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UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of:

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,
SUBCOMMITTEE MEETING ON BABCOCK AND
WILCOX REACTORS

Place: Washington, D. C.

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UNITED STATES

NUCLEAR REGULATORY COMMISSION

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Room 1167
1717 H Street, N.W.
Washington, D.C.

Tuesday, April 29, 1980

The Advisory Committee on Reactor Safeguards, Subcommittee on Babcock & Wilcox Reactors, met, pursuant to notice, at 1:00 p.m., Mr. Etherington, Chairman of the Subcommittee, presiding.

PRESENT:

Mr. Mathis

Dr. Lawroski

Mr. Tam

Mr. Ray

Mr. Ebersole

Dr. Zudans

Mr. Tedesco

Mr. Capra

Mr. Taylor

Mr. Thatcher

P R O C E E D I N G S

CHAIRMAN ETHERINGTON: The Subcommittee on B&W Water Reactors.

I'm Harold Etherington, Subcommittee Chairman.

The other ACRS members present today are Mr. Ebersole, Mr. Ray; and we're expecting Mr. Mathis and Dr. Lawroski later in the afternoon.

We have also present today as consultant Dr. Zudans.

The purpose of this meeting is to review NUREG-0667, Transient Response of Babcock and Wilcox designed reactors. The report was published an NRC task force formed to study the apparent high frequency of transients at B&W plants.

This meeting is being conducted in accordance with the Federal Advisory Committee Act and the Government in the Sunshine Act.

It may be necessary for the Subcommittee to hold one or more closed sessions for the purpose of exploring matters involving proprietary information.

Mr. Peter Tam, on my right, .is the designated Federal Employee for this meeting.

The rules for participation in today's have been announced as part of the notice of this meeting previously published in the Federal Register on April 14, 1980.

A transcript of the meeting is being kept, and it is requested that each speaer first identify himself or herself and speak sufficient clarity and volume that he or she can be heard readily.

We have received numerous statement on requests for time to make oral statements from many members of the pblic. I don't feed and hear feedback from the microphone.

Can people hear? Well, there's no PA system.

(Brief discussion.)

We'll have a short executive system, which will be recorded.

I think the Subcommittee will remember that we met last month for the primary objective of reviewing Mr. Denton's recommendations to proceed with construction of B&W for which permit construction permits had been issued.

And several pertinen t topics were discussed, including a brief review the draft of NUREG-0667, the members had only just received and had not had time to read.

The purpose of today's meeting is to complete the review of NUREG-0667, including Chapter 7, which we still have not received. So we have have it, I think it's on the table, isn't it? -- which we had not received as of five minutes ago?

NUREG-0667 is scheduled for review by the full

Committee on Friday, and the Commission will, of course, will be advised by the usual members that tie topic was included in the ACRS May agenda.

But inasmuch as ACRS has already supported Mr. Denton's recommendation to proceed with construction of B&W reac ors, I don't see any need for an ACRS letter addressed specifically to NUREG-0667. If contrary opinions, we would like to hear that now before we go into the regular session.

And do the Committee members, Subcommittee members, have any comments or, or any feeling, let's say, on whether the Committee needs to write a letter.

We'll hear probably from the Staff.

(Pause.)

We'll go right into the agenda then, which -- mislaid, slightly.

(Pause.)

Mr. Tedesco, I think, is first on our --

(Pause.)

MR. TEDESCO: Mr. Etherington, we're prepared to start a background of where we are. Subsequent to our meetings last, of earlier this month, where we met with this Subcommittee, as well as the full Committee, we have since that time met with the owners, on April 23d.

At that meeting we had an opportunity to hear each owner with comments and thoughts that they had about recommendations of -- in our report.

We have revised the report in certain areas, as an editorial type of change. We have made no substantive changes in S-22 recommendations. So they still are pretty much as they appear in the draft report.

CHAIRMAN ETHERINGTON: Now, these 22 recommendations, they're kind of scattered through the report on the --

MR. TEDESCO: They are, but section 2 is a place where they're all, they're all kind of together.

CHAIRMAN ETHERINGTON: Yes.

MR. TEDESCO: Section 7, which we indicated to you was being prepared by the probabilistic analysis staff, had been completed. It has been provided to you this morning. And we are prepared -- a briefing on the substanc, of the section, and at some of the bases of how we arrive at certain of the conclusions that were drawn from this.

Now, in section 7, that will complete the overall report; and we are now -- completion of the report as early as the latter part of this week. We want to issue the NUREG.

Now, the convention that we are going to recommend, that any implementation of the recommendation be included into a -- class. Subsequent to that, the decision was made that NRR wanted to -- that action plan, on the basis that if it would represent our response to the Presidential and the report, it would not be left as an open document.

That it would represent a closed-out action, and that our report now -- even though it contains a lot of related recommendations -- we will provide a separate implementation or supposition.

It will be phased, but not necessarily a part of it.

Now, what we'd like to address before we start our other aspects is to request that the Committee does -- expressing their comments on NUREG-0667. Mr. Denton was sure that he will adopt the Committee's comments in response to that report. So we would encourage you -- as an expression of your thoughts on that -- that I understand the program is supposed to appear this coming Friday afternoon with the full Committee. At that time -- where we are, where the report is, to help you in any way we can, so that we can have --

CHAIRMAN ETHERINGTON: And if you do want a letter, but you're not insistent, is that what you're saying?

MR. TEDESCO: I think Mr. Denton would be very pleased to have the report.

CHAIRMAN ETHERINGTON: Okay.

MR. TEDESCO: And I would encourage you --

(Pause.)

DR. ZUDANS: I had a copy of the agenda. And we did have a brief statement to make; it wasn't going to last more than 5 or 10 minutes. Effectively, I've finished right now. And we were going to have our people talk about section 7; and they won't be here until 2 o'clock.

MR. TEDESCO: That's all I had to say, unless you have questions.

DR. ZUDANS: Remember at our last meeting we had -- Ron asked about range bank indicators. Are you going to do something about that?

MR. TEDESCO: We -- I was talking about it among the task force. And we didn't come up with a -- why we should differ from the first high-level priority -- certainly recognized the degree of its importance, but we felt that there were other -- that would provide backup information for that. And it was not necessarily, in our opinion, be required to be that recurrent step.

DR. ZUDANS: I wonder whether it did. If there is at times, contain primary coolant. Maybe if you want to account for primary coolant in -- some indications wouldn't be bad.

MR. TEDESCO: I think I mentioned before that we have 18 related or safety-range types, as indication of the discharge lines of all of the valves of your relief valve and discharge to the --

DR. ZUDANS: That I understand.

MR. TEDESCO: Yes. So we would have an indication of whether or not the valve was discharging; we would not know how much.

DR. ZUDANS: That's, that's the whole issue I am raising: how much? is the question. And how much might be or might not be important to know what's going on, is not important.

MR. TEDESCO: We're looking for something that would give a, the operator some very quick reliable information. It's not necessarily meant that he perform a complete analysis with it. But it'll give him a very quick assessment of the status.

And that has been our guideline in making our recommendation.

And I have -- we have people who have too few in E4 -- and other people have said, "Well, gee, I don't know how -- I wouldn't know how to handle it."

So we're dealing with a rather subjective type of thing.

DR. ZUDANS: Well, as long as it's not completely for --

CHAIRMAN ETHERINGTON: The French -- and the architect engineer supply --

So they vary it from time to time in capacity?

(Pause.)

DR. ZUDANS: Well, after reading the report, if you could comment a little bit on this proposed sensitivity study to evaluate the once-through steam generator in the electrical system.

What did you intend to recommend? The report is not very explicit to that.

MR. TEDESCO: Well, we've, we've done that perfectly. It did not want to be prescriptive to the extent that we are telling the licensees what they --

Their plant, they're more familiar with the design aspect and the operation -- and we wanted them to look at things like the change in the power level, change in the water level, the importance of super heat, change in the secondary size of atmospheric valve setting -- that type of approach.

Another example of -- we hope that there would be others.

DR. ZUDANS: Some other locations -- and I think these two things are tied together -- you say that it would be desirable to achieve certain states without operator's interference -- and it would be desirable to use the excursions of parameters in a specific --

All of this is so obviously coupled to what the system can do. If you did the sensitivity study, you may be able to find out. That's if the study might concluded it's enough, to reach, you know, shutdown stage without large excursions. Or you can't reach them without human intervention; that might change --

So the priority really would be to find out what the system can do without human interaction, because as I read the report there is no single record that would show how a plant would react if people would not interfere. Interaction.

MR. TEDESCO: Well, and some of the actions that the operator is told to do now -- that's kind of a routine instruction that he follows in that to make sure he can maintain the level in the pressurizer, but --

DR. ZUDANS: He would use the excursion, but this is your requirement or your recommendation mean that this should be achievable without starting the second pump? without doing this --

MR. TEDESCO: Yes, but -- you have to do it --

DR. ZUDANS: So I would say for one, I would be extremely interested to see the analysis result that shows what can or what cannot be done, because it's certainly a matter which should -- I'm pretty sure that B&W must already have such analysis. They could not design a reactor without having it.

MR. TEDESCO: They might be --

DR. ZUDANS: That's right. That's the way I read the question.

CHAIRMAN ETHERINGTON: Don't like to sit and waste 45 minutes waiting for Mr. --

Is there anything --

May we have comments by the industry, B&W, Toledo Edison, and the Owners Group?

The gentlemen involved are available. And would it be a hardship to make your presentation now?

MR. TAYLOR: Mr. Etherington, B&W doesn't have any final comments at this meeting.

CHAIRMAN ETHERINGTON: I see.

MR. TAYLOR: We don't have anything different to say at this time.

MR. RAY: Harold, could I ask Mr. Tedesco a question?

CHAIRMAN ETHERINGTON: Yes, please do.

MR. RAY: It would seem to me that the capacity of the quench tank would be an important element in the design of a plant.

Do you have any idea how widely this varies between specific plants?

MR. CAPRA: I do not know.

MR. RAY: Do you have any fe l from a Staff view-point as to why it varies?

MR. TAYLOR: In -- criteria.

CHAIRMAN ETHERINGTON: I suspect there isn't any. In fact, some have just a quench tank; and some have a quench tank and a, another tank, some tank, from the quench tank.

MR. RAY: Should there be criteria as to what it should be?

MR. TEDESCO: Well, I, I guess, you know, there must be some -- there must be some event that you're designing for.

MR. RAY: If there is one, it's --

MR. TEDESCO: You usually have a --

CHAIRMAN ETHERINGTON: The experience is in the incidents we've had is that when they overflow they overflow real good, and it doesn't matter how big they are.

MR. TEDESCO: One of the recommendations that we shou.. develop a set of criteria for the transient situation. And when B&W responded to that, they mentioned that an example of a criteria might be that the reactor fluid just contained within the system and the quench tank -- I think that was one of them that you mentioned --

DR. ZUDANS: That's why, that's why you would have to know what is in a quench tank.

MR. TEDESCO: Yes. Yes.

DR. ZUDANS: That's why I raised the point.

MR. CAPRA: My name is Bob Capra. I'm a member of this task force, also.

I don't have the information with me now, but I know that the size of the quench tank is an item that we looked into specifically related to the Rancho Seco hearing. That was one of the contentions that the quench was possibly designed too small.

And a member of the Staff in the Auxiliary Systems Branch, Earl Matthews, was the individual that, that researched that. And to the best of my recollection, the size of the quench tank is based on a continuous rod withdrawal accident. And the expected release from that.

And the capacities or any more details than that I don't know. But I know the Staff has looked into it. I remember talking with Bill Matthews about it. And the quench tanks do vary from size to, from plant to plant. Why? -- if they're all based on the, the same design basis or the same accident. I'm not really sure, except I do believe, as Mr. Tedesco said, that that is within the scope of the architect engineer.

MR. TAYLOR: Yes, it would be -- that's correct. The thing that I'm not sure about the accident design basis for the quench tanks, and I think there's information in the FSARs about those. But in the early days the quench tanks were designed -- and I think it's fairly -- although we didn't supply -- that the quench tank was designed, as I recall, to take two back-to-back transients involving the lifting of the PORV without overpressurizing -- we still stayed comfortably below the safety valve set point on the quench tank and the --

Then to contain within the quench tank, there's a cooling core which is designed to bring the contents of the tank back down to atmospheric within something like an hour or an hour and a half.

So there were specific criteria. And I believe in our case the, the criteria that were passed on to the architect engineers were based on two consecutive back-to-back transients which -- the way the PORV was set before, it would have accommodated that transient, plus -- without overpressurizing the pressure tank.

MR. RAY: Well, if that were the controlling element in the design, Mr. Taylor, would they not come up with consistent sizes?

MR. TAYLOR: Yes, and I, I believe the, there is some difference in the power level force on the B&W plants, with -- I guess, within 10 percent of the percent. But they should be the same size, but I suspect that the architect engineers have added margin in some cases, which is different than others. But --

MR. RAY: Well, this would be in the direction of conservatism --

MR. TAYLOR: Yes.

MR. RAY: -- rather than short --

MR. TAYLOR: Now, I, I think you'll find that they're not all that different, really, on the B&W plants.

Now of course, it would be different on the Westinghouse plants than it is on ours.

MR. RAY: Is the Staff satisfied that there is an adequate interface between the architect engineer who is responsible for this design, evidently, and the B&W people?

MR. TAYLOR: Well, I think that has been an outstanding question right from the start. I think we, we made reference to th... in our report in a very general way. I think we have to improve the interface --

MR. RAY: You don't think it is adequate?

MR. TEDESCO: Well, based on the review we have done on the operator reactors, I think that has been a concern that we expressed.

MR. RAY: On the receiving end of it, Mr. Taylor, do you think that in general you do have an adequate, B&W has an adequate opportunity to comment on the design?

MR. TAYLOR: Well, I think the opportunity is quite adequate, although, you know, it has changed over the years. Now, these, these quench tanks were designed 12 or 14 years ago. And I think it also must be recognized that at the time these plants that we're talking about were designed, there was, as I recall, a pretty clear demarcation between two separate systems. The reactor coolant system ended at the discharge nozzle of the primary relief valves. And the, in some cases, there was a separate system called the primary relief system, of which the quench tank and the tail piping was a part. And in other plants that was a part of the waste disposal system.

But in any event, it was not a part of a safety-related type system. It was designed to ASME class 3C. But the whole, I think the answer is, there's never been any hesitation, as I know, on the part of the architect engineers or the customers to, to accept comments on these things.

But it was a, there was a separation between what you'd normally consider the reactor coolant system and the relief system or, or waste disposal system.

MR. RAY: That isn't prevalent today.

MR. EBERSOLE: Oh, yes, it is.

MR. RAY: I'm saying, I'm saying it with my tongue in my check.

MR. EBERSOLE: That's the old parochial or channel -- function of the design division. And, and it's a deadly system that has to be replaced.

DR. ZUDANS: Mr. Chairman, I'd like to ask questions, if we have time.

(Pause.)

With respect to this recommendation where you say that NRR should develop a set of criteria that would kind of specify or describe what kind of excursions or should happen during transients, it occurs to me, just a thought: shouldn't it be better that such criteria -- knowledge of N-triple-S system be developed by --

And then reviewed by you and, you know, then you could identify what you consider inadequate, rather than telling them what they should do, asking them what it can do; and then make a judgment.

MR. TEDESCO: I really think that's what our, what our recommendation is.

DR. ZUDANS: Well, you said that --

Let me see: in one place I think you said that NRR should develop it.

(Pause.)

SPEAKER: That was 79P, wasn't it?

DR. ZUDANS: Yes, I don't know the numbers. I said page, page 5 dash 26.

SPEAKER: Yes, here's what we, what we put on the board here.

DR. ZUDANS: You say: "We recommend that a program be established within NRR to develop the successful criteria."

MR. TEDESCO: Look at 19 up here. These criteria that we're talking about should be developed with industry --

that's our real recommendation.

(Brief discussion.)

DR. ZUDANS: Well, I don't know. The, 526, five dash 26, which says: "We recommend that a program be established within NRR to develop these criteria."

But what you are showing me here, you are doing what, what I recommended.

(Pause.)

MR. RAY: Mr. Tesdesco --

(Brief discussion.)

-- are these performance criteria, criteria that the Staff alone used for evaluating designs? Or will it be the performance criteria against which the industry will design the plant? Will it be the latter?

MR. TEDESCO: Well, we'll agree that the position for basic design criteria for light-water reactors for anticipated transients --

MR. RAY: Yes. So they're not, they're not limited to design review criteria. That is, they're not criteria for just design review by the Staff.

MR. TEDESCO: They're design criteria.

MR. RAY: For the design and construction of a plant.

May I ask you something:

How about the criteria that exist today? Evolutionary history on that would be interesting. How do they --

MR. TEDESCO: Here's what happened:

We, we had criteria for anticipated transients that would say that, well, if you didn't reach a DFE of 1 or 1.3 correlation and your reactor pressure didn't go above 10 percent of --

That would be acceptable. And as far as safety goes, that would probably be adequate criteria.

But if you look at the embarassment, the sensitivity things that are happening, of blowing the quench tank of less than 40,000 gallons of water in the containment, they may not be matters or issues that endanger health and safety, but they're not results of, of transients that you like to see occur on the frequency that we're having them.

So therefore, when you say, "Our criteria -- as far as the public goes, as far as really stabilizing the behavior, I think we'd do a lot better."

MR. RAY: Practical point.

MR. TEDESCO: Now as far as B&W --

Do you have them?

MR. TAYLOR: No, I don't have them with me. I can call them out.

They were the pressurizer level remaining on scale, LHPI actuation, no safety valve, no safety valve actuation, reactor coolant system, steam generator level remaining on scale, and temperature decrease remaining within the tech spec cool-down limits or tech spec change limits.

CHAIRMAN ETHERINGTON: Would you, would you repeat those again? I've only got four of them down.

MR. TAYLOR: Pressurizer level remaining on scale.

CHAIRMAN ETHERINGTON: Yes.

MR. TAYLOR: AHPI actuation.

CHAIRMAN ETHERINGTON: Right.

MR. TAYLOR: Code safety valve actuation.

CHAIRMAN ETHERINGTON: I've got -- one.

MR. TAYLOR: Steam generator level remaining on scale.

CHAIRMAN ETHERINGTON: I've got that one, too.

MR. TAYLOR: Reactor coolant system temperature change rate with the tech spec limits. That's a hundred degrees per hour.

(Pause.)

CHAIRMAN ETHERINGTON: Yes.

MR. TAYLOR: Let, I'll think about the last one. I don't have it right on the tip of my tongue. But those were what, when those criteria were satisfied, we would consider the transient behavior effort anticipated to be within normal bounds.

(Brief discussion.)

MR. TAYLOR: Reactor coolant system within the boundaries of the, of the reactor coolant system and the quench --

CHAIRMAN ETHERINGTON: Mr. Tam says we have these in the minutes of the last meeting, and we will --

MR. TAYLOR: It would be in the slide I had at the last meeting.

CHAIRMAN ETHERINGTON: Yes. Right. We'll get them.

MR. EBERSOLE: But this would not be included in what, all the things you would call transients, only a certain fraction of those.

MR. TAYLOR: That's right.

MR. EBERSOLE: Which you would identify on some sort of a probabilistic base.

MR. TAYLOR: Yes.

MR. TEDESCO: And then you have to talk about how, a no-failure case, do I talk about the single failures? the double failures? --

MR. EBERSOLE: You, you march out so far and then draw a line.

MR. TEDESCO: Yes.

MR. EBERSOLE: And that line has not yet been drawn.

DR. ZUDANS: I have the last question, if I may: When, when a discussion of reactor coolant pump release time is given in this report, one of the main reasons why one wants to restart the coolants on this to get pressurizers straight, it occurred to me -- and that's not a criticism or anything else -- couldn't, couldn't the industry provide a pressurizer spray with a separate pump, a cam-modeled pump, that would sit in the system without --

MR. TEDESCO: I guess you could, but the present plants don't accommodate that.

They are relying upon the main coolant pump pressurizing.

DR. ZUDANS: Yes, I know that. So I'm just saying that this is a future problem.

(Pause.)

MR. EBERSOLE: May I make a few comments?

Mr. Tedesco, I was somewhat surprised at the mild way in which you handled the matter on page 2.4, paragraph 2.2, in your discussion of, about the characteristics of the aux feedwater system in respect to whether it be safety-graded and, in particular, to whether it be seismically qualified.

I can contrast this with the recent hullabaloo we had about finding certain pipes qualified to withstand seismic stresses. And I think we have to realize that the aux feedwater pump in a seismic incident is probably going to be well among the very root few systems that have to work.

And without it, you don't stand a ghost of a chance of surviving a seismic incident, which would seem to me to make it absolutely mandatory to make it fully competent in all aspects to seismic events.

MR. TEDESCO: We will talk about that in section 7. But for the time being, let me just share that, the scenario or in terms you express what we have in our bible and talk about it. But we also recognize the uniqueness of B&W plants with their high-pressure ejection system and its capability to feed and bleed that would not require in the aux feedwater system --

MR. EBERSOLE: I admit we --

MR. TEDESCO: -- would have, would have a little more capability in these plants to deal with that situation.

The exception was Davis-Besse, which doesn't have a seismic --

So these are all aspects upon, for balancing --

MR. EBERSOLE: You mean you invoked feed and bleed as a seismic cooling method, after --

MR. TEDESCO: Taking that capability --

MR. EBERSOLE: Ah, but everybody, I think is currently agreed, no one's going to really test any of the plants in a realistic way, except at Idaho.

MR. TEDESCO: Well, we've already done some tests --

(Laughter.)

MR. EBERSOLE: Yes, we have some.

DR. ZUDANS: The only question is that you couldn't do that with current -- that you have.

MR. EBERSOLE: That, that assumes, by the way, the existence of certain things that you don't now have. That was not discharginining through the safeties, I don't think; it was through the PORVs, and it included the full discharge rate of what both, both primary, for the all, high-pressure injection systems.

So it was a pretty tenuous set of escape.

If you intend to sweeten it up, it might be better. But that involves looking at the PORV designs.

SPEAKER: And we are doing that.

MR. EBERSOLE: And I think maybe in that connection PORVs are maybe misplaced, if we're going to look at them in the context of providing feed and bleed. They are, after all, classical valves that are designed to upset and unseat, to go through some performance maneuvers which give them a blowdown of so many PSI. They're not particularly well designed to handle two-phase flow, if at all; and I think it would be well worthwhile to take a hard look at some of the PORV intrinsic design, is suitable for this kind of use.

MR. TEDESCO: Well, one, one of the things that developed out of Lessons Learned was that these valves be tested for single-phase or two-phase --

MR. EBERSOLE: Yes.

MR. TEDESCO: -- effluent, including solid water.. There's a test program going on now --

MR. EBERSOLE: Let me suggest that that's like testing a vehicle that you know is not likely to pass the test. And it would be better to test the valve that you knew would pass the test.

And I refer to a kind of a valve which I'll call a ported plug valve, which would pass the test, we know now.

But I have strong doubts that the PORV in their present design configuration will ever pass that test.

MR. TEDESCO: Well, I guess we have to rely on the criteria, the testing criteria. In other words, if we're going to go through a test -- or solid water, and then we say, "Well, a valve should restore itself to the condition it was bef[illegible] the test," that means --

[illegible] EBERSOLE: We are following -- I'm, I'm saying you are ask ng the machine to do an off design performance, and it's m[illegible]ch better to do, have the mac[illegible]ine do an on-design performance, which we could have, which we've got now.

MR. RAY: Well, Jessie, what do we need, a proper valve?

MR. EBERSOLE: Yes, proper valves -- to do this thing. I can suggest a design which I have a great deal of faith in, which is a rotary perfect valve, which hardly seems what it's doing. It's so insensitive to the modes of flow it can be throttled; it's very reliable.

Tapered to it.

Well, sure. But that valve is designed for that purpose.

DR. ZUDANS: And actually, you provide a valve like that with, say, a capability to discharge amount needed for the created moon.

So you can forget about PORVs and --

MR. EBERSOLE: All right. It is, it is a function for that purpose.

MR. TEDESCO: Remember last month I mentioned that people from Consumer Power Company -- an alternate proposal -- and they would demonstrate that capability -- I'm not sure, they may be consuming that nuclear valve design, I, I don't yet.

MR. EBERSOLE: Their proposal tended to defeat feed and bleed.

DR. ZUDANS: The only only one to change --

MR. EBERSOLE: Yes. All they were doing was backing up closure.

So they, they were in direct contradiction to being able to feed and bleed.

MR. CAPRA: Yes. Well, I think, I think their proposal also included the override capability --

MR. EBERSOLE: Well, they had 3,000 series in their design: a PORV and two block valves.

MR. CAPRA: But you have to design those with an override capability to open the block valve.

MR. EBERSOLE: I'm saying it's a string of three valves; you've got to open all three of them, for feed and bleed. That's hardly a reliable system for feed and bleed.

MR. TEDESCO: They may have a seismic --

MR. EBERSOLE: But you couldn't, you couldn't, you couldn't claim bleed and feed on a three valve in series rig.

MR. TEDESCO: Well, you won't take a single failure.

MR. EBERSOLE: Well, no. You won't even take a double failure.

(Pause.)

Another comment on feedwater:

I happened to go through TMI-1 looking at the DC power problem. And I found a curious opportunity for improvement which I certainly suggest we look at, regarding aux feed in addition to the other improvements. It appeared there that they weren't quite sure, but in any case it would only take modest modifications to make the aux feedwater control system respond to appropriate level controlling without any DC power -- in short, to fully mechanize it, using pneumatic or hydraulic controls, and make it self-contained, an aspect to holding an appropriate rate of feedwater flow, without any electrical functions at all, which was in my view a substantial improvement, considering they only have two batteries at those plants.

Matter of fact, the engineer there said he wasn't quite sure but he thought that would not be extreme modification and make it fully mechanical.

MR. THATCHER: Did you discuss whether they were going to --

MR. EBERSOLE: No. I didn't. It could be stored for a while, and then made up by an engine. It's just getting word from the susceptibility to -- a DC power failure, which was the issue at hand then.

MR. RAY: And what was the source of this suggestion?

(Pause.)

MR. EBERSOLE: I asked for it.

MR. RAY: Oh, this was your suggestion.

MR. EBERSOLE: Yes, to get it off DC, since DC was the problem.

(Pause.)

What else?

(Pause.)

Oh, on the matter of the fast cool-down transients, which are unique to B&W and are related to aux feedwater, is there any advantage in using pump trip to inhibit those things?

I have some horror of a B&W plant suffering a failure and surviving a run-on of the lengthy drop from the standpoint of containment pressure, which it would not do if it were into the containment.

But anyway, what results is, you have a substantial depresssurization of the primary coolant, and we look at that in the LOCA event, interestingly enough, the cool-down of the main vessel, but we don't look at it in this instance, where it is fully repressurized to the safety set front by the high-pressure injection.

And therefore, it is really challenged again to high pressure at the point after chilling.

Do you follow me?

This is an old issue: whether the main steam depressurization with the compounded effects of main feedwater/aux feedwater run-on, which produced the worst chilling effect and then, compounded by the follow-on automatic response of the system to high-pressure eject with cold water, clear on up to the safety-valve set point -- whether that imposes a primary vessel stress level.

MR. TEDESCO: Dr. Weinberg asked that very question on his --

MR. EBERSOLE: I'm not surprised.

MR. TEDESCO: But I think that -- I asked the cause.

And they investigated the overcoolant effect, overcooling transients which I --

They knew, was that they would not -- their early operating cycle, they would not go down below and continue --

I don't know how much analysis --

MR. EBERSOLE: Well, there's a gradient in the vessel.

And the question was asked at Pebble Springs, but it was given the same quality answer that the other questions were given, which was not very high.

DR. ZUDANS: Well, that means you would have to have undercooled state; and then you would start.

MR. EBERSOLE: Then repressurizing with cold water.

DR. ZUDANS: With HPI.

And what does it mean in terms of reactor undercooled? By how many degrees?

MR. EBERSOLE: There's a, that's a pressure gradient; and at one time the, the --

DR. ZUDANS: But how much is the temperature?

MR. EBERSOLE: Oh, quite a -- well, it, it is a gradient.

The interface of the vessel is chilled. And I was told one time -- in a very casual way, by the way -- that the conductivity rate, or the conductivity characteristics of the pressure vessel steel were limiting, such that an insufficient mass of metal was chilled to --

DR. ZUDANS: Well, it's a scheme effect --

MR. EBERSOLE: Yes, it's a scheme -- and whether that is a crack propagator or not, I never knew.

DR. ZUDANS: But if it's only a few degrees, what would be the cold --

MR. EBERSOLE: It's more than a few degrees.

DR. ZUDANS: It's more than?

MR. EBERSOLE: Yes.

You'd get cold water.

DR. ZUDANS: Really cold?

MR. EBERSOLE: HIPSI (phonetic spelling) is cold.

CHAIRMAN ETHERINGTON: Locally, of course.

MR. EBERSOLE: Yes. Well, locally; true.

I think it bears some review.

CHAIRMAN ETHERINGTON: But doesn't the design of the plant call for a number of HIPSI injections?

MR. EBERSOLE: Not under this condition.

This is --

(Brief discussion.)

MR. EBERSOLE: Not in the degree to which --

CHAIRMAN ETHERINGTON: What is the difference in the condition, then?

Tape 2:

MR. EBERSOLE: This is a secondary side --

But it implies a prodigious shrinkage in the primary cooling system. And a cold water coming in from the HIPSI pumps to replace the shrinkage and fully pressurize it to safety valve set pressures -- afterward.

CHAIRMAN ETHERINGTON: But still it's only a local cool-down, isn't it?

The large amount of --

MR. EBERSOLE: No. No, it's a general cool-down.

CHAIRMAN ETHERINGTON: Well, but there's not very much cool-down in something, a thousand gallons a minute into the system when it mixes the --

DR. ZUDANS: Yes, but the unfortunate thing is that -- from stress -- and it could crack.

MR. EBERSOLE: It's local. It's local to where the incoming cold water is.

CHAIRMAN ETHERINGTON: Yes, that's what I say: it is local.

MR. EBERSOLE: Yes, but -- true. It's system nozzle, really.

DR. ZUDANS: It's where your nozzles crack.

(Pause.)

It's a good question.

(Pause.)

MR. EBERSOLE: Oh, in, in your instrumentation improvements you made no mention of primary coolant level indication, or of avoid meter or any other inventory --

MR. TEDESCO: That's being worked on. That's not --

MR. EBERSOLE: Well, let's see: I guess the report didn't indicate any additional instrumentation in the quench tank.

In the electrical world, Bob, you use differential C2 measurements to figure out where the inventory is --

Can't you do this with the liquid measurements?

I've got a water input, and I've got a water loss. And I do some rapid computing, and I say: "Well, I know where it's all coming in; and I know where it's all going out. And the difference is where I don't know where it's going," which is a break

Isn't that sort of monitoring appropriate to a system like this?

Using the electrical analogy.

MR. THATCHER: Yes, I know. If you thought about the level in the vessel --

MR. EBERSOLE: I'm trying to track inventory.

Yes, I'm talking about vessel invetory.

I'm saying, "I know what water input is, and I know what water output is, through defined paths; and any difference is through undefined paths."

DR. ZUDANS: They can't measure flow through --

MR. EBERSOLE: What's that?

DR. ZUDANS: They cannot measure the flow rate.

MR. EBERSOLE: Can't measure flows?

MR. THATCHER: Can't --

MR. EBERSOLE: Is it the flashing problem?

(Pause.)

I'm talking about during accident condition -- well, these, these types of mild things like we had at Crystal River, which appeared to be monitored by inventory -- or monitorable by inventory -- flows, could have been. We would have known that 45 gallons were going out without measuring it on the floor level --

MR. TEDESCO: No, we made some calculations, based on containment pressure. Based on the estimate that we made on the partial pressure of air and the partial pressure of water, and then causing a feed -- you have to, how much would flashing water and flashing it in the --

We made a rough estimate of it, and it didn't turn out too bad. But that, I think that was very fortuitous.

MR. EBERSOLE: Well, I, I -- it's just an idea that assumes that you could measure input and outgo.

MR. TEDESCO: You have a mass inventory --

MR. EBERSOLE: Yes, right.

DR. ZUDANS: Flow metering and what? Do you have any place in the --

MR. THATCHER: In the high pressure injection there's

DR. ZUDANS: No, no. But in the main pipes you don't have --

MR. THATCHER: Sure. Those are active --

DR. ZUDANS: Where is this --

(Pause.)

MR. TAYLOR: It's about two-thirds up the hot leg.

DR. ZUDANS: Two-thirds up the hot leg.

(Pause.)

MR. THATCHER: But -- no.

DR. ZUDANS: But it wouldn't measure a, a -- type of rate. That's the problem with natural circulation. You can't make it --

(Pause.)

MR. EBERSOLE: Again, one of the four topics instrumentation, will this gnawing problem of how you handle contradictions and so-called redundant systems, wherein you have bidirectional response to execute, I don't understand how you sort that out.

Maybe you could tell me.

(Pause.)

I have redundant instrumentation. One tank, one indicator says the tank is high; and the other says it's low. Or one says that the flow is high, and the other says it's low or normal. I don't know which one to believe. I don't know what to do.

MR. TEDESCO: Well, at Crystal River the operators ignored all of them.

MR. EBERSOLE: All of them? Maybe that's the solution. If you --

MR. TEDESCO: And that's why we just kept --

MR. THATCHER: Well, are you assuming that "redundant" means "two"?

MR. EBERSOLE: I mean "redundant" means "two." Well, that's what the general -- "redundant" in this business means the minimum, which means two.

MR. THATCHER: Well, I admit: if you put two in, you might have that problem.

MR. EBERSOLE: Yes. I take it "redundant" means two.

MR. THATCHER: Reactor protection systems typically have more than two, i.e. --

MR. EBERSOLE: They don't on the reactor trip; they have got two breakers.

MR. THATCHER: On the what?

MR. EBERSOLE: Main power circuit breakers to the magnets, they ultimately converge to circuit breakers on the magnet supplies -- that's all.

MR. THATCHER: Oh, I thought we were talking about --

MR. EBERSOLE: Well, we are. I swtiched to control.

But anyway, when you're in the indicating area, I'd think you have problems when you just define "redundant," because it'll be interpreted as two by, you know, instinct. And it will leave you, leave you hung.

Now, I think all the instrumentation here, by the way, was -- the connotation, it was all analogue instrumentation, the way you talked about. And I couldn't help but go through here and say, "Well, one way to get some confirmation by diverse techniques is to do some step flash measurements with ERDA detectors -- non-analogue.

And anyway, anyway, get away from the problem of pure two-train redundancy in indication -- or provide some of the answers to how you cope with conflicting displays.

MR. THATCHER: The recommendation was mostly in the problem you run into when you lose your normal restrictions.

MR. EBERSOLE: Yes.

MR. THATCHER: Now, if we're postulating an addition to losing that normal train, we're going to lose one of the redundant --

MR. EBERSOLE: No, that's not so.

MR. THATHCER: -- back-up indicators.

MR. EBERSOLE: No, I would not want to do that.

My, the implication I, I heard only two indicators in the first place, two total in all. That's all I had. And I think you'll find that's the case.

MR. THATCHER: Well --

MR. EBERSOLE: Mr. Taylor, is that right? Would you interpret "redundant" indicating on recording equipment as being two trained?

MR. TAYLOR: Normally, I would, yes.

MR. EBERSOLE: Okay.

I'm getting a little noise that maybe it's more than two.

MR. TAYLOR: Well, Mr. Thatcher is right in terms of the protection system, which may contest four channels, three or four channels.

MR. EBERSOLE: Yes.

MR. TAYLOR: Two out of three or three out of four.

But primarily, if you're talking about indication, I, I would think of this too.

MR. EBERSOLE: And how would you handle the interpretation of contradictory information?

MR. TAYLOR: With difficulty.

(Laughter.)

MR. EBERSOLE: Well, that's an honest answer. I would, too.

There may be some cases where you would have a clear course or some course of action you could take at some -- inconvenience or cost would be all right.

MR. TAYLOR: I think it's a question of whether -- or I, I would think it would be a question or not the difference is significant.

MR. EBERSOLE: Yes.

CHAIRMAN ETHERINGTON: You're, you're suggesting we should have three --

MR. EBERSOLE: Not, not, not entirely. Diversity would be all right, if you could --

CHAIRMAN ETHERINGTON: Well, but even diversity, you've got two different readings. Which one do you --

MR. EBERSOLE: Oh, you will have diversity. You have that means two sets; two sets of two different kinds will give you three at least. I mean, you could use flows for levels or temperature for level -- whatever.

MR. THATCHER: But of course, if those diverse parameters are only minimum redundant, i.e., two -- power supplies and you lose those, one of those power supplies, like I think Mr. Etherington --

MR. EBERSOLE: You could infer that --

MR. THATCHER: -- you could be in as bad a shape.

MR. EBERSOLE: True.

(Pause.)

Well, let's see: the consumers' power proposal is non-safe in the context of using feed-bleed. It's non-conservative because it tends to defeat feed-bleed.

In this connection, I surely would like to point out what I consider the decided advantage of the B&W bores. They do not face the problem that we're currently discussing in the feed-bleed and concurrent flow or -- what do we call it?

SPEAKER: The reflux.

MR. EBERSOLE: -- the reflux flow, that the steam generators do. They have a, an excellent system with venting to condense the boiling coolant off the core into the steam generator and get a driving head of water, to get normal unidirectional circulation. They don't have to have counter-current flow, which is a substantial safety advantage when you're really in trouble.

On the other hand, at the moment they have no means to go to low pressure with safety-grade equipment on both the primary and secondary. Therefore, they can't claim an easy way to get water on both sides.

If one were to go to an easy way to get water, which means low pressure on both primary and secondary, you could even go subatmospheric on the secondary side and bring the primary coolant temperature down to very -- well, to cold conditions, which of course is the natural state that TMI-2 fell into, because it couldn't go any other way.

And I say we should set the stage -- for doing what TMI had to do, but do it deliberately, not accidentally.

(Pause.)

Do you follow me?

MR. TEDESCO: No, not completely.

MR. EBERSOLE: Okay: I can, with a B&W boiler, because I don't have to worry with plugging the convection process, which is an extreme advantage in my view, because it has the capability to vent at the hairpin bend -- at the, rather, candy cane. And it is not faced with condensation in a rising set of tubes, which requires counterflow. It achieves its condensation in a forward direction of flow and provides a natural unidirectional flow back to the primary system out of the steam generators.

Then it has a unique advantage, bordering on being as good as a boiler, which it would be in this case, for cooling to very low pressures and temperatures.

MR. TEDESCO: It depends on what the isolation --

MR. EBERSOLE: Exactly.

MR. TEDESCO: -- and the --

MR. EBERSOLE: And you draw the secondary side down by the condenser at your leisure. And in the meantime you would survive at high pressure and temperature.

You would only have to provide a qualified means to get the low pressure to enhance the way of getting water into the primary and secondary and keep the cooling process going, I think anyway, to pretty much lay its emergency cooling problem at rest.

I would never say that for combustion in Westinghouse.

(Pause.)

That's what I want to get on the record.

MR. TEDESCO: Because of your steam generator.

MR. EBERSOLE: Yes. It's just -- well booked for that purpose.

(Pause.)

That's all I think I care to say here. Very sensitive aspects of this design in meeting transients of a milder kind suggests that ways of control that invokes ways and spargers and various other things, rather than inventory control, using high-pressure systems into the secondary system and also high-pressure spray pumps like you mentioned a while ago, rather than reactor coolant pump bleeds --

MR. TEDESCO: That would make sense to me.

MR. EBERSOLE: Yes, sure.

(Pause.)

Mr. Etherington, that's all I had -- in marginal notes, anyway.

CHAIRMAN ETHERINGTON: Are there any further comments on this --

(No response.)

You're expecting your people in about five minutes, are you?

MR. TEDESCO: Yes, sir.

(Brief discussion.)

The Chair was talking about the, the full Committee action? Or do you want to wait?

CHAIRMAN ETHERINGTON: Do you want to talk about it?

MR. TEDESCO: Well, we wondered how you felt. You, you indicated that there might be a question on your mind whether or not you would write a letter, but that there was a need for one.

CHAIRMAN ETHERINGTON: Well, the Committee will decide whether it wants to write a letter. But usually, if the Staff urges strongly that a letter be written, the letter is written. This is the usual --

MR. TEDESCO: Okay. If that what it takes, then we will.

CHAIRMAN ETHERINGTON: Yes.

Or if the Subcommittee recommends that the letter be written, that probably carries even more weight.

MR. TEDESCO: Yes.

CHAIRMAN ETHERINGTON: So you've got to persuade us.

(Pause.)

The Committee may, in fact, want to write a letter.

MR. TEDESCO: Okay.

(Pause.)

CHAIRMAN ETHERINGTON: Well, is -- we may as -- do you have --

DR. LAWROSKI: Well, I would like to ask why, in view of the fact that they haven't gone -- that resumes construction --

why the light is ever needed by the --

CHAIRMAN ETHERINGTON: Yes.

(Pause.)

MR. TEDESCO: Well, I guess none of us associated with that letter to mean, well, it was all right for construction, it's okay for operator reactor -- we can agree with that.

But more specifically, should what the Committee's view that the 22 recommendations that we have made -- this is what we're working for.

Whether or not the Committee has a view as to importance, as to improvement in implementation. I think as far as the continued operation, continued construction, we're all right.

CHAIRMAN ETHERINGTON: I see.

MR. TEDESCO: We appreciate that --

CHAIRMAN ETHERINGTON: Well, that's a good distinction.

DR. ZUDANS: Since we have time, I'd like to --

MR. TEDESCO: Well. Okay.

DR. ZUDANS: -- on your, when you discuss this steam generator, secondary, or the capacitor being smaller than B&W or any of the others, and they can dry out by full power in 27 to 30 seconds, while the others would last for 90 seconds, and that the set point would be reached in eight versus 20 seconds, what's the real significance in terms of operator interaction?

MR. TEDESCO: If we have to make a -- to restore feedwater, you would have a much stronger time on one than you have on the other.

DR. ZUDANS: Seventeen seconds?

(Brief discussion.)

MR. CAPRA: But that, that's full power that you're talking about. I mean you're going to get a reactor trip long before that. That doesn't mean you're actually -- that would be if you left the main steam valves wide open, which isn't realistically going to happen.

DR. ZUDANS: Good. That, that comparison would be what would be the case.

MR. TEDESCO: It's still a question of -- you're still talking about, about --

(Brief discussion.)

MR. CAPRA: No, but then, it would range anywhere from around three to four minutes -- B&W steam generator, compared to maybe 10 to 15 minutes to a Westinghouse steam generator -- or maybe even longer.

DR. ZUDANS: That could make a difference.

MR. CAPRA: It makes, it makes a difference if you set the criteria: no operator action within 10 minutes, you know, which is, is fairly standard.

MR. EBERSOLE: May I ask a question in this area?

Have we exhausted to the appropriate extent the process of what used to be called power setback or runback, fast runback, less than SCRAM, to x percent? -- in this reactor. It seems to need it worst than most.

MR. TEDESCO: That's all on the ITS.

MR. EBERSOLE: Yes, is it, is it driven as hard as it should be?

MR. TEDESCO: Too much.

(Laughter.)

MR. EBERSOLE: In other words, that's the way it is.

MR. TEDESCO: Trying to get away from that now.

MR. EBERSOLE: Yes. That is, the power runback. I'm talking about rod run-in.

MR. CAPRA: You mean of the actual speed of the rods?

MR. EBERSOLE: Yes -- well, or the number.

MR. THATCHER: It's pretty accurate --

CHAIRMAN ETHERINGTON: I wonder whether everybody would try to speak a louder, please.

MR. EBERSOLE: I mean, at least it used to be called a set-back -- in years gone by.

MR. THATCHER: They do run a certain amount of --

MR. EBERSOLE: Yes, I'm saying it is the degree of use of that.

Mr. Taylor, could you comment?

MR. TAYLOR: Well, I really don't have any numbers in my head; but the, that was of course the purpose of putting in the anticipatory trip.

MR. EBERSOLE: But that goes all the way.

MR. TAYLOR: That goes all the way.

MR. EBERSOLE: I'm saying what about an intermediate stage?

MR. TAYLOR: I have the impression that once you have flipflopped these set points between the PORV and the SCRAM set point, that just doesn't make any difference.

In, in the kind of thing you're talking about, it was possible on many occasions to keep the reactor tripping when the PORV would lift. But once those set points are reversed, rod run-back takes on a different, a different ballpark as far as capability for change in your system.

It was used, and that's the contributive to keeping the reactor on the line.

MR. EBERSOLE: I'm merely asking: has it been used to the most appropriate extent, fully?

MR. TAYLOR: I can only say: perhaps not. I, I just don't know.

MR. EBERSOLE: Yes. Okay.

DR. ZUDANS: But the approach by Consumers Power seems to be very good, at least in current, at least the current thinking, because it will reduce the SCRAMs, which seem to be receding already design life, on the basis of what you have this time -- and in general provide, maybe if you put the right vial in there, provided it didn't leak --

But I think that Jessie is right: you shouldn't even attempt to qualify those --

MR. EBERSOLE: That's like trying to qualify a concrete airplane.

(Laughter.)

I mean you don't start with a bad, with a bad sample.

SPEAKER: There's an awful lot of money being put out in --

MR. EBERSOLE: I know; and I think it's misplaced.

SPEAKER: What? On concrete airplanes?

MR. EBERSOLE: No, we're trying to make these things do things they were never intended to do.

DR. ZUDANS: Well, even a plywood airplane.

(Laughter.)

MR. EBERSOLE: We have a history of that, don't we?

Well, it got off the, got off the water.

(Pause.)

CHAIRMAN ETHERINGTON: Well, are we ready to continue?

Right. Section 7.

SPEAKER: Do you want to take a break first? Or do want to go right on?

CHAIRMAN ETHERINGTON: Pardon?

SPEAKER: Do you want to go right now?

CHAIRMAN ETHERINGTON: Yes, go right into it. Oh, do you want to take a break first?

SPEAKER: It's on the schedule.

CHAIRMAN ETHERINGTON: Well, it's on the schedule at what time?

SPEAKER: 2:00 o'clock.

CHAIRMAN ETHERINGTON: Okay, we take a break.

End Tape 2

(Brief recess.)

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CHAIRMAN ETHERINGTON: The meeting will reconvene.

MR. ROWSOME: The Probabilistic Analysis Staff was asked to prepare the seventh chapter, which was to be a review of the risk reduction potential or effectiveness of the recommendations prepared by the Task Force.

What I will give for you is a brief outline of what we did and how we got there, and what the findings were.

The people who participated were myself, Frank Rowsome, Matt Taylor and Mark Cunningham.

The technique was to fill in a number of tables using engineering judgment since we only had about two weeks in which to work. It was too -- far too short a period of time to do any actual probabilistic safety analysis, and we do not yet have the benefit of the risk assessment work on Crystal River at hand.

We do have the eventries for that work, but we don't havethe system reliability models that are qualification we trust.

So, that on such a short time scale all we had to bring to bear on the problem was the engineering judgment we have developed in the course of working on system reliability analyses and risk assessments over the years.

We decided to fill in several tables, one which tabulated the influence of B&W plant characteristics on the

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likelihood of severe accidents and incidents; tabulate effect of each recommendation on a number of distinct accident scenerios catalogued by the initiating event; and to do the same thing for the twenty-two recommendations tabulated according to the likelihood of the ..verity of the outcome.

The catalog of accidents into severe accident -- accidents and incidents is based upon a consistent finding from not only the reactor safety study but the other risk assessments and core melt consequence analyses we have done on plant -- TWR plants with dry containments. And that is accidents which correspond with the WASH-1400 release categories 1, 2, and 3 are the only accidents which can give lethal dosage, that is, prompt or acute fatality. And these are also the only accidents that are -- will contaminate significant -- or have any probability of contaminating significant amounts of land.

These stand out as being qualitatively much more severe in their consequences than accidents that belong in the WASH-1400 release categories 4 through 9.

So -- and in addition, not only there are -- are there neat distinctions in terms of the severity of the consequences when you draw the line between TWR-3 and TRW-4 in release categories, but there are convenient system distinctions here too. To get an accident that belongs in the severe accident category, that is, TRW release categories 1 through 3, you must not only melt the core but also cause prompt, early severe containment rupture--a big puff release--fairly early on in the accident.

If you merely have a leaky containment after a core melt, or the containment holds until base mat melt through or one of those outcomes, you do not get even at the ten to the minus nine level any acute fatalities or much ground contamination in our consequence analyses.

So, that the lesser core melts and the TMI-like scenarios belong in what we call here accidents characterized by potentially significant numbers of latent cancers and potentially troublesome groundwater contamination, but relatively little ground contamination through the atmospheric pathway and no acute fatalities.

The systems failures associated with this can include core melt or core damage but without prompt, early containment failure. They also might include LOCA with gross containment failure and TMI-like scenarios and the rest.

The incidents are the ones that have relatively small effectively negligible radiological consequences on site.

Now, our studies have indicated that the kinds of accident scenarios giving rise to these three classes of accidents belong in different populations. For example, let me see if I've got another slide here. I think I -- yeah. Severe accident scenarios. I have another slide. I think it might be out of order.

To get a release category one, two, or three incident, what I have called here the severe ones, the ones that can give lethal doses to people offsite, you must melt the core and breach containment early on in the accident, and you could do that by internal missles or external missles. You could do that through structural collapse of the containment. You could do that with a loss of coolant accident which flows down outside of containment. It bypasses containment and cannot be isolated. That's the event V of the reactor safety study, and it's a triple common mode failure because it constitutes in one such incident a breach of containment at LOCA, and it intrinsically fails ECCS because you cannot go into the recirculation mode to close the loop on recirculation.

You could get such an accident if you loss core cooling and containment sprays and fan coolers in which case the containment would burst on overpressure or possibly through a hydrogen burn early enough in the incident -- in the accident to give the severe consequences we are speaking of here.

A borderline case is the case in which you lose core cooling in the core melt. Containment sprays and

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fan coolers are running, but the containment vents are open and it fails to isolate containment.

In some scenarios the dose reduction factor obtained with the sprays and the coolers may be enough to get you out of the severe accident category; sometimes they will not. It's a -- that's a borderline case.

But of particular interest here is the fact that one, two, and three have really nothing to do with the design of the nuclear steam supply system per se, except insofar as perhaps there may be a propensity in the reactor for the vessel lid to blow up.

They're all balance of plant features which govern the susceptability to those accidents.

In four and five you are dealing with the failure of somewhere between eight and twelve front line trains of engineer safety features or systems if you do the counting. Two or three trains of ECCS, two or three trains of containment sprays and two or three trains of fan coolers have to fail to get you into a -- a -- the fourth-line failure, essentially all of the engineer safety features.

And the likelihood of all of those failures happening to random coincidental faults in those front line systems is clearly negligible involving so many failures.

Where they could be caused with non-trival

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probability is through common-cause mechanisms such as a fire, or a flood, or an earthquake, or possibly a failure of one of the support systems which underlies almost all of the active engineer safety features such as essential AC power, DC power, or in some plants essential service water systems, or things of that kind.

So, that what governs the susceptability of a plant to core tend to be the common mode failure mechanisms that -- that are shaped by the design systems in the auxiliary building--AC power, DC power, susceptability to earthquake--that sort of thing.

Again, not terribly sensitive to the design of the nuclear steam supply system, likewise for FROG.

That gave us a clue that perhaps in a severe accident scenario category B&W plants would not look any different from Westinghouse and CE Plants. But it appears to be balance of plant features that govern the susceptability to this kind of thing.

So, we look case by case through the list of characteristics -- unique characteristics of the B&W and Triple S to see if that tentative hypothesis would hold up. That they would in fact not look any different than the Westinghouse. And we concluded for the most part that that's true.

Here is a table of characteristics on the left and

3/ the three severity categories across the top. Severe accidents in the middle column, radiological accidents that are not severe; that is, no lethal doses in the middle right column, and the non-radiological incidents on the extreme right column.

Most of these plant characteristics surfaced in this context because they have been a nuissance, because they had caused incidents, because they had attracted attention in LERs or abnormal occurrence reports.

So, almost by definition they have a significant effect on the frequency of incidents in these plants. That's how they got to be on the table in the first place.

We went through them and plowed our way through each of these characteristics to see what effect it would have on these classes, and we came to the following conclusion that the fact that the steam generators dry out more promptly in a B&W plant than the CE and Westinghouse plant has very little bearing on the likelihood of severe accident and very little bearing on the likelihood of even modest reactor accident.

The time to dry out does indicate a time to a disruption in the normal heat dissipation path. But it is not a point of no return for core cooling. You can resuscitate steam generator cooling after dryout for a good while even after you lose the ability to cool the core through

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the steam generators you may still be able to save or prevent core damage by resuscitating high pressure safety injection particularly in the plant that have high head high pressure injection pumps which are capable of running pressures all the way up to the safety valve set point.

They can make up the deficit in primary coolant after a good deal of it has boiled away. So that the point of no return for restoring core cooling after an interruption in both feedwater and ECCS may be as late and perhaps later in B&W plants as it is in CE and Westinghouse plants.

CHAIRMAN ETHERINGTON: You mentioned CE and Westinghouse a couple of times. Are these supposed -- are these to be construed as relative to --

MR. ROWSOME: They are relative to -- to the picture that has emerged for Surry in WASH-1400.

CHAIRMAN ETHERINGTON: I see. So, they are relative to U-Tube events.

MR. ROWSOME: Yes. Yes.

MR. EBERSOLE: But you had to invoke feed/bleed to say that; didn't you?

MR. ROWSOME: I didn't have to in this column, but I did have to in this one, yes.

Now, the frequent undercooling transients associated with the prompt dryout, we think, again does not relate very well to the kind of common mode failures that are likely to give you not only core melt but containment systems failures as well. But they do relate to the kinds of scenarios that can get you into a core damage situation.

The undercooling -- what really distinguishes B&W plants here is not the outright failure of the emergency feedwater system which is equally serious in any PWR, but the delay -- delayed start of emergency feedwater.

And a CE or Westinghouse plant a few minutes delay in getting emergency feedwater start it will not ential a challange to the pressurizer valve, the PRV, or code safety. Whereas in the B&W plant it will.

So, the -- a penalty associated with the once-through steam generators that shows up on this line is the fact that delays in starting auxiliary feedwater emergency feedwater translate almost on a one-to-one fashion with challenges to the valves in the pressurizer. And since transient-induced LOCA we now believe to be one of the more statistically prominent routes to core damage in all PWR's, particularly so in B&W plants, we think that signicance may be large.

However, I should point out -- should reiterate that the difference has to do with start times of the order of one minute versus ten minutes. The B&W Plant put a premium on prompt auto start of the well time auto start of the emergency feedwater system.

The heightened trip frequency particularly since the TMI ratchets have gone in, there have been a higher frequency of scrams and nuissance trips in B&W plants. It's not so great a factor of two, but it is statistically significant.

The kinds of things that have been causing the enhanced trip frequency in B&W plants have been small routine upsets in feedwater flow or power or cooling mis-match sorts of things that wouldn't have been troublesome at the old set points, but have become troublesome since the lower adjust scram and anticipatory scram set points have been put in.

We think they have a fairly small corrolation with the kinds of scenarios that led to core damage and negligible corrolation with the kinds of scenarios that are likely to lead to severe accidents.

The text describes a couple of hypothetical exceptions to this. We examined the logic underlying that conclusion in the text. If you are interested I can go into it, but if we go into such details we'd be here for two days.

Perhaps I should just --

DR. ZUDANS: Just -- just one question.

MR. ROWSOME: All right, sir.

DR. ZUDANS: Why in -- from this type of analysis

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point of view you are quite right, but you are really exhausting the reactor vessel's life if you do that on the other components. You can see the design number of such trips then you aren't finished with the percentages.

MR. ROWSOME: Well, that's true. It could -- could mean problems with vessel qualification near end of life. It could entail earlier or maybe even qualitative difference between having to anneal a vessel and not. It could be hideously expensive to the utility.

I am giving the agency credit with tracking vessel life well enough that --

DR. ZUDANS: It would detect it if anything happens.

MR. ROWSOME: That it will be detected before the risk of vessel rupture becomes substanially -- a substanial contributor to the risk.

But the ecnomic penalty associated with rapidly running through the life expectancy of the reactor vessel and it's equipment of course is a real problem for the owners and is a real cause for alterness on the part of the regulators.

NNI/ICS faults -- they do have the common -- common-cause failure characteristics that they have historically blinded operators to what was going on in the plant when they took place, some of them. And some of them have led

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to schizophrenic behavior on the part of the integra:ed control systems.

I think that the potential for core damage associated with that is very, very much smaller now that we have had the educational experience of having several of these incidents and called attention to them and have the educational experience of TMI than it was at the time of the Rancho Seco transient, as you all know. I believe you had a copy of the memorandum I wrote on that subject saying that that was a serious safety flaw in the climate that prevailed before TMI.

I no longer think it is a large contributory, but it may remain a non-trival contributory to the danger of core damage. I believe it is a negligible contributor to severe accidents.

Frequent overcooling transients. We didn't see any reason to believe that drawing the bubble into a reactor coolant leg would pose a serious challenge to loss of off-core cooling for a critical period of time. We didn't think the nuissance to ECCS actuations were much of a problem except insofar as they affect operator behavior. To the extent that operators come to anticipate frequent nuissance ECCs actuations from nuissance scrams. They will be conditioned to try to trottle back or delay the auto start of auxiliary feedwater and will be conditioned to quickly, and perhaps cavalierly turn off or turn back ECCS in the belief that it is in fact a nuissance challange.

I think these -- that's not a significant contributor to the risk with the TMI learning experience fresh in people's minds. The likelihood that such errors even if made would be continued long enough to produce core damage, I think, is very small.

But nevertheless, the experience of having a high frequency of nuissance actuations cannot but diminish the seriousness with which operators take the real thing.

It could be countered either with training or with actually addressing the frequency of nuissance ECCS actuations. One could get rid of the safety implications of this scenario either through plant design or through operator training. And since both are being worked on at the moment, I think that's a --

DR. ZUDANS: I have a question -- you were not here when we discussed this -- I guess Jessee brought up the question that ECCS or HPI would still be -- in an undercooling case still might be acquissent all the way to pressure so -- for safety valves with cold water. And that would have structural implication of some sort of a lesson.

MR. ROWSOME: It would certainly use up a lot of the typical life. There's no question but what such --

DR. ZUDANS: Well, it might even exceed Appendix JD1. GE I mean.

MR. EBERSOLE: Let me set perspective on that. You know we've had post-LOCA examination of the cracked potential of the vessel even though it was unloaded after the LOCA. A much more serious state of affairs if you depressurize the secondard and then compound that by run-on of the main feedwater and carry out the most absolutely terrible quenching rate you can get and then compound it by fully pressurizing the primary loop with a high pressure core injection to the safety valve set point under the chilled condition.

That is the scenario that we are talking about which challenges of the integrity of the pressure vessel because of the thermal effects. I don't know how much life it would use up or --

DR. ZUDANS: Well it might just take one cycle if it --

MR. EBERSOLE: It may be one cycle and that's it. I don't know.

MR. ROWSOME: We didn't consider that scenario.

MR. LAWORSKI: Since this is a closed meeting, we will just ask for a yes or no. Did you consider the relative vulnerability to sabotage to these reactors?

MR. ROWSOME: We didn't I don't offhand see

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any reason to believe they are anymore susceptable, but I haven't really thought of that.

MR. LAWORSKI: May be worth looking into though.

MR. ROWSOME: We did look at the overfeed of the steam generators potentially leading to main steamline rupture. If that takes place in the containment it will look to the containment with the exception of slight differences in temperature and pressure relation rather like a LOCA except that the primary system would be intact. You still have all of your options except perhaps main feed and condensation in the condensor.

All the options for decay heat dissipation you would normal have including aux-feed. Even you chose to use the effected steam generator as a heat sink for the cool down you would be dumping steam into the containment atmosphere where it would be condensing on the fan coolers just as it would in a LOCA situation. You can cool down and then go on to the decay heat removal.

We don't see any reason why that would degrade the reliability of -- with which one could get to cold shutdown. In other words, it is not a scenario which puts core cooling at peril.

MR. EBERSOLE: May I question that in this respect. I understand B&W has -- is supposed to be an extremely reliable system to avoid runon of main feedwater. I take

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it that's what you mean by overfeed main steam?

MR. ROWSOME: Yeah. The hypothesis is that in ICS failure of some sort or some other failure --

MR. EBERSOLE: If that's coupled to a main steamline rupture it extracts both the primary -- the secondary system energy and the primary system energy and runs both of these inventories of energy into the containment beyond the capability of the containment to sustain the pressures and temperatures; is that not correct?

MR. ROWSOME: Well, the conservative design basis codes would probably predict higher than design temperatures or pressures. I would -- it would very much surprise me if one got -- if one exceeded the actual failure pressure. Even if you did and bursted, unless you cause structural collapse on the reactor coolant system you still have core cooling capability.

MR. EBERSOLE: Oh, no, you don't either because of the compounded effects of disabling the pipes and paraphernalia that provides the continuous flow of water to the primary system.

Mr. Taylor, am I off base here? I think you cannot stand a containment failure and by any stretch of the imagination hope to maintain a continued water flow to the primary -- to the core proper.

MR. ROWSOME: You don't know what's going to break.

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It might break in such a way that you cause the whole building to collapse, and you might blowout a --

MR. EBERSOLE: Well, it could empty, for instance, into the secondard environment that has the equipment which is supposd to provide this runon cooling.

MR. ROWSOME: Well, what I was going to say is the only scenarios we identified which we saw as posing a direct hazard to -- understand that this is a scenario in which we have not postulated a preory common mode failure of containment fan coolers or sprays, or that kind of thing.

We don't see that as statistically corrolated to the kind of failures that would leave main feedwater on full and run the steam generators water solid which was the scenario we had heard described to us both in the ACRS and by the Task Force as one of the concerns about this plant that it takes such a short time to run that thing water solid.

That if you were to get a failure mode with the plant trips and feedwater not tripped, that it would be a matter of one or two or three minutes before you would in fact run water solid. And that the lines were not qualified for it.

We assumed that using realistic assumptions and under the assumption that fan coolers and sprays would be operable that the containment would survive the challenge.

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Although it is not a design basis challenge, that's true. And there are ways in which the containment might fail, although I won't argue that it will fail in these ways, by which you could go on cooling afterwards.

We don't see a high probability -- a high transition rate given that you've gotten into the situation for going on to core damage or core melt.

If on the other hand the break is in the auxiliary building, then we see a hazard if and only if the consequences of that break get to the common support systems for the -- for core cooling, which would mean the feeding both auxiliary feedwater and high pressure safety injection. And to the extent that one can through a deterministic analysis which would depend upon the details of the balance of plant design postulate that such break would in fact fail all trains of high pressure safety injection and all trains of aux-feed. Then, maybe you've got something worth worrying about which is why we've got the question mark on negligible. Otherwise not.

We looked at the pros and cons of having high-head safety injection pumps capable of lifting the code safety, although as I understand it, it's not necessarily clear that they can carry away enough heat unless steam is coming out of the valves to keep up the decay heat. So, that if you were depending upon code safetys you might have to boil some of the primary coolant before you got to a situation in which you would keep up with decay heat.

MR. EBERSOLE: Mr. Rowsome, I understood that as -- is the way it would naturally occur anyway. You would have two-phase discharge with a net loss of inventory --

MR. ROWSOME: Right.

MR. EBERSOLE: -- until you reach some undefined state.

MR. ROWSOME: Right. Right.

MR. EBERSOLE: But, we -- before you came in we were talking about an immediate thing ahead of seven, which we've been calling reflux condensation, which I think it's fairly clear that the geometry in design of the B&W plant has a clear superiority over CE and Westinghouse --

MR. ROWSOME: Yes.

MR. EBERSOLE: -- to do this. Even to the point where one with negative pressure or vacuum on the secondary could run to quite low levels of temperature. Advantageous rather than a --

MR. ROWSOME: Yes, I think that's true.

MR. EBERSOLE: -- a disadvantageous aspect of this design.

MR. ROWSOME: That's true. What we said in the text, we didn't talk about that at any length. What we said in the text is that we believe that the plants, at least those with the high head safety injection functions, perhaps those without; really do have a genuine point of no return for core damage all the way up to core damage. That if you get back to capability of building a heat sink in the secondary side or get back to capability of making up on the primary side with an HPI pump, you don't have any artificial points of no return in this reactor design before the core damage actually commences.

It's not so clear that's the case with the -- some of the other designs.

MR. EBERSOLE: Yes, right.

MR. ROWSOME: You may hit a point of no return long before you've actually incurred core damage in which the installed equipment could no longer bring you back.

And we think it's worth a good deal of reducing the susceptability of the plant or enlarging the window to recover a firm scenario that might otherwise go to severe releases or to -- to an accident.

MR. EBERSOLE: In seven are you incorporating feed/bleed in the context of just primary feed/bleed or feed/bleed in both primary and secondary with evaporative cooling on the primary to the secondary?

Is that -- we've been identifying it in the feed/bleed and reflux condensations. It was two modes of cooling--one, there is no net discharge from the primary

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system.

MR. ROWSOME: Yes.

MR. EBERSOLE: But there's a condensation process of transport that --

MR. ROWSOME: Right.

MR. EBERSOLE: -- occurs to the secondary in which the case B&W plant looks good.

The other is just direct water transport and steam out of the PRV's or hopefully a better valve than PRV's.

MR. ROWSOME: Yes.

Now, I think they are in better shape in those respects. And perhaps your comment about the -- the reflux condensation of being another advantage that perhaps belongs down here along with seven is -- is a good point.

Now --

MR. EBERSOLE: If it's not reflux in this case, it doesn't have to be reflux. It's for --

MR. ROWSOME: Yes.

There is in the handout some of the footnotes associated with the --

DR. ZUDANS: Could you put it on 9? Put it on number 9 and see what the last -- I think I understand what you want to say but you didn't --

MR. EBERSOLE: I think you would like to put reflux --

DR. ZUDANS: As you say a point of no returns

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means you've reached the point where nothing else can be done. But what you mean is even beyond that point this feed/bleed would provid you with some capabilities.

MR. ROWSOME: There may be a window, I don't know, beyond which restoring a heat sink in the seconardy side would no longer save the core. And -- but where restoration of ECCS could.

MR. EBERSOLE: Well, along that line of progressive degradation, shouldn't 9 be for combustion and Westinghouse reflux condensation? From B&W it would be a boiling condensation in the primary loop and -- and a repritative cooling in the secondary?

MR. ROWSOME: Well, I was really thinking of the point of no return for melt rather than just --

DR. ZUDANS: What does that sentence really mean? Is it correct? Is it saying what you are telling?

MR. ROWSOME: What I mean to say here is the provision of high head safety injection pumps that are capable of reducing some net flow for any plausible pressure that you'd likely find in the primary coolant system during a core melt scenario can provide an option for arresting the core damage or preventing core damage in the event that the steam generators go dry and stay dry indefinitely.

DR. ZUDANS: Okay, that's --

MR. EBERSOLE: Would you discuss interposing an

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eight and a half in there which I will call qualified pressure reduction in the primary and secondary side to quite low pressures and subsequent boiling of the primary to the secondary and low pressure boiloff of the secondary. That's a modified B&W design which would permit low pressure evaporative cooling. Do you follow me?

It would require feedwater but not in the context it would be high pressure feedwater as you infer here. It could be any old water.

MR. ROWSOME: Yeah. Yeah.

MR. EBERSOLE: A kind of ECCS on the secondary.

MR. ROWSOME: Right.

MR. EBERSOLE: For the express purpose of getting out of this thing which is more likely than a LOCA.

MR. ROWSOME: We've given a little thought to whether you could take credit for let -- let's say the scenario might be station blackout, and an hour may go by and you get back offsite power, and you have a plant like Davis-Bessie which has turbine-driven auxiliary feedwater pumps, and you would have had dry steam generators in the last hour. The steam pressure might well have decayed away if you had not succeeded in starting the pumps right away You would not be able to restart auxiliary feedwater if you had lost it near the outset. But you do have a low head startup feedwater pump, motor driven, non-essential

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offsite power, could you use that to booster yourself back up?

MR. EBERSOLE: To pull the pressure back up.

MR. ROWSOME: And it would require the use of the steam dump valves which would have to be operable under the circumstances. But in fact that it may well be that that's --

MR. EBERSOLE: That's a window.

MR. ROWSOME: A window for saving the core in a plant like that.

Many plants -- well, Browns Ferry saved itself by using its condensate pump. B&W may be a little harder off in this regard because most of them have de-airators, which would have to be reflooded before you could use the full head of the condensate pump to translate that into head in the steam generator if you really wanted to use condensate pumps. But startup feedwater pump in such a design could -- could make a difference, yes.

And I think we're tracking in the same terms.

MR. EBERSOLE: Yes.

MR. ROWSOME: Now, the B&W concerns upon the Task Force and gave shape to many of it, but not all of its recommendations, are closely associated with the frequent incidents that have been recorded in the -- They're one removed from the scenarios that lead to core

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damage and is still one further removed from the scenarios that lead to major releases. So that one doesn't expect a one to one correspondence between Task Force recommendations and vulnerability to major releases.

Nevertheless, there are some of their recommendations that do -- that do make that contact.

The view graph that I have of these tables are quite illegible. But you have the tables in your handout so you will be able to -- to look at them.

I don't think I really want to talk about this table at all unless you all want to. I will go on to the one that relates to -- to signifigance of outcomes.

MR. LAWORSKI: Would you give us a summary of what that table says?

MR. ROWSOME: I don't think it can be summarized.

MR. LAWORSKI: The high points? There are no high points in it?

MR. ROWSOME: Well, let me go through the high points of this one and that may lead us back there if we get to ones we're interested in.

The recommendation to upgrade the emergency or auxiliary feedwater fluid system to safety grade, we got a high evaluation in two contexts--one was in the diversity of power supply and one was in other alternations we've suggested such as carrying the diveristy through to support systems, valve acuators and low boil cooling systems and the like extending the single failure criteria to mis-aligned manual valves and to the criterias from the normal test and routine to the pumps and the like.

And to the suggestion of a addon dedicated shutdown system such as the Ebersole current proposal for the dedicated safe shutdown system.

We think they can have high value in very high consequence accidents scenarios as well as the core damage accident scenario.

The only ones in which we saw significant competing risks that could be made worse by the imposition of this requirement is through the imposition of the steamline and feedwater line break criteria on Oconee which was not designed with that in mind. And we think designing -- forcing the auxiliary feedwater system in to a design that isolates the effected steam generator as the existing disigns do could provide more risk enhancement for scenarios like blackout and -- and just loss of main feedwater scenarios than it buys you in risk reduction on the accidents for which these features are provided.

So, that's -- infinitesimal, negligible -- you can't see it against the background.

MR. EBERSOLE: May I ask a question about G up there. The seismic and external event qualification where

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you have a low potential benefit. We have a tremendous investment to seismically qualify these plants. I think it's fair to say that we experience a moderate to severe seismic event. And one of the systems that must work because other heat rejection systems will probably fail is that system which provides offspeed water and discharges the steam to atmosphere whether you use electric power or you use steam turbine. If you don't have that system then all of the other civil and structural expendures that we have made are to no avail.

MR. ROWSOME: Well, if the core melt that are to no avail, we think feed and bleed is a viable --

MR. EBERSOLE: You are going to invoke feed and bleed and that's why that becomes low --

MR ROWSOME: That's right.

MR. EBERSOLE: -- came out earlier.

CHAIRMAN ETHERINGTON: -- in the context of proving that it works.

MR. ROW: Certainly NRR would not be about to embrace the viability of feed and bleed as the design basis way of coping with a safe shutdown earthquake without a lot more analysis and a lot more qualification testing than we've seen.

On the other hand, I think we know enough to at least have an engineering judgment that it would probably work most of the time.

CHAIRMAN ETHERINGTON: Remember, it will only work at the present time using all of the unqualified relief apparatus, this is pure RV's and block valves which are currently being proposed by some utilities to be put in series, so you have to open three things to get a flow.

And, it would have to be using all of the charging pumps. There's no redundancy in coping with feed and bleed with just a fraction of the mitigating systems.

I think that's correct. You have to use all the high pressure feed pumps, for feed and bleed.

MR. ROW: I don't know whether that's true or not. I would imagine that if one did a realistic analysis, one might get away with very limited or no core damage unless -- But one certainly couldn't get to appendix K with anything less and you may not even be able to get to appendix K with all three transient, all the valves operable.

CHAIRMAN ETHERINGTON: I was thinking of it in the context of no damage.

MR. ROW: Right.

CHAIRMAN ETHERINGTON: Thank you.

MR. ROW: What stands out in the second recommendation for qualifying the control systems, the most valuable improvement we see is providing an auto start for the auxilliary feed water system which is free of common mode failure susceptibility with the integrated control system and non-nuclear instrumentation.

The last thing you want in the plant is a set of failure modes that will not only cause loss of main feed water, but also defeat the starting of auxilliary feed water.

Under the competing risk, the only thing that rises to prominence is the potential that a feed water regulation system that throttles main -- throttles auxilliary feed water as well as main feed water, to avoid overfilling steam generators, of course, has a malfunction potential that could cause a loss of all feed water, and so you have to be very careful about the competing failure modes and such provision is designed in.

The provision of diverse auxilliary feed water pumps, very important to Davis Bessie, so long as they do not have high head injection capability and pretty important, even with it.

Modifications to the steam and feed water line break logic to avoid the adverse systems interactions, the potential to shut off auxilliary feed water, we think is moderately important.

The long list of suggestion recommendations to improve the integrated control system in the NNI -- We had a little difficult evaluating these because most of them are recommendations that are in the form of suggestions to go look in a particular context for ways to make things better.

They do not go so far as to say, scrap the integrated control system and start over. They say, go and see if you can make it better with little adjustments here and there, little minor alterations.

And, I don't know how much room for improvment there is in that system without in fact scrapping it into -- And, I'm not also really convinced that one really has to improve it.

One can design around it. One can design to live with it. Both are options.

The loads we assigned here reflected, I think, our pessimism that there was substantial room for improvement in the present design along the lines we have suggested, but that's just a matter of opinion, and we could well be -- well be wrong.

We like, in particular, the I&E bulletin that asks each licensee to examine the functional effects of bus outages for all their instrumentation buses, safety and nonsafety related, learn to identify the systems, learn to identify the causes, learn how to cope with them.

We think that recommendation is a very valuable one in many respects because it not only teaches operators how to recoginize those particular failures, but it should be an excellent training experience on the way things inter-act, and are tied together in the plant. So, it should have additional value for operator training.

As I mentioned in the memorandum on the light bulb incident, the emphasis in this bulletin on getting the cold shutdown, I think is a distraction from what I would give to be the principal emphasis, and that is for our cooling.

Safety grade panel of vital instruments -- Another good recommendation, if ever there was one. I think it could conceivably be important in the scenarios in which there are common cause failures in offsite power, service water, seismic condensor, -- the kinds of accidents we expect to dominate the severe accident risk.

Although, it doesn't rise to the importance that it does in the scenarios like the light bulb incident scenario which may remain potentially a significant contributor to core damage.

Others that stand out -- The operator training on the Crystal River incident and the development of plant specific procedures for the loss of ICSN&I, we think would have a broader application and be of benefit even in the high risk accident scenario.

We question in the text why confining that to the Crystal River incident. There have been alot of others that are equally troublesome.

And, none of the others rise to particularly prominent evaluation, one way or the other.

The bottom line which we didn't put in the chapter, but which we are going to put in a memorandum to Harold Denton, are these recommendations.

Neither the 22 recommendations produced by the task force, nor the B&W characteristics that spawned the concern in the first place, really focused squarely in on the very high -- high consequence and the accident spectrum.

The fact that B&W plants, we don't think are anymore or less susceptible to these accidents than CE or Westinghouse plants doesn't mean it's all right to forget about them.

We think rather more regulatory attention should be devoted to that and to the spectrum.

For our part, we're attempting to do the interim reliability evaluation program, to highlight how susceptible plants are to that kind of accident.

The NUREG and other NRR, and I&E activities, I think, could stand to have a little more focusing placed upon the severe end of the accident spectrum.

There's a suggestion among the recommendations to provide performance criteria for anticipated transients, and by implication, some of the abnormal transients as well.

If those performance criteria deal with the extent of the excursion in pressures and temperatures and other parameters, that's all well and good. But, we think much more value would be had by extending them to reliability criteria that may be deterministic, may not necessarily be probabalistic, but that deals with diversity, redundant -- redundancy analysis for susceptibility to common cause failures and whatnot, that such criteria could go a long way to patching the loopholes that TMI and Rancho Seco and the other -- NNI, ICS incidents have suggested may exist in our safety regulations and our safety review.

MR. ETHERINGTON: In that connection, may I bring up the interpretation of GDC-19, as an opportunity to do some of these things.

As you know, the least adequate interpretation of what you do with that requirement is to put a bunch of instruments off in some other corner of the plant, distance from the control room, and then low and behold, back wire it to the terminal boards in the spreading room under the control room and essentially retransmit the same signals that were derived for the signals in the control room to a distant point, which makes them commonly susceptible to such things as fires.

A better interpretation is to go to the instrument routes and provide independent routes of instrumentation for that distant control point. An even better step is to provide independent DC supplies and make an integral shutdown function out of the remote shutdown system.

In short, I'm merely saying there's a basis now, which is a conservative and proper interpretation of GDC-19 that would go a long way to laying alot of these things to rest.

MR. ROW: Good thought. I hadn't thought of that one. That's a good point. Thank you.

I think many of the recommendations that the task force has come up with are valuable suggestions of places to look for improvements but in many cases we do not now know what the feasibility or advocacy of those would be.

And so, we recommend that the instrumentation be a collaborative venture between the agency and the owners of B&W.

Finally, we recommend serious consideration of add-on emergency feed water and high pressure injection systems in the form of a dedicated safe shutdown system which we think would go a long way to putting many of these concerns to rest.

CHAIRMAN ETHERINGTON: Are the other chapters to be modified in any way as a result of this late contribution by chapter 7?

MR. CAPRA: We've had a couple of meetings with B&W and B&W licensees. The most recent meeting was on the 23rd of this month, in which Mark Cunningham basically gave the same presentation as you heard today, and then as a follow on to that meeting, we discussed the -- some plant specific comments on the individual recommendations.

We have modified a couple of the recommendations a little bit, not necessarily based on this work here, but on discussions with the licensees.

For instance, the original recommendation brought about B&W merit guidelines for loss of non-nuclear instrumentation and the integrated control system.

After exploring it a little bit, we found that really plant specific procedures was the way to go and cut out the generic guidelines. It was too plant specific.

We've cut out the -- originally in one of the recommendations, for the equipment that we would require on a safe shutdown --not shutdown panel, but the panel of vital instruments, we had containment temperature listed.

After discussions with the licensee, we found that that may not be very productive to try to do that, there's too many places you can measure temperature and we're really not sure that was really valid.

We really needed that. Do you want to talk about the implementation?

MR. TEDESCO: We want to now somehow tie the report that we've done in section 7 and try to tell you where we're going with it.

When section 7 was prepared for Mr. Denton, because he was confronted with -- We have given him 22 recommendations

How do we know, that we should go ahead and do all of them or just parts of them or do half of them or do none of them. We want to know in some way, what benefit he might derive in improving the sensitivity of the B&W plant if he went ahead and implemented them.

And when they have to come up with that table, 7.3 -- Yeah, 7.3, we have used that now to guage in our judgment how we might com e up with a recommendation on implementing our 22 actions.

You see, some of them have a high benefit and others are practically nothing, based on that evaluation, and realizes that we're not going to be guided absolutely by that but it's certainly helpful for us in -- their own approach towards a recommendation.

So, we have developed a table --

CHAIRMAN ETHERINGTON: Was it 6 and 7, something of an after thought or was it originally --

MR. TEDESCO: Yes, an after thought. After we had gone through our work, accelerated effort that we had, came up with these 22 recommendations, we were briefing Mr. Denton and the safety directors on our recommendation and that's when the question came up, well, what do I do now, how do I go ahead and implement these or don't do anything. How will I know what benefit might be derived from them.

CHAIRMAN ETHERINGTON: Did the question come up, which of these items apply to all pressurized water reactors?

MR. TEDESCO: No, we were only geared toward the B&W operating plants on a generic basis.

CHAIRMAN ETHERINGTON: But some of these would be?

MR. TEDESCO: Some of them are --

CHAIRMAN ETHERINGTON: Would be?

MR. CAPRA: I think you mentioned, Mr. Etherington, that section 6 and 7 were after thoughts. Section 6 was always in the report, the Crystal River studies. It was Section 7 that was --

CHAIRMAN ETHERINGTON: No, I was only referring to 7.

MR. CAPRA: Also, as Mr. Tedesco pointed out, the Probabalistic Analysis Staff did do this assessment and we took it into consideration in trying to prioritize these recommendations.

However, this is -- We only use this, I'd say, a factor of one-third in our consideration. There were two other things that we wanted to look at.

Originally we had -- were seeing these recommendations being implemented into the action plan, like Mr. Tedesco covered that earlier in the very beginning of the presentation when it was decided that these recommendations would have to be implemented outside the action plan, that also made us have to go back and consider the action plan in our prioritization of these items.

There are similar, more existing requirements which may cover some of these recommendations in whole or in part. You have to look at the detailed scope in the action plan.

So, in trying to prioritize these, we look at the priority grouping and decision grouping, which these similar recommendations appear in the action plan itself, and the third item that we use in considering the priority was the meetings that we've had or the feedback that we've gotten since the draft report was issued on April 2nd, and that's from the NRC Staff, from the previous two meetings we had with the ACRS, from the two meetings with the B&W licensees and B&W itself.

So, we have attempted now to try to prioritize them basically into two categories, priority 1 and priority 2. The way we foresee or the way we intend to recommend to Mr. Denton that these recommedations be implemented, is by and large on a plant specific basis.

Now, the way we intend to do that is to give Mr. Denton some generic implementation guidelines by which we'll prioritize them into either category 1 or category 2.

And, we've also broken it down a little bit farther, which we'll show you probably on Friday, and two different level -- action level. But, who has the initial action or whether it's a joint venture by industry and the NRC Staff, --

MR. TEDESCO: Would you clarify what we mean by 1 and 2?

MR. CAPRA· Okay. I'm going to. Priority 1, out of the 22 recommendations and taking those other three things that I mentioned into consideration and prioritizing them, 10 out of the 22 recommendations, we feel are in this priority 1, and they're items that we feel should be scheduled and implemented and commenced in the very near future, and if necessary, and restructuring of priorities and resources to accomplish those is warranted.

For the remaining 12 recommendations, we feel that those recommendations should be implemented, however, they should be implemented in a manner that's consistent with existing priorities and resources.

Now, as a result of the meeting that I mentioned took place on the 23rd of this month, the owners have agreed to submit to us detailed comments, plant specific comments, on the individual recommendations.

The usefulness of that meeting, I think, was important, by the fact that we did find out that some of the work on these items has already been gone, not necessarily complete, but it is underway and it varies from plant to plant.

Other plants where the recommendations in the report themselves may have been very perscriptive in nature, some of the utilities proposed alternative means of accomplishing essentially the same goal.

So, rather than blindly going out and recommending some false implementation schedule, based on January 1st, 1981, '82, '82 --

MR. TEDESCO: They wouldn't be so arbitrary.

MR. CAPRA: No, they wouldn't be arbitrary. We feel that Mr. Denton can take our priorities and action levels, give them to the new division of licensing, couple those with the plant specific comments that we'll get in and go out with individual implementation letters to the licensees.

It may involve meetings for some of the more generic ones, with B&W and B&W owners group if they decide to form a subcommittee to handle these recommendations.

CHAIRMAN ETHERINGTON: What guidance are you giving us to what reactors they apply -- these recommendations apply to?

MR. TEDESCO: They apply to all the operating B&W.

CHAIRMAN ETHERINGTON: All operating B&W?

MR. TEDESCO: Yes, sir, that's the whole purpose.

CHAIRMAN ETHERINGTON: And any additional ones for those which are not operating included here?

MR. TEDESCO: I guess. We'll look at them all.

CHAIRMAN ETHERINGTON: Pardon?

MR. TEDESCO: We will consider them all in all the plants. But right now the immediate problem --

CHAIRMAN ETHERINGTON: In all the operating plants?

MR. TEDESCO: The operating plants, yes.

MR. ZUDANS: On this table 7.3 that you made reference --

MR. TEDESCO: Yes, sir?

MR. ZUDANS: Is this L, M, and H classification stricly by the Probabalistic Branch or is it input from task force?

MR. ROW: It's all ours.

MR. ZUDANS: And how does the task force feel about these? -- how this picture would really look like?

MR. TEDESCO: We have quite a bit of discussion on them, and yet, -- If we had a choice of doing it, we probably would have made some changes, and I know yesterday we did -- on some changes that they had agreed to.

But when we overlay our own prioritization, they came out pretty good.

MR. ZUDANS: Systematic rather than accident?

MR. TEDESCO: Yeah. -- I have a personal concern about, what was in the deposition that was changed, about the seismic design and the high -- and so on.

But I understand how they were approaching it, and so --

MR. CAPRA: We don't have a slide on it, or even a handout, but I can tell you that out of the 22 recommendations which we considered this priority 1 and which we consider priority 2, he could write them down to quantify their recommdations.

The first four recommendations all dealing with auxilliary feed water are classified as priority 1.

CHAIRMAN ETHERINGTON: Wait a minute. What are we referring to?

MR. TEDESCO: 7.3 --

CHAIRMAN ETHERINGTON: The tables?

MR. ZUDANS: 7.3, table --

MR. CAPRA: I think if you take -- 1, 2, 3, and 4 are all priority 1. 5 and 2, 6 is 1. 7 is 2. 8 is 2. 9 is 2. 10 is 2. 11 is 2. 12 is 1. 13 and -- Hold on just a second. I can't remember 13 and 14.

Yes, 13 and 14 are both one. 15 is 2. 16 is 2. 17 is 3. 18 is 2. 19 and 20 are priority 1, and the remaining two recommendations, 21 and 22 are priority 2.

On a couple of the recommendations where the Probabalistic Analysis Staff has not necessarily assigned a high potential benefit associated with the recommendations that we had given a priority 1, it was based on the other two items that I had mentioned, seeing where it falls within the action plan and the interaction we've had with the staff, the HRS and the licensees and feasibility of doing these.

MR. ZUDANS: They compare 9 and 19?

MR. CAPRA: Pardon me? 9 and 19?

MR. ZUDANS: Yes. Somehow I feel that you couldn't do 9 until you did 19 and 19 is priority 1.

MR. CAPRA: Well, we've also had discussions with the licensees on that and I believe one of the things that came up, -- I don't know how firm it is now -- The licensees believe they can best accomplish three of these, they can best accomplish three of these recommendations if taken together and that's 9, 10 and 19, by coupling those.

One of the suggestions was the identified performance characteristics that Mr. Taylor had mentioned earlier. They would take these performance characteristics and go back and take a look at operating history and find where it misses the mark and then in what system, what areas, and look at potential improvements for 9 and 10, based on 19.

But 19, we had envisioned originally, as a long-term solution to the problem, equally applicable to all light reactors, not just B&W plants.

MR. EBERSOLE: May I ask some questions that maybe address the problem in a different way. I'm going to go forward toward less conservative methods of cooling in emergency.

I'll say, -- The first question is, are we going to do anything to these plants based on the assumption that we can, in fact, always put feed water into the secondary side.

Are we going to satisfy ourselves that that will be a mode of operation that we can assure that we do not have to give serious consideration to a mode of operation wherein we cannot put feed water in.

Before you answer that, I'm going to say I crossed the full spectrum of possible pressures that we might get by providing design modifications to invoke blowdown of the secondary system with qualified equipment.

If we do that, I'm asking, can we say yes, we'll always put water in the secondary? I'm going to build on that -- If you say yes, then I'm going to build on it differently than if you say no.

MR. TEDESCO: Let me just -- Clearly, the ration that we in the task force moved, was to give ourselves the greatest assurance that we would have available a secondary -- the aux feed water systems and do all that we can to insure the availability and that's clearly what we have moved for.

MR. EBERSOLE: Would that include qualified blowdown to increase the possibility of getting water in?

MR. TEDESCO: Well, what that means -- First of all, I want to make sure I answer your question, so I don't guess at what you're saying.

I'm saying, does that include a provision, much as the Board has used on the primary and only circuit they have, to blowdown the secondary to increase the or enhance the possibility of putting water in the secondary of a low pressure pumping system?

Yes, other -- river pumps.

MR. TEDESCO: I haven't looked at it in that way.

MR. EBERSOLE: Well --

MR. TEDESCO: I know we have atmospheric dump valves that are like a BWR --

MR. EBERSOLE: They're not qualified. The electric and power supplies are presently on --

MR. TEDESCO: In many directions, maybe not as qualified as much as other --

MR. EBERSOLE: I'm saying now they are not qualified even to the extent of having them on diverse electrical power supplies.

MR. TEDESCO: But, there's also -- They have a capability of being manually operated.

MR. EBERSOLE: Well, I think -- I guess what I would say is, if you do blowdown the secondary side, I could be comfortable on the basis that you would always get water into the secondary.

MR. TEDESCO: That's some low pressure by some means.

MR. EBERSOLE: Now, then, that leads to two more problems. Can I always guarantee the transport process from primary secondary, and here I have to diverge from -- get away from the boilers and consider separately B&W and the combustion Westinghouse plants.

Would the combustion plants and Westinghouse plants which were similar, can I, in fact, depend on that for conviction to always provide a transport mechanism, from primary to secondary?

Can we -- Based --

MR. TEDESCO: I believe that your -- That that capability is there.

MR. EBERSOLE: Well, the problem is --

MR. TEDESCO: And I think -- Under certain conditions, yes.

MR. EBERSOLE: Well, the problem is at the moment, which is many questions remain, the influence of noncondensables blocking this process. And, we need to invoke reflux condensation which is a questionable process at this time.

MR. TEDESCO: But that's only when you start to void the primary system.

MR. EBERSOLE: Say it again?

MR. TEDESCO: If you start voiding your primary system --

MR. EBERSOLE: This would be the result of voiding the primary system, in part?

MR. TEDESCO: Um-hum.

MR. EBERSOLE: But final cooling would have the assistance of the secondary?

Well, anyway I'm saying -- I don't know whether I can depend on the transport process in the CE Westinghouse designs or not.

Now, if I jump to combustion, -- and consider it as -- system, I think I can say, I can believe evaporative cooling off the primary to the secondary, because it doesn't need a reflux condensation.

This is with the primary system partly filled, and with the appropriate level instrumentation and again, the perrogative to blowdown to low pressure on both sides.

Now, I haven't got yet to the final thing which nobody wants to test in real plants and we're getting very slowly along in the reliable tests, and that is bleed feed off the primary alone.

And, at the bottom of the line, we have to say, are we going to have to depend on bleed feed?

MR. TEDESCO: I don't think we're going to require it absolutely, but I think the capabilities exist that we recognize it.

MR. EBERSOLE: In a vague way.

MR. TEDESCO: -- and we do give credit for it, it's there.

MR. EBERSOLE: What do we need to do to bring the reliability of that process up to an appropriate level?

I don't know.

MR. ABBOTT: It's something that needs alot of study. -- inadequate core cooling guidelines, will naturally lead into a feed and bleed mode. So, whether NRR wants to admit it or not, if sufficient subcooling is not verified, the operator will, in fact, end up in a feed and bleed mode.

Those are the current requirements as they exist today. He can't throttle that pump, the HPI pump, until he verifies a 20 to 50 degrees subcooling.

And, if he loses auxilliary feed water, and if the primary side goes to a pressure such that PORV or the safety valves lift and HPI pump comes on, that pump will stay on until he gets his auxilliary feed water back in the secondary side.

MR. TEDESCO: That's Crystal River?

MR. ABBOTT: That's Crystal River, that's Three Mile Island.

MR. CAPRA: That's in the B&W small reg guidelines to direct the operators to --

MR. ABBOTT: You keep hearing from the Staff that the NRR has not recognized feed and bleed as a viable mode of core cooling.

MR. TEDESCO: I think the Staff -- is not required, and we have not required that of --

MR. ABBOTT: You are requiring it in accordance with the short term lessons learned, Sections 2.17, inadequate or cooling.

MR. EBERSOLE: Well, Ed, there's a method preferable to feed and bleed which is evaporative cooling to the secondary, which implies you don't have a continuous -- through the primary.

MR. ABBOTT: All I'll point out is that all this discussion about feed and bleed is rather new, given a point that it's already being required of plants.

MR. EBERSOLE: But it isn't. The grounds for that are not well-established.

MR. ABBOTT: You don't call it that, but the eventual loss of feed water, the eventual line up of the plant will be, in fact, that these --

MR. TEDESCO: I know -- We asked licensing to evaluate transferability without feed water systems and we have a series of analysis back.

It was my understanding, from guys like Brian Sheron that we have not established a requirement that all plants have a design basis.

MR. ABBOTT: All I'm saying --

MR. TEDESCO: I know what you're saying.

CHAIRMAN ETHERINGTON: Well, then, with serious transients having been associated with the ICS and the NNI, I'm supposed to find that on the criteria -- number 5.

I notice that you show small effects there.

MR. THATCHER: I think Mr. -- that a little bit. was one of the problems that we don't really know how much we can improve upon those systems. They are a single, basically a single channel or a single track system.

Now, we're -- We've been exploring ways to auctioneer inputs to that system or have fast transient capabilities on some of the power supplies, but still we come down to certain single point vulnerabilities no matter what you do, unless, like you said, scrap the whole system and start over.

MR. EBERSOLE: May I --

CHAIRMAN ETHERINGTON: It's not really that the benefit is small, it's the benefit versus effort, effort ratio that's small then?

MR. THATCHER: Yeah, that's probably true because there may be alot of -- involved and your gain is very small.

MR. EBERSOLE: Mr. Chairman, he can improve it basically because no matter what he does to it, it still has single point vulnerability the way it's being done now.

I thought the improvement or one, the significance of the Crystal River and the Rancho Seco event was revealing on a generic basis that NNI contained instrumentation critical to safe shutdown, which was not present in the safety grade, but austere configurations against other safety systems.

I'm talking about just instrumentation and indication. I mean, the recorders and indicators.

MR. THATCHER: That's true.

MR. EBERSOLE: So, I saw where the improvement would be in extending the scope of indicating and reporting in safety complexes to back up the NNI.

MR. TEDESCO: And that's number 6.

MR.EBERSOLE: And that really -- That was rude to not saying you could improve NNI per se.

Is that right?

MR. THATCHER: Can I just but in here for more time? Those panel of vital instruments, is that the same thing as the safety --

MR. EBERSOLE: I thought that's what that was.

MR. THATCHER: That instrumentation may not necessarily be there for -- next to the HPI pumps or the safety related equipment which the operater is going to be controlling.

For example, in the Crystal River accident, the operator balancing should be closed without having an accurate instrumentation because one of -- of it's scale.

MR. CAPRA: The items we've selected -- This is related to the safety factor and the action plan, but it's not as encompassing. There is not -- There is no control mechanism on this panel to control any of those things.

It's a place that the operator can find out plant status. This is not meant to replace the safety state vector.

MR. ABBOTT: If the information is not going to be used to control anything, why is it there?

MR. TEDESCO: This is telling the operator there is a rapid indication of the plant status.

MR. ABBOTT: Exactly. And it'll tell him something, right?

MR. TEDESCO: Right. And then he'll know where to go, --

MR. ABBOTT: The only point I'm making is that if it tells him that he's not subcooling, he has to verify HPI flow, he may have to go on the other side of the control room, to adjust his flow in order to -- separated the control functions from his indication.

MR. EBERSOLE: In many cases -- Ed, that has a problem. If you localize the indication and the control, they become a new source of vulnerability to some incident.

So, it's deliberate sometimes that you make these systems passive and use voice transmission to distant points to avoid a new common point of damage.

Otherwise, that becomes a new focus where you can create simultaneous damage by fire or whatever. You can't extend damage by voice due to fire.

DR. LAWROSKI: Jessie, wasn't it by a voice transmission that there was a misunderstanding in what the temperature on the hot pipe was.

DR. ZUDANS: Pressure?

DR. LAWROSKI: No, no. It was a temperature.

DR. EBERSOLE: I'm sure there could be.

DR. LAWROSKI: That was misunderstood by 50 degrees instead of being 280 -- stated to be 230.

DR. EBERSOLE: Well, it might be a good idea to televise the output of the remote instruments to the control point. At least a television system wouldn't transmit this --

DR. LAWROSKI: The tail pipe -- I was thinking.

DR. EBERSOLE: It was a tail pipe.

DR. LAWROSKI: Of a bad transmission. Later it gets verified that it did read right. That it was much higher than was believed to be the case by the guy at the control room.

DR. EBERSOLE: Well, the first reaction is to use voice transmission off of a passive set of instruments.

DR. LAWROSKI: Yes.

DR. EBERSOLE: But the second is to put on remote TV. You know -- and put that output right at the

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control points.

DR. ZUDANS: We call it out.

DR. EBERSOLE: Right.

CHAIRMAN ETHERINGTON: Well, we may revert to this topic, but are there many impressions at the moment? Now, we are having some industry presentation -- are we? Yeah. Are you Novak?

MR. NOVAK: Yes. I'm Julian C. Novak, General Superintendent of Power and Engineering and Construction at Toledo Edison. The comments I've got touch upon some of the areas we've already discussed today, but I think I'll present them anyway, and maybe I have a slightly different slant than some of the things you may have heard.

I'm here today on behalf of Toledo Edison and also as chairman of the B&W Owners Group Executive Committee. My statements will reflect the viewpoints of both. It will provide some brief comments relating to the NUREG 0667; its status and a general assessment.

I'd like to first make a couple of comments regardingthe B&W 177 Fuel Assembly Owners Group. I think at times there is confusion as to what we are and what our purpose is. The group is an informal organization of representatives of utilities owning 177 fuel assembly

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B&W nuclear steam supply systems. The group is not a legal entity. It has no funding, and it cannot make commitments to the NRC or to anyone else. Only the licensees can do that.

It exists to provide a forum for the members to address mutual problems. It consists of a standing executive committee and appointed subcommittees on specific topics of general, mutual interest.

The TMI-2 subcommittee was one of our subcommittees. There is no Crystal River 3 subcommittee since we felt that Crystal River issues were plant specific rather than generic.

We can continue to exist and effectively address truly generic licensing matters of the NRC will allow us to decide what matters to pursue jointly after initial contact through established licensee-NRC communication paths.

If the NRC chooses to decide on in its camp which matters are owner group matters, then the owners group concept will fail.

CHAIRMAN ETHERINGTON: Is Florida Power in your group? You mentioned that you don't include Crystal River?

MR. NOVAK: We don't have a Crystal River 3

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subcommittee.

CHAIRMAN ETHERINGTON: But is Florida one of your members?

MR. NOVAK: Yes. All the utilities having a 177 fuel assembly plant including consumers are in our executive committee.

CHAIRMAN ETHERINGTON: Right.

MR. NOVAK: As I say, we set up subcommittees for whatever topics we feel are of a mutual generic interest.

DR. LAWROSKI: And you said that the Crystal River event was not a -- you didn't consider it be a generic?

MR. NOVAK: That's right. We have set up no subcommittee on that. The Staff has made contact with our Three Mile Island 2 subcommittee, and we have respected responding.

DR. ZUDANS: It's generic.

MR. NOVAK: We don't consider it generic. There are specific plant differences even in ICS and NNI. You take, for example, a Davis-Besse reaction to what happened at Crystal River, and you plan an entirely different set of circumstances.

DR. EBERSOLE: What about Rancho-Seco?

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MR. NOVAK: I can't personally speak for Ranch-Seco.

DR. ZUDANS: Is the utility unique in the INE arrangement or not unique? I thought there was very few exceptions to that.

DR. LAWROSKI: I thought Rancho-Seco was another.

DR. ZUDANS: But that's okay if you consider it does not.

DR. LAWROSKI: I just wanted to make sure I heard correctly.

MR. NOVAK: That's the way we view it as owners. That there are significant differences that we do not feel we can address them on a generic basis. And I believe I heard the Staff saying that, too. But they will be making presentations, recommendations plant specific.

DR. TEDESCO: We did not address Crystal River as being unique.

CHAIRMAN ETHERINGTON: I'm sorry, Bob. I don't--

DR. TEDESCO: We did not think that Crystal River was unique. I think there are generic implications of that clearly.

CHAIRMAN ETHERINGTON: It doesn't mean it has to apply to all, I suppose. It means it can apply to more than one. Is that your interpretation?

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DR. TEDESCO: The whole question on the NNI has been found at Oconee and Rancho-Seco and Crystal River.

DR. EBERSOLE: I thought there was a generic implication in Crystal River in that there were too many NNI instruments that failed.

DR. TEDESCO: I feel that way.

DR. LAWROSKI: Yeah. That's exactly what I thought. That's why I wanted to make sure I heard correctly.

MR. NOVAK: Well, what I'm saying is that not the same failures would occur in all the plants. You might also say that there's generic, you see, in Westinghouse.

DR. EBERSOLE: Yes, you would.

DR. TEDESCO: That's why we set up --

DR. NOVAK: And what I'm saying, then, is in the context of our owners group structure, we would not consider that to be enough generic for all of us to try to address it in a mutual manner.

DR. EBERSOLE: Let me ask you this. Would you consider that aspect of that incident which is the fact that NNI or similar instrumentation may well be subject to single failure effects which would deny the operator of adequate instrumentation, indication and recording to

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properly shut the plant down.

DR. ZUDANS: That's in a special context.

MR. NOVAK: I'm not sure of the question.

DR. EBERSOLE: I'm saying what Crystal River showed was there were too many indicating and recording instruments that failed so the operator, in effect, was blinded.

There was a generic implication that might affect all plants including CE and Westinghouse and any others. Did you -- does your group look at it in that context that it revealed a possible common deficiency in what we call NNI at Crystal River, and you might call something else?

MR. NOVAK: We look at in the context as to whether there are sufficient similarities in our designs to approach the matter in a joint fashion.

MR. EBERSOLE: Yeah. Okay.

MR. NOVAK: And our finding in this case is that there is not. With that, let me say that we as utilities do appreciate the opportunity to meet with the ACRS subcommittee and express our feelings on the report and other matters.

Since your last subcommittee meeting, representatives of B&W utilities met with Messrs. Capra, Tedesco and the staff as was mentioned last Wednesday on

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the 23rd. Specifically, this was to discuss Section 7 of the report generated by the Probabilistic Assessment Staff. As was indicated, Section 7 wasn't completed at that time. We reviewed the methodology of how the Probablistic Assessment Staff was proceeding.

And several tables of the preliminary results were made available at thatmeeting included a tabulated effect of each report recommendation on the frequency of several events.

Some initial reactions were provided by the utility as Mr. Tedesco indicated. Those representatives were present. The remainder of the Wednesday's meeting was spent discussing some of the technical bases and background of each of the recommendations of the report.

As I said, the meeting was extremely enlightening to us. I'd also like to reiterate the general impression that was expressed at the last subcommittee meeting, and that is for the time period allotted to the Staff, the development and issuance of such a document is a commendable effort.

Two of the recommendations, in particular, are now being evaluated by B&W owners and B&W to see what new type of program could be developed to help meet these recommendations. Specifically Recommendations 10 on the steam generator sensitivity to secondary coolant conditions and 19 on the performance criteria for anticipated transients.

DR. ZUDANS: Are you going to add 9 to that, too?

MR. NOVAK: They are rather together so they are being addressed. We feel these are appropriate for further study on a historical operations approach. The details are still being looked at, but we feel a short term effort in this area could be quite rewarding to the owners regardless of the regulatory aspect of it.

However, the basis for which the report is commended, namely that the short turn around time, afford to it we feel must also be the basis for which it might be criticized. The concern is that the scheduling of deadlines on a non-technical basis and the inflexibility of these schedules. These characteristics or this characteristic is specifically and more generally evident in that most recent NRC activities collectively established unattainable goals for both the staff and the industry.

This practice is and will continue to result in rapid implementation of requirements that may be unrelated to overall plant safety, unimportant to risk evaluation, and actual counterproductive due to diversion of industry resources from truly significant activities.

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To illustrate, I'd like to use the NUREG 0667 as only one example of the problem. As we know on March 12, NRC management set up a task force to discuss the generic aspects of the operating experience of the B&W plants.

Two weeks was assigned to this activity. This study was to be made in conjunction with all the actions already taken or proposed in response to TMI-2 as referenced on page 1-2 of the report.

On April 3 over 200 pages excluding Section 7 were used -- were issued including 22 recommendations. Table 2-1 cross-correlated each recommendation to a Section in TMI-2 Action Plan. Now, a generic based consensus of engineering judgment has assessed the effect of each recommendation on their frequency of selected events and a likelihood of incidence and accidents.

This is latter assessment we've just now seen. The overall results are interesting, but it's our contention that the recommendations, comments and evaluations are excellent only as to input to other c-going implementation activities for further evaluation. They need not be acted upon independently. Some general comments to elaborate: one, the qualitative consensus assessment of Section 7 is through our understanding

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generically based. This certainly would be appropriate in looking at implementation of recommendations in a plant that has not previously implemented any specific or alternative approaches.

I think we've kind of agreed on that. Many of the recommendations are interrelated with other recommendations in this document as well as with other activities that could effect the overall comparative value for a specific plant.

We're really extremely disturbed to here that while the task force was charged to perform their study in conjunction with all the actions already taken or proposed in response to TMI-2 accident, which really we feel should have been also encompassed in non-TMI-2 accidents being taken or proposed -- we're disturbed now that we're hearing that it's now been told its recommendations will not be incorporated in the TMI-2 Action Plan, which, in fact, was a task force recommendation on addressing the recommendations.

We feel these things are really all integrated together. That they really need to be handled together rather than separated out. The other comment I've got is several of the recommendations are over prescriptive and you've heard a little of that this afternoon, too.

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In some cases, detailed fixes are enumerated. These do not take into account difficulties of installation or operation of that one item as compared to alternatives or even the real need of the modification, as determined by analyses or operational restrictions.

Although discussions with the task force members show a flexibility in this area, since TMI-2 obvious oral interpretations of published recommendations have been increasingly difficult to attain from an implementation audit group after a task force has disbanded.

And I'm concerned about some of the things I've heard today in that regard because the word --

CHAIRMAN ETHERINGTON: Did you say inflexibility?

MR. NOVAK: Inflexible. We have heard some good words as to some flexibility on that, but we're concerned that the printed word is what's going to be incorporated two months from now or whenever it may be.

DR. ZUDANS: In other words, when a task force is disbanded, you think you won't have anybody to go to?

MR. NOVAK: Yes.

DR. ZUDANS: That's a good assumption.

MR. NOVAK: The basis of the --

DR. ZUDANS: There should be additional action.

MR. NOVAK: The basis of the technical concern really needs to be the recommendation is what we're saying, not the prescriptive approach that we're hearing.

How the basis is resolved should be a matter such that the basis -- or should be done in a matter the basis is addressed adequately by analysis and/or operational equipment or hardward.

Another comment is the recommendations need to be integrated. And this a little bit picks up on previous comments, too, with other ongoing approaches for continuity and priority. In discussions with the task force members, some recommendations appear to rely on fault fixing and even an event oriented response is made by station personnel during an event. These activities do not appear to be consistent with our major post-TMI thrust.

An extremely large effort in design operation and analysis by the industry is trying to obtain symptom oriented responses to assure that safe, stable shutdown can be reached and maintained relying on as little as possible on the operator or maintenance personnel to correct the initiating problem.

We believe that it's more -- we believe this is the more effective prioritization of industry resources to work on the symptoms.

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DR. ZUDANS : I would like to understand one -- everyone of these 22 items was correlated to some action plan. How is now to be perceived and to do implementation that I'm saying, one, through the action plan or through this independent action? How is to be understood this? It's not being incorporated?

DR. TEDESCO: One thing this has happened that all the commissioners that they are complaining the task action plan -- they are closing it out because it basically represents their response to a presidential commission. So because of that you are closing them out and say here's our response to it.

All of a sudden you come along with other actions that are related to it, and although our original recommendation was, and I think we ultimately feel the same way -- that it should be part of that -- Mr. Denton is reluctant to reopen the task action plan so he wants us to recommend to him an implementation program recognizing that there's a relationship between our recommendation and the plan.

And it's necessary we do phase into it.

CHAIRMAN ETHERINGTON: No. Yours will not be in the action plan. But will the pertinent action plan items be more or less in your recommendations?

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Is anything falling down the cracks there be- tween --

MR. NOVAK: Not that I know of, right, for all phases.

MR. CAPRA: That Table 2-1 may be a little misleading where we tied it to the Action Plan. What we intended -- what we had intended by doing that is if you actually go and look at the title of what that is the recommendation fits very well that we have made into that section of that action plan, but if you look at the scope of the presently existing requirements, some are close but none are the recommendations we have made.

So what the purpose of that table was supposed to be is to show when either the taks force or actually what we had perceived as Roger Mattson's TMI Action Plan, if he took these recommendations, this is guidance to him to take these recommendations and now address the scope of our recommendations in the Action Plan.

So it was not meant necessarily to be a one-to- one correlation there. There are some recommendations that we have made that are very close, as I mentioned, to things in the Action Plan.

DR. ZUDANS: They are very much one-to-one in many cases, but the biggest issue -- just think yourself in the position of industry. There are two items now. Each of them may supplement each other or may be the same things and if I take an incremental fraction of total objective, the total objective of the same ladder, and for action here, that sounds rediculous.

MR. CAPRA: Let me give you a good example. I can understand industry's concern.

DR. ZUDANS: Also my concern. I think it's everybody's concern.

MR. CAPRA: The task force is concerned also.

DR. ZUDANS: Just because what you mentioned doesn't have time --

DR. TEDESCO: Let's condense things now. It wasn't only Roger Mattson --

DR. ZUDANS: Well, of course, he has pressure just like everybody has pressure, but it's something that he just cannot be --

DR. TEDESCO: No, we're not under illusions. I mean it's more convenient to put them together, but it doesn't mean that just because we don't do it, we're going to lose it. I think it places a much higher demand upon the management of our program now to make sure that indeed we do phase these things together.

MR. RAY: But you may squeeze industry in

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priority reconciliation.

MR. CAPRA: We're tried to take that into account in our priorities here. That's made fairly clear in our implementation memo we're proposing to forward with this document to Mr. Denton to insure that the two requirements are not mutually exclusive. And that you can't look at one without the other.

DR. ZUDANS: Okay. If you make that, that sounds okay.

DR. TEDESCO: Well, our priority in implementation do not have a date on it. They do not say "do x" by 1981 in the Task Action Plan of 1980. We're not saying that. All we're saying that we have broken these down into priorities one and two. And one means that what we think is very important and is necessary to have to readjust your schedule and priorities to do it.

But we're not saying what action should be; what they do.

DR. ZUDANS: Well, if your transmittal of your input to Dick -- if you would state that you assume that no action plan corresponding to this item will be implemented individually and vice versa that these should be implemented together. That would solve industry's problem and also solve my problem.

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MR. CAPRA: Words to that effect -- in the cover memo there is also a closure which lists the priorities of these action levels that I talk about, and I have -- we have also indicated the Action Plan item again right along with it to insure that it is cross-correlated. And in addition, it's not just the Action Plan.

There are a couple of other recommendations with bear close similarity to other staff documents.

DR. ZUDANS: I think what you need is a Super Action Plan.

MR. CAPRA: You know. We brought up before an ongoing living type of document to take care of these.

DR. TEDESCO: And in a few more weeks you're going to have a Crystal River report, and it's going to have some more recommendations.

DR. ZUDANS: What will industry do if they can't talk to Bob? I see your concern.

CHAIRMAN ETHERINGTON: I think we've interrupted Mr. Novak for a long time.

DR. ZUDANS: I'm sorry.

CHAIRMAN ETHERINGTON: It's all right. I said we, I said.

MR. NOVAK: Well, it does bring out our concerns of really coordinating and correlating these efforts, and it's not just the Three Mile Action plans. There are also bulletins; 7927 is in there. There are many activities that we have ongoing that need to be brought together.

Several of the recommendations are based on items that, as we've indicated before have been previously addressed by utilities, but the Task Force may not have had the time or information to review in their schedule to evaluate in detail.

In some cases, generic industry activities are ongoing now to address in more detail requirements that need to be provided to cope with plant status evaluation. The comments that I've made are not really new revelations. In my opinion, they are comments that can be attributed almost entirely to a new wave of commitment without knowledge, as we're calling it.

Schedules are made without flexibility and knowledge of impacts. Commitments to schedules are made requiring expedited activity from every support group involved. Conflicting priorities tax all available resources, and the orderly resolution of items reverts to implementation by crisis.

The confidence level of the overall result is

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degraded. Whether it be a report generated by the NRC or the implementation of modifications by utilities, the analogy is appropriate. This essentially seems to become a new way of doing business.

What I proposed previously is that the task force effort should be an input to the evaluations and implementations ongoing in other programs. These can then benefit from the positive results of the task force effort and assure their proper prioritizatian and integration.

I'd like to make a couple of comments maybe reiterating a little bit what I've said as candid observations that I'll attribute solely to myself rather than as a representative of the owners group.

When it comes to the interactions between licensees, and I'll presume applicants too and the NRC staff, we don't really feel we've learned many of the lessons from the post-TMI 2 reviews. In fact, through the disappearing task force approach, as I'll call it, we've polarized ourselves even more.

Scheduled pressure task forces seem prone toward pride of authorship; put themselves in print before getting broad viewpoints, and then perhaps ask for comments with the propensity toward defending their work rather than being objective. I think I've seen some of that today with Section 7.

Then as I've mentioned before, the task force as it stands, and the auditors come along and say, it wasn't our idea. We're just here to enforce it. Such an approach isn't enhancing a spirit of cooperation between us. Thank you.

CHAIRMAN ETHERINGTON:Would it be possible for you to take Table 7-3 and go down the items and tell us which gives you particular trouble, and perhaps in a couple of words why? Do you have a table 3 handy?

DR. ZUDANS: 7-3 you mean.

CHAIRMAN ETHERINGTON: 7-3. Is this difficult or not?

MR. NOVAK: It's difficult from the standpoint, again, that I really can't -- I can state for Toledo Edison on the items, but I can't really speak for each utility because they are all different in different degrees. As far as the way they are approached, we feel like for example, the first four really are all essentially one item because we're saying there were some interactions on these issues.

As we mentioned to the staff, one, I guess, that's bothering me right now is the one that says there should be diversity of contaiment isolation by using the radiation monitor. Well, as I see it now, those words are going to stay in print.

But the utilities have indicated other possible approaches. Now, how are the people who are going to implement these task force recommendations going to look at that? It says put in a radiation monitor as an isolation segment.

DR. TEDESCO: That is very clearly specified in the task action plan. Clearly that one there is open page.

MR. CAPRA: That is a one-to-one correlation with an item under 2E-42 or whatever.

MR. NOVAK: It doesn't make me feel any better.

DR. TEDESCO: No. That one there does not apply to all.

CHAIRMAN ETHERINGTON: Of course, three of these are just priority two anyhow. It's only item -- well, no, I'm on the wrong page.

MR. NOVAK: I said really not able to go down each one because I don't have --

CHAIRMAN ETHERINGTON: Then, if it doesn't seem a practical thing to do, forget about it.

MR. NOVAK: We will be doing it on a utility

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basis on our response as Bob had indicated.

MR. CAPRA: Can I make a couple of comments on this?

MR. RAY: Could I ask a question first? Are you going to prepare a submittal of your comments? Is there a record anywhere of these comments other than going in this transcript? Do you have a handout? Are you going to have a handout?

MR. NOVAK: My comments are broad on the concept. Each individual utility will submit our comments on the 22 recommendations to the staff.

MR. RAY: But as of today, there's no document?

MR. NOVAK: That's correct.

MR. CAPRA: I just wanted to make a couple of comments on Mr. Novak's talk. We labored with the problem of being overly prescriptive or overly general and being goal oriented on some of these recommendations.

One of the problems that we have seen in the past is that by being too general, you come up with too many questions about what do you really mean. For instance, we got one today: what do we mean by a sensitivity study to reduce the sensitivity of the one steam generator. Whereas, it isn't very clear if we say we want a high radiation monitor that's going to isolate the containment

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vent and purge. Now, that is very prescriptive.

There are alternative proposals, I'm sure which may even be better to meet the goal and maybe the task force has developed. We have put that in the forwarding letter to Mr. Denton to make him aware that we see that those proposals are feasible to meet the same goal without necessarily installing the exact piece of equipment.

But in order to get the meaning across to the reader of the document, you do, in some cases need to be fairly prescriptive. The second comment that Mr. Novak made about that there is a great deal of effort underway on the part of the B&W licensees to institute emergency procedures on a symptom-oriented basis is true.

I think you may in the past had a presentation on the ATOG Program, Abnormal Transient Operational Guidelines. That comment came out in the comment the other day in the meeting with the licensees in which they said you have recommended here that procedures be developed for loss of ICS and NNI. And I say that appears to be in conflict with our ATOG program.

We don't want a procedure for loss of NNI and ICS. We want a procedure based on the symptoms, and I've been involved in the ATOG program for almost a year now, and I still don't see how the ATOG program can go much farther than immediate operation action to get the plant in a safe condition. You're still going to need a procedure for how to get the plant from this safe shut-down condition to fix the plant and get on with business, whether it's restart or shutdown or whatever.

It depends on what the fault is. You can only go so far with controlling on individual symptoms so I'm not down playing the ATOG program. It's a very significant program. It's involved a lot of effort, and I think we're going to reap huge benefits from it.

But to say that you don't need individual procedures for individual casualties -- I don't think that's the case. It may use your symptoms to get you out of trouble to begin with, but you have to go some-where, depending on what the fault was throughout the systems.

DR. ZUDANS: The sympton oriented procedure, of course, is one of the best facts on these items.

MR. CAPRA: Yes. It's under -- well, it's under item -- recommendation 219 of short-term lessons learned which is abnormal transients and accidents, and has been incorporated into the Action Plan.

DR. ZUDANS: This mostly relates to thermal hydraulic type of symptoms. Not instrumentation.

MR. CAPRA: What the purpose of it is is the operator sitting there and all of a sudden an event happens. He gets a reactor trip. That's the first thing he sees. What does he do? He's presently maybe varies from facility to facility. Maybe he has 20 emergency procedures. Maybe he has 30 or 40. He doesn't know which emergency procedure to go to.

So rather than thumbing through all of those, he trained the operator on maybe six or seven basic symptom oriented actions that he could take to get the plan --

DR. ZUDANS: No, that's a very good approach. I don't know how you could do symptom oriented NNI and ICS?

MR. CAPRA: No, you can't. But you can -- you can have the problem or you lose power to NNI or ICS, you can use the symptom-oriented approach to get you to an point. But after that you have to go somewhere else. But you have time to do it then.

DR. ZUDANS: I'd like to point out maybe it wasn't quite clear from the agency point to say that. It might be difficult time limit. I think it's an excellent piece of work and whether or not it should be

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together with a task action plan or should be separated it's not important. It's really the question to make sure that will it succeed. And I guess the response is to implement it -- we should know -- should be aware of the situation so that they should be able to. Nothing is cast out in my opinion.

MR. CAPRA: Well, it has been difficult in the past both on the NRC staff and on the licensees because as Mr. Novak pointed out individual task force produced individual documents and had individual implementations to get detailed implementation schedule. You will have this by January 1. You will have it by September 1 or whatever and individual licensees receive letters from various sources. I was on the Bulletins and Orders Task Force.

Our recommendations happened to go through-- our letters happen to go through operating reactors now. Lessons learned necessarily do that. The emergency planning task force didn't do that. So they didn't receive the requirements to know where to prioritize it. I think one of the advantages or at least a lesson that this task force has learned from the various task forces we've on before is that our recommendations are not date oriented. They are priority oriented.

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And they're going to the people who will implement the Action Plan. We perceive Mr. Denton giving these, as I mentioned, to the Division of Licensing under Mr. Tedesco and Mr. Novak. Those are the individuals who will be implementing the Action Plan.

There is no group of implementation auditors as Mr. Novak alluded to. The implementation of individual will be the Division of Licensing.

DR. ZUDANS: Okay. So it's really the same contact.

MR. CAPRA: Yes, sir.

DR. ZUDANS: Thank you. I --

MR. EBERSOLE: Mr. Novak, it occurred to me that maybe you could be prescriptively critical of a prescrip. on? In short, you could give us some precise examples. Erroneous prescriptions quantitified and illustrated in hard terms to illustrate your point?

MR. NOVAK: Well, one I mentioned on the containment isolation and radiation -- high radiation signal.

MR. EBERSOLE: You have a better way maybe?

MR. NOVAK: Some of the utilities in connection with Bob and Mr. Tedesco and Mr. Capra have indicated on the reactor trip signal, for example, could be one --

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one of the proposals made.

MR. EBERSOLE: You imply that you need -- you will always have a trip when you need to insulate. Is that necessarily true?

MR. NOVAK: These are alternatives that we look at. Another possibility could be, say, none are in operation.

What we're saying is what we be looking for as one of the objective that we're after.

MR. EBERSOLE: Well, don't these prescriptive requirements leave those loopholes?

DR. TEDESCO: They would not limit purge in operation.

MR. EBERSOLE: I mean I can see a need for definitive rebuttal to prescriptive requirement. I mean not a general rebuttal, but a definitive one -- to each one.

MR. NOVAK: What I'm saying is the words of the report though should not have such words if they're all -- if there all alternatives that you should put high radiation trip.

MR. EBERSOLE: Well, isn't it in a general sense if you got a better way, cut it out. If it isn't, that's simple to fix.

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MR. ABBOTT: It's been my experience -- if it's writing and the auditor comes up from INE or from NRR to decide whether or not --

MR. CAPRA: Oh, that's too late. That's too far down the road. I mean earlier on.

MR. ABBOTT: I think that's his complaint.

MR. CAPRA: That's one of the reasons that we've asked for the individual comments on the recommendations in the form of letters on the record so that these can all be considered. I would expect that there would be meetings with the licensees before detailed implementation schedule is imposed.

MR. EBERSOLE: All I see is if you can do that is that the prescriptive method to avoid such ambiguity that you get a poor job and nobody ever knows it. I refer to the interpretion of GEC-19 as a case in point. You could define that to get virtually no benefit at all out of ite.

MR. CAPRA: Lessons learned was another good example. You had two detailed letters plus dog and pony shows going throughout the country to try to explain what was meant by the recommendation because they were general, but the licensees didn't necessarily know how to comply in a way that we would necessarily accept.

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MR. EBERSOLE: The question always comes up -- what do you really mean, and I think you should put down what you really mean however prescriptive it may be and give the other side an opportunity to shoot it down.

DR. TEDESCO: I think we are doing the best that we can to meet these problems to implement it.

MR. NOVAK: I appreciate your efforts. Bob. I'm concerned what gets in print, though.

DR. ZUDANS: Is this document going out for a mutual comment?

MR. CAPRA: No. It's our final recommendation.

DR. ZUDANS: You didn't ask for comment?

MR. CAPRA: It's been out since April 2. We've had two meetings with the licensees since.

CHAIRMAN ETHERINGTON: Have you made any substantial changes as a result of your discussions with the licensees?

MR. CAPRA: The two recommendations that I mentioned earlier on the instrumentation for the vital instrument panel and the -- plant specific procedures for NNI-ICS guidelines. But also whether the actual recommendations have changed is not as important. I feel, is the way we have now after meeting with the B&W and the licensees have come to the methodology of implementing

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these recommendations based on what we had said prior to these meetings or comments or whatever we may have taken another approach.

We may have been date oriented to get -- implement these recommendations that way. But based on the interaction we've had with the licensees and B&W this is the best approach that we know.

DR. TEDESCO: Don't forget these are also generic things, and they will vary plant by plant. Some plants have already done a lot of these things, and I think you have to recognize where we're all coming from in that basis. Now, Mr. Novak said they're not going to turn. Well, I don't think they're all going to say that. That's fine for him. The other ones may have something different.

So I think you've got to know what things are based on what we are coming from. This is a way of solving the problem. It's a good way. We already have a safety system in there. We have to change it to satisfy it. It's not to eliminate any plant specific hard work done.

MR. ABBOTT: Another one talks about the method of training effort by giving lectures. Perhaps there is better ways than pure lectures. And it says

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lectures. Someone is liable to say having set through the lecture -- it sounds trite, but that's what we look at. That is what happens in the real world.

It will happen. There's no if about it. That is what will happen.

MR. CAPRA: I don't see anything wrong with that. You need to document your training. I think it's very important.

MR. NOVAK: That's already a requirement of the Commission anyway.

MR. CAPRA: Right. The licensees brought up the possibly a better way to do the training is on shift. And my experience in operating is on shift is not the place to get training. It's fine if nothing is going on, but sometimes there's a lot going on, and the consistency that you get from on-shift training is not the same as you can get in a formal lecture with a qualified instructor and an improved lesson plan, whether a quizz is given or not -- I don't know if there's any benefit to that, but certainly a formal lecture to me -- my own personal opinion is a much better way to train individual than on shift.

And that was the proposal that was offered.

MR. NOVAK: But we're still getting back to the may be avoided by it.

DR. TEDESCO: We really tried not to over prescriptive on it.

CHAIRMAN ETHERINGON: Any more questions?

(The meeting was adjourned April 29, 1980.)

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philosophy of the prescriptive part of it. The oh, I need -- is there a way to determine whether the man has learned a lesson or not. Telling that it must be by lecture, sit down or quizzes, I don't feel that that is necessary in the approach that you have to take.

You sit in judgment and say that your way of thinking about is the way --

DR. TEDESCO: We have to justify the continued operation and why we think it's all right. And we come up with recommendations and -- that's a responsibility we have, too.

MR. NOVAK: They could just as well say I'm sure that the operator is cognizant of the Crystal River 3 event --

DR. TEDESCO: And then the next thing you tell them -- tell me what does that mean? You've got to have some special --

DR. ZUDANS: I don't know why you're so excited about this aspect. It's an easy thing to do anyway.

MR. NOVAK: I'm really talking about broad concepts.

DR. ZUDANS: But there should be better examples than this precription. Why is this prescription -- it doesn't really matter. There are other things that

RISK-BASED PERSPECTIVES

ON

THE RECOMMENDATIONS OF

NUREG-667

I. WHAT WAS DONE?

II. WHAT ARE THE FINDINGS?

III. WHAT ARE THE FURTHER RECOMMENDATIONS?

I. WHAT WAS DONE?

A. WHO: FRANK ROWSOME
MERRILL TAYLOR
MARK CUNNINGHAM

B. HOW: CONSENSUS OF ENGINEERING JUDGMENT OF RISK ASSESSMENT ENGINEERS.

C. WHAT:

1. BACKGROUND RISK PICTURE
2. TABULATE INFLUENCE OF B&W PLANT IDIOSYNCRACIES ON LIKELIHOOD OF:
   a. SEVERE ACCIDENTS
   b. ACCIDENTS
   c. INCIDENTS
3. TABULATE EFFECT OF EACH RECOMMENDATION ON FREQUENCY OF:
   a. LOSS OF MAIN FEEDWATER
   b. ICS FAULTS
   c. LOSS OF OFFSITE POWER
   d. SMALL LOCA
   e. STATION BLACKOUT
   f. ATWS
   g. OTSG OVERFILL
4. TABULATE EFFECT OF RECOMMENDATIONS ON LIKELIHOOD OF:
   a. SEVERE ACCIDENTS
   b. ACCIDENTS
   c. INCIDENTS

II. WHAT ARE THE FINDINGS?

A. RISK PICTURE

1. SEVERE ACCIDENTS

   CONSEQUENCES:

   - Potentially Lethal Doses
   - Potentially Extensive, Severe Land Contamination
   - Dominates Health and Safety Measures of Risk

   SYSTEM FAILURES:

   - Core Melt <u>AND</u> Gross, Early Containment Failure

2. ACCIDENTS

   CONSEQUENCES:

   - No Acute Fatalities Possible Offsite
   - Nc Extensive Offsite Contamination
   - Latent Cancers or Need to Interdict Groundwater are Possibilities

   SYSTEM FAILURES:

   - Core Melt With or Without Basemat Meltthrough
   - LOCA with Gross, Early Containment Failure
   - TMI-Like Scenarios

3. INCIDENTS:

   o No Abnormal Radiological Releases

TABLE 7.1

EFFECT ON FREQUENCY OF INCIDENTS OF B&W

PLANT CHARACTERISTICS OR CONCERNS

B&W PLANT CHARACTERISTIC OR CONCERN	EFFECT ON FREQUENCY* OF: SEVERE ACCIDENTS (LARGE RELEASE)	ACCIDENTS (SMALL RELEASE)	INCIDENTS (NO ABNORMAL RELEASE)
1. SHORT TIME TO SG DRYOUT FOLLOWING LOSS OF FEED	SMALL[1]	SMALL[1]	LARGE[2]
2. FREQUENT UNDER-COOLING TRANSIENTS	SMALL[3]	LARGE[4]	LARGE[4]
3. HEIGHTENED TRIP FREQUENCY	NEG	SMALL	LARGE[5]
4. NNI/ICS FAULTS	NEG	MEDIUM[6]	LARGE[2]
5. FREQUENT OVERCOOLING TRANSIENTS			
A. LOSS OF PRZR LEVEL	NEG	NEG	LARGE[2]
B. NUISSANCE ECCS ACTUATION	NEG	MEDIUM[7]	LARGE[2]

TABLE 7.1 (CONT.)

B&W PLANT CHARACTERISTIC OR CONCERN	EFFECT ON FREQUENCY* OF: SEVERE ACCIDENTS (LARGE RELEASE)	ACCIDENTS (SMALL RELEASE)	INCIDENTS (NO ABNORMAL RELEASE)
6. OVERFEED MAIN STEAM LINE RUPTURE	NEG?[8]	NEG?[8]	?
7. FEED AND BLEED CAPABILITY (HIGH HEAD HPI)	MODERATE IMPROVEMENT[9]	LARGE IMPROVEMENT	LARGE

NOTES:

*BASELINE OF COMPARISON IS THE WASH-1400 RISK PICTURE FOR SURRY.

[1]LOSS OF STEAM PRESSURE TO DRIVE TURBINE-DRIVEN EMERGENCY FEEDWATER PUMPS OR RESTORE MAIN FEEDWATER MAY BE MORE LIKELY WITH THE OTSG DESIGN.

[2]FAULTS OF THIS KIND INTRINSICALLY QUALIFY AS ABNORMAL OCCURRENCES OR DISRUPTIVE EVENTS.

[3]THE DIRECT EFFECT ON THE FREQUENCY OF DOMINANT SEQUENCES IS NEGLIGIBLE, HOWEVER, THE PRONOUNCED EFFECT ON THE FREQUENCY OF CORE DAMAGE IN CONJUNCTION WITH COINCIDENTAL CONTAINMENT FAILURE MIGHT RIVAL DOMINANT SEQUENCES IN PROBABILITY.

TABLE 7.1 (CONT.)

[4]DELAYED START OF AUXILIARY FEEDWATER FOLLOWING LOSS OF MAIN FEEDWATER IS MORE LIKELY TO LIFT A PRESSURIZER VALVE IN B&W PLANTS. THIS INCREASES THE FREQUENCY OF TRANSIENT-INDUCED LOCA IN POSITIVE ASSOCIATION WITH FAULTS THAT MIGHT DEGRADE THE RELIABILITY OF HPI AS WELL AS AUXILIARY FEEDWATER. THE LESSONS OF TMI HAVE ALREADY REDUCED THIS LIKELIHOOD OF SERIOUS OUTCOMES FOR THESE SCENARIOS. TOTAL FAILURE OF ALL FEEDWATER AND OF HPI IS EQUALLY PROBLEMATIC IN ALL PWRS.

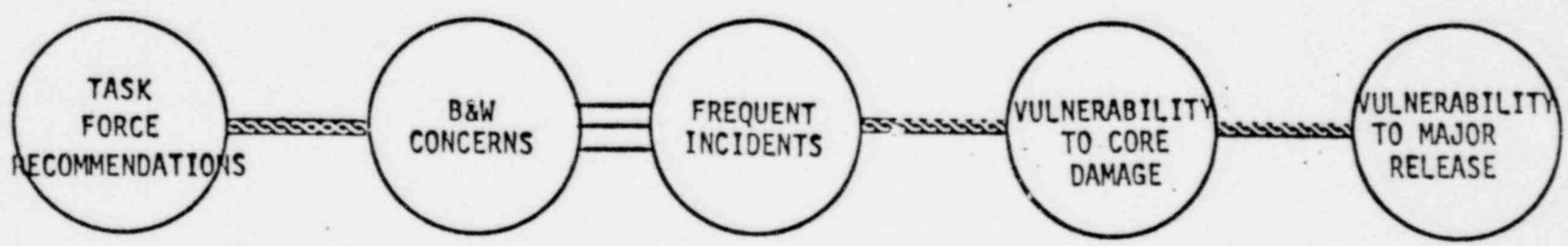
[5]FREQUENT TRIPS ARE INTRINSICALLY A CAUSE FOR CONCERN.

[6]EFFECT VIA OPERATOR ERROR OR TRANSIENT-INDUCED LOCA.

[7]EFFECT VIA LONG TERM INFLUENCE ON OPERATOR BEHAVIOR.

[8]NEITHER THE POSSIBILITY NOR THE LIKELIHOOD OF THIS HYPOTHETICAL GROUP OF ACCIDENTS HAS BEEN VERIFIED.

[9]FEED AND BLEED CAN PROVIDE AN OPTION FOR CORE COOLING IN THE EVENT OF A TOTAL LOSS OF FEEDWATER. IT MAY ALSO PROVIDE A LATER POINT OF NO RETURN FOR SAVING THE CORE DURING PRIMARY COOLANT BOILOFF.


TASK FORCE RECOMMENDATIONS
B&W CONCERNS
FREQUENT INCIDENTS
VULNERABILITY TO CORE DAMAGE
VULNERABILITY TO MAJOR RELEASE

III. RECOMMENDATIONS

A. FOCUS REGULATORY ATTENTION ON COMMON FAILURES IN ESF'S THAT AFFECT SUSCEPTIBILITY TO SEVERE ACCIDENTS

1. IREP
2. IMPLEMENTATION OF NUREG-0667
3. OTHER NRR, I&E ACTIVITIES

B. COMPLEMENT PERFORMANCE CRITERIA WITH RELIABILITY CRITERIA (DIVERSITY, REDUNDANCY, COMMON-CAUSE FAILURE ANALYSIS, ETC.)

C. EXPLORE FEASIBILITY AND EFFECACY OF RECOMMENDATIONS WITH B&W - IF NECESSARY ALTER CONTROL SYSTEM REQUIREMENTS

D. CONSIDER ADD-ON AFWS/HPSI DEDICATED SAFE SHUTDOWN SYSTEM

SEVERE ACCIDENT SCENARIOS

FOR

DRY CONTAINMENT PWRS

(CORE MELT AND EARLY, GROSS CONTAINMENT FAILURE)

1. MISSILES THAT BREACH CONTAINMENT, REACTOR COOLANT SYSTEM, AND FAIL ECCS, E.G., AIRCRAFT CRASH, REACTOR VESSEL LID.

2. STRUCTURAL COLLAPSE OF CONTAINMENT BUILDING LEADING TO FAILURE OF REACTOR COOLANT SYSTEM.

3. LOSS OF COOLANT ACCIDENTS WHICH BYPASS CONTAINMENT AND ARE NOT ISOLATED.

4. FAILURE OF CORE COOLING, CONTAINMENT SPRAYS AND FAN COOLERS

5. FAILURE OF CORE COOLING AND OPEN CONTAINMENT VENTS (BORDERLINE CASE - OPERABLE SPRAYS AND COOLERS MAY REDUCE RELEASES BELOW "SEVERE" THRESHHOLD).

Table 7.2

Effect of Task Force Recommendations on Particular Plant Transients

Task Force Recommendation	Loss of MFW		Loss of MFW From ICS Faults		Loss of Offsite Power		Small LOCA		Station Blackout		ATWS		OTSG Overfill	
	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg
1. AFWS Upgrade to Safety Grade														
a. Fluid System Upgrade	M-H		L		M-H		M-H		H		M-H			L
b. External Event Qualification	L		L		L		L		L		L		L	
2. Automatic Initiation and Control of AFWS	H		H		H		H		H		H			L
3. Diversely-Powered Auxiliary Feedwater Pump for Davis-Besse	H		H		M-H		M-H		L		H			L
4. Modifications to the Steam and Feedwater Line Break Detection and Mitigation Systems	H		H		H		M		?	?	H			
5. Improvements to the Integrated Control System														
a. Channelizing sensors, etc.	L	L	L		L	L	L	L	L	L	L	L	L	M
b. Meter fa'lure position	L	L	M		L	L	L	L	L	L	L	L	L	L
c. Annunciating failed bus	L		M		L		L		L		L		L	
d. Reversion to manual control	L	L	M		L	L	L	L	L	L	L	L	L	L
e. Loop indication separation	L	L	M		L	L	L	L	L	L	L	L	*M	M
f. Recommendations from ICS reliability study	L	L	M	L	L	L	L	L	L	L	L	L	M	L
g. Recommendations from INPO Crystal River report	L		M	L	L		L		L		L		L	
h. Follow-up to IE Bulletin 79-27	H		H	L	H		H		L		L		M	M
6. Installation of a Safety Grade Panel of Vital Instruments	H		H		H		H		H		H		H	

Table 7.2 (Cont.)

Task Force Recommendation	Loss of MFW		Loss of MFW From ICS Faults		Loss of Offsite Power		Small LOCA		Station Blackout		ATWS		OTSG Overfill	
	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg
7. Improved Use and Display of In-Core Thermocouple Indication	L		L		L		L		L		L		L	
8. Safety Grade Vent/Purge Isolation on High Radiation Signal	L		L		L		L		L		L			
9. System Response Modifications to Prevent Pressurizer Level Loss and ECCS Actuation	L		L	?	L		M		L		L		H	
10. Study of Means to Improve the Response of the OTSG	?		?		?		?		?		?		?	
11. Elimination of Post-Reactor Trip Operator Actions	L		M		M		M		L				H	M
12. Instrumentation and Control Technician Be Assigned to All Shifts	L	L	M	L	L	L	M	L	L	L	M	L	L	L
13. Operator Training on the Crystal River Incident 14. Development of Guidelines for Loss of ICS/NNI	M		H		M		H		L		L		* M	
15. Increased Simulator Training	M	L	M	L	M	L	M	L	M	L			M	L
16. Criteria for Restarting Reactor Coolant Pumps	M		M		M		M		M		?			

Table 7.2 (Cont.)

Task Force Recommendation	Loss of MFW		Loss of MFW From ICS Faults		Loss of Offsite Power		Small LOCA		Station Blackout		ATWS		OTSG Overfill	
	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg
17. Alternative Solution to PORV Unreliability and Safety System Challenge Rate Concerns	M		L		L		M	M	L		L	L	L	
18. Completion of IREP Crystal River Study	H?				H?		H?		H?					
19. Performance Criteria for Anticipated Transients	?		?		?		?		?		?		?	
20. Requirements for Reactor Coolant Pump Trip in Small LOCAs	L		L		L		L	L	L		M			
21. Reevaluation of AFWS Injection Point into the Steam Generators	L	L	L	L	L	L	L	L	L	L	L	L	L	
22. Study of Operator Errors in B&W ˜lants	L		L		L		L		L		L		L	

Table 7.3

Effect of Task Force Recommendations on Severe Accidents, Accidents, and Incidents

	Potential Benefit			Potential Detriment		
	SA	A	I	SA	A	I
1. Upgrade the AFWS Fluid System to Safety Grade						
a. Single Failure Criterion*	L	L	L	ε	ε	ε
b. Technical Specifications	M	M	M	ε	ε	ε
c. Pedigree (N-Stamp, QA)	ε	ε	ε	ε	ε	ε
d. Safety Grade Power Supplies*	L	L	L	ε	ε	ε
e. Diversity of Power Supplies	H	M	L	ε	ε	ε
f. Main Steam and Feedwater Line Break Criteria	ε	ε	ε	M	L	L
g. Seismic and External Event Qual.	L	ε	ε	ε	ε	ε
h. Other Alterations (see text)	H	H	L	ε	ε	L
*Most plants already comply; improvement might be large in those (if any) that do not.						
2. Safety Grade Initiation and Control of AFWS						
a. Safety Grade Control and Instrumentation Independent of ICS/NNI	M	H	H	ε	ε	L
b. Autostart to avoid dry steam generators	ε	M	M	ε	ε	M
c. Throttle AFWS to avoid overcooling of steam generators	ε	L	M	L	M	L
d. Feedwater termination to prevent overfill	ε	L	L	M?	H?	M?
3. Diversely-Powered Auxiliary Feedwater Pump for Davis-Besse	H	H	M	ε	ε	L
4. Modifications to the Steam and FW line Break Detection and Mitigation System	M	M	H	ε	ε	ε
5. Improvements to the ICS and NNI						
a. Channelized signals	ε	L	L	ε	ε	L
b. Evaluate mid-scale instrument failure mode	ε	L	L	ε	ε	L
c. Indicate multiple failures	ε	L	L	ε	ε	ε
d. Reversion to manual control	ε	ε	ε	ε	L	M
e. Loop indication separation	ε	L	L	ε	ε	L
f. Recommendations from ICS reliability study	ε	L	M	ε	ε	ε
g. Recommendations from INPO Crystal River report	ε	L	L	ε	ε	ε
h. Follow-up to IE Bulletin 79-27	M	H	L	ε	ε	ε

Table 7.3 (Cont.)

	Potential Benefit			Potential Detriment		
	SA	A	I	SA	A	I
6. Installation of a Safety Grade Panel of Vital Instruments	M	H	H	ε	ε	ε
7. Improved Use and Display of In-Core Thermocouple Indication	ε	L	L	ε	ε	ε
8. Safety Grade Vent/Purge Isolation on High Radiation Signal	ε	L	M	ε	ε	ε
9. System Response Modifications to Prevent Pressurizer Level Loss and ECCS Actuation	ε	L	M	ε	ε	ε
10. Study of Means to Improve the Response of the OTSG	?	?	?	?	?	?
11. Elimination of Post-Reactor Trip Operator Actions	ε	L	L	ε	L?	L?
12. Instrumentation and Control Technicians Be Assigned to All Shifts	L	M	M	ε	L	L
13. Operator Training on the Crystal River Incident 14. Development of Plant-Specific Procedures on Loss of ICS/NNI	M	H	H	ε	ε	ε
15. Increased Simulator Training	ε	M	M	ε	L	L
16. Criteria for Restarting Reactor Coolant Pumps	L	M	M	ε	ε	ε
17. Alternative Solution to PORV Unreliability and Safety System Challenge Rate Concerns	ε	L	M	ε	L	L
18. Completion of the IREP Crystal River Study	?	?	?	?	?	?
19. Performance Criteria for Anticipated Transients	?	?	?	?	?	
20. Criteria for Reactor Coolant Pump Trip in Small LOCAs	ε	M	M	ε	L	ε

Table 7.3 (Cont.)

	Potential Benefit			Potential Detriment		
	SA	A	I	SA	A	I
21. Reevaluation of AFWS Injection Point into the Steam Generators	ε	ε	L	ε	L	L
22. Study of Operator Errors in B&W Plants	ε	ε	ε	ε	ε	ε

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

April 29, 1980

MEMORANDUM FOR: R. Fraley, ACRS

FROM: R. Tedesco, Chairman
B&W Reactor Trainsient Response Task Force

SUBJECT: TRANSMITTAL OF DRAFT SECTION 7 TO NUREG-0667

Attached is a DRAFT copy of Section 7 to NUREG-0667 (Transient Response of Babcock & Wilcox-Designed Reactors). On April 2, 1980 DRAFT NUREG-0667 was issued pending completion of Section 7. Section 7 was developed by the Probabilistic Analysis Staff to evaluate the effectiveness of the Task Force recommendations using perspectives derived from probabilistic safety analysis and risk assessment.

The information contained in Section 7 will be discussed this afternoon with the ACRS B&W Water Reactor Subcommittee. Sections 7 is presently be being incorporated into the main report of NUREG-0667. The Task Force final report is expected to be issued in final form on April 30, 1980.

Copies of subject document will be distributed to the ACRS Subcommittee at this afternoon's meeting.

R Tedesco

R. Tedesco, Chairman
B&W Reactor Trainsient Response
Task Force

cc: H. Denton
E. Case
NRR Division Directors

7. RISK REDUCTION POTENTIAL

7.1 Introduction

The Probabilistic Analysis Staff was asked to evaluate the effectiveness of the Task Force recommendations using perspectives derived from probabilistic safety analysis and risk assessment. This chapter reports this review.

It is not possible to obtain a quantitative measure of risk reduction effectiveness for the recommendations. To do so would have required a thorough knowledge of the likelihood and consequences of the many competing accident scenarios in the plants before the alterations and a thorough knowledge of the implementation and effects of the recommendations. This is clearly far beyond the known at this time.

On the other hand, many qualitative insights that shed some light on the potential value of the recommendations can be developed against the background of past attempts at realistic analyses of the likelihood and consequences of nuclear accidents using probabilistic risk assessment methods. These include relationships between B&W plant characteristics and the likelihood of accidents, and judgments of the range of benefits and disadvantages of the recommendations. In many cases the recommendations suggest studies and directions in which to look for improvements rather than prescriptive fixes. The risk-based perspectives add another dimension to the definition of these suggestions. The observations about B&W safety issues and about the recommendations reported here originated in the professional judgment of experienced nuclear risk assessment engineers. They are not based on probabilistic safety analyses performed for this specific purpose.

The technique employed to arrive at these observations was to develop several tables (7.1, 7.2, and 7.3). The entries in the tables were arrived at by consensus. The assumptions, observations, and arguments that surfaced in the course of arriving at this consensus became the source for the footnotes and text.

In Section 7.2 the broad outlines of the risk picture are sketched for Babcock & Wilcox reactors. The study addresses B&W plants as they are being operated since TMI but before the recommendations contained herein are implemented. We find that the characteristics of the B&W nuclear steam supply system design and operation makes these plants much more prone to minor incidents, somewhat more prone to core damage, and no more prone to severe accidents than are other PWR designs.

In Section 7.3, the twenty-two recommendations discussed in Section 2.0 of this report are evaluated for their range of effects on the frequency of a number of particular accident scenarios and for their influence on the likelihood of incidents, minor accidents, and severe accidents.

It should be clearly understood that these observations reflect the opinions of risk assessment engineers and not the results of detailed calculations or a formal research program. As such, they should be regarded as uncertain.

7.2 Risk Perspectives for B&W Plants

A number of studies have been performed or are under way which address the realistic consequences of core melt accidents at pressurized water reactors having large dry containment buildings. These studies include the Reactor Safety Study, WASH-1400, the alternate sequence and consequence analyses done in conjunction with the Kemeny and Rogovin inquiries into the accident at TMI, and some studies currently in progress on Indian Point, Zion, Calvert Cliffs and Oconee.

These studies suggest that there is a "natural" classification for accidents in dry containment PWRs. In this scheme, lines of demarcation in accident consequences correspond with lines of demarcation in terms of the functional failure of systems. There are three levels of severity in this classification. We might call them:

1. Severe Accidents,
2. Accidents, and
3. Incidents.

The basis for the distinctions are as follows: all accidents that produce any acute fatalities beyond the site boundary are predicted to entail both severe core damage or meltdown <u>and</u> gross, early containment failure. Accidents of this kind are also the only ones to produce substantial ground contamination by fallout. Such accidents dominate the risk as measured by public health and safety criteria and by offsite property damage.

Accidents - the intermediate class of incidents - may entail core damage or meltdown but do not entail gross, early containment failure. The accident at the Three Mile Island is an example. Also belonging in this class of incidents are design basis LOCA events with gross containment failure. Such accidents do not cause acute fatalities. They will not cause fallout that severely contaminates offsite land. They may - in their more serious variants - cause latent cancer casualties or groundwater contamination warranting interdiction. Accidents like these are not irrelevant to public health and safety, but they are very much less severe than the ones we have called "Severe Accidents." Unless these accidents were to be - and were to remain for a long time - very much more probable than severe accidents, they would be overshadowed in public health risk significance by the severe accidents. These accidents are, however, the dominant contributor to the economic risk borne by the plant owners relating to on-site equipment damage, as the accident at Three Mile Island indicates.

Incidents have virtually no offsite radiological consequences associated with them. Their contribution to public risk - as measured by health effects or offsite property damage - is negligible. The economic risk for the utility and its rate-payers associated with incidents tend to be smaller than or comparable to that associated with accidents. They include anticipated transients, events like the Browns Ferry fire, design basis LOCAs, etc. They do not entail significant core damage nor do they include LOCA in conjunction with abnormal post-accident containment leakage.

Accidents fall into the "severe" category only if the containment fails and the core releases much of its radioactivity. The causes of such accidents may be described as follows:

1. External missiles (e.g., heavy airplane crash) or internal missiles (e.g., the reactor vessel head) that breach the reactor coolant system, disable emergency core cooling systems and breach containment.

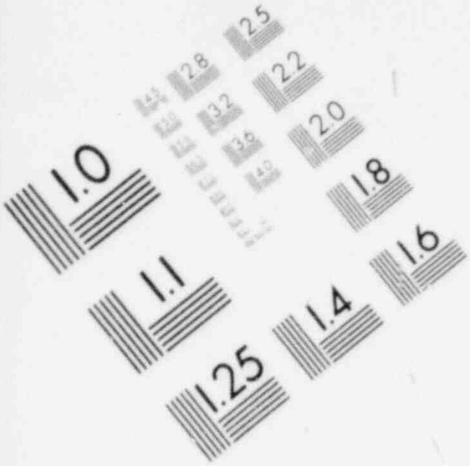
2. Structural collapse of the containment building which defeats the core cooling systems.

3. Loss of coolant accidents that are not isolatable and which bypass containment. (Event V in WASH-1400)

4. Failure of core cooling, failure of containment sprays, and failure of containment fan coolers.

5. A border line case is failure of core cooling and failure of containment isolation with operable containment sprays and coolers. Such scenarios may fall in either the "severe accident" or "accident" spectra of consequences.

Accident scenarios of the first two kinds (missiles and structural collapse) have been extensively analyzed in nuclear power plants. They are believed to be extremely improbable. Probabilistic risk assessment suggests that the third kind of scenario, the interfacing system LOCA that blows down outside containment, may be among the dominant contributors to the risk from any PWR. The susceptibility of a plant depends upon the design, administrative controls, and surveillance of the reactor coolant pressure boundary valves on the larger lines that attach to the reactor coolant system and penetrate containment. It does not importantly depend upon the particular reactor design.



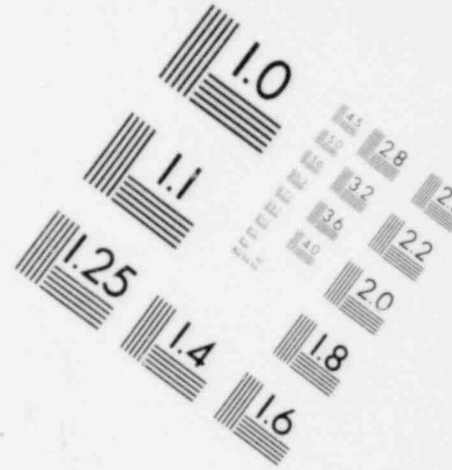
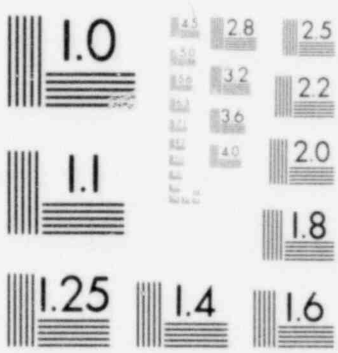
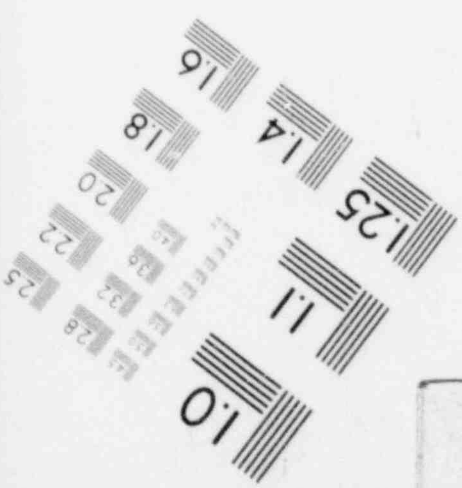


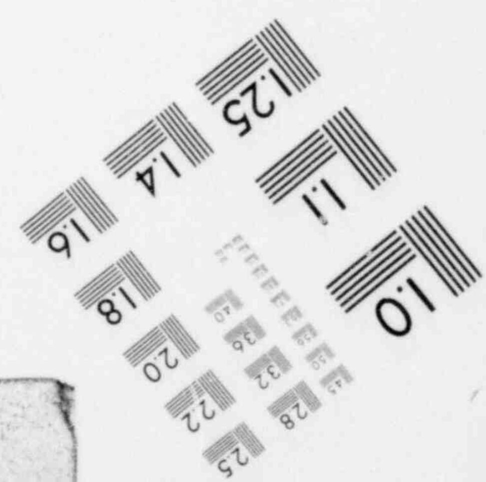
IMAGE EVALUATION TEST TARGET (MT-3)



6″

MICROCOPY RESOLUTION TEST CHART





Risk assessment studies suggest that the fourth group of severe accident scenarios may also contain dominant contributors to the risk. These are accidents entailing failure of core cooling (leading to severe damage or melt) and also failure of containment fan coolers and sprays (leading to gross containment rupture on overpressure). Many failures in the "front line" engineered safety features are required for this to happen. For example, failure of all trains of containment fan coolers, failure of all trains of containment sprays, failure of the safety injection function and either a LOCA or a failure of main and auxiliary feedwater. The coincidental or random failure of all trains of all these "front line" engineered safety features is clearly much too unlikely to affect the risk. However, common cause failures such as fires, floods, earthquakes, or the failure of support or auxiliary systems, such as AC power, DC power, control and actuation systems, auxiliary cooling water systems, etc. can produce the many functional faults in "front line" systems from one or a very few root-cause failure events.

One example of this group of accident scenarios was found to be a dominant contributor to the risk in the Reactor Safety Study PWR. It entails loss of offsite power, the failure of both emergency diesel generators, and the failure of the turbine-driven auxiliary feedwater pump. All feedwater is lost, leading to the boil-dry of first the steam generators and then the reactor core. Containment sprays and coolers are also defeated by the failure of AC power sources, so this scenario belongs in the group of severe accidents.

The likelihood of these severe accident scenarios is governed by the susceptibility of the front line engineered safety features to common failure mechanisms, not to the details of the design of the nuclear steam supply systems. Therefore, there is little reason to believe that B&W plants are any more or less likely to be subject to such accidents than are other PWRs.

It is well known that the once through steam generators employed in B&W plants hold a small inventory of secondary coolant. They boil dry more quickly than other PWR designs following a loss of all feedwater flow. Dry steam generators implies an interruption in normal reactor heat dissipation but it does not mark a point of no return for core cooling. Later restoration of feedwater may restore normal cooling for some time after steam generator dryout. Most B&W plants also have HPI pumps with high shutoff head; these pumps can drive open the pressurizer safety valves. This capability is very useful in extending the time-window within which core damage or meltdown can be avoided following an interruption in primary and secondary side cooling. Thus, most B&W plants may have as long or longer points of no return for the restoration of successful core cooling than do some other PWR designs.

Undercooling transients are more likely in plants with highly responsive OTSGs than in otherwise comparable plants with recirculating steam generators. Brief interruptions in the heat sink provided by the steam generators may cause a challenge to one of the pressurizer valves (PORV or safety valves). Thus, the B&W design tends to be more susceptible to transient-induced LOCA. The difference between B&W and other designs is confined to the case of <u>delayed</u> auxiliary feedwater starts. Prompt AFWS starts do not cause undercooling transients. Outright (sustained) failure to start is equally serious with or without responsive steam generators. Thus, B&W plants place a premium upon the reliability with which the auxiliary feedwater starts are properly timed. The penalty for late starts is an increased likelihood of transient-induced LOCA.

The most prominant common-cause failure mechanism we can identify that causes both delayed auxiliary feedwater starts and sustained ECCS failures lies in operator error. A practice of trying to avoid over-cooling incidents tends to make such errors more likely. On the other hand, the experience of having had a TMI accident, the operator retraining it spawned, and the other changes made since the accident have gone a long way to reduce the likelihood that such scenarios would start or would progress to core damage once started. Nevertheless, our event tree-fault tree studies suggest that transient induced LOCA which cannot be isolated and which occurs in conjunction with ECCS failure may be among the dominant routes to core damage, i.e., to an accident, although we think it very unlikely that such a scenario would also entail the failure of containment fan coolers as well as sprays. Thus, transient-induced LOCAs should not be prominant causes of severe accidents.

It is known that B&W plants have somewhat more frequent trips than do other PWRs, particularly since the TMI-inspired alterations to the trip setpoints. These excess trips seem to be originating from minor secondary side transients and non-safety-grade instrumentation faults. These transient initiators do not correlate with the occurrence of massive, common-cause failures in the engineered safety features - with a couple of noteworthey exceptions - so they are not expected to increase the frequency of the risk-dominant severe accidents in B&W plants above the level expected for other PWR designs.

The two exceptions deserve closer scrutiny. The Non-Nuclear Instrument (NNI) bus faults that occurred at Rancho Seco and Crystal River caused massive faulting of the instruments upon which the operators depended to understand the status of the plant. It could be postulated that such faults could lead to the kinds of operator errors that could give rise to severe accidents. For a number of reasons, severe accidents via such routes seem very unlikely: (1) In the post-TMI environment, it is unlikely that operators would override the autostart of engineered safety features while their instruments are obviously faulted; (2) It is unlikely that operators would shut off containment fan coolers, even under circumstances in which they might mistakenly shut off ECCS or containment sprays; (3) All historical instances of NNI failures have been repaired before the point of no return for a severe accident; and (4) The attention given to the recent Crystal River and other incidents has alerted operators to the symptoms, consequences, and the ways to deal with NNI failures.

Another hypothetical way that the somewhat higher transient rate at B&W plants might affect the frequency of high-risk accident sequences is through failures of offsite power. Loss of offsite power may originate outside the plant or be precipitated by a plant trip. Studies performed for WASH-1400 suggested that most instances of loss of offsite power originate outside; the overall frequency of the loss is quite insensitive to the plant trip rate according to industry statistics. There may be exceptional sites where this is not true, however. To the extent that B&W plants trip more often than other PWRs, they place a correspondingly greater safety premium upon the reliability with which the grid, the switchgear and the startup transformer picks up plant auxiliary loads. We expect for most B&W plants that the somewhat higher trip rate has a negligible effect on the likelihood of severe station blackout accidents.

In summary, then, the enhanced frequency of transients in B&W plants is not believed to importantly affect the likelihood of severe accidents.

Another concern with B&W plant design and operation is the comparatively high frequency of overcooling transients following reactor trip. In some of these transients the shrinkage of reactor coolant causes the pressurizer level to go off-scale low and/or the pressure to fall to the ECCS actuation point. Even if the pressurizer bubble is drawn into a reactor coolant loop and the reactor coolant pumps are tripped, we see no difficulty in sustaining convective circulation in the unaffected loop and sustaining or restoring it in the loop with some of the steam bubble. Frequent ECCS actuation in such events is significant in the ways it affects operator behavior. Frequent spurious ECCS actuations could tend to induce operators to disable or override actuation signals important to safety.

In the post-TMI environment, we think that operators would correct such errors long before they resulted in core damage in all but the fastest-moving accidents and would correct such errors before containment failure results in a severe release in virtually every case. Thus, the "cry wolf" effect of overcooling transient-induced spurious ECCS actuations might have some effect on the frequency of core damage (accidents) but a negligible effect on the frequency of major releases, i.e., severe accidents.

ECCS actuations in overcooling transients - apart from their effect on operator behavior - are expected to have very little effect on the likelihood of core damage. If ECCS fails to start, no harm is done as it isn't really needed in an overcooling accident. There is a very slight chance that HPI or the affected makeup pump might be critically needed before it could be repaired. On the other hand, such challenges provide experiences more closely resembling genuine demands than do surveillance tests, so these nuissance demands also help to debug the system. On balance the prospect of ECCS failures in these overcooling transients has very slight and counterbalancing effects on the likelihood of core damage and a negligible effect on severe accidents.

If ECCS does start in these overcooling transients, the operators may leave it on long enough to lift the pressurizer PORV or possibly a safety valve. This, in turn, opens the possibility of a spillage of reactor coolant and perhaps a stuck-open valve, i.e., a LOCA. In the worst case of a stuck-open, non-isolatable pressurizer valve, ECCS must work to sustain core cooling. However, ECCS will have higher-than-normal reliability under these conditions because its successful start caused the LOCA in the first place. There is no reason to believe that such incidents are likely to be coupled with ECCS failure or with the failure of containment fan coolers or sprays.

It has been suggested that a reactor trip together with a failure to throttle main feedwater in a B&W plant would rapidly fill the OTSG's and result in water in the main steam lines. No such instances have occurred but comparable upsets in the Integrated Control System have been observed. The main steam lines and valves may not be qualified for the weight or the water-hammer potential associated with this scenario; they might rupture. The characteristic range of times to fill the steam generators and main steam lines is a very few minutes, perhaps too rapid to give much confidence that the operators would consistently trip the feedwater pumps or stop valves in time to avoid main steam line breaks.

Such scenarios would affect the risk of severe accidents only if the break produced flooding that defeats support systems for essentially all of the active engineered safety features, i.e., essential DC power, AC power, or possibly essential auxiliary cooling water systems, and do so with a probability that rivals station blackout or Event V. Such scenarios would have a significant effect on the likelihood of core damage only if the flooding defeats emergency feedwater and HPI (feed and bleed cooling) and does so with a probability that rivals other common-cause or multi-fault scenarios such as loss of all feedwater and HPI failure.

In either the case of accidents or severe accidents, the significance of the water-solid main steam line break scenarios seems to rest upon the potential for massive flood damage in essential compartments of the auxiliary building. If such flooding does not take place, there appears to be little direct threat to ultimate core cooling or containment integrity.

The susceptibility of B&W plants to loss of all essential AC or DC power or loss of all HPI and EFW due to water-solid main steam line breaks and subsequent flooding should be reviewed. If a deterministic analysis suggests a real possibility of such a scenario, then a probabilistic evaluation should be performed.

These considerations of B&W plant characteristics are summarized in Table 7.1. We conclude that B&W plants are not significantly different from other PWRs in their vulnerability or susceptibility to severe accidents - those that dominate the nuclear risk.

B&W plants have a different profile of susceptibility to core damage accidents than do other PWRs. They are more likely to incur transient-induced LOCA but the ones with high head HPI pumps may be less likely to incur core damage from a loss of all feedwater. B&W plants are more likely than other PWRs to have over- or undercooling incidents, transient-induced LOCA, etc.

Table 7.1

Effect on Frequency of Incidents of B&W Plant Characteristics or Concerns

B&W Plant Characteristic or Concern	Effect on Frequency* of: Severe Accidents (large release)	Accidents (small release)	Incidents (no abnormal release)
1. Short time to SG dryout following loss of feedwater	small[1]	small[1]	large[2]
2. Frequent undercooling transients	small[3]	large[4]	large[4]
3. Heightened trip frequency	negligible (neg)	small	large[5]
4. NNI/ICS faults	neg	medium[6]	large[2]
5. Frequent overcooling transients			
a. Loss of PRZR level	neg	neg	large[2]
b. Nuissance ECCS actuation	neg	medium[7]	large[2]
6. Overfeed main steam line rupture	neg?[8]	neg?[8]	?
7. Feed and bleed capability (high head HPI)	moderate improvement[9]	large improvement[9]	large

Notes:

*Baseline of comparison is the WASH-1400 risk picture for Surry.

[1]Loss of steam pressure to drive turbine-driven emergency feedwater pumps or restore main feedwater may be more likely with the OTSG design.

[2]Faults of this kind intrinsically qualify as abnormal occurrences or disruptive events.

[3]The direct effect on the frequency of dominant sequences is negligible, however, the pronounced effect on the frequency of core damage in conjunction with coincidental containment failure might rival dominant sequences in probability.

Table 7.1 (Cont.)

[4]Delayed start of auxiliary feedwater following loss of main feedwater is more likely to lift a pressurizer valve in B&W plants. This increases the frequency of transient-induced LOCA in positive association with faults that might degrade the reliability of HPI as well as auxiliary feedwater. The Lessons of TMI have already reduced this likelihood of serious outcomes for these scenarios. Total failure of all feedwater and of HPI is equally problematic in all PWRs.

[5]Frequent trips are intrinsically a cause for concern.

[6]Effect via operator error or transient-induced LOCA.

[7]Effect via long term influence on operator behavior.

[8]Neither the possibility nor the likelihood of this hypothetical group of accidents has been verified.

[9]Feed and bleed can provide an option for core cooling in the event of a total loss of feedwater. It may also provide a later point of no return for saving the core during primary coolant boiloff.

7.3 Observations on the Task Force Recommendations

Table 7.2 reports the judgment of the review group from the Probabilistic Analysis Staff of the effect of the Task Force recommendations on the likelihood or severity of a number of accident scenarios: loss of main feedwater, loss of main feedwater due to ICS or NNI faults, loss of offsite power, small LOCA, station blackout, anticipated transient without scram, and steam generator overfill.

Table 7.3 is very much like Table 7.2 except that the columns treat incidents by the severity of outcome rather than by the kind of initiating event. In this Table, we have assessed the potential of each recommendation for reducing the likelihood and/or severity of the three categories of events (incidents, accidents, and severe accidents). That is, each entry in the table may be interpreted as the potential for the specific recommendation reducing (or increasing) the likelihood of the particular event category and/or improving (or harming the plant's capability to cope with the events in that category. Thus, some recommendations may be of high potential benefit in reducing the likelihood of a severe accident but of low potential benefit in coping with an ICS/NNI fault like that experienced at Crystal River. Others may be of some moderate benefit in reducing the frequency of overcooling incidents, of moderate benefit in reducing the likelihood that such an incident will propagate into an event causing core damage (the "accident" category), but of negligible benefit in reducing the likelihood of severe accidents.

Table 7.2

Effect of Task Force Recommendations on Particular Plant Transients

Task Force Recommendation	Loss of MFW		Loss of MFW From ICS Faults		Loss of Offsite Power		Small LOCA		Station Blackout		ATWS		OTSG Overfill	
	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg
1. AFWS Upgrade to Safety Grade														
a. Fluid System Upgrade	M-H		L		M-H		M-H		H		M-H			L
b. External Event Qualification	L		L		L		L		L		L		L	
2. Automatic Initiation and Control of AFWS	H		H		H		H		H		H			L
3. Diversely-Powered Auxiliary Feedwater Pump for Davis-Besse	H		H		M-H		M-H		L		H			L
4. Modifications to the Steam and Feedwater Line Break Detection and Mitigation Systems	H		H		H		M		?	?	H			
5. Improvements to the Integrated Control System														
a. Channelizing sensors, etc.	L	L	L		L	L	L	L	L	L	L	L	L	M
b. Meter failure position	L	L	M		L	L	L	L	L	L	L	L	L	L
c. Annunciating failed bus	L		M		L		L		L		L		L	
d. Reversion to manual control	L	L	M		L	L	L	L	L	L	L	L	L	L
e. Loop indication separation	L	L	M		L	L	L	L	L	L	L	L	M	M
f. Recommendations from ICS reliability study	L	L	M	L	L	L	L	L	L	L	L	L	M	L
g. Recommendations from INPO Crystal River report	L		M	L	L		L		L		L		L	
h. Follow-up to IE Bulletin 79-27	H		H	L	H		H		L		L		M	M
6. Installation of a Safety Grade Panel of Vital Instruments	H		H		H		H		H		H		H	

Table 7.2 (Cont.)

Task Force Recommendation	I of .W		Loss of MFW From ICS Faults		Loss of Offsite Power		Small LOCA		Station Blackout		ATWS		OTSG Overfill	
	Pos	Neg	Pc-	Neg	Pos	Neg	Pos	Nen	Pos	Neg	Pos	Neg	Pos	Neg
7. Improved Use and Display of In-Core Thermocouple Indication	L		L		L		L		L		L		L	
8. Safety Grade Vent/Purge Isolation on High Radiation Signal	L		L		L		L		L		L			
9. System Response Modifications to Prevent Pressurizer Level Loss and ECCS Actuation	L		L	?	L		M		L		L		M	
10. Study of Means to Improve the Response of the OTSG	?		?		?		?		?		?		?	
11. Elimination of Post-Reactor Trip Operator Actions	L		M		M		M		L				M	M
12. Instrumentation and Control Technician Be Assigned to All Shifts	L	L	M	L	L	L	M	L	L	L	M	L	L	L
13. Operator Training on the Crystal River Incident 14. Development of Guidelines for Loss of ICS/NNI	M		M		M		M		L		L		M	
15. Increased Simulator Training	M	L	M	L	M	L	M	L	M	L			M	L
16. Criteria for Restarting Reactor Coolant Pumps	M		M		M		M		M		?			

Table 7.2 (Cont.)

Task Force Recommendation	Loss of MFW		Loss of MFW From ICS Faults		Loss of Offsite Power		Small LOCA		Station Blackout		ATWS		OTSG Overfill	
	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg
17. Alternative Solution to PORV Unreliability and Safety System Challenge Rate Concerns	M		L		L		M	M	L		L	L	L	
18. Completion of IREP Crystal River Study	H?				H?		H?		H?					
19. Performance Criteria for Anticipated Transients	?		?		?		?		?		?		?	
20. Requirements for Reactor Coolant Pump Trip in Small LOCAs	L		L		L		L	L	L		M			
21. Reevaluation of AFWS Injection Point into the Steam Generators	L	L	L	L	L	L	L	L	L	L	L	L	L	
22. Study of Operator Errors in B&W Plants	L		L		L		L		L		L		L	

Table 7.3

Effect of Task Force Recommendations on Severe Accidents, Accidents, and Incidents

	Potential Benefit			Potential Detriment		
	SA	A	I	SA	A	I
1. Upgrade the AFWS Fluid System to Safety Grade						
a. Single Failure Criterion*	L	L	L	ε	ε	ε
b. Technical Specifications	M	M	M	ε	ε	ε
c. Pedigree (N-Stamp, QA)	ε	ε	ε	ε	ε	ε
d. Safety Grade Power Supplies*	L	L	L	ε	ε	ε
e. Diversity of Power Supplies	H	M	L	ε	ε	ε
f. Main Steam and Feedwater Line Break Criteria	ε	ε	ε	M	L	L
g. Seismic and External Event Qual.	L	ε	ε	ε	ε	ε
h. Other Alterations (see text)	H	H	L	ε	ε	L
*Most plants already comply; improvement might be large in those (if any) that do not.						
2. Safety Grade Initiation and Control of AFWS						
a. Safety Grade Control and Instrumentation Independent of ICS/NNI	M	H	H	ε	ε	L
b. Autc tart to avoid dry steam generators	ε	M	M	ε	ε	M
c. Throttle AFWS to avoid overcooling of steam generators	ε	L	M	L	M	L
d. Feedwater termination to prevent overfill	ε	L	L	M?	H?	M?
3. Diversely-Powered Auxiliary Feedwater Pump for Davis-Besse	H	H	M	ε	ε	L
4. Modifications to the Steam and FW line Break Detection and Mitigation System	M	M	H	ε	ε	ε
5. Improvements to the ICS and NNI						
a. Channelized signals	ε	L	L	ε	ε	L
b. Evaluate mid-scale instrument failure mode	ε	L	L	ε	ε	L
c. Indicate multiple failures	ε	L	L	ε	ε	ε
d. Reversion to manual control	ε	ε	ε	ε	L	M
e. Loop indication separation	ε	L	L	ε	ε	L
f. Recommendations from ICS reliability study	ε	L	M	ε	ε	ε
g. Recommendations from INPO Crystal River report	ε	L	L	ε	ε	ε
h. Follow-up to IE Bulletin 79-27	M	H	L	ε	ε	ε

Table 7.3 (Cont.)

	Potential Benefit			Potential Detriment		
	SA	A	I	SA	A	I
6. Installation of a Safety Grade Panel of Vital Instruments	M	H	H	ε	ε	ε
7. Improved Use and Display of In-Core Thermocouple Indication	ε	L	L	ε	ε	ε
8. Safety Grade Vent/Purge Isolation on High Radiation Signal	ε	L	M	ε	ε	ε
9. System Response Modifications to Prevent Pressurizer Level Loss and ECCS Actuation	ε	L	M	ε	ε	ε
10. Study of Means to Improve the Response of the OTSG	?	?	?	?	?	?
11. Elimination of Post-Reactor Trip Operator Actions	ε	L	L	ε	L?	L?
12. Instrumentation and Control Technicians Be Assigned to All Shifts	L	M	M	ε	L	L
13. Operator Training on the Crystal River Incident 14. Development of Plant-Specific Procedures on Loss of ICS/NNI	M	H	H	ε	ε	ε
15. Increased Simulator Training	ε	M	M	ε	L	L
16. Criteria for Restarting Reactor Coolant Pumps	L	M	M	ε	ε	ε
17. Alternative Solution to PORV Unreliability and Safety System Challenge Rate Concerns	ε	L	M	ε	L	L
18. Completion of the IREP Crystal River Study	?	?	?	?	?	?
19. Performance Criteria for Anticipated Transients	?	?	?	?	?	
20. Criteria for Reactor Coolant Pump Trip in Small LOCAs	ε	M	M	ε	L	ε

Table 7.3 (Cont.)

	Potential Benefit			Potential Detriment		
	SA	A	I	SA	A	I
21. Reevaluation of AFWS Injection Point into the Steam Generators	ε	ε	L	ε	L	L
22. Study of Operator Errors in B&W Plants	ε	ε	ε	ε	ε	ε

For each accident grouping, there are two columns in Table 7.2 or 7.3 labeled "Pos" and "Neg." "Positive" denotes the benefit to be expected from the sound implementation of the recommendation. "Negative" denotes the potential for increased competing risks that might arise from the recommendation. For example, an alteration to the Integrated Control System could make one failure mode less likely and other failure modes more likely. We record both effects, the improvement under "Pos," the degradation under "Neg." The comments and interpretations underlying these judgments are summarized in the text below.

The entries in the tables are interpreted as follows:

1. H - High

   The recommendation is judged to have a substantial effect on a dominant contributor to the likelihood of accidents in the group of accidents.

2. M - Medium

   The recommendation is judged to have a moderate effect on a dominant contributor or a major effect on contributors that are only moderately likely to have a significant influence on the overall frequency of accidents of the type under consideration.

3. L - Low

   The overall effect on the likelihood of accidents is judged to be low. That is, the recommendation may have little effect, or it may have a strong effect on factors not bearing directly on the dominant contributors to the class of accidents under consideration.

4. Blank or Epsilon (ε)

Negligible effect.

A discussion of each recommendation follows.

1. <u>Upgrade the Auxiliary Feedwater System (AFWS) Fluid System to Safety Grade</u>

In this recommendation, the Task Force calls for the improvement of the "fluid-moving" aspects of the AFWS to "safety grade." The actuation and control aspects are treated in recommendation 2. Safety grade qualification entails several facits:

a. Single failure criterion

We believe almost all B&W plants have an AFWS already meeting the single failure criterion for its mechanical aspects. Thus, we think the effect of this recommendation is small. Nonetheless, its imposition is desirable, because a violation of the single failure criterion could severely compromise the reliability of the AFWS.

b. Pedigree requirements

Safety qualification normally entails a number of quality assurance and code requirements. As applied to pipes, pumps and valves, these criteria tend to bear upon pressure boundary integrity rather than active failure reliability. Since pipe breaks are a negligible contributor to the functional unavailability of the AFWS, there is very little benefit to be gained from a retroactive requirement to upgrade the pedigree of piping, valves and pumps (presuming that the equipment now installed is already of good quality).

c. Class IE power supplies for motor-operated pumps and valves We believe most plants already comply so that the effect of the recommendation will be small. Nonetheless, this recommendation is important as an instance of non-compliance could compromise system reliability.

d. Seismic Category I qualification Seismically-induced loss of main feedwater is sufficiently probable to warrant a requirement to provide an engineered safety feature qualified to cool the core under this circumstance. However, it is not so common an initiating event that diverse as well as redundant means are needed. We recommend that license applicants be given the option of selecting either primary system cooling (feed and bleed) or secondary system cooling (emergency feedwater) as the designated, qualified method of cooling the core following a seismically induced loss of main feedwater.

e. Technical specifications Safety qualification implies the imposition of technical specifications and finite allowable outage times for periods during which portions of the AFWS are out of service. These can have a moderate to large effect on AFWS reliability and thus on risk.

f. Main steam and feedwater line break design bases

Main steam and feedwater line breaks have been taken as design basis challenges for the AFWS in some but not all operating PWRs. AFW must be isolated from the affected steam generator and yet AFW must be supplied to the surviving steam generator(s) despite a single active failure.

Such accidents pose very little risk. They are rare and they do not directly threaten core cooling. We see virtually no risk reduction potential in extending these requirements to all PWRs, and the requirements might safely be relaxed where the provisions for automatic isolation of the "affected" steam generator or the valving necessary to satisfy the single failure criterion is found to degrade AFWS functional reliability for the very much more common loss of feedwater events.

g. Diversity of power supplies

Branch Technical Position ASB 10-1 currently requires diverse power supplies for AFWS pumps. The concept of designing out the susceptibility of the AFWS to failure in the event of a common cause failure of all sources of motive power, such as all AC power or all steam, can have a very large risk reduction potential. However, the requirement needs strengthening to include not just pump power supplies but valve and support systems as well. There should be at least one train of the

AFWS that is capable of starting and running for each of the following circumstances:

1. Loss of power on all essential and non-essential switchgear buses.
2. Loss of steam pressure in both steam generators.
3. At least one train should fail on rather than off if the corresponding control power supplies (DC or AC instrument power) were to fail off.

h. Other requirements

Most B&W plants have a two train AFWS. There is a limit to the reliability improvement that can be achieved without adding a third train. Loss of main feedwater is a very frequent challenge. With two train AFWS designs - even ones of comparatively high reliability - loss of all feedwater is a rare but distinctly credible event. We judge that a return interval of once in a thousand reactor years is about the best one might confidently expect for loss of all feedwater in PWRs having two train AFWS designs. A case can be made for the provision of an add-on, third train of the auxiliary feedwater system which does not depend upon the same support and auxiliaries as does the principal two-train system. However, the case for such an add-on may be less compelling in B&W plants with a demonstrated feed and bleed cooling capability than it is in plants with comparatively low head HPI since they have alternate means of core cooling.

2. <u>Safety Grade Initiation and Control of the AFWS</u>

This recommendation is primarily concerned with the need for a safety grade system for initiation and control of the AFW system independent of the ICS/NNI. Also included within the recommendation are: a call for an appropriate selection of initiating signals such that the undercooling and overcooling transients experienced during the transition from main to auxiliary feedwater are minimized in severity; an inclusion within the steam generator level control of an overcooling protection capability; and a feedwater termination signal to prevent overfilling of the steam generators.

The most important part of this group of recommendations deals with the provision of an AFWS autostart system that is capable of responding in the event of a loss of main feedwater and which is independent of the ICS or its power supplies. The key to a large improvement in safety is to assure that the kind of failure events that may cause a loss of main feedwater will not also disable the AFWS.

Apart from this elimination of common cause failure susceptibility - which has large risk reduction potential - the redundancy and IE qualification requirements associated with safety grade actuation is expected to produce a small improvement in system reliability.

The selection of autostart actuation points to minimize the likelihood or severity of over- or undercooling incidents is clearly desirable provided that it doesn't introduce new system failure modes. That is, a provision to delay or disable an autostart to avoid an overcooling transient ought not to have, as a failure mode, the outright disabling of the autostart system.

The recommendation to provide throttling of the AFWS to prevent overcooling is directly related to the discussion above concerning the safety grade level control. We believe that providing such level control is desirable, will help to some degree to reduce the frequency of overcooling events, and to a lesser extent reduce the likelihood that such events propagate into accidents involving core damage.

The recommendation to terminate feedwater supply to prevent an overfill condition appears to be more appropriate for the case of the main feedwater system rather than the AFWS. However, even for the former system, provisions to override the ICS and trip or throttle to avoid grossly overfilling the steam generators may - through nuissance trips - degrade plant safety by as much as this proper action may increase it. If such a protective system is deemed to be necessary, great care should be employed to design it for a very low nuissance trip rate.

Provisions to throttle or trip the auxiliary feedwater system to avoid grossly overfilling the steam generators (beyond that provided by the upgraded AFWS control system) is even more subject to adverse side effects. "Protective" systems that have the effect of isolating a reactor from its heat sink - as these do - should be avoided if possible, and entered into only with great care, thorough reliability analysis, and a careful investigation of adverse side effects. We expect that a system to trip or throttle the AFWS on very high steam generator level may have the net effect of increasing the risk.

3. Diversely-Powered Auxiliary Feedwater Pump for Davis-Besse

In this recommendation, the Task Force has noted and addressed their concern about a unique feature of the present Davis-Besse AFWS. In this plant, both AFWS pumps are driven by steam drawn from the main steam lines. The Task Force concern about this configuration was that temporary interruptions in feedwater flow to the steam generators can result in dry-out; subsequent attempts to initiate the AFWS may then be compromised by lack of motive steam. Potentially aggrevating this problem is the failure to isolate the steam lines. During the February 26, 1980 Crystal River incident and the March 20, 1978 Rancho Seco "light bulb" incident, the steam generators dried out. The remaining steam mass trapped within the steam generators was depleted by the continued operation of the main feedwater pump turbines, although the feedwater discharge valves were closed so there was no water mass replenishment by feedwater injection.

Other recommendations of the Task Force address the reduction in frequency of events which would result in steam generator dry-out. However, because such events cannot be eliminated completely and because the AFWS is a critical feature for coping with feedwater transients and some small LOCAs, we believe that a diversely-powered AFW pump for Davis-Besse is of high value in reducing the likelihood of severe accidents and accidents, and moderate value for incidents. This is further reinforced by the more limited capability of the Davis-Besse plant to cope with a total loss of feedwater because of the relatively low shutoff head of their HPI pumps.

4. <u>Modifications to the Steam and Feedwater Line Break Detection and Mitigation Systems</u>

Installed in most of the B&W pla ... are systems intended to cope with the effects of a main steam line break inside the reactor building. These detection and mitigation systems are designed to detect the affected steam generator and isolate feedwater flow to it. Licensing calculations indicate that, for the assumed conditions, continued flow of feedwater presents the possibility of reactor building pressure exceeding its design pressure and a possible return to criticality in the core (due to the severe RCS overcooling combined with a stuck-out control rod). This recommendation of the Task Force addresses the concern that such systems can initiate feedwater transients (by spurious operation) and, under certain circumstances, prevent feedwater delivery during a (non-steam line break) transient.

We believe that these detection and mitigation systems can be highly significant common-cause failure mechanisms, being both the cause of a feedwater transient and interfering with the subsequent necessary delivery of emergency feedwater (as occurred during the September 24, 1977 Davis-Besse transient). For this reason we believe that this Task Force recommendation is of moderate value in reducing the likelihood of severe accidents and accidents, and high value for incidents. We note, however, that the goal of the recommendation, to eliminate the potential for adverse interactions resulting from these detection and mitigation systems, may be very difficult to accomplish. We believe that it is important not only to consider design changes for these systems but to also reconsider the actual need for such systems. If the requirement for automatic isolation of the auxiliary feedwater system (vis a vis operator intervention) is an artifact of conservative reactor building pressure calculations, it may be preferable to remove the detection and mitigation system's control of the AFWS, rather than attempting to design a more sophisticated system.

5. Improvements to the Integrated Control System and Non-Nuclear Instrumentation

It is clearly evident from the Crystal River incident and other similar events that the ICS and NNI in B&W plants can be both the initiator of a transient event and a compromising agent in the plant's and operators' attempts to mitigate the transient's effect. While other Task Force recommendations deal with ways to improve the mitigating capabilities of the plant and its operators, this recommendation addresses means for improving the reliability of the ICS/NNI so that its frequency of failure is reduced and its failure not so severe.

Because this recommendation deals strictly with means to improve the ICS/NNI, we believe that it can provide significant benefit only for transient events initiated by faults in these systems. Thus (as Table 7.2 illustrates), we feel that these recommendations are, in general, of relatively low merit for events such as "normal" losses of the main feedwater system, small LOCAs, etc. In some cases, we also believe that specific recommended modifications might have slight negative implications. For example, modifications in meter failure position may impede operator actions in other events (until such time that the operators become thoroughly familiar with the new indications and the altered system is debugged).

For the case of ICS/NNI-initiated transients, we believe that the specific Task Force recommendations are generally of low to moderate importance in reducing the likelihood of incidents, while of generally low value for accidents, and negligible value for severe accidents. Again, since instrumentation and control equipment modifications will inevitably require some time for adjustment on the part of the operators and the I&C technicians, some increased likelihood in human error and frequency of ICS/NNI failures can be expected for some time.

We also believe that certain recommendations are of relatively more importance for the ICS/NNI-initiated type of transient. Specifically, we believe to be more important the capability for bus transfer in the event of power supply faults and the follow-up actions to IE Bulletin 79-27, which addresses on a plant-specific basis the capability to cope with power-failures to the ICS/NNI. We also note that recommendation 5d (reversion to manual control) could be of some low to moderate value (for accidents) if this change were to remove the possibility that faults could disable both automatic and manual control of the plant secondary side. If the recommendation does not accomplish this, then we believe it to have negligible importance.

6. Installation of a Safety Grade Panel of Vital Instruments

This Task Force recommendation is similar to the Lessons Learned Task Force recommendation 7.2 and calls for a safety-grade panel of instruments in the control room which is independent of other instruments, their power supplies, etc. and their associated potential for common-cause failures.

The installation of such a safety-grade panel would provide the operating crew with a credible source of information during events which affect other plant instrumentation. Other Task Force recommendations have as a goal the reduction in frequency of such losses of instrumentation; however, since such losses cannot be eliminated (or even substantially reduced in frequency), we believe that such a safety panel is important. Since it is a virtual certainty that operating crews will in the future be faced with faulted non-nuclear instrumentation during a transient, such a safety panel can significantly improve the likelihood that the operators will correctly diagnose and cope with the transpiring events (presuming that these instruments are powered from appropriate supplies, e.g., batteries). For this reason, we believe that this recommendation has high value for incidents, high value for accidents, and moderate value for severe accidents.

7. Improved Use and Display of In-Core Thermocouple Indication

This Task Force recommendation has two aspects: the improvement in the capability to use the in-core thermocouples (as one input to the subcooling meter); and the improvement in the display capability of the thermocouple indications, so that trend information in core outlet temperature (temporal behavior, regional variations, etc.) is available to the operators. Apparently, thermocouple indications were used by the Crystal River operators during the February 26, 1980 incident while much of the other instrumentation was failed or of questionable credibility.

As we have discussed above, it is highly likely that instances of large-scale instrumentation failures will in the future be experienced by operating crews, so that reliable information from diverse sources such as the in-core thermocouples will be important to the operator response to the events. In this sense, this recommendation is coupled with Task Force recommendation 6 (Safety-Grade Vital Instrument Panel). Because the latter recommendation calls for the provision of several indications of RCS status, we believe that it overshadows the potential benefit resulting from the improved use and display of the thermocouple indication. Thus, while we feel that better use and display of the thermocouple indication would be a desirable capability, we believe that the installation of the "safety panel" is distinctly more important. In this context, this recommendation appears to be of low importance for incident and accident mitigation and of negligible importance for severe accidents.

8. <u>Safety-Grade Vent/Purge Isolation on a High Radiation Signal</u>

This Task Force recommendation calls for the installation of safety-grade isolation equipment on the reactor building vent/purge system which would be actuated on high radiation levels in the reactor building. This is of concern because, for some events, isolation of the vent/purge system on high building pressure or low RCS pressure might not occur until after the release of some radioactive material. For example, for a total loss of feedwater accident (i.e., both main and auxiliary feedwater fail), RCS pressures would climb rather than drop sufficiently to cause the building isolation on low RCS pressure. Further, the operation of the purge might prevent building pressures from reaching the other isolation setpoint; thus, automatic isolation might not occur. Under such circumstances, operator actions to isolate the vent/purge system might not occur until some material (e.g., radioactive gases released from the expelled coolant) has escaped through the system. To cope with such a situation, a vent/purge system isolation on high radiation level in the reactor building has been recommended.

In essence, the intent of this recommendation is to substitute automatic isolations (on high radiation) for operator-initiated isolations for that class of accidents where the "normal" isolation-initiating signals would not be received. The consequences of not providing such an isolation can be thought of as the difference in the magnitude of release if an automatic isolate were to occur and if the isolation were dependent on operator action. Since the concentration of radioactive material in coolant is relatively low, we believe that the increased time required for human actuation of the vent/purge system isolation would result in only a small difference in the radioactive release. For this reason we believe that this recommendation is of negligible value with respect to severe accidents, and low value for accidents. We also believe, however, that it could be important (in the severe accident category) to assure that these valves fail closed on loss of power, so that isolation occurs in the event of such potentially severe accidents as station blackout.

We note that the above conclusions on the relative merit of this recommendation are based on the conclusion that small releases of radioactive material during an incident will result in negligible health effects within the surrounding public. If, however, the objective is to prevent <u>any</u> release of radioactive material, this recommendation clearly is more desirable; for this reason we believe it is of moderate value with respect to coping with incidents. We also note that an anticipatory trip of the containment purge isolation valves could also be triggered on high pressure in the reactor coolant drain tank.

9. <u>System Response Modifications to Prevent Pressurizer Level Loss and ECCS Actuation</u>

Following a reactor trip in a B&W plant, the reactor coolant undergoes significant contraction as it cools; as a result, the pressurizer level and RCS pressure drop substantially. To cope with this, operators are trained to quickly isolate letdown flow and start an additional make-up (HPI) pump, so that shrinkage is accounted for by additional coolant injection into the RCS. Even with such operator intervention, however, these plants have a history of occasional secondary side malfunctions leading to reactor trips, losses of pressurizer level, and ECCS/HPI actuations (on low RCS pressure). This Task Force recommendation calls for the examination of means to reduce the severity of the post-trip RCS transient, so that the frequency of level loss and HPI actuation is reduced.

A reduction in the frequency with which pressurizer level is lost and/or ECCS is actuated in overcooling accidents is useful in several ways. Frequent ECCS actuations due to overcooling transients may condition operators to expect all ECCS actuations to be spurious and encourage them to disable the autostart of emergency feedwater (to avoid the overcooling) or to override the ECCS start without positively determining that there is no genuine need for it. Thus, it is important to avoid or counteract (with training) this effect on operator behavior.

Apart from the effect on operator behavior, the frequency of overcooling transients leading to loss of pressurizer level or spurious ECCS actuation has little bearing on the likelihood of core damage and still less on public health and safety. The failure of ECCS under such challenges has almost no safety penalty since ECCS is not really needed in this scenario; it offers an opportunity to gain experience and debug the system. The success of ECCS under such challenges may lead to increased challenges to pressurizer relief and safety valves, which might then fail open. However, the ECCS system needed to mitigate such failures must be accorded higher-than-average reliability in such situations because its operability was responsible for the opened valve in the first place.

Thus, virtually all of the moderate significance (with respect to the incident accident category) attributed to this recommendation relates to its effect on operator behavior. We also believe it 's of low value with respect to reducing the likelihood of accidents, with negligible value in the severe accident category.

10. <u>Study of Means to Improve the Response of the Once-Through Steam Generator (OTSG)</u>

In this recommendation, the Task Force has addressed the concern of the relationship of the relatively small OTSG secondary side coolant inventory to the overall "sensitivity" of the B&W plant. The recommendation suggests that both active and passive means to improve the OTSG response be investigated.

We recognize as the Task Force did that there are a number of ways possible to improve the OTSG responsiveness. Such design changes to the OTSG obviously have the potential for significantly improving the overall behavior of the plant during feedwater transients (or, if poorly designed, having negative impact). Equally obvious is that, since we do not now know what the study results would show, we cannot pass judgment on its relative merit. For this reason, we believe that it is sufficient that we concur on the Task Force recommendation that such a study be undertaken.

11. <u>Elimination of Post-Reactor Trip Operator Actions</u>

As was described in our discussion of recommendation 8 above, following a reactor trip in B&W plants, the operators are required to take certain actions to help minimize the post-trip pressurizer level and RCS pressure decrease. Additional operator actions are also required in the event of a small LOCA to balance HPI flows, etc. This Task Force recommendation calls for decreasing the burden placed on the operators during this time period by reducing or eliminating (automating) the immediate manual actions required by the emergency procedures.

By removing those requirements on the operator to act, one allows the operator the opportunity to think more broadly about his situation. For this reason, we believe that the reduction in the demands placed on the operating crew during the early phases can have an important impact on their capability to cope with the accident, i.e., reduce the likelihood of errors during the event.

Thus, we believe that this recommendation has negligible potential for reduction in the likelihood of severe accidents, and low benefit for accidents and incidents.

We note that, under certain circumstances, the automation of post-trip actions can also produce adverse effects. Care should be taken when automating certain functions (e.g., letdown isolation) to avoid potential adverse interactions with ICS/NNI. Since we do not believe it possible to eliminate the occurrence of large scale instrument failures, etc. resulting from ICS/NNI failures, prudence dictates that newly-automated functions be subject to thorough failure modes and effects, common-cause failure, and interactions analyses.

12. <u>Instrumentation and Control Technicians Be Assigned to all Shifts</u>

This recommendation addresses the Task Force concern that power faults, etc. which result in severe ICS/NNI failures can be sufficiently complex that trained instrumentation and control personnel are required to study and correct the problem. Since it is not now the practice of all plants to have such personnel on all shifts, there exists the potential for extended fault rectification times if staff must be brought in from offsite in an emergency. Because of this concern, the Task Force recommended that appropriate personnel be available on-site during all shifts.

We believe that this recommendation has both positive and negative aspects. On the positive side, we agree with the Task Force that having trained personnel available would be somewhat beneficial - probably of moderate value for incidents and accidents, and low value for severe accidents. However, consideration of the data on the causes of large scale ICS/NNI failures indicates that roughly one-half of the events were a result of errors made by these same personnel as they performed their surveillance and maintenance duties. Since presumably these personnel would be performing their routine duties during their shifts, the likelihood of experiencing an ICS/NNI failure on back shifts would be increased somewhat by requiring the appropriate personnel to be present. On balance, we believe that the positive aspects of this recommendation slightly outweigh the negative aspects; however, we also believe that the "net gain" is of low value. Recommendation 14 is more to the point.

13. Operator Trainir ... the Crystal River Incident

14. Development of Plant-Specific Procedures for Loss of ICS/NNI

We have c... en to consolidate Task Force recommendations 13 and 14 into one for the purposes of this risk evaluation because of their similarity in intent. Recommendation 13 of the Task Force calls for specific operator training on the events of the February 26, 1980 incident at Crystal River. Recommendation 14 addresses the need for plant-specific procedures to assist operating crews when ICS/NNI failures occur in the future.

We believe that the reduction in the likelihood of operator errors during ICS/NNI-caused transients requires operator training involving both retrospective and forward-thinking views. The Task Force's recommendation on Crystal River training provides one aspect of the retrospective training; however, this specific training alone does pose questions regarding the need for training on other similar events, e.g., the Rancho Seco "light bulb" incident or others identified from LERs as having the potential to be accident precursors. We believe that this type of training could be highly valuable in "preparing" the operators for possible future accidents.

The Task Force recommendation on plant-specific procedure development addresses the need for forward-thinking training. Since it is a virtual certainty that operators will be faced with ICS/NNI failures in the future (which may be similar to or different from past events), we believe it important that more general training on coping with such events be provided.

We believe that this combination of training for ICS/NNI faults can be of relatively high effectiveness for this type of transient. Other recommendations reduce the significance of these incidents, e.g., recommendations 2 and 6. We believe that on an overall basis, these recommendations are of high value for incidents, high value for accidents, and moderate value for severe accidents.

15. Increased Simulator Training

This Task Force recommendation calls for the requirement of a one week per year simulator training course for all operators in B&W plants (this training is now optional).

We believe that this recommendation has both positive and negative aspects. On the positive side, such simulator training can be important to the understanding of plant behavior during transient events, LOCAs, etc., and thus be a useful means to reduce the likelihood of operator error during real events (e.g., Crystal River type "incidents" and TMI-2 type "accidents"). We believe that making such training mandatory, rather than optional, is of moderate value for incidents, moderate value for accidents, and negligible value for severe accidents.

The negative aspects of this recommendation result from our concern about the limitations of the available simulator capability. First, the B&W simulator is made to resemble the Rancho Seco control panels, so that operators from other plants may have difficulty in fully melding together their training with their own control room. Second, present simulators tend to have difficulty in accurately recreating some transient events, so that the training can again be somewhat counterproductive. Overall, however, we believe that these negative aspects do not overshadow the gains achievable by the simulator training, so that we agree that this training should be pursued.

16. Criteria for Restarting Reactor Coolant Pumps

This Task Force recommendation is concerned with guidelines provided to the operators of B&W plants with respect to the restart of the reactor coolant pumps during non-LOCA transients. B&W has provided these guidelines to the operators; however, the NRC staff has yet to conduct their review. The recommendation calls for the expeditious completion of the NRC review.

We believe that appropriate guidance on the restart of the reactor coolant pumps can be an important aspect in the prevention of core overheating and damage. Forced-flow cooling of the fuel can be highly advantageous during events where malfunctions have interrupted decay heat dissipation, so that clear criteria for re-establishing this flow appears to be of significant merit. Because of the potential merit of quickly re-establishing reactor coolant pump flow, we believe that the completion of the NRC's review of the restart guidelines is of moderate value for improving the capability of the plant to cope with incidents and accidents, and low value for severe accidents.

17. <u>Alternative Solution to PORV Unreliability and Safety System Challenge Rate Concerns</u>

This Task Force recommendation addresses the concern that, since the post-TMI switch of the PORV setpoint and the reactor trip setpoint on high RCS pressure (and other related plant modifications), the frequency of reactor trips in B&W plants has increased. It appears that transients which formerly would have been accommodated without causing a reactor trip now do result in trip. Since this increased trip frequency has some negative impact on plant safety (e.g., increased likelihood of an ATWS event), the Task Force has recommended that a proposed plant modification plan (submitted by Consumer's Power Company) which would allow a return to the pre-TMI setpoints be considered by the NRC staff. If determined to be acceptable by the staff, the Task Force recommends that such modifications be required in all B&W plants.

It is apparent that the return to the pre-TMI PORV/reactor trip setpoints has both positive and negative aspects. On the positive side, the return to the original setpoints could reduce the likelihood of ATWS events to some limited extent, and allow the plants to operate in a way more like that to which they had been originally designed. The latter aspect may help somewhat to minimize unusual behavior of the plants during transients (i.e., it allows them to respond more smoothly during such events).

On the negative side, the return to the original setpoints will increase the frequency of use of the PORV; with this increased frequency the likelihood of experiencing a stuck-open valve (a small LOCA) increases commensurately. While the installation of an automatically-closing PORV block valve may alleviate this aspect, it also presents other problems. In some accidents (e.g., a total loss of feedwater), the PORV is the only controllable means for energy removal from the RCS. In such instances, an open PORV can be advantageous, in that it permits RCS depressurization with the associated increased HPI flow. Further, for plants with relatively low-head HPI pumps (e.g., Davis-Besse), a stuck-open (or commanded open) PORV is the only means for the critical RCS depressurization. In such situations, automatic block valve closure can be distinctly counterproductive. Also, the automatic closure of the PORV block valve could, for events such as a total loss of feedwater or the Crystal River incident, result in unnecessary challenges to the (unisolable) safety valves. Thus, block valve auto-closure can increase the challenge rate of the safety valves, resulting in an increased likelihood of a <u>bona fide</u> LOCA. It is noteworthy that during the February 26, 1980 Crystal River incident, operator actions to close the PORV block valve (as required by NRC) resulted in the opening of the safety valves, with the resulting increase in coolant release to the reactor building.

The return to the original setpoints appears to have merit. Improved PORV block valve reliability is also clearly desirable. However, the automatic closure of the block valve(s) appears to have undesirable side effects. While not as critical as some other Task Force recommendations, we nonetheless believe that the resolution of this issue is still important. We believe that this recommendation is of moderate value for the incident category, low value for the accident category, and of negligible value for the category of severe accidents.

18. Completion of the IREP Crystal River Study

This Task Force recommendation relates to the Probabilistic Analysis Staff's risk evaluation of the Crystal River plant, which is the first part of the overall IREP study of all operating plants. This study has as its goal the identification of those factors of the plant design which are important to the public risk from that particular plant. The recommendation calls for the expeditious completion of the Crystal River study, with prompt consideration made by the NRC on the need for plant modifications suggested by the study.

The IREP Crystal River study has as a goal the identification of those plant faults which have the greatest potential for causing core damage and risk to the public for events initiated by transients and LOCAs. For this reason, we believe that such an identification can have high value for accident sequences resulting from "routine" losses of feedwater, station blackout, and small LOCAs. Since other initiating events have not been as thorou hly evaluated (e.g., losses of ICS/NNI, etc.), the potential f equency reduction potential for such sequences is less significant. Since the results of the study (and the subsequent regulatory actions) are not yet completely clear, we cannot now determine the importance of the study results on plant safety.

19. Performance Criteria for Anticipated Transients

This Task Force recommendation calls for the development of performance criteria to define the acceptable limits of plant response to anticipated transients. The purpose of the criteria is to assure that those plant functions critical to coping with transient events are designed to adequately protect the core during such events.

Without knowing what factors will be considered in the development of these performance criteria, we find it difficult to assess the relative merit of this recommendation in relation to others made by the Task Force. Development of criteria for system performance, such as reliability, human and systems interactions potential, etc. could provide significant payoff; for this reason, we agree that this relatively long-term Task Force recommendation should be pursued.

20. Criteria for Reactor Coolant Pump Trip in Small LOCAs

In the post-TMI reconsideration of small pipe break accidents, a concern arose that for certain sizes of pipe breaks, the running of the reactor coolant pumps might aggrevate the break flow to the extent that licensing requirements on acceptable accident fuel temperatures would be exceeded. As a result of this concern, the NRC now requires that the reactor coolant pumps be tripped under certain conditions when it is believed that a small LOCA exists. The NRC staff has acknowledged that such a requirement may impede the capability for recovery from other types of events and as such has recommended that the question of the relative merit of pump trip continue to be pursued. This Task Force recommendation endorses the previous staff and industry recommendations on this matter.

We agree that the present requirements for pump trip are less than ideal. While for some small LOCAs it may be preferable to trip the reactor coolant pumps, clear benefit in continued pump operation may be seen for other sizes of LOCAs and for non-LOCA transients which have some symptoms similar to those of LOCAs. We believe that this concern is of moderate value in the capability of the plant to cope with incidents and accidents, and of negligible value for severe accidents.

21. Reevaluation of the AFWS Injection Point into the Steam Generators

In general, B&W plants inject the AFWS water into the steam generators through a feedwater ring at the top of the steam generators, so that the water sprays directly onto the steam generator tubes. In contrast, Westinghouse and Combustion Engineering plants are designed such that AFWS flow enters through the main feedwater rings, filling the steam generator from the bottom. Because of top-entry of AFWS water increases the potential for an RCS overcooling transient, the Task Force has recommended that reconsideration be given to the relative desirability of top-entry and bottom-entry of AFWS water.

We believe that both points for AFWS entry have positive and negative aspects. Top-entry has the advantage of providing a higher effective thermal center in the steam generator, so that natural circulation cooling would be enhanced. Prospects of recovering from situations entailing degraded core cooling are better with top-entry injection. It is thus important to safety not to lose this option. As noted above, this entry point does, however, have the disadvantage of increasing the likelihood of overcooling the RCS. Bottom-entry does reduce the overcooling potential, but also lowers the steam generator's thermal center. The latter entry point may also pose problems of thermal shock of the feedwater lines, nozzles, etc. We strongly recommend against eliminating the top-entry injection option. Further, the added complexity of top and bottom injection point options is probably not warranted by the small risk reduction potential in reducing overcooling events. In our judgment, we believe this recommendation to be of low value in the reduction of incident frequency, and negligible importance to the categories of accidents and severe accidents.

22. Study of Operator Errors in B&W Plants

In reviewing the operating experience of B&W plants for instances of ICS/NNI failures, it became apparent to members of the Task Force that the frequency of operator errors in these plants tended to be somewhat higher than that for other plants. This Task Force recommendation calls for an evaluation of the compiled data to assess the statistical significance of this apparent difference.

The Probabilistic Analysis Staff has determined that the differences in operator error rates in Table 5.3 of this report are not statistically significant. However, PAS has under contract a research program to study the kinds and frequencies of operator errors being reported in LERs, to relate these to plant, vendor, and circumstance. These studies may lead to insights that can be used to reduce human error contributions to the risk.