICENSE
RENEWAL RULE

During the 366th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 1990, we reviewed draft Regulatory Guide, Task DG-1009, "Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses," and associated draft NUREG-1299, "Standard Review Plan - License Renewal." Our Subcommittee on Plant License Renewal also reviewed this matter during its meeting on October 2, 1990. During this review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced. These documents are an important part of the program to implement the proposed license renewal rule, 10 CFR Part 54, that was published for public comment on July 17, 1990. We commented to the Commission on this proposed rule in our report of April 11, 1990.

We believe that the general approach proposed by the staff for implementation of the license renewal process is reasonable, and we agree that both of the subject documents should be published at this time for public comment. However, we have a concern, discussed below, about control of the process for selecting structures and components important to license renewals (SCITLRs). We believe that this matter should be considered further as public comments on the rulemaking are evaluated. We also offer several comments on the implementing documents.

There is justification for the general philosophy of the proposed license renewal rule. Aging-degradation issues should be dealt with by more explicit programs as the plant age passes beyond the general target age for which it was designed. Our understanding is that a 40-year operating life has been used for most structures and components in nuclear power plants. However, that target age and the design were not so precisely defined that there should be a step increase in licensing requirements as the plant passes its 40th anniversary of operation. As we said in our April 11, 1990 report, "no specific form of plant aging becomes magically decisive at forty." We have a concern that the license renewal process under the proposed 10 CFR Part 54 will permit or encourage a

significant expansion of regulatory requirements as a plant phases into operation under a renewed license. We had hoped and expected that the implementing documents would provide some clear indications of how such regulatory expansion would be constrained. They do not. Introductory material in the proposed 10 CFR Part 54 indicated that the backfit rule would somehow be used in controlling the extent to which regulatory requirements would be expanded. However, the rule itself does not make it clear how this is to be done, nor do the draft implementing documents. We recommend that the rule or the implementing documents be revised to ensure that the process for selecting SCITLRs and developing new requirements is sufficiently disciplined.

In addition, we have several specific comments on the proposed implementing documents:

- (1) In the proposed process for evaluating age-related degradation, the draft Regulatory Guide indicates that a decision about classification of a given structure or component should be made on the basis of whether the structure or component is routinely replaced or refurbished (see Block 12 of Figure 1B in the draft Regulatory Guide). We recommend that satisfactory results of inspection or monitoring should also be credited at this decision point.
- (2) Many of the unresolved safety issues and generic safety issues that have been analyzed over the past several years have had assumptions about expected plant life factored into their resolution. The staff has indicated that, in general, an expected life of 60 years instead of 40 years would make little difference in cost-benefit analyses, given the large uncertainty inherent in the calculated results. However, the staff also indicated that a review of all such resolutions will be made, in the light of new expectations about plant lifetimes, given the changes of 10 CFR Part 54. We would like to be kept informed about the results of this review.
- (3) Certain industry topical reports on the subject of aging degradation are being developed by NUMARC, and are expected to be approved by the staff as acceptable references in license renewal applications. We encourage the development of these industry reports as a means of providing a comprehensive technical base for license renewal reviews. Because the license renewal process can be expected to extend over many years, much technical information about aging will be in need of revision, and some means for formally updating these industry reports and their approval by the NRC should be provided.
- (4) Perspectives gained from applicable risk assessment should be used in the selection of SCITLRs.

- (5) Consideration should be given to including physical security systems in the SCITLR program.

We plan to continue our review of this important subject after public comments on this proposed rule, the Regulatory Guide, and the proposed Standard Review Plan are received and assimilated.

Sincerely,



Carlyle Michelson
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Draft Regulatory Guide, Task DG-1009, "Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses," Revision 5A dated August 1990, and U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Draft NUREG-1299, "Standard Review Plan, License Renewal," dated August 1990, transmitted by memorandum dated August 31, 1990, from Eric S. Beckjord, RES, and Thomas E. Murley, NRR, to Raymond F. Fraley, ACRS
2. U.S. Nuclear Regulatory Commission, Rules and Regulations, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Proposed Rule Making, Published July 17, 1990



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

April 11, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL

During its 360th meeting, April 5-7, 1990, the Advisory Committee on Reactor Safeguards reviewed the staff's proposed rule on nuclear plant license renewal. This matter was also discussed during a meeting of the Regulatory Policies and Practices Subcommittee on March 26, 1990. During these meetings we had the benefit of discussions with representatives of the NRC staff, NUMARC, EPRI, Northern States Power Company, and Yankee Atomic Electric Company. We also had the benefit of the referenced document and its enclosures.

The decisive issues for license renewal and associated plant aging, and the potential for further aging during the proposed license extension, should be addressed throughout the life of a plant. Attention to aging phenomena, and the criteria for safe operation (adequate protection of the health and safety of the public), should be the same just after as just before license renewal. There may be components or systems which are not aging issues during the first forty years, but become so later, and which therefore may require special attention.

At the time that the forty year period for a license was chosen, there was no special technical rationale for its choice, and no specific form of plant aging becomes magically decisive at forty. The regulatory job for license renewal is to identify the aging elements of the plant, and ensure that they receive timely attention during the extended license period.

In that context, we were surprised by the lack of emphasis on pressure vessel integrity during our briefings. This is surely one of the driving technical issues for extended life, and we assume that it will move to a more central position as the plans develop.

April 11, 1990

The staff proposes to use the "current licensing basis" of a plant as the basis for license renewal, but there seems to be some ambiguity about the interpretation of the term. The industry seems concerned that this may provide an opportunity to impose arbitrary new requirements. It is important that this terminology be clarified, so that any future conflicts of interpretation are minimized.

With these observations, we concur in the approach being proposed by the staff, which emphasizes attention to aging phenomena, avoids the temptation to treat license extensions as relicensing, and makes a timely start toward providing an integrated policy for dealing with aging phenomena.

Sincerely,



Carlyle Michelson
Chairman

Reference:

Memorandum dated March 6, 1990 from Warren Minners, Office of Nuclear Regulatory Research, NRC, to Raymond F. Fraley, ACRS, Subject: Proposed Rule on Nuclear Power Plant License Renewal, w/enclosures: Draft Commission Paper, "Proposed Rule on Nuclear Power Plant License Renewal"

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