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NUCLEAR REGULATORY COMMISSION

In the matter of:

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

303rd General Meeting

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards

303rd General Meeting

1717 H Street, N.W.

Room 1046

Washington, D.C.

Thursday, July 11, 1985

The Advisory Committee on Reactor Safeguards met in open session, pursuant to notice, commencing at 1:50 p.m., David Ward, Chairman of the Committee, presiding.

ACRS MEMBERS PRESENT:

David Ward	G. Reed
H. Lewis	C. Wylie
J. Ebersole	F. Remick
D. Moeller	P. Shewmon
W. Kerr	C. Mark
M. Carbon	C. Siess
D. Okrent	R. Axtmann
H. Etherington	

ALSO PRESENT:

R. Savio, ACRS Staff Member

E. Igne

H. Alderman

R. Fraley

PRESENTERS:

V. Stello	T. Spies
T. Murley	F. Rowsome
F. Gillespie	S. Brocoum
S. Israel	D. Brand
E. Jordan	A. DeAgazio
B. Sheron	G. Rivenbark
D. Powell	H. Wong

PROCEEDINGS

MR. WARD: Our next agenda item is No. V, quantitative safety goals. I think we will start out with the subcommittee report. Dr. Okrent.

MR. OKRENT: Well, Dick is handing out now a slightly revised version modified, or whatever you want to call it, of the draft we were talking about on the Saturday of the last meeting, and one can see what the changes are by crossout and underlining. Okay?

Some of them are just editorial, some are for purposes of discussion, some sort of seem like they might be improvements as a result of things that have transpired yesterday and today, et cetera.

I should note that at the back of this, at the clip you will see something in blue which is a draft of a possible letter that I was asked to prepare, so that is something that we would presumably look at Saturday.

Since it is not completely disconnected, in my opinion, from some of the things that relate to safety goals, and since I managed to get an idea for a possible draft, it is available now.

Other than that -- well, let's see --

MR. WARD: Aren't we going to hear from the Staff?

MR. OKRENT: We are in a minute. There is an agenda you have. Has that been handed out?

MR. WARD: Yes.

MR. OKRENT: Okay. And I would propose that, you know, we had about a four-hour subcommittee meeting yesterday, but rather than my trying to provide a summary of it now, since we have most of the principal actors from yesterday here today, I would propose unless someone else wants to make a comment first, that we move into the part with the NRC Staff now.

MR. WARD: Okay. Good.

MR. OKRENT: I assume we still have two hours with this agenda item; is that correct?

MR. WARD: That's right.

MR. OKRENT: So in that case, Vic Stello is the first one up.

MR. STELLO: Well, let me start by saying I had the opportunity to sit and listen to the committee's discussion of this item with the Commission, and I wanted to start by saying it is somewhat troubling to me for two reasons:

The committee --

MR. REMICK: I'm sorry, I'm having trouble hearing you, Vic.

MR. STELLO: Well, it works.

The Committee continues to remind us that when there are issues that they are particularly interested in, they want to make sure that we make them aware that we are working on

those issues and give them copies of the appropriate papers and reports so they would have early input.

We did that with respect to the safety goals. We clearly have not evolved to where we are prepared to say what the Staff's view is on this issue.

We wanted to make sure that the committee was aware of the variety of views on a number of issues that are contentious.

Sitting listening to the discussions, it seems that there were a number of comments that puts the Staff in what I would characterize, I guess, fairly as a bad light, and I don't think that that's fair.

If the committee wishes to hear what the Staff is thinking before the Staff has a view, then I think that's healthy and I think it's proper and I think we ought to do that. Then there has to be an understanding that you are hearing fairly rough ideas that have not had even a chance to be adequately discussed within the Staff, and it is unfair to ever bring the result of those discussions in any unfavorable light.

It would be unfortunate if the net result is that the committee -- in the way it came across at least to me -- was suggesting that the only time they want to talk to the Staff is when there is a final polished complete decision by the Staff and the Staff is ready to stand behind that.

I think that would be a big mistake. I think that the committee and the interaction of the committee with the Staff in working out some of these very difficult areas is very, very beneficial, and I quite frankly find it very, very useful -- and I have repeatedly said and I would start by suggesting again that there are a number of these areas that if the committee cannot write a letter on this generic topic of safety goals, there are at least a number of parts of it which no matter what the safety goal finally says are particularly difficulty issues, and perhaps the committee could find a way in which to state its view regarding some of these difficult issues.

Those are the difficult issues, I guess, that I wanted to go to next and would start by hoping that the committee, if it is unable to write a definitive letter -- and I understand there are a variety of views on the committee on this subject -- at least we could focus on a number of difficult subjects and get the committee's advice, counsel, if it has to have half a dozen additional comments. I don't think that that's bad, either, in trying to at least bring into light some help on some of these issues.

The two that I emphasized when we were here last time, the same two issues that I guess I want to emphasize again today, to avoid any controversy, let me say that anything we say, we are going to mean "mean," whether it turns

out that way or doesn't turn out that way. I don't think it's relative. I think a discussion of mean and median -- well, I just don't think it's very productive.

MR. LEWIS: If I could just interrupt you for a second, I agree with that. And the reason for my frustration -- and as we both know, I said the nastiest things this morning -- is that it has been around for at least two years, to my personal knowledge, probably longer than that, and so it is hardly an issue which has just come up and on which the Staff has not reached agreement.

I applaud what you just said.

MR. STELLO: Let me just suggest that it be in the first paragraph you write in the letter. Get the committee to agree that from now on whenever we talk about any of these results, we mean "mean." Fine. If that's the committee's view, say it. I think it would be useful. It may have made that thing go away a year ago, I don't know. If you can't deal with the whole of the subject, at least deal with the parts of it that continue to be a problem and we can get them out of the way. I don't think it's worth very much discussion.

But there are two areas that I do believe are worth a great deal of discussion, and that's the whole issue of where you set the performance criterion at. At 10 to the minus, 10 to the minus 5. Do we mean by 10 to the minus 4 the

classical results of PRA which are predictors of the capability for the core to be severely damaged, which may lead to a full scale core melt, where the core physically melts through the vessel, in contrast to a goal of 10 to the minus 5, for which the core is already calculated to have left the vessel and its fission products then are inside the containment.

I think that's a fundamental, philosophical point for which one has to go back and look at what can you get at, as a result of the calculation? Can you ask the PRA technology to take that next step?

If we do, are we then really putting off using any in any meaningful way safety goals until we have developed yet even further PRA technology to be able to discriminate between those two kinds of calculations?

My preference is, I would rather we didn't do that, we would use the PRA technology for what exists today and take those numbers and use those numbers and just recognize that there is some degree of conservatism in using them in that manner, whether it's two or 10. I don't know that anybody really knows. You've had numbers offered to you as to what the differences are.

It would be very, very helpful if the committee can deal with that subject. That is a policy matter that the Commission clearly has got to come to grips with, what ought

that policy be, and we ought not to use any surrogates for how to get there. That is, we ought not to try to decide that a cost-benefit analysis doesn't produce the kinds of changes we'd like to make and therefore let's make it some other number. Let's just do it strictly on the matter of policy.

It is a policy issue as to where that ought to be set. I think we ought to decide that issue and move on with it.

Cost-benefit analysis. We spent a great deal of time talking about cost-benefit analysis. There is a great deal of controversy behind it. Where ought we go with this issue?

I was, I guess in terms of summarizing it -- we go where we were before with the area of cost-benefit, which would not include any of the averted costs. That's what the draft safety goal was all about. Or we could go to the end of the spectrum, which was suggested, as all costs, all benefits, all of the time.

I think this again is a matter of policy as to whether it's important to have that kind of information displayed to decisionmakers, and the decisionmakers in this agency may reside in a lot of places. This room -- you-all are decisionmakers. Eventually you reach a conclusion and you write a letter that says what it is that you have decided on an issue.

The Director of NRR has to reach a conclusion or a decision on a particular licensing matter.

Hearing Boards reach conclusions.

The EDO has to reach conclusions. The Commission.

What information ought that decisionmaker see with respect to these issues before he decides? I can say it as simply as I know how:

I'd like to know everything that there is know about the issue that's reasonably obtainable before I decide it. And I say that simply as saying personally I like to see all costs, all benefits.

Saying it another way, I'd like to know everything that I can find out about the issue before I decide it.

Now, one other point that I don't know how to emphasize enough, so it isn't misunderstood, as I gather there is a lot of misunderstanding about:

There's nothing magic about safety goals. The safety goal is not going to be a mechanism whereby we replace the current regulatory process. It will be an additional element in the regulatory process. The report from the steering group, in fact, I think, used language something like "we should not supplant the current traditional regulatory methods that we use," the defense-in-depth concept.

I think that's stronger than we ought to say it. But even in the extreme, it is clear that you are not going to

use what you get from safety goals to change that regulatory process from the fundamental way we do things today.

If you look at the back-up report from the steering committee, you will find that most of the decisions that have been made that were tested with the safety goal would not have passed the safety goal standard of $1000 a manrem.

Those requirements issued, even though they didn't meet that cost-benefit balance. And I suspect that that's probably the kind of thing that will continue in the future. But there ought to be a rationale. You ought to know what the results of that cost-benefit analysis are. If you decide not to go that way, there ought to be a reason for doing it.

On the other hand, if we get results of PRA that show that our traditional methods of regulation suggest that we're not doing something that's needed for safety, then by definition it ought to supplant those traditional methods and we ought to require something where PRA methods show us we ought to. If they clearly show us that we ought not to do something, then too we ought to give that some substantial weight.

I don't know that this is written out plain enough, because it seems to create a great deal of confusion, at least when I listen, as though there's some fundamental change that's going to take place, should we ever adopt safety goals.

I don't think that that's advocated or suggested yet

by anyone. I think we are a long way from where we would make a transition from our current regulatory process to a completely new regulatory process derived from the kind of methodology embodied in the safety goals. We are not there.

But we're never going to learn a great deal, in my view, until we get started. I prefer to start, and start slow, do what we can, use it intelligently and rationally. If it's being used improperly, I'm sure the ACRS will remind us of when the use is improper. We all ought to be watching each other. But I would urge that we get on with it. I think it's very, very important that we do that.

And again I want to plead that the ACRS try to pick these -- at least these several difficult issues and try to decide where it comes out on these difficult issues and at least put down some sort of comments back so that we would have the benefit of the counsel of the committee in these areas. If it can't bring itself to deal with the whole issue, I urge it at least deal with parts of the issue so we can make as much progress as we can possibly make.

I think I've probably said more than enough already. There are a number of other changes that I think should have had more discussion when we gave them to the implementation section, in particular, for which if you will read that one very carefully, it says that that is to be developed later by the EDO, and I don't think we ought to

spend a great deal of time about it.

If there is any particular issue about the implementation -- and that's the one that describes how the Staff would use results that are bounded from 10 to the minus 3 to 3 times 10 to the minus 5 in terms of results, and how we would implement that. If we need to talk about that, then I think we ought to.

We certainly didn't in the subcommittee. There are a number of other changes that we talked about that I briefed you about last time that we're thinking about.

With that, I'd like to ask Themi Speis to summarize the NRR position on this matter.

[Slide.]

MR. SPEIS: I am Themi Speis from the Office of NRR.

I am here to summarize some of the views that were provided to the EDO by Harold Denton. Unfortunately he is on vacation and decided to stay there. He is somewhere in North Carolina. We did talk to him last night, and he was briefed on what happened at the subcommittee meeting, and basically he told me to relate to you people that he has been personally involved in our safety goal deliberations, he has been briefed by members of the steering group that came from the Office of NRR. He has been briefed by Dr. Murley. He has been discussing this with Mr. Dircks, but he still is reviewing all these important issues that you have been hearing and talking

about the last few days, and maybe the last few years.

So his views are still evolving, even though he is not withdrawing the letter or anything like that. But, you know, his work is input to the EDO and his final concurrence will depend on the views that the other offices in EDO and what views you might have and things of that sort.

Frank Rowsome briefed most of you yesterday. I will repeat some of the same things.

One of the things that Harold told me was to separate the two viewgraphs that were presented yesterday and make sure that in the first viewgraph I have the one serious difficulty, so I put the other things he said about the safety goal in the other viewgraph. He doesn't want to mix the big things with the small things. So I did that. It was very easy to do it.

So you see that one of the difficulties that he has, that he is suffering with this issue, is the goal itself, whether it should be 10 to the minus 4 or 10 to the minus 5. You are familiar with the 1 minus E to the minus-lambda d type of calculations. If you assume you have 100 plants and if you assume you have a coremelt probability of 3 times 10 to the minus 4 and plot the numbers there for the next 20 years, you'd come up with a 45 percent chance of having a serious accident in the next 20 years, and a 10 percent chance of two or more such accidents.

So this kind of bothers him, you know. He thinks that the number looks too big.

The other thing that bothers him is we can all say that we can have a core melt accident but nothing will happen because the containment is there and the appropriate systems included within the containment will prevent any type of radioactivity getting out, and even though he believes that substantial progress has been made in this area, in both analytical and experimental to some extent, and we know more about what type of challenges are generated inside the containment as a result of a serious accident and how containments respond to the challenge, he still feels that maybe we are putting too much reliance on those types of analyses and how the fission products behave and how the containments perform.

Therefore, from that viewpoint, maybe the performance guidelines should be higher than 10 to the minus 4.

The other problem that he had when he put this together was that it wasn't very clear from the Steering Group's report what the 10 to the minus 4 was. This was discussed today and was discussed at length yesterday, whether it was a core damage type of number or if it was a core melt with no vessel penetration, so he wanted to tighten this up, and his 10 to the minus 5 in both reactor years is a

large-scale core melt where the vessel in the primary system has indeed been penetrated and the containment is challenged.

You heard numbers like factors between 2 and 10, variations between core damage and core melt where the vessel has been penetrated, depending on who the analyst is. Of course, as you all know, it is very hard to precisely identify which are the sequences that will -- that won't progress from core damage to core melt. That depends on the sequence and many other characteristics of the sequence and the system that you are dealing with. But we also know that the PRA numbers are numbers that come -- the characterization of those numbers are inadequate core cooling.

So there are some sequences that if they come from things like station blackout where probably there isn't that much difference between core damage and massive core melt where the primary system has been penetrated. But on the other hand you can have some other sequences where possibly you lose recirculation or something like that, and the chance of preventing that sequence from going from core damage to core melt is probably -- there is a chance to do that with that sequence.

MR. KERR: Is the implication of the last line in connection with the rest that although he thinks the 10 to the minus 4 is too large a number, that he would be satisfied with 10 to the minus 5 for an upper limit to a probability of large

scale fuel and fission product release?

MR. SPIES: Well, he has come up with this number. Let me show you some calculations that have been performed and are very near target.

[Slide.]

The 50/50 number that I mentioned earlier was based on 3 times 10 to the minus 4, which really was a 45 to 55 percent. If the goal -- if the number is one times 10 to the minus 4, then the probability that in the next 20 years in the population of 100 plants you will have a core melt is 20 percent. If you go down to one times 10 to the minus 5, it's really 2 percent. So there is an order of magnitude between this number and this number [indicating], but if you stop -- if the goal, say, is three times 10 to the minus 5, you are still substantially below the 50 percent.

He thinks that the 50/50 number is too high and possibly we should have some other goal that gives you some number that has a better confidence.

MR. KERR: I guess I must not have made my question clear. I thought you were saying that the 10 to the minus 4 for severe core damage was too big, but that if one adopted 10 to the minus 5 for release of the core, melt through of the core through the vessel and release of fission products into the containment, that that would be acceptable to him.

Is that what the first slide means?

MR. SPIES: Basically yes, yes. But, you know, based on this spectrum of calculations, you know he is not dogmatic that this should be the number, okay.

MR. KERR: No, I'm not trying to agree or disagree with him; I just want to make sure I understand him.

MR. SPIES: We discussed this last night. Yes.

MR. REMICK: A question for clarification. Where does the 3 times 10 to the minus 4 come from? What's the origin of that?

MR. SPIES: Okay. This is based on six industry PRA's and in two or three of them the staff did its own review, and this is an average of those numbers from the six PRA's. Some of them are industry numbers; some of them have been revised as a result of the study.

MR. LEWIS: Weren't they, in fact, all claimed to be means?

MR. SPIES: Yes. At least that's what they claim. But there's always a big difference between what they claim and what the real world is. If you remember, you did the review of the WASH-1400. They had this number that they call a mean and they put a distribution around and then a normal distribution, but that distribution could have been something else.

MR. LEWIS: But all those claimed to be means.

MR. SPIES: That's all claimed, yes.

MR. WARD: Themis, a question. Do you recall the letter that the ACRS wrote in, I guess, October of last year? I'm not sure it was October. In which we provided an estimate -- it was an answer to Commission Asselstine's questions. We provided an estimate of --

MR. SPIES: I recall that. I have read so many letters that I don't --

MR. WARD: Yes, I know how you feel.

Well, I just wondered. The number that you seem to be using and that has been floated around as kind of the staff estimate of the probability of an accident in the next 20 years is something more like this 45 percent or 50 percent. The number we estimated was more, as I recall, like 10 or 15 percent. Do you understand what the differences were there?

MR. SPIES: The only thing I can that I remember very well is we came up with this calculation. As I say, it's one minus E to the minus lambda d. If I remember correctly, Commissioner Asselstine took the same numbers and he put some uncertainties plus or minus, and that number went up to 80 percent, and on the other side it went to 10 percent or 5 percent. But I don't recall how you arrived at the 10 or 15 percent.

Dr. Okrent, you recited yesterday -- I don't recall the numbers.

MR. WARD: Maybe it's as simple as we used 10 to the

minus 4 as our eyeball average or something -- is an eyeball average a mean or a median?

MR. SPIES: If you used 10 to the minus 4, then --

MR. LEWIS: We used 10 to the minus 4 and stated that we were using the proposed safety goal as the basis for that. Although that didn't specify whether it was median or mean.

MR. MARK: But I think we treated it like a mean.

MR. LEWIS: Yes. We used it like a mean.

MR. SPIES: All I did was use was the paper from my college days of 30 years ago.

MR. WARD: They're probably still good.

MR. SPIES: So enough of this. This is the issue that he is suffering with more than any other issues right now.

MR. OKRENT: Can I ask a question, Themis? Suppose the Commission were to adopt that proposed alternative. Do you have reason to think that it's approximately achievable, for the large majority of the existing reactors -- and by existing I mean operating or under construction -- by making necessary changes?

MR. SPIES: I would give you one view, and I would like most of the experts sitting back there to add or subtract to what I say.

I think based on my perception, if we look at the

sequences that really take you to massive core melt and vessel melt through and do something with them -- you know, like station blackout issue, -- you know, if you can come to grips with such an issue, then I think that the delta between core damage where the stuff is contained inside the vessel and massive melt through the vessel and challenging the containment would be large, or would be maybe a factor of 5 or 7, something like that. And then in that sense, you might get closer to achieving this goal. But that's one view, and I would like to have some of the other experts --

MR. STELLO: I think the correct answer is we just don't know, because you're asking to go back into all of those analyses and ask the question, redo the calculations and put them into two bins; those that go to full melt through of the vessel and those that don't, and then, what are the two numbers.

I think until at least you have done with the calculations that exist, I don't know that you can answer the question. You need to really know what scenarios there were that went to the full core melt, and what it is that we are or aren't doing anything about. But until you've done that, I don't think you can really answer that question.

MR. LEWIS: You will add more uncertainty.

MR. STELLO: You clearly are going to add more uncertainty in the calculation because you are asking the

technology to be able to discriminate something that I don't think it is yet ready to discriminate.

MR. MURLEY: And it is heavily dependent on human action.

MR. KERR: The implication, I thought, of Mr. Denton's message was that reactors that complied with his criterion of 10 to the minus 5 for release of large amounts of fission products would be safer than existing reactors. When he says, I am not comfortable with 10 to the minus 4 but I would be comfortable with 10 to the minus 5, either he has a gut feeling or somebody has done some calculations which say that the use of that criterion would produce or would at least convince me that reactors are safer than they are. Now, I must be missing some point of logic.

MR. STELLO: You know, we're talking about what it is that Mr. Denton thinks, and I apologize. He promised me he'd be here today.

MR. KERR: He has not discussed this with anyone else?

MR. STELLO: I have not had any discussion that would allow me to give an answer which I feel comfortable represents his thinking. I'll give you what I view the difference to be, as I see it.

The 10 to the minus 4 are calculations that we have today that give us the information we have been measuring

against 10 to the minus 4.

MR. KERR: Now, the other thing he could be saying -- and I just hate to believe this -- is that if I use a 10 to the minus 5, then the likelihood that I will reach that is smaller, and therefore, things look better without my changing anything. But he's not saying that, surely, is he?

MR. STELLO: I doubt it.

MR. SPIES: No, I don't think he's saying that.

MR. KERR: Well, then, he must be saying that he wants some changes, and that the changes will demonstrably make reactors safer.

MR. STELLO: Well, let me give you my opinion. The results that we have today that generally are in the 10 to the minus 4 ballpark, if you look at whether the judgments are or aren't correct that the difference between that number and a full-scale melt is a factor of 2 to 10; if the factor of 2 is correct, then if you adopt 10 to the minus 5 you will be making reactors five times safer. If it turns out it were a factor of 10, there'd be no change.

So the answer, as I see it, is if we had infinite knowledge we would know to what extent you would improve safety by the changed language. But I think the intent that he had in mind, in my view, was that there would be an improvement in safety by adopting this rephrased language; that is, reactors would be made safer, that not only would the

likelihood of a melt-through of the reactor vessel be lower, but the likelihood of severe core damage also would be lower. The net intent being that if you applied this and reactors met this new standard, that there would be some improvement.

Based on what I've heard, I'd say it's between a factor of 0 and 5, in terms of improvement.

MR. WARD: One and 5, yes.

MR. STELLO: One and five, right, I'm sorry.

MR. REMICK: Dave, I have a procedural question. I have a comment I would like to make on this 10 to the minus 4 question, but I don't know if you want those now or if you want Themis to go ahead and make his presentation and we can come back. So I ask just from the standpoint of efficiency whether you want me to do it now or if you prefer I hold off.

MR. WARD: Why don't you go ahead and make it now. I think we're in a general discussion.

MR. REMICK: Okay. I think it might be helpful to look at this historically. When the safety goal was developed it was developed from the standpoint that the Commission's responsibility is protecting health and safety. And therefore, the primary goals were public risk guidelines. The idea being that the Commission ought to be able to tell the public that if you live in the vicinity of one of these plants, it's our best judgment that you will not be subjected to an increase of deaths from an accident at that facility,

prompt fatality, greater than one chance in 1000, an incremental increase.

And you should not suffer a chance of a cancer fatality sometime in your lifetime greater than one part of 1000, and that was the concept. And then people said, yeah, but you're going to have people come in and say okay, we'll meet that by building the perfect containment vessel. We will worry about ECCS and all that type of thing. So they said okay, well then you should anchor that and make sure people don't put all their eggs in the mitigation basket, but that they put some in prevention. There ought to be some guideline, and it was called a secondary guideline, and I think it still is, that gives people a feeling for where they should be on the prevention to give them some order of magnitude.

Now, I must admit the Steering group is coming out and saying that they think that should be elevated to a full-fledged guideline. So I just wanted to give that historical perspective on 10 to the minus 4.

The other thing is there was a phase in going from NUREG-0880 to Rev. 1 in which this question of what do you mean by large-scale core melt was raised, and it was proposed, and actually it was proposed to the Commission if I recall that this be changed to a loss of core cooling capability to better describe what the intent was, but that was not

accepted. Large scale core melt stayed in.

But I thought perhaps this historical perspective might help in this discussion. If it is still a secondary guideline I'm not sure it's so important. If it's what the Steering group recommends, that it be a primary guideline, then whether it's 10 to the minus 4 or 10 to the minus 5 or some other number.

Now, there's one other thing I'd like to say, I guess, while I have the microphone. There's a question in mind if it goes to something like 10 to the minus 5, has the staff thought of whether this would require public comment to get full-fledged comment on that point?

MR. STELLO: Well, what I had hoped was that we would get some sort of discussion as we are having today now on the issue, and try to come out where the staff ought to be in terms of this, and then the question will have to be raised: if you do change, is it important to again provide an opportunity for public comment. But I don't think you want to deal with that issue until we've decided it. I think it's a good question.

MR. REMICK: No, my only point is perhaps it would be good to have your views on that.

MR. STELLO: Yes, I agree. I also think that you have raised the point -- and I was going to come back to it later -- if you are really interested in changing a plant

performance in terms of safety, the safety is really the consequence of the business. You could have changed the full order of magnitude at the other end, which was quite arbitrarily chosen; there was nothing particularly scientific about a 10th of a percent, and made it a 100th of a percent, and then you would have been able to say you have clearly set as a goal an even tighter standard as that goal for public health and safety because that's the business we're in, protecting the health and safety of the public.

That is the other end of this argument. But there is the middle ground that we haven't talked about that I think we need to come to, and that's part of the defense in depth. It's trying to achieve the balance. And I think that's part of also what was driving Harold, is that middle part of this business that says, you know, -- and Themis made the point earlier -- let's not try to rely too much on the containment performance and the dispersal of the fission products. We have so many other pieces of information that are coming together that make this even more difficult.

If the source term results, based on what they look like, -- they also have suggested that the consequences are indeed also lower. So in terms of the actual safety of the public, the research that we have done suggests that plants indeed are safer, which is also the conclusion I think of the American Physical Society than we had previously believed them

to be.

This suggests, again, making them, say, for we are clearly coming to the ultimate question of when have you really achieved "safe enough"? If "safe enough" means another factor of 10, there's a variety of ways to say that, and this is one of them. There are others, however.

MR. WARD: Well, I'm glad you brought that up, because I think we have lost sight of the original idea of the safety goal, as you stated it, because what is driving Mr. Speis here to want to lower the number is not a concern about causing more offsite cancers or accidental fatalities, but that he doesn't like the 45 percent chance of a coremelt, whether or not it causes offsite fatalities.

MR. REMICK: And I'm not sure we should be referring to 45 percent. If it met the safety goal, it would be 20 percent. Three times 10 to the -4 is based on a current set of PRAs, an average of those.

MR. LEWIS: It would actually be 18 percent.

MR. WARD: But it's got nothing to do with offsite effects, except, of course --

MR. SPEIS: This number got notoriety because we addressed the letter to some Congressman, so I put it there for historical continuity and perspective.

MR. WARD: Yes, a lot of history today.

MR. OKRENT: Well, there are several points that I'd

like to make.

In the first place, Mr. Denton does make the point that he is not prepared to be as confident about the predictions on containment performance as the PRAs are claiming. So that's one reason for his being uneasy.

Let me finish, Themis. I have several points I want to make -- uneasy with this coremelt, and I am equally uneasy, and, in fact -- well, so far as history is concerned, just like in the Judeo-Christian tradition, there is more than one time when history began.

In fact, before NUREG-0880 began, there was an ACRS document which proposed some trial criteria. There was a mean large-scale coremelt guideline in it of 10 to the -4 per reactor year. It clearly meant melt through the reactor vessel, because there was a lesser hazard stated in it, which showed a rather modest release of fission products involved from the core -- in other words, a TMI type sort of thing -- and in fact, it suggested that that might be acceptable at a higher frequency.

In the ACRS proposal, there was an effort to achieve "defense in depth" by asking for both a mean coremelt frequency and a containment performance guideline, given a large-scale coremelt, and that is a part that we are still trying to get the Staff to pursue.

I don't think it's practical to go to 10 to the -6

on coremelt; in fact, my understanding of what transpired at Sizewell B is, the designers there originally thought, we will try to show coremelt frequency at 10 to the -6 and decided they'd better back off to 10 to the -5, and I think in Italy as well, to get some credit for containment, to get the 10 to the -6, the serious release outside containment.

So I would be quite skeptical, let me put it this way, of anyone trying to show me 10 to the -6, and I don't think I would take on the job of designing the reactor that gave you 10 to the -6 mean coremelt frequency with -- all right, mean -- and it asked them to give it with some confidence, it would be, you know, obviously even worse.

The reason why I asked the original question about 10 to the -5 mean is, I guess I, myself, am at this moment skeptical that you will find a factor of 10 universally, reactor to reactor, between the likelihood of an interrupted coremelt accident and the large-scale coremelt.

It may occur in some reactors that the dominant event is one where, when you look at it with certain modest things or whatever it is, you can decide you can restore something in the time allowed -- it wasn't permitted in that particular PRA, but because -- and you alluded to this yourself -- that because there are a lot of different kinds of scenarios and they vary from reactor to reactor, I wouldn't, myself -- I'd be reluctant to count on it, and one of the

reasons why I then asked about the 10 to the -5 is, I think it would take probably substantial changes -- I don't know what kind -- and even assuring a mean, 10 to the -4 -- and let's leave out small reactors like Big Rock -- it may not be straightforward, let me put it that way.

MR. STELLO: I would like to make a point now, because I think it's important to make right after Dave made that comment.

I don't believe I've heard Harold say what you said. I read his words again in terms of lack of confidence in the containment. I don't think it says that.

I think that the sentence says -- and maybe we ought to read what he said, because he's not here, rather than trying to put words in his mouth that he may have to write letters to try to get out of them -- let's read what he said.

MR. LEWIS: Wasn't he going to be here?

MR. STELLO: He told me he would be here, yes, and he is not.

"The accident prevention guideline is a quantitative corollary of our "defense in depth" concept and should include a margin to allow for the imperfection in the current methods of predicting coremelts, fission product behavior, and containment performance."

That's what he says, and I don't believe that's what you said he said, so I think we ought to rely on the written

words.

MR. OKRENT: I am perfectly willing to stand with what you read, because it may be --

MR. STELLO: Okay. One more point, since I already rudely interrupted, and I apologize.

If you talk 10 to the -5, Dave, you ought to understand that there's roughly a half a dozen to a dozen scenarios which make that up, which means you are talking about 10 to the minus -6 scenarios that you are trying to deal with, and you are also getting pretty far out in the spectrum in light of the uncertainties, and you may even spill into the 10 to the -7 scenarios.

And you are asking this technology to do, I think, maybe a touch or two more than really it's prepared to do.

MR. WARD: Well, I am concerned about the same question.

MR. WARD: Carson?

MR. MARK: I wanted to add one thing to Forrest's history. The secondary criteria, they didn't make enough of it in the Steering Group report, but the real fact is that that's the only criterion on which they can lay their hands and make any conceivable application. It isn't a secondary criterion; it's the only operable one.

MR. LEWIS: I just want to comment on Vic's rereading of Harold's basis and remind us that, if I remember

the words correctly, he said to allow for imperfections in the calculation. That is a basically conservative position. It doesn't say allow for bias in the calculations, but to allow for imperfections. So he wants to move a criterion down which should be realistically calculated to provide conservatism against imperfections in other calculations. And that's a generic disease, that these things have to be done realistically.

And I read Harold's letter as betraying the routine, every where around here, conservative bias applied to realistic criteria.

MR. WARD: Themis, did you have anything else now?

MR. SPEIS: I have -- this is really the main thing, you know.

MR. WARD: Okay. Well, just go ahead. I think we sort of interrupted you.

MR. SPEIS: I have a second Vu-graph now.

[Slide.]

Some of the other comments that you have read -- and Frank explained them very well yesterday, so I can't add more to what he said -- we do agree with the Steering Committee's recommendation that averted onsite losses be included. As Frank said yesterday, since this issue has been discussed very extensively and there are all kinds of ideas and nuances and statements, we suggested that, you know, maybe a separate

document should be put together and articulate in great detail all the pros and cons, whether it's economic regulation or safety regulation or under or over-regulation. That's basically what he said.

The other thing we are uneasy with to some extent are the implementation guidelines. I have a separate Vu-graph --

MR. KERR: May I ask a question, since I think we are going to a different topic?

You say you agreed to the safety goal Steering Group's inclusion of onsite. Suppose as an alternative, one said, "All reactors must meet the 10 to the -4 guideline," period, and you would use ALARA to improve on that.

Would you feel the same way about including onsite costs? What I'm asking, is the onsite cost simply a lever to get to the 10 to the -4, which is where you think things should be, or is it a matter of principle that you think that the onsite costs ought to be considered.

MR. SPEIS: Well, I think as a matter of principle, all costs should be displayed.

MR. KERR: Now wait. Let's not talk about "all costs," because we'll never be able to -- we can talk about onsite costs, maybe.

MR. SPEIS: Well, all important costs, all important, as someone said yesterday. And when we talk about

onsite cost, what are we talking about? Averted radiological cleanup and replacement of power?

MR. KERR: Okay. So it's a matter of principle, and not just an effort to get plants to the 10 to the -4, say? I'm not trying to be critical. I'm just trying to understand how the Staff reached the conclusion, because, you know, it helps me to consider it, if I understand it.

MR. SPEIS: Well, everybody always, you know -- their motives are not 100 percent pure. Maybe there is a 1 percent impurity in Harold's motive. If I get this, then I can --

MR. KERR: I don't see anything impure about wanting to have a lever to accomplish something you think should be accomplished.

MR. SPEIS: I think on this one, I will ask the specific question of him to make sure that I don't speak for him. But I think it's a matter of principle, but as I say, maybe that 1 percent impurity is always important. I don't know.

MR. KERR: Thank you.

[Slide.]

MR. SIESS: If it were a matter of principle, would you have any problem if it showed you that 10 to the -3 was okay?

MR. SPEIS: Excuse me?

MR. SIESS: If it were simply a matter of principle, would it give you a problem if the cost/benefit analysis showed you that 10 to the -3 was okay?

MR. OKRENT: Excuse me, Chet. I have to ask a question of you first, because if they say there's a criterion of 10 to the -5 that should be met, then I don't see how cost/benefit can show that something less than that is okay.

MR. SIESS: I agree. But I don't know why you need to make cost/benefit at all, if you've got 10 to the -5 as a limit.

MR. OKRENT: But you see Harold is proposing a speed limit or whatever you want to call it of 10 to the -5. So if that is adopted, then what --

MR. SIESS: Well, by "principle," maybe he meant that if you are going to use cost/benefit, then you should include all costs and all benefits. But I was assuming that there was a matter cf principle involved in using cost/benefit it. Maybe I mistook it.

MR. KERR: Well, your comment puzzled me a bit, because I thought, given that, let's suppose we had a 10 to the -4 goal. I thought that cost/benefit was to be used to determine whether plants that did not meet the 10 to the -4 must make changes to meet it, and you made that decision based on a cost/benefit analysis, is that not correct?

MR. OKRENT: Well, there are different ways in which

one can use cost/benefit analysis, and one is -- let's talk about 10 to the -4, and let's say a plant doesn't meet it, and you can say, "I can't find any cost-effective way to get from 10 to the -3 to 10 to the -4; therefore, I won't make any changes." That could be one way to use cost/benefit analysis.

Another is to say, "10 to the -4 is our design objective, and we intend to meet it, if it's practical." Then you find the most cost-effective way, if you will, of getting up to 10 to the -4, but --

MR. KERR: But regulatory people don't have to find the most cost-effective way. That's left up to the designer.

MR. OKRENT: But it may still have a ratio which is -- where the cost is larger than the benefit. But you said, "This is a goal that we intend to meet, to the extent one is able to calculate it and so forth."

MR. ROWSOME: I wanted to insert a couple of points.

MR. KERR: I yield to the Senator from --

[Laughter.]

MR. ROWSOME: Cost/benefit already has a well-established role in this agency in at least three arenas -- in the backfit policy, in the regulatory analysis of new generic reactor safety standards, and in NEPA. The safety goal arena has been chosen as the arena in which the agency will codify how it implements that. But it has a policy existence quite apart from implementing the quantitative

design objectives in the safety goals.

No one in the Staff has entertained the idea of making any coremelt frequency guideline a requirement that all plants must meet, so that some of these comments -- the discussion has gotten rather academic and far afield.

MR. KERR: As Mr. Lewis pointed out this morning, I don't consider "academic" a pejorative term.

[Laughter.]

MR. ROWSOME: I don't either. I am just pointing out that we talking about applications of cost/benefit in arenas other than just implementation of the quantitative design objectives.

MR. WARD: Well, I heard that. But for some reason, we seem to have the same fundamental difference of opinion that we had yesterday between -- I think it's between Bill and Dave, and I am amazed.

MR. KERR: I never disagree with Dave.

MR. OKRENT: And I wasn't offering an opinion. I was trying to say, I have seen cost/benefit used in different ways.

MR. WARD: Yes, but maybe we are about to hear how NRR proposes to use it. Is that what we are about to hear, Themis?

MR. SPEIS: Well, I was going to bring the point that we made, illustrated graphically. It will take a minute

or less than a minute.

MR. WARD: Okay.

MR. SPEIS: You are all familiar with the implementation to exclude consideration of regulatory actions between 10 to the -3 and 3 times 10 to the -5, unless -- but you check the mortality goals, of course, and if met, then you check whether a specific sequence exceeds 10 to the -5.

I was going to make the point that, you know, when PRAs are done and we have to review them, there is always the problem that, you know, the analysts, they can play games with the sequences. They can combine them this way or that way. And I have an example at the bottom of the page here that maybe somebody will come up with 100 sequences, and all of them will be slightly below the 10 to the -5f, and when you put them together, you come up with 9 times 10 to the -4, which will be slightly below this 10 to the -3.

So this goes to the heart of uncertainties in how one performs this calculation and what the reviewers accept and don't accept. So our point is that those things are not very well articulated in this implementation. But maybe we have to do more by experience or by some other vehicle.

MR. SHEWMON: Back to Bill's comment, or the use of cost-benefit, I know I read some place, though whose position it was, I don't know, that there was a window some place around 10 to the minus 4 to 10 to the minus 3 in which

cost-benefit would be used, and above that you fixed it no matter what, more probable, less probable, something else.

Is that still --

MR. SPEIS: If a specific sequence has an arrival rate of more than 10 to the minus 5, then you look it over and make sure that there isn't something to be done.

MR. KERR: Is the window that Prof. Shewmon talks about between 3 times 10 to the minus 5 and 10 to the minus 3, that's the window I believe to which you refer?

MR. SHEWMON: And it was only that window that cost-benefit would be used. Is that still the policy or proposal? Whose proposal was it?

MR. STELLO: Let me say what it is that we are proposing as a rule, and the rule before the Commission right now, basically says if ever you conclude that a plant is not adequately safe, that is that you can reach the conclusion that there is no risk to the public health and safety -- if you reach that conclusion you don't do cost-benefit. You impose it.

If it is a matter of compliance, the proposed rule that is before the Commission says you don't do it.

What Themi has in front of you is the part of the implementation program for safety goals and how to try to use that as a way in which to develop a more meaningful process to try to deal with some of these questions.

Perhaps in 10 years we will be able to then change the language I just gave you into a more quantitative form. The language I gave you is the language of the regulations. That's the requirement.

Let me remind you again, let's not continue to characterize these safety goals as requirements or as standards. They are not meant to be that. They are not going to be used for that purpose.

MR. SHEWMON: Well, if I used safety goal, I misspoke.

MR. STELLO: Well, I gave you what the rules are, and the rule that is now proposed and before the Commission. As I recall, that is in fact --

MR. KERR: But it seemed to me there seemed to be exceptions to what you said, and perhaps you can explain why there are no exceptions. There was a branch technical position, for example, that had to do with behavior of auxiliary feedwater systems. I have been told in the past that branch technical positions are not rules. One could satisfy all the rules without conforming, therefore, with this branch technical position and yet I think new reactors at least are being required to conform to that branch technical position.

So when you tell me that as long as they meet all the rules, nothing more has to be done, I am puzzled.

MR. STELLO: I have just given you what is in fact the requirement. The way in which we impose requirements takes on a new meaning when you talk Staff positions. What are all the Staff positions?

Now the Staff positions are those things that are embodied in the standard review plan, and we tell a plant now if you wish to get a license, show us how you conform to the standard review plan.

Now listen to the next sentence. It's very important.

If you choose not to follow the standard review plan, you wish to deviate, tell us where you wish to deviate and then tell us how you meet the basic underlying regulation.

MR. KERR: Vic, we're playing games.

MR. STELLO: No, we're not. It's very, very important. What the requirements of the Commission -- they're what they are. They're our only legally binding requirements and obligations of the Licensee.

The way we do our review process is a standard review plan, which you've seen, is a very thick document, and we basically tell the Licensee do it this way, and if you don't, you have got to be able to justify why not.

MR. KERR: And the likelihood of his being able to do that is what? 10 to the minus 6, 10 to the minus 7, and he knows it?

MR. STELLO: No, I think the Beaver Valley contest has shown that of those areas where he raised the contest -- I think, if my memory serves me right now, about 12 of the 16, the Licensee, we agreed with him and he departed from the standard review plan.

Am I right, about 12 out of 16?

So the answer to your question, the most recent example I have is where the contest was raised, 12 of the 16, and they all aren't decided yet, the Licensee departed from the standard review plan and the Staff accepted it.

So the real world, if we can make it ever work sensibly and manage it correctly, it ought to allow for that. To allow to deviate from these practices when there is a case made to do that.

MR. LEWIS: In that case, did the Staff accept it cheerfully?

MR. STELLO: I never asked. I don't know.

MR. LEWIS: It took a long time.

MR. STELLO: It did indeed, because, you know, we began a process, it was the beginning of a new way in which to manage backfit, and it was just beginning. That process has only been in place about a year.

Themi, I would suggest that in light of the opening comment that we are not going to discuss mean and median, and I plead with you do not, so the next slide I would not use.

MR. SPEIS: I have brought an additional expert with me to help in the discussion, Mr. Thadani.

MR. STELLO: We are not going to discuss it.

MR. KERR: I thought we had a caucus just before lunch and decided that we weren't going to talk about it any more today at least, didn't we?

MR. WARD: Yes.

Let's see, Dave, we've got, you know, 45 minutes more.

MR. OKRENT: Is there a representative of Mr. Minogue here?

MR. STELLO: We made a call to Mr. Minogue's office. I don't know whether his representative is here yet. I guess not. He is on his way.

But let me, before you move to that, if I could ask Themi one question.

MR. SPEIS: I want to make one more point. Harold told me to stress to the committee here that he will continue to write his views to Dircks and to the Commission and to the other offices and to the ACRS, even before an integral view has been packaged by the EDO, because, you know, even though he feels that maybe his letter was misquoted and he received larger notoriety, but he still told me to relate to you that's the way he will always do business.

He wants to express his views because maybe in some

circles the letter was mischaracterized. I am not referring to the ACRS.

MR. STELLO: In order to help, Themi, let me ask you a question. Based on my understanding of discussions with Harold, it is my understanding with Harold that one issue he thought needed more discussion and ventilation is the 10 to the minus 5 issue, and given that that got the proper ventilation, that he was prepared to support the steering committee report; is that correct?

MR. SPEIS: Yes.

MR. STELLO: Okay. Thank you.

MR. OKRENT: Well, anyway, I must say I hope the Commissioners and the EDO encourage their senior staff to express their strongly held or important opinions, and that there isn't anything done that would diminish this expression of opinion.

MR. STELLO: I agree with you, with the exception of this particular sequence that we have had with the ACRS, where we tried very much not only to get Harold to put down his comments, but to bring him down and discuss it with you.

I don't know that this experience has been one which would warm the cockles of one's heart to do it repeatedly.

MR. KERR: Vic, I must -- I guess --

MR. WARD: Well, did he have a bad experience at ACRS? What are you talking about?

MR. KERR: Yeah, what's the problem?

MR. STELLO: I sat in front of -- well, I guess behind you-all and I listened to the characterization of the Staff, in the way in which the safety goal issue has been handled, and some of the comments made about what the Staff has done and where it is, and I don't think they were complimentary.

MR. KERR: Oh, I thought you were talking about Harold's letter.

MR. STELLO: Harold's letter being one part of it. That's part of the controversy.

MR. KERR: I did not hear a criticism of Harold's letter. Lewis said something about being complimentary.

MR. LEWIS: Well, this is a serious issue.

MR. STELLO: I think it is. I think we are coming down here to try to get discussion and advice, and we weren't finished by a long shot, and I think some of the discussion up there, you know, left me haunted. But I too will not hesitate or shy away from coming back to the committee when we are not decided.

MR. LEWIS: No, I'm not talking about you, but I hope you are not telling us that there are elements of the Staff that will not speak to ACRS unless they get favorable reviews from ACRS.

MR. STELLO: Oh, I don't think anybody on the Staff

has that view. All of us have been around too long.

MR. LEWIS: Well, wait a minute, now. You mean you didn't mean what you just said?

MR. STELLO: I mean what I just said.

MR. WARD: Okay. Well, let's go on to the next speaker.

MR. OKRENT: Okay. I also hope that the EDO will not, on important issues, take steps that would diminish the kind of technical interchange that has been possible, in fact, in large part because of the presence of serious thoughts by various members of the senior staff.

MR. STELLO: Dave, let me make it very clear. The EDO sought out those views. He asked for them in writing, and he wanted them, because he thinks it's very important to get those views from his senior managers, and he thinks it's important to have adequate discussion and ventilation of them before he decides issues.

MR. OKRENT: That's fine.

MR. STELLO: So let's make it absolutely clear on the record that not only does he support and does it, and he will continue to, I assure you.

MR. OKRENT: Well, we are in perfect agreement.

I have a small question for Mr. Rowsome.

MR. KERR: While we are on this issue, Vic, I don't know what significance this statement will have, but I want to

make it.

Just as the ACRS is not of one mind on many issues, I don't think you should take comments by individual members of the ACRS as representing ACRS views.

MR. STELLO: I know better than to do that also.

MR. KERR: Okay.

MR. OKRENT: Frank, when you were standing, you said something I didn't quite understand about 10 to the minus 5, and let me pose an iffy question to you:

If in fact the Commission did adopt the suggested alternative in Harold Denton's memo, how would you envisage its -- well, I won't say implementation -- pursuit, since it's an objective?

MR. ROWSOME: Let me start out by answering that by stepping back to the discussion we had about half an hour ago about how accurately you can measure that, or whether you can measure it.

The credit given in PRA for corrective action in repairing faulted systems is now calibrated to what we believe to be the time of core uncovery. If we calibrated it instead to the time we believe a core meltthrough would occur, that estimate would be no more nor less accurate than it is now.

In fact, there are a lot of sequences in which -- theoretically possible sequences in which equipment is nonfunctional by virtue of cognitive errors by operators, or

the system is operable, but the operators are unaware that it is not functioning, perhaps because of faulted control instrumentation and the like, so that in fact - and those sequences are normally not considered or not effectively predicted in PRA.

So I think the case could be made that one could be more accurate in predicting the frequency of vessel meltthrough than one has been in the past in PRA, in predicting the frequency of severe core damage.

Now as to implementation, there are many ways one could contemplate implementing it. Let me set you an example. The FAA sets a target accident frequency for a particular make and model of aircraft on the basis that there is a fairly small percentage chance, less than a 45 percent chance, that any one of those planes will have a serious accident in its whole service life.

They know they cannot in fact realize that. The frequency of human error giving rise to aircraft accidents is about two orders of magnitude more frequent than that goal. Nonetheless, it is a goal, and where they do know how to implement it, they do.

So we could, without question, on some sequences, on some classes of sequences, some classes of vulnerabilities, make intelligent use of that goal. Where there are particular large uncertainties, as is frequently the case with external

events, one simply doesn't know whether one has met that goal or not.

I do not anticipate and would not recommend that the Staff go ultraconservative and start moving heaven and earth trying to beat down those frequencies. I don't see the Staff doing it, and I don't think it's necessary or appropriate.

On the other hand, I think it does make good sense to target a frequency that we can really be comfortable with, and that does set as a goal some improvement on where the historical experience would suggest we are today.

MR. WARD: Let's see, Frank. You said there was not -- you didn't think there was a difference in the uncertainty in the ability to recover before the point of core meltthrough and the uncertainty and ability to recover before the point of core uncovery? Is that what you said?

MR. ROWSOME: Well, obviously there is a little phenomenological uncertainty having to do with how long a core has to be uncovered before it's likely to go through the bottom of the vessel. That will likely be uncertain to within 10 or 20 minutes on the basis of the way we can do phenomenological analysis now.

MR. WARD: But I thought there were important questions about adding water to an overheated core. In the case where the core is just beginning to uncover, I guess it is fairly clear that simply adding water at the maximum rate

is obviously the best thing to do. With a core that is severely overheated and beginning to melt, I thought there were some concerns that that may not be the optimum thing to do.

MR. ROWSOME: Whether those would ever appear as important in the overall frequency profile of the plant depends on the nature of the sequences. There are some sequences in which you know once you have entered them, you are intrinsically beyond the point of no return. Vessel rupture, gross interfacing system LOCA, sequences for which there is no potentially operable system that could turn around core damage. And so those problems don't enter at all.

There are other sequences in which you know you have some core cooling throughout. The sequence by definition entails 50 gpm or 100 gpm or something like that into the reactor vessel, and there may be some marginal questions on some of those depending on exactly what the flow rate is, whether you might go through the vessel or not, but those would be a distinct minority sequence.

That kind of sequence has not proved to be important in PRAs in the past. Usually risk is governed by sequences in which all core cooling has failed outright. But there are some borderline sequences where whether in fact you go through the vessel at all regardless of operator action is in doubt, sure.

But I think on balance for most sequences, you have a disabled but theoretically capable system, and you are trying to estimate the likelihood that operators will manage to jury-rig them into operation, even though they did not start on demand for some reason or failed in service.

And the uncertainties associated with that are typically orders of magnitude. And I think that overwhelms the phenomenological uncertainties involved in this in practical applications.

MR. KERR: Frank, you said something in your earlier comments which I interpreted to mean that you thought all existing reactors or the population of existing reactors needed to achieve a significant improvement in safety, but I wasn't sure that was what you meant.

MR. ROWSOME: I said I thought it would be sensible to set goals that implied improvement. I do not, as I have said to this committee before, believe that commercial nuclear power plants on balance pose undue risk to public health and safety, and I continue to feel that they do not.

MR. STELLO: I guess I understood his comment in light of the airline analogy is that we clearly ought to set the goals beyond where we think we are going to achieve them, just to have them as goals, as aiming points, rather than --

MR. KERR: I was asking for clarification because I wasn't sure I understood him. If that is what he is saying, I

understand --

MR. STELLO: That is what I was going to ask. Is that what you intended?

MR. ROWSOME: Yes.

MR. WARD: Well, we heard these described yesterday as aspirational goals, and I thought the response yesterday was that clearly the goal we are talking about here or have been talking about for two years is not an aspirational goal, it is a goal that we expect the plant to realize. So I think I just heard something different from that.

MR. KERR: I think I just heard something different, too, but at least it was clearly stated and understandable.

MR. WARD: That's right. Okay.

MR. STELLO: I don't know that I ever remember the goals ever being characterized more than targets or aiming points rather than in any sense a standard. I will say, though, that I think we ought to recognize that once we come out with this safety goal, should we ever come out with one, that there will be a significant force that will develop that will want or move to have all reactors meet whatever these goals are, exactly, at least.

MR. LEWIS: Didn't the Commission two years ago when it first started make an explicit statement that we do not expect every plant to achieve these goals?

MR. STELLO: That's correct.

MR. LEWIS: I am not disagreeing with your prediction.

MR. WARD: Okay. Anything else? Dave, do you have anything else?

MR. OKRENT: Well, we have a choice. I guess at the moment there are no further comments from the Staff; is that correct?

MR. STELLO: Well, I was hoping that we could continue to find a way to talk longer and that our researcher would be here. He indicated he was on his way.

MR. WARD: Well, it looks like they are not going to be. We are already behind our agenda time.

MR. OKRENT: We are not behind the allowable agenda.

MR. STELLO: He is here now.

MR. OKRENT: Well, why don't we then hear from Research and Denton. Let's see. When did you reconvene the meeting, a quarter of?

MR. WARD: Yes. You have got another half-hour. It owuld be nice to make up a little bit.

MR. OKRENT: We can make up some time. I would rather make up some time now and use it when we are going to review the letter tomorrow or whenever it is, next up.

Frank Gillespie is in Research. He is the Head of the Division of Risk Analysis and Reactor Operations, or something like that.

MR. WARD: And Human Factors.

MR. GILLESPIE: I apologize. We thought we stated our position yesterday, and inadvertently, due to another task force I'm on, I didn't come down today.

The Research position is fundamentally in support of the group's report. The one comment that we wrote in in May on the group's report has been, at least in substance in the reading of the new paper, our concern on defense in depth has partially or maybe totally -- because I only got the paper yesterday -- been addressed in the new paper that has been developed by Vic's group.

MR. OKRENT: Which? We don't have the benefit of it, or at least I don't.

MR. GILLESPIE: Oh, I got it from here.

MR. STELLO: Excuse me. Yes, you do.

MR. GILLESPIE: The draft Commission paper.

MR. STELLO: The one that we referred to as the draft is the one that he is talking about, which I assure you he has.

MR. GILLESPIE: Yes, you had it much longer than I did. It's dated June 19th.

MR. OKRENT: Oh, I apologize.

MR. STELLO: We accept.

MR. GILLESPIE: I only saw it yesterday, which reacted to some of our concerns. Our concern was -- I think in

our letter we said this. We are not concerned with cost-benefit. We are more concerned with the idea that defense in depth not be lost, and we fully support that. I think in the committee's report in the examples that we used in there, there was some emphasis put on the case studies on the fact that you really had to consider the uncertainties in some of those case studies to have them fall into a positive answer under the safety goal proposal.

I think you really have to consider that, and that is where we were coming from yesterday. So our concern was one that you don't want to lose the ability to make things better just because a plant meets 10 to the minus 4. There may be safety fixes which really are safety effective below that.

MR. LEWIS: Could I just understand that comment? You surely don't mean that you don't want to not consider improvements in safety when the plant is quite safe. You said almost that, and I think you didn't mean it. This is the open endedness of safety improvement that I'm concerned with.

MR. GILLESPIE: Yes, and this is what I said yesterday. I don't have in mind a structure on how to put an entirely closed end on it, but indeed the safety goal or the way the paper, as I understand it, reads now from quickly looking at it, you know, it identifies this is something that would be used in parallel with the normal staff the way we

have done it in the past. It would be used as another yardstick.

MR. WARD: Do you just mean ALARA? Is that all you are saying?

MR. GILLESPIE: No. I don't know if I am saying ALARA in the terms that it is written up. I guess I am saying ALARA in the sense that if there is a fix -- and someone asked yesterday about do you have a list of fixes. No, I don't have a list of fixes in mind. If there is a fix whose significant safety benefit could be shown, even though the plant met a particular core melt frequency safety goal, the policy should have enough flexibility to allow the Staff to show that significant increment in safety is, in fact, worth it; and I think the paper now says that.

MR. SIESS: Is it always worth it, Frank? Is there any point below which you wouldn't go even if it were cost-beneficial?

MR. GILLESPIE: Well, if it is that diminimus, I don't think you could make the case that it was a significant safety fix.

MR. SIESS: I know, but there are lots of definitions of diminimus. Some people wouldn't think 10 to the minus 5 was diminimus.

MR. GILLESPIE: If there was --

MR. SIESS: So I am asking: you must have some

diminimus in mind, at least by your judgment.

MR. GILLESPIE: The diminimus would depend on the benefits you think you are going to gain from it.

MR. SIESS: Well, in that case you would keep going as long as the incremental benefit exceeded the incremental cost; right?

MR. GILLESPIE: As long as the incremental benefit exceeded the cost. But I mean cost in terms of other than necessarily just dollars and sense.

MR. SIESS: Okay, as a total incremental benefit and total incremental cost, you would keep going, and there would be no diminimus. There would be no how safe is safe enough and safe enough when it doesn't cost anymore as long as you started low enough.

MR. LEWIS: This is an important issue that is being raised.

MR. WARD: I don't understand this. This is either ALARA or unrestricted ratcheting. I haven't figured out what you are saying. Will you pick one of those?

[Laughter]

MR. GILLESPIE: What we have not done is attempted to put a diminimus quantity on. If you want to call it -- there is a point when you are not going to be able to make a case that a safety benefit is worth whatever the cost of doing it is. Now, if I don't have a specific safety benefit in mind

and therefore I don't have a specific cost of installation in mind, I can't tell you what diminimus quantity that particular fix would have to cross to do it.

MR. KERR: In your consideration of this issue, do you give some thought to the fact that when you get down below 10 to the minus 4 or 10 to the minus 5 or somewhere, you are in a region of uncertainty in which not only is the core melt frequency uncertain, but the effect that you have on it by doing something to the plant is also uncertain, and indeed, the SIGN maybe uncertain?

MR. GILLESPIE: In which case showing a definite safety benefit is going to be very difficult, so I am agreeing --

MR. KERR: Well, I agree showing they have any benefit at all --

MR. GILLESPIE: 'Is difficult.

MR. KERR: And indeed, one may have a considerable amount of concern that doing something makes things worse. It has happened, you know.

MR. GILLESPIE: Yes, it works both ways.

MR. EBERSOLE: I am going to drag out an ancient but classic case to illustrate the point. For many years there has been a flap about the 10-inch high pressure steam line that feeds the HPSI pumps at the Brown's Ferry and similar plants. If the pipe bursts on the outboard side of this, the

potential is to inject steam from a 10-inch main into the entire three-unit station and produce appalling damage effects.

The thesis that prevents that is that the valves, the isolation valves will close if such an event takes place, and even that event itself is of low probability. But little is known about the reliability of those valves to do that. The operation under dynamic loads is poor. The original test was poor, the post-operational testing and duration testing is virtually negligible, and we just simply don't know whether those valves will intercept that flow.

At one point in the history of the plant, one of these very pipes was attempted to be straightened by a water plug that raced through the system. The valves did shut. Had it not, it was an inviting place to start a three-unit catastrophe.

The cost of fixing this is not very much. You just close the valves in the first place and leave them closed, and then you open them when you need them. The cost is zilch. Yet it is not being done because of the rigidity of the ideology we have in the regulatory process.

MR. STELLO: May I try? I don't know that I am going to help, but let me use this example to illustrate that in the papers you have, we have tried to say there are safety goals that we are trying to develop, but there is a regulatory

process. I think Frank is trying to talk about both sides of these issues.

If the traditional regulatory process -- and I will use Jesse's example. Say that the wisdom is that those valves ought to be closed and you don't have a detailed comprehensive cost-benefit analysis that shows that they, in fact, ought to be closed, although I think perhaps one would show that they ought to be as well, but let me just assume that they didn't, and Frank is, I think, suggesting that we ought to be able to use the traditional regulatory wisdom to make a decision where we believe that that particular decision will substantially enhance safety, even not without the benefit of a comprehensive or a cost-benefit analysis at all, using traditional methods.

The paper does, in fact, say that that is what we are going to do. We are not supplanting the traditional methods with safety goals. Where there is such a case and it does arise in conventional wisdom of the Staff and coming to the ACRS and getting the regulatory advice, the best we know how, that says an action is an appropriate action to take, we ought to take it, even though the cost-benefit analysis may not even be available.

I think that is the point that Frank is making.

MR. GILLESPIE: Yes. It's a parallel process.

MR. SIESS: Well, that's clear enough until you

start confusing us with the cost-benefit analysis.

MR. STELLO: Did I help to straighten it out, Frank?

MR. GILLESPIE: Yes. That was it. I think if you read our comment letter, Vic articulately put one particular sentence into the paper from it which said we thought the cost-benefit issue was just kind of -- I forget the exact words, but it was taking people off the point. It was distraction.

MR. OKRENT: I will read the sentence from Minogue's memo, which I think Gillespie didn't give, which perhaps says what Minogue is thinking most directly. "For the reason cited above, I believe a straightforward approach is to overtly adopt the prevention policy and use the numerical guidelines of core melt frequency and the safety goal as a means to this end."

MR. GILLESPIE: Yes.

MR. SIESS: It doesn't mention cost-benefit.

MR. OKRENT: No. He has been talking about cost-benefit above, and he says that is a red herring in it.

MR. GILLESPIE: That happened to be one way it looked to us, that the committee had proposed attacking that problem. We are not hung up on the cost-benefit. We are hung up on the need for a philosophy that says we are still going to have the ability to go in on the prevention side of things, and the traditional approach is to do that.

MR. KERR: What does not being hung up on cost-benefit imply? Mr. Rowsome has just told me that it is a very deeply embedded part of the Commission philosophy. He didn't say hung up on it.

[Laughter.]

MR. KERR: What does it mean when you say you are not hung up on it?

MR. GILLESPIE: It means if conventional wisdom says to do it, whether you have got some goal like a plant worth fixed at $4 billion or $9 billion or $2 billion, if the conventional wisdom says that it really does look like there is a significant safety benefit to a particular plant from doing something, then we should get on with doing it.

MR. WARD: And ignore the safety goal, in that case.

MR. GILLESPIE: You consider the safety goal, but don't let it override your decision if the safety need exists; that's right. There are two parallel processes in operation.

MR. SIESS: You take off the bottom, take off the floor under the safety goal, leave the ceiling on, which is essentially what the British have done. They go to as low as practical once they get below their quantitative line.

MR. WARD: Now, when you said as practical, though, he is not talking about practical. They are saying if judgment indicates that something ought to be fixed, they are not going to worry about how much the cost --

MR. SIESS: I would have to define practical without bringing judgment in. It's either cost-benefit or judgment, and sometimes, judgment -- it usually isn't one man, and sometimes it is a collective judgment that has been arrived at after six months of arguing with the utility or the industry. So I think judgment covers it. You know, I don't like conventional wisdom. Judgment isn't wisdom.

MR. OKRENT: Well, Mr. Chairman, I think we have had a chance to hear from Mr. Gillespie, which would complete the Staff views, and since you are behind your agenda, I would propose we thank the Staff very much for helping to keep us stimulated and confused by ourselves and go on with your next topic for now.

We are supposed to come back.

MR. KERR: Would you be willing to accept a message to Mr. Stello that we really do love him?

[Laughter.]

MR. STELLO: Only if you give me a big hug and a kiss.

MR. WARD: We'll have to do that during the break. Let's take a 10-minute break.

[Recess.]

MR. WARD: The next topic is the Diablo Canyon seismic 10-year review, and I asked Dr. Siess to take over.

MR. SIESS: We don't call it the 10-year review, we

call it the long term seismic program. This requirement for the reevaluation of the seismic design or seismic design basis for Diablo Canyon was included in the license condition for Unit 1 chiefly, I believe, as a result of concerns -- to pick up on a comment we put in our letter of July 14th, 1978, which was seven years ago almost, which recommended that the seismic design of Diablo Canyon be reevaluated in about 10 years, taking into account applicable new information.

At the time we wrote that, we thought it would have been operating for a few years rather than a short time. The license condition required this reevaluation of the seismic design basis involving a number of things, starting with geology, seismo-tectonics, the magnitude of the earthquake, the ground motion of the site and earthquake engineering. All the aspects that we believed would affect the seismic design basis, including the tau value, soil structure impaction, et cetera.

The Licensee, Pacific Gas & Electric, came up with a plan and this is what it looks like. It came out in January on schedule. The Staff has undertaken a review of it.

Now the committee heard from the Licensee and their consultants at a subcommittee meeting that was held around Los Angeles on March 21. The subcommittee meeting was attended by two members, Dave Okrent and me, and seven consultants, covering the total range of programs from geology on up

through the plant.

The Licensee presented to us at that time their plan. Our consultants viewed it in general with favor. There were some compliments made and no significant concerns expressed.

The Staff at that time had just begun its review which they have since completed, and our meeting yesterday was to hear from the Staff their evaluation of the Licensee's plan and to come up with our recommendations regarding it.

In the several months between the time the plan was submitted and the time the Staff finished its review, there were meetings with the Licensee and the Staff submitted a rather long list of comments and questions and suggestions to the Licensee, all of which were responded to, and as a result of those exchanges, the Staff has concluded, I think fairly straightforwardly, that they find the program acceptable, that they think it will do what the Licensee has been asked to do.

They have suggested that it be kept flexible so that a change in emphasis is possible, so the Licensee is still working on what he calls a scoping program to attempt to narrow some of these, I believe most of the geologic aspects of this down to a level that is manageable for the three-year period. This must be completed in three years. It's a part of the licensing condition, and the clock starts running when the Staff has issued its approval.

The Staff has issued a draft evaluation. They say it is not final until they have heard from the ACRS, at which time they will either agree with us or disagree with us. I think those are the only two alternatives.

Now the subcommittee -- or first of all, the Staff has consulting help. They have a geologist, Dr. Slemmons. They have the U.S. Geological Survey. Our consultants, geological consultants, were particularly pleased that Dr. Slemmons will be working with the Staff. They have a great deal of confidence in him.

All the rest of the geologists and seismologists are working for the Licensee, and I think between the Staff and the Licensee, there is not a California seismologist left, is there? Although I think Dr. Okrent suggested one that might be considered.

MR. LEWIS: We breed them in California, you know.

MR. SIESS: Yeah, I know.

Now, this is a complex program and what I would like to do is simply tell you that with one exception, as far as the program is concerned, the Staff is satisfied that it is a workable program, that it will achieve the objectives.

Of course, this is not the end of it for the Staff. They will have continuous interaction with the Licensee.

Progress reports at six-month intervals, or reports at six-month intervals --

MR. BROCOUM: Progress reports are quarterly. This is Steve Brocoum, Geosciences Branch. And meetings at a minimum every six months.

MR. SIESS: Minimum of six months. So the Staff has found this acceptable.

The subcommittee and consultants in March found it acceptable. The subcommittee in its meeting yesterday -- and that was Jesse Ebersole, Carson Mark, Dave Okrent and me -- found the program acceptable, with one exception, and the exception is one taken by Dave Okrent to the proposal by the Licensee in satisfaction of the requirement that says PG&E shall assess the significance of the conclusions drawn from the seismic reevaluation studies utilizing a probabilistic risk analysis and deterministics studies as necessary to assure adequacy of seismic margins.

And this has been interpreted in various ways, but one possible way of looking at it is that if it turned out from all the geologic, seismologic and other studies that the original seismic design bases were adequate, those to which the plant was redesigned, then there would really be no reason to assess the significance of the conclusions, except to say that they confirmed previous findings, and that nothing would be necessary.

The Licensee has not been that optimistic and has proposed making a PRA beginning essentially now, having it

ready to use in the assessment of the significance. He has proposed not just a seismic PRA, but a full scope PRA, taking into account internal as well as external initiators which we think is useful.

But he has proposed only what has been called a Level 1 PRA.

Now as I understand it, Level 1 carries through core melt scenarios, develops a core melt -- the core melt frequencies, and what is referred to as plant damage states corresponding to each of those scenarios, whether it's an early or late or whatever, and somebody can explain that in more detail.

Level 2, I believe, goes to source term within containment.

MR. OKRENT: I think we had better have a better definition of Level 1 and Level 2.

MR. SIESS: Well, I am reading this from something that the Staff gave us at the subcommittee meeting.

MR. ISRAEL: I will go into it later.

MR. SIESS: Well, what I have seems strange because it says Level 4 goes to consequences, and that sort of left Level 3 in limbo. But Sandy will explain that to you.

The issue is simply this:

The Staff is satisfied with a Level 1 PRA as a basis for assessing the significance of whatever conclusions may

come out regarding the groundmotions. The subcommittee at the March meeting -- or at least the two people that were there -- suggested that there would be some advantage to the Licensee in going to consequences and not stopping with core melt, and the Staff said well, if they go through Level 1, we can extrapolate that to the consequences.

This was discussed at length at the meeting yesterday. In fact, it was almost the only subject of discussion. The three members of the subcommittee other than Dave agreed or decided they would agree with the Staff that Level 1 would be adequate.

Dave believes that they should go to Level 2 or 3, depending on what the proper definition is. Dave would like to see them go to consequences.

MR. OKRENT: No.

MR. SIESS: Well, okay, Dave will have a chance to say and we will figure out how to go about it.

So what I am proposing is that you accept the recommendations of the four subcommittee members as far as the program plan is concerned on matters other than the PRA level, and that you hear the story from Dave, his reasons, the Staff's reasons, the Licensee's, if you wish, why they think Level 1 is adequate, and then let the Full Committee decide whether they want to accept this, the Level 1 PRA, or want to make a recommendation for the higher level. That is the issue

I would like to put before the committee.

First I am asking you to accept our recommendation on the other aspects of the reassessment of the earthquake effect, and to hear and decide on the issue of what kind of PRA is needed in order to assess the significance of this.

Now the other members of the subcommittee, Jesse and Carson, have I spoken for you? Or you can speak for yourselves.

MR. EBERSOLE: I just thought I would mention that Level 1, as I understood it, did include consideration of containment damage as an initiator to core damage.

MR. SIESS: Oh, yes.

MR. EBERSOLE: And that's as far as it went.

MR. MARK: Well, I don't think I have anything to add to your account, Chet. There are, of course, other reasons which might appeal to the Applicant to make a more extended PRA. There is a very good chance of learning something, and being able to explain things in connection with a plant for future arguments.

I guess a PRA has never been done on Diablo Canyon, full scope.

MR. SIESS: Full scope. We did a partial seismic PRA -- well, a risk analysis, seismic risk many, many years ago, but not really what we are talking about now.

MR. MARK: Well, there are a number of things which

I'm sure he has in mind more clearly than I, like the severe accident policy and all kinds of things where a PRA might be handy to have in your briefcase or in your trailer truck, whichever they go into.

MR. SIESS: Dave, which would you think would be best? To let the Staff say why they think Level 1 is enough and then you give your arguments?

MR. OKRENT: Well, I think it would be well to have someone define in the first place what Level -- what the Staff thinks is adequate and --

MR. SIESS: What it is and why they think it's adequate?

MR. OKRENT: Yes, what it is and why they think it's adequate, and then they can define Level 2 as they interpret it and at that point we could have a discussion, because right now we don't have it.

I will wait until their --

MR. ISRAEL: Let me briefly go over the analysis process. Most of you are familiar with it, but at least to put everything into perspective. PRA analysis usually starts off with what we usually call the front end, where the plant systems are analyzed in an event tree, fault tree fashion to determine those sequences that will lead to core melt.

The sequences may have various initiators and what you are concerned with are failures of various systems so that

in effect you would end up without core cooling and you would end up with a core melt.

In the past this analysis has resulted in determining various plant damage states. Plant damage state in effect tells you something about the core condition, core melt and the containment condition. The one of particular interest in terms of the seismic analysis would be containment failure prior to core melt which would occur if, in fact, the earthquake failed the containment which would then obviously -- or maybe not so obviously -- fail the piping running into the reactor and core melt would ensure from there.

You have other plant damage states that could occur. One is core melt without any containment melting, and this usually occurs because you lose all electric power in the plants, both offsite and onsite. This can occur because of the seismic or it could occur for other reasons as well.

There are other plant damage states that occur. Event V, which you have all heard about, which may not be specific to seismic, but in this situation we have containment bypass prior to core melt. Event V is failure of the low pressure ECCS piping outside of containment so that you have a path outside of containment, with potentially subsequent core melt.

There other damage states dealing with transients and LOCAs, which basically are early core melts if you had a

large LOCA and didn't have injection, or you could have later core melts because of small breaks or transient situations where you lose core cooling and ultimately the core would fail. And the concern here is the availability of containment cooling, and this would be determined by the analysis of the plant systems in the fault tree process.

This is basically what Level 1 is. It is essentially a plant damage state analysis. That is the front end. The back end, which will differentiate between Level 2 and Level 3, the back end tries to characterize or characterizes what happens to the containment given any one of these plant damage states.

Let's take, for instance, core melt without any containment cooling as a plant damage state. You would be concerned about the split fractions as to where the potential containment failures would occur, how they would occur. You have heard about steam explosions, the alpha mode where the vessel would explode and go through the containment.

You have heard about early hydrogen burns that could fail the containment. There are late hydrogen burns that could fail the containment. There are overpressurization failures of the containment as the pressure just builds up and there is an overpressure of the containment. Or you could go to basemat melt-through or no failure at all.

So all of those are different split fractions dealin

with a given plant damage state. So for each one of the plant damage states, I could then go into various potential containment failure modes or no failure at all.

Those split fractions are usually determined under Level 2. Along with each one of those split fractions, which are usually characterized by a release category, if you will, the analysis then calculates a source term for each one of those release fractions.

So at the end of Level 2, I now have plant damage states which I now multiply by various failure mode, containment failure mode split fractions, and for each one of those I have a source term. I can now go over into Level 3, which is the consequence analysis, and the consequence analysis takes each one of those source terms for each one of those containment failure modes and determines the offsite consequences.

Our past experience in terms of offsite consequences has been that early fatalities are dominated by containment failure prior to core melt. I say this, and that is based on the spread of plant damage state frequencies that we have seen in the past. Obviously, if we had some benign plant damage state that suddenly had a frequency of 1, potentially that may take precedence.

But based on the spread of plant damage state frequencies that we have seen in the past in the various

plants we have analyzed, early fatalities are dominated by containment failure prior to core melt, and those types of sequences are basically the one that you could potentially get with a seismic event where the earthquake rocks the containment away from the auxiliary building or fails the containment somehow. It could also be Event V.

You also potentially could get early fatalities from long-term containment failure, and the way we arrive at that is usually through the station blackout sequence or what I will call core melt without any containment cooling, so that over a period of time the pressure builds up and the containment fails and that is released to the environment. You could get early fatalities from that.

There are other plant damage states, but those that seem to have containment cooling do not seem to contribute to early fatalities.

In terms of latent fatalities and person rem, the dominant contributor appears to be, from past experience, core melt without containment cooling, again, which also looks like an extended station blackout type of sequence.

There are several reasons why the Staff didn'lt require the Licensee to extend his analysis beyond Level 1, which would be the plant damage state level. The foremost reason, and this goes back a year ago and we are still in this mode, was that we are in this source term upheaval. There is

the old source term and the new source term, and we are in the middle of the breach here and this study is not going to be done for three years, and I saw now way how to indicate to the Licensee what he should be doing with the source term.

For that reason, that was one of the primary reasons for not requiring him to do the back end.

Let me tell you that in terms of the source term effort, evidently there will be a Staff report coming out within the next six months, I believe, or maybe in the next couple months, NUREG-0956, dealing with wherever the Staff thinks it is on the source term. I think within the next year or year and a half, the Staff will also come out with a report called NUREG-1150, which will be a mini-update, I think, of WASH-1400 in terms of source term, where they will analyze five surrogate plants: Surrey, Zion, Sequoyah, and a couple boilers.

MR. EBERSOLE: May I ask a question, Sandy?

MR. ISRAEL: Yes.

MR. EBERSOLE: How do you interrelate Event V to the seismic event? If you just --

MR. ISRAEL: Potentially the seismic event could take out that pipe as well.

MR. EBERSOLE: But if it's going to take that out, then it's not merely Event V, it's a host of other similar events. For instance, reactor water cleanup system. There

is an event similar to V. Or the one we were just discussing earlier, the main steam line failures to HPSI systems, which doesn't apply to Diablo.

MR. ISRAEL: What you pointed out was that I characterized that Event V obviously comes from internal events, but potentially --

MR. EBERSOLE: There are quite a few Event Vs.

MR. ISRAEL: Yes. If you fail the containment, you fail the whole thing. But you pointed out a good point. The seismic calculations they are doing for Level 1 would then give frequency for this containment failure prior to core melt, which is probably going to be a dominant factor in terms of early fatalities, and the most likely type of sequence for seismic is the -- well, the one of significance is station blackout because a large seismic event takes out offsite power, and once you start getting into the electrical area, it will wipe out a lot of other things.

Those are probably the two sequences of concern coming out of the study, and those would be identified in plant damage state.

But anyway, to get back, within about a year and a half, we should have this five-plant study using new source term. The new source term work includes considerations of new containment or whatever the containment failure modes are on the table at the time of doing the work, and there are some

new ones since WASH-1400. Also, it would come up with containment failure split fractions: you know, what fraction would go into the various bins such as hydrogen burn failure or long-term overpressurization, what have you, and they would also calculate source terms for each one of these based on whatever the technology is that they are into. And presumably the Staff would be embracing this sort of thing.

So this, then, gets to one of the next concerns as to why we didn't require a level -- anything more than a Level 1. Ordinarily when people start doing Level 2, Level 3 analyses, they are very expensive. There is a lot of computer calculation, a lot of phenomenological calculations in terms of containment loading, sensitivity studies dealing with core debris coolability and core/concrete interactions, et cetera.

These can become very expensive, and even our review of these becomes expensive. As I mentioned yesterday, we spent over $500,000 reviewing the Indian Point work several years ago.

There is an alternate and less expensive way of estimating offsite consequences, and here again, I want to stress that these are estimates, and I also want to stress that the whole purpose of this Diablo Canyon study is to investigate seismic characteristics of the site as opposed to some of the other PRA work which we have done which has been more geared to looking at containment potentials.

Certainly we have today the siting study work that was done at Sandia several years ago, which looked at every site in the country and came up with surrogate source terms. They also looked at Diablo Canyon, and people who are familiar with the back end analysis with a little clever extrapolation and interpolation of this information could come up with estimates of what the consequences would be at Diablo Canyon using whatever the front end core damage frequencies are, but would be based on old source terms.

I suspect that three years hence what we really will have, and another shortcut methodology, we will have this five-plant study, which includes Surrey and Zion, which probably encompasses the internal workings of the containment concerns, and that will provide source terms and containment failure split fractions, and I suspect that what we can do then is we will use that along with the Diablo Canyon plant damage states to come up with consequences.

MR. EBERSOLE: I wonder if you would explicitly tell me what you mean by split fraction.

MR. ISRAEL: Okay. Some people call them split fractions and sometimes they call them conditional probabilities. For instance, the alpha mode, which is the steam explosion inside the pressure vessel blowing the head off, et cetera, has a conditional probability of like 10 to the minus 4 that the containment will fail in that mode.

For a plant that has a core melt without any containment cooling, the conditional probability for long-term overpressurization failure may be like .4 or .5, and the probability that basemat melt-through may be around .3, .4 for that, and maybe .1 there would be no failure at all. So really the conditional probabilities of this plant damage, this scenario, which is in some plant damage state going into some sort of containment failure mode.

So the thing I was trying to stress is that we have an alternate way that is not detailed and less expensive for coming up with estimates -- and I stress estimates -- of what the offsite consequence is.

I suspect that at the time that the Diablo Canyon study is completed, depending what we see in terms of the plant damage state frequency may push us in one direction or another in terms of how much scratching we do in the consequence area.

A third reason that we didn't push them into doing back end analysis is that traditionally the Staff has performed these back end analyses for PRAs that have come in, and even though the licensees or the applicants have submitted back end analysis, we seem to do these audit calculations, and it is like paying twice for the same amount of work, in addition to which, at the end of all of this study, the licensee will come in and say, hey, this is what my plant

looks like, and the plant damage states will obviously identify weaknesses in terms of the seismic situation and it may suggest all by itself potential fixes.

At Indian Point there were buildings being together. I think at Zion there was concern about loss of the service water pumps. I think Indian Point 3 had trouble with the fuel oil line going to the diesel generators.

The Staff -- you heard earlier Frank Rowsome talking about the fact that the Staff is responsible for coming up with a cost-benefit analysis if, in fact, we want to backfit the plant. They say, hey, we looked at your results and we think that you ought to do X, Y, Z, and therefore we are on the hook for doing a cost-benefit analysis that would require us to p...orm an offsite consequence analysis.

The other feature that struck me was that this happens to be a low population site. There are approximately 200,000 people, roughly, within about 50 miles of the plant. The average site in the United States is about a million. Indian Point had about 17 million. So just to give you some sort of perspective on the potential societal risks that we are talking about.

There is another reason that I have that I think is quite valid, that I stress that the whole thrust of this Diablo Canyon study is dealing with the seismicity and geology and what have you of the site, and it really isn't geared to

examining -- it's not good form for resolving the ongoing source term severe accident work.

I am somewhat concerned about that because, as I mentioned yesterday, the Indian Point PRA that came in had 12 volumes. Three volumes dealt with core melt from internal events, one volume dealt with core melt from external events, one volume dealt with a summary of the whole shebang, and seven volumes dealt with containment analysis, source term analysis and consequence analysis. Somehow I think that is the wrong emphasis for the effort at hand.

Basically the summation of all those were the reasons that the Staff felt that it was acceptable to just stay with the Level 1.

MR. SIESS: Thank you. I am pretty sure those are the same reasons we heard yesterday that the three of us found compelling.

I would suggest that we hear Dave before we question.

MR. OKRENT: I must say I am not sure that you found them compelling, plus, in fact, I'm not sure you had a clear picture of the difference between what is required at Level 1 and Level 2, but I will just leave that as an aside.

MR. SIESS: I didn't change my opinion.

MR. OKRENT: First, let's look at the difference between Level 1 and Level 2. And I am not talking about Level

3.

For Level 1, as they say, they would calculate the accident states. They would not have evaluated containment failure modes, except those that resulted from the earthquake itself. They would not have evaluated the containment capability to withstand pressure and temperature, nor would they have evaluated whether this design basis, I will call it, or as-designed capability to withstand pressure and temperature in any way is weakened by a severe earthquake, because it's going to be, in their study, either the containment fails -- and I don't believe for a minute that a building is going to fail -- or some penetration fails or not. So, in fact, they're going to have a big hole in their ability to assess what I would call -- what's called the likelihood of different release categories.

Obviously, the containment capability is vital in this.

Secondly, Diablo Canyon has different configurations than Indian Point and Zion and Surrey with regard to containment cooling capability, with regard to AC power, with regard to service water, for example, and one particular way in which it will be different from these is that since, in fact, we are particularly assessing the effect of seismic, the review presumably will consider a series of more complex transients than is initiated by the different kinds of

failures or combinations of failures that you ordinarily just don't look at in an ordinary PRA, because you say that many things won't occur randomly, whereas you can lose all of your alarms and much of your control room instrumentation and a few steamlines, et cetera, et cetera, all with the one earthquake here. And it is, I believe, going to be a different problem to try to evaluate the operators' ability, for example, to restore a lost diesel generator or a lost cooling pump or so forth, as compared to what is done in the -- that you cannot just translate from the other PRAs.

So I think the Staff will be in a poor position to make estimates of release -- of the mode of containment failure and the frequency of containment failure in release categories.

Furthermore, I don't believe for a minute it should be the Staff that's making that estimate. I think the Applicant, the Licensee, in the first place, is the one who has better knowledge of the plant and so forth and will have done the PRA work that would help you somewhat in doing the Level 2 work, and it doesn't make sense to me for the Staff to do it. And I think what they do would be very -- it would be suspect, because of their lack of the kinds of things I've just indicated.

I think the basis given by the Staff as the primary basis for choosing Level 1 is just -- just doesn't hold

water. He said the source term upheaval is the one for only going to Level 1. I must say, we have been getting Level 3 PRAs from a lot of Licensees in the last few years.

MR. KERR: Level 3 or 2?

MR. OKRENT: We've been getting Level 3. That means they not only do -- I meant Level 3. In other words, if you look at Limerick and Shoreham and Zion and Indian Point and Millstone -- and I haven't seen Susquehanna -- Midland -- they been doing Level 3, which means they had to do Level 2, of course. I'm not suggesting in any way that Level 3 be done, where there's a lot of calculation of consequences, a lot of these volumes that he's been talking about, a lot of the computer time that you've been tal'ing about, but I do think it's relevant to understand the behavior of the containment, given a coremelt, and how it affects containment failure likelihood.

On the one hand, to say the source term upheaval leads us to say, "Only go to Level 1," but on the other hand, we're going to have a method of estimating all this shorthand again seems to me to be somewhat contradictory.

With regard to the expense of Level 2 added to Level 1, I think what we heard yesterday -- I can't remember the exact number -- but if the total cost of this entire study is eight to ten million, I vaguely recall, and the cost of the PRA, I assume, is two or three million -- and correct me, if

I'm wrong -- or higher, four million, I think going from a Level 1 to a Level 2 will -- let's say it is three million -- will not increase it by more than 25 percent, I think, and my guess is that it would be less than 25 percent -- the PRA. Maybe I'm wrong, but I don't think so, because they have the advantage of the work done by Bowsey & Associates and so forth for Zion and Indian Point, and the other utilities have used this in their PRAs.

And just what additional new knowledge is developed between then and whatever is the time when they start trying to evaluate the exact build-up of pressure and so forth is -- they are not going to be running experiments or developing new codes or anything like this, and the ground was broken, really, at Zion and Indian Point.

So for a variety of reasons, I think it is the Staff's position, it is the wrong one. I think, I'm sorry to say, I think my fellow subcommittee members maybe didn't appreciate the points I was trying to make --

[Laughter.]

MR. OKRENT: But I think the committee, in fact, should recommend a Level 2.

MR. SIESS: Dave, help me on something. In the session preceding this one and in some other discussions, I've gotten the impression that you thought a coremelt guideline was a very important, almost overpowering safety goal. And if

that impression is at all correct, why isn't it good enough for evaluating the seismic design basis at Diablo?

MR. OKRENT: Wait. In the first place, I have in writing made the point to the Staff that a coremelt guideline is not adequate and that when they talk about meeting a coremelt guideline without looking at the risks, that is an incomplete look. So I won't accept the original attribution, okay?

I think it's relevant to know the coremelt frequency.

MR. SIESS: The health effect guidelines are more important, then.

MR. OKRENT: No. I have in general -- and I'm not changing my position here now -- I have in general said, if we could get an estimate of containment release frequency, we have the essential information for estimating the safety of the reactor, and it's the same position I had a year ago.

MR. SIESS: Incidentally, I might point out that the additional cost is not negligible, but it is not a prohibitive cost by any means. It might have an impact on the time.

MR. OKRENT: It will have an impact on the time, if it is decided three years from now that it is needed. If it is decided early on, it can be -- much of the work can be done concurrently.

MR. SIESS: Unless the source term changes in three

years.

MR. OKRENT: No. But again, they would make some estimate of source term with uncertainties. I know how PLG does this. And if there is new information, they would either change their distribution and come up with new numbers or whatever. The methodology is set up, and they can use new information on a particular isotope or whatever they are dealing with.

MR. SIESS: Mr. Chairman, I am going to turn it back to you. We have here from PG&E a representation that they will answer questions. I don't think they have anything they want to say. They have proposed the Level 1. The Staff has accepted a Level 1. And they answered some questions yesterday about possible costs and impact on time.

They have indicated in response to the Staff that depending on how the results come out, after they get the new seismic design basis and look at the Level 1 PRA, they may want to go on beyond that, as well you might, because the farther out in the consequences you go, I think the better this plant is going to look. But that's just my guess.

So what I'd like to do is turn it back to the Chairman to get the sense of the full committee. I'm in a position to draft a letter which says we approve the program and comment that the proposed Level 1, the Staff has accepted it, and we agree, to which they can add additional comments or

anybody else. Or I am prepared to write a letter that says we approve the program, and we think that the PRA ought to be beyond Level 1, to Level 2 or Level 3 or whatever we might say, and I propose that those are the alternatives, and I think if you could find out how the committee wants to go, even now or Saturday.

MR. WARD: All right. Thank you, Bill.

MR. KERR: What would you anticipate one would do with the results of the Level 2 PRA that one could not do with the results of a Level 1?

MR. OKRENT: Well, the Level 1 doesn't --

MR. KERR: I understand the difference between the two, I think.

MR. OKRENT: It doesn't give you a good handle on the releases and --

MR. KERR: But suppose now we have a good handle on the releases. What does one then do? This is an operating plant.

MR. OKRENT: Yes. But they might --

MR. KERR: I'm not trying to be critical. I'm trying to understand.

MR. OKRENT: Well, they may have a coremelt frequency like Indian Point, which the Staff estimates would be about 4 times 10 to the -4. And if you didn't have an evaluation of containment capability, that could look quite

threatening or more threatening than it is, but you can't just say that containment is equivalent to Indian Point. In fact, the Staff argues that Indian Point has a better than average one.

MR. KERR: I did not sharpen the question enough. One consequence of carrying this out would be that you would decide to shut down the plant. I sort of went to an extreme. Now short of that, what would one anticipate that one might do? Rebuild the containment, for example?

MR. WARD: Could I -- there might be some procedure changes or some instrumentation or something that would be --

MR. OKRENT: Well, penetration. There may be one penetration that will go only to five psi above the pressure at which it was tested. And if they don't look for containment pressure capability, you don't know that.

Now you know at Sequoyah, they fixed up one major penetration, for example, and that may not be the only reactor that did it after looking at containment pressure capability. So I really can't understand, in fact, spending eight to ten million dollars, of which three or so is on this PRA, and not spending this additional whatever it is, 300 to 500 K, to get to Level 2, and I'll tell you, as I say, it's stopping at the wrong point.

MR. WARD: Could I make a comment? I have to leave for a few minutes.

I guess from what I have heard, I am very much in sympathy with what Dave is saying -- you know, my personal position. It seems to me, the point of doing the seismic PRA is to look for what might be some new relationships, I mean, that are unique to the sort of common failures you can get from a major seismic event that could effect, you know, both the ability to cool the core and the ability for the containment to perform as designed. And it just seems to me that if the PRA isn't probing for those things, it isn't of an awful lot of use.

MR. SHEWMON: It is certainly probing for core damage or coremelt, isn't it?

MR. WARD: It is, but that's just half the story.

MR. SHEWMON: I know, but that's not quite the way you worded your statement. So you are only concerned about what you feel is left out, is the performance of the containment.

MR. WARD: Yes. And I know they are looking specifically at, I guess -- I think what I heard -- at containment failures from seismic. But it strikes me that there are some interrelationships which the PRA is the tool to use to examine those.

MR. SHEWMON: Now the reason for this whole exercise was that there is a possibility of higher seismic loading here than other places, and therefore they are supposed to go back

and look harder. Is that sort of a price for the license?

MR. OKRENT: If one wants to be a little bit lengthier, when the committee first reviewed the operation in 1977 or '78, we agreed with the design basis, but we didn't agree with the methodology that the Staff was proposing, whereby they arrived at a design basis. And at that time, we suggested, look, we are still in a learning stage with regard to earthquakes and in particular earthquakes around here and so forth. It would be good to take another look at the seismic design basis in ten years; that's what we said in '77 or '78.

And then whenever we were reviewing something, quality assurance or something, we reminded the Commission that we had made this recommendation, and the Commission picked it up, not when we first made it, but when they were reviewing the quality issues, okay.

MR. MARK: Well, there was another point, as I recall it. When we wrote that first letter, there was a really wide spread of opinion as to just what the seismic setting was at Diablo Canyon, and it was also known that there was a lot of further exploration going to occur in the next few years. And while it looked all right on the basis of what we sort of guessed was the case, it was possible that it should be looked at again.

MR. SHEWMON: That is a major part of this program.

MR. OKRENT: Yes.

MR. EBERSOLE: Can I ask a question? Dave, maybe I'm just beginning to see the light. Are you saying that what you are bothered by is that even if the containment damage itself was not an initiator, you could have a damage state, some other kind of damage state, that led to core damage, but you would have a containment which wouldn't work?

MR. OKRENT: Well, in what they are proposing to do at Level 1, they won't evaluate the containment capability, even its pressure capability.

MR. EBERSOLE: But it could be damaged by the earthquake, not having --

MR. OKRENT: Furthermore, it could be damaged but not fail, and I don't know how one would get at that. But again, they wouldn't look at that, since they are not going to look at the containment.

MR. EBERSOLE: But the containment can be damaged like anything else, and then you can have a damage state, but it will not contain.

MR. OKRENT: Well, if the containment -- if they can predict a loss of containment integrity due to the earthquake, that will be done in Level 1.

But degradation of ability to withstand pressure, I don't know. Not in this. They are going to have to look at pressure capability.

MR. EBERSOLE: Well, I thought that's what containment failure would be, a loss of ability to do anything it's supposed to do.

MR. OKRENT: Well, Jesse, you are the man who frequently says things aren't always on or off, or black or white; right?

MR. EBERSOLE: Yep.

MR. OKRENT: Okay. Well --

MR. SHEWMON: To what extent are we telling them to go up for common mode -- and I am using this only as a paraphrase, you know, bring me a rock.

MR. OKRENT: No, no.

MR. SHEWMON: But you are talking about there is some kind of damage, we're not sure what kind of damage, and there is now an interaction between this damage which didn't cause failure in some future event, which may cause leakage.

MR. OKRENT: No, I'm sorry, but there are some things mixed up in here. If you do a PRA, you want to estimate the releases from the containment. You have to evaluate the capability of the containment to withstand pressure and temperature. You do that in a Level 2, not in a Level 1.

You furthermore have to evaluate them, the build-up of pressure with time, to estimate, so you look at the cooling capability and if it is lost, will it be restored and so forth

in time before the pressure exceeds whatever is the failure point and so forth.

Again, that would be part of a Level 2 examination. There may be some special things in a severe earthquake that have not been looked at. That's what Dave Ward was suggesting a little bit that they would also look at.

I suggested one thing, namely, there is going to be a lot more room for confusion in the control room since they will have lost much of their instrumentation automatically, and there may be much more complicated transients, furthermore, because you have multiple failures of nonsafety systems.

MR. SIESS: But that's still covered in Level 1.

MR. OKRENT: Well, not ordinarily, and how they will do it, I don't know.

MR. SIESS: But you don't have --

MR. OKRENT: The ability of the operator to recover from these, you don't have to do unless you want to find out, you know, how long does this transpire, and when does it get water back to the containment sprays, if that's the cooling mode, and so on.

So for a Level 2 you have to examine the ability of the operator to recover under these circumstances, to fix the diesel generator under these circumstances or whatever. Whereas if what you have to do is the damage state, you can

stop at an earlier point, you don't have to get to the test of whether the containment has failed or not.

MR. SIESS: Let me try to make a point. The basic objection of this reevaluation was to look at the seismic design basis which is a spectrum input at the base of the structure. It's arrived at by looking at what happens at the site, looking at the soil structure interaction, et cetera, and you find out what happens at the site by starting somewhere out with an earthquake and transmitting it in and so forth.

The first result of this thing, the first three items on the list, will give them a new design basis or a spectrum of design bases, maybe with probabilities on them. If those design bases are different, then there needs to be some way of determining what is the significance of that difference, and this fourth requirement was put in that once they get the results of this, they assess the significance of it.

If it is half the previous value, I don't even see why you would have to do a PRA. If it is greater than that, you certainly want to assess the significance to see if you have to shut the plant down, because it's now somewhat larger design basis.

I feel confident that if it comes out significantly different and the assessment based on a Level 1 shows there is

significance, nobody is going to have to tell them to go to a Level 2 or 3 or certainly we're not going to have to tell them. I think somebody is going to look at that because the low population and a number of other factors are going to be in their favor, that the significance, I think, is going to be smaller, the farther down the line they go in the PRA.

Now that's a judgment. So I would rather see them get on with this job in three years, come up with a revised design basis, and then look at the significance. And I am convinced that if it looks significant, we are going to find out about it and something will be done about it.

Now they have to get shut down for a year or two while they go to that Level 2 or 3. That's another story.

Incidentally, Paul, this is an insurance policy in one respect because as new geological evidence is developed 10 years from now, what they are doing will be in such a form that they could factor that new evidence in very easily and determine its effect.

So I think it is a very good insurance policy against future improvements and knowledge which are inevitable.

MR. EBERSOLE: Do new geological findings always have a tendency to make things worse?

MR. SIESS: New geological findings tend to make worse, but the assumption here to start off with on the Hosgri

was so far in one direction that I think there is at least a decent chance that it will end up less.

Then they are going to do a lot more in terms of getting from the earthquake into the site, it's more refined than what they did before.

I think our consultants felt that there was about a 50-50 chance. There is some evidence that the fault may be a different kind, but there is also some evidence that it wasn't as long as it was to get it up to 7-1/2. So I wouldn't say I'd clearly go in one direction on that. By that time they may find the tau effect doesn't exist, but something else may exist. Well, the tau effect took this from .75 down to about two-thirds, as I recall, or six-tenths or something. But there is something there that probably is going to show up, anyway.

MR. LEWIS: Well, where are you, Chet? Are you looking for guidance on what to do?

MR. SIESS: I am looking for a vote on -- I had turned it back to the Chairman because I am not a neutral in this affair. I wanted the three people who disagreed with Dave -- and it's now turned over to the Full Committee, unless PG&E would like to say something.

MR. BRAND: I don't believe so.

MR. AXTMANN: I've got a question. At the subcommittee meeting someone pointed out that the Japanese go

for stiff piping and Diablo Canyon calls for flexible piping.

MR. SIESS: No, nobody can have a simpler piping system than Diablo Canyon.

MR. AXTMANN: Oh, excuse me. The statement is -- the question was, why are we going from stiff to flexible?

MR. SIESS: There are moves aboard that will be coming in to the committee on stiff vs. flexible piping, but Diablo Canyon as it exists has got piping supported as stiffly as anybody in the world, I would guess.

MR. AXTMANN: So we will be edified in 10 months?

MR. SIESS: The proposed change is something we will be debating.

MR. KERR: There is a two-volume report on a Staff study of pipe stiffness and its relationship to seismic behavior.

MR. SIESS: Yes. But this plant as built and as rebuilt has got an awful lot of pipe supports.

MR. LEWIS: Chet, are you looking for an instruction to go prepare a letter? Are you looking for a soft vote? What do you want?

MR. SIESS: A soft vote would help. I can write a letter with two paragraphs in it and we can debate it Saturday if you wish.

MR. LEWIS: Well, we can certainly conduct a soft vote if someone will make a motion on which we ought to vote.

MR. SIESS: Well, as chairman, I would move that you accept the majority opinion of the subcommittee and we write a letter accepting the Staff's acceptance of the PRA at Level 1.

MR. LEWIS: Okay. The consequence of that, if passed, is that you will prepare a letter which we will then vote on. Okay. Is that a reasonable motion to vote on? We have been talking about it now. Okay, let's have a show of hands. Who would like to support Chet's motion?

[Show of hands.]

MR. LEWIS: Opposed?

[Show of hands.]

MR. LEWIS: The majority, but three against.

MR. SIESS: Well, I will draft a letter and we will see what it looks like.

Now I'm not sure that we can guarantee that the Staff will accept our recommendation.

MR. LEWIS: Well, that seems to be a general rule of life.

[Laughter.]

MR. SIESS: Well, it would be advice from the Commission.

MR. MARK: I just wanted to find out if my impression is correct. I have the impression that whenever an Applicant does a PRA, the Staff refuses to accept it without doing it over again, and getting bigger risk numbers. Would

that be the case here?

[Laughter.]

MR. ISRAEL: Yes, that's part of the risk the Licensee takes.

MR. MARK: Is the Staff going to be in 1988, when this thing is with us, with enough money left to be able to do a review of a Level 2 PRA?

MR. ISRAEL: Well, no, there is the one-week evaluation, the two-month evaluation and the three-year evaluation. It depends upon the money we have at that time.

MR. LEWIS: Okay. Are we done with the subject, Chet? You have your instructions and you will write a report.

Okay. Gentlemen, the next item -- Dave is missing for a few minutes, but he'll be back.

The next item on the agenda is recent events at operating reactors. Do you want to head right into it or take five minutes?

Five minutes.

[Recess.]

MR. LEWIS: There are two options. We are coming up to a group of, as I read the agenda, five or six -- five recent significant operating events. I see two possibilities. One, if we go through them in the order in which they are listed, we will not get to Davis-Besse, which many people think is the most significant one. The other

possibility is that we start with Davis-Besse and we may not get to the other four because it is one of the more interesting ones.

The Staff is willing to do it either way, so we only need a statement of preference from the Committee. What do you want to do, start with Davis-Besse or end with it?

MR. REMICK: I support that. Start with it.

MR. LEWIS: Is that agreeable, to start with Davis-Besse? But at some point arbitrarily we will cut it off so that we get to the other ones. We will get only a preliminary report on that now because the investigation team hasn't come in. We will hear more about it next time. Okay.

MR. JORDAN: My name is Ed Jordan. I'm the Division Director Emergency Preparedness and Incident Response in I&E.

The five events that we plan to present to you are selected out of eight by the Subcommittee on Operations, selected out of 11 -- I'm sorry -- and these events we will go through as briefly as we can.

I would like to say just something about the incident investigation program before we do regarding Davis-Besse.

The Staff put together a Commission paper. This is SECY 85-208, entitled "Incident Investigation Program," which proposed to the Commission a manner of dealing with events, and it is response to the Brookhaven study and also to the

ACRS findings.

The Commission has not adopted this; however, the Staff advised the Commission in this paper that for the interim the Staff was going to proceed with this method. The date of this paper is June 10th. The Staff had prepared it, and the event a Davis-Besse happened June 9th. So it was in this fashion that we established this multi-discipline team that is made up of technical experts from the various NRC offices.

Ernie Rossi, who normally chairs this particular meeting for I&E, is the team leader. He reports directly to the Executive Director, Bill Dircks, in this fashion. A memorandum was prepared by Bill Dircks identifying Rossi as a team leader and identifying J.T. Baird of NRR, Larry Bell of the I&E Training Center, and Wayne Lanning of AEOD as the experts in the various areas that seemed appropriate for this particular investigation.

MR. LEWIS: I might just remind the Committee, for people who worry about the ITT proposal, it is in your package called 4.2, which is under Anticipated ACRS Activities. There is a long proposal on that.

MR. JORDAN: Good.

These team members were relieved from any other assignments until this particular task is completed, and that is scheduled to be done July 22nd with a presentation to the

Commission immediately thereafter, and I would expect a presentation to the ACRS as a specific event in your August meeting.

The special investigative team is to prepare a single report, and this represents the single investigation that the Staff if preparing. That report would focus on fact finding. It would identify the root causes and provide findings and conclusions, but it would not contain, normally, recommendations. The recommendations subsequent to that would be developed by the program offices responsible for the various areas. If it impacted on inspection or impacted on licensing, then I&E or NRR, respectively, would be responsible for coming up with those recommendations.

The investigation was conducted or is being conducted in a very factual manner. The very first effort was to take statements from plant personnel who were involved on shift at the time of the event, review strip chart records and then go to procedures and logs and manuals, and finally looking at the equipment.

They are in the process now of examining what they feel may be the causes of the event, and it is obviously multi-faceted. There was imposed a confirmation of action letter that caused the plant insofar as possible to leave the equipment in the "as-found" condition, which is one of the issuesa that the ACRS, I believe, is interested in.

The program is being administered by the AEOD office, but Dr. Rossi is reporting directly to the EDO in terms of the findings.

So with that, I would request Al DeAgazio to give the discussion and refer you then to the handout, and there is page 12 of the handout that starts the presentation.

We did attach to the package an information notice issued July 8th to all licensees summarizing what we know about the event at this time.

Al.

Oh, yes, one other thing. We have received copies of the licensee event report, and we will distribute extracts of that to the Committee.

Okay, Al.

MR. DeAGAZIO: I'm Al DeAgazio. I'm the project manager for Davis-Besse in NRR.

The sequence of events that I have has been put together from information that has been available from the team and information that was available from the resident inspectors. The sequence of events will leave out a lot of the detail that is available but doesn't particularly affect the sequence as we understand it now.

Before I get into the description of the event, I think there are two packages that are being handed out to you. One package has a slide which has the sequence of

events. That looks like this.

[Slide]

That package has a diagram of the feedwater system, but it's a little more simplified than the diagrams that appear in the second package that is being handed out now. The second package has the feedwater systems with the valves in various positions, depending upon the condition of the plant, and as we go through, I will refer to those. So if you will, refer to the second package for the diagrams.

Before I go into the description of the sequence, let me first describe what the feedwater system looks like at Davis-Besse, or the feedwater systems look like at Davis-Besse, and what the normal alignment of valves is.

The Davis-Besse plant has two steam generators, two once-through steam generators. There are two steam-driven main feedwater pumps. They can provide flow to either steam generator. There are two turbine-driven auxiliary feedwater pumps that provide auxiliary feedwater flow to either one or both of the steam generators, depending upon valve alignment.

This plant also has a small capacity electric motor-driven startup feedwater pump. This is used normally just during start up, and the later stages of plant shutdown. The capacity of the electric motor pump is less than one turbine-driven auxiliary feedwater pump. The characteristic curve of the pump is kind of flat, so it is difficult to say

whether it is half capacity or what it is because it is very dependent upon the pressure in the steam generator.

So I think we will say it is no more than half of an auxiliary feedwater pump as far as capacity is concerned, and depending upon the pressure of the steam generator, it could be considerably less.

MR. EBERSOLE: At the time of the occurrence of this accident, was it not contemplated that additional pumps would be installed at some point in the future?

MR. DeAGAZIO: There is a license condition that is in effect now that would require them to provide a new startup feedwater pump in a new location. The startup feedwater pump would have a capacity that is equal to one of the auxiliary feedwater pumps. The piping arrangement for that startup feedwater pump would not be the way it is shown here but it would have the capability to also provide feed to the higher level auxiliary feedwater nozzles in the steam generator.

MR. EBERSOLE: When was that condition imposed and how long has it been more or less dormant?

MR. DeAGAZIO: We started working on that license condition approximately last summer, about a year ago, and it was finally put in place on the plant in January of this year, after evaluation. The plant was in a refueling mode from -- I think it was about September to January.

MR. EBERSOLE: But that was not operational when

this occurred?

MR. DeAGAZIO: The new startup feedwater pump is not in the plant now yet. It is scheduled to be in, according to license condition, before startup from the next refueling outage, which would nominally be sometime next spring.

MR. EBERSOLE: Thank you.

MR. DeAGAZIO: During normal operation at power, the feedwater pumps are providing flow through the feedwater heating trains through a main flow control valve, and past the isolation valves into the steam generators. The auxiliary feedwater system, there are four valves that are closed so that there is no open path from the pumps to the steam generators. These are containment isolation valves.

These are steam emission valves from -- I didn't have enough room and I was going to get complicated, so Point A feeds here, Point B feeds here (indicating). The steam admission valves are closed. Flow control for auxiliary feedwater is by controlling the speed of the turbines. There is not a flow control valve.

At the time just prior to the event, the plant was operating at 90 percent power, and there were no special tests going on at the time. The plant was stable. One main feedwater pump was on automatic. One main feedwater pump was in manual control. The purpose of that was that they had been experiencing difficulties with the speed governors on the main

feedwater pumps.

At the last refueling outage, both of the feedwater speed governors had been replaced with new models. The initiating event was the tripping of the main feedwater pump that was in automatic on overspeed.

The reduction in feedwater flow initiated a power runback, and approximately 30 seconds later, the power level had been down to 78 percent, but the runback wasn't fast enough and a high pressure reactor trip occurred.

Just shortly after, about one second later, a steam and feedwater rupture control system signal was generated, and this signal was generated on low steam generator level, apparently. It seemed to be spurious, and the cause of it is speculated at the moment to be oscillations created by the tripping of the turbine as far as the level control system for the steam generators.

I'm sure that that's not at all firm at this point, but that is the latest theory at the moment.

This trip cleared itself approximately three seconds later. Whether or not it was related to that trip of the steam and feedwater rupture control system, the main steam isolation valves tripped. This valve and this valve (indicating).

This had the effect of stopping all steam flow from the steam generators, and since the one main feedwater pump

that was running in manual at that point was deprived of steam and yet it coasted down on the amount of steam that was stored in the system beyond the isolation valves, it took approximately four minutes for that pump to coast down and trip.

MR. LEWIS: Could you remind me why the main steam isolation valves have to trip at that sequence?

MR. DeAGAZIO: Why do the main steam isolation valves trip in that sequence?

MR. LEWIS: It was normal that they should have at that point in the event, or was it abnormal?

MR. DeAGAZIO: It really depends upon what kind of signal was generated from the steam and feedwater rupture control system. First of all, there should not have been a signal from the steam and feedwater rupture control system at that time since steam generator levels were normal. But the purpose of the steam and feedwater rupture control system is to provide protection to the plant in the event of either a steam line break, a feedwater line break.

It provides for starting of the auxiliary feedwater pumps in the event of low steam generator water level, and provides actuation for the auxiliary feedwater system in the event all four reactor coolant pumps are lost to promote natural circulation.

MR. KERR: But isn't the answer to Dr. Lewis'

question that you don't know exactly why the main steam isolation valves closed at this point?

MR. DeAGAZIO: Exactly what kind of a trip was generated, whether it was a half-channel trip or a full-channel trip, I don't think we know at the moment. If it was a full-channel trip, other things should have happened besides the main steam isolation valves going closed. If it was a full-channel trip, then some valves that should have worked didn't work. If it was a half-channel trip, then the main steam isolation valves should not have closed.

MR. LEWIS: Is that the answer to Dr. Kerr's question?

MR. DeAGAZIO: I think so. I think the answer is we don't know just exactly why the main steam isolation valves closed.

MR. EBERSOLE: Well, isn't it a fact, though, that if you have a main steam line failure, it is obligatory that you must shut one of these big boilers off and permit only one to blow down on the pain of having excessive containment pressure?

MR. DeAGAZIO: If you have a steam isolation break, both of them are isolated.

MR. EBERSOLE: They are both isolated?

MR. DeAGAZIO: Both of them.

MR. EBERSOLE: Both mains are isolated?

MR. DeAGAZIO: Both mains are isolated.

I am a little ahead of myself, but why don't I go ahead and describe what happens on the steam and feedwater rupture control system actuation on a low steam generator --

MR. EBERSOLE: WEll, that is what gives you redundancy to prevent discharge of both of them. You isolate both of them.

MR. DeAGAZIO: That is correct.

[Slide]

If there is a low steam pressure trip of the steam and feedwater rupture control system, the main steam isolation valves close on both steam generators and it doesn't matter which steam generator has the low pressure as far as the main steam isolation valves.

Oh, I have got the wrong diagram here.

[Slide]

This diagram is for a low pressure trip on steam generator No. 1. Both main steam isolation valves close. The main feedwater trains are all isolated. The steam admission valve for the feedwater pump that is normally dedicated to feeding that steam generator is opened, so that steam generator No. 1 -- I'm sorry. We have a break in steam generator No. 1. Steam generator No. 2 is aligned to feed -- to obtain its feed from aux feedwater pump No. 2. No. 1 is not aligned to No. 1 steam generator.

The steam admission valves from that steam generator, which presumably has a break, have not opened. The steam admission valves from the No. 2 steam generator are opened. That is this valve (indicating). And the discharge flow from the No. 1 pump is aligned to feed the No. 2 steam generator, as is the No. 12 feedwater pump.

So that this steam generator, if it has the break, if it is the one that has the low steam pressure, it is totally bottled up and feedwater is provided only to No. 2 steam generator. If No. 2 fails, has the low pressure, then the No. 1 steam generator gets all the feedwater flow.

If both of them see a low pressure signal, they will both get isolated. There will be no steam admission, no feedwater flow path until one of the steam generators repressurizes. At that point the signal clears and the one that recovers pressure will get the feed.

[Slide.]

Since we are discussing the flow paths on a low steam generator water level, or on a high feedwater to steam pressure differential which would be indicative of a feedwater line break, the action is similar but a little different.

The isolation valves to the steam generators are not closed, so that feedwater flow can be provided from the pump that is dedicated for that particular steam generator. Steam admission to the auxiliary feedwater pump turbines is also

from the respective steam generator.

Going back to the sequence now, the main steam isolation valves had closed, all main feedwater flow is off, at this point the steam generator water level is declining.

I'm sorry, I'm getting ahead of myself.

We had the spurious trip of the steam and feedwater rupture control system, and that had the effect of -- let me back up. I am still in the wrong sequence.

The spurious trip had the effect of closing the main steam isolation valves and cleared approximately four seconds later. The main feedwater pump took approximately four minutes to coast down and at that point the steam generators are without feedwater.

At about six minutes into the event, there was an actual, a real low level steam and feedwater actuation signal generated for steam generator No. 1, and that aligned the valves as shown here [indicating], forgetting the fact that it was No. 1.

I'm sorry, it didn't matter.

The valves were aligned as they are shown here. The operator, recognizing that he was getting into a situation where level was dropping, attempted to initiate auxiliary feedwater and not depend upon the steam and feedwater rupture control system, and the automatic system beat him by just a second or so, but when he made his action, he unfortunately

selected the wrong buttons on the control panel and selected the low pressure trip instead of the low water level trip and so it had the effect of aligning the valves this way.

[Slide.]

But since he punched the buttons for both steam generators, both steam generators were isolated from feedwater and there was no feedwater flow to either steam generator. These valves were closed also, and these and this valve was closed along with this and this [indicating].

So there was no flow path into the steam generator and the steam generator water levels are now going down. At this point they are drying out and approximately one minute later he recognized that he made the error and tried to correct it.

When he did correct it, what should have happened was that the valves that were closed should have opened according to this diagram.

However --

[Slide.]

-- the system should have aligned this way. However, this valve and this valve failed to open.

MR. LEWIS: Could I just jump ahead and be sure that by the time you are finished, you will tell us how many equipment malfunctions there are or there were because I'm having a little trouble distinguishing what is normal response

to the event and what is abnormal.

MR. DE AGAZIO: This is a very complicated system, and I do have a list of what we know has failed so far.

The operators then were dispatched throughout the plant to attempt to get these pumps started again, which tripped just shortly after they started on the first real low level signal from the steam and feedwater rupture control system to attempt to open these valves and to attempt to get the electric motor-driven start-up feedwater pump in operation. And this was a manual operation. You have to go to the valves locally and manually open the valves.

Also, to replace some control fuses which were pulled to prevent starting of this pump against a closed suction valve. By approximately --

MR. OKRENT: Excuse me. Were the positions of those valves on the feedwater line indicated in the control room? The two that opened.

MR. DE AGAZIO: These valve positions are all indicated in the control room.

MR. OKRENT: Thank you.

MR. DE AGAZIO: This one is not [indicating]. The others are just manual local valves.

The operators were successfull in getting the start-up feedwater pump started about the same time they were successful in resetting the turbine-driven steam water --

turbine-driven feedwater pumps, so that by about 16 minutes or so they had started restoring feedwater to the steam generator.

Just prior to restoring feedwater to the steam generators, the pressure in the reactor coolant system had risen to the point where the PORVs had actuated. They actuated three times. There was indication that the PORV failed to recede on the third time. Whether this was actually a failure of the valve or not, I think there might be some question, because just about that time the feedwater was being restored to the steam generators, so whether the drop in pressure was what caused the operator to believe that he had a stuck-oven PORV or whether it was some other indication, we don't really have that information at the moment.

The latest information from the site is that the valve has been disassembled and there was nothing abnormal about the valve. So the reason why it stuck, if did stick, is not known.

Once the feedwater had been restored, the plant entered a normal cooldown.

[Slide.]

There was at least 13 or 14 different malfunctions or failures or unexpected occurrences.

First of all, the initiating event was the main feedwater trip on overspeed.

Following that was the unexplained either spurious half trip or full trip of the steam and feedwater control system. The unexplained action of the two main steam isolation valves which should have closed, if this was a full trip, should not have closed if it were a half trip.

Not shown on here is if this were a full trip of the steam and feedwater control system, then other valves should have actuated which did not.

The two auxiliary feedwater pumps operated for a few seconds after the initial low level signal from the steam and feedwater rupture control system. Then they tripped out on overspeed.

Incidentally, one of those speed governors on the auxiliary feedwater pumps had been replaced at the last refueling outage with a new design, and the other one was the original design, but they both did trip out.

The two isolation valves in the auxiliary feedwater line that failed to open when reset --

MR. LEWIS: Is it known that they got the signal to open?

MR. DE AGAZIO: Yes, when the operator recognized he had made an error and corrected his mistake, he pushed the reset button which should have removed that signal and caused the valves to open again.

MR. LEWIS: But it isn't known whether the valves

got the signal; it is only known that he pushed the button? I am trying to find out whether it was a valve failure or a control failure.

MR. DE AGAZIO: I think the indication was that it was a valve failure, not a control failure, because when the operator went down and moved the valve to slightly off the seat manually with the hand wheel, then the valve continued to open.

The PORV failing to reseat -- and there is a question mark that might be associated with that -- there was a start-up feedwater valve that failed to open and had to be opened manually. After the auxiliary feedwater pumps were restarted, one speed governor failed to respond to the control from the control room and had to be operated manually.

When the auxiliary feedwater system was restored, there was a low suction on the system and it caused a switchover to a back-up service water supply, and somewhere in the sequence was a damaged turbine bypass valve.

MR. LEWIS: You know, I am an ignorant person, but if -- and of course the final report on this isn't out yet, but if I really am to believe that on a challenge produced by the tripping of one main feedwater pump, there were subsequently nine independent failures of components in the following sequence, I would take that very seriously. You know, I reserve judgment until I know what actually happened,

but that would have serious implications.

MR. EBERSOLE: Yes. In that connection, do you intend to put together an analytical study of what is the probability of this combination of events and use it as a backdrop to consider PRAs in the future?

MR. DE AGAZIO: I'm not sure a decision has been made on that yet. I think that until the report is issued by the team and we know more about this with certainty as to what happened and what failed, I think it would be premature.

MR. EBERSOLE: It may be that this is a milestone occurrence in this context, and that we have to look at PRAs in a somewhat less optimistic manner than we currently do.

MR. LEWIS: Well, Jesse, just in defense of PRAs, we should wait until we find out what happened. There is zero probability that there would be 10 independent failures. So we will find that out.

MR. MARK: Maybe this is a precursor for TMI-1.

MR. LEWIS: It may well be. In fact, since this is the first time the IIT system has been exercised, even before it's been approved by the Commission, it would be very interesting to see if they can get at the root cause of this event.

MR. JORDAN: Yes, and that is the purpose, of course, of the way the team is constituted and managed. I sincerely hope that works.

MR. LEWIS: I am holding my breath.

MR. REMICK: The title there says known equipment failures or malfunctions, and you have the main feedwater trip. Was that a malfunction? You said it was due to overspeed of the turbine that was on automatic. Was that a spurious malfunction, or was there a reason for the overspeed trip?

MR. DE AGAZIO: The plant had been experiencing difficulties with both of these main feedwater pumps the prior weeks, so I count that as a malfunction in the plant. It is the initiating event here, but they had been having considerable difficulty with these pumps, which is the reason why they were running one of them at manual, so they could minimize --

MR. KERR: It's my impression from another source, and perhaps mistaken, that they had indeed called new control systems on these pumps recently. Is that not the case?

MR. DE AGAZIO: I'm sorry.

MR. KERR: I got the impression from another source that they had recently installed new control systems for both feedwater pumps.

MR. DE AGAZIO: Both main feedwater pumps had been equipped with new speed governors on the last outage.

MR. KERR: And they were having some difficulty with getting them adjusted or making them work properly?

MR. DE AGAZIO: They were supposed to be improved models, and one auxiliary feedwater pump had also been equipped with a new speed governor.

MR. REMICK: Do you know if the pump actually did overspeed and that is the malfunction and the trip worked, or the pump didn't overspeed but tripped by a false signal that was overspeed? Do you know which?

In other words, was there an actual overspeed of the turbine pump, of the turbine and pump?

MR. DE AGAZIO: I don't believe I have that information.

MR. LEWIS: Forrest, there was a comment in the LER which has a lot more information in it, which I can't find at the moment, and I don't know whether it's the answer to your question, that somewhere there was frequency to voltage converter trip that had failed on an overspeed control car. And I may not be quoting it exactly, but there was something like that.

MR. EBERSOLE: How many minutes were left before core damage would have occurred?

MR. DE AGAZIO: I'm afraid I don't have the answer.

MR. SHEWMON: There was almost no loss in the primary system, was there?

MR. DE AGAZIO: Not much, a blowing down three times to the PORVs for a short period of time. So at that time they

hadn't lost much, anyway.

MR. SHERON: Brian Sheron from the Staff.

This leads obviously to the question about feed-and-bleed on this plant, what mitigative actions could have been taken. We have done a fair amount of hand calculations since the event and we are also in the process of doing more sophisticated calculations with RELAP-5 right now, and we are going to report this to the ECCS subcommittee I think on the 31st.

But what we determined is that if the operator had taken no action whatsoever to either turn on makeup pumps or turn on the start-up feed pumps, open a PORV, we probably would have seen core uncovery in about 50 minutes, five oh, 50 minutes. Generators dry out very shortly in a few minutes, and the rest of it is just -- first it is displacement of water by steam collecting in the high points, until you uncover the vent path, which is the surge line to the pressurizer. And then it is just a boil-off.

We have also looked at the capability -- we also looked at the capability of what the makeup pumps, the start-up feed pump and the PORV could have done, and we have generated a table that's what you find out is there is a matrix of a combination of things that could have been done.

Had they started both feed pumps -- I'm sorry, both makeup pumps, opened the PORV and turned on the start-up

feedwater pump, they most likely would have been able to keep the core covered. There was sufficient capacity there to remove decay heat.

The calculations also indicate that if they had just turned on the two makeup pumps and opened the PORV and had not even started the start-up feedwater pump, that they still could have kept the core covered.

If you just had one make-up pump, you would probably have uncovered the core, but you would have extended the time considerably.

Also, remember the event occurred at 90 percent power, so a lot of these numbers also will change at 100 percent power.

In terms of how much time did an operator have to initiate these actions, right now what we see is that -- and these are just various numbers we have -- if the plant had been operating at 100 power, rather than 90, if they had initiated just two makeup pumps and no start-up feed pump, if they had initiated makeups at the time of steam generator dry-up, which I think is about two or three minutes, they wouldn't have uncovered the core, but had they waited 20 minutes to initiate the make-up pumps, they would have gotten core uncovery.

I beg your pardon?

MR. ETHERINGTON: With damage to the core or not?

MR. SHERON: Well, what we are calculating is core uncovery. Typically damage occurs some time 15 or 20 minutes later. Once you uncover say about half the core and you start getting substantial clad heat-up -- we haven't done the calculations into the core. These are just until when the core uncovers.

MR. REED: Your numbers, most of them surprise me. Your 50 minutes surprises me. Your two to three minutes to dry out on the secondary side of the steam generators does not surprise me, because that number has been floating around a lot.

And, of course, I was quite surprised that the system went for 11 minute without secondary feedwater, and apparently there was no core damage and you didn't get above 594 degrees.

Now I have to conclude that the reason here that you are giving me such numbers as 50 minutes to top of core is because this particular B&W plant has high set steam generators vs. the others with low set steam generators, and therefore you are getting the drain-down advantage of the loop piping in the steam generator primary side; is that correct?

MR. SHERON: No. I asked the very same questions of the individual who did the calculations, and just to lend a little bit of credibility, the person who did the calculations worked for about 10 years for B&W in the analysis area, so

he's --

MR. REED: I think you already answered my question.

MR. SHERON: The answer is that volume of the core, as I understand it, comes out about the same. So the raised loop plant really doesn't behave that much differently than the lowered loop in terms of time to core uncovery.

My understanding is, it also has to do with the relative location of the surge line on the hot leg. During the boiloff process, once you've dried out the generators and you saturate the primary system and you generate steam in a primary system, that steam will first collect in the top of the vessel, and what it does is, it displaces water. And the displacement process is very quick, and so you're pushing water right out of the pressurizer.

Once you fill the upper head, you then collect the steam at the top of the candycane, and you still displace water until you uncover the surge line. Once you uncover the surge line, then steam can escape out the pressurizer, and then it becomes just a boiloff process.

MR. REED: All right. What I think I'm seeing -- and every time I see this PORV thing, I see one PORV; I don't see plural here, PORVs -- I see one series, in-series block valve; is that correct?

MR. SHERON: Yes.

MR. REED: That is pretty thin ice for getting an

exit path, isn't it? I mean, he closed the PORV. Suppose he never could have gotten the block valve, suppose he never got it open again?

MR. SHERON: That's correct. Again, you have to remember that the whole issue of feed and bleed and the fact that B&W plants -- and this plant is not unique in that respect -- all B&W plants only have one PORV. You know, the PORV, HPI, or makeup system was not originally designed for feed and bleed, and the PORVs were put on these plants, as you well know, just to protect the lifting of safety valves.

So you're correct in the sense that obviously a plant that has a higher PORV capability with more PORVs does have, perhaps, a much higher capability of feed and bleed.

MR. MARK: I got the impression from reading one of these things that there was pretty hectic fifteen minutes in that control room, and with the exception of the one operator error, it looks as if they must have done everything just about right and very quickly. Is that the proper impression?

MR. DeAGAZIO: That's reasonable. I think it was not only hectic in the control room, but to recognize that to start up the startup feedwater pump, the auxiliary feedwater pump and get the valves open to both the startup feedwater pump and the stuck-open valve or the stuck-closed valves in the auxiliary feedwater system path took a lot of running around in this plant. They had to go to vital areas with key

access, card access.

MR. MARK: They did a number of things within a period of two or three minutes.

MR. DeAGAZIO: In about nine to fourteen minutes, they were moving through that plant quickly.

MR. MARK: Here is a list of things which malfunctioned, and there was one operator error. Sometime it might be worth saying that this crew was rather good and did very well.

MR. DeAGAZIO: With regard to the operator error, the control panel where he made the error looks like this, and I could pass this around.

MR. MARK: Yes. That was at fault.

MR. DeAGAZIO: This has been identified in the control room design review as a human engineering defect. What he did was, he punched the two top buttons, when he should have hit this one (indicating).

MR. KERR: I thought the aux feedwater system was automatic start; is that not the case?

MR. DeAGAZIO: The auxiliary feedwater system is automatic started by the steam and feedwater rupture control system.

MR. KERR: Well, why did he have to punch anything?

MR. DeAGAZIO: It's my understanding --

MR. WARD: Isn't that just confirmatory?

MR. DeAGAZIO: It's really not to depend upon the automatic signal, but to, when you recognize that you're going to need an action, to take the manual action and not depend upon the automatic action.

MR. WARD: It is just sort of follow-up, which is, I think, in most plant procedures, and unfortunately he followed up with the wrong button.

MR. DeAGAZIO: I think that what happened is that if you know where the location of this panel is with respect to the indicators that he was looking at, he had to move around the control room, around from the central horseshoe and get around to the back panel, and the automatic signal just caught it first.

MR. KERR: I'm not trying to be critical. I'm trying to understand.

My impression was that after TMI, the Staff decided that aux feed systems should have automatic startup, because this was likely to produce less operator error than manual startup.

Now what you have here is a combination of the two, it seems to me. You still have the possibility of operator error, even though you have automatic startup. Maybe there should be a rule that says once you need aux feedwater, you should handcuff the operator for ten minutes, so that he can't foul things up, because he did, I think, what any normal

operator or well-trained operator would do He tried to do something, even the wrong thing

MR. LEWIS: But you know, they are telling us, Bill, that the operators made one mistake, but the machinery made ten.

MR. KERR: But we don't know whether the machines made mistakes or not.

MR. LEWIS: Well, I really -- if I have to predict anything, it is that the final story on this will be quite different from this preliminary version that we are hearing now.

MR. EBERSOLE: In the Palo Verde case, they are counting on secondary blowdown.

MR. SHEWMON: Wait. Let me stay with that for a minute. The operators' procedures are, "Don't see if the machine is working; punch buttons," is that it?

MR. WARD: As I understand it. I don't know, Glenn. Almost all emergency procedures call for confirmatory actions to follow up presumed automatic actions. And I gather in a lot of cases, instead of looking for an indication that a valve is open or a pump is started, the confirmatory action consists of doing manually what's necessary to open the valve or start the pump.

Now, you know, I'm not arguing whether that's the right thing to do or not.

MR. KERR: It may be the right thing. I was just curious.

MR. WARD: Well, that's what's pretty standard.

MR. LEWIS: I will confirm that.

MR. WARD: Well, this is a case where it looks like that is not a good way to do it.

MR. LEWIS: I will confirm that. I have checked some specific ones recently, and the procedure says, "Verify," and what the operators are taught is that "verify" means push it manually to make sure it happened.

MR. KERR: Well, in the long run, that may be the thing to do.

MR. WARD: It is sort of like with a scram. You know, there is some argument and discussion after Salem, whether when there's an automatic scram, should the operator instantly hit the manual scram button?

MR. SHEWMON: In that case, it was the right thing.

MR. WARD: Yes. There, it would have been right.

MR. REED: I would like to support a little bit what Carson Mark said. It looks like the operators did a good job here, except for the one error, which they are entitled to, which is the basis of design from way back, equipment failure and one human failure.

I look at this design, and I've been shocked ever since I looked at it, because I don't -- I can't imagine why

steam-driven pumps were used as auxiliary boiler feedpumps here and used only. That just bothers the hell out of me as a design issue, and I don't understand -- of course, this is a very complex valving arrangement, automated and otherwise, with respect to introducing water into one or the other steam generators.

And the other thing that bothers me is, I see all kinds of closed valves. And as Jesse knows, I mentioned that with respect to System 80. Valves have habits of not functioning, and they are quite unreliable. And whenever you can create systems with checkvalves, without closed valves, your much better off to do that from a reliability of performance point of view.

So I see a lot of design things that bother me, and I shall be listening to the final report very closely to see whether the design issues are really raised this time around.

MR. EBERSOLE: Well, let me enlarge a little bit on that, Glenn.

These turbine-driven pumps unfortunately are also buried in the bowels of the plant and are cooled by AC-driven environmental cooling fans, which makes then interdependent with the AC system, unless that's been fixed, and I don't think it has.

MR. DeAGAZIO: I verified that. It is AC only.

MR. EBERSOLE: So either electric power failure or

steam failure or any other kind of failure can knock these pumps down. It is a design disaster.

MR. WARD: In addition to what Carson said, isn't it fair to say that after the initial error the operators made, they seemed to fully understand what was going on? I mean, in some other incidents we have had, they have been compounded because the operators in the control room didn't understand what the condition of the plant was.

Is it fair to say that here they understood it all the way through?

MR. DeAGAZIO: I haven't heard of any information that would indicate that they did not understand what was happening and that their actions, at least as far as the sequence is concerned, indicates that they did understand.

MR. SHERON: Excuse me. I have no idea what you're saying.

MR. DeAGAZIO: I was saying, there is no information that would lead me to believe that they did not understand what was happening. To answer the question specifically. It is my understanding that they were, at that point, following that the ADOG procedures were in place, and they were following those procedures.

MR. EBERSOLE: It's fair to say, however, that this plant meets all regulatory requirements, doesn't it?

MR. DeAGAZIO: I believe so, yes.

MR. JORDAN: With respect to the operators and their understanding in the control room, we were disappointed in the Operations Center in Bethesda with the information we received from the plant. We were called 36 minutes after the turbine trip, and we were advised that the plant was stable. They had had a main feedwater pump trip. We were unable to get details for some time, and it was a third phone call before an unusual event was declared, which would be the lowest of the emergency classes. And the Licensee did identify in this third phone call with the Shift Supervisor that they were, indeed, in a site area condition, although they had not declared it during that timeframe.

So the individual we were talking to was the Shift Technical Advisor. The Shift Technical Advisor at this plant is on a 24-hour call situation in a building outside of the control room area. He came into the control room area about ten minutes into the event and really never got caught up in terms of providing the Operations Center with anything like the obvious concerns that the personnel that were actually coping with the event had.

So we don't --

MR. KERR: But if you had to choose between their looking after the accident or informing the Information Center, you would sort of take the former, wouldn't you?

MR. JORDAN: I'm sorry, but I have to insist on

both, that if this event had gone awry, then the NRC would not have been in a position to initiate the emergency response capability and make determinations as far as what offsite measures should be taken.

So we do really need both, and it wasn't provided in this case.

MR. LEWIS: Wait a minute. Aren't emergency measures determined locally? They didn't declare an emergency, and that is bad, but doesn't that declaration trigger the emergency procedures locally?

MR. JORDAN: Yes, it does.

MR. LEWIS: Okay. It doesn't have to go through Washington.

MR. JORDAN: No. The Licensee identifies the extent of the emergency. He classifies it. The information to the NRC gives an opportunity for a second look at whether he has, indeed, classified it correctly and whether the appropriate actions are being initiated.

MR. LEWIS: Yes, but that's a surveillance role. You used the word "initiation."

MR. JORDAN: I'm sorry. In this case, it was okay. But I'm just postulating that if it had gone sour worse than it did, if they had not gotten the feedwater back on, then the agency was quite behind in its response.

MR. REED: How many minutes did you say it was

before you got an understanding of the event off the red phone? How many minutes was it?

MR. JORDAN: The initial call came to us 36 minutes after the trip. Our actual reasonable understanding didn't occur until later when the Resident Inspector got to the site. We had only bits and snatches of what had happened. So it was an hour and a half or so after the event.

MR. REED: Well, did they meet the regulatory requirements?

MR. JORDAN: Well, I mean in terms of that the plant was stable, that the knew what had transpired from the superficial respect.

No, we don't know what happened yet. That's correct.

MR. REED: Did they meet the regulatory requirements as soon as possible?

MR. JORDAN: They met the regulatory requirements.

MR. REED: Within one hour, right?

MR. JORDAN: That's correct. I'm only indicating that from the agency's knowledge, what was conveyed to the NRC in 36 minutes was not representative of what the operators themselves knew, clearly, and that's all that I'm trying to get across.

MR. REED: I'm still going to vote with the operators, that they should have concentrated absolutely, as

they did, and let the red phone come second, as long as they met the regulatory requirements.

If you're on an airplane going down, you are not going to necessarily get on the red phone. You would rather tend to your airplane going down.

MR. JORDAN: I agree. But if it crashes, we'd like to know about it. It's too late; that's right.

MR. REED: Well, we hit the ground. You can recover the data boxes weeks later.

[Laughter.]

MR. WARD: Okay. Thank you very much.

Jesse, we are going to lose five of our members at 6:30, so I think we really ought to plan on ending today's meeting at that time. We have about -- Mr. Fraley, will you need a full half-hour?

MR. FRALEY: No. Fifteen minutes ought to be plenty.

MR. WARD: So do you think we could point toward finishing this item at about 6:15?

MR. EBERSOLE: Well, all I can ask is for the individual presenters to expedite it, I guess.

MR. JORDAN: We will go ahead in the order listed.

MR. EBERSOLE: I think you can march quite quickly through this. This is an absolutely unbelievable system interaction.

MR. JORDAN: This is the Hatch-1 stuck-open relief valve. George Rivenbark of NRR will give the presentation. He is the Licensing Project Manager for Hatch.

[Slide.]

MR. RIVENBARK: Well, this was on May the 15th and it happened in the evening. Unit 1 was operating at full power. It is surmised that a crane was passing overhead and ruptured a line that provides water pressure to the charcoal filter deluge valves, the fire protection valves. When the crane hook drug across a line, it cracked it, bled pressure off of the line.

This pressure normally kept the valve closed that caused the valve to open that spray water into the charcoal filters. The operators did not know this at the moment. They knew it some minutes later, maybe 15 or 20, possibly, when they saw water dripping into the control room through the air conditioning ducts.

MR. EBERSOLE: This crane was out in the turbine hall, was it not?

MR. RIVENBARK: The crane at the time was passing from one turbine building to another turbine building, and it goes over the control room. And above the control room, on the floor immediately above the control room is the location of the air conditioning and the equipment.

So -- and this has no protection against outside

influences. It is open to the crane area, and so the crane hook which was presumed to have been drug across there, bumped this line and cracked it.

As a result of the water dripping into a instrument panel, it happened to be into one of the newly installed analog transmitter trip system panels, and it resulted in one of the SRVs opening. The SRV opened several times and closed and then opened and stayed open.

At the point that the SRV opened and stayed open, the operators scrammed the reactor. After the reactor was scrammed, the feedwater pumps -- initially when it was scrammed because the SRV was opened, the level went down, the feedwater pumps quickly recovered that level.

They tried to close the SRVS by pulling fuses, but the procedures were incorrect, they had the fuses listed incorrectly, and before they could finally locate the right fuse some 30 minutes into the event, the SRV closed by itself, and they don't know why. They don't really know exactly why the SRV opened.

They attribute it naturally to the moisture getting into the instrument panel. The moisture that went into the instrument panel, in addition to just causing the SRV to open, also caused eventually one of the power supplies that were in there to burn out.

The reason that the water dripped into the control

room was determined -- well, aside from the fact that the deluge valve went off, was that the vents that are in the plenum -- and the plenum looks like roughly this.

[Slide.]

The air conditioning plenum. The charcoal filter -- this is in the room above the control room, immediately above it, and here the ducting comes in from the outside of the building air intake, and here is the charcoal filter sitting here, and then the discharge line out into the control room goes this way and the intake back from the control room comes immediately into the edge here.

This is pretty much a straight drop into the control room.

Well, these drains which were located at the bottom of this box were plugged and the water filled up to this point, and at this point it flowed over this way and down into the control room [indicating].

The action that was taken subsequently was to dry out the instrument panels and replace the power supply and clean the air conditioning -- the drains for this particular box and they checked the auxiliary box and it did not have its drains plugged.

So --

MR. EBERSOLE: It would appear that just without the crane dragging its parts around, that there is a potential for

liquid egress from a system inside there, and therefore a straight potential to run right down into the control room from a water pipe, which is almost close to the old classical scenario of having a toilet on the ceiling which overflows after you flush it, and the whole set-up seems to be loaded.

Is that a characteristic of the general condition at Hatch?

MR. RIVENBARK: If the drains plug up and the water floods into the box, it is going to run over.

[Laughter.]

MR. LEWIS: The safety relief valve is electrically operated from that panel, so when you say you don't know what caused it, you mean you don't know exactly --

MR. RIVENBARK: They don't know how the water affected the circuits, what it did or exactly how it caused it --

MR. LEWIS: But the control for the circuit is at that point?

MR. RIVENBARK: Oh, yes.

MR. EBERSOLE: Don't we need to get the water ingress potential completely away from the panel board? Not just patch up the drain holes and all, but just get the potential out. And aren't there easy ways to do that?

MR. RIVENBARK: I'm not prepared to answer that.

MR. EBERSOLE: I guess what is going to be done is

just what you said, you leave it cocked again for this.

MR. RIVENBARK: Would you repeat that, please?

MR. EBERSOLE: I'm saying what I heard you say, you are going to fix the drains that were plugged --

MR. RIVENBARK: No, you didn't hear me say I was going to fix it. What you heard me say was that the drains were cleared, and having fixed the cause and put everything back into working order, the plant was allowed to restart.

MR. EBERSOLE: Yeah, right, so you fixed the drains.

MR. RIVENBARK: I do know that they are looking at the possibility -- the plant is looking at the possibility of removing the water altogether from the plant.

MR. EBERSOLE: Good. That's all I was after.

MR. REED: I want to make a point here, because we frequently talk about valves and malfunctioning. I believe he said that the safety relief valve, even though it got the signal, the fuses had been pulled or something, to go closed, it didn't.

I want to make a point that this was in -- even though this is a BWR, without boron in the water, that this was an internal pilot operated electromagnetic relief valve, and I have great concerns about their reliability on PWRs, but I find on BWRs they can be reliable.

MR. RIVENBARK: I would like to correct one little point. I did not say that they pulled the correct fuse. As a

matter of fact, I don't believe that they did. I think their inference to me is that they did not ever find the correct fuse before the valve went and closed by itself.

MR. REED: Thank you for that clarification.

MR. KERR: Do you feel better now about PORVs in BWRs?

[Laughter.]

MR. EBERSOLE: I think the next one now has to be considered in the previous light of what happened at -- oh, it was at Hatch also, wasn't it?

MR. JORDAN: No, this is the Oyster Creek.

MR. EBERSOLE: But there was an earlier one in which we had -- these are scram dump volume leakage phenomena.

MR. JORDAN: This is Dave Powell, one of the operations officers from I&E.

[Slide.]

MR. POWELL: Okay. Understanding that I have to be a little bit brief on this, I'm going to go ahead and forget about going over the event sequence, if that's okay.

I think there were quite a few members here last time when I went over it, so if there are any questions involving that, I will go ahead and entertain them.

MR. MOELLER: When it says that the plant is at 99 percent power, was it steady state there?

MR. POWELL: Steady state, 99.

Okay, this event is of interest to us because it mimics an event that occurred at Hatch in August of 1982, where they had a similar event which was an uncontrolled leakage out of their scram discharge volume drain valves.

The reason that occurred was again they were unable to clear the scram signal that they had in at that time which was a high drywell pressure. And in this particular case there were two valves in series. I will point them out right here, these two. In the Hatch event, there was only the one.

This was a backfit modification, these two particular valves. I'm not sure whether it was a generic letter that required that, but I understand most of the BWR plants put this in at their first refueling outage following that particular requirement.

In this particular event following the scram, and in the signal that was input, they were unable for 38 minutes to clear the scram signal due to a 600 pound interlock which seals in when they have an MSIV closure at this plant. That is a rather plant-specific interlock. Most of the BWRs do not have that.

These two particular valves failed. This particular valve, they both had different failure mechanisms. This particular valve failed to go fully shut. It went down to about 1/8th of an inch from its seat. So there was about a 1/8th inch gap right there between the seat and the disk.

This particular valve is hypothetized to have originally gone fully shut, but due to the pressure buildup from this leaking valve, it forced this particular valve open.

Now the reason that occurred was they had an improperly sized spring in the valve actuator. They had a 400 pound spring and it should have been -- well, I don't know exactly what it should have been.

What they decided to do to fix it was to put in an 1100 pound spring.

Those two errors combined allowed an uncontrolled leakage of reactor coolant. The estimate is about 500 gallons that flowed from the scram discharge volume to the reactor building drain tank. From there it flashed. The fumes came up through various vents in the reactor building at the 23-foot level, and the combination of the fumes from the paint that was blistering -- and I'm not sure exactly where the paint was. I believe the gentleman yesterday said it was upstream of the valves here -- and in combination with the steam it ignited off the fire protection deluge system. And as he indicated last time, it was on the 51-foot level, not the 23-foot level.

I think that lasted for approximately five minutes before they got that secured. There was no damage to any equipment inside the reactor building due to the actuation of the deluge system or due to the steam.

There was some radioactive contamination of the 23-foot level. Most of it was short-lived isotopes, and they had to recover.

The potential significance for this is, of course, the uncontrolled leakage of radioactive coolant outside of containment. They also had, like I said, the fire deluge system took off and actuated and, of course, any time you get water around electrical equipment -- I'm not sure if there was any in that particular area, but any time you have the system actuate, that potential is always there.

So it's a possibility at some other point, you know, some other event, that that kind of a problem may show up, and you might lose some secondary balance-of-plant or safety-related equipment.

The third thing that occurred was we had CRD seal high temperature alarms. They came in intermittently and the problem with CRD high temperature alarms is not so much a problem for this particular event. It is for later operation of the plant. Should those seals degrade due to the high temperatures, there's a possibility that they would be unable to operate properly, either withdraw or insert properly or possibly scram properly due to the degradation of the seals.

The two causes, as I noted earlier, was -- I spoke about these two particular failure mechanisms. I believe the ultimate problem -- and I have to back up a little bit. There

is no requirement to leak-rate-test these valves, all right? These particular valves are categorized as B valves, Category B valves under ASME Section 11 and because of that, these particular valves were never leak-rate-tested after the post-installation of the backfit mods, and had they done that -- you know, 20/20 hindsight -- these particular problems would have been probably picked up during the leak rate test.

Basically what they did when they put it in was just hydroed the system to whatever the requirement was. To do that, I would imagine they blank-flanged off the outlet valve, opened up these two and just tested the piping within that whole system as required under their ISI inspection program.

MR. EBERSOLE: You had mentioned earlier, and I think the Full Committee would like to know, that the rationale for not testing these is that the minimum boundary is considered to be at the rod seal itself, and all this junk down here is just standard equipment, when in fact that's not real life.

MR. POWELL: Well, I have the NUREG that addressed that particular problem, and the Staff words on that particular matter. Initially they came out and indicated that the boundary should be at the scram discharge pilot valve, which would be this, all right? This would still be outside of that particular category A system, safety-related type portion of it.

MR. EBERSOLE: Yes, but there's been a tremendous controversy about potential leakage from the scram dump volume even due to metallurgical failure of the volume proper, not to mention valve failure as being a potentially serious event which should have logically caused these valves to have been recognized as safety-related valves.

MR. POWELL: Well, I'm not sure how each plant chooses to look at their scram discharge volume system, when it comes to the requirements, but --

MR. EBERSOLE: Isn't it a standard problem for all BWRs?

MR. POWELL: I can't answer that. We haven't seen this type of problem, other than the Hatch event, which I guess precipitated putting on the series valves, hopefully to preclude this kind of problem from occurring.

As far as the probability studies that were done on the system, in terms of core melt sequence, this particular NUREG addressed that also, and the conclusion was that failure of the piping -- and that's what they addressed, not the valves at that time -- would not lead to -- let's put it this way:

The probability was not high, compared with other types of events that would lead to core damage.

MR. EBERSOLE: Oh, yes, the probability of piping failure was virtually zero.

MR. POWELL: Well, they said it was less than 10 to the minus 6 per plant year, so --

MR. EBERSOLE: But they ignored the valves standing right in the pipe.

MR. POWELL: Beg your pardon?

MR. EBERSOLE: They ignored the presence of the valves which made an aperture through the pipe.

MR. POWELL: Okay. But I don't know. We're putting out an information notice to show people that these kinds of events can occur. I would imagine that most utilities -- and I would say probably mostly the -- I should say the prudent utilities -- will probably go ahead and change their IST program to include those in a leak rate test on a yearly basis.

But again they are still not under any requirement, according, you know -- as far as I know, to do that.

MR. EBERSOLE: The Staff will not upgrade these valves? That's not --

MR. POWELL: Well, I can't speak for them. I understand that NRR is not really pursuing this matter much more at this point, but they might. I don't know.

MR. JORDAN: There is not a decision yet to upgrade the valves. The prompt action is to put out the information notice and then make a subsequent decision on whether action is needed.

MR. EBERSOLE: By the way, I don't believe you told us how they stopped this leakage. There's a little note here. Did they try to blow down the system and get the pressure down to reverse the interlock, and then did that fail or what?

MR. POWELL: Well, as part of their normal procedures for shutting down under MSIV closure, normally the isolation condensers would have automatically set off for them. The operators had to override that function because of the high level that they had in the vessel at the time.

So what occurred was they got a high level with high pressure and the electromagnetic valves, the A and D valves, automatically actuated.

That basically blew down the system. It also, later on in the event, they started up the reactor water clean-up system and it began the letdown portion of that to the condenser. That eventually resulted in some other problems. I don't want to say problems, but they started getting oscillations in the levels and they got other scram signals that they picked up.

But, you know, I think that they had the plant pretty much under control. They just couldn't isolate -- they couldn't override the signal with the load switch at that time until they got the pressure down below 600 pounds.

MR. JORDAN: The next event is the Rancho Seco

reactor coolant system high point vent leak. Howard Wong of I&E, who is one of the reactor construction engineers --

MR. EBERSOLE: I think the Full Committee should note that this is an event borderline to a one-inch small LOCA on a B&W plant.

MR. JORDAN: It is appropriate that Howard Wong should give the presentation. Number one, he was responsible for the Bulletin 79-14, which was the "as-built" review, and two, he went to Rancho Seco to assist Region V in follow-up of this particular event, so he has physically been around this piping and looked at it and followed upon the Licensee's action.

MR. WONG: Let me set up the conditions that were initially in the plant at the time of the event. The plant was in hot shutdown. The correction on the slides says hot standby was hot shutdown, the difference being that it was a subcritical mode. We were starting up from a refueling outage.

The event was a 20-gallon per minute, non-isolable primary coolant system leak on the high point vent system on the B steam generator.

Initial identification of the leak was by personnel who happened to be inside the containment who heard a pop noise. On exiting containment, control room personnel were informed, and by remote operated cameras inside containment,

located the general location of the problem.

Control room personnel identified a small steam leak using this camera. Additionally, the containment entry was made to precisely locate the source of leakage, and it was identified as a non-isolable leak. At that point in time, the normal cooldown procedure was ensued.

The actual leak portion occurred in a TMI modification that was performed in the 1983 outage, which was the high point vent lines.

What can be seen here is the RCS loop top of the candy cane and the additional piping above was original design.

What occurred was a 120 degree through-wall crack just below the weld. The enlarged diagram shows it, and actually it was located on the outside portion of the T. This weld was made during the 1983 outage. The old welds were this piece in here and additionally up above.

The cause appears to be missing supports and fatigue and therefore fatigue failure.

As a result of the RCS vent line addition, two additional pipe supports had to be modified, and an addition of one cross-brace member was required. Investigation revealed that these support changes had not been performed, although records stated to the fact that work had been done and, indeed, inspected.

[Slide]

I would like to show exactly what modification was made, and this is perhaps a little more detailed than the handout that is provided as an isolated sheet.

The modification that was supposed to be made was additional lateral bracing. Let me set it up first as the piping system. Here is the RCS loop, the nozzle coming off. The TMI modification was this bottom line coming around. That is the vent line.

The line that was the old piping was nitrogen supply line, only used during refueling outages.

MR. KERR: Mr. Wong, that is an impressive diagram, but what am I supposed to -- what are the salient points that I am supposed to learn from this presentation?

MR. WONG: The important points here are this is the vent line that was added for TMI. Somewhat darker are the modification work that was supposed to be made. The actual break --

MR. KERR: So what I should learn is that somebody didn't do what they were supposed to do?

MR. WONG: That's correct. The supports that were affected were these here, this cross-brace here, and also notice this spool piece, removable spool piece in the middle.

MR. SIESS: Where was the crack?

MR. WONG: The crack is down in this portion here,

right off the nozzle from the --

MR. SIESS: All that new stuff doesn't have any braces on it?

MR. WONG: It does, but they are not shown on this diagram. It does have pipe supports.

MR. SIESS: And this is braces on the old piping?

MR. WONG: That's right. This is more just to show the pipe supports off of the existing system. It is important to note --

MR. SIESS: Even if this change hadn't been made, there was a nozzle there with pipe going into it before they made it into a high point vent, right?

MR. WONG: That's correct. If you take away this portion, basically this is as was designed originally back in 1974.

MR. SIESS: So that nozzle would have failed or that pipe would have failed even if this additional pipe hadn't been put on it?

MR. WONG: Well, it's not so much additional pipe. The point I want to get -- and maybe I can jump ahead right now. The important is this removal of the spool piece in the middle. As designed -- and this was to provide that the nitrogen system would not be contaminated during operation, so it was planned to be removed during operation, during outages, and would be in place for nitrogen blanket purposes. And if

that would have happened, the purpose of this was to transmit loads across both sides of the pipe.

The dummy spool piece was designed so that it would go back during operations, basically giving rigidity to the pipe, decreasing flexibility so that loads would be transmitted across.

MR. KERR: Let me ask again what I think Dr. Siess was asking. Suppose that the TMI change had never occurred. Would this leak have occurred anyway?

MR. WONG: I think so. The fact of the matter is that during operation, this piece was not here, basically leaving the one-inch pipe with about 60 pounds of distributed weight, including a flange at this end, basically as a cantilever.

MR. SIESS: It wasn't the new piping that put the stress on there; it was the old piping.

MR. WONG: It was most likely probably the fact that this was not adequately supported. Without this spool piece, it was hanging out as a cantilever.

MR. SIESS And that's the old pipe.

MR. WONG: That's correct.

MR. KERR: This is referred to as a high point vent leak. The high point vent was not even there before TMI-2.

MR. WONG: Well, it is an existing system as established, as it was last month, the vent line was there.

MR. EBERSOLE: Yes, but originally it was not called that. It was the nitrogen supply line.

MR. SIESS: But when did they take that spool piece out?

MR. WONG: It looks like from investigation now -- this was actually designed back in '74, this spool piece, to be removed during operations.

MR. SIESS: When did they take the spool piece out?

MR. WONG: They have always put it back during refueling outages, so it appears from commercial operations -- during an outage it was there, but during operation, it was not, although a dummy piece should have been put there.

MR. SIESS: Now, was the weld that cracked in there all that time subjected to the vibration and stress from the spool piece?

MR. WONG: No, it was not. I will get back to the original slide that I had of the weld.

MR. SIESS: Well, the original weld didn't fail; it was some new one that was put in?

MR. WONG: That's right. This is the old weld. It's a small pipe piece. Here is the T up in here. This weld here, where the crack was initiated, it was a 1983 weld.

MR. SIESS: Well, why didn't the old one fail if it was the old configuration that caused the vibration?

MR. WONG: There is a postulation that this is the

stainless to Inconel weld at this point, and this is stainless to stainless.

MR. KERR: Okay. So it probably wouldn't have occurred had the TMI-2 change not been made because you wouldn't have had the stainless-to-stainless weld.

MR. WONG: That is one postulation, that's right.

MR. SIESS: You could think that way if you wanted to.

MR. EBERSOLE: I think you are driving into that point with some effort.

MR. WONG: The metallurgical examination of the piping that had cracked showed evidence typical of high cycle fatigue. The consultants which Rancho Seco used -- Bechtel, GE, IMPEL and their own metallurgists, agree to that fact. The primary crack is seen to be transgranular in an area of high residual stress. Although it cannot determine the point of crack initiation or the direction of propagation, evidence characteristic of stress corrosion is missing.

MR. EBERSOLE: Well, what you are doing is going through the physical details of why it failed, but I think the real point of essence is did the paper records show that all these pipes were in place and the hangers were in place and all was in order?

MR. WONG: That is correct.

MR. EBERSOLE: Well, the fact that that was not true

was the real root cause of this problem, wasn't it?

MR. WONG: That's correct. If it had been implemented as designed, it would probably have not failed.

MR. SIESS: So if the paper says it is there, Jesse -- if the paper says it is not there, you know it's not there. If the paper says it is there, it must be there. But somehow we have got to tell that pipe --

MR. EBERSOLE: That's right, we've got to tell the pipe it's got to be there.

MR. SHEWMON: What sort of administrative action is likely to be taken against the person who signed off on it incorrectly?

MR. WONG: From the licensing standpoint, I can't make a statement on that. I will get into corrective actions, and that might clarify a little bit. Licensee actions, I don't really know.

MR. SHEWMON: It is not a misdemeanor.

MR. WONG: I don't know.

MR. KERR: Well, that's QA. That doesn't have anything to do with what is in the plant.

MR. WONG: It's a combination of not just QA/QC but also the field engineer in this mode that had a package to be sure it was properly done. The field engineer did sign off, and then, of course, QC signed off.

MR. SIESS: You mean QC signed off on the paper

without looking at it? The only person that is supposed to look at it is the field engineer, and everybody else just looks at paper?

MR. WONG: That is not correct.

MR. SIESS: I hope it isn't. It ain't right.

MR. EBERSOLE: I understood that you could reach up with your hand and swing this pipe back and forth a foot or two; is that right?

MR. WONG: I don't know about that. It was cut up by the time I got there. Dead weight analysis on this piece of pipe in the "as-built" condition shows a deflection of .8 inches at the end of the piping run, at the end of the flange, just on dead weight alone.

MR. SHEWMON: Well, before you leave that one, where did the crack start, outside or inside?

MR. WONG: On the outside portion.

MR. SHEWMON: Then stress corrosion cracking isn't really germane.

MR. WONG: It was on the outside. It appears to be basically --

MR. SHEWMON: Well, you have answered the question. Go on.

MR. WONG: An additional point I want to make. There are axial indications along this full piece, the small portion -- it might be only about three-quarters of an inch.

There are some slight axial indications. They are still trying to investigate exactly what that is. They might be, they consider, possibly from the manufacturing process.

MR. EBERSOLE: I think this is probably as far as the Full Committee wants to hear this in light of the time requirements here, unless I am wrong. Is the Full Committee satisfied with this degree of presentation here?

MR. WARD: Yes.

MR. EFERSOLE: I think we can terminate this in the interest of time and jump to the next one.

MR. JORDAN: The next one is Sequoyah Unit 2, and --

MR. SIESS: I think the one we just heard about is a QA problem that is much more interesting than the fact that an overstressed pipe failed with fatigue, which is sort of nice to know since that's the way we expect them to fail.

MR. EBERSOLE: This is very brief here but very pertinent, this one.

MR. WEISS: Sequoyah Unit 2 tripped on May 22nd from 100 percent power on overpower delta T. This event demonstrates how a plant can trip following an approved procedure, although a new one, despite all the precautions that are in place to prevent maintenance or surveillance activity from causing this sort of thing, such as communication between the control room and the people performing the surveillance or maintenance activity.

The particular surveillance or maintanance activity under way at this time was the primary system calorimetric which was being performed for the first time. An instrument technician had to take temperature readings from four protection sets. The protection cabinets were located approximately 15 feet apart, and he was using a digital voltmeter to take the readings.

MR. LEWIS: Reading what?

MR. WEISS: Temperature.

MR. LEWIS: Well, what was he reading?

MR. WEISS: He was supposed to be reading voltage; and in fact, he had the leads connected to the digital voltmeter in the ammeter connections.

MR. LEWIS: Mr. Ebersole just said RTDs. Is that correct?

MR. WEISS: Yes.

MR. SHEWMON: But you missed his point. He had them in the amp --

MR. WEISS: So the internal resistance of the voltmeter was much lower than it should have been. He had it connected in the ammeter holes instead of the voltmeter holes. So he tripped one channel, and since he is doing a calorimetric and he has to get a snapshot of the plant conditions, he has to go to all four protection sets within three minutes.

So he closes the door on one cabinet, moves to the next cabinet, opens that door, puts in the probes and takes a reading. Although the reactor operator in the control room noticed the dropping and noticed the trip, it happened too quickly and the plant went down on two out of four coincidence.

MR. SIESS: Well, that is improper use of test equipment. You could have labeled it use of improper test equipment just as well. I mean if you checked out a voltmeter and somebody was checking on the other end, he couldn't have done this. So you say all precautions had been taken. I think there is one I could have thought of.

MR. LEWIS: Well, you know, they blamed the other voltage trip on the same thing.

MR. WEISS: Yes. You will recall there was an event similar to this that we discussed before the Full Committee some time ago where the use of a digital voltmeter caused a shorting of the output transistors and the RPS system, and we issued an information notice on the subject. The corrective actions taken for this particular event include a precaution about procedures not only regarding the consequences of making a mistake but also for looking for the proper expected values of voltage before proceeding to the next piece of equipment.

MR. EBERSOLE: It would appear that TVA should outlaw multipurpose meters.

MR. KERR: Well, there is also something to be said for doing this sort of testing when the plant is not operating, it seems to me.

MR. EBERSOLE: But this was a full power thermal measurement.

MR. KERR: It doesn't have to be full power. It's a temperature measurement, Jesse. That can be made at other than full power.

MR. EBERSOLE: Well, this was a thermal heat balance. They were on line. As a matter of fact, you remember you said they had to do it within what?

MR. WEISS: Within three minutes.

MR. EBERSOLE: Three minutes channel to channel.

MR. WEISS: Yes, and they were taking their readings off of temperature resistance to voltage modules, they are called.

MR. KERR: Well, you can surely take thermal readings without voltmeters.

MR. EBERSOLE: This is high and low temperature. They are getting delta T. I think they had better outlaw multi-meters.

MR. LEWIS: Well, or do less testing.

MR. SHEWMON: But Glenn will tell you these people were selected maybe by not the right criteria and they should be capable of knowing what a voltmeter is and what an ammeter

is.

MR. EBERSOLE: It's the people who select the meters.

I think that does it. Thank you very much.

MR. WARD: Okay. We are finished with that.

I suggest that the subcommittee that has to leave could leave now.

MR. EBERSOLE: Thank you, Ed, and all your fellows.

MR. JORDAN: By the way, t.. ACRS has fed back to the Staff the "thumby maintenance." We are using that word now.

MR. WARD: This will complete the transcript for today.

[Whereupon, at 5:27 p.m. the reported meeting was concluded.]

CERTIFICATE OF OFFICIAL REPORTER

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Name of Proceeding: 303rd General Meeting

Docket No.:

Place: Washington, D. C.

Date: Thursday, July 11, 1985

were held as herein appears and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission.

(Signature) Suzanne B. Young

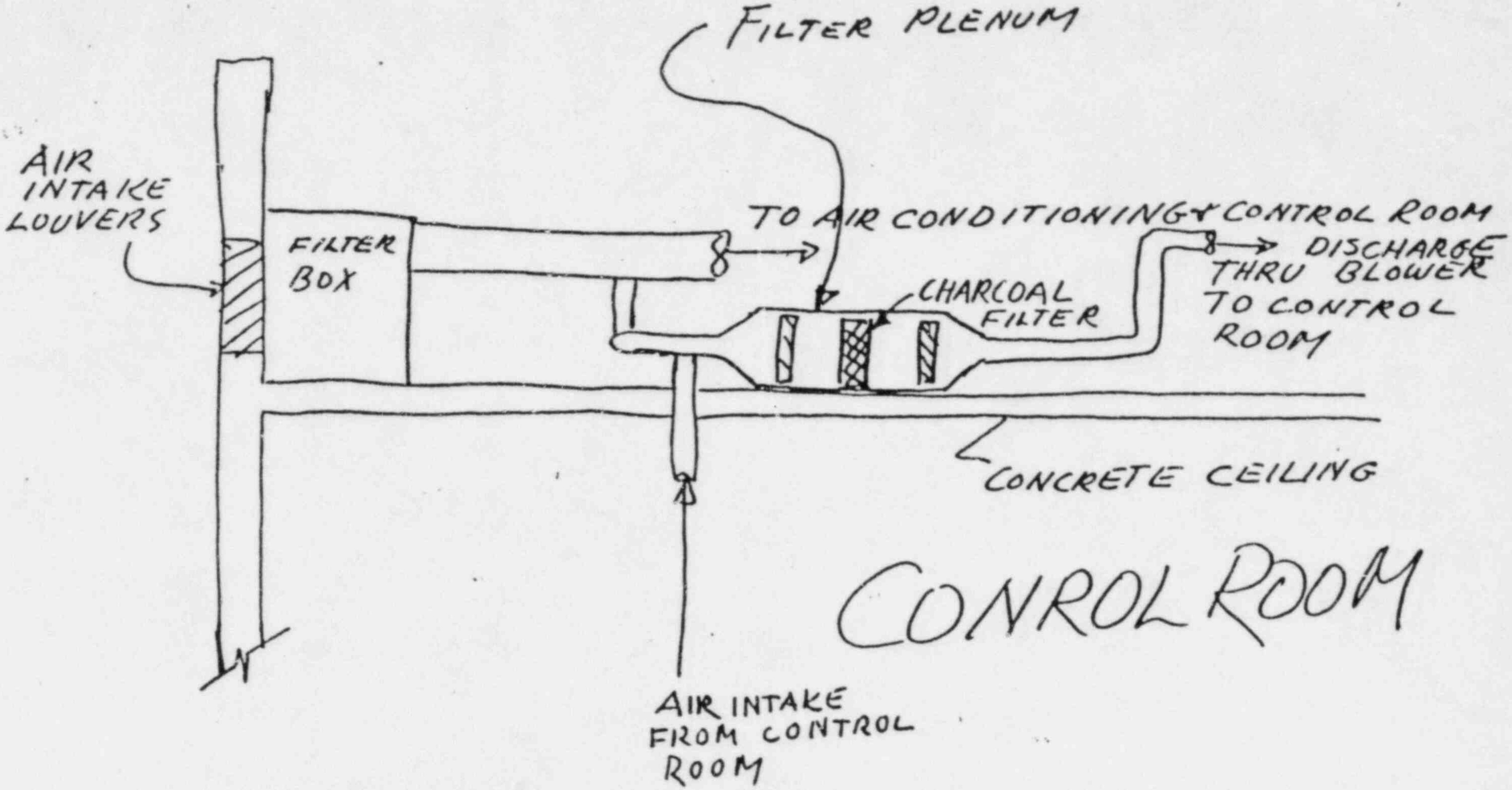
(Typed Name of Reporter) Suzanne B. Young

Ann Riley & Associates, Ltd.

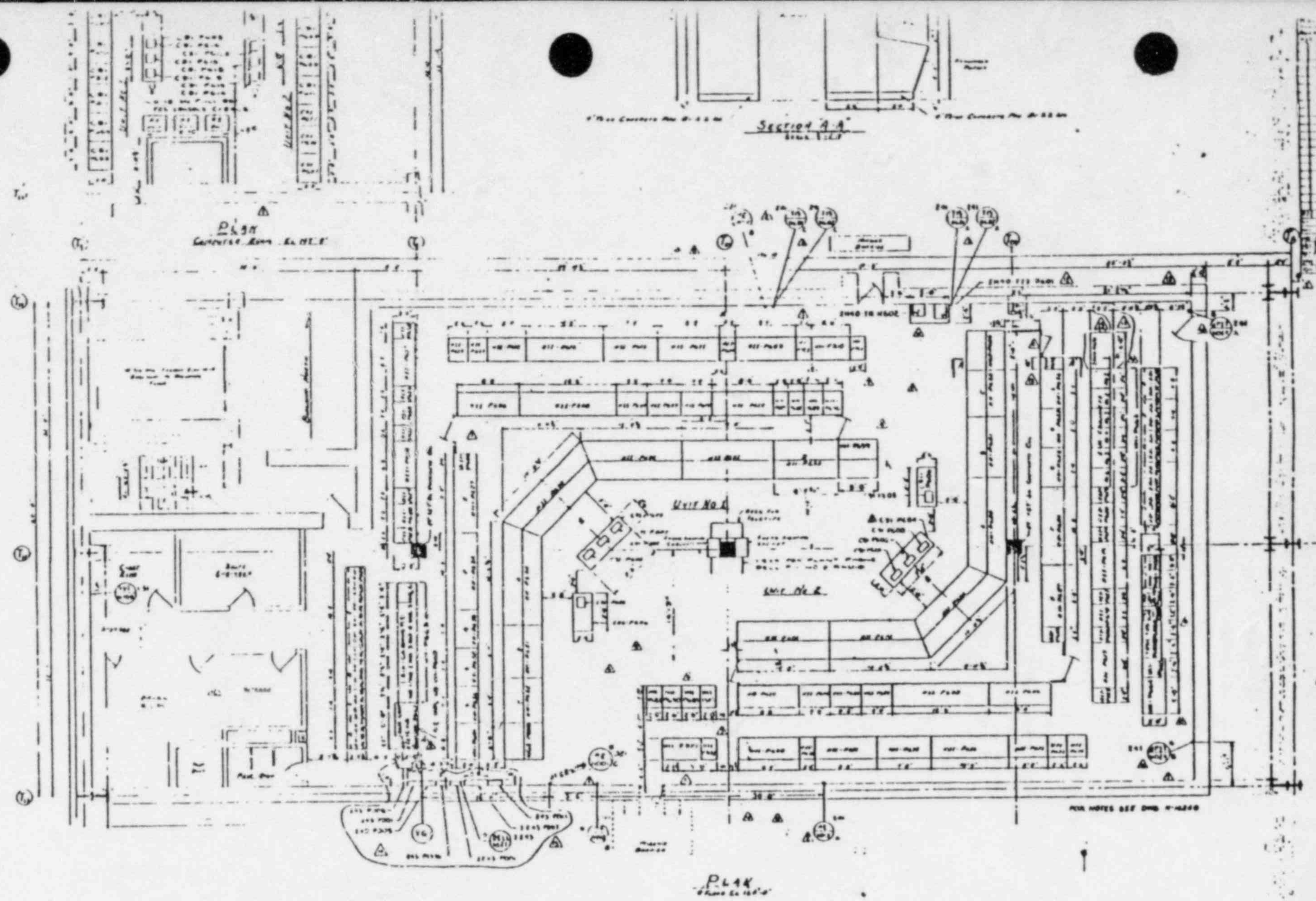
HATCH UNIT 1 - STUCK OPEN SAFETY RELIEF VALVE OF MAY 15, 1985

(G. RIVENBARK)

- A SYSTEMS INTERACTION EVENT
  - UNIT 1 OPERATING AT FULL POWER
  - CONTROL ROOM EMERGENCY VENTILATION SYSTEM CHARCOAL FILTER DELUGE VALVE ACTUATED
  - WATER LEAKED THROUGH VENTILATION DUCTS INTO A HATCH UNIT 1 ANALOG TRANSMITTER TRIP SYSTEM (ATTS) INSTRUMENT PANEL CAUSING SRV TO OPEN
  - REACTOR MANUALLY SCRAMMED
  - FEEDWATER PUMP RECOVERS REACTOR WATER LEVEL
  - SRV CLOSED - WITHOUT OPERATOR ACTION
- CAUSE
  - LOSS OF INSTRUMENT WATER SUPPLY CAUSING DELUGE VALVE TO OPEN TOGETHER WITH PLUGGED DRAINS
  - NOT SURE HOW WATER CAUSED THE SRV TO OPEN
- ACTION
  - REPLACED ATTS POWER SUPPLY, CLEANED PLUGGED DRAINS AND INSPECTED DRAINS IN REDUNDANT FILTER UNIT
  - LICENSEE PROPOSES TO ADD CLEANOUT CHECK PROCEDURES FOR PLENUMS AND THEIR DRAINS
  - ORAB WILL DEVELOP TIA TO COORDINATE:
    - IE NOTICE
    - FURTHER INVESTIGATIVE EFFORTS
    - GENERIC REVIEW


FILTER PLENUM
AIR INTAKE LOUVERS
FILTER BOX
TO AIR CONDITIONING & CONTROL ROOM
DISCHARGE THRU BLOWER TO CONTROL ROOM
CHARCOAL FILTER
CONCRETE CEILING
CONROL ROOM
AIR INTAKE FROM CONTROL ROOM

W

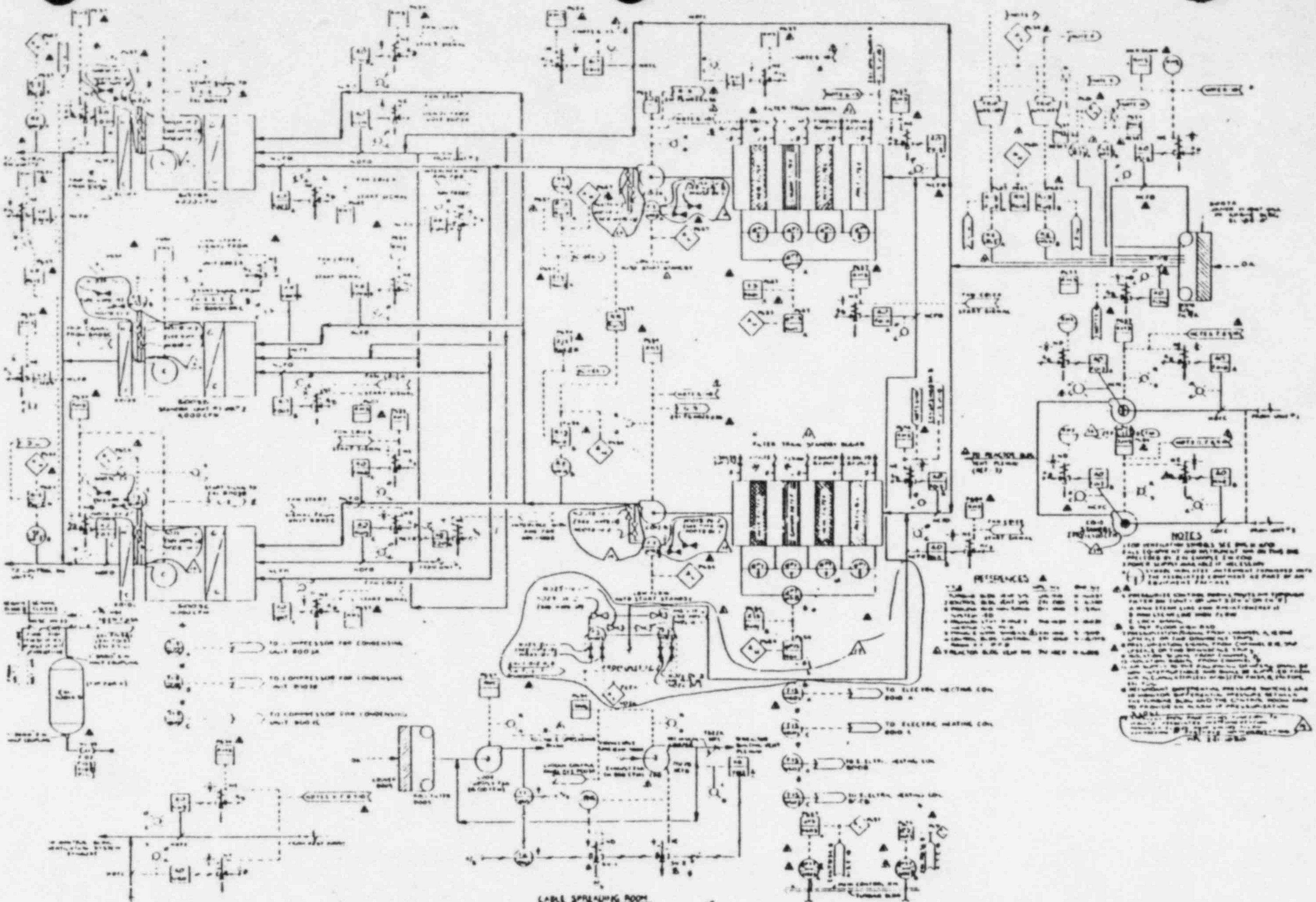


Georgia Power

EDWIN I. HATCH
NUCLEAR PLANT – UNIT

FIGURE 6.4-1

4



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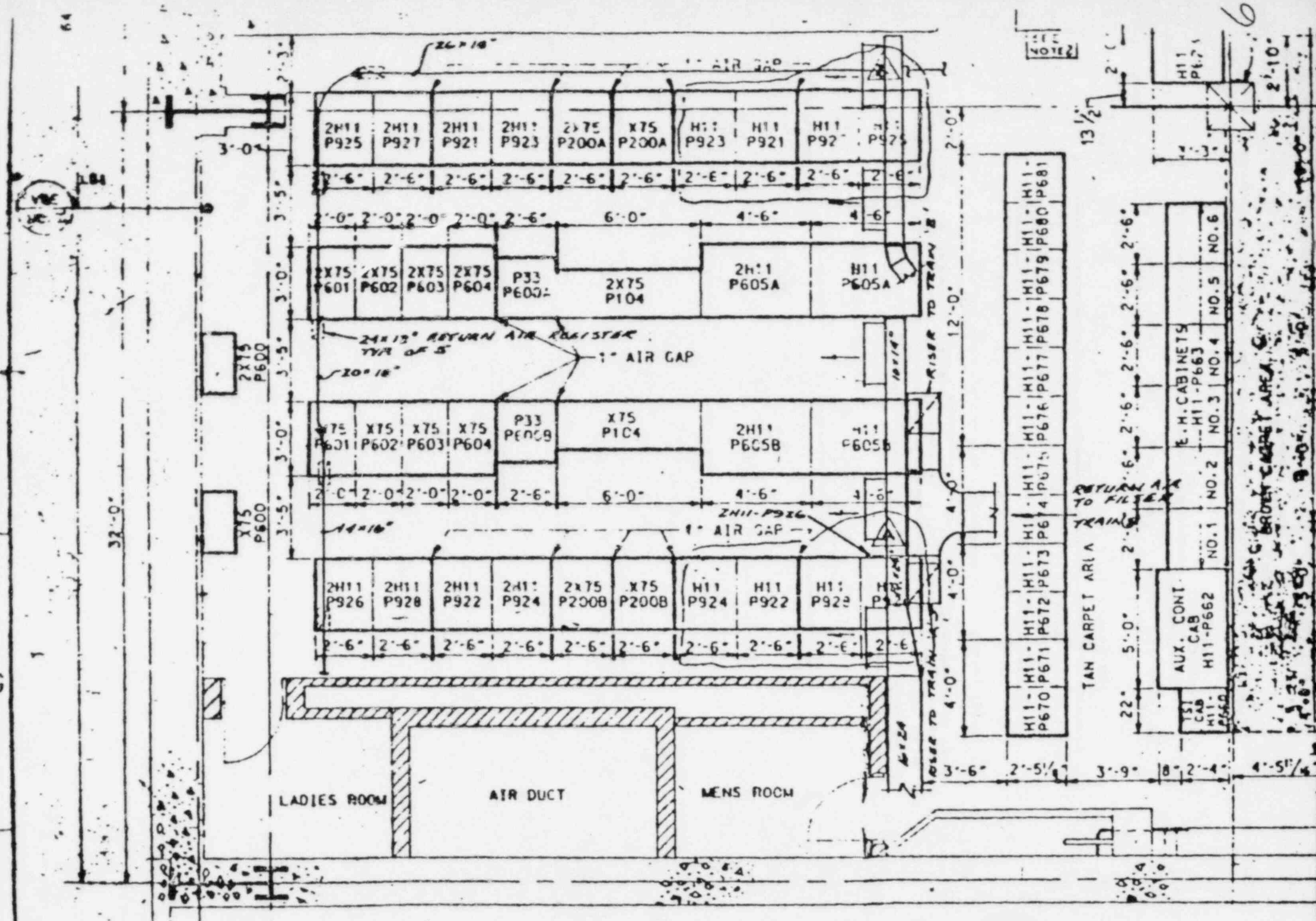
Georgia Power

EDWIN I. HATCH
NUCLEAR PLANT - UNIT 2

CONTROL ROOM VENTILATIO
TEMPERATURE CONTROL D

FIGURE 9.4-1 (SHEET



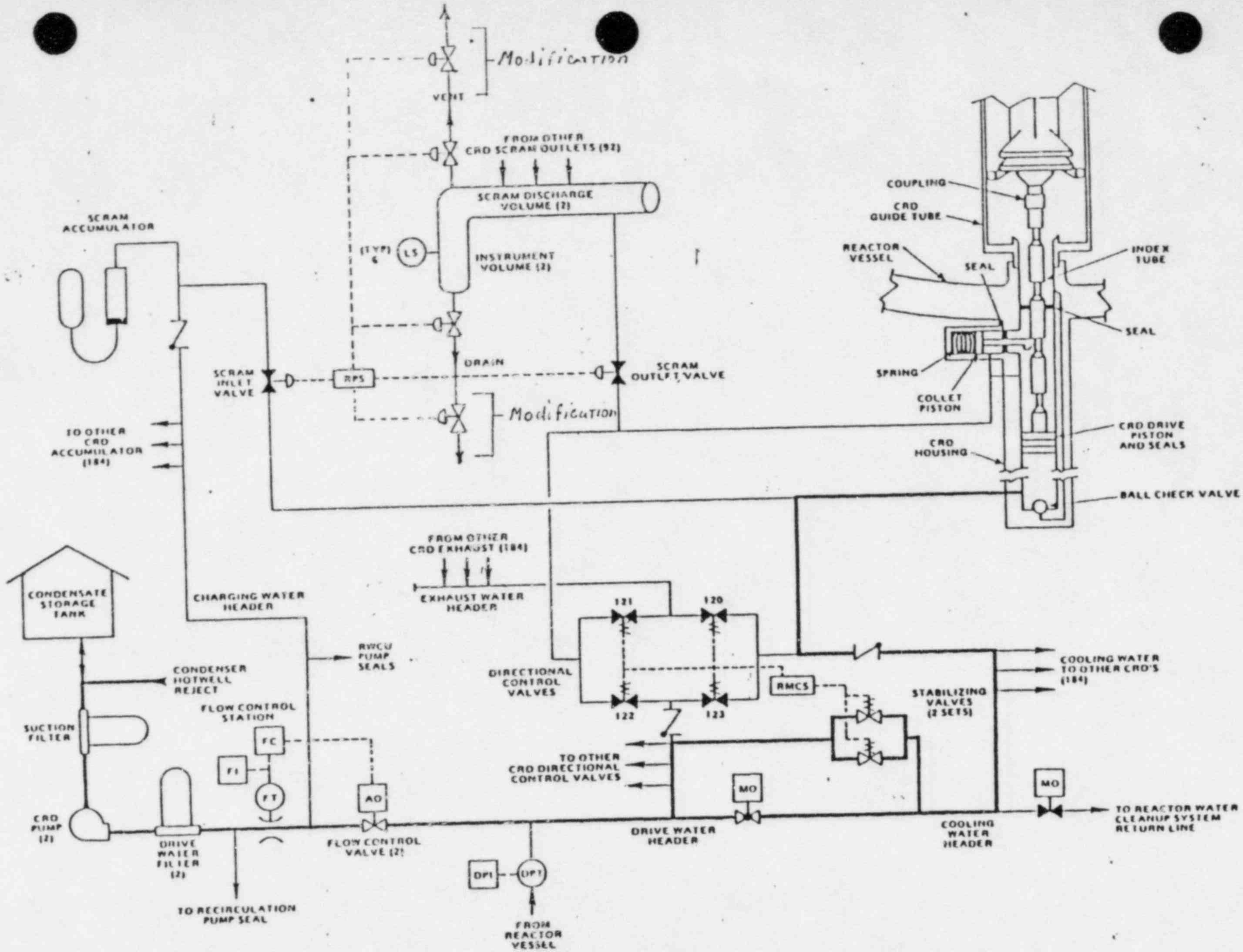
PLAN VIEW

DUCTWORK/ATTS PANELS

OYSTER CREEK - UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE CONTAINMENT

JUNE 12, 1985 (D. POWELL, IE)

- WITH REACTOR AT 99% POWER, FAILURE OF THE ELECTRIC PRESSURE REGULATOR CAUSED A TURBINE BYPASS VALVE TO OPEN RESULTING IN A REACTOR PRESSURE DECREASE, FOLLOWED BY MSIV CLOSURE AND REACTOR SCRAM.
- SCRAM DISCHARGE VOLUME DRAIN VALVES FAILED TO FULLY SHUT/SEAT CAUSING REACTOR COOLANT TO BE DISCHARGED TO THE REACTOR BUILDING DRAIN TANK.
- RELEASE OF STEAM FROM FLOOR DRAINS AND PAINT BLISTERING ON HOT PIPE CAUSES PORTION OF REACTOR BUILDING DELUGE SYSTEM TO ACTIVATE
- SCRAM SIGNAL NOT RESET FOR 36 MIN ALLOWING CONTINUOUS REACTOR COOLANT FLOW TO THE DRAIN TANK. CAUSE WAS 600 PSI INTERLOCK ON MSIV CLOSURE/LOSS OF CONDENSER VACUUM.
- SAFETY SIGNIFICANCE - (1) LOCA OUTSIDE CONTAINMENT, (2) POTENTIAL EQUIPMENT MALFUNCTION DUE TO FIRE DELUGE SYSTEM, (3) EXCESSIVE CRD SEAL TEMPERATURES.
- CAUSE-VALVE SPRING ON VALVE V15-134 (VELAN) VALVE UNDERSIZED
  - -VALVE V15-121 (VALTAK) STROKE DISTANCE INSUFFICIENT TO TIGHTLY SEAT THE VALVE. (1/8" OPENING)
  - -IMPROPER POST-INSTALLATION TESTING OF VALVES
- CORRECTIVE ACTIONS - REPLACED 400 LB SPRING WITH 1100LB SPRING, ADJUSTED VALVE STROKE DISTANCE CHECKED CRD SEALS FOR DAMAGE, CHECKED EQUIPMENT, NO DAMAGE FOUND.
- NRC FOLLOWUP ACTION - IE NOTICE IN PREPARATION.



(Typical)
FIGURE 2.3-1 CONTROL ROD DRIVE HYDRAULIC SYSTEM

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RANCHO SECO - RCS HIGH POINT VENT LEAK

JUNE 23, 1985 (H. WONG, IE)

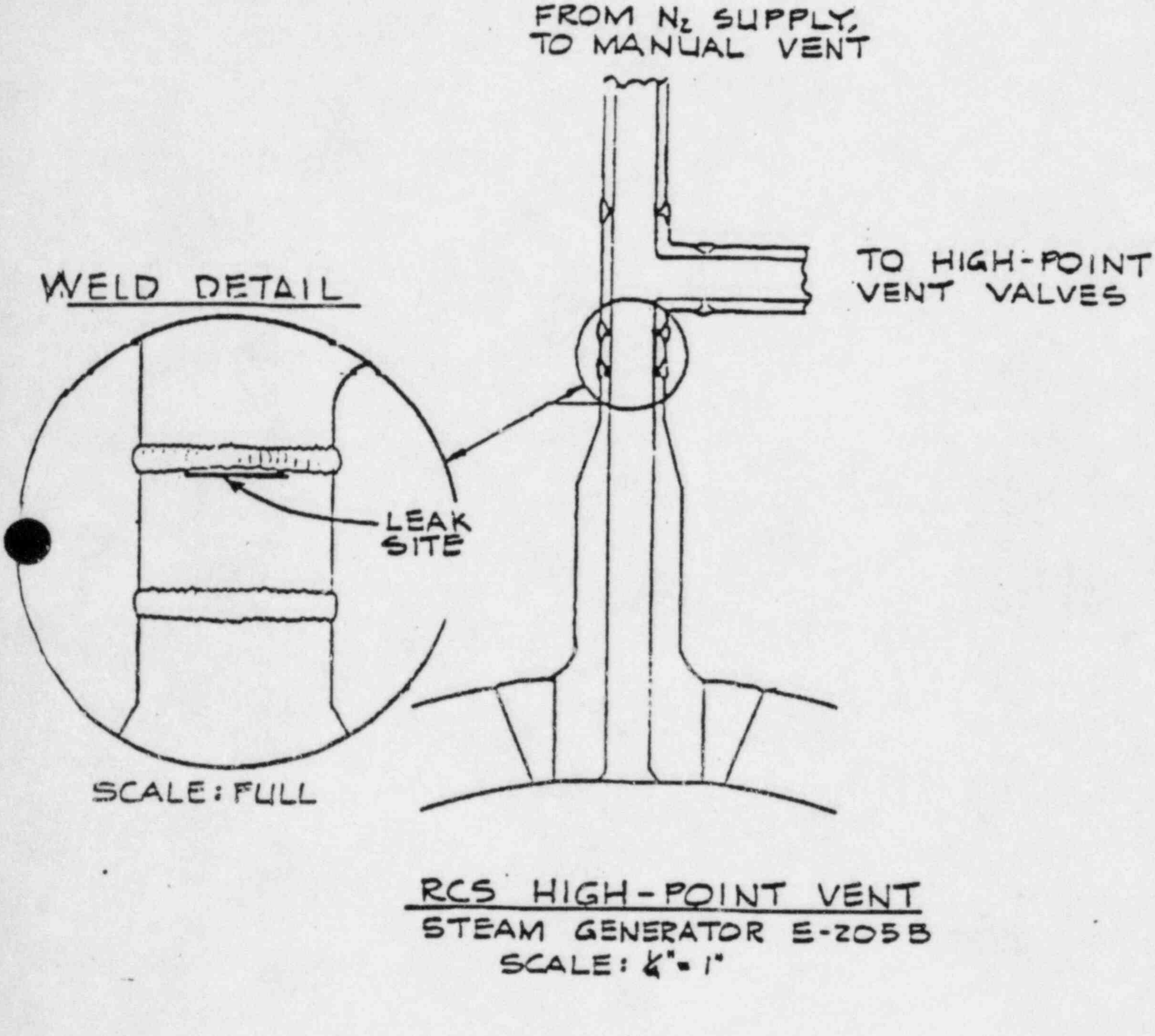
- PLANT IN HOT STANDBY RESTARTING FROM A REFUELING OUTAGE
- 20 GPM NON-ISOLATABLE PRIMARY COOLANT LEAK ON HIGH POINT VENT ON B STEAM GENERATOR HOT LEG
- TMI MODIFICATION INSTALLED 1983 REFUELING OUTAGE
- 120° THRU WALL LEAK AT WELD
- CAUSE APPEARS TO BE MISSING SUPPORTS AND FATIGUE FAILURE
- LICENSEE ACTIONS:

  STRESS ANALYSIS TO IDENTIFY OVERSTRESSED AREAS (BOTH HOT LEG VENTS)

  REPAIR SYSTEMS

  INSTALL SUPPORTS

  WALKDOWN TO INSPECT AND EVALUATE OTHER SYSTEMS
- REGION V, IE TEAM PARTICIPATING IN WALKDOWN.



RCS HIGH-POINT VENT
STEAM GENERATOR E-205B
SCALE: ¼" = 1"

SEQUOYAH 2 - IMPROPER USE OF TEST EQUIPMENT AND REACTOR TRIP

MAY 22, 1985 (E. WEISS, IE)

- REACTOR TRIPPED FROM 100% POWER ON OVERPOWER DELTA T
- DIGITAL VOLTMETER (DVM) USED TO MEASURE T_{HOT} AND T_{COLD}
- LEADS FROM DVM INCORRECTLY CONNECTED TO ONE RPS CABINET
- BEFORE TIME CONSTANT RECOVERS, LEADS FROM DVM INCORRECTLY CONNECTED TO 2ND RPS CABINET

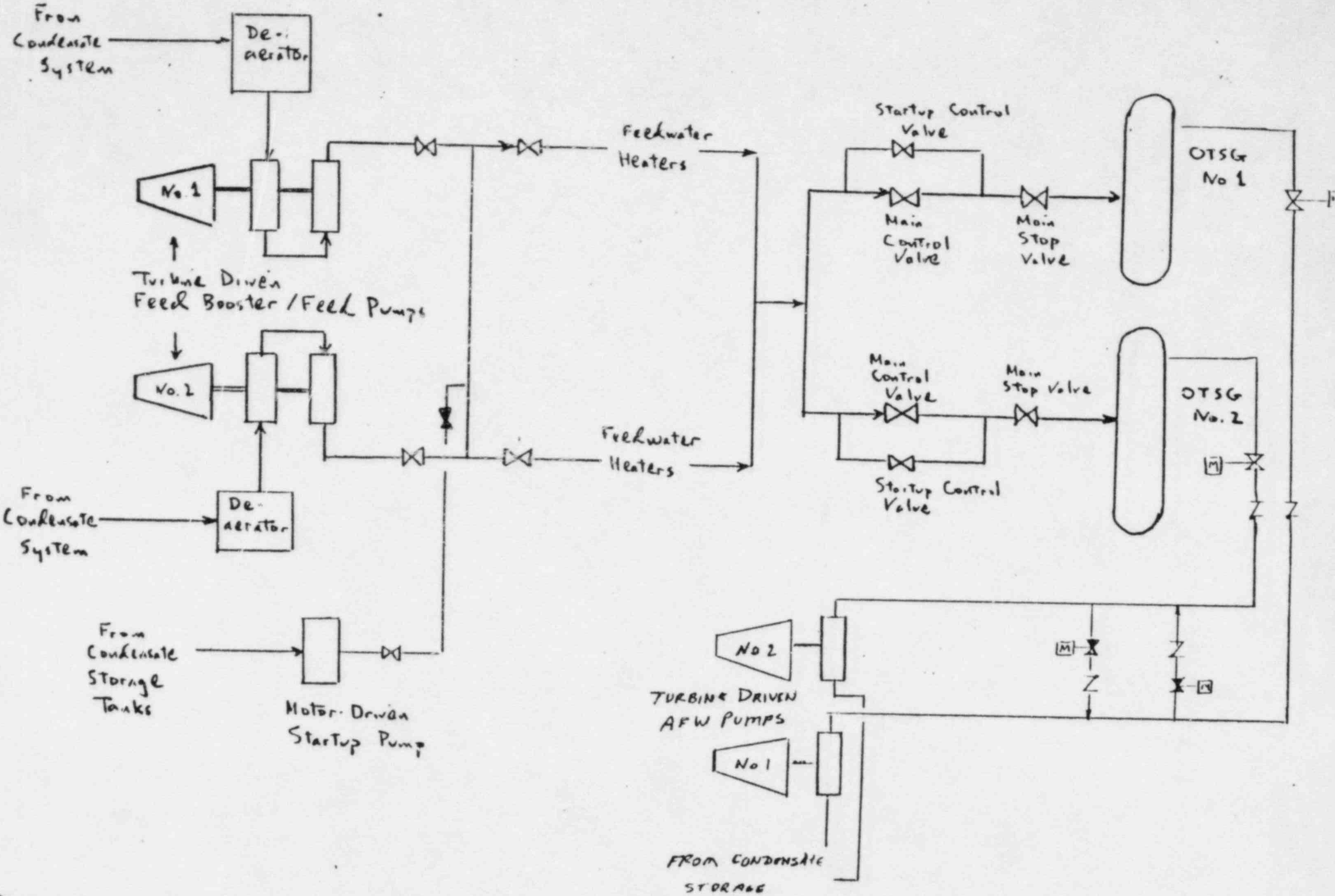
DAVIS-BESSE - LOSS OF ALL MAIN FEEDWATER AND AUXILIARY FEEDWATER

JUNE 9, 1985 (A. DEAGAZIO, NRR)

- Loss of one main feedwater pump at 90% power.
- Reactor trip at 78% power on high pressure.
- Both MSIVs close spuriously tripping remaining main feedwater pump.
- Steam generator low level starts both auxiliary feedwater pumps but both trip on overspeed.
- Operator erroneously manually trips SFRCS on low pressure
- No feedwater available for about eight minutes and steam generator levels fell to about eight inches.
- PORV cycles three times - did not reseat on third cycle, operators close block valve.
- Startup feedwater pump used to feed one steam generator.
- Operators restart auxiliary feedwater pumps and restore normal post-trip conditions.
- No indication that subcooling margin was lost or that reactor coolant activity was abnormal.
- Plant now in cold shutdown.

KNOWN EQUIPMENT FAILURES OR MALFUNCTIONS

- Main feedwater trip
- Spurious half trip of SFRCS
- MSIV closures (2)
- AFW pump trips on overspeed (2)
- AFW isolation valves fail to open (2)
- PORV failure to reseat
- SUFP valve failure to open
- AFW speed governor after reset
- Switchover to service water backup supply
- Damaged turbine bypass valve


From Condensate System
De-aerator
No. 1
Turbine Driven Feed Booster / Feed Pumps
No. 2
From Condensate System
De-aerator
From Condensate Storage Tanks
Motor-Driven Startup Pump
Feedwater Heaters
Feedwater Heaters
Startup Control Valve
Main Control Valve
Main Stop Valve
OTSG No 1
Main Control Valve
Main Stop Valve
Startup Control Valve
OTSG No. 2
M
M
M
M
No 2
Turbine Driven AFW Pumps
No 1
From Condensate Storage

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

July 8, 1985

IE INFORMATION NOTICE NO. 85-50: COMPLETE LOSS OF MAIN AND AUXILIARY FEEDWATER AT A PWR DESIGNED BY BABCOCK & WILCOX

ADDRESSEES:

All nuclear power facilities holding an operating license (OL) or construction permit (CP).

Purpose:

This information notice is being provided to inform licensees of a significant reactor operating event involving the loss of main and auxiliary feedwater at a pressurized water reactor. Information in this notice is preliminary and was obtained from the special NRC fact finding team which is investigating the event. A complete report of findings will form the basis for further communications or actions related to this event. The NRC expects that recipients will review this notice for applicability to their facilities. Suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On June 9, 1985, the Davis-Besse plant was operating at 90% power with Main Feedwater Pump 2 in manual control because problems in automatic had been experienced. A control problem with Main Feedwater Pump 1 occurred, and it tripped on overspeed. Reactor runback at 50% per minute toward 55% power was automatically initiated. Nevertheless, 30 seconds later, the reactor tripped at 80% power on high pressure in the reactor coolant system.

One second after reactor/turbine trip, one channel of the Steam and Feedwater Rupture Control System (SFRCS) was automatically initiated due to a spurious signal indicating low water level in Steam Generator 2. Both Main Steam Isolation Valves (MSIVs) closed. Three seconds after the actuation, the SFRCS automatically reset. Closing of the MSIVs isolated the turbine of the operating main feedwater pump from its source of steam. The pump continued to supply feedwater to the steam generators for a few minutes as it coasted down.

Four and a half minutes after reactor trip, water level in the steam generators began to fall from the normal post-trip level which is 35 inches. After MSIV closure, steam release to atmosphere continued to remove decay heat. One minute later, Channel 1 of SFRCS actuated when the water level in Steam Generator 1 actually reached the SFRCS setpoint at 27 inches (See Figure 1). SFRCS started Auxiliary Feedwater Pump 1 and initiated alignment of it to Steam Generator 1.

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Within seconds after automatic initiation of Channel 1 of SFRCS, the operator actuated both channels of SFRCS; however, he inadvertently actuated both SFRCS channels on low steam pressure instead of low water level. When an SFRCS channel is actuated on low steam pressure, a rupture of the steam line associated with that channel is presumed to have occurred. The SFRCS closes the steam generator isolation valves, including a valve in the auxiliary feedwater line, and aligns the auxiliary feedwater pump to the other steam generator. Because both channels had been manually actuated on low steam pressure, both steam generators were isolated from both auxiliary feedwater pumps. Five seconds after the operator's inadvertent actuation of both channels on low steam pressure, SFRCS Channel 2 received an actual low water level actuation signal. Because low pressure initiation takes precedence, alignment of the auxiliary feedwater pumps remained unchanged. At six minutes into the event as both auxiliary feedwater pumps were accelerating, they tripped on overspeed.

In summary, all main feedwater had been lost, both steam generators were isolated from feedwater and were boiling dry, all auxiliary feedwater pumps were tripped, pressure of the reactor coolant system was rising, and reactor coolant system temperature was increasing.

Within one minute after the operator's inadvertent actuation of the SFRCS on low steam pressure, the mistake had been recognized and the SFRCS had been reset. If equipment had performed in accordance with system design requirements, the operator's error might not have had a significant impact on the event. The auxiliary feedwater isolation valves should have reopened automatically, but the valves did not reopen. The operator then tried to reopen the valves from the main control panel, but the valves would not reopen. Operators were dispatched to locally start the auxiliary feedwater pumps, open the auxiliary feedwater isolation valves, start the nonsafety-related motor-driven startup feedwater pump, and valve it to the system.

Pressure and temperature in the reactor coolant system continued to rise because there was not sufficient water in the steam generators to provide an adequate heat sink. At 13 minutes after reactor trip, reactor coolant system pressure reached 2425 psig, and the Pilot Operated Relief Valve (PORV) opened three times to limit the pressure rise. On the third lift, the valve remained open. The operator closed the PORV block valve and reopened it two minutes later after the PORV had closed.

Approximately 16 to 18 minutes after reactor trip, the operators had the startup and auxiliary feedwater pumps running and the valves aligned. Water levels were beginning to rise in the steam generators. Reactor coolant temperature reached a maximum of 594° F and then started to decrease to normal. Refilling of the steam generators caused the reactor coolant system to fall to 1716 psig and about 540°F before returning to normal (See Figure 2).

At 30 minutes after reactor trip, plant conditions were essentially stable.

Discussion:

For several minutes after reactor trip, the steam generators were unable to cool the reactor coolant system adequately.

The first problem contributing to this event was the loss of all main feedwater due to closure of the MSIVs. The licensee's hypothesis, based on information from Babcock & Wilcox, is that turbine trip caused a pressure transient upstream from the turbine stop valves which caused the outputs of the redundant steam generator level instrumentation channels to oscillate widely for several seconds. The licensee believes that this caused a spurious low level actuation of SFRCS which closed the MSIVs.

Three additional problems contributed to this event by affecting the availability of both trains of the auxiliary feedwater system. The first occurred when the reactor operator pressed the wrong SFRCS buttons. The second occurred when both auxiliary feedwater pumps tripped on overspeed. The third occurred when both auxiliary feedwater isolation valves did not reopen when SFRCS was reset.

Control buttons for the SFRCS are arranged in two vertical columns. Each column of buttons controls one SFRCS channel. The operator should have pressed the fourth button from the top in each column. Instead, the operator pressed the top buttons causing isolation of both steam generators.

Both auxiliary feedwater pumps are driven by Terry turbines which tripped on overspeed early in the event. When this occurred, steam was being supplied to the turbines via crossover lines, which are longer than the normal supply lines and include long horizontal runs. The licensee believes that significant condensation may have occurred in the crossover lines. Further, the licensee believes that the quality of steam arriving at the turbines may have been affected significantly by the configuration of the crossover lines and may have caused the overspeed trips.

The auxiliary feedwater system isolation valves have Limitorque motor operators. The motor operators have torque switches which prevent overtorquing of the valves by disconnecting power to the motors. When the valves are being opened, additional torque is required to overcome friction while the gates are being unseated and while a significant pressure differential may exist across the gates. During the initial part of the opening stroke, the torque switch in the motor operator is bypassed by a bypass switch so that full motor torque is developed if necessary. The licensee believes that these bypass switches went off bypass too early. The valves did not reopen until an operator unseated them by hand.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate NRC regional office or this office.

E L Jordan
Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: R. W. Woodruff, IE
(301) 492-4507

Attachments:
1. Figure 1 - Steam Generator 1 Level and Pressure
2. Figure 2 - RCS Temperature and Pressure
3. List of Recently Issued IE Information Notices

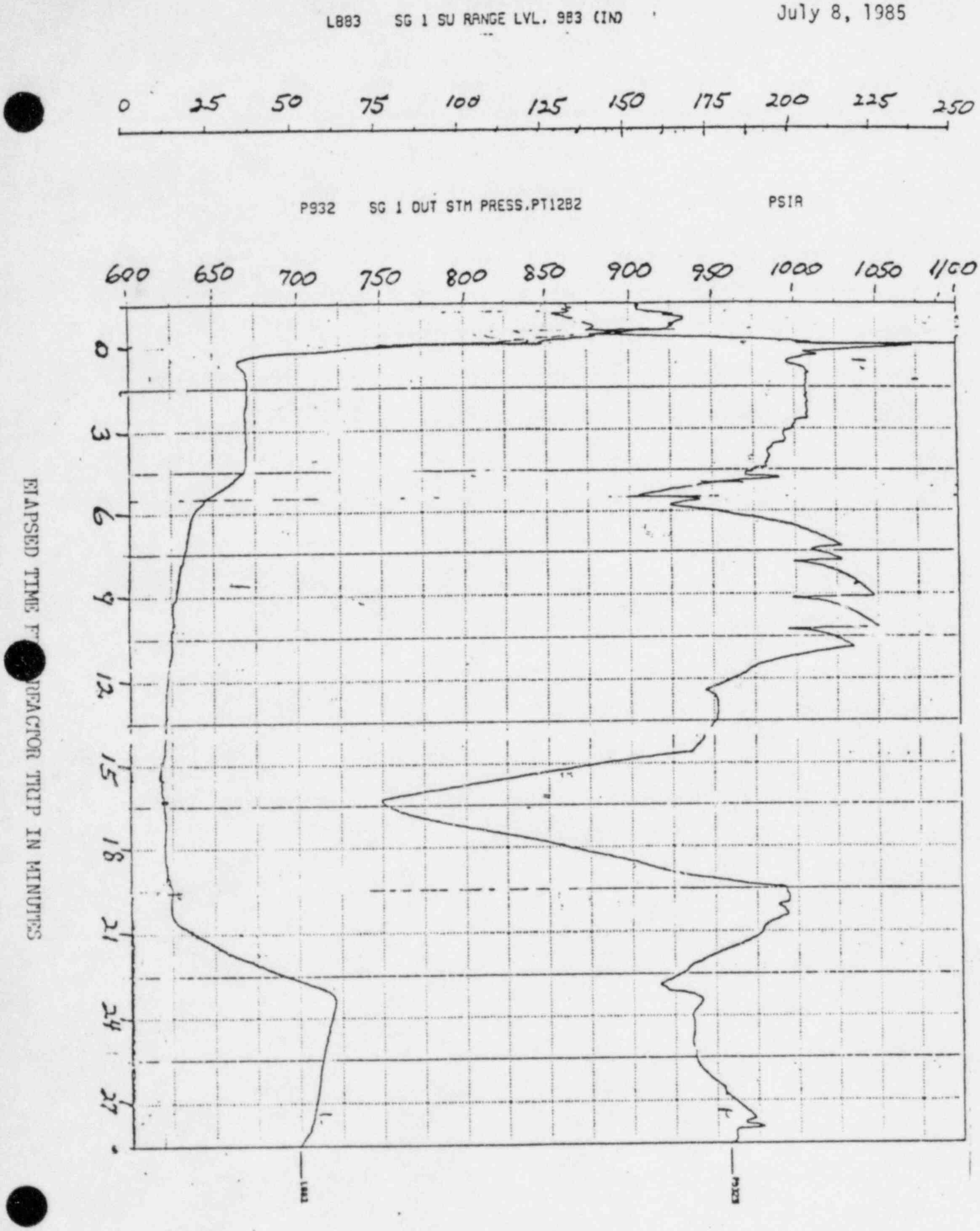

FIGURE 1: STEAM GENERATOR 1 LEVEL AND PRESSURE

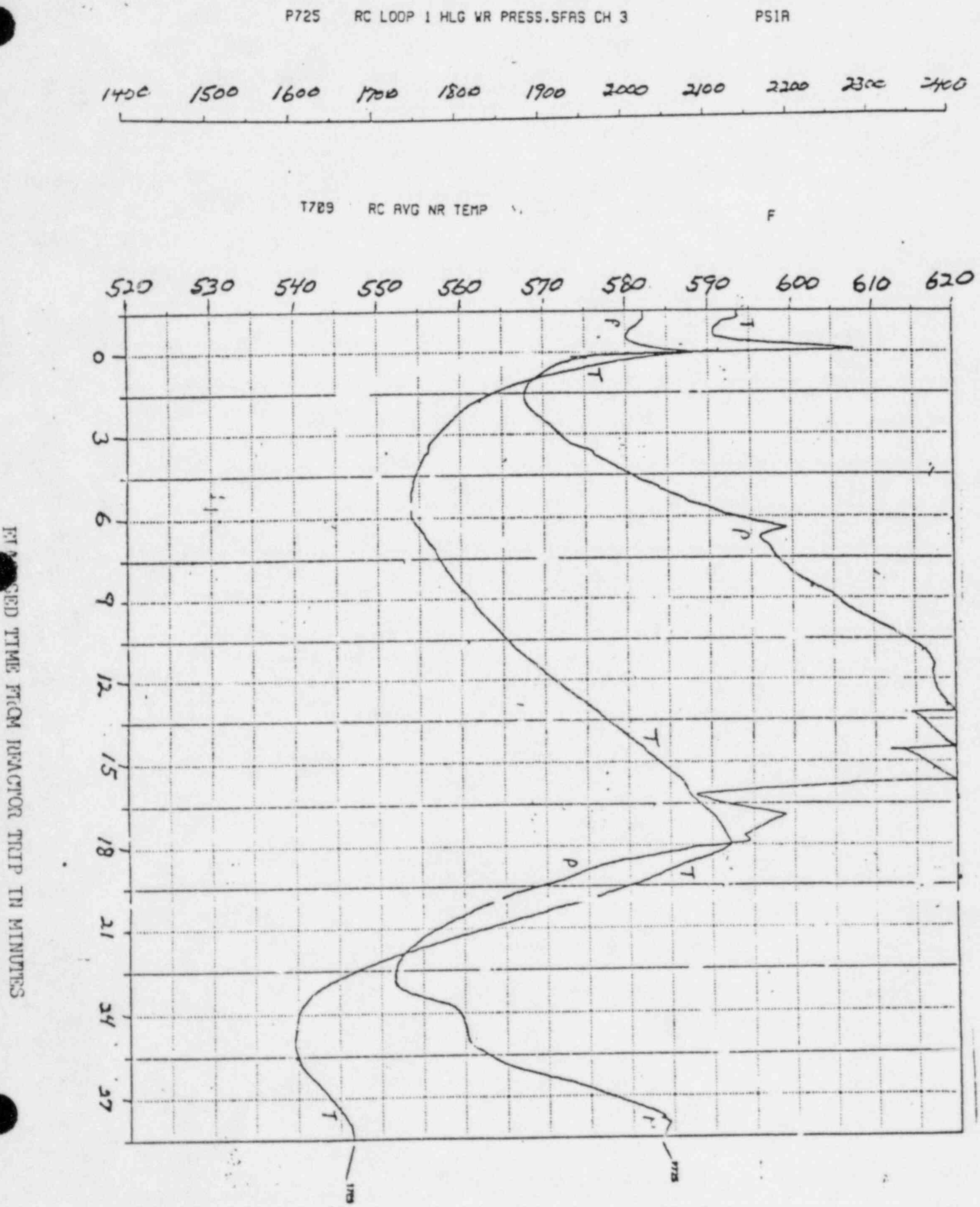

FIGURE 2: RCS TEMPERATURE AND PRESSURE

LIST OF RECENTLY ISSUED
IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
85-49	Relay Calibration Problem	7/1/85	All power reactor facilities holding an OL or CP
85-48	Respirator Users Notice: Defective Self-Contained Breathing Apparatus Air Cylinders	6/19/85	All power reactor facilities holding an OL or CP, research, and test reactor, fuel cycle and Priority 1 material licensees
85-47	Potential Effect Of Line-Induced Vibration On Certain Target Rock Solenoid-Operated Valves	6/18/85	All power reactor facilities holding an OL or CP
85-46	Clarification Of Several Aspects Of Removable Radio-active Surface Contamination Limits For Transport Packages	6/10/85	All power reactor facilities holding an OL
85-45	Potential Seismic Interaction Involving The Movable In-Core Flux Mapping System Used In Westinghouse Designed Plants	6/6/85	All power reactor facilities holding an OL or CP
85-44	Emergency Communication System Monthly Test	5/30/85	All power reactor facilities holding an OL
85-43	Radiography Events At Power Reactors	5/30/85	All power reactor facilities holding an OL or CP
85-42	Loose Phosphor In Panasonic 800 Series Badge Thermo-luminescent Dosimeter (TLD) Elements	5/29/85	All power reactor facilities holding an OL or CP

OL = Operating License
CP = Construction Permit

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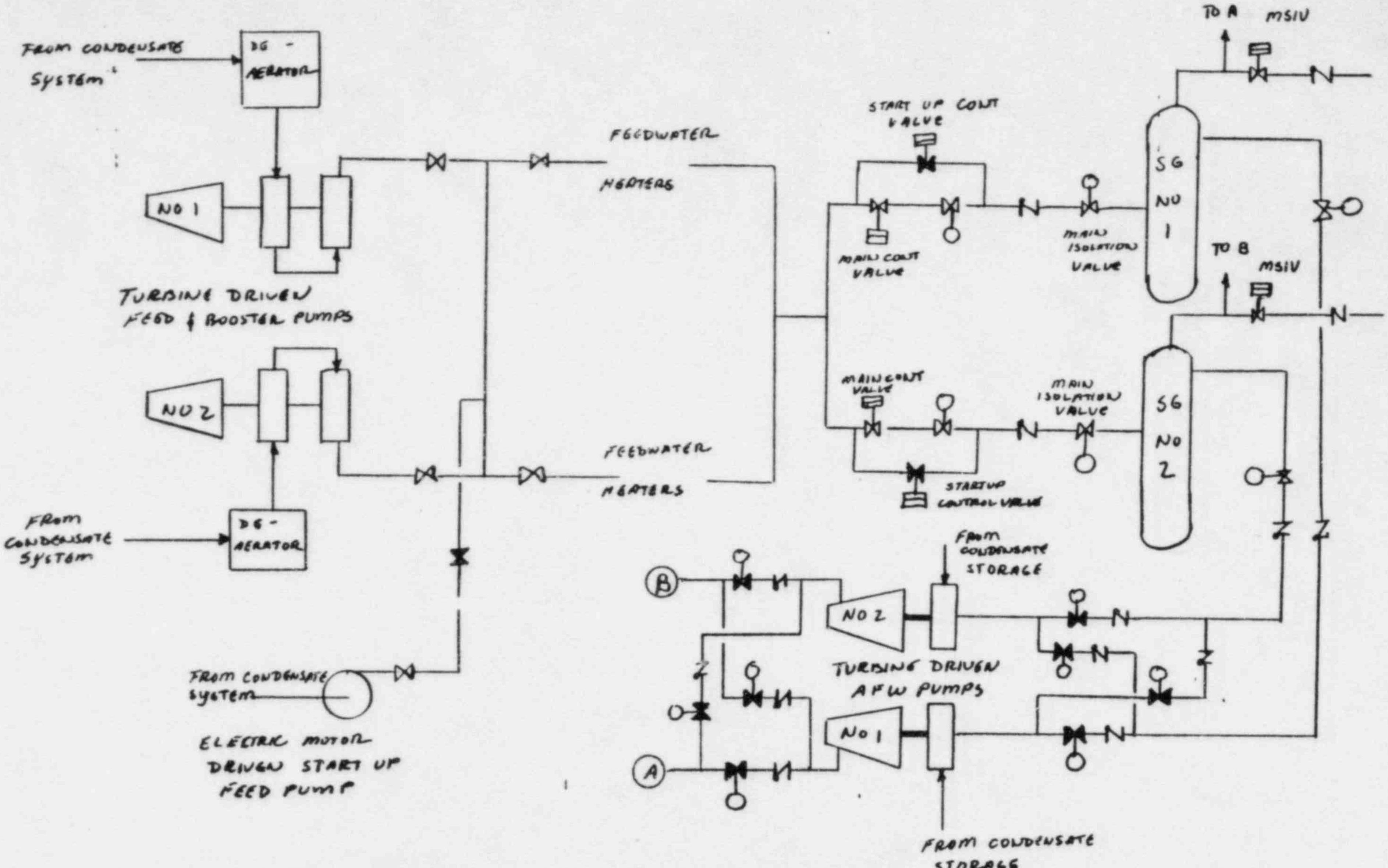
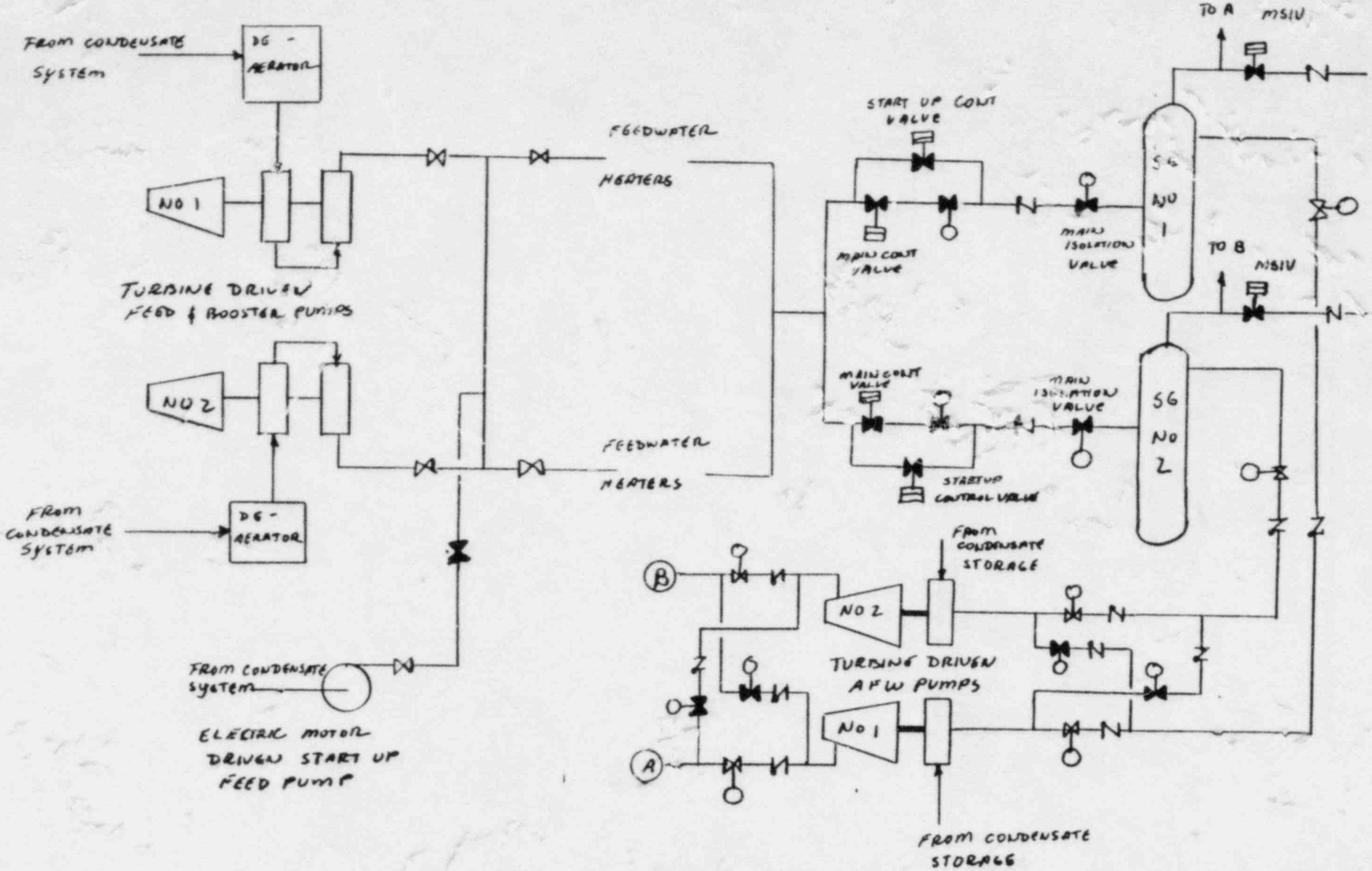


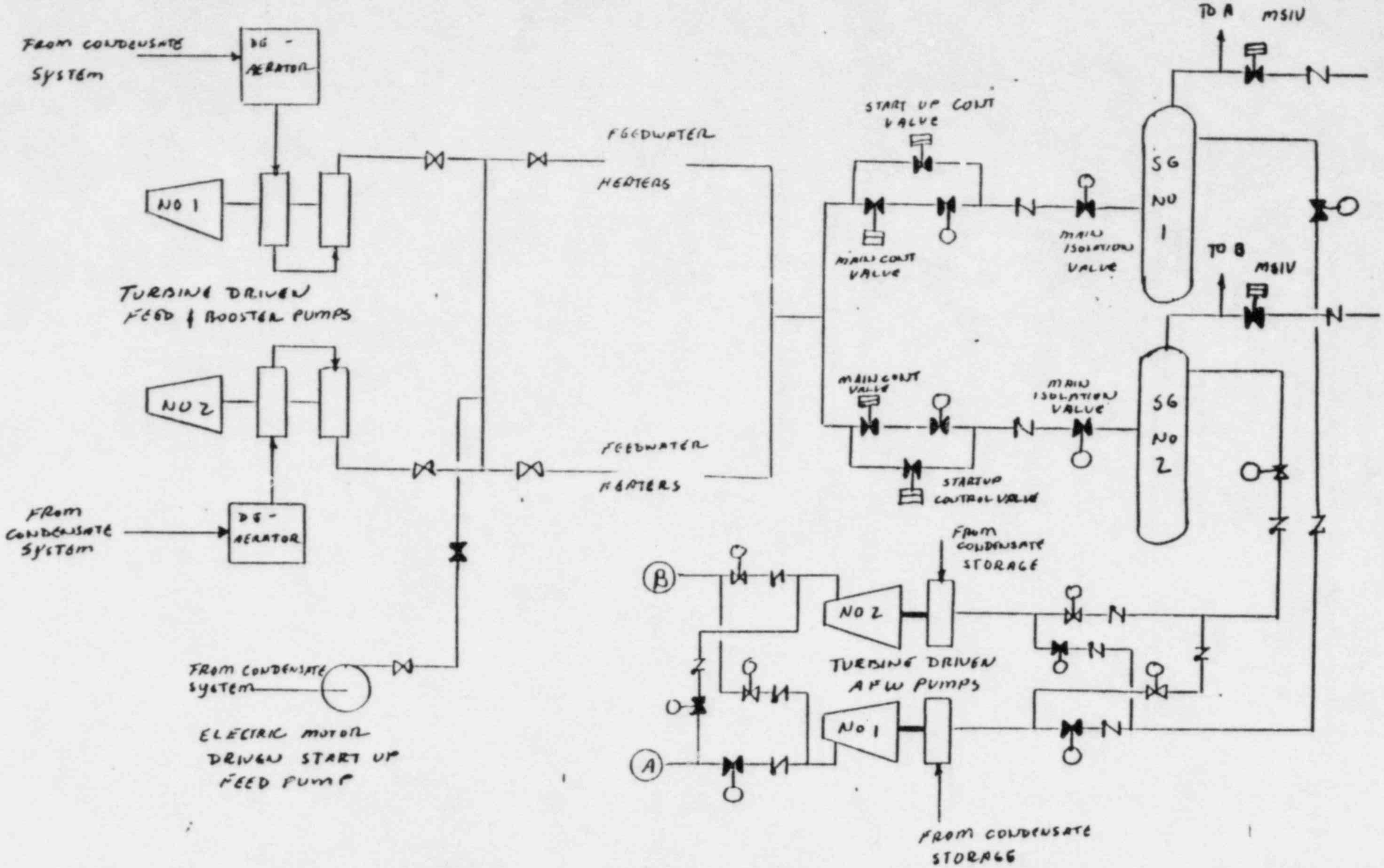
DAVIS BESSE FEEDWATER SYSTEMS

NORMAL OPERATION



DAVIS BESSE FEEDWATER SYSTEMS

AFTER SFRCS TRIP ON LOW ST GEN LEVEL OR HIGH STEAM - FEED PRES DIF



DAVIS BESSE FEEDWATER SYSTEMS

AFTER SFRCS LOW PRESSURE TRIP
ON ST GEN NO 1

TO: George Lanik 1 of 7

NRC Form 366A
U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Davis-Besse Unit 1	0 5 0 0 0 3 4 6	8 5	— 0 1 3	— 0 0	0 3 OF 2 2

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Description of Occurrence: Davis-Besse Unit 1 was operating at 90 percent of full power with the No. 1 Main Feedwater Pump, MFP, operating in automatic and the No. 2 MFP in manual control. This configuration was established to limit the susceptibility of the No. 2 MFP to control problems which had previously occurred. The control problems occurred only after a reactor trip and appeared to be connected to the automatic mode of operation. This configuration, therefore, permitted automatic feedwater control during operation and offered improved availability of at least one MFP in the event of a reactor trip.

At 0135 hours, the No. 1 MFP tripped on overspeed due to an unrelated control problem. The Control Room operators increased the No. 2 MFP speed, but it did not have adequate capacity, for the existing reactor power. The reactor tripped on high Reactor Coolant System, RCS, pressure at 0135:30 hours, tripping the turbine. Reactor power was at approximately 80 percent of full power at the time of the trip.

Immediately following the trip, a spurious Steam and Feedwater Rupture Control System, SFRCS, low steam generator level full trip occurred on Channel 2, an SFRCS full trip alarm was received, and both main steam isolation valves, MSIVs, closed. An actual low steam generator level did not exist at this time. This spurious trip resulted in a partial actuation of the SFRCS components since only the MSIVs actuated. When the MSIVs closed, the main steam supply was isolated to the MFPs. The No. 2 MFP continued to supply feedwater until approximately 0140 hours at which time its discharge pressure was not high enough to supply feedwater to the steam generators. The level in the steam generators which was being maintained at the low level limit setpoint (35 inches) began to decrease. SFRCS Actuation Channel No. 1 then automatically initiated on low steam generator level, starting the No. 1 Auxiliary Feedwater Pump, AFP, to feed the No. 1 Steam Generator (see Attachment 1 for a diagram of SFRCS actuated components).

At 0141:08 hours, a Control Room operator attempted to manually initiate the SFRCS, however, he incorrectly actuated the SFRCS on low steam pressure instead of the desired low steam generator level. Therefore, each SFRCS actuation channel sensed that its respective steam generator was depressurized. SFRCS Actuation Channel No. 1 then attempted to align AFP No. 1 to feed Steam Generator No. 2. SFRCS Actuation Channel No. 2 attempted to align AFP No. 2 to feed Steam Generator No. 1. Both actuation channels closed their respective Auxiliary Feedwater Containment Isolation Valves (AF599, AF608), which prevented any auxiliary feedwater flow from reaching the steam generators. At 0141:31 hours, AFP No. 1 tripped on overspeed. At 0141:44 hours, AFP No. 2 tripped on overspeed.

At 0142:00 hours, an operator recognized the manual initiation error and reset the low pressure SFRCS buttons, and pushed the low steam generator level SFRCS manual actuation buttons. Since both SFRCS actuation channels were already tripped on low steam generator level, the SFRCS automatically began to realign the AFPs when the low pressure buttons were reset. However, the Auxiliary Feedwater Containment Isolation Valves (AF599, AF608) did not automatically open. The operators attempted to open these valves from the Control Room by operating their control switches and by reinitializing the SFRCS. These attempts failed to open the valves. Equipment

NRC FORM 366A
(9-83)

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Davis-Besse Unit 1	0 5 0 0 0 3 4 6	8 5	— 0 1 3	— 0 0	0 4 OF 2 2

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Operators were sent to open these valves locally, and when the valves were moved off their closed seats utilizing the manual handwheels, the motor operator responded and fully opened the valves. During this period, attempts were also being made to restart the AFPs and preparations were underway to start the motor operated Startup Feedwater Pump.

The RCS average temperature was increasing due to the lack of primary to secondary heat transfer. RCS pressure was increasing due to the decreasing density of the RCS water and increasing pressurizer level. RCS pressure increased to the Power Operated Relief Valve, PORV, setpoint (2425 psig). The PORV cycled a total of three times, relieving pressurizer pressure to the Quench Tank. Following the third opening, the PORV failed to reclose at the proper RCS pressure. The Control Room operator observed the primary plant conditions and closed the block valve on the PORV. RCS pressure was at approximately 2075 psig when the block valve closed. The Quench Tank contained the discharges from the PORV.

At approximately 0151 hours, the operators placed the Startup Feedwater Pump in operation to supply the steam generators. Steam Generator No. 1 pressure had decreased to approximately 750 psig. Steam Generator No. 1 repressurized to approximately 900 psig from the Startup Feedwater Pump. Steam Generator No. 2 had decreased to 920 psig. At 0152 hours, the No. 2 AFP was returned to service by the operators locally. Maximum RCS temperature had reached approximately 592 degrees Fahrenheit. At 0155 hours, the No. 1 AFP was returned to service locally by the operators. Control of the AFP turbines was maintained locally by an operator at the turbine trip throttle valve. At 0158 hours, RCS average temperature was restored to the normal post trip temperature. The cooldown of the RCS lowered RCS pressure to a minimum of approximately 1720 psig. Operators manually started the No. 1 High Pressure Injection, HPI, Pump in the piggyback mode (Decay Heat Pump No. 1 supplying the suction to the HPI Pump No. 1) in precautionary anticipation of the rapid cooldown. Only a slight amount of water (less than 50 gallons) needed to be injected.

Several other equipment malfunctions occurred which did not affect the physical plant response. One source range nuclear instrumentation, NI, channel was inoperable prior to the trip. The remaining source range NI channel failed to indicate properly when it was automatically energized after the trip. The display units for the Safety Parameter Display System, SPDS, were inoperable in the Control Room at the time of the trip. At 0158:40 hours, the suction of the No. 1 AFP automatically transferred from the Condensate Storage Tank, CST, to the Service Water System. The operator manually realigned the pump suction back to the CST. No significant amount of service water was added to the steam generator during the recovery from the transient. It was noticed that the pneumatic operator on one main turbine bypass valve was damaged, preventing the valve from being opened. This did not affect the post transient response of the plant.

Additional details of the plant transient and corrective actions will be provided in the restart report response to the Region III Confirmatory Action Letter (85-06). Attachment 2 provides a chronological listing of the event. This report is being submitted in compliance with paragraph 50.73(a)(2)(i), 50.73(a)(2)(iv), 50.73(a)(2)(v),

NRC Form 366A (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

50.73(a)(2)(vi), and 50.73(a)(2)(vii). This report also satisfies the reporting requirements for a Emergency Core Cooling System Actuation Special Report, Section 6.9.2(a) of Technical Specifications. This was the fourth high pressure injection actuation cycle to date.

Designation of Apparent Cause of Occurrence: This transient was initiated when the No. 1 MFP developed control problems and tripped on overspeed. The plant tripped on high RCS pressure due to inadequate feedwater being supplied from the No. 2 MFP during the plant runback. The cause of the MFP overspeed tripping was determined to be due to a bad speed summation and valve lift reference circuit board card in the MFP control. A frequency to voltage converter chip had failed. The board is being returned to General Electric for further analysis on the root cause of the failure.

The root cause of the MSIV closure has not yet been determined. It is presently believed that the MSIVs properly responded to a momentary low level SFRCS trip. Further investigations will follow once an action plan is completed.

The cause of the SFRCS spurious trip on low steam generator level has not yet been positively determined. Troubleshooting will begin in accordance with the action plan. However, it is presently believed that the steam generator level sensing channels are sensing an extremely rapid secondary side pressure transient that occurs in the steam generator following the turbine stop valve closure on a turbine trip. These level transmitters share a common set of sensing lines with transmitters which were replaced during the 1984 Refueling Outage. Prior to the 1984 Refueling Outage, Bailey BY level transmitters were installed which have now been replaced by Rosemont Model 1153. Since these Rosemont transmitters have no significant displacement required for operation, while the Bailey BYs required a volume displacement to operate the bellows, it is postulated that the responsiveness of the sensing line and transmitter arrangement has been greatly increased by this change. This increased responsiveness allowed the SFRCS to sense the rapid secondary side pressure transients which previously were undetected. Further analysis of this condition is underway.

The cause of the incorrect manual SFRCS initiation was personnel error attributed to a poor switch layout. These SFRCS manual initiation pushbuttons had been identified in the Detailed Control Room Design Review as one of the principal items needing human engineering improvements. There are two adjacent vertical columns of buttons with five buttons in each column (see Attachment 3 for arrangement details). Each column represents one SFRCS actuation channel. To manually initiate both channels of the SFRCS for steam generator low level, the operator should have depressed the fourth button from the top in each column; instead, the two top buttons were depressed. A design change had been developed prior to this event to improve the switch layout and will be implemented during this outage.

The cause of the AFPs tripping on overspeed after initiation has not yet been positively determined. Water flashing through the nozzles of the AFP turbines is thought to be a contributor. The governor was inspected on both AFP turbines, and no contributing factors to the overspeed were seen. Further investigations and testing are planned.

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The cause of the Auxiliary Feedwater Valves (AF599 and AF608) not opening by the motor operator was determined to be a combination of a high differential pressure and an improperly set torque switch bypass limit switch. With this torque switch bypass limit switch improperly set, the motor operator was allowed to torque out during the opening stroke. These valves are open during normal operation and were closed by the incorrect manual initiation of the SFRCS. If the AFPs had been operating at the time the valves attempted to open, the differential pressure across the valves would have been significantly lower, and the valves should have opened to allow the auxiliary feedwater flow to occur. These valves were stroked following the transient and ability of the valves to open (without significant differential pressure) was verified. Recent testing also verified that the valve operators torque out under high differential pressure with the improperly set torque switch bypass limit switch. Further investigations are in progress.

The cause of the control problems with the AFPs after the overspeed was reset is presently attributed to the difficulty in opening the trip throttle valves. No mechanical deficiencies were found while investigating the resetting of the overspeed trip device/linkage. Further investigations are in progress.

The cause of the PORV not properly reseating has not yet been positively identified. Operator observations at the time of the transient indicate that the electronic controls signal was calling for the valve to reclose. A visual inspection and disassembly of the PORV failed to identify the cause. Further investigations are in progress.

The two, independent SPDS display units were inoperable due to separate but similar failures in the data transmission system between the Control Room terminals and their respective processors. The failures are of an intermittent nature and the exact cause is still under investigation.

The cause of the source range NIs inoperability has not been positively identified. The failure of the source range NIs has been a repetitive problem at Davis-Besse with repeated investigations failing to determine the root cause. Since 1977, the boron trifluoride detectors, preamp, and cable in Containment have been replaced, along with the modules in the Reactor Protection System and a reworking of the grounding on the preamp and count rate amplifier module connections. No positive effect on the total elimination of the spiking, nor the erroneous/elevated count rate has occurred from these corrective actions. Further review is being performed on the possibility of ground loops, induced current or voltage from adjacent cables, or intermittent problems with the count rate amplifier module.

The cause of the Turbine Bypass Valve 2-2 damage has not been identified. The valve was disassembled and the actuator stem extension piece was found bent, four parts were missing, and the valve internals were found loose. Several valve parts were shipped to the vendor for further analysis. Further review of the turbine bypass valve failure is underway.

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The cause of the inadvertent AFP No. 1 suction supply transfer from the Condensate Storage Tank to Service Water has not yet been determined. Testing and other investigations are currently being performed.

The cause of MS-106 apparently cycling in about one third of the expected stroked time is still under investigation.

Analysis of Occurrence: This event involved a temporary loss of feedwater to the steam generators. This event was bounded by the analyses previously performed (see Toledo Edison submittals to NRC Serial No. 506 dated May 22, 1979, and Serial No. 517 dated June 15, 1979), which analyzed a loss of all feedwater for 30 minutes following a reactor trip. These analyses showed that as long as either:

1) Auxiliary Feedwater is restored within 30 minutes of the loss of main feedwater,

OR

2) Within 30 minutes, at least one makeup pump and the PORV are available for primary cooling (feed and bleed) and the Startup Feedwater Pump is available to supply a steam generator,

fuel cladding temperatures would remain within a few degrees of saturated fluid temperature and no cladding rupture or metal water reaction would occur.

Operator interviews indicated that the shift was fully aware of the core status and were prepared to implement the "feed and bleed" core cooling method if the auxiliary feedwater was not restored. The Startup Feedwater Pump was available throughout the event and in fact was placed in service within ten minutes of the tripping of the AFPs. Auxiliary Feedwater was restored within 12 minutes of the loss of feedwater. These response times and equipment availability are well within the loss of feedwater analyses.

At no time during the event was the required subcooled margin (20 degrees Fahrenheit) lost. The reactor coolant pumps continued to operate throughout the event. The primary code safety valves were not challenged and at no time during the event did the RCS pressure or temperature exceed the allowable values. The maximum temperature reached was below the normal operating temperature for the hot leg temperature. There is no indication of any fuel cladding degradation based on the reactor coolant radiochemistry analysis.

An analysis has been performed by Babcock & Wilcox to determine if the transient adversely affected the steam generators. Conditions and components specifically analyzed include: (1) Main Feedwater Nozzles, (2) Auxiliary Feedwater Nozzles, (3) Steam Generator Tubes, (4) Tube to Shell Delta T's, and (5) Lower Tubesheets

The results show that the transient had no adverse structural effect on the steam generators.

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APPROVED OMB NO. 3150-0104
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Corrective Action: The failed circuit board will be replaced in the No. 1 MFP. As a precautionary measure, the No. 2 MFP speed control circuit will also be inspected for a similar failure.

The corrective action on the MSIV closure has not yet been determined since trouble-shooting has not yet begun.

The corrective actions for the SFRCS spurious trip on low steam generator level have not yet been determined since troubleshooting has not yet begun. The proper method of manual actuation of the SFRCS buttons will be reviewed with all licensed operators. The switch layout is being modified to add additional demarkation of the actuation buttons, and to add actuation guards over the switches (see Attachment 3).

The corrective actions to be taken to prevent the AFP trip on overspeed have not yet been determined.

The torque switch bypass limit switch will be reset on the Auxiliary Feedwater Valves AF599 and AF608. Maintenance personnel will receive additional instruction, and the procedure for setting the motor operator valve limit switches will receive additional clarification. Other nuclear safety related motor operated valves at Davis-Besse will be evaluated.

The corrective actions to correct the control problems with the AFPs after the overspeed was reset have not been identified.

Corrective actions to be taken on the PORV have not yet been identified.

Corrective actions for the repair of the data transmission systems affecting the SPDS Control Room displays have not yet been identified.

The corrective actions for repair of the source range NIs have not yet been determined.

The pneumatic actuator for Turbine Bypass Valve 2-2 will be replaced. Additional corrective actions may be necessary after further investigation to determine the root cause of the failed valve actuator.

A tabulation of the causes and corrective actions determined to date is summarized in Attachment 4.

Corrective action details for the No. 1 AFP suction supply transfer from the Condensate Storage Tank to Service Water has not yet been identified.

Corrective action details for MS-106 have not yet been identified. Further investigation is in progress.

Failure Data: This is the first occurrence at Davis-Besse of a loss of both main and auxiliary feedwater.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

This is the first failure that has occurred at Davis-Besse on the MFP turbine electronic controllers which has caused an overspeed tripping of the pumps. A new electronic control system for main feedwater pumps was installed during the 1984 Refueling Outage.

Spurious closures of the MSIVs have occurred previously at Davis-Besse before time delays were added to the steam to feedwater pressure differential trip circuitry.

An SFRCS spurious half trip on low steam generator level has occurred on two previous trips since the 1984 Refueling Outage. Spurious trips on low steam generator level have not occurred prior to the 1984 Refueling Outage.

Incorrect manual initiation of the SFRCS has not previously occurred at Davis-Besse.

The AFPs tripping on overspeed after initiation has not previously occurred at Davis-Besse.

The Auxiliary Feedwater Valves AF599 and AF608 are normally open. One previous occurrence of one of these valves not opening with high differential pressure occurred after the March 2, 1984 reactor trip.

The operators do not normally attempt to control the AFP turbines locally. Problems with controlling these pumps do not appear to have been repetitive, however, some problems have been experienced previously with proper resetting of the trip throttle valve.

The PORV has not been challenged since the pressure setpoint was raised in 1979. Prior to 1979, several deficiencies were noted in the valve operation. In September 1977, the valve stuck in the open position, causing an overpressurization of the quench tank.

The diversity of the SPDS display sources (Ramtek and Chromatics display devices) has normally allowed at least one SPDS display to remain operable. The failure rate of these units is higher than is acceptable. Efforts are underway to increase the system reliability.

The failures of the source range NIs have been a repetitive occurrence at Davis-Besse even though exhaustive evaluations and corrective actions have been taken.

The damaged pneumatic operator on the turbine bypass valve has not previously occurred at Davis-Besse.

There have been several cases where the AFP suction inadvertently transferred from the Condensate Storage Tank to the Service Water supply.

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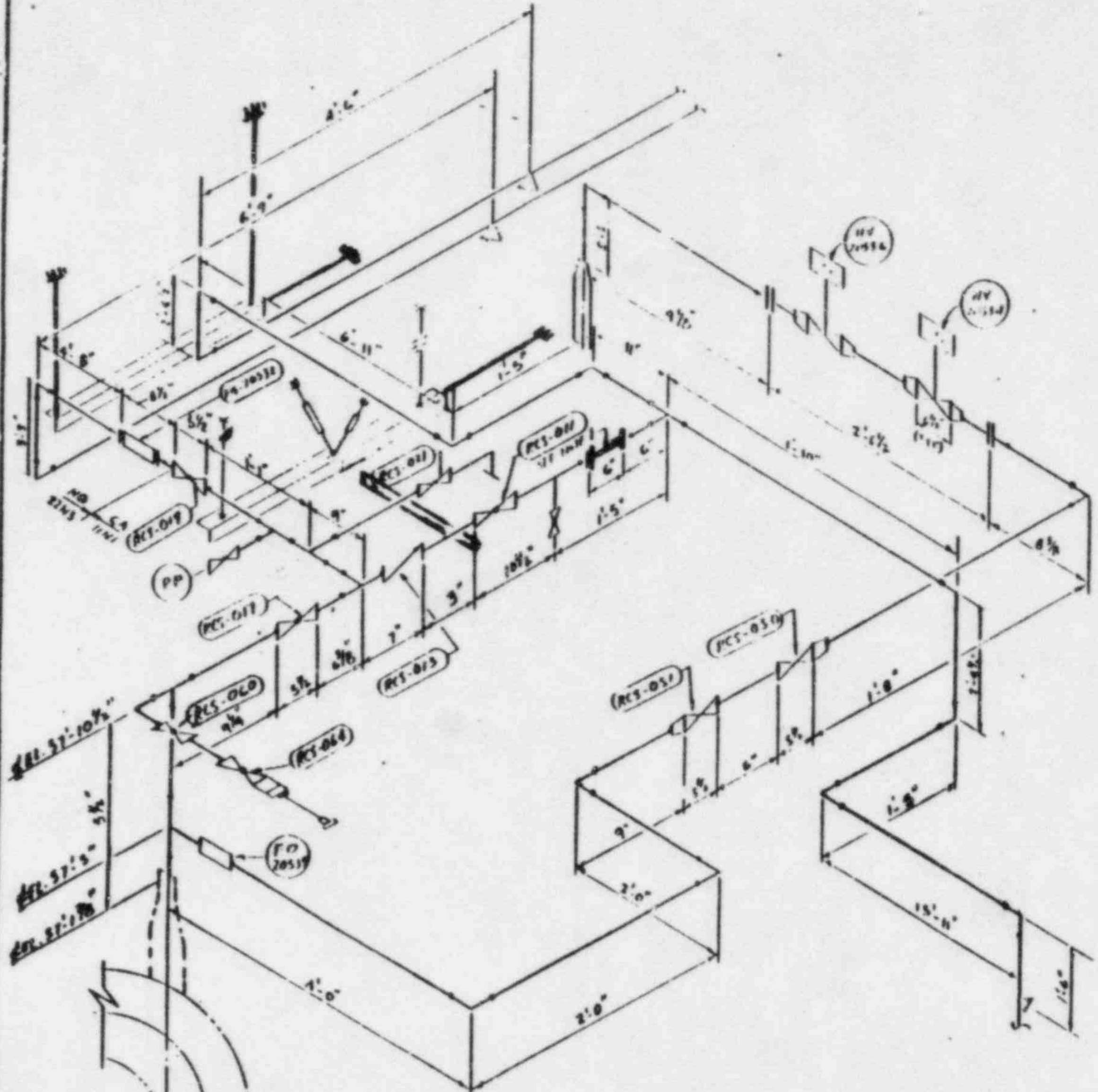
AS REQUIRED BY
1983 DCN PACKAGE



NOTE 1

FIELD TO WELD RIGID MEMBER BETWEEN TWO PLAS.

ISOMETRIC DIAGRAM
RCS HIGH POINT VENT
STEAM GENERATOR E-210B

RANCHO SECO NUCLEAR GENERATING STA., UNIT 1