NUCLEAR REGULATORY COMMISSION

ORIGINAL

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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In the Matter of: 244TH MEETING

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

244th MEETING

Room 1046
1717 H Street, N. W.
Washington, D. C. 20555
August 7, 1980

The Subcommittee met, pursuant to notice, at 8:30 a.m.

MEMBERS PRESENT:

M. PLESSETT, Chairman, Presiding.
J. C. MARK, Vice-chairman
S. LAWROSKI
D.W. MOELLER
W. KERR
M.W. CARBON
W.M. MATHIS
J.C. EBERSOLE
H.W. LEWIS
D. OKRENT
J.J. RAY
P. SHEWMON

DESIGNATED FEDERAL EMPLOYEE:

R.F. FRALEY

ACRS CONSULTANTS:

I. CATTON
Z. ZUDANS
W. LIPINSKI

ALSO PRESENT:

J.M. JACOBS, Secretary

P R O C E E D I N G S

MR. PLESSETT: The meeting will come to order.

This is the 244th meeting of the Advisory Committee on Reactor Safeguards. Reviewers for the meeting will be Paul Shewmon and Jerry Ray, and I must say that they have an unusual distinction. The full committee approved the action of the Procedures Subcommittee. They will be the last singled out reviewers because we are going to suggest that after this meeting, there be 15 reviewers of the full committee. We will discuss this in more detail later.

During this meeting the committee will hold discussions on the following: Regulatory Guide 1.97, Revision 2, instrumentation for light-water-cooled nuclear power plants to assess plant and environs conditions during and following an accident; two, status report on the NRC review for startup of TMI Unit One; three, NRC report on unresolved safety issues; four, NRC report on items related to the Trojan Nuclear Plant; five, discussions regarding quantitative risk criteria; and six, discussions relating to proposed actions and positions regarding several safety-related matters.

The specific items on the agenda today are Regulatory Guide 1.97, TMI review status, unresolved safety issues, and the Trojan Nuclear Plant. We shall also discuss various other topics as well as future schedule.

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

Mr. Raymond Fraley is the designated Federal employee for this portion of the meeting.

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

We have received request for permission to make oral statements with regard to Reg. Guide 1.97 from representatives of General Electric Company, the ANS 4.5 Working Group, and the Atomic Industrial Forum. These presentations will be made at approximately 9:30 this morning at the beginning of this session.

(The Chairman's Report was presented beginning at 8:35 a.m. and ending at 8:50 p.m.)

(Recess)

MR. PLESSETT: We have a report from the ACRS Subcommittee on Regulatory Activities. Dr. Siess is ill and could not get here, and the report will be made by Bill Kerr.

Bill, would you please make you report?

MR. KERR: The Regulatory Activities Subcommittee spent a great share of its meeting yesterday in a discussion of Reg. Guide 1.97. I will remind you that the original regulatory guide involved a considerable amount of long-expressed ACRS interest as well as NRC interest in instrumentation that would be helpful in trying to follow the course of a serious accident.

The particular regulatory guide that we are now considering is a second revision of the regulatory guide, and the earlier versions confined themselves primarily to accidents class 1 through 8 and did not take any cognizance, or at least very little cognizance, of the possibility of a more serious accident.

ACRS periodically urged that consideration be given to more serious accidents, and I think some consideration was given in various forums, but not very specific. After TMI-2 occurred, as part of a general reaction to reactor safety, a considerable amount of change in the scope and content have occurred in the present version of Reg Guide 1.97.

In the first place, there certainly is now, I think, consideration of accidents more serious than the design basis accident. I say I think that is the case because we got at least one response yesterday from a staff member indicating the design did stop at the design basis accident; but I believe that is probably not the case.

However, there is a good bit of mention made in discussion of the single-failure criterion. It seems to be one of the principal bases for determining reliability of the systems. In addition to a change in that kind of scope, a broader spectrum of accidents is now being looked at . there has also now been a change in the expected use of the informtion that will be collected and about which 1.97 is primarily concerned.

Initially, I think, it was felt that the information would be used primarily by people who are operating the plant who were concerned with accident recovery, perhaps primarily people in the control room. The present version has been expanded as an effort to collect information that will be useful not only to that group but also to those who might be responsible for advising emergency evacuation or other activities that might take place off-site; too, information that might be needed in the operation of an emergency control center if the control room became unhabitable or unavailable or inoperable for some reason; too, information that would be needed in a technical support center.

Such centers are being and will be set up in connection with operating reactors and for information that would be available for and probably would be transmitted by and made use of by the Nuclear Data Link. Indeed, there are statements associated with the Nuclear Data Link which say those variables will be transmitted that come from -- I guess make up that list of variables in Reg Guide 1.97.

Reg Guide 1.97, Revision 2, purports to endorse a draft ANS standard, which is being developed and is apparently near its final form, although it has not yet been finally approved, and it deals with accident mitigation. I say purports to endorse because although 1.97 says it endorses the standard, it then goes on to take a good many exceptions to the standard. So it is rather difficult for me to tell whether it endorses the standard of not. It certainly goes considerably beyond the standard, both in scope and in requirements.

I think in addition to details there are some fundamental differences in the industry approach to the problem and the present approach being taken by the staff in Reg Guide 1.97. The differences are -- in one case I think a difference in approach which is not so serious -- the difference in approach is that there does seem to be a feeling in industry that the various tasks ought to be separated because they could be approached more logically if they were separated.

I mean, for example, one ought to concern oneself first, say, with control room displays, or at least a method of getting information to the operator which would be useful in an emergency situation. That perhaps is a first priority task and one ought to concentrate on this and one ought to separate that from these other tasks. That is sort of a matter of approach.

I think one could argue that it makes some sense to separate because the tasks are more manageable if one does separate. On the other hand, since many of these data being arked fnq amd ltcq ne sgd hmenql'shnm hr khjdky tn ad common to the various users, there is some argument that says one gets a better integration and perhaps is more efficient if one tries to solve all these problems simultaneously.

It seems to me there is a second, more fundamental difference, at least as I try to interpret the two approaches. The industry people say that they think one ought to try to define what it is one wants to do with the information, how it is going to be used -- for example, in a Safety System Display or off-site center or whatever -- try to determine what it is one wants to accomplish, and on the basis of that, what sort of information and what sort of information processing will be necessary. Then after one has determined that, one then asks for specific variables, temperature, pressure, whatever. They insist -- I don't think this is entirely true, but maybe it is partly true -- that the staff, on the other hand, has said what sorts of measurements should we be able to make.

Now, we are not quite sure what we are going to do with them except in a general way that we are going to use them for the Data Link or the emergency center. From that one compiles, in effect, whole list of variables. This is the nomenclature used in 1.97.

Without necessarily at this point having a completely coherent picture in the way in which the variables are going to be processed and used, it seems to me that if this perception is correct, this does represent a fairly fundamental difference in approach. I think you want to look for the possibility that this difference in approach does exist and see, if it does, how serious you think it is, and make any suggestions that might come from that perception.

We discussed the guide after the subcommittee had listened both to the staff and to a number of industry representatives who spoke. You have received at one point, I think, a very thick list of industry comments and a not quite so thick but nevertheless formidable compilation of staff responses to the industry comments.

It is certainly an issue that has caused a good bit of concern in the industry and has led to a good many comments. The subcommittee did not suggest that we come to the full committee with a recommendation. We had at various times in the subcommittee meeting -- I was there as Chairman in Chet's absence, and Jerry Ray was present for most of the meeting. Bill Mathis was present, I think, for most of the meeting. Dade Moeller was present for some parts that particularly concerned him. Steve was present for while. I thought I mentioned Jerry, did I not?

So there was considerable representation. We also had as consultants Lipinski, Catton and Zudans, and they certainly made significant contributions.

I have nothing further to say, but I would like to ask if there are comments from those committee members who were present and from the consultants.

MR. PLESSETT: Thank you. Bill.

Dade?

MR. MOELLER: I think Bill has done a very good job of summarizing what we heard yesterday. I do have several comments which I hope will be taken in a constructive sense. This has been a long, drawn out affair, as most of us know, and the staff does currently have a guide which they are proposing to be accepted.

As a person reading it over the last week or so, I must say, though, that it does seem to be in some senses a hodge podge of thoughts, and you do note within the NRC staff a sensitivity and, in fact, sort of a resistance to changing what they have set down. I think that is exemplified when you go through the public comments because if you do statistical summary on some of them, you will find that perhaps 80 percent have been rejected without any change in the Guide.

Not that these are all correct comments, but I think that gives you some idea of the status. At the same time, though, progress is being made because they are communicating and they are discussing these issues. Even though you do not have a guide completed, nonetheless, as I say, progress is being made.

To me the industry in its presentation seemed to be quite systematic and scientific in their approach, and I did gain one what I thought was a very important thought from what the industry people said. That is they asked, you know, just what is it we are doing? Are we monitoring variables or are we assessing the status of systems? I believe the ANS standard, in fact, says it is to monitor variables, whereas the Reg Guide says it is to determine the status of systems.

They acknowledge what they are doing and make no greater claims, whereas the Reg Guide would imply that they have reached the end point. Obviously, there is no easy answer to it; otherwise, it would have been completed a long time ago.

MR. PLESSITT: Jerry.

MR. RAY: I don't think I could add any

constructive thoughts except just this one. I think that it was expressed repeatedly, and I personally feel that it is quite complicated and that therefore its practicality of application by operating companies might be influenced as a result.

MR. PLESSETT: Charlie.

MR. MATHIS: I did --

MR. SHEWMON: Would you explain what you just said, Jerry? I thought you were going to talk about a particular part being complex.

MR. RAY: The whole report, in my opinion, is complicated, and it will be difficult for a designer, for instance, in engineering a new plant to really clearly derive from it what he should do; and from this viewpoint, simplification might very well be profitable.

MR. PLESSETT: Charlie.

MR. OKRENT: I don't understand that either, Jerry, because either the designer understands a subject so well he does not need to be told what he is going to do -- and usually the people who are designing instrumentation are not specialists in how serious accidents progress, and my experience is they do not really have a big background in the control room and so forth -- or they need somebody to tell them what they should do.

So,, whi.. it sounds nice at first blush to give them just some general guidance, I think if that is what you are doing, you also have to say go to school long enough so you fully understand the situation so you will do the job right.

MR. RAY: Well, the detail that is in there in many cases is confusing. There are various categories. There is a multiple indication of a condition the plant, a status of the plant, and the relation between these is not clear. Therefore, one could go several ways in the course of applying this guide to what he was doing in detail.

MR. SHEWMON: Are you saying it is ambiguous?

MR. RAY: In some respects it is. It could be ambiguous from one's interpretation of it.

MR. MATHIS: I might make one other point that came out several times yesterday, and that is the importance of dividing this whole array of instrumentation into specific objectives, such as the Safety Parameter Display System, which we have talked about before as being a necessary aid to the operator.

If you take and break the whole series of instrumentation into groups that you are trying to accomplish the functional objective of, then it becomes more easily implemented in an orderly fashion. I think that is something we should consider because, as has been pointed out, this has been kicked around for a long time, and if we are going to get it off dead center, then maybe breaking it up in smaller parts may accelerate, if you will, the implementation, and I think that is very important.

MR. PLESSETT: Carson.

MR. MARK: I don't have a very clear picture of the total package, of course. For years one has been thinking of the instrumentation which would be important to have in the control room in order to do the right things and know what was going on. One could have imagined what was needed in the Data Link would be a subsequent fact; what was needed in the c ntrol centers might be nearly equivalent.

So my question is whethe or not the addition of the data link seems to have added to the data required or in what way has it affected the proposed Guide.

MR. KERR: Carson, it is a good question, and I do not know the answer because the parameters that are listed are not identified specifically by saying this is needed for the Data Link or this is needed for STF.

MR. MARK: You said the changes have been added --

MR. KERR: I have an idea, for example, from having listened to people who discussed the Data Link, and I remember from our early discussions the number of 100, I think, was introduced as an appropriate number of variables. We need about 100 variables transmitted in order to fully understand the status so that we can give advice, and perhaps if needed, maybe go beyond giving advice.

I therefore have an idea that to some extent, if the number of variables listed in Reg Guide 1.97 -- there have been a number of inputs to the Guide -- one of the inputs from those people who are responsible for the Nuclear Data Link, whether they looked and said those things that we need as part of the Nuclear Data Link are a subset of what is needed in the display center.

I have not seen -- in fact, one of the comments that I made to the staff yesterday was that the information that had been made available to us did not permit me to understand from whence came the specification of variables. I am sure there exists, either in somebody's mind or on paper somewhere, a logical justification that one could follow.

The information made available to me has not permitted me to share in this understanding, so I think your question is quite appropriate, but I do not know the answer. I think you might want to ask our consultants.

MR. PLESSETT: Would it be all right with you if we did it in alphabetical order?

MR. KERR: If you start with the lower end of the alphabet first, I will agree to that.

MR. PLESSETT: All right, Zenon. You are certainly --

(Laughter.)

MR. ZUDANS: Most of the items have been already mentioned. However, I will try to summarize. I think there has been a lot of work done, naturally, and the selection is very broad, and I cannot think but of one variable that is missing from the list, and that variable was discussed and I will repeat it again. I consider reactor coolant inventory as an important variable, and somehow one would have to find a way to handle it. I was told it is very difficult.

What appears to me is that by designing this Reg Guide, engineering judgment was made at the tail end. People sat down and decided, take this parameter, do I want it or not, and on the basis of the systems, the decision was made, yes, I want to do that.

I would like to have seen something like AFI did, for example, where the engineering judgment was made at the front end, where they decided what the objectives are, set up a set of criteria and set up some mechanical procedure which then resulted in selected sets that satisfied the objective, satisfied the criteria, and that would be a structured type of approach. I am sure you will hear that presentation.

I am not necessarily in agreement with their criteria. That is another question. If there is judgment, it should be at that level and not at the tail end.

MR. OKRENT: Are there any variables that you think are superfluous?

MR. ZUDANS: No, I don't think so. I think what Mr. Mathis mentioned here is an important thing, namely, they should be reclassified at the tail end. They should be broken up. Some are for Safety Parameter Display Systems, some are for support center, some are for accident monitoring. I would like to see them simply make four or five columns and classify them.

I still think that this is not a correct way of proceeding. I still think they should start from the front end and make an engineering judgment at the beginning and not at the end. I think now specifically that many, many comments -- there are so many comments that I sympathize with Dr. Ray that you read and you are sent to note 14, and note 14 says go there. By the time you read all those notes, you have to be a mastermind, you know, to lay out the scheme and understand where it should go.

What it really tells me is that whoever wrote those notes, I don't think he had a schematic laid out completely. I think industry will have a great deal or problems with qualifications. I think the biggest problem is not the number of instruments. Most of the instruments are already in the plant I tried to get an answer to that question, and later on I was told that if you looked at all the instruments and forget whether or not they are qualified, 1E or something, then maybe there would only be about 20 additional instruments needed.

If you look at the instruments as required now, you need between 163 and 180, and the main reason for having such a large number required is because of qualification. I think the qualification could be spelled out in a more general way, and I don't think I have found anyplace that you have to qualify the instrument at the location where it is installed. Those parameters should be covered in one way or another.

So I am left in the dark in the sense that I do not have a full picture which would say here are the instruments already in, these functions are being performed by those intruments; here are the instruments we are adding, these will perform additional functions. That is not there. That picture is missing. So, in a way it is too prescriptive; in other ways, it is simply too loose. That is about all.

MR. EBERSOLE: Pardon me. Did you say, when you mentioned the inventory, that it could be a parameter of interest, that you were told it is hard to measure?

MR. ZUDANS: That is correct. They pointed out to me that they are measuring the level in the sump, the containment, the drain tank. I asked whether they measured level in the auxiliary building. This is not a single item; there are several items that may lead you to that.

MR. SHEWMON: I would be very interested in knowing whether or not the core and pressure vessel has water in it. But when you say you would like to know the inventory and then you talk about sumps and other things, what do you mean by inventory?

MR. ZUDANS: You have to know how much water you have in the primary coolant system.

MR. SHEWMON: Is that the inventory you mean, or do you mean every inventory inside the containment?

MR. ZUDANS: I mean the reactor coolant inventory, whether it is sitting in the primary coolant system or has been spilled over into a drain tank. It was also discussed, another item that would be equivalent, it might be a level indication in the reactor vessel.

MR. EBERSOLE: Is it fair to say in the boiling water reactor it is necessary to say, you know, the water in the pressure vessel, it is certainly necessary, you know, the pressure, that you don't destroy that mechanism because otherwise you will prematurely attempt to open the low pressure valves and you will fail. So there is a survivability requirement on inventory and pressure in the PWR vessels, and I don't know why it could not be extended to the PWR in the light of new requirements post-TMI.

MR. ZUDANS: We were told there were now no instruments available to measure the level in the reactor vessel.

MR. EBERSOLE: Is that inventory, in your view?

MR. ZUDANS: I would say that would be equivalent. Not inventory, no.

MR. EBERSOLE: I guess I do not understand why the matter is at issue, then, as to why you cannot measure level in these vessels.

MR. ZUDANS: I guess we were told it is not easily derived.

MR. EBERSOLE: Is that the staff's position, that you cannot measure the level in a boiler or pressurizer where the pressurizer has the new post-TMI requirements?

MR. BENAROYA: I did not hear the question.

MR. EBERSOLE: We are discussing whether you can practically and feasibly measure level in reactor vessels in light of the fact that you have to do it in a boiler at present, and presumably with the new additions made by the Action Plan, we will be doing it in the pressurizer.

MR. BENAROYA: That is correct.

MR. EBERSOLE: So the matter of measuring inventory --

MR. ZUDANS: Pressure vessel.

MR. EBERSOLE: That refutes the thing that you said, that inventory is not measurable. We are going to do it, right?

VOICE: (inaudible)

MR. EBERSOLE: You must refer from the level that you had a void.

MR. ZUDANS: Whatever the resolution, the Guide does not address that item. I pointed that out.

MR. EBERSOLE: I think that seems to be an omission, then.

MR. BENAROYA: It is not missing. If I may add a few words there, what we are saying is that in NUREG 0660, we are saying we need the level in the reactor vessel. It is under development. Now, Guide 1.97 says that anything that has not been developed yet does not include it. That is why it is in the cover letter. As soon as it is developed, then it will be installed.

MR. EBERSOLE: That is PWRS.

MR. BENAROYA: The boiling water reactors already have that set of requirements.

MR. EBERSOLE: Right.

MR. ZUDANS: There was another item discussed. Maybe I could mention it. In GE's presentation on thermocouples in the reactor outlet, I feel very strongly they have a good point. That is all.

MR. MARK: Who has a good point?

MR. ZUDANS: GE has a good point. I do not think you need them.

MR. LEWIS: I am impressed by your comment that you don't think any of the variables are superfluous. I have not looked at the list, but in looking at this list of the comments, I find a number of so-called public comments, industry comments, in effect, that certain variables are unnecessary, to which the staff answer is sometimes they are necessary, other times they are defense in depth. Defense in depth seems to me -- correct me if I am wrong -- to be another way to say I don't know why we need them.

MR. BENAROYA: That is not correct. Every one of them has a very specific meaning. We can answer the question for the requirements to each and every item that we have in the Guide.

MR. LEWIS: But you chose not to.

MR. BENAROYA: It becomes a monumental task to include the requirements for each one and the different conditions because each guy has a different idea what should be there.

MR. KERR: Let me give a slightly different interpretation because I asked what was meant by defense in depth. I think the answer I got was that in order to provide some diversity and additional reliability, there will be occasions on which one might trust what I would call the primary indicator, and therefore a secondary or tertiary indicator would be helpful.

Defense in depth, I think, means, in effect, the assurance of additional reliability and a method of eliminating ambiguity that might occur in either some expected or unexpected situations.

MR. ZUDANS: Now I will answer the question. It is my impression that disagreement between industry and staff comes from a different interpretation. Industry is talking strictly about accident monitoring instrumentation. The Guide covers definitely a much, much broader scope. It is not defined as such. It is defined -- we want to monitor system status. To do this, not an individual reading will do. You may have to combine the readings. So there is no talk about processing signals.

I think this discrepancy comes from different objectives, and I am not against covering broader objectives. I think it is better to cover it in one place than to set up a group for SPDS, a group for technical support center, a group for control room, a group for AMI. I think those groups will never speak to each other very well.

So it is a good thing to have everything together, but why not call it by the right name.

MR. LEWIS: That is the reason I asked the question. I thought this was instrumentation to follow the course of an accident.

MR. ZUDANS: It is not.

MR. KERR: Incidentally, Hal, with the exception of Zenon, who wants to add one additional parameter, I did not find anybody else, either within industry or staff, who thought that any variables were missing.

MR. OKRENT: I am sorry. I have to take issue with the statement because, in fact, there is the report that we talked about recently, NUREG CR-1440, which takes the different approach to looking at instrumentation following the course of an accident, as you know, in which they -- if you have not looked at the report -- have looked at several specific sequences like, for example, the check valve accident and some others. In fact, this is not unlike part of an ACRS recommendation and report sometime ago suggesting, namely, that we take some of these specifically and follow through and see what happens.

MR. MOELLER: Could you use the mike?

MR. OKRENT: I am wearing one. I cannot help it if it is not on.

(Laughter.)

MR. OKRENT: What they try to do here is follow through these sequences to see at what point it would be potentially useful for the operator to have certain kinds of information. In fact, they have identified a considerable number of items which were not in the previous version of the Regulatory Guide, and some of which are not in this version of the Regulatory Guide as far as I can tell. And this is not a complete study of all potentially interesting sequences.

In any event, I think it is relevant to note that the industry approach was not this type of approach, and therefore, in fact, would not have picked up this kind of information.

MR. KERR: In a generic sense, Dave, it seems to me that the industry approach is more nearly that approach than the staff's approach. They did not follow accident sequences, necessarily, but they did, it seems to me, say, you know, what is the task we are trying to accomplish.

MR. OKRENT: Then you have to get back to Zenon's comment. They said what is the task we are trying to accomplish, and then they set certain criteria which did not encompass this type of thing, so they arrived at a list which in their previous version, in my opinion, was incomplete. I don't know what the new standard is, so I cannot comment on it. But I remember that their earlier result was incomplete. I am willing to stand on that position.

MR. KERR: Back to my statement about variables.

It is not clear to me from the comments made yesterday whether this report would introduce any new variables or not because I think we have not gone through and made a complete listing. At least we found a number of cases in which that report referred to an earlier version of the Guide, and the later version does include the variables.

MR. OKRENT: I think, again, there were some that were not in and I asked if they could tell us today about these as to why and so forth. Again, this is not, I think, furthermore not a complete review, but it is a different kind of logic which has not been, in fact, pursued systematically by either the industry or the group preparing the Reg Guide.

The staff has factored some of this into their current Guide. There are some things in here that they have not.

MR. SHEWMON: The staff would like to comment.

MR. MOELLER: Mr. Benaroya and Mr. Zudans.

MR. BENAROYA: I have to differ with Dr. Okrent's statement that we did not take into consideration -- we certainly did. We looked at it and we incorporated all the ones that we thought were necessary. The ones that we did not incorporate are for good reason. It might be subjective, but there is good reason.

MR. OKRENT: Again, the question was are all the parameters in there? I will stand with what I said. They are not all in there. There are suggestions for other parameters that you have decided, and I am waiting to hear the bases for why, but they are not all in there. Furthermore, this is only a partial study of this type. So I do not want the impression that you have covered all of them. I am not arguing that any of these additional ones should not be in there. I am waiting to hear, in fact, what the reasons are that some of these are not sufficiently important. I am willing to be convinced.

MR. ZUDANS: We certainly don't have all the instruments in the plant in the Guide. That is a certainty. All I can say is if you have any questions on that report as to why we don't have them -- as to whether we have that in writing, no, we don't.

MR. MOELLER: I believe, Mr. Okrent -- the question or the promise yesterday was that you would be explaining that today in your presentation, if that is correct.

MR. BENAROYA: That is incorrect, because we said we could not.

MR. MOELLER: Mr. Lewis.

MR. LEWIS: I just want to understand, because twice now you have said that everything in there has a good reason and that you are prepared to defend it, and that if we look to the omission or inclusion of any parameter, there is a good reason. Yet, the question that I thought Dave was raising was the approach that leads to the suggestion you have made, and to address that question by going through each thing, the condensate level or tank, that is the wrong way to get at a philosphical basis for the selection of particular parameters.

I feel a little reluctant to come into this not having looked at the list, but I have not perceived the underlying structure.

MR. MOELLER: Mr. Ebersole.

MR. EBERSOLE: I just happened to pick up page 18. On the bottom of that, comment 37, it says letdown flow does not perform a Type D safety system function and should be deleted. The staff response is it provides a backup for feed and bleed cooling. I believe it is true that this would be a straight liquid loss. It would be perhaps useful to detect the loss of inventory, but so would seal leakage at the pump seal. So I think there is a little bit of a twist in that answer that I do not understand.

MR. MOELLER: Why don't we leave that as a comment.

Mr. Zudans.

MR. ZUDANS: I just wanted to state how I perceived those different methods. The ANS set of parameters and the way I understood them resulted by consensus of the committee based on stated objectives. The AIF or NSEC set started with all the sets available from everybody else, including the staff set, the ANS set, WASH-1400, and then they set the criteria and worked through the procedure and came up with something that resulted in what they got. The staff set came up, again by consensus, but at the tail end. So they are different.

I think the most logical procedure -- and I stated before that I do not necessarily agree with the set of criteria used -- is that one of AIF. We did not see that report before the meeting.

MR. MOELLER: Thank you.

Let's move on to the remarks of our other consultants and ask Mr. Lipinski for his comments.

MR. LIPINSKI: In my letter to the committee commenting on Reg Guide 1.97, I did express concern with the fact that the Reg Guide provided the measurements but did not discuss how these measurements were to be used. The staff acknowledged that they agreed with this concern and that was not the scope of the Reg Guide. They said there were a lot of human factors to be considered. It was not within the scope of this document.

Now, one of the purposes of the document was to adequately describe the ranges that the instruments had to cover for accident conditions, and this document has done that. Where errors have been committed in the past, the document shows what the expected ranges should be, particularly, say, radiation within containment or some of the peak pressures that should be measured within the containment.

The second consideration was the environmental qualification of these instruments. Even though the instruments may be in the plant now, they are not completely environmentally qualified to withstand the conditions of an accident. The power supply considerations also come out of this as to how necessary some of these measurements are and how they should have their power supply. So, the Reg Guide does identify these requirements fairly explicitly.

Now, one of the differences between the ANS standard and the Reg Guide is the ANS standard took the basis of looking at the functional requirements for the various measurements and then proceeded to define basically prime measurements. But it also says that one should utilize diversity and leaves it simply as a statement in the document but does not proceed to provide examples of diversity to these prime measurements.

The Reg Guide does. It provides the prime measurements that give indication of functional performance as well as backup measurements that are diverse that would support that. If the primary system was not available, the operator then has to turn to secondary sources of information.

The Reg Guide also looked at the functional performance of safety systems such that if the function did not occur, it was up to the operator to determine whether the problem actually existed. Take auxiliary feedwater as an example, whether the valves were open. So there are other pieces of information that show why the steam generator level is not where it belongs. The Reg Guide does this.

On the subject that Zenon brought up, I will use the term "mass balance in the primary system." There are a series of measurments that are available, but one thing the Reg Guide does not do is talk about computational procedures as to how you would use these measurements to derive additional information. The refueling storage tank level is available. Flow rates are available. Level in the condensate storage tank is available.

You can put this all together in a dynamic model coupled with differential equations. If you do these computations, then you can show that everything makes sense with respect to all of the measurements involved. This subject is in a program sponsored by EPRI and DOE. I have been serving as adviser to that program. There have been four or five meetings on that subject.

The purpose of this program is to see how you can take the measurements in the plant, utilizing computer techniques, to provide the operator with assistance in operating that plant. So this feature of providing dynamic models and operating on the information and getting the operator decent information as a result of these calculations is part of that program.

It is not part of Reg Guide 1..97. It is beyond the scope.

MR. MOELLER: Thank you.

Mr. Ebersole.

MR. EBERSOLE: On diversity, one of the main concerns now is whether some of this instrumentation can survive in containment. Do you think Reg Guide 1.97 should consider as an aspect of diversity getting some of this (inaudible).

MR. LIPINSKI: That would be the ideal case, but most of the sensers had to be within the containment where the measurement is to be made. If you have any processing equipment, then you have to look at the transmission of the signal from the senser to outside containment.

MR. EBERSOLE: You say most of it.

MR. LIPINSKI: You are going to make a delta p measurement.

MR. EBERSOLE: You can take it out of the static lines. The reason I say that is that is routinely done in the older boilers. You have virtually no electronic instrumentation. It gives you a great deal of freedom from these tremendous concerns we have about viability of instrument in PWRs.

MR. LIPINSKI: That is a consideration as to what represents the best approach to the situation, whether you do it internally or try to draw the signals to the exterior of the containment.

MR. EBERSOLE: I think that is within the diversity considerations.

MR. MOELLER: Thank you.

Let's move on, then, to our last consultant, Mr. Catton.

MR. CATTON: I really don't have much to add, other than I would like to emphasize the fact that supplying a whole bunch of measurements to somebody (inaudible.) I am not sure what use 1.97 will serve.

MR. MOELLER: Any questions?

Mr. Shewmon.

MR. SHEWMON: I guess I have a comment here from reading one of the staff's comments. My note was question overkill on information, and the staff's response was the operator and the utility must learn to cope.

(Laughter.)

So I guess I am in agreement with you some, but evidently the staff's position is once they have all this information, then it is the utility's problem as to how it gets worked into what control use and what presentation to the operator.

MR. BENAROYA: I think there is a misunderstanding here. Most of the instruments, if not all, in Reg Guide 1.97 are now in existence in the plants. Therefore, I don't understand, really, the question. I would like to be shown how many of these instruments are not in today.

MR. KERR: I might add that I got this response yesterday to a question I asked, and I do not understand the response. There seems to be a feeling, even though there might have been too much information available earlier, as long as you do not make any more information available, you do not have too much.

I don't know what is too much, but I remember that a number of comments on TMI-2 seemed to indicate that too much information was being thrown at people at one time. I don't know whether it was a valid comment, but the staff response to questions of that sort seems to be we are not making any more information available than was made before, so it must be okay. But I do not follow that logic, necessarily.

MR. SHEWMON: Ivan's comment was there is still the job of how somebody is going to use all this good information to increase the safety of the plant. I guess for today's purposes we can simply say 1.97 does not address that. Is that a fair statement?

MR. KERR: It is not clear to me that it does.

MR. BENAROYA: I do not know of any document that could answer that question today. We don't know what accident would happen next.

MR. LEWIS: How can you justify instruments except in the context that they are going to be used?

MR. BENAROYA: That is right. We look at the instrument, we look at the information that will be provided. It is that kind of information that is useful under circumstances that might arise. If we define what the information is, then it goes in. But we do have, I can tell you in each case, what kind of information it will be provided, either to the operator or to his management, not only to the operator. The control room operator is only one part of the problem.

MR. SHEWMON: I think that misses part of the problem in the TMI-2 control room, namely, there were thermocouple readings available to them. They were right. They just did not believe them. There were lots of gongs going off. They just could not cope with all of them.

MR. BENAROYA: I don't think it was not coping with them. The first one -- they did not believe them, that is true. Number two, they were not trained to use them, and that was a problem.

MR. SHEWMON: Part of the training is what we are getting into, but part of the display is also part of the problem. If we can coin a phrase, there is, you know, human factors and man-machine interface. And I think maybe you would agree also that that is part of the problem and has yet to be addressed. I thought that was true.

MR. BENAROYA: That is true. 1.97 only provides the tools to management or to the operator to make any decisions. It does not provide the guidance that you are saying in training. It certainly does not.

MR. MOELLER: Why don't we -- Mr. Lipinski -- and then we will switch and have the staff do their presentation, and perhaps they can address some of these points.

MR. LIPINSKI: There are other activities going on in NRC that do address this question. The plant safety system vector, the concise display to aid the operator to reduce the confusion that went on at TMI is designed to do this. If I had a long list of measurements and then I had a column that said plant safety system vector, which one of these measurements go on to that display panel? You put an x in the column.

I look at the technical support center. The one measurment I can see going to everyplace, including NRC, would be neutron flux. I would only have one set of sensers coming out for neutron flux. It is very reliable. But how I process this information and send it around to different display panels is the subject of another document.

MR. KERR: I do not see -- that is a good example, but when you say neutron flux you have not said anything unless you know how you are going to use it. Invariably the way you get neutron flux is to get the output of a sensor, and what that sensor tells you is the flux in the sensitive region of that sensor, and that is all. That is not what one wants. What one wants is an idea of what the parttern of flux is in the reactor core.

Now, how much detail you want on that pattern and what you are going to use for it and what assumptions do you make in the normal operation of a reactor. You typically assume you have a fairly good idea of what the flux shape is, so you only need to check it at a few points. That is what you mean by neutron flux. In an accident situation, is that still what you mean? One has got to define what sort of accident and what one is going to use the information for before one can know what one means by neutron flux.

MR. LIPINSKI: It is going to tell you whether you are at 100 percent power or whether the rods have gone in.

MR. KERR: It will not unless you know something about the flux shape.

MR. LIPINSKI: In this particular case where it is a gross change --

MR. KERR: That is precisely the point.

MR. LIPINSKI: Up 8 decades or down 8 decades is an important piece of information. Whether high voids is another consideration.

MR. KERR: That gross information is not meaningful unless it tells you something about the flux shape.

MR. LIPINSKI: Okay. Now, as I pointed out, this program is looking at the subtleties for all of the different measurements in terms of what they mean, in terms of validation.

MR. MOELLER: Okay. The staff has a comment, and then we are going to call on Mr. Al Hintze to give the formal staff presentation.

MR. BELTRACCHI: The safety parameters in place, that was evolved from the Lessons Learned Task Force in terms of study in the Three Mile Island accident. Its attempt was basically for detection of safe or unsafe conditions. It was to establish a minimum set of parameters by which the operator could perform this detection function, that he be able to look at it and determine whether it was operating safely or unsafely.

In terms of implementation we were greatly concerned on the human factors aspect of it and therefore we called for such things as the use of meter cooling techniques. I think it is also important to recognize that in integrating the safety parameter display system of this type, you would evaluate the functional aspects of the plant.

For example, you could take such things as a series of temperatures, core exit temperature, steam generator exit temperature, RHR exit temperature, things of this nature, depending on the mode of operation, and put these all on one plot or one graph or one CRT, whatever have you. The functional relationship between these could tell you if, for example, power were decreasing. You would expect them to converge, and if power -- if you lost the coolant source, you would expect them to diverge. These are the types of things you would expect them to encounter in terms of design problems of establishing the safety parameter display system.

I think it is important that the staff does recognize that there are human factors considerations to be brought into play in this design. Again, I want to emphasize that it was basically for detection.

MR. MOELLER: Okay. Let's move on, then, with the formal staff presentation on Reg Guide 1.97, Mr. Al Hintze.

MR. HINTZE: In the interest of time, I am going to delete a good share of the background I was going to give on this regulatory guide to state where we are or set the setting as to where we are. But let me say that the Guide has been out for public comment, and the proposed draft you have now is the result of the consideration of these comments.

The Guide does take a systematic approach. There has been a lot said this morning regarding, well, you cannot understand what we have done. If you understand the ANS standard and the approach that it took, then you should be able to understand the Guide. We have defined the safety systems that need to be of concern to the operator. Is reactivity under control? Is the core being cooled? Is the reactor coolant system integrity being maintained? Is the primary reactor integrity being maintained?

Once you define those functions, then you pick up the variables that will tell you are those functions being performed or are they not being performed. And we have added some backup variables to help or assist in that process.

The next thing we do is tell the operator how his systems are operating and can he use them in mitigating the consequences of an accident, or are they being used.

Now, if you think he has too many monitors to look at, the only thing you can do is delete some of the systems that are there for him to use. If you want to say he does not need to know about certain systems, then fine, tell us what they are, and we will take those out of the list.

But we went through the plant systematically and said these systems need to be operating or can be brought into operation. These are the parameters that are necessary to tell the operator that they are functioning, and they are included in the list.

The staff and the industry differ in some fundamental areas with respect to Regulatory Guide 1.97, and these areas are in the scope of the Guide. In our view the differences are not so great that the ANS-4.5 standard cannot be used to meet the enlarged scope as proposed by the NRC staff.

The same principles outlined in the standard apply to the increased scope. First, the ANS standard limits consideration to accident monitoring required by the control room operator. The Guide accepts the premise that monitoring needs of the control room operator is paramount and should be given prime consideration. However, in the process it is no great step to add to that list the other monitoring needs of the plant operating organization, so that event monitoring requirements are pulled together in one document in an integrated approach to the requirements. Then the measurement of a single variable can be coordinated into a single set of requirements without measurement. It can then be designed once to meet all requirements for its use. However, this expansion in scope is mostly included in the addition of the type E variables which fit nicely into the ANS-4.5 format.

Second, the Guide expands the scope of the standard in expanding the definition of type C variables. Just to restate, type C variables are those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases -- the difference being the potential for breach of the fuel cladding and the primary coolant boundary, which ANS-4.5 excludes from its consideration.

It would seem imperative that if the operator can be informed that a barrier is being challenged severely, he should be informed so that mitigating actions can be taken.

Admittedly, the potential for breach of the fuel cladding is a more difficult one, but the potential for breach of the primary coolant pressure boundary is as straightforward as the potential for breach of the containment, which the standard includes

Third, the Guide expands the scope of a design basis event accident, which is defined in the standard to exclude operational or anticipated operational occurrences. Anticipated operational occurrences are part of the events for consideration given in 10 CFR Part 50, but are explicitly deleted from consideration by ANS-4.5, including operational occurrences include only type A variables which neither the standard nor the Guide attempts to list because they are considered plant specific. Hence, there is no list to discuss any disagreement. However, the staff would -- however, the standard would require the designer to look elsewhere for the monitoring criteria for such pre-planned manual actions associated with mitigating some events.

The Guide keeps together an integrated approach. Further, the Guide defines -- fourth, the Guide defines two additional variable types not included in the standard. They are types D and E.

Type D variables are essential to provide the operator with information in order for him to use plant assistance in mitigating the consequences of an accident. If only types B and C were required, the operator would know whether safety functions are being performed or things are getting out of hand; but he would not have the information to know what to do about it or what actions to take. He would not know which safety system is functioning and which is not.

If the operator is to be given any responsibility in being the last-ditch stand in performing actions to mitigate the consequences of an accident, that is, if things are not going according to plan and he must take actions accordingly, then the control loop he is in must be closed, and he must be given the information on which to base the decisions and actions.

It is essential that type D variables be addressed in accident monitoring considerations. The addition of these variables does not corrupt the ANS standard, as some have suggested. They fit right into the format of the standard. Specifically, a definition is provided, a design basis is given, guidance on the method of selection is given, guidance is given for determining the performance requirements.

These are the same steps given in ANS-4.5 for determining the type A, B, and C variables. And if those steps are understood, then the increased scope should be -- present no problem.

The staff senses great urgency in seeing that this Guide go forward. The efforts of many people for the past year have gone into its development, including voluntary industry personnel participating in the development of ANS-4.5.

Yesterday and perhaps again today AIF proposed that we call a systematic -- proposed what they call a systematic approach to determine accident monitoring variables; and it sounded pretty good. However, this proposal had a familiar ring.

In July 1979 ANS proposed developing an accident monitoring standard by the process they called a systematic approach also. It is now over a year later, and the standard still is not officially accepted.

This is not said to demean the efforts of the ANS people, because from personal knowledge I know that many competent men jeopardize the effectiveness of their -- the effectiveness of their efforts to their employer because of the ambitious schedule set in developing the standard by the working group.

Voluntary consensus standards do not come easy. It is my guess that the systematic approach proposed by AIF will not produce consensus results any faster than the systematic approach taken by the ANS organization and adopted by the NRC staff. And the results probably will be little different.

It is true the Guide has a broader scope than the ANS standard, and the instrument list is, as a result, longer. There has been expressed agreement that the broader scope is necessary. However, industry has suggested that the additional scope be addressed by other groups and other standards, thus delaying consideration of these vital concerns. However, an instrument is an instrument is an instrument, and we can see no reason for not including the requirements of all accident monitoring instruments in one integrated document, with the exception of one variable, and that is the core exit temperature for BWRs.

There has not been a single variable listed in the Guide that is not already included as a variable being monitored in existing plants in some form. Therefore, we are not talking about a long list of variables that need to be added.

We recognize, as does industry, that while there may be no difference in the number of measurements that are required for plant operation, there will certainly be a difference in the qualification requirements that some of the instrumentation measuring -- instrumentation measuring -- for some of the instrumentation measuring these variables. Consequently, for plants -- for operating plants there may be a significant impact. However, this is no different than what is currently being done with other equipment in the plant that is important to safety.

It has been necessary to institute a review of all Class 1E and associated equipment to verify its suitability for continued use. Most certainly not all reviewed equipment will have to be replaced, because acceptable assurance of qualification will be made available.

Accident monitoring instrumentation will have to be reviewed either under the same program or a similar one, and much of it may be found acceptable with perhaps some changes on range of instrument readout.

Some graduate students at the Ohio State University took the project of evaluating proposed revision 2 to Regulatory Guide 1.97 and concluded that all but four of the variables listed in the Guide were considered by them as essential for accident monitoring.

They concluded that there was one additional variable that should be monitored, and the staff agrees with that addition.

In summary, a voluntary industry group along with the NRC staff has put a lot of effort to produce the standard and the Regulatory Guide. We differ widely in scope, and hence the list of variables, with one exception. No new variables have been added to existing instrumentation. An independent group with no biases has essentially agreed with the list. I fear that more time will not bring industry and NRC closer together, and thus we are specifically directed to reduce the scope of the Guide.

We strongly urge that the Guide be issued. Its application is referenced in the task action plan, and it's therefore of vital concern. Let AIF do their work and then see where the differences are. The Guide can be revised if the evidence of further studies shows that it should be.

MR. WENZINGER: I think Mr. Hintze made the point, and I would like to make it again, that there probably is not a lot of disagreement on what the final list of instruments ought to be for accident monitoring purposes or for monitoring other events of concern that might not be legalistically characterized as accidents.

And one of your consultants has indicated that the AIF method for determining what the list ought to be appears to be a more systematic method. Regardless of which systematic method you use, I think the list is going to turn out to be nearly the same no matter how you do it. Sure, there may be an occasional difference or two here or there. I don't think there is any dispute over that at the present time.

I would like to direct your attention, though, to the implementation section of this Guide and to what we think is probably a reasonable schedule for getting what we now have in the Guide completed, and that is something on)rder of June 1983. Hey, guys, that is a few years from now, e a few; and that is just to get the sensors and the signal conditioning equipment for this list of instruments in place in the plants all over this country.

That does not speak to all the additional work that I am sure you have heard of today as well as previous to today on how the individual displays of this instrumentation will be carried out.

I think it is extremely important that we have installed in these plants the equipment that is satisfactorily qualified in order to withstand the environments to which it will be subjected during these events; and that primarily aims at sensors and perhaps some signal conditioning equipment.

We would like to get as much of that out of the harsh environment, and we will do as much as we can. We need to get started now. If we wait for somebody else to prepare another list and then another list and the lists do not have much difference, what have we gained by waiting so long? Not very much, I think.

What is the cost? Not having qualified sensors and signal conditioning equipment installed in these plants at a reasonable point in time. I do not think that delay is worth it. We can, in parallel with the installation of the sensors and signal conditioning equipment, get on with, and we do intend to get on with how is this instrumentation readout -- that is, how are these signals going to be used by the control room operator, by the support staffs, and many others? And that work should continue and I am sure will continue.

But if we wait to get all of this in place, it is going to be more years before we get on with installing the basic sensors and the equipment that goes along with that that will be subjected to the harsh environment.

These equipments that are going to be in the control room and the emergency response center are not going to see the harsh environments that the sensors and signal conditioning equipment will.

Thank you.

MR. MARK: Bill.

MR. KERR: Do I understand correctly that you are saying we should go ahead and get the sensors installed even though we are not sure exactly how we are going to use them? We are pretty certain we are going to need them, and so we should go ahead and order them and install them.

MR. WENZINGER: I think I said that no matter what list you come up with -- and you can set this group to making up a list, or the AIF to making up a list, or send us back to make up a list -- I don't think there is going to be a great deal of difference between those lists when you are all done.

There may be a difference or two or maybe even ten out of perhaps a total of 100. It depends on how you count them.

MR. KERR: I'm not trying to be critical of what you said. I was trying to understand whether I understood you to say -- I thought you said even though we are not certain how we are going to analyze, or display, or use these data, we should go ahead and order the instruments and install them.

Did I understand or --

MR. MINNERS: Could I answer your question, Dr. Kerr? I think we are working on a new reg which addresses your question of how to display that information.

MR. KERR: That was not the question I asked, Mr. Minners. I asked if I understood his statement correctly.

MR. BENAROYA: The thing is, for each parameter we have a justification, the reason for the information; that will be available for each one, for each and every one of them.

MR. HINTZE: I think we can answer your question, yes. The answer is yes. The list that has been proposed, we have found nobody disagreeing with any particular parameter that is there, so that we have a fairly good assurance that we can proceed at least in getting these sensors and the signal conditioning equipment in.

MR. KERR: Your statement was if the AIF comes up with a different list that you are willing to change the Regulatory Guide. It seems to me if one is going ahead and ordering the equipment right away, which one would have to do in order to get it installed by June of 1983, that a change in the Regulatory Guide may affect future plans, but it is unlikely to have very much effect on existing plants.

Is that about what you had in mind when you said that you would be willing to change the Guide?

MR. BENAROYA: What we are saying is -- how to improve philosophies, how to develop this guide. The next one would be on probabilistic studies. Improve it then. It certainly would be a better method and a better guide.

MR. KERR: Mr. Benaroya, I think Mr. Hintze made the statement. I am curious as to what he considered to be the significance of his statement that you would be willing to change the Guide in response to the AIF suggestions. It seems to me such a change would not have any effect on the ordering and installation of the instruments now listed in 1.97.

Is that your understanding?

MR. HINTZE: I think what I said was the Guide can be revised if evidence of further studies shows that it should.

MR. KERR: If a Guide were released now and were put into effect, then existing plants would go ahead and order so that the changed Guide would have no influence on existing plants, would it?

Mr. Hintze, answer the question on the statement you made.

MR. HINTZE: I have almost forgotten the question now.

MR. KERR: Well --

MR. HINTZE: The answer is it would act like any other Regulatory Guide or any revision to a Guide or any revision -- any increased safety that is in any plant that is already operating. If you come up with some new ideas and you find out --

MR. KERR: That is not like any Regulatory Guide. This is a fairly sweeping guide which affects all operating plants.

MR. HINTZE: The revision of it --

MR. KERR: Your point was that we needed to go ahead and get these instruments ordered, I think, and that was the reason that the Guide needs to be approved.

Now, if one follows that logic, it seems to me that any change that occurs in the revision is not going to affect existing plants because they will have already ordered this equipment, won't they?

MR. HINTZE: That is absolutely right.

MR. KERR: Okay. I mean, I am not trying to argue with your statement. I just wanted to make sure that I understood what you considered to be the implications of it.

MR. HINTZE: That is true in any case, any case that you give approval and find something needs to be changed. You go ahead and make the changes.

MR. KERR: I have a final question. Since this is just a Regulatory Guide and since I am told that there are alternate methods of satisfying the intent of Regulatory Guides, what are the alternates that the staff would find acceptable to 1.97, or are there alternates that the staff would listen to if this were issued as a guide?

MR. BENAROYA: If you give me an example, I will give you an answer. I am not a designer. I gave you the best parameters that I thought would be. We are still limited to the ingenuity of engineers.

MR. KERR: You would listen to proposed alternatives to methods of satisfying, even though it is not quite clear what it is we are satisfying. 1.97 is just a list of instruments. It is not a requirement on monitoring systems or information systems.

MR. BENAROYA: If the systems engineer and the people that have gone through it would know what the requirements are for each condition -- and given the alternative they -- I am sure they would be able to evaluate it.

MR. EBERSOLE: I think I had better refer to -- in the Regulatory Guide, to the first page, the introduction where criterion 19 is mentioned, and to say something about a tremendous hassle we may be into if we do not look carefully at what criterion 19 means and how it has been applied in the field.

One of the features of criterion 19 is that, "Including the necessary instrumentation at appropriate locations outside the control room be provided with the design capability for prompt hot shutdown of the reactor."

That requirement has been interpreted by industry with a tremendous breadth of conservatism. The least conservative interpretation of that is to provide some additional terminal boards in the main control room and simply extend some wires off to some distant dial some place and still be totally dependent upon the integrity of the control room and the apparatus within it, thus leaving you completely dependent on the viability of the control room.

When this was first noticed -- it was about 12-odd years ago -- we thought that interpretation was a little short of immoral, and that it was necessary to take the view which would be taken by the man on the street that surely the requirement for operating from a point outside the control room would take into consideration the loss of that focus in the plant as a means of controlling the plant, not the fact that simply you could not want into the room because it was filled with something like a stench from a skunk.

We are about to get on the same track again with the instrumentation following the course of an accident. We're not saying anything about how the system is being put together except visualizing it as a bunch of dials some place, and we are not looking at the interconnected network, the source of the signals, or the degree of dependence or independence that we are going to align the plant, whether or not we are going to repeat the generality contained in GDC 19 here, again on instrumentation following the course of an accident.

This goes to a third room called the Accident Response

Center. I see nothing in this guide that gives us any indication how we are going to solve a standing dilemma on GDC 19 as it affects this guide.

MR. WENZINGER: You are absolutely correct, sir. We are not going to solve that problem with this guide.

MR. EBERSOLE: Unless you do, then this guide will have the weakness at its foundation, that it will be -- it will have little or no effectiveness. You have to solve it. You have to say something about the independence of how you weave these instrumentation systems together or make them independent.

MR. MINNERS: I think that is a different question, Mr. Ebersole. The first question of whether you are going to have a means of shutting down the reactor outside the control room is, as was said, not addressed by the guide. You now seem to be questioning -- are we going to have some criteria for how this instrumentation is going to be put together into an information system that can be used during accidents, and we are working on such criteria.

They are very difficult to write, and they are impossible to write, I think, unless you have the list of instrumentations first.

MR. EBERSOLE: You are going to do it in two steps then?

MR. MINNERS: Yes, sir. It would be nice if we could do it all at once, but I do not think that is a practical solution. I am sure that it exceeds my abilities and most other people's abilities to do it. It is too big a problem to do in one big --

MR. EBERSOLE: I think you have to acknowledge the presence of the problem within the context of this thing here to assure everybody you are going to put it together right.

MR. MINNERS: Well, if that would give people comfort, I guess that is -- could be done. But I think there is documentation that shows we are working on the other parts of the problem. The Action Plan addresses the question of first we are going to do the Reg Guide, then we are going to do the integrated systems of the safety parameter display, technical support center, emergency operations facility, and the nuclear data link, which all are related closely because of the data needs, which is going to be based on Reg Guide 1.97.

When we get that done, we are going to go on to the diagnostic system, which is another step that you go beyond, and then possibly we may go beyond that and talk about having system status, Reg Guide 1.47, and approving that. There are lots of things we can do.

What we have tried to do in the Action Plan is put some order and priority to them. There may be disagreements about that, but we think the Reg Guide is the highest priority, the first thing that ought to be done, and is necessary. We are trying to attack that problem with a limited scope so we can get that done, and then we take another step.

MR. EBERSOLE: We can say now it is just sufficient we consider the dials, the instruments, the recorders, the instruments on the face of the board here in this, not how we are going to put them together.

MR. MINNERS: I don't know what you mean by sufficient. Obviously it is not sufficient in the long term, but you have to start somewhere.

MR. EBERSOLE: For the purpose of this guide.

MR. MINNERS: For the purpose of the guide the scope is stated, and I think it fulfills its scope. And then there is a broader scope that has to be addressed, and we are addressing it with other documents.

MR. EBERSOLE: All right.

MR. SHEWMON: Your name, sir, is Hintze?

MR. HINTZE: Hintze, yes, sir.

MR. SHEWMON: Hintze, okay. You made a point, you said the report had been independently evaluated. Did you refer -- there you are talking about this whole issue which nobody seems -- you have seen somehow apparently. I teach at OSU. I have a lot of respect for the students, but I have some reservations about the completeness of a homework assignment done in my own department.

(Laughter.)

I might even -- okay. I just wanted to be clear that that was the independent assessment you were alluding to.

MR. HINTZE: Yes.

MR. SHEWMON: The other thing was that you talked about the importance of the C and D variables or D and E, which would apparently monitor the status of systems, and you wanted to get on so that the operator would have this. Yet, we are not putting in new instruments, you tell us, and I cannot quite imagine -- I am unclear about what it is that the operator does not have now that he would have under this system with the addition of new instruments or something.

end t3

beg 4

MR. HINTZE: I think this relooks at the instruments which we arrived at the list in an entirely different way from what the operator -- what the designer of the plant did. In coming up with the instruments required for operation, we said that these instruments are important for accident monitoring under the D class. And we put certain qualification criteria associated with those instruments -- now, the operator will have --

MR. SHEWMON: This will be the same monitoring equipment had before, but now it will be a redundant or more stable power supply.

MR. HINTZE: You would relook at them to see if they will withstand the criteria which you would say is necessary for accident monitoring. You may have to upgrade them some, but what we are saying is it is not a new measurement.

MR. KERR: Yes. There might be entirely new sensors, new systems. There might be new installations, Paul. I think the point was one would be measuring the same variables, but one would not necessarily be using the same physical instruments that are now in the plant, isn't that correct?

MR. HINTZE: Yes. In most cases it probably will be just an extended range on the instrument readout.

MR. MARK: Bill, would you say that brings us through the NRC staff presentation?

MR. KERR: Does that bring us through the NRC staff presentation?

MR. HINTZE: Yes.

MR. KERR: Yes, I would say it does.

MR. MARK: I think we would want to hear the comments -- Dave.

MR. OKRENT: I would like if I can again to briefly ask Mr. Benaroya if he could go through the V sequence again and look at those things that this NUREG/CR-1440 indicated could be useful for this V sequence, and tell us quickly how the staff reacted to these and why.

Is that a fair request?

MR. BENAROYA: It is an unfair question, and it is not a short item.

MR. OKRENT: Take five minutes instead of two.

MR. BENAROYA: How about half an hour?

MR. OKRENT: I thought you could look through this table, that is Table 5.1, and select those that are important to the V sequence and comment on that.

MR. BENAROYA: How about you selecting the items and I will give you the answers?

MR. OKRENT: Well, I assume that you knew what was in already, where you had reacted, that these were already included; and it might be more efficient, but if you prefer, let me try a few.

MR. BENAROYA: I did this with you last June, two months ago.

MR. OKRENT: Let me say in general I think when the staff comes down to meet with the Committee it would be useful if they would at least try to answer questions in a direct way, and while you did it in June, I was only able to get this report -- in fact, the Subcommittee got it two days ago or a day ago, so you have had a head start.

But anyway, let's try a few and see what the logic is.

MR. LEWIS: May I interrupt long enough to say that I agree with what Dave is saying. I don't think you are being very helpful.

MR. BENAROYA: I certainly will try, but what Dr. Okrent is saying is not a two or five minute item.

MR. LEWIS: Splendid.

MR. BENAROYA: I am most willing to go through each and every one of them.

MR. OKRENT: Well --

MR. BENAROYA: Why don't we start at the top and go one by one?

MR. OKRENT: I was hoping you could pick out those that you thought were the most significant and where there might be a question, but let me look through and make a few quick guesses.

Now, on page 54 there is an item, the second item on the table, positions of key valves and safety-related systems. Then under the V sequence it says "indication of capability of systems to operate when called upon, diagnosis of failure."

Now, I am not arguing for or against this. Is this in the guide? If not, why is it not, and so forth?

MR. BENAROYA: Dr. Okrent, the valve positions, as I said yesterday, are somewhat of an ambiguous condition. We do not have all the valves in the Reg Guide. We have very few of them. We have some. We could not include all of them because it is a very extensive list. We have to put all the isolation valves in the containment really. We don't -- that is why we did not include all the isolation valves, the position of key valves, whatever key valves means.

MR. OKRENT: Well, if I understand correctly, what they are interested in here is is it possible by monitoring certain valves to tell that the V sequence has occurred, and if so, in what leg. Is that part of this?

MR. BENAROYA: Yes, for the one they have evaluated in this case. You are absolutely right. But there are other cases which would involve other valves just as important. And you see we made a very arbitrary decision as to which valves we were going to include, which is a very small number in comparison to the total number of valves that we should be monitoring and are monitoring.

Let me add that in many cases we do monitor flow which gives the equivalent of whet r there is an isolation or not.

MR. OKRENT: Well, let me pose the question more generally. For the V sequence, which I suppose --

MR. BENAROYA: Which kinds of plants?

MR. OKRENT: The V sequence, the failure of the check valve or isolation valve, high pressure, low pressure system either in a BWR or PWR. Do you feel that the things that are in the guide give adequate information at a proper time so that there is not any additional help to the operator were he to have some of the things recommended that are not included, or that in fact although it might be nice to have the valve positions indicated or other things, this is really just so much additional effort that you are not sure it is worth doing it, or in fact it is just hard to do? It would be nice to have it, but it is hard to do.

I am not quite sure I understand which of these three or maybe a fourth position is the one. Could you help me?

MR. ROSENTHAL: We did go through that document. Okay. For event V, in particular, they decided a valve position would be useful. In this case the guide requires LPSI flow, and it requires some level indication, so what you have is you have a subcompartment sump level that says you are filling up the sub-compartment and you have runoff flow through the LPSI. That is another way of saying you have that problem, and we are saying it would be appropriate for the operator to look and then turn off that valve.

I view this as in a sense several mechanistic tests of how good the 1.97 list is for the event V that they looked at for this one PWR as a test -- it is a test of how good the list is. They are saying that gee, it would be nice to have that specific valve, which would be different on a different plant.

My response is to the sense that I have flow and sump level indicated, I pass the test because the operator ought to know what action to take.

Now, we could go through and decide on a plant-by-plant which valves are important, but if you go through the SAI exercise in a different plant you will come out with a different list of valves.

MR. OKRENT: Can I comment on that briefly? First, I am sympathetic with the point of view that time -- it is long past time to move in this area, and for that reason I for one am not going to push changes of my own. I think if there is something wrong in there, I would want to hear about it and have it deleted, fine. But with regard to the event we have just been talking about and your reaction, my own feeling based on let's say limited exposure in trying to look at transients in a simulator and so forth is that here, while it sounds very nice to say yes, if he looks at flow and if he looks at what is in the sump, he will -- he ought to be able to figure out, and certainly in the post mortem he will have figured it out.

The status monitoring panels that tell you which valves are closed or open or certain other things are a much more effective way of communicating information to the operator. And I would suggest that in connection with this type of an event and others of this sort, once hopefully you have a Reg Guide 1.97 in process -- as part of your next stage as to how you use this information, you might also try to bring some of your own operator trainers into the act and say now, suppose this event occurred. Do you think the operator would look for this, or would it be better to have other information, because your trainers are not even thinking about event V, and none of the operators are hearing about event V so far as I can tell.

I think in the next round of thinking I would suggest that you not ignore the kinds of things that come in here as to whether there in fact may be some significant gains if the measurements can be done and if they can be displayed in an effective way.

MR. ROSENTHAL: May I make one more comment? I believe you are using event V as an example.

MR. OKRENT: Exactly.

MR. ROSENTHAL: To the degree SAI has described a sequence of tests we have not ignored the document. We did go over the document. We tested that document against our list, so we have done our homework. Your comment about involving the operators I think is very well taken.

MR. MINNERS: I think we are trying to involve the operators. One of the items that is in the Action Plan on the NTO list is to have operators training to try to -- to show them how to mitigate accidents with the equipment they do have. I think that is an important thing that should be pressed harder possibly, and in that training exercise I would think that people would find more examples of where maybe we have open places in our instrumentation and control requirements.

And I do not think by sitting around in an office in Bethesda that you are going to find all these holes. Hopefully, the industry is going out and training their operators to try to control accidents, and during the training process they may find out they need more information, or less information, or different information. And I think that is an effort that ought to be pressed a little harder than I think it is being pressed.

MR. OKRENT: Maybe within the NRC staff overall you have some people who are in fact training resident inspectors and so forth, and if some of them interacted with some of you, that might be also useful.

MR. MARK: Max.

MR. CARBON: Mr. Hintze, do I understand correctly that you believe in the big majority of the cases this will involve extending the range over which variables are read for existing instruments, rather than providing completely new instruments with bigger ranges?

MR. HINTZE: I said yes, that perhaps most changes will be in the extended range area. Certainly all of the rest of the instrument panel will have to be reviewed and evaluated, as we are doing the rest of the Class 1E equipment in each plant. But as a judgment factor I would say that that is probably right.

MR. CARBON: I am not sure I understand. Is this basically -- do you expect this to be a modification of existing instruments or replacement of existing instruments?

MR. BENAROYA: Most probably in some cases it would be replacement. In some cases they might extend the range. It depends. In most cases it will be a new instrument. I would assume it would be with my own experience. I don't know. My own experience says it would be a new instrument in many cases because the ranges are extensive.

Dr. Okrent, to give you a fast answer to your questions, excluding the valve locations because that we can argue for a long time, on page 54, auxiliary building temperature and radiation level, yes, they are both in the guide. The boron concentration on page 58, yes, it is in the guide.

MR. OKRENT: Okay.

MR. BENAROYA: The others are all valve positions which depends.

MR. MARK: I think perhaps we could plan to hear from the other people who have comments, and we will take before starting that a break until five to 11:00.

d tp 4 (Brief recess.)

MR. MARK: May we resume?

We have comments from the ANS 4.5 working group which I believe will be given by Mr. Sommers.

Is he here?

MR. SOMMERS: Yes. My name is Dave Sommers, Consumers Power. I was a member of the ANS 4.5 working group.

I apologize tht the chairman of our committee could not be here to make the presentation. I am the designated hitter.

(Slide)

ANS 4.5 has been discussed by the NRC and relayed by comments by various members summarizing yesterday's presentation.

The major points of difference with the reg guide -- inspecifically discussng the reg guide, I would like to make four major points in terms of points of contention and then go into some specific detail.

First off, the points of agreement are too few in number and in content between the reg guide and ANS 4.5. This is a result of the scope.

The aras of difference have not narrowed since the December 1979 issue. We felt that was an unexpected result. We had some feedback yesterday from the NRC that some of our areas of concern in terms of variables that we had designated as barrier monitoring, type C and type B, that were missing in Reg Guide 1.97 sections were due to the NRC restructuring or format difference as opposed to a content difference.

Although that is a format and a minor problem, we would point out, going along with some of our philosophical differences, i is an important thing for the designer that the various variables, even if they are repeated, should be brought forth as part of -- under the various functions, functnal requirements that have been specified.

Point three: we would like to mention, and especially since there were some remarks to the contrary, ANS 4.5 does have a rather broad base of industry support in terms of accident monitoring variables and requirements.

Now, specifically I will give you a time frame of what we are talking about.

(Slide)

Our group was formed in July of last year in a rather expedited manner and developed a draft standard which is going up before NUPPSCO for reconsideration, and at this time we do not anticipate any additional comments.

And the bottom line is we expect an ANSI standard by the end of this year and distribution by February.

MR. CKRENT: What is NUPPSCO?

MR. SOMMERS: Would somebody help me with that acronym.

MR. WENZINGER: Nuclear Power Plant --

MR. SOMMERS: But -- thank you.

The point being, however, that in terms of the standard, the standard is a consensus standard. It has been balloted, received by NUPPSCO, and it is going through its last,final stages. It should be issued as an ANSI standard by the end of the year.

(Slide)

Finally, our point four, a general, overall view, we feel that Reg Guide 1.97 does require a rather severe overhaul in terms of the scope, which you have heard summarized by various ACRS members in terms of addressing the variable types that are pertinent for accident monitoring for a crisp presentation to the operator, in terms of the audience who the information is going to and in terms of the purpose, whether it is accident monitoring, per se, or whether it is accident monitoring -- emergency planning informaton to the NRC.

We feel it is very important to break out in the reg guide, again from the designer's standpoint -- the utility's standpoint of what we are supposed to do with the information that we are be requested to provide.

We feel in terms of the guide itself, as I alluded to, that the requirements should be tied to the objectives and functions; and that is omissions should not be left in the guide, since it is such -- if it is going to remain a very detailed and prescribed guide, then your -- I feel obligated to put in, even if it does tend to be redundant, the functions that you are using for diversity under the various functional requirements, under type B and type C.

In terms of the format clarity and jut the general ambiguity, in the reg guide, as it stands today, the example of format that I have given you at this stage of the game, 10 of the 20 possible items that we have in ANS 4.5 do not show up in the type B and C requirements.

This is, as I stated, a format problem more than a technical problem, although it would be very confusing to a designer trying to use the document right now.

Also, in terms of the general ambiguity statement, it applies just to the -- if you take a look at the notes on the various tables, between tables one, two, and three, in the reg guide we have 58 notes as opposed o six in ANS -- the ANS document, and again to indicate the complexity, which is going to be also difficult for a designer to handle.

I would like to just say in terms of the last item that I had up there which was reasonableness, I think Dr. Kerr asked the question yesterday -- and I could not think of the words and I spent a half hour -- you asking me the question in terms of the different objectives --

reasonableness is, I think, an objective that you have to have.

You have to ask yourself, what is the operator going to do with this information, and it should be applied to each parameter.

In terms of reasonableness, the man-machine interface has to be addressed, and you have to come at accident monitoring from a systematic approach which identifies the functonal requirements first and then goes about identifying the parameters later.

Thsi is lat lest the ANS 4.5 position.

(Slide)

Since, Dr. Kerr, I think you obligated me to keep this under 10 minutes, I will start jumping from my presentation from yesterday to go to some of our specific comments.

Significant differences between the two documents, just a brief overview: I think a significant thing to point out in addressing the tables on the bottom here, the technical requirements, that the table -- the specific technical requirements are organized by table and by qualification criteria as opposed to the ANS document which is organized by function and varaibles.

That is, you can have under a functin that you have stated as of safety significance --

(Slide)

Just throwing this up very briefly, the three types that we have, the critical safety functions and the barrier integrity, that when you look at these functions, we have tried to address these functions or the technical requirements with regard to the express functions as opposed to addressing under the various -- to address them by the qualificaton criteria used in table one.

And one of the things that leads to confusion is that after expressing an important safety function -- identifying the function, we find variables -- kind of a mixed bag of variables with different qualification criteria.

Again, that becomes difficult if it is prescribed for a designer to really ascertain how he is to approach this particular problem.

(Slide)

I would like to delve into some of our detailed -- the specific, detailed comments at this point.

First off, we felt that the reg guide was not extremely systematic in its approach. Well, you have heard this in the summary, and if you had been here, you would have heard it for about five hours yesterday from industry.

We really feel that identifying the functions and going from there is important for the designer to be educated. The form analysis based on the functions in defining instrument criteria is the next logical step.

We felt the scope expansion was unsupported; specifically, ANS 4.5 was control room operator oriented. Reg Gude 1.97 addressed emergency planning and emergency planning facilities. Consequently, as a base reference, ANS 4.5 is really not applicable to the other establishing functional requirements or data requirements for the other locations.

The same methodology could be approached for those different locations, but we neither had the expertise at those meetings or the expressed scope of ANS 4.5 to be expanded into these other areas.

Specifically, we did not feel that in the time frame as I had thrown up in under a year turn around that we could address all the problems in a functional sense.

And so we set about grasping only the control room and the control room operator as something we could attempt to handle.

Finally -- likewise -- excuse me -- the scope expansion also blurs the AMI focus. Accident monitoring is primarily for the operator. We have to establish what is wrong to get the operator back within the established safe boundaries for safe plant operation.

type D and E variables are not really within the AMI scope. they tend to blur the focus of accident monitoring.

I would pint out that Reg Guide 1.97, Rev 1 recognized the scope limitation when we first were talking about accident monitoring, following the course of an accident, when it stated in the last paragraph of its discussion: "It should be noted in the safety analysis, many parameters may be identified that will be desirable but less essential information for the operator.

"Any instrument used o measure these less essential (i.e., backup) parameters is outside the scope of this guide"

In ANS 4.5, that was our basic premise when we initiated the ANS 4.5 effort; we took it from that pont and went on. We specifically have a problem with the requirements being overly prescriptive in nature.

(Slide)

Our approach was to define functions, to come up with what we defined -- what has been -- has been stated as a minimum set with the understanding that an analysis had to be performed to ensure the plant uniqueness and also generally to be looking at diverse requirements, that an analysis would be performed by the designer to establish what the requirements were.

We were trying to find in the minimum list that we had generated in ANS 4.5 where we could get common ground in short term turn around within industry. Admittedly, ANS 4.5 does not address all the parameters that one might come out with for accident monitoring.

In terms of applying our criteria, it met our criteria in terms of our systematic approach. However, if you open it up to a few additional bits of information, as faras criteria, I think some additional parameters could be added.

We have discussed that within the group, but at this point in time, our feeling was, again, ANS 4.5 has a broad consensus within industry. We can go out and do something. We can move on 4.5 right now.

We regards to requirements being overly prescriptive, since the reg guide -- or at least it is our allegation that the reg guide is not based on funcional requirements in a plant analysis as its basis. The listing with ranges and extended -- the extended ranges and equipment qualification requirements result in a designer really being stuck between either blind compliance or trying to establish some ground rules that were not specifically defined in the reg guide to argue his case against the requirements.

We have such things as position C5 in the reg guide which requires identification of variables and parameters by the designer for defense in depth. We have kicked and bandied those words around, defense in depth, and I am not sure after the conversation that I still know what that means or how I would go about identifying parameters for defense in depth in any systmematic way.

Complexity is not necessarily a virtue. by specifying a number of detailed instrument requirements, we start getting -- we run into anomalies. Position C7 states that criteria -- excuse me -- positin C8 states that electrical isolation must be used for AMI channels -- between the AMI channels and any non-safety ue of the sensor.

Since Reg Guide 1.97 specifies the use of non-1E AMI equipment, we run into the situation that we are isolating between two non-1E systems, which does not make a lot of sense.

Position C8(b) requiring operational availability checking states that one must perturb the variables to show that the channel is functioning. While that is good practice, in general, such things as the high range radiation monitors, which will invariably end up as being separate monitoring channels, we run into the situation that there is a lack of consideration of whether this is even feasible or ALARA considerations of having the tupe of check source you may need to be able to perturbate the variable to get the thig on scale.

We have heard some remarks today that most of the equipment is either available or near being available; I might add that getting the literature from vendors who make statements that they comply to Reg Guide 1.97 is a little premature and dangerous.

There are a number of instruments that are listed, equirements that are listed. the instrument is not available. It is not available today and more than likely will not be available by June of 1983.

I submit as a number of examples environs radiation monitoring, 10^{-6}, 10 per hour range to be environmentally qualified.

This device is presently unavailable. I might note that the range that is specified, the lower end of which monitors the decade below ambient background, which we do not feel is terribly reasonable; it is not AMI oriented for such a range, and potentially it ends up with an ambiguous indication to the control room operator in a substantial accident where background shine is evidenced in the case of a LOCA. The readings may be higher than normal without having unplanned releases to the environment.

We would note that the range that we had put in ANS 4.5 of 10^{-3} to 10^{2} r per hour was determined -- suggested to the NRC following the working group efforts -- early working group efforts, including the work of the input of Dr. John Posten from Georgia Tech and Dr. Monty Schultz from Penn State where we looked and considered a number of the specific problems with this device.

And yet we had this prescribed requirement in the reg guide. I might also add we have such -- we had problems in terms of high ranges for effluent monitors where we have requirements for 10^5 microcuries per cc monitoring capability.

It is my understanding that this reflects vaporizing the core and dividing by containment volume. That certain is conservative in its approach for an upper range. It does not really take into consideration the reality of a stage meltdown -- stage 100 percent meltdown as has been indicated by Sandia reports.

And in some cases you can vaporize the core and divide by the containment volume and you still cannot get 10^5 microcuries per cc for a number of plants, and not necessarily the very small ones.

RCS radioactivity specified 10 curies per cc a real time measurement; I would pint out that that may be possible with a small sample in a lab, but with a 36 inch pipe at 10 curies per cc, that device is not available. It will probably be available in the near future.

ANS 4.5, I would point out, has suggested something on the order of radioactivity levels based on a 100 percent gap activity release followed by sampling. We feel that was an answer to our reasonableness criteria as opposed to what you see here; requiring reliable power indication to provide status of non-1E power supplies seems to be in our mind unrealistic or not practical, not consistent with the power supply itself.

And the basic thing I am trying to get across to you is that in terms of a designer, when you have this type of detailed requirement, the designer is hard pressed not to just lay down and say I will comply.

In the Kemeny Commission, there was an admonition that this Commisson stated -- it stated there was absorbing concern with safety that will bring about safety, not just the meeting of narrowly prescribed and ccomplex regulations.

It is our ANS 4.5 position, the way we are goig about this right now, is this is prescribed and complex and it takes away the responsibility, not legally, but it does in practicality -- it takes away the responsibility of safety from the designer where it should be.

The last item terms -- in terms of problems we have with the guide, is the human engineering aspect.

(Slide)

Again, we have -- if you attached the reasonableness criteria in determining what variables must be displayed in the control room, you find that you have to address the human considerations.

At TMI, where the Kemeny Commission had stated that, it was there consideration that between 50 and 100 alarms represented a severe imposition on the control room operator; we feel that you have to address the impact that the display requirements that are being implemented by Reg Guide 1.97 will have on the control room operator.

Human factors consideration even goes further than that in terms of just the impact on displays, but also to , t r s o how many channels we are talking about that would be added in the control room.

One of the things you run up against as a designer is that you have existing panels that are non-1E and to provide the separation criteria required by the reg guide, you cannot physically go ahead and just change the instrument. You may not be able to put the -- i.e., you may not be able to put the 1E instrument where the non-1E instrument used to be on the panel.

This leaves you in the anomaly or the anomalous situation that you leave the non-1E display in that panel and you put the AMI instrument someplace else.

And you end up with an oprator relying on a non-1E instrument where his controls are and having an AMI instrument locatedin the back of the control room because you have physical constraints with the existing control rooms today.

Other backfit anomalies that you have is the extended ranges giving you more than multiple displays for a given instrument. And you have the situation again from an operator's standpoint, that you either can have one range detector, which is insensitive and cannot be used or should not be used by the operator for normal operations, or you have two displays: one that is used by the operator and the other which is not.

And the one that is not is the one you want to use during the accident, again, to get across from the standpoint that there is a cross purpose or there is an interchange. Accident monitoring, by its very nature, is at cross purposes with human engineering.

Accident monitoring says, give me all the information I can possibly get. Accident monitoring wants to be able to provide as much infomation as is possible. Human engineering says, give it to me in a way I can assimilate it and don't feed it to be faster than I can handle.

MR. OKRENT: Now, I really wonder if you are not choosing your definition of human engineering to suit your argument. It seems to me if you were in the control room and there were an accident going on and you wanted to be able to find out what the pressure in te containment was, if it exceeded the current range of reading or if you wanted to know whether some valve was open or closed and there was no way of finding out, you would be very unhappy with the designers who did not supply the information.

MR. SOMMERS: Correct.

MR. OKRENT: Or the ANS standards group that says it was not necessary, and so forth.

MR. SOMMERS: Dr. Okrent, that really comes under what we were talking about in terms of scope. We talk about accident monitoring in terms of getting the operator's attention, that he has a problem with a small set that characterizes in an overview sense the status of the plant.

ANS 4.5 takes no disagreement that type D instruments have to be addressed, but they are no accident monitoring. They are more diagnostic and should be done on a system basis and should be done and fed in, perhaps, into a data acquisition system so you can go ahead and get something out of this as opposed to having everything identified as accident monitoring and important and too much to comprehend.

MR. OKRENT: There are some plants, as I understand, that do not even have computers now, and those thaet do, they are not required to have them operational, and so forth.

MR. SOMMERS: The point is, the things we see are being generated today in terms of the new requirements, they are going to -- the only way will be computers.

MR. OKRENT: Oh, I think, in fact --

MR. SOMMERS: We are saying, let's coordinate that activity.

MR. OKRENT: Well, what bothers me a bit, as you well know, the industry argued we did not need any of this kind of instrumentation.

MR. SOMMERS: We have been educated.

MR. OKRENT: And now I still find plants running without it and control rooms without it and you go in and you cannot really find out in these control rooms most of the kinds of things we are talking about.

And in what you are proposing, you yourself said, although you tried to make it logical and meet some criteria, it is not a complete set. You have not told us what other things not in the ANS standard, in your opinion, you think should be in there.

What we have is something industry has agreed on. But industry agreed on a fire protection standard before Browns Ferry that was really, I would say, not what they might agree on after Browns Ferry. and the fact that there is an agreement by industry does not necessarily tell me it is adequate.

As I say, it would help me if, instead of telling me -- not instead, but in addition -- the right word is "addition" -- in addition to saying what you think are the things that are wrong or not useful or whatever it is in the current standard, what it is that one can do to accelerate getting the instrumentation in, instead of going through another round of discussion.

And if the ANS standard is incomplete and if that is, let's say, over-specified, then we ought to have at a very early stage a definition in fact of what is the right thing.

I do not think the public -- and I know I don't relish the thought of one or two more years of trying to discuss just where to go.

MR. SOMMERS: To reply to you with respect to ANS 4.5, we could not more heartily agree with your statement. What we are basically saying is that we can take the first step to get off the dime by an endorsement of ANS 4.5 and let's move from there and negotiate.

MR. OKRENT: It may not be adequate ground; an endorsement, I have found, as you know, that is the plateau from which you do not move ovver the next five years. You endorse it if it meets minimal rquirements, but if it is under minimal, then you don't.

MR. SOMMERS: In our scope, we feel we meet minimum requirements within our scope.

(Slide)

There are additional parameters, and you -- we have discussions before. You are well aware that we have individual members having differences of opinions, but in terms of the functions that we are wanting in an overview, a safety overview of the plant for the control room operator, we think we are close to --

MR. OKRENT: I was asking the staff if they had looked at this report and asked themselves are there kinds of things that you get out of this kind of a study, NUREG/CR-1440 that, say, it would be useful to have things that were not on the list.

Has your group taken the approach taken in this report and gone through actual sequences to ask yourself, are there kinds of things that would be nice for the operator to have that did not appear in our list from our criteria?

Have you done this systematically to see whether your criteria are good?

MR. SOMMERS: We have not done it. I say no from the limited standpoint of what I was able to read of that document last night.

MR. OKRENT: Well, all right, so --

MR. SOMMERS: Again, please remember the time frame by which we were operating.

MR. OKRENT: Oh, but let me say, I don't think the ANS group should have waited for EG & G or SAI or whoever did this report for the NRC; it seems to me the industry should have done this kind of thinking for itself.

MR. SOMMERS: I trhink a subsequent speaker is going to give you some idea of some industry input into that. But ANS 4.5 specifically did not.

MR. OKRENT: All right.

MR. SOMMERS: Finally, in terms of human factors, Reg Guide 1.97 does play a significant part and does have a significant impact, and if I might throw up a flimsy that gives a rough idea -- I know I am going to get some differences of opinions from the NRC side.

(Slide)

Independently, I cross checked with a designer. Basically, I just gave the reg guide document and asked what was the impact. and this has been cross checked with myself doing it on a plant I am familiar with and cross checked again with a designer.

The bottom line, I think, is more important when we talk about the total Class 1E displays that we are proposing, the additional one, the upgraded ones. The difference in scope is where you see the differrence in the numbers.

Really, the righthand column is the more important. Significant points are the fact that 95 trend recorder points -- and from a human factors standpoint, depending on how many points on a recorder you accept, you are talking about a rather large number of analog strip chart recorders in a control room.

The power upgrades, tht is just a matter of information for making reliable power for non-1E displays.

The one I would like to bring your attention to is the 163 to 175 instrument channels, and again, depending on what is called out of that information in the control room, that is a lot of additional displays in the control room.

Now, it is one thing to say you have 30 or 40 variables, but when you talk about redundant trains and redundancy on redundancy, and so forth, the bottom line is the instrument channels.

To specifically address a comment by Mr. Zudans, I went home and tried to do a quick check on my numbers, and I came up with 66 brand new instrument channels that would not be in a plant today.

I understand, Vic, that as of January 1, 1981, theoretically that number will shrink by Lessons Learned requirements.

MR. BENAROYA: Thank you, because that is what I meant.

MR. SOMMERS: Again, from a human engineering standpoint, an additional channel is an additional channel, and again in terms of comprehension for the operator.

MR. SHEWMON: Let me understand what an additional channel means to you.

If I want some redundancy so that I want three thermocouples coming in that I can choose between instead of one, is that -- but they all indicate the same temperature -- is that another channel?

MR. SOMMERS: I will have to beg off for interpreation.

MR. SHEWMON: It is your graph, and you are talking about the number of them.

MR. SOMMERS: I will tell you what we assumed, and that was individual displays for individual thermocouples by the requirements as specified in table one. We do not agree with that approach, but that is the way, if you take the letter of the law, that is the way a designer would have to approach it.

MR. EBERSOLE: In that connection, rather than just talk about displays, did you also include information source and degree of dependence or independence and make a more cohesive picture of what you are talking about? You see, all you talk about are displays, the receivers, and awhile ago we were talking about the whole document really just being based on a discussion of the receiving end of all this instrumentation.

Inevitably, we will have to interface with how we will get the power, whether we will retransmit from existing sources, whether we go to intermix with standard instrumentation or not.

MR. SOMMERS: You are reiterating all the questions that I personally would have.

MR. EBERSOLE: Here is a case in point: you talk about displays, but it does not say anything about the source and whether or not it is dependent, independent or 1E ornon-1E or whatever.

MR. SOMMERS: Okay. Let me be specific, then. In terms of the channels we are talking about, if you rack up what we are talking about, the entire reg guide, new 1E channels, non-1E channels and then the 2E channels, the bottom line is wrapped up taking a look at the overall impact.

I did not break out between what was 1E or non-1E, although I do have on the outside the total class 1E displays being 53 out of the 175. The total class 2E's are 110.

MR. MARK: I think we will have to move on unless there are some questions.

MR. KERR: What is the relationship between the 66 number you referred to and the 163 to 175 additional instrument channels?

MR. SOMMERS: I was asked yesterday how many of these instrument channels would be brand new display panels that are not otherwise -- the information is not otherwise displayed in the control today.

The difference is 66 are new bits of information. The total I have up here can represent the difference of -- I cannot get into an existing control room and existing panels and I have to put the new AMI channel someplace else.

MR. KERR: Okay. I understand.

MR HINTZE: Does your tabulation there take into account that -- only the type B and C instruments are required for continuous display, that all others are on demand.

MR. SOMMERS: I cannot say about the independent review, Al. I just gave them the reg guide and asked them to go ahead and do it.

MR HINTZE: What do the numbers mean, then?

MR. SOMMERS: These numbers reflect, if you want, a compilation between the two; they were running very close.

MR HINTZE: You were treating the two as if there were that manay different dials on the panel; that is not true.

MR. KERR: It says channels. That means an information delivery system.

MR. SHEWMON: He did not have any graduate students available so he went to a different, independent source.

(Laughter)

MR. SOMMERS: Again, some of that goes down to the interpretation -- A1, to be specific, the interpretation of that instrument panel or whether that will end up as a display depends on the discussion we had in terms of the uniform building code being used as a seismic category for classification of your non-1E devices.

You may eliminate the use of the plant computer by use of that requirement; that has to be understood.

MR HINTZE: I just wanted to make sure that the guide does not require everything to be displayed, that there are on demand --

MR. SOMMERS: Where we cannot present it on demand because of other requirements, environmental qualification requirements, I am still stuck with the display. Not to confuse the situation, let's just talk instrument channels because that is what I had brought up yesterday.

MR. MARK: I wonder if we should not move on. We have two --

MR. SOMMERS: Could I have just 60 seconds?

MR. MARK: Exactly.

MR. SOMMERS: Exactly.

(Slide)

Just to give you a quick overview of what we would recommend, how to get the next step, and that is it is ANS 4.5's position, whether it is reasonable or not, that Reg Guide 1.97 be split in terms of its content.

We feel there are ways this thing can be done in a more sensible manner that will be able to be implemented by a designer.

One is the AMI, the accident monitoring part being in Reg Guide 1.97; I don't know how practical, but the safety system status may be in the bypass reg guide, the inoperable reg guide.

The effluent discharge items are in Reg Guide 1.21. And again from the standpoint of trying to pull the display requirements together outside of the control room, those are in NUREG-0696, if that is the intention.

But our bottom lines are that each topical section should be self-sufficient; it should have its criteria requirements and variables specified so a designer can go and do his job without being confused.

And I think the last point I would like to reiterate, in terms of encouraging solution flexibility and the use of CRTs, graphics and other tradeoffs for human engineering, it just has to be factored in this reg guide to be effective and to represent an increase in safety.

Did I make it?

MR. MARK: Thank you, Mr. Sommers.

We have a presentation from the General Electric Company.

Mr. D'Ardenne or Sawyer or both.

(Discussion off the record)

MR. MARK: We will switch to the AIF and have GE follow.

MR. COLEY: My name is Bill Coley. I am manager of engineering services steam production department at Duke Power Company, and I am here today representing the AIF Subcommittee on Safety Parameter Integration.

I am also chairman of the AIF Subcommittee on Control room Considerations.

The purpose of our presentation is to attempt to offer a way to approach the intent and spirit of Reg Guide 1.97 and at the same time fulfilling the requirements and placing into operation the emergency facilities in nuclear stations.

The approach that we suggest is an outgrowth of some interchange that has taken place between our subcommittee and the NRC technical staff over the past few months, I think commencing in probably April.

Thhis has involved a considerable amount of interchange between our organizations and has involved industry experts from a large, broad base of the industry.

The basic problem that we have with Reg Guie 1.97 stems, I think, from the time in which the rewrite of Reg Guide 1.97 was initiated. At the time revision two work began, industry did not have in place the kind of comprehensive emergency plan that we have today.

We did not have the emergency organizations that we find that we have today.

Further, the industry had no efforts and no plans for facilities such as safety parameters display system, the technical support center, the emergency operations facilities, and those other facilities.

Consequently, Reg Guide 1.97 was not selected to support those facilities, even though the intent and purpose of Reg Guide 1.97 is to support a station for the detection, the assessment, the mitigation, and the response to a site incident.

Consequently, we feel the document, as structured, is not necessarily in concert with the industry efforts -- NRC and industry efforts on these emergency facilities.

It is our feeling it starts at the end of the problem -- starts from the end instead of starting from the front end and defining what is necessary and then implementing that.

This disconnect with what we perceive as the requirements of Reg Guide 1.97 and the implementation of emergency facilities has become extremely important to us, the reason being that Reg Guide 1.97 is now being stated as the basis for the minimum parameter set for the emergency facilities.

In essence, we feel this is not in the best interest of establishing the optimum emergency facilities, primarily from the fact that the emergency facilitis must be structured to meet these minimum parameter requirements and not the reverse, which is the logical case.

Further, we are concerned that there are areas that are not covered in Reg Guide 1.97 that are essential to be addressed. We discussed human factors; we discussed the need for an importance of information -- the use of information; where the information is to be provided, how it is to be provided, in what form, and what reduction or computation must be mae in order to simplify the operation so that an operator or someone in the emergency center can understand what is being presented to him.

In our efforts with the NRC, the AIF has embarked on a systematic approach to defining the data requirements and the functonal requirements for the emergency facilities and this systematic approach integrates all of the factors that I have mentioned.

It is our suggestion that we take a phased, systematic approach to implementing these facilities, that we sequentially apply the methodology we have developed, and determine a set of accident parameters that should be monitored in each of the facilities.

Further, we believe this approach will allow us, as an industry, to respond more quickly to needed changes in plant safety. Quite frankly, I think there are improvements and substantial improvements that we can make before June 1983. And I think we ought to get onto our business in doing that.

Accordingly, we would suggest a different alternative for implementing the intent and spirit of Reg Guide 1.97.

Now, the AIF subcommittee has taken as its first step the first matter of importance, the control room. And what we have done is identified a minimum parameter set for the safety parameters display system.

This parameter set has been arrived at through a systematic methodology, and within the space of approximately one month's working time, we hafe made the analysis and received industry concurrence in a subcommittee consisting of a wide and diverse segment of our industry. We think it is a productive approach, and simply the fact that we have been able to achieve that consensus and agreement by looking at the logical development of the parameter list we think is of major importance.

We, as an industry, have been criticized for not getting our act together, for not developing industry consensus, for being too diverse and individualistic on all our approaches.

This has proven to be a very powerful tool in pulling us together.

The next part of our presentation is to be given to you by Dave Cane. Dave is from the Nuclear Safety Analysis Center, and what Dave will present to you is an outline of the methodology that was used in developing the parameter list for the safety parameter display system.

Now, one thing you have to keep in mind: this safety parameter display system is a control room aid to give the operator an idea of what has happened, an overall safety review of the plant, and as a detectiuon device to tell him when something has happened.

Therefore, the specifics of the methodology and the examples are related to that one example. But we think the same generic approach can be aplied to the control room next -- as a next order priority; then to the technical support center; then to the emergency operations facilities and then to the other facilities that are necessary.

Dave.

MR. CANE: Good morning. My name is David Cane and I work for the Nuclear Safety Analysis Center, and I am concerned with plant modfications and improvements, safety related data acquisition and display.

In this context, I would like to discuss briefly the thrust of the recent efforts at NSAC, which have been to develop a structured safety parameter identification, selection, and work with industry to determine what safety parameters are needed for display at varioui s emergency facilities.

In our work we found that without a formalized approach to parameter selection, there is no way to reconcile the differences between the lists drawn up by varous industry groups. Indeed, without a structured rationale, we find we are left with no option except to adopt the largest parameter list supplied to us in order to finesse the issue.

This makes everybody happy, but we think that this is far from being the most desirable alternative. What I would like to do is briefly highlight the approach that we used to come up with and establish the list of parameters for the safety parameter display system which is required by the NRC action plan as is being described by a document that is in the process of being developed at the NRC, NUREG-0696, functional requirements for these systems.

We selected the safety parameter display system because we feel that within NSAC it is the highest priority tool for coping with an accident that has been proposed after Three Mile Island and has the largest leverage in terms of safety benefits.

And we have chosen to focus our attention on it first. The method that we have developed has played a substantial role, we feel, in achieving n industry consensus as to what parameters are needed for pressurized water reactors display system design.

We feel that the same methodology can be readily extended to select and supply a rational basis for the parameters to be used by any safety facility to monitor an accident. And this includes the control rocm.

(Slide)

Procedure for safety parameter selection involves three basic ingredients. First of all, one should identify the set of functon requirements for the display facility and then proceed to specify particular parameter selection criteria, which embrace these funcional requirements, and finally to develop a decision logic that combines the selection criteria together to provide an acceptance test to be applied to each of the parameters.

Funcional requirements for a facility -- for safety display systems facility are concisely stated; the concise rendition is the safety parameter display system should present to the oeratcr a select grouping of key plant indicators to give the oerator a comprehensive overall view of the safety status of the plant.

such a system should serve principally the purposes of detection rather than diagnosis. From this functional specification a list of selection criteria were prepared.

(Slide)

These are shown here; I will go down each one of them in turn.

The first criteria that we use is that the parameter -- proposed parameter could qualify for inclusion on this data set for safety panel display systems if it were a leading indicator in the event tree for a dominant accident sequence.

A dominant accident sequence is that listed for pressurized water reactors in WASH-1400. the methodolcgy used to determine leading indicators is extremely siilar to the one that was used in NUREG-1440.

This work has been done under contract to the Nuclear Safety Analysis Center. But in addition to being a leading indicator, the parameter could qualify if it is a fundamental parameter for selecting the satisfaction of a critical plant safety function; that is, determining whether you have reactivity control, heat removal, so forth, and so on.

But in addition, a plant parameter could qualify if it were a primary indicator for the status of the radioactive barrier.

Now, as I said before, the safety parameter display system is not for the purposes of indicating -- providing diagnostic capability. It is primarily a detection tool; for that reason, detection is indicated here as one of the selection criteria for this facility.

MR. OKRENT: Would you help me and tell me the difference between detection and diagnosis.

MR. CANE: Okay. I have a loss of coolant accident. Do I or do I not have a loss of coolant accident, but where is the break? What system do I have a failure in that is causing this problem?

In the first case we are talking about detection and in the second case, in my view, we are talking about diagnosis.

MR. OKRENT: So you are not trying to look as to where the break is in your system, only that there is a break?

MR. CANE: Yes. And this is the funciion of the safety parameter display system, and it is consistent with the functional requirements being developed, and it is not to supplement -- it is to supplement, not to be used in lieu of the information on the control board where the operator can conceivably go to diagnose the problem.

MR. OKRENT: But if he does not have the informaton on the control board now to diagnose this, he will not get it from your safety parameter display system. There would still have to be some third source, some new source of this information.

MR. CANE: That is exactly the case. And what I am trying to descrige here is the methodology, not so much the lit of parameters that you come up with for any particular facility.

MR. COLEY: As he indicated, that system is the first step, and it is strictly limited to overall safety status of the plant in detection with regard to certain functions.

The next step is the review of the control room, and that diagnosis function must be addressed in the control room.

MR. OKRENT: It is not clear to me whether you are proposing something instead of what the staff has in the Reg. Guide 1.97 or something in addition.

MR. COLEY: Okay. The bottom line of what we are proposing is this. The sum total -- well, the whole objective and intent of Reg. Guide 1.97 is for you to be able to detect, to assist -- to assess and to diagnose and to mitigate the consequences of a site accident. That same information is being used to support the emergency facilities in the plant, the facilities with which you do that.

All right. Our objective is this, that through a logical and systematic approach of defining independently the requirements for each of these facilities, the net or sum total of those requirements will be the intent of Reg. Guide 1.97, so this is the first step in that process. The safety parameter display system. Next would be the control room, the tech support center, emergency operations facility, and so forth.

So, that is the approach that we are suggesting, and this is just really an example of a detection system which is really not intended to be used for diagnosis. Quite obviously, the control room would do that, the control room instruments.

MR. EBERSOLE: In the context that I use for accidents, I include accidents which in themselves are loss of instrumentation, and I need instrumentation to follow loss of instrumentation. What do you do about this?

MR. CANE: Instrumentation, to follow laws of instrumentation.

MR. EBERSOLE: Case in point, a control room fire, DC power failure, or whatever.

MR. CANE: I think there are provisions for redundancy as well as diversity, as well as qualifications, requirements that are applied to key instruments which minimize the possibility that you lose instruments, but if you lose all the instruments, you lose all the instruments.

MR. EBERSOLE: I am thinking about the case here where I am not going to lose all the instruments, but I may lose a large fraction of them and then depend on these instruments to follow the course of an accident. Crystal River is a case in point, or I could define a worse one which Brown's Ferry nearly had, which was loss of control room.

MR. CANE: I feel that the parameters that are displayed here for the safety parameter display system, to borrow words that people use in the industry, are your gold plated instruments. You do what you possibly can to protect integrity.

Now, as we get away from detection and diagnosis and we get into things like system status, perhaps measures that are less stringent, but it has to be considered on a case by case basis.

MR. EBERSOLE: How do you relate this to the present criterion for requiring we be able to shut a plant down from outside the control room?

MR. CANE: That is what you are talking about -- it is a control function which is a very important consideration, but the facility we are talking about is not a facility that is designed to enable the person to control the plant. Rather, it is there to permit him to get an overview status of the plant. I am not going to try to degrade control because that is a Number One issue, but not this issue in my view.

MR. EBERSOLE: You are just giving us visual input to the operator in this discussion.

MR. CANE: In this case, we are making sure that the operator -- it is impossible for the operator to blindly get into -- for instance, the Three Mile Island situation, not seeing the forest for the trees. We think that is a very critical area that needs to be improved.

MR. EBERSOLE: Having got that visual input, he has got to do something with it, and I think I am talking about now, what does he do if he has it?

MR. CANE: Okay. I think that is an important question, but I am not sure that falls within --

MR. EBERSOLE: It has been mentioned before as another --

MR. CANE: I agree with you. I agree with you. But you have to divide big problems into little ones, and this is a little one which is a big one.

MR. OKRENT: First a comment, then a question. It is not completely clear to me that if we have some kind of a well-defined safety parameter display system, that this will prevent the operator from concentrating on something that is bothering him on the panel, but he may still get into the same position.

This may help, but it is no cure. This is just a side comment. I was interested that you distinguished between diagosis and status display as if status display is not important for diagnosis.

MR. CANE: It is necessary but not sufficient.

MR. OKRENT: Well, okay. All right. I will buy that definition.

MR. CANE: Okay. I will get to the matter of operator interaction with control panel at the end, but let me get through the end of this, and then I have a couple of extra comments if there is time relative to that question.

Where we were is talking about the parameter selection criteria, and we are talking about detection. The last three -- direct measurement is the provision that a parameter be directly measured, not a derived variable. It is important to distinguish that I am not proposing you don't do something with the parameters.

If you want, in an advance system you can compute mass inventory. You can do a heat balance, any number of things. But that is not the thrust of our effort. We are strictly looking at parameters that are measured. We feel it i importnat that a parameter, if it is going to go up on this critical display, that it be highly reliable, that it does not fool the operator.

MR. EBERSOLE: Does this mean you will not re-transmit it from a so-called standard instrumentation source? I think Reg. Guide 197 implies you can take a standard source and put an isolation device in it and retransmit from that to this so-called set of post-accident instrumentation. You are saying you do not endorse that. Is that correct?

MR. CANE: We feel that this is the first accident monitoring system that should be developed and implemented in a very timely manner. We do see a need to install it, possibly in multiple locations. It makes sense to isolate it.

MR. EBERSOLE: Thank you.

MR. CANE: So we are talking about reliability. The parameter should be useful for finding out the plant conditions and not be keyed to a particular event or a particular plant condition.

Okay. These considerations dictate that a certain selection logic with an optimal parameter set be developed to reconcile two competing display characteristics. Simply stated, they are, more is better versus design efficiency. The more is better attribute assures that the display system will be responsive to any conceivable accident situation.

(Slide.)

MR. CANE: And what this means, almost in the academic sense, is, you use an unlimited number of parameters. The design efficiency attribute emphasizes the information overload human factors concern and the overriding need for streamlined, finely tuned emergency facility design.

Therefore, efficiency for most of the use of a fairly limited set of parameters. So you have more is better versus efficiency. You have to reconcile this.

MR. OKRENT: These words are tricky, and I could easily use the argument against instrument -- you know, more by saying we should not have additional displays of pressurizer level, for example, or level in a BWR vessel or something. If you have it in one part of the control room, you certainly would not want to put it in another, because you are just complicating life, and you are having more displays, and this is likely to confuse the operator. Therefore, let's only put it in one place.

On the other hand, you go into the control room and you find the operator may be actuating with something he wants to control level, whether it is pressurizer or vessel, and in fact it is at the opposite end of the room from where the level instrumentation is, and so he cannot even see it, and he may have to go across or whatever, unless you put it convenient to the actuating device.

So, I have a little bit of a problem with this generalized discussion. It leads us into a kind of unproductive area. I just have to put it that way.

MR. SHEWMON: Some of us think there is finiteness to people's minds and what they can agree on and find what is useful.

Shall we let him get on with it?

MR. CANE: This is the safety parameter facility.

It is one unit. If you take an operator and say you are going to display time histories, you put 20 time histories on there, it looks like spaghetti. You cannot make any sense out of it. If you divide the thing up to make a little more sense out of it, you say, I have to be looking at specific functions like reactivity or heat removal. Even within heat removal, if you just take that function, so the guy gets an overall view of how much heat he is removing, or is he removing heat, you don't want to put 20 parameters up there.

MR. KERR: Mr. Cane, do you think the NRC staff would disagree with your objective that one should look for an optimal set?

(General laughter.)

MR. CANE: No.

MR. KERR: I don't, either. I don't think we need to belabor that, because it seems to me it is short of a general objective.

MR. CANE: Okay.

MR. KERR: I agree. I think it is important.

MR. CANE: In any event, considerations here dictate that you develop a decision structure that incorporates basically and as well as or components.

(Slide.)

MR. CANE: The flow diagram I am showing here fully describes the procedure that we used to determine the safety parameter display system safety set for PWR's, and in the process, we input the union of six candidate parameter lists which includes the parameter list that was originally prepared under contract by the Technology for Energy Corporation on the left, and extends clear across to the interim list.

The same procedure could be adapted by applying a specific list -- the book list of parameters for any particular plant being put through the same process. This is what we did to achieve robustness. In concert with the idea that more is better, it was decided that a candidate could qualify for a leading indicator for events in a dominant accident sequence, or as a primary indicator accomplishment of a critical plant safety function, or indicates the status of a radioactive barrier, fuel cladding, pressure boundary, containment building, and so what we have at the top is a parallel path where a candidate can qualify by meeting either of those criteria.

MR. KERR: It seems to me that it is worth noting that Professor Okrent's problem is somewhat ameliorated by the robustness of the system.

MR. CANE: The attempt was not to tie the parameter selection to dominant accident sequences per se or safety functions per se, but to include all of the possibilities. To succeed in becoming a safety display system parameter, a variable must provide useful information under diverse plant operational conditions, and this constitutes the end component.

So, the flow chart really expresses the algebraic formula, binary algebra for the parameter selection. I am sorry, we went through this yesterday. Candidate parameters can be analyzed through the use of a selection matrix showing a selection criteria satisfied by each parameter. This is exactly what we did. We broke up the candidate parameter list and a union of all those inputs developed a selection matrix, simply put an "x" in each column where a parameter was believed to satisfy each of the criteria.

(Slide.)

MR. CANE: For example, the parameter at the top is the hot leg temperature. The "x" is there, and if you go through the decision line, it qualifies, and there is an asterisk there indicating it was accepted as a safety panel parameter.

Under the terms of -- if we continue sequentially to apply this acceptance formula down the list, the PWR safety parameter display system data set is a direct end product. The data set is the industry consensus on data requirements for safety parameter display systems. The end product is the listing which is a repeat of the asterisk items on the selection matrix.

(Slide.)

MR. CANE: And this slide has been broken into functional requirements, because we presently believe that the grouping in a safety panel design should be by function as opposed to, for example, radioactive barrier.

MR. EBERSOLE: Would you put your previous slide up just for a moment, please?

(Slide.)

MR. EBERSOLE: Under Vessel Level it says, "State of the art precludes reliable unambiguous level measurement at this time." What have we been doing with BWR's all these years that we cannot do with PWR's?

MR. CANE: It measures the level above the core. I do not think that it --

MR. EBERSOLE: That has been a standing problem with the boilers, too.

MR. CANE: That is a problem.

MR. EBERSOLE: Are you saying we have ambiguous information at present with the boilers? I might agree with you to some extent.

(General laughter.)

MR. CANE: Okay. We could get into a side discussion on that. It is a complex one. Be r in mind that you may disagree. This turned out to be the same problem yesterday, I am trying to demonstrate an approach. I think the approach is missing in 1.97, as far as a formalized presentation on how they got there. I think there is a way to do it. We can talk about the criteria. We can talk about the judgments. But what I am really trying to describe is the method.

MR. EBERSOLE: Okay. Thank you.

MR. CANE: In conclusion --

MR. LEWIS: I would like the record to show that I did not bring up the subject of void indicators at this point.

(General laughter.)

MR. CANE: In conclusion, I tried to show there was a logical progression for developing data system requirements that begins with functional specification and maks use of formal selection criteria. The structured approach maximizes the opportunity for arriving at the optimal data set while minimizing subjective arguments as to relative safety significance, required numbers and kinds of parameters, and so forth. Step by step procedure which would lead to a consistent, fully justified, and in short a much better guideline for accident monitoring instrumentation.

I want to point out this is not an academic exercise that we did because we simply thought it was fun to select parameters, but an attempt to move along -- I think this is in conjunction with the industry effort -- move along for timely implementation of the safety facility that we feel is very important, and that is, of course, the safety parameter display system. We tried to use dominant accident sequences. That is, we have used event trees that have been pre-analyzed in a systematic fashion borrowed from other work which is WASH 1400 incorporated it. However, we did not rely exclusively on pre-analyzed events.

Now, in the course of development of the safety parameter display system, we have not stopped with parameter selection, although that is a very important first start after you define what the functional requirements are. We had plans, and are working with the Electric Power Research Institute to develop a prototype system and to demonstrate it on a reactor simulator.

The simulator will be the Zion simulator This is in concert with the disturbance analysis program. There is an attempt to provide a continuous path between this system, which is obviously detection oriented, with advanced display systems to provide for disturbance analysis, and this evaluation will be done with real operators. We will try to iterate and learn from what information is obtained in those tests. That is all.

MR. MARK: Thank you.

MR. OKRENT: One or two questions.

MR. MARK: Make it very short.

MR. OKRENT: Suppose we had the check valve accident.

MR. CANE: It is a large break LOCA, small break --

MR. OKRENT: No, no, no. With your minimum list, would you have any -- where would the operator first learn from this list that that was the event. He would know there was a loss of coolant if there was a large break, but where would he learn from this that it was that type of loss of coolant?

MR. COLEY: If you look at the function, we were trying to accomplish with this, if the operator knew he had a loss of coolant accident, we consider our safety parameter display system successful. Remember, that was the intent of the system to start with.

(Slide.)

MR. COLEY: Where does he go next to find out? To the control room instrumentation. And we think the same kind of review needs to be performed on the control room, so that he learns here first that he has had an accident or this is one source he can go with if he is lost. Then he can go to control room instrumentation, which has been similarly analyzed to give him the diagnostic capability.

Again, this particular safety parameter display system, I guess there is a draft NUREG-0696 which is to be issued some time in the near future, I think. The safety parameter display system, I guess, supersedes the safety display panel and the safety state vector, which -- I believe that was the original terminology that was applied.

The whole purpose of our presentation today -- and I want to make clear what we would like to see out of this, because quite obviously we are here and we want something. Our objective is not to knock Reg. Guide 1.97 and the approach taken -- I think it is logical -- had the intent that the NRC has in that document. We as industry have the same intent.

We feel that taking the specific course of action we recommend would give both the NRC and industry what all of us want and need faster. We would suggest that instead of implementing 1.97 as it currently stands, that NRC and the industry continue the dialogue we have already started on the safety facilities for the stations, that we apply a systematic evaluation to those facilities, factor in human factors, the need for the use for information, where the information ought to go, what computations ought to be included to simplify that information so that we fully define what we are doing where.

We would suggest that we start with this facility, move next to the control room, then the tech support center, and so forth. It is our feeling as industry that we can implement on a faster schedule the total changes that are required in improving the ability of us utilities to respond to accidents by this kind of approach.

As an example, if the polish-up work is done on safety parameter display systems, I find it hard to believe that industry or most of industry certainly could not respond and have this facility in place well ahead of June, 1983. That is my personal opinion. We would suggest that implementing in this phased manner, we can make those safety improvements that need to be made much faster.

Now, our committee is continuing on its effort. As we indicated yesterday, we have ballparked a figure for completing the same kind of analysis on the control room instrumentation. We feel reasonably this could be done in about two months, and I think that would put us well on the road to making those plant improvements that we really need to make.

Thank you for your time.

MR. MARK: Are you in close collaboration with the ANS group. That I think must have some common interests.

MR. COLEY: The composition of the AIF subcommittee includes ANS 4.5 representation. The composition of that committee includes ANS 4.5. The IEEE control room standard, P566 or 567, I forget which. It includes all owners' groups, all vendors'-owners' groups, and pretty diverse elements of the architect engineering field and operating utilities.

MR. MARK: Bill?

MR. KERR: Mr. Coley, there is some logic, it seems to me, in the approach that you suggest, which is solving one problem at a time of several larger problems. However, in the installation of censors, instrument lines, instrument displays, and so on, it seems to me that the mere task of going into an existing system and pulling out equipment and replacing equipment, if done four or five times, as one progresses towards the goal, could have some disadvantage compared to in a sense doing it all at one time.

Could you comment on -- I am sure you have thought about this.

MR. COLEY: As I say, I have not seen the draft of 0696 that has been issued, but in that draft, the NRC recommends a sort of common data base or common approach to providing that information to all the facilities, and I think that -- I think for the kind of approach that is being suggested and discussed, I do not think we are talking about ripping out instruments and re-installing them and running wires to other locations.

I think we are talking about a technical approach in which you make the assumption first of all that most of the facilities should probably have access to the data anyway, and you define a technical approach that will allow you to do that.

Now, one thing, on installing the sensors, and this is not a trivial issue, quite frankly, there are not a lot of sensors that are qualified to meet the requirements that we need. We quite frankly cannot buy the sensors. The idea of whether or not a parameter is in a control room is sort of a side issue, because it has to be in the control room at the range that you ask for, and it must be qualified for what you ask -- for what you would like to have.

Just two brief examples. Containment pressure. Reg. Guide 1.97 specifies a range for containment pressure several times the burst pressure of the containment. All right. We have a containment pressure sensor right now. That sensor is used by the protective system, and it is used to isolate the containment and to line off the containment building spray pumps. Increasing the range of that sensor will decrease the resolution such that that sensor cannot perform the protective function, so even though the parameter of containment pressure is in the control room, you must add an additional sensor. Otherwise, you would compromise the safety function.

Reactor coolant flow is another one. Reactor coolant flow is in the control room, and it is a variable we read. That is true. But the guide specifies that you would like to be able to read plus or minus 12 percent flow as a means of detecting natural circulation, and if you take a look at some plant designs you are attempting to read that 12 percent flow in a 36-inch pipe with elbow taps, and I submit that does not give you a measurable signal.

It is simply a very difficult thing to do. So the parameters is in the control room, granted, but it is not an easy issue getting plus or minus 12 percent flow. The existing instrumentation just will not do it.

MR. MARK: Thank you.

Mr. Coley?

MR. MINNERS: Mr. Coley, I think you misspoke. You said we were requiring containment pressure which is four times -- many times the burst pressure. It is three to four times the design pressure.

MR. COLEY: Okay. In any event, the range is not adequate to still use that large range sensor for the protective function.

MR. MINNERS: I agree.

MR. MARK: Paul?

MR. SHEWMON: Some time before we quit this, I would like somebody to discuss what is going forward as a result of the TMI 2 action plan, while 1.97 is being refined. I don't know whether -- there has been an allusion to what sorts of things are coming out of the action plan here. Maybe the staff could do that best.

MR. MINNERS: I tried to answer that question before. Maybe you were not in the room. I can repeat it if the Committee would like.

MR. SHEWMON: Maybe one of the people who heard it before can paraphrase it for me.

MR. KERR: You would not mind having it repeated, would you?

MR. MINNERS: That is well within my ability, to put it in simple language. The staff in the action plan has outlined a staged action for this whole subject of emergency response.

MR. KERR: Would you mind saying which part of the staff that is, Mr. Minners?

MR. MINNERS: It is various parts of the staff, depending on what action has been taken. I will try to bring that into my discussion. Realizing that Reg. Guide 1.97 was well on its way, we had given that first priority for issuance, so that is being done in this forum.

As Mr. Coley has indicated, the need to provide an integrated set of criteria for these emergency response facilities became apparent, and in cooperation with the industry we are starting to develop such a thing which is now in a draft NUREG document, 0696, which tries to give the functional criteria for the technical support center and the emergency operations facility, the safety parameter display which was just discussed, and also the nuclear data link. They are very closely related, because they are all going to take essentially the same set of data that is coming out of the plant. You want to do that as an integrated system. So, those things are under way.

Now, also proposed is probably a better way of talking about it, and that effort is being done by a task force which has people from inspection and enforcement for nuclear data link, from NRR for the emergency planning operations and division of human factors is on the task force, and we also are in contact with the people in research who were administering the contract for the nuclear data link.

MR. SHEWMON: Let me deflect you into shorter range items. Was there any new instrumentation required as a result of the action plan? For example, level indicator?

MR. MINNERS: Yes, sir. As part of the lessons learned, there is a requirement to put on additional accident monitoring instrumentation, and you have to have it all on according to the action plan by January 1, 1981. It includes vessel instrumentation, radiation instrumentation, extra instrumentation which the lesssons learned task force thought was the real hard nut of what extra stuff ought to be put on right now.

MR. OKRENT: So there is additional instrumentation which is going in if not on a crash basis at least an accelerated basis, while we are discussing these other things that would be desirable.

MR. MINNERS: Yes, sir, and that has been integrated with Reg. Guide 1.97.

MR. OKRENT: The presentation the staff gave, they are urging we do something while we were dead in the water or something.

MR. MINNERS: We are moving forward on that. The longer range thing is --

MR. SHEWMON: You have gone far enough for me. Thank you.

MR. CARBON: Mr. Coley indicated he thought they could move faster and accomplish more safety quicker going at it with their approach than with yours. Would you comment on whether you tend to agree with that?

MR. MINNERS: I don't know how to answer that. We took what was available. Reg. Guide 1.97 was available. What was available we took. We did not try to re-invent the wheel.

MR. CARBON: It is not a matter of taking what is available. I think he said it is a matter of a different approach.

MR. HINTZE: The approach they are taking will not produce a list until they get their complete analysis done. We have the list now. We have looked at these various applications that came along after we started on the list. We reviewed what is going to be required by the nuclear display -- parameter display. We reviewed what the technical support center is going to require and matched it against the list which we have, and we find that we are covering all of the ones they are coming up with.

MR. CARBON: He is not talking about a list. He is talking about accomplishing an appreciable increase in safety at an earlier date.

MR. BENAROYA: Let me say that in our implementation, there is -- there is a point of Januar, 1 -- I mean, June 1, 1983, completing the implementation. There is nothing that prohibits it from starting earlier.

MR. MINNERS: In fact, there are some licensees who have already started designing their systems, computer-based systems, and they will have a real problem if we do not start putting some requirements on it, because they may have a system that does not fit the requirements, or vice versa, whichever way things work out.

MR. MARK: I think we must get on with the GE people who were kind enough to hold back with their presentation.

MR. D'AREDENNE: My name is Walter D'Aredenne of General Electric. I am filling in this morning for Dave Waters who gave this portion of the presentation yesterday on behalf on the BWR owners group. Mr. Waters is with the Carolina Rower and Light Company, and as I said, he was speaking on behalf of the owners group yesterday.

We have had a history of commenting with the staff on this revision of Reg Guide 1.97. Both the BWR owners group and also GE, and we have been having interactions with the staff since last July when they initiated this revision.

(Slide.)

We still have a lot of comments, and our purpose today is to bring your attention to the fact that we still have numerous comments. We have prepared written comments which we have submitted to you, and I think have been distributed to you. I think we have, in addition -- I think we have 14 comments all told.

We also intend to discuss the technical aspects of core exit temperature measurement that is required in the current draft. We are going to give you a technical presentation and not an emotional one or one on the basis that it costs too much.

Technically we do not feel that the core exit temperature measurement is warranted or justified. Our concerns with the contract -- our biggest concerns are the fact that we do not feel that the current draft is at a stage where it is ready for issuance. And this has been brought out, I think, by the previous commenters, ANS and the AIF.

The fact that functional criteria for the parameter list have not been established and that this leads to a problem with the practicality and the implementation of the current Reg Guide, we brought this out in a letter that is referenced in our comments.

Roger Mattson said that we felt the Reg Guide should be deferred until these functional criteria could be developed, and we feel that the approaches discussed by AIF and ANS and also that are addressed in the NUREG/CR-1440 provide -- identify appropriate approaches.

We are not going to discuss our concerns with the criteria, because we feel that that has already been adequately addressed. Our other major concern is with the core exit temperature, as I mentioned before. And Craig Sawyer who is with us will go into the details of our technical concerns about that.

As I mentioned before, we do have additional comments which are in our written comments.

With that, let me turn it over to Craig, and then I will go into my last remaining slides afterwards.

MR. SAWYER: I am Craig Sawyer. I work with General Electric Company. As Walter mentioned, we have a particular problem coming to grips with the need for core exit temperature measurements in BWRs, so let me start by addressing the reasons that the current version of the Reg Guide cite for core exit measurements in BWRs.

Specifically, they are to indicate the potential for or actual fuel clad breach, and by means of a footnote to measure the extent and trend of core damage down to the five percent -- five to ten percent core blockage level assuming no ECCS functioning.

(Slide.)

That is part of the criteria.

(Slide.)

The paper we prepared on this subject for your detailed reading approaches the problem from two points of view. One is if we had temperature measurement, what do we do [illegible] it, and the other one is how could we implement temperature measurements given that we fit the requirements as mandated, and both aspects of the problem are addressed in a paper in some detail.

For the time I have here let me merely state we have a number of current variables in BWRs which can indicate either the potential for or actual cladding breach. By way of introduction, as you know, cladding breach occurs with a combination of high cladding temperature and high stress, or by means of excessive oxidation; and it is not clear that be measuring core exit temperature that you are getting a one-to-one correspondence between -- at least in a quantitative sense between cladding breach or amount of cladding breach and the temperature that you are indicating.

Quantitative variables that do exist already or are planned in other parts of Reg Guide 1.97 include high hydrogen levels, high steam line radiation monitors, fission product measurements in the reactor coolant containment air and suppression pool water, off gas radiation levels, low water level, which I will go into in some detail in a moment, complete loss of makeup. And these currently measured variables provide diversity, unambiguous indication that you have of a cladding breach and are already qualified and tested.

(Slide.)

Let me just quickly flip this up. I will go into the subparagraphs to these in the scenario that I want to go through with you as an example.

Basically we are talking about three periods of time during an event that we looked at in assessing what kind of information core exit temperatures could give you, and these are prior to core uncovering, core heatup, and then after recovery.

Without going into further detail on that chart which you have also in the handouts, we have looked at a large number of scenarios after TMI and reported on them to the staff covering degradations that accrue from transients plus multiple failures to provide adequate makeup water, small breaks, stuck open relief valves, and even large breaks.

And although in detail the scenarios might be different, in a general framework they always proceed on a BWR as follows. You have an event which threatens to uncover the core. In this particular case the event we chose for the example was a stuck open relief valve for a BWR without makeup systems coming on when demanded automatically because of additional failure.

Other events can be postulated. The time scale may be shifted, but the idea is always the same. What I have shown during this phase of the event is the period of time in which the core is covered. During this time the reactor is operating in a saturated mode. I show from the top figure water level, which is indicated to the left, referenced to an arbitrary scale, although I have shown where on that scale the top of the fuel and the bottom of the fuel are, and on the right temperature, and we have chosen for this example to talk about fuel temperature for an average bundle and a thermocouple located in the bypass zone between fuel channels directly adjacent to where the fuel temperature is being measured.

Indicated on this chart are the kinds of information that the operator will have during the period of time that the core uncovery is being threatened. There is a low level scram; at a lower level there is a signal given to turn high pressure ECCS on. Beyond that point the operator might make a decision because he can confirm that high pressure ECCS or other high pressure events -- the feedwater system is not available to inject water and take action to depressurize the reactor. But in no case later per the guides we have written for the staff's review for BWR emergency operation -- in no case is it indicated on the chart.

Low pressure ECCS gets a signal to start also as indicated, and the operator must depressurize by the guidelines we have submitted in order to make sure that the low pressure ECCS can inject, because the vessel will be depressurized before the core is uncovered.

MR. EBERSOLE: Before you leave that, there is one aspect of your post-accident cooling that is not shown on that, and I again refer back to the recent flap about the hold-down bolts on the injection pumps, to bring it into focus.

One mode of your cooling says that you are happy with two-thirds core height, and you build the injectors at that height to maintain that level in case you have a major pipe failure outside the downcomer on the outside. You do not show the condition of cooling which you claim in your SARs for the case where you are at two-thirds core height and you in fact have a water level below the top of the fuel.

MR. SAWYER: Let me postpone that until I get to the third phase. I will be happy to address that then.

MR. EBERSOLE: All right.

MR. SAWYER: In this phase the water level has not gotten down to the two-thirds level.

MR. EBERSOLE: All right.

(Slide.)

MR. SAWYER: The second phase is what we call core bundle?

MR. SAWYER: For this example that is right. We could be, in this particular case, once the water level has penetrated into the midplane of the reactor, if you had a number of thermocouples located at all of the LPRM strings, for example, those thermocouples would take on a shape qualitatively like the power shape of the reactor. We are just showing a typical example here.

Also shown on here are the time at which we would predict that there would be detectable hydrogen production from reaction of the cladding, significant hydrogen production in terms of containment pressurization by means of additional hydrogen, and the time at which we expect due to delays the getting of readings above background fission products in the air space relative to this scenario.

MR. SHEWMON: Would you comment on which of those -- do you have instrumentation now in plants for all three of those to detect hydrogen? Is it only significant or wet well air space fission products?

MR. SAWYER: We have monitors right now that will detect hydrogen and somewhere -- in terms of accuracy just about where we say detectable hydrogen.

MR. SHEWMON: That is in place on plants and has been tested? It is not in a laboratory some place that you have in mind to install?

MR. SAWYER: I am not an expert on hydrogen monitoring equipment.

MR. SHEWMON: I am not an expert on BWR plants either, but I am trying to find out whether this is hypothetical or whether this is indeed what is in the field now.

MR. SAWYER: This is intended to be a generic design. I believe that most, if not all the plants, have such instrumentation.

MR. SHEWMON: Which would detect hydrogen.

MR. SAWYER: Yes.

MR. OKRENT: I don't think Browns Ferry has an on-line hydrogen detection device, does it, or some of the old BWRs?

MR. SAWYER: There may be some of the older plants that do not have these devices.

MR. EBERSOLE: You show temperatures at the core mid-plane. What would the temperatures be along that same time curve, say in the upper 12 inches?

MR. SAWYER: As I said, the moment that the water level gets below the top of the fuel, you will begin to produce some superheat, and so thermocouples located at the top of the reactor will see such superheat. But the trajectory that a thermocouple located, let's say, right at the top zone would see, it would look something like this. That is intended to be qualitative.

I don't know if this is exactly the trajectory it would take, but it would start sooner, but it would flatten out because of a lower power density.

end tp 9

MR. SHEWMON: How do you get hydrogen generated copiously with cladding at 1000 degrees F.?

MR. SAWYER: I think the hydrogen is generated -- that arrow should be equivalent to around 1500 degrees Fahrenheit.

MR. SHEWMON: Okay.

MR. SAWYER: I achieve about 1500 degrees about right here.

MR. SHEWMON: Good.

MR. SAWYER: Okay? Now, for the sake of example, we have assumed that you don't go to complete core melt. I don't think it is necessary to have core thermocouples to watch the progress of a core melt. It would be assumed that if they were to have utility, they would be used to help the operator recover and try to prevent such an occurrence from happening.

So we have continued the scenario on, assuming a delayed makeup injection till late in the event. For this purpose, we assumed that one of the low pressure injection systems is recovered and turned on. Until this time, we have assumed that there is no makeup systems providing water whatsoever. That is either the normal makeup from the feedwater system or from the RCIC or any of the ECCS systems until now.

Assuming this delay, the bypass thermocouples in the core overall will quickly be quenched. If you assume that the scenario went, depending on the course and which bundle it is and how long you waited to provide an uncoolable geometry, a blocked channel, you would have an upward course. A channel which had still cooling area available to pass water through the channel would also quench. We have done analyses which we have provided in the writeup which say that because of the multiple cooling paths that the BWR thermohydraulic design has for getting natural circulation and maintaining natural circulation, you actually have to have more than 99 percent of the cross-sectional area of the channel blocked in order to end up having the channel continue its upward course.

If as little as one percent of the original area is available, the channel will be retained in a safe mode after the water level is restored. It is primarily because of the interaction of the ECCS equipment and what effect it has on core thermocouples, and the fact that the core thermocouples don't necessarily follow the same course as the fuel will that we have a hard time coming to grips with why it is that this would provide useful information to the operator in following the course of an accident.

In conclusions, as I pointed out, we have identified a number of variables which indicate the potential for cladding breach and are actually better suited to provide a quantitative measure rather than a qualitative measure that you would get by knowing that the temperatures are out of line and where they should be.

One point that I didn't mention, although it is discussed in the text, is that we have done some analyses of what would happen following a recovery if a channel were completely blocked, and would that channel then cause a continued propagation of core damage into other channels, and have concluded that the heat fluxes are so low once you have recovered that even if you can't get cooling to an individual channel, that the channel heat-up will eventually, inside the channel, the fuel heat-up and slumping will eventually cause a breach of that channel and provide that extra path necessary for cooling without a steam explosion.

This has been documented also in our licensing topical report on flow blockage which we provided to the staff three years ago.

Finally, the point we want to make is we don't believe the temperature measurement is a reliable indicator of the extent and trend of core damage and may, in fact, provide a confusion factor to the operator.

MR. MINNERS: One more clarification.

MR. SAWYER: Yes.

MR. MINNERS: The water level that you have plotted is calculated (inaudible) water level?

MR. SAWYER: It is actually a density for the instruments that we use. So if there is two-phase water, it basically reads density times height. So if you have two-phase water, the top of the frothing mixture will be higher than that indicated here. I guess the answer is yes.

MR. OKRENT: Which of the radiation instruments, in your opinion, would give unambiguous and interpretable information if you have your isolation valve closed, as you might very well?

MR. SAWYER: Right. The reason we have instruments on there that would be lost under some scenarios -- I'll go back to this chart -- in an isolation event, that and that will not be there. The main reason those are there is to protect you while a sudden change change, like blockage of an inlet channel, during normal operation of a plant. So those will be gone if you have an event. In the one which we have postulated, for example, in which this reads the low water level, which provides an automatic signal to isolate the plant, those won't be there.

So the fission products you will be measuring will be those due to the area radiation monitors that we have located around the containment. Many plants have a gross gamma monitor qualified to high radiation levels to measure post-accident postulated gamma levels, too.

MR. OKRENT: You know, one might think of another scenario where the primary system were relatively bottled up, losing its inventory perhaps only intermittently or something and the level dropping.

MR. SAWYER: That is exactly what happens in this case.

MR. OKRENT: All right.

MR. SAWYER: The safety relief valves are piped to the suppression pool, and if there is no break, that is exactly the course that the fission products will have to take.

MR. OKRENT: Right. It is not clear to me that you will get too much from radiation level that tells you a lot about what is happening in the core. That is all I am getting at. And the hydrogen part, again, is also -- you really will be relying, I think, on water level. Of all the parameters you have there, I suspect that the one that --

MR. SAWYER: Well, the operator, to protect the reactor and take the necessary actions to recover, I agree, will be relying primarily on water level. These other parameters are shown as a way of saying that if you want to know how much -- tell me 5 percent, 10 percent of the core has been consumed or of the cladding -- that those other parameters are useful more in a post-accident phase to assess that.

MR. OKRENT: Yes. One could look at this figure and get the feeling that, gee, there are all kinds of things that will give you maybe what I will call on-line information about how hot the core is and so forth. What I am suggesting is that in fact these others are not necessarily equivalent to temperature, water level if you know it; and if you don't have some local blockage or change in level for whatever unspecified reason -- within the core, in other words, a non-uniform level, that should tell the operator quite a bit.

But I think that --

MR. SAWYER: Qualitatively fission, by having area -- if you have a release or a cladding breach, the noble gasses which will be released will not dissolve in the suppression pool and they will give you an instant reading in the monitoring equipment out there, at least qualitatively, that, hey, I have got --

MR. OKRENT: Right now there will be an alarm that says high radiation level in suppression pool --

MR. SAWYER: Correct.

MR. OKRENT: And after that you won't really be able to tell whether it is two fuel rods or 1000. Maybe he can go over somewhere -- and then the shift technical adviser can do a calculation.

MR. SAWYER: I am not sure that that information is necessary, his knowing that he is at 2 percent 10 percent, for him to take the appropriate action.

MR. OKRENT: I'm not arguing for or against any specific temperature measurement. Don't misunderstand me. I know you have difficulties, furthermore, in making certain kinds of measurements. I am just looking, really, at how many of these are equivalent or provide really early information.

MR. SAWYER: In fact, you are absolutely right. The operator guidelines for emergencies which we have written and the staff is reviewing right now tell our operators to base their decisions for either backing up the automatic functions or depressurizing and attempting to get other equipment to pump water in primarily on a combination of water level and reactor pressure, what to do.

MR. EBERSOLE: When you are talking about core exit thermocouples, how many are you thinking about?

MR. SAWYER: In this particular example, I don't think it matters whether you have one or fifty.

MR. EBERSOLE: Well, it matters to cost.

MR. SAWYER: Yes, it will. I think we and the staff have always agreed that during a core heatup, having a temperature measurement will tell you something because there is super-heated steam around. But we have always maintained that that temperature measurement, you could as well measure the temperature in the steam line or in the dome of the reactor and get the same information.

MR. EBERSOLE: In the light of our previous discussion on PWRs, do you consider your level measuring instrumentation to be ambiguous in its output?

MR. SAWYER: Let's talk about which product line. I think the earliest, the two BWRs we have that are now operating prior to installation of jet pumps did not have level instrumentation which goes all the way to the bottom of the pool. One of them at this time has undertaken to redesign and provide that by using a lower tap which is available on the vessel to provide that additional range.

I am not sure of what the other BWR is doing, but I presume that they will be following suit.

MR. EBERSOLE: So there are just two, then, that are --

MR. SAWYER: That did not have this range, that is right.

MR. OKRENT: Even with the newer ones, most of your instrumentation really is level instrumentation reading above the top of the fuel. Is it about one, or --

MR. SAWYER: Two.

MR. OKRENT: Two that go to the bottom of the fuel?

MR. SAWYER: Yes. There are two that go to the bottom of the fuel. There is a reason for picking the ranges we do, because the wider the range you attempt to expand with a given instrument, you have an accuracy problem.

MR. OKRENT: I understand what you need for control, but the bulk of them, in fact, don't give you the information that --

MR. SAWYER: However, the operator should be doing everything he could to keep the water level from even getting down into the core.

MR. OKRENT: And those that are full range -- by that I mean they go to the bottom of the core -- are they calibrated for operating temperature or cold or --

MR. SAWYER: No. The original intended purpose prior to us thinking about this problem as a result of TMI was that those were to be used when the reactor is shut down.

MR. OKRENT: So they could give him misleading information now.

MR. SAWYER: They would be inaccurate by a couple of feet, out of 12 feet total range. Now, the instrumentation which is being provided for the BWR-2 that wants to extend the range, they have also asked for microprocessing equipment to perform this compensation automatically for the operator. There is no reason that that can't be done. It is just that we don't see a need for it.

MR. MINNERS: Does pumps on and off change that reading? The one that measures the bottom of the core. Do you get a different reading when the pumps are on and when the pumps are off?

MR. SAWYER: No. Now, it is a question of what you mean. With the pumps off, the water level in the core zone is the same as the water level outside in the downcomer region. When the pumps are on, the effect of that would be to have the water level in the core zone high relative to water in the downcomer. In fact, that instrument will be reading the water level in the core zone, which is where you want it to read.

People have had conceptual difficulties thinking about that before, and they have thought, well, gee, you are measuring inaccurately. But we are not trying to measure downcomer water level; we are trying to measure core water level. So it just turns out that when the pump is running, it is, in fact, reading core level.

MR. SHEWMON: Have you finished?

MR. SAWYER: I'm finished.

MR. SHEWMON: I have one question, then. This is a topic which has been around for some time. In this discussion of public comments for Revision 2, the staff didn't have time to provide an index. Can they tell me which one of the pages I can find their response to this question on? I'm sure somebody commented that it would be nice to get rid of these thermocouples.

MR. KERR: If you look at the cover letter, which you may not have, to Reg Guide 1.97 --

MR. SHEWMON: You're right.

MR. KERR: In it the author says that that is still an open question and that they will tell us as soon as they have made a decision. The letter is dated July 7, and I don't know whether the decision has yet been reached or not.

MR. SHEWMON: So if I look at this, I won't find any discussion of the thermocouple in it. Thank you.

MR. KERR: Does the staff have a schedule for reaching a decision on that question?

MR. HINTZE: The answer is we intended to have this resolved before this meeting. We met with GE on the 10th or 11th of July, which was not completely adequate in terms of information to where a decision could be made. This is the additional information which GE came up with as a result of that meeting. We haven't evaluated it yet.

MR. CARBON: But as the Reg Guide is written right now, it requires the use of BWR core exit thermocouples.

MR. HINTZE: We had no alternative but to take a conservative course, right, and we left it in.

MR. WATT: Jim Watt. I might mention that the requirement was set as an objective rather than as a specific requiremen .

MR. SHEWMON: Indeed, the general objective was that you be able to indicate the probability of rupture of the cladding on the fuel.

MR. D'ARDENNE: We also did an evaluation of the cost of the thermocouples, both in dollars and in dose. The basis for the evaluation we get was for putting the thermocouples in the ends of the LPRM tubes that are inside the core. These are tubes that go up in between the bundles and are in the bypass region between the bundles and between the channels, and there are about 40 of those. So we are talking about 40 thermocouples, and they are, I believe, about 12 inches below the top of the core?

The dollars came out to $400,000 per plant for plants under construction, and $600,000 per plant for plants that are operating. The total of that cost was $28 million for all of the plants. This included cost of the equipment, the engineering, the field engineering, and the installation work. But it did not include any developmental cost.

The dose -- for the maintenance this would involve all plants whether they are new plants or plants already operating. We estimated that it would take 2 to 15, and the number here of 8 is an average of that 2 to 15 range. This increased maintenance is for the increased time that it would take you to change LPR strings, the increased time that it would take you to service control rod drives, and then also just the thermocouples themselves.

For all of the plants this would amount to a total of 18,500 man-rem for the life of all of the plants that are now operating or under construction. You would have an additional dose effect from the installation in operating plants, and we estimated that that would be about 100 man rem per plant on the average, for a total of 2500 man rem or a total dose impact of 21,000 man rem.

Now, this cost is high and we feel it is unjustified for a parameter which we feel has very marginal benefit, at best. However, even if the cost was zero, we still don't think that thermocouples would be a justified parameter. It would confuse the operator in some situations, and in general they are not useful and we can't justify their installation.

Another problem exists, and that would be the fact that if they are required to be installed, the fact that we cannot design a thermocouple system that would measure what the intended purpose of the thermocouples would be, so we would be stuck with trying to fulfill a promise that we couldn't keep.

MR. OKRENT: Would you leave that on a minute? I believe you said there were of the order of 40 LPRM assemblies per plant.

MR. D'ARDENNE: Correct.

MR. OKRENT: Now, per thermocouple can I divide the 28 million and say if I were, for example, to use only four per plant, then I would divide -- instead of 40 that would be 2.8 million, or not? I am trying to see if it is linear. And also, is the man rem linear with the number of thermocouples roughly?

MR. SAWYER: It is approximately linear.

MR. OKRENT: Approximately linear. Okay.

Let me mention one possible scenario where it might be handy to have at least some thermocouples in the core. If you got to a situation where over some period of minutes you didn't have water in the core and it started heating up, the temperatures you measured elsewhere, if you could measure them elsewhere, might not tell you as much about what the temperature in the core region is, as some thermocouples located in these LPRM-2s. I have now gone through a differ nt scenario than the one you sketched out, but that might be interesting information.

I agree that for the reasons you showed, there could be reasons where the operator looks at these and says, gee, everything is real cool, because there is some water coming down between subassemblies, and he would have to be taught and he would have to understand what they measure and what they don't measure.

But there may be some scenarios where this gives you one more thing that --

MR. KERR: We discussed this sort of thing yesterday, and at least I couldn't -- there may be something. But my question is what would the operator do differently if he knew that a piece of the core had gotten very hot. There is an obvious answer to me as to what he would do differently.

In terms of eventual recovery or what one might do later on in the accident, perhaps having the information that pieces of the core had been very hot would be helpful, but -- I don't mean that I answered the question exhaustively. There may be something obvious that he would do.

MR. OKRENT: I haven't tried to thing it through, but you are not automatically in a situation where you are going zip right up to melt. It might be --

MR. KERR: But you are automatically in a situation where what you want to do is get water back in the core, it seems to me.

MR. OKRENT: Yes, but that is not the only thing you have on your mind. You might damn well know you want to get water back in. You knew that before, too, presumably. There may be other things that people are thinking about, like is the core heating up at one degree a minute, one degree an hour, or whatever?

MR. KERR: Now, this is one of the reasons that I think it could be very important to give some thought to this sort of thing before one specifies the instrumentation. I think you need to ask yourself what sort of information could you get and what would one do with it -- what would an operator do differently if he had it than if he didn't have it -- before you can really know what sort of instrumentation --

MR. OKRENT: But it can tell you different things about the time you have to do certain things, also the time the governor has to do certain things, to hook up the fire hose or whatever it is.

MR. KERR: Dave, I agree that it may. What I am saying is somebody needs to look at this. You certainly aren't going to cover all the situations, but it might give yoi a much better guide as to what you put in and where you put it if you did have some scenarios.

MR. OKRENT: Right now, if I had to guess, of the order of a half-dozen of these that go up to the normal range of -- not just to your saturation temperature. It could be a reasonable compromise. You can't get all the kinds of different kinds of information you might ever dream of, particularly if you go to the single subassembly problem. That you are not going to be able to get, and you have to sort of write that off, I think. I don't see how you will get that.

MR. MARK: Does this bring us to the end of the topic?

MR. D'ARDENNE: I have one more slide.

MR. MARK: Oh, excuse me.

MR. D'ARDENNE: It is just a summary to say what we have discussed focuses in on the PWR unique features because we felt that generic comments were really addressed adequately by the AIF and ANS.

What we feel is necessary is that we need to have a variable selection criteria established, and that these should be based on procedure operator guidelines which the Owners' Group has developed, and they are independent of the event; and that these should be integrated with the functional criteria that the staff is working on, NUREG 0696, and that they should focus in on key variables; and finally, that we should eliminate marginal variables like the thermocouples. As we have stated before, for the reasons either stated by the staff or for any other reasons, we do not feel that the thermocouples are either warranted or reasonable on a technical basis.

Thank you.

MR. MARK: Thank you, Mr. D'Ardenne.

I have one question on which I would like to get comments from the staff. It was suggested earlier that you are requiring instrumentation to measure radiation at a tenth of natural background. Now, I heard that and I am sure that was said, but it can't possibly be true, can it?

MR. STODDART: That is essentially correct.

MR. SHEWMON: Which, that it can't possibly be true, or you want him to specify a tenth of background?

MR. STODDART: That it specified approximately a tenth of background.

MR. MARK: That is 10^{-6} R per hour.

MR. STODDART: 10^{-6} R per hour.

MR. MARK: I find 10^{-6} microcurie per cc. That is not the same thing.

MR. STODDART: No. Microroentgens per hour, basically. Ambient background radiation can run from, oh, 10 to 20 microroentgens per hour, depending on the particular part of the country. Around here it would be about 10.

MR. MARK: And this is to follow the course of an accident that we wanted to --

MR. KERR: No, it is to see if you have had some strange situation occur in which the background has dropped to 10 percent of its normal level.

(Laughter.)

MR. MARK: Thank you. I think we can adjourn for lunch, and I expect we will want to take off at 20 after 2:00.

(Whereupon, at 1:20 p.m., the meeting was recessed, to reconvene at 2:20 p.m. the same day.)

End 1

AFTERNOON SESSION (2:20 p.m.)

MR. PLESSETT: Let's reconvene and we'll call on Mr. Silver. Would you present your discussion of Three Mile Island Unit 1 status?

MR. SILVER: Surely. This slide is my total presentation, and let me explain some of the items on it. As I'm sure you know, we issued an SER on June 11th of this year. We have previously issued a status report covering the same ground essentiall ... comparing the SER to the status report we reduced what I've called the number of entries in the status summary, which is essentially the number of open items but not quite, from some number in the nineties, depending on how you count, to a number like 35, again depending on how you count.

To break these down, there were approximately 17 design and analysis items, 6 items in the management-financial area, and 12 in the general category of procedures, tech spec, outstanding tests and the like.

Several items that attracted some attention during the last ACRS meeting are indicated under items of ACRS interest. This reflects the status as of the SER essentially, and in fact the status as of now. There has been very little change from June to this date.

The first item is an unambiguous indication of inadequate core cooling or water level measurement for simplicity's sake. The licensee is not yet in compliance, has not made a commitment to do this despite our statement in the SER that what they have done, which is state that they don't think it's necessary. They still have not satisfied this requirement. My understanding verbally is that that is still their opinion; that is, that this is neither necessary nor do they know how to do this in an unambiguous way that would not cause more trouble than it resolves.

MR. KERR: Mr. Silver, is the disagreement about water level or an unambiguous indication of inadequate core cooling? Are the two being treated as if they were synonymous, or is there a distinction?

MR. SILVER: I guess I indicated they are synonymous. That's not the case, of course. If there were some method of unambiguous and direct indication other than water level measurement, that --

MR. KERR: No. My question was is their position that they can indicate inadequate core cooling without measuring water level, or is it just that they say one doesn't need to know whether there is inadequate core cooling?

MR. SILVER: They believe that the existing instrumentation does in fact indicate inadequate core cooling, and to some extent the onset or the imminent onset of inadequate core cooling. Our position is that it does not adequately indicate the --

MR. KERR: Okay. So there is not necessarily a disagreement about the need for an indication of inadequate core cooling as a disagreement about the method? I haven't seen the

SER. I'm not trying to put words in your mouth.

MR. SILVER: In fact, we are in the throes of preparing testimony for the hearing which also addresses this subject in a somewhat different way than the SER does. We feel that the existing instrumentation does not indicate adequately the imminent inadequate core cooling. It will indicate inadequate core cooling. It will not indicate that we may be approaching inadequate core cooling. We may be approaching the condition which needs some modification, some correction, and therein lies the difference.

I believe the licensee in effect is saying that they have no way, they don't believe, to indicate this without possibly making the situation more ambiguous than it might be without the instrumentation.

MR. KERR: Thank you.

MR. SILVER: The next item is reactor vessel head vent which we have required and the licensee has committed to install such a vent, but we have not yet seen either a conceptual or a detailed design in this area.

The third item which I've listed is management, and of course, their management and organization has been in a state of flux essentially since shortly after the accident. It has changed several times, and we are expecting this week in fact, probably tomorrow, another submittal changing the organization again. This is a conscious stepwise change on their part.

This new submittal is expected to establish something called the GPU Nuclear Group which is, they hope, we hope, the immediate precursor to the GPU Nuclear Corporation, which is still in the throes of PUC, SEC, and other kinds of approval necessary for the establishment of an actual corporation.

The Nuclear Group, according to advance information we have, will be essentially the same organization as the Nuclear Corporation with different titles, and because of the amalgam of organizations of managers essential to GPU and what have you, many of the individuals who have multiple titles in order to accomplish their job, but it will be essentially the immediate precursor to the Nuclear Corporation, requiring only the final changes in job titles and that sort of thing.

We are completing this week a series of inspections by I&E at the site in an attempt to get a better handle on some of the questions raised by the Commission in their March 6th order which amplified the August 9th order as far as management issues the Commission was concerned about -- was and is concerned about.

Schedule.

MR. OKRENT: Excuse me.

MR. SILVER: Sir.

MR. OKRENT: Before you leave the subject titled "Items of ACRS Interest" --

MR. SILVER: It was not intended to be a comprehensive list.

(Laughter.)

MR. OKRENT: Are there any things that have arisen out of recent experience since the TMI-2 accident that the staff has felt needed early action and which might apply particularly to TMI-1? Was anything learned out of Crystal River, for example, that applies to TMI-1?

MR. SILVER: Much of what happened at Crystal River applies to TMI-1, yes, and again, essentially anything that happens to operating plants or any information received could very well apply to TMI-1. And such requirements, if requirements do result, will be laid on TMI-1 in the same way as other operating plants except those which have particular -- have a nexus to the TMI-2 accident.

The TMI-1 restart program is being separated, if you will, from the other matters that may very well apply to TMI-1 but have no nexus to the accident.

MR. OKRENT: Well, I don't know quite how to interpret your words, but I'm not sure whether you're telling me you're treating TMI-1 as the same as all other operating reactors or operating reactors currently shut down, but operating reactors, or whether you're giving it some kind of special treatment requiring more --

Did you answer that question before I asked it? If not, would you?

MR. SILVER: I think for items resulting from the

TMI-2 accident, for example, the Action Plan -- and I had intended to get that under the heading of additional items later, in a moment or two -- for those items that have a direct relationship to the accident we may or may not be treating them in a special way depending on the item and the nature of the requirement and a variety of things. In general we have treated them in a special way. For example, we have -- and I would tell you in a moment -- essentially required the full list of NTOL requirements to be applied to TMI-1. Most of them have already been applied in one way or another anyway, but we have proposed -- I shouldn't say we have required this, because we have not officially yet -- the staff has proposed to its management that such be done. I would say it will happen or will not happen as the case may be within a matter of days.

But such items, for example, as, just to pick one, fire protection which, as I understand it, is due to be resolved by October of this year. This is proceeding and presumably will be resolved by that time, of if not by that time, certainly prior to restart, but not directly as a requirement for restart.

MR. OKRENT: Well, let me pose the question a different way. If TMI were to restart as you now envision it, would it be subject to the same loss of information that Crystal River experienced or will there have been changes made before their startup, at least not by the same number of faults, let me put it that way. I suppose it's always subject to the same loss of information but --

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MR. KERR: The same scenario, you mean?

MR. OKRENT: Yes. Could the same scenario lead to the same loss of information and so forth, or has the latter been fixed -- I'm just trying to understand the situation.

MR. SILVER: Yes. There have been no unique requirements for TMI-1 because it's TMI-1 as a result of the Crystal River incident.

MR. OKRENT: But there might be some unique requirements for TMI-1 because of TMI-1 and it's next to TMI-2, so let me pursue that point a minute. You know, the staff is looking in an extra way at Zion and Indian Point and now Limerick. Has there been consideration as to whether and to what extent they should look in a special way at TMI-1?

MR. SILVER: The reasons, of course, for Zion and Indian Point are siting requirements which the situation does not exist to the same extent at least at Three Mile Island.

MR. OKRENT: But it's not a wonderful site in the sense of being 5? miles from anybody.

MR. SILVER: That's correct. I would say no. Aside from its nearness to TMI-2, it has not been considered uniquely with respect to the Crystal River accident, and I don't directly see any reason that it's proximity to TMI-2 would necessarily cause a problem in that respect.

MR. OKRENT: Well, I suppose it depends on how you combine things in your mind. Now, along the lines of ACRS interest, the ACRS in commenting on the Action Plan and at other occasions has suggested that each operating plant do a study on possible mitigating features for containment like vented filtered containment. This is aside from however the rulemaking hearing goes, to look at the pros and cons. And they've also suggested that each plant or groups thereof do studies, I guess what you would call some kind of IREP, to see if there are any places that are particularly meet points or whatever you want to call them.

Has the staff in any way looked at these recommendations in terms of TMI? I know you're doing it at Indian Point and Zion and Limerick.

MR. SILVER: We have considered adding TMI-1 to the list of clients which are doing or will be doing an IREP in the near future, and the current staff recommendation that I mentioned a moment ago does not include an IREP for TMI-1.

MR. OKRENT: Well, just so we're talking in the same language, when the staff says IREP, to me it means this is a study the staff will do. And what the staff recommended was that at least for plants that the staff wasn't going to do, the utilities do them, and the utilities might even do them for the ones that the staff were going to do, too, but that would be a separate question.

So I just wanted to make sure that we're talking the same --

MR. SILVER: I'm not aware of that Met Ed or B&W

are doing these. I cannot answer your question directly.

MR. OKRENT: But has the staff considered this and rejected it? Has the staff considered it?

MR. SILVER: I cannot answer that for my management. I have not, frankly, and I have not proposed it.

MR. OKRENT: All right. Well, why don't you assume I'll be interested in hearing about it.

MR. SILVER: Okay. To continue with schedules, I had mentioned that we are expecting a management submittal this week. Based on that schedule we would expect to issue an SER covering the management issue in late September.

The current schedule I have verbally from Met Ed on open items resolution differs significantly from previous verbal schedules I had received over the past couple months and now talks about August 31 as a target date for submittal on information which presumably would resolve some open items or all open items.

Based on an August 31st submittal we would produce an SER, barring interference by the hearing itself, in early November.

The financial issue is the schedule of December 15th for an SER is based on a September 15th submittal by the licensee of the new financing plan. This date is not a firm commitment again from Met Ed but just an expectation based on questions we have raised, the meeting we in fact have scheduled early next week to discuss these questions. And I think that date is reasonable.

To just mention additional items, again TMI-1, of course, is subject to all operating reactor requirements, as I've already mentioned, and in addition, the staff has proposed the full NTOL list be implemented by Met Ed.

The attachments I have -- I don't have a slide -- but the rest of the attachments that I believe you have is a list of all the items in NUREG-0694, the NTOL requirements document, and indicates where each of the items has been required, either in the order or in various letters from the staff, or in fact by the document which is not yet promulgated, those that are called new in this list.

And as a final note for your information, the hearing is scheduled to start approximately October 9th in Harrisburg. I have no way to predict how long that might be or when the actual restart might occur, assuming that would be the result of the Hearing Board's recommendation after hearing the case.

MR. OKRENT: On the schedule is there some time when you expect that the ACRS would be reviewing something and providing an opinion or not?

MR. SILVER: I've been discussing this with the ACRS staff, and we have contemplated a subcommittee meeting, I am told, in late October and a full committee meeting in November.

MR. OKRENT: Okay. Getting back to the point I was raising earlier, I guess I'd like to maybe restate what I was suggesting in a more positive vein. Speaking as an individual obviously, not for the ACRS, it seems to me that the staff and the Commission should at least consciously consider whether TMI-1 should be looked upon as the same as any other operating plant, excluding the high density plants, or whether it should be re-viewed in some kind of special status, possibly the special status accruing from a combination of the fact that this is a moderately high density population site, that there has been an accident, that there may be policy reasons for providing an extra degree of assurance, if one wants to put it that way.

MR. SILVER: I am certain this has been done and will continue to be done, but the point I was trying to make is not necessarily as part of the restart proceedings.

MR. OKRENT: Well, if you don't do it as part of the restart proceedings, I'm not quite sure how one does it, but maybe you could explain that all to me some time.

MR. SILVER: That's my presentation.

MR. MOELLER: As Mr. Silver mentioned, the SER now is scheduled for November the 3rd, and we had also indicated the possibility of a subcommittee meeting the end of October. However, that now does not seem too logical, because certainly the sub-committee should not be meeting before the SER is --

MR. SILVER: I can offer this. On the assumption that the applicant, or the licensee rather does make appropriate submittals on a schedule that would satisfy this, we would certainly have at least a firm draft for the subcommittee meeting and probably before it.

MR. MOELLER: Fine. Thank you. I think, though, we must keep that in mind because that is a very crucial item in the schedule.

MR. PLESSETT: Any other comments?

(No response.)

I guess not. Thank you, Mr. Silver.

MR. SILVER: Thank you.

MR. PLESSETT: Now, I think we'll proceed to the next item on the agenda. Let me see if I know what this is about. What's involved here, I guess, is the fact that you had to prepare a list for the Congress, right?

MR. GEORGE: Yes, that's correct. To satisfy NRC's statutory requirement for identifying unresolved safety issues. I'm sorry.

MR. SHEWMON: The agenda says 15 minutes summarized by the subcommittee chairman. Are we scratching that and going directly?

MR. PLESSETT: No. I'm just going to ask -- well, I'll ask you, Mr. Subcommittee Chairman. How come this didn't go through the full committee?

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MR. SHEWMON: It's before the full Committee.

MR. PLESSETT: No, it's already gone up to Congress. My impression was that it has not gone to the Congress. It went to the Commission and came back down here because they asked if we commented. Maybe it did. Can somebody tell us?

MR. SCHROEDER: Frank Schroeder from the staff. Yeah, maybe I can clarify that. You remember when we came down here last and described to the committee the methodologies we were using to identify USIs we pointed out that we had a commitment in last year's annual report to get a special report to the Congress in July identifying any new USIs we had identified particularly as a result of (unintelligible) but which we have not been able to complete in time to put it in the annual report where it would normally have been.

We explained to the committee then that because of the extremely tight schedule and commitment to get the report to Congress in July we would not have time to get back to the committee for review of those items. We prepared our report -- or at least our Commission paper -- which summarized what would be in the report to Congress and came over to the Commission with it. And the Commission decided, after briefing by the staff, that they wanted some extra time to consider it themselves and that they would like to hear the advice of the HBRS on this matter.

So the Chairman decided to inform the Congress that we would not be able to supply the report in July as promised and that we would supply it later. So the report has not gone to Congress.

MR. PLESSETT: Fine, thank you. Please take over Paul. You seem to be on the track at this time and I --

MR. SHEWMAN: Well, that's what I heard a week or two ago. Okay, I guess partly because of that schedule there has not been a subcommittee meeting on this because this came on the agenda only a week or two ago -- the fact that there would be a full committee review.

In looking over the thing here it seems to me that there's two questions that the subcommittee or the committee has to look at. One -- to look at the issues we have to see if there shouldn't be some deletions or omissions. A more philosophical point that bothers me some is that, at least in my words -- or my impression is that the definition of USI, or unresolved safety issue, used by the staff this time is to say these are the highest priority items for the study of the staff. Therefore, they will get the resources. They will get the special scheduling and things of that sort. And to say that a high priority item is an unresolved safety issue is, I guess, one way to choose high priority items.

But I'm a little bit concerned about the fact that if we keep taking all the high priority items we have and calling them unresolved safety issues that, well, it is one political comment that certainly lays us open to the criticism which was partly responsible for starting this -- that we never get rid of our unresolved safety issues because we add them as fast as we move them. And therefore we still end up with 42-1/2 or whatever it is unresolved safety issues this year, next year and the year after.

So I don't feel particularly comfortable with it and I don't know whether this change in semantics is going to get us in trouble whether we can say something which is low priority next year is now a resolved safety issue. And so maybe how things get resolved by this new procedure might be worth a discussion as well. I think the unresolved safety issue should be high priority, but whether everything that's high priority should be an unresolved safety issue.

So you might keep that factor in mind also as you hear the presentation of what the staff has indeed come up with. You can judge your own way which way you think they ought to go. Unless there's questions, that's all I had.

MR. GEORGE: Okay. What I had planned to do was, in light of the fact that --

MR. PLESSETT: Would you identify yourself?

MR. GEORGE: I'm sorry. Hank George from the

Generic Issues Branch, Division of Safety Technology. I have a few handouts here which it may help to follow

I think that in light of the fact that we did go over the process a couple of months ago I hadn't planned to go into too much detail on that -- just an overview to refresh what we had done. As you recall, the NRC does have a statutory requirement to identify unresolved safety issues, report these to Congress, and to annually report on the progress toward their resolution.

In response to this, a definition of an unresolved safety issue was developed approximately a year and a half ago. It was a definition that was developed by the staff originally. The Commission changed it significantly. The definition we're working with is one which essentially was given to us by the Commission.

MR. SHEWMON: Now is that what -- an unresolved safety issue is a matter affecting a number of nuclear plants and posing important questions concerning the adequacy? Is that the one they gave you?

MR. GEORGE: That's correct. That definition shows up in NUREG 0510 and this is essentially the definition we're working with. Now I think I recognize some of your concerns as to if we're going to resolve the issues we have now, how are we ever going to get finished with unresolved safety issues. As soon as you resolve those a new group moves to the top.

Part of the problem, I think, is related to the way we have the definition. It says "poses important questions concerning the adequacy of existing safety requirements". And when we developed -- tried to elaborate on what that meant with the Commission these are the words that were agreed upon. And if you look at items 1 and 2 you see that it says that it "compensates for a possible major reduction in the degree of protection or (2) provides a potentially significant increase in the risk." Now this may sound a little bit alike. But the intent is, number one -- the first one -- should be those things that are deficiencies to existing regulations or deficiencies in existing criteria.

We found, for example, from operating experience that we need to make some changes in our criteria to bring protection up to where we thought it was.

Number two is forward-looking. It's saying that if we do this additional protection it'll significantly decrease risk. Things perhaps like emergency preparedness, evacuation -- things that are forward-looking. We say that if we do those -- consideration to melted cores is another example.

The problem with this definition, then, is that as soon as you take care of those things that are the greatest contributors to risk then other issues now move to the top and they're the greatest contributors to risk and by this definition you have to pick out some of those things as unresolved safety issues. So the definition that we're having to work with is one which, you know, you're really not going to get an end or one final level of safety.

MR. SHEWMON: That definition's not in your handout and it's not the definition that the staff gave us which came out of section 210 of NUREG 0410.

MR. GEORGE: Okay, let me explain what we have.

Section 210 is -- obviously that's the words out of the amendment to the Energy Reorganization Act. And all that really says is identify unresolved safety issues. And it doesn't provide -- at least what's in the law does not provide any clarification on what that means. It was the Commission that developed that definition. That one shows up in NUREG 0510. It also shows up in a SECY paper, 78-616.

If you also look at this most recent Commission paper we have, that one also includes this current -- this definition that we had here a second ago, as well as this elaboration.

MR. LEWIS: Can I go back to your interpretation of that definition? You said that this implies that if you remove something from the top of the list something else will pop up and take its place. And I don't read that in there, because while there will also be a most significant unresolved issue -- it's just a matter of definition -- that most significant issue may not be very significant. So I can envisage a world -- it's not the one we live in now -- but a world in which although there were plenty of questions around none of them were very significant. Do you preclude that? Do you have a criterion for precluding that?

MR. GEORGE: If I follow you line, what you're saying is at some point we may get to where there are a whole number of issues that are of minor significance -- collectively they are of large significant?

MR. LEWIS: No, no. I'm getting at the question of quantitative safety standards in an oblique way. That is, the buzz words for these things are always potentially significant, or major reduction, and those are quantitative statements made non-quantitatively.

MR. GEORGE: Sure.

MR. LEWIS: But I inferred from what you said earlier -- and I may have been wrong -- that you are taking the view that there will never be no issues. That the most important existing issue will always be potentially significant. And I don't see that as the logical consequence.

MR. GEORGE: Well, I guess what I'm saying is that you know you're always going to have something that is going to be the major contributor to whatever the real risk is.

MR. LEWIS: True. But that major contributor may not be very important. In the best of all worlds -- if the NRC does its job -- the major contributor will still be 10^{-123} probabilities or however else per year. If the NRC does its job...

MR. GEORGE: Well, I guess what I'm saying is that I don't see that out built into the definition. I agree. I think it should be. Somewhere we should set a level that says once you get this low you don't have to worry about the most significant ones. Even if you're going to decrease it.

MR. LEWIS: I was only commenting because you seem to preclude that.

MR. GEORGE: No, it was not intended.

I guess the point is that -- in response to what Paul was saying -- I agree we are picking, you know, some that are the highest priority and the most significant. You know, somewhere, perhaps, we ought to work into this definition some stopping point.

MR. LEWIS: Well, (unintelligible) at the issue. Because the industry keeps saying that the NRC doesn't know when to say stop.

MR. SCHROEDER: Clearly it would be very helpful in applying this definition to have a statement of safety rules which provided some way of determining what's significant. (unintelligible). So he sees potential difficulties with continuing the nature of this definition.

MR. SHEWMON: Could we get a copy of the definition you referred to? And, you know, it should be part of the handout if that's what you're putting on the screen. I've got SECY 80-325.

MR. GEORGE: This is the definition. Okay that's what was termed the definition in NUREG and then the Commission elaborated, actually working with the Commission.

MR. SHEWMON: Well, that one we have. Now the other one that you said you used?

MR. GEORGE: Okay, let me go back to the first one. You see the words up in here "that poses important questions concerning the adequacy of existing safety criteria". And that was expended to "matters imposing important safety questions" with this description.

This refers back. It's expanding on.

MR. BARNES: I'm Chris Barnes from the staff. This is the following paragraph in closure to SECY paper and is an expansion and clarification of the definition (inaudible).

MR. GEORGE: Both of these are quoted out of NUREG 0510 which were used a year and a half ago in identifying current unresolved safety issues.

MR. BARNES: The closure to the SECY paper after the close, the following paragraph begins "in applying this definition..."

MR. SHEWMON: Okay. We found it. Thank you.

MR. GEORGE: I apologize for not having this in the handout. I wasn't planning to use it in just going over this briefly.

MR. KERR: You've got to recognize that these college professors always want definitions.

MR. GEORGE: Going back to some of the background discussion, then, a year and a half ago, as we mentioned, we had NUREG 0510, which used those definitions and identified the current set of unresolved safety issues.

The issues that were in there -- 17 individual unresolved safety issues -- the staff has, over the past year and a half, expended considerable time and resources towards resolving these issues. Several of them are resolved or close to resolution.

Since NUREG 0510 I'm sure almost everyone's aware there's been a large number of recommendations, concerns, issues identified -- things that have come from TMI, some things that have come from operating experience. So what we had to do was go back and take a look at all of these and try to identify which of these potential issues were generic -- were satisfying the definition. Which ones were generic, posed important questions concerning safety.

To simplify the review process that we looke at, it was essentially the three steps that are here. The first one is identifying all these various issues and they came from sources -- the TMI action plan, ACRS recommendations and reports over the last year and a half, recommendations from staff. A number of these were recommendations from operating experience, abnormal occurrence reports. These were the issues that we're now taking a look at.

The second step is the initial screening. At that step what we were doing was taking a look at all of these issues. The issues identified in step number one were on the order of 425 various recommendations and concerns. At step two what we're identifying is which ones of these are generic, throwing out those that are just plain specific, which ones are not yet resolved. They may be ones where the staff position is in process or may have already been developed.

The result of this initial screening was a set of 44 candidate issues and the final step -- step three -- to apply, you know, the final element of the definition for a USI, was to evaluate the safety significance of these issues. Now what we did in taking a look at these was, where possible, we would try to relate it back to whatever information we had available -- quantitative information, risk analysis -- and look at it in those terms.

Unfortunately, because of the time limitation that we had, we certainly didn't have the time to go back and do detailed risk analyses of all 44 issues.

We also had some limitations in that the Probabilistic Analysis Staff is pretty heavily worked right now and we didn't have, you know, a lot of people there that we could turn to. So the process that we used was one that -- there is an attachment that's in the SECY paper that elaborates on it. But it was essentially one that asked a set of specific questions concerning the issue trying to determine things like the extent to which this deficiency may exist. Is it just an idea in someone's mind? Or do we have operating experience that shows that it can be widespread? Is it an issue that, if it did exist, could cause loss of some safety functions or rupture some fission product barrier or could even lead to degradation of some emergency preparedness plans?

And collectively, then, looking at what answers that we developed to these questions, we would make a determination of the significance of the issue. The answers to those questions were initially developed by a core group of individuals that are in the Generic Issues Branch and Safety Program Evaluation Branch.

MR. KERR: Did the person who wrote those definitions, which would seem to me to talk about needed decreases in risk or possible decreases in risk, have in mind that one was going to be quantitative in his evaluation? I would have thought, seeing the definitions, that one might have expected to be as quantitative as one could be in determining if significant risk exists or in determining whether a decrease in risk would be followed by some action.

MR. GEORGE: Well, I guess, you know, trying to infer the intent from the definition may be difficult. The definition that's there is qualitative. There's not a quantitative number in there that says you need -- it must be an issue, for example, that reduces the issue by a factor of four the risk. That sort of problem.

MR. KERR: It talks about significant decrease of risk.

MR. GEORGE: So someone has to judge whether it's significant. What we tried to do was use some individuals that were familiar with the risk assessment and make a determination -- a judgment -- as to whether it appeared that this issue would have a significant impact on whatever risk models we had -- the information we had available.

MR. KERR: Well, insofar as your qualitative judgment permitted, did you consider a decrease by a factor of two to be significant?

MR. GEORGE: If we look at the output on this --

just to jump ahead a second -- it may help in answering the question. We identified some issues that we felt were clearly unresolved safety issues and some that we were clearly not unresolved safety issues and some for further study. And I think this third category of further study somewhat reflects the uncertainty in these judgments of risk and I see where this further study is more the in-between group, where maybe it is only a factor of two. maybe six. We're not sure.

And just discussing this further study a little bit more. What we're saying on most of these things that are in further study is because they fell into this group we're not sure just where they fall relative a threshold and we need to look at them a little closer and try to develop more quantitative data on these issues so we're better able to determine them. They're more in the gray area rather than the black-and-white.

MR. OKRENT: First, I'm a little surprised that when people mention things like a factor of four or a factor of two you didn't say that if it were a factor of ten percent we would consider it significant. Because I don't think you're going to find very many things that you can attach a factor of four to on overall risk. If you can isolate them you'll probably do something about them. I think you're going to talking about one of many contributors. And unless it really stands out, in which case you're probably already doing something, it's going to be buried somewhere.

Your criteria used for screening, there's one that says -- the second one -- the staff position on the issue for recommendation has been developed -- that I understand -- or could be developed within six months. That part is a little less clear to me because things could be developed but they may not be developed for two years and six months or four years and six months. Now does that mean they are unresolved safety issues? They may meet all the other qualifications. In fact they may meet the wording that the Commissioners approved specifically. And just to say you could arrive at a position in six months. I must say, is that what's resolved?

MR. GEORGE: Yeah, I see the question that you're raising.

MR. SHEWMON: Near resolution because a report is in progress or being issued.

MR. OKRENT: I know what the word -- I find that particular case that's to be used in screening a curious one and another one -- when you said the definition of the issue requires long-term confirmatory, exploratory search -- it's number five -- and indicating in a sense if it's long enough term we won't put it on here even if it meets the criterion of raising important safety issues.

MR. GEORGE: Well, I don't think the long term is what we are really hanging our hat on. It's more the confirmatory-exploratory. It's an issue where you don't have a specific concern. You just want to do some research in an area to see what behavior is. You know, fuel behavior and under certain conditions. But you don't have a specific conern identified.

I might point out that we didn't use that one very often. Some issues -- just because an issue involved research we certainly didn't screen it out, even if the research took more than six months. There's a lot of research associate with core melt that, you know, we're saying that's really a part of the degrade core issue.

MR. OKRENT: Well, I'll go back and see if I have any which had a five that I wanted to ask you about. You certainly used two quite a lot.

MR. GEORGE: We certainly did. I think in most cases where we were using two were where we either had a position or we knew one was being developed and the six months may have been arbitrary. We tried not to rely on that where it looked like something was going to take maybe five or six months and we said well we're not going to rely entirely on the schedule. We know it can slip, so we're not going to screen it out on that basis.

The intent on that six months really went back to the definition where it said something to the effect that where resolution has not yet been developed and we felt, I guess, that by the time you go through this evaluation process, you get the NUREG report written, you get the Commission paper out. By the time you report it to Congress the issue is resolved or may be resolved within a month. So that's what the intent of that six months was, but if it was something that the current schedule showed it taking on the order of five or six months we didn't rely entirely on that.

It was more of one where it was on the order of two or three months.

MR. OKRENT: To come back to ATLAS for a minute, let's assume there is not a staff paper or so forth. Would you say that's one that requires a policy decision or it's an unresolved item in the sense of needing something else?

I don't think -- because your number seven is "the issue or recommendation requires the policy decision rather than a technical solution." Let me give you another example. We've been talking about is the single failure criterion adequate for about a decade, I guess. Was that a policy decision, or what? I'm trying to understand what these words mean.

MR. GEORGE: Okay. On a policy decision we were relying more on things where maybe a management decision related to how NRC operates or the ACRS interfaces -- organizational type things. On something like single failure criterion, that one specifically we were saying well, the way it's written into the action plan it's relying on IREP and NREP. See what the results are and then make a determination as to how we want to change that.

So we didn't say that that was the policy decision.

MR. SCHROEDER: I think clearly the intent was that if we felt that the information was well enough in hand -- the technical information -- that what was really required was NRR management or the Commission itself to simply look at what was now available and decide are we are aren't we going to impose this requirement. We categorize that as not warranting putting it in the unresolved safety issues.

MR. OKRENT: So you're telling me there may be an issue that meets the criteria, namely it could have an important effect on any existing plans or so forth. And the information may be developed, but if the NRC can't make up its mind it's now unresolved. Curious.

MR. CRONIN: I'm Chris Cronin. I think a perfect example of the policy decision application is siting criteria. The staff has done technical work on that and issued a report that develops some recommendations. They also go forward in the second policy task force --

MR. KERR: Excuse me. Are you having trouble?

THE REPORTER: Yes.

MR. CRONIN: I'll try it again. On siting policy we screened that out on policy decisions, and we did so because we feel the bulk of the technical work has been done and now it's just a matter of selecting what criteria should be applied for siting. And certainly I believe that this committee and probably a number of other people would agree that siting criteria is a significant issue and may significantly affect power plant design in the future and maybe even for operating plants. But we screened that one out on the basis of policy decisions.

MR. SCHROEDER: Let me add one more thought. I think we have to admit that one of the things that inevitably is in the back of the staff's minds is that one of the corollaries is once you've defined something as being an unresolved safety issue is that we establish a task manager to manage the progress, we establish schedules which we document and report to the Congress, report the status on, and issue a report finally. And if you're faced with an item that a lot of that work has already been done and what remains is for decision-makers to make a decision, I think there's a tendency on our part not to want to put it into this list.

MR. OKRENT: It seems to me that those are the ones that most need identification.

MR. RAY: I have a brief question on your category --

MR. KERR: Excuse me. Have we resolved what we're going to do with that resolution? It seems incredible to me that an unsettled issue is considered resolved. Is it?

MR. SHEWMON: If it's low priority in the staff's eyes for their scheduling.

MR. KERR: I'm sorry. I didn't hear any low priority attached to it. Indeed, I --

MR. SHEWMON: Poor scheduling.

MR. KERR: I gather that siting which is a very important issue is resolved in this definition. Did I misunderstand?

MR. SCHROEDER: No. I didn't intend to imply that just because we didn't put it in the definition of "unresolved safety issues" that warrants all the trappings of such an issue, that it automatically meant that it was resolved. There are a number of generic activities that did not make it under this definition. They are not resolved. They are matters being worked on.

MR. KERR: I'm sorry. There is a third class. There are unresolved issues, resolved issues, and those that are in neither category, and I hadn't realized it. Okay. I understand now.

MR. SCHROEDER: There are other designated generic issues which are unresolved.

MR. GEORGE: Okay. The result of, as we indicated, this process was identification of a set of unresolved safety issues. These are the six that are shown in the Commission paper. The letter which we received from you, or the memo concerning this meeting indicated that there might be some interest in going through these and discussing them, which from the standpoint of how do we know that we're done with these issues, how do we know that we're ever going to finish with these.

MR. SHEWMON: My impression was that the A's were ones you thought merited action before, weren't they? Maybe you should start with telling me what A-45 means. That comes from when you divided all these things into A, B, C, and D back a year or two ago?

MR. GEORGE: What we're doing right now is really abandoning that A, B, C, or D. I think we're really saying that putting them in those separate categories really does not have much impact on deciding whether it's a USI or not. We're really going through a process of identifying which ones are in the top category, because most of the issues we have now are all A's; just to keep the designation same we were assigning A numbers to these also.

MR. SHEWMON: These are new A's then, not old A's.

MR. GEORGE: These are new A's, that's correct.

MR. SCHROEDER: You really have to look a little bit at the history of how we got the A, B's, and C's. Before the enactment of Section 210 in the Act, NRR had set up a program to manage generic issues. In the course of that we categorized those issues on our list into A, B, C, and D, which was some sort of a relative --

MR. SHEWMON: I had not realized these were additions to the A list now instead of -- all of a sudden you're saying some things are more A than others.

MR. SCHROEDER: But then when Section 210 came out and we made our first selection for the Congress of unresolved safety issues, all of the ones that we picked were out of the A list, but not all of the A issues made the definition of unresolved safety issue. So, you see, the fact that we're up to A-44 now in one of the unresolved safety issues doesn't mean that there were ever 44 unresolved safety issues. There were 44 type A issues that preceded the unresolved issues.

MR. SHEWMON: Well, how many are anointed currently?

MR. GEORGE: We currently have 17 specific unresolved safety issues in NUREG-0510, but that is based or that includes 22 different tasks. There are a number of A tasks that are combined into the unresolved safety issue on Mark I containments, for example.

MR. SHEWMON: So there are 22 tasks to resolve issues?

MR. GEORGE: That's correct.

MR. SHEWMON: Apparently more than 22 A items then.

MR. GEORGE: That's correct.

MR. SHEWMON: And this will make it 28 unless you get rid of a few.

MR. GEORGE: We have gotten rid of a few. We have some information on it if you're interested.

MR. SHEWMON: I think I've read it.

MR. GEORGE: Okay. This first one, the longterm upgrading of training and qualifications of operating personnel is a specific issue that relates to making improvements to the qualification requirements, training requirements for not only operators but non-operator personnel -- maintenance individuals, facility technicians. So it covers a large spectrum.

Now, what's intended to be covered under this Unresolved Safety Issue is the following items. First would be making revisions and improvements to Reg Guide 1.8. This is going to be both some short-term fixes, which I understand I think some of those were discussed with one of the subcommittees yesterday. Those changes, the short-term changes for Reg Guide 1.8 were going to incorporate some increased staffing and better definition of qualification requirements.

Another element of this Unresolved Safety Issue would be standards development, factoring into Reg Guides regulations criteria related to maintenance personnel, technicians, things that have resulted from a study by BEDA Associates. It's NUREG-1280.

Another item under this issue will be making recommendations to the Commission and factoring some decisions into Reg Guides or regulations based upon an NRR study -- it's 80-17 -- study of operator qualification, operator license -- very broad scope to that study. It covers everything from the selection of operators or their qualifications, requalificational testing, simulator training, additionally qualification requirements for NRC examination personnel.

That study is due to be completed November 1980, this year, and the result of that would be some subsequent changes in, as we said, Reg Guides or regulations.

Another element of this Unresolved Safety Issue would be to make some changes to 10 CFR 55 to incorporate requirements on simulator training, NRC administration of requalification examinations, and mandatory operating tests as simulators.

A fifth element that is part of this Unresolved Safety Issue would be to develop criteria related to NRC training workshops. Another component of this issue is I&E developing inspection procedures for these improved training programs.

Now, what we see is this issue involves those six elements, and we're saying that we feel that this Unresolved Safety Issue would be resolved and completed when all of these changes to either the regulations or the Reg Guides are made to upgrade the criteria related to operating personnel, operators as well as their support staff.

MR. SHEWMON: At the bottom of the discussion of that in SECY 8325 you say, "The revised requirements on subsequent rulemaking activities are expected to be completed in two years." Are there always rulemakings to resolve each of these, or why would there be rulemaking changes on this?

MR. GEORGE: There are a couple of elements in here where it's considered that it would be beneficial to make rulemaking changes. It's not necessary for all of the issues, and it's not necessary for all of the complements of these issues.

The reason that there are some rulemaking changes is that parts of this introduce requirements in areas where the regulations right now don't have any coverage -- things like qualifications for maintenance individuals, so that would be something that they're contemplating adding to the regulations. In many of the areas it's just changes to Regulatory Guides.

MR. SHEWMON: But a rule is one step more formal than the guide, isn't it?

MR. GEORGE: Sure.

MR. SHEWMON: And your answer is since we're doing it differently than we did before, we have to make a rule, whereas before a guide worked when we didn't cover these items.

MR. GEORGE: Well, even in these areas there's going to be a combination of both. It will be introducing something into the regulations. There will be a rule change, in addition to the changes that Reg Guides that amplify on what the intent of that rule is.

The second item under operating procedures, this is one that of course, you know, the NRC and industry recognize that if there were a number of reported events that directly related to procedural deficiencies -- certainly TMI-2 was a good example of some problems. There are going to be some short-term improvements in operating procedures. A lot of this is already started and in some cases already completed.

The longer term effort, and that's basically what would come under this Unresolved Safety Issue, is considered necessary to effect fundamental changes in both the content and the format of operating procedures. And the operating procedures we're talking about are both normal and emergency procedures.

The proposed Unresolved Safety Issue work activity is going to include both the NRC and the industry. The effort is going to be a longterm one to develop what the staff has termed a procedural program plan that is going to identify the requirements that the staff feels should be incorporated into procedures.

This effort relates to 1C(9) out of the Action Plan, and the basic elements of this task are going to be studies that define the organization and format of the procedures, studies that will result in the development of criteria for the format content. And the other element would be studies to assure that the content is comprehensive, to make sure that it covers transient analyses, takes into account information from reliability analyses, that it also includes consideration of some administrative prohibitions to prevent unwanted operator actions during accident conditions.

We feel that this Unresolved Safety Issue then is going to be completed once we get this definition of both the required format as well as the required content, and also address some of the problems that have been found in cross-referencing the procedure that sort of thing.

MR. SHEWMON: Would you tell me again what the Unresolved Safety Issue is here?

MR. GEORGE: Okay.

MR. SHEWMON: Or is this since you can do better, you should?

MR. GEORGE: The concern, I think, is basically one where individuals have taken a look at what the operating procedures look like, and they find that they don't feel that an operator could reasonably follow those type of procedures in an emergency. The format is such that it is very difficult to follow the information. A lot of procedures are cross-referenced so it's difficult for the operator to pull it all together and to use it.

So the issue is really to make these procedures more usable.

MR. SHEWMON: And this has primarily to do with emergencies? I mean, we've been operating with these for many years, and now as a result of TMI-2 we appreciate that these are particularly difficult to use in emergencies, or have they been poor all along and we just haven't realized it?

MR. GEORGE: I think it's a combination of both. We're saying that when we go back now and we look at LERs, we find that a lot of these have resulted from procedural problems, procedural deficiencies, and WASH-1400 pointed that out, that that was a significant contributor.

I think what we're saying is we should have been giving more attention in the past for these normal procedures, and then we found that from experience with TMI and some other events that even the accident-type procedures need improvement.

MR. MOELLER: Excuse me. I think I share Dr. Shewmon's question, or the fact is certainly I don't understand it completely. I've seen your definitions and so forth, and regardless of the definition it seems to me that in the past, Unresolved Safety Issues were issues that had been discussed and batted around for many months and had proven to be very difficult to resolve, and therefore, they became Unresolved Safety Issues that you were working on trying to resolve.

It would seem to me we should avoid just adding to the list any problem that we face, because are you having tremendous difficulty in the six items that you have listed here in bringing about a conclusion to them?

MR. GEORGE: I wouldn't say that that was a criterion for any of these, or to put it the other way, I guess I would say that some of these may not be that difficult to resolve, but they were on the list more, I think, because of a judgment that because of deficiencies that, you know, we feel exist in these areas that there could be a significant contribution to risk due to these.

The next item on control room design is one that --

MR. OKRENT: Excuse me. Would you guess that it might or might not be easier for the staff to lay out the broad approach and the skeleton that you follow for item 2 in six months, perhaps not knowing what all the procedures are, but being able to define the situation?

Might that not be easier than trying to decide whether the DC power systems on more than one operating plant is adequate? Which do you think would be easier to decide or to handle in six months? I'm just trying to get a feel for it.

MR. GEORGE: You're comparing the operating procedures, for example, with DC power reliability?

MR. OKRENT: In other words, do you have a handle on the procedure, one, in the sense that you know what the nature of the problem is, and you know how to approach it, and in general -- and one could get more specific -- recognizing that without having some information on this transient or that transient, which has to come from some studies, you can't write just the right information in the procedure?

MR. GEORGE: I'm not sure, Dick, if you'd be able to address that or not.

MR. COX: My name is Tom Cox. I was one of the group of people who contributed to the overall assessment of these candidate issues, and I'd like to point out there just to reiterate something that Hank George said a moment ago, that if we're looking for why this particular issue has become a proposed Unresolved Safety Issue, I think it's because of the perceived level of risk, which admittedly is a qualitative judgment assessment at this point. However, the panel did feel that while, yes, Dr. Okrent, the ways of arriving at the ultimate conclusion of this or the resolution of it may seem technically more resolvable than loss of DC power, nevertheless because of what came out of TMI-2 and all the major investigations showing and stating that there were significant procedural deficiencies that set up a significant level of risk.

As we went through this we, too, agreed that there was a significant risk involved in these procedural deficiencies and that it should be taken care of in the near future, or at least given a significant priority level and looked at.

I don't think we're saying that anything more should be done than has been defined as a task action plan -- that is, as an action plan task in NUREG-0660, but that it should receive the priority commensurate with what is the perceived risk level.

MR. OKRENT: Fine. In fact, I think there is an important area in the area of procedures, and the ACRS in fact early on identified that as an area to be worked on, so I'm not disagreeing in that context. But it seems to me it's a more straightforward problem at least to define in general what one wants to do and then to get the necessary work done, and in some cases to get some information which is perhaps not in hand to write a specific procedure than some other things which are more complex perhaps and which may represent what you call policy decisions that could be made in six months, but in fact sometimes are not made in 60 months.

MR. KERR: Is one of the requirements for an Unresolved Safety Issue that you be able to tell when you have it resolved?

MR. SCHROEDER: That's not written into the definition, but it's very much in our minds, Dr. Kerr.

MR. KERR: I wonder how you will know when you have number two resolved.

MR. SCHROEDER: Let me add that we're at a bit of a disadvantage in discussing these proposed issues as compared to when we discussed with people the issues that are now in the Unresolved Issue category, because at that time we had already laid out a task plan.

In some cases in retrospect we recognize now that even those task plans did not sufficiently address the concern you have, and we're very sensitive that in developing plans for the resolution of these issues we define rather clearly for ourselves and everybody else what constitutes resolution of that issue, and that we bound it so that it isn't a never-ending task.

MR. KERR: Well, is this one of the things you tried to do before you recommended that these become unresolved?

MR. SCHROEDER: No, sir. We applied the criteria, as Hank indicated. The next step in the process once the Commission agrees that these meet the definition is to assign a task manager and develop such a plan for each one and get it approved by the management.

MR. KERR: So once you've defined a plan then completion of that plan resolves things.

MR. SCHROEDER: Yes. With one caveat I'd like to add, that because of the statutory requirement to report these to Congress, there will be issues that we will consider resolved generically in the sense that the staff has completed its study and made recommendations and issued a report on it. We will continue to report the status of those to Congress until such time as those staff new requirements have been implemented on all plants to which they've been recommended to be implemented.

So we may actually reach a point where there's no more active action under the task action plan, but we're still reporting where we stand on implementing them in plants.

MR. MOELLER: I find, too, I guess, I'm not fully understanding, because like on your initial screening criteria you say that under item four that you would screen out an issue that is only indirectly related to nuclear power plant safety, and that makes sense; but then you say that an example is recommende changes in the licensing process. In other words, a change in the NRC's procedures. But here our number two item is a change in the procedures of the plant staff does count as directly applicable, but a change in the licensing process is not directly applicable.

Do you see why that would be confusing to me?

MR. GEORGE: Well, I guess the intent of number four, changes in the licensing process, refer to things like, for example, deciding that NRC is no longer going to issue limited work authorizations. You know, a decision like that --

MR. MOELLER: Some minor advisor change in the licensing process.

MR. GEORGE: Well, not necessarily minor, but I'd say in the steps or the procedures that are normally followed in getting the license, but I don't see that that's the same thing that we're really talking about here on number two. Operating procedures we're referring to are procedures that are at the facility that an operator will follow in performing his functions, either normal or emergency.

I don't really see that as one that falls under number four. If the procedure is incorrect and tells him to do something wrong, or if it doesn't have sufficient information, the operator, you know, makes an incorrect step, takes an incorrect step, you know, I can see that that's very directly related to safety, and I'm not sure that the types of things we're talking about in four

MR. MOELLER: Well, like control room design. I can see that the information you display and the manner in which it's displayed and where it's displayed could be important. Is tnat control room design?

MR. GEORGE: Sure. It covers that. Both layout as well as method of providing the information. Control room design also picks up a couple of other items. These are areas where they show up in the Action Plan as new requirements. One is a plant safety parameter display console, so that's something that's also included under number three, developing the criteria that a display console must meet.

MR. EBERSOLE: Mr. Chairman, within the subject of operating procedures isn't there an internal priority system that one ought to pick up? I'll take three examples as case in point.

It's now coming into view that we consider loss of all power as a possible or whatever you want to call it, an accident, not a design basis -- I don't care what you call it, it's an accident, looking at DC power loss. Up to about a month ago we thought ATWS was just way out in the stars, but now we don't think that any more.

If I take these three things as examples and you go to the field, I think you're going to be pretty miserable when you find out that you have virtually no procedures to cope with these things, and I think these ought to be at the top of the priority within that category.

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MR. GEORGE: Yes, that's exactly correct. One of the items that is covered under this, and it's stated in the Action Plan -- in summarizing what we had in there I didn't mention that, but that's correct. There will be some criteria developed on what events must be covered under the procedures, both ones that have been considered in the design basis previously as well as events that are not covered in the design basis.

MR. EBERSOLE: Well, I didn't say covered eventually. I meant covered quick, while we sort out what we're going to do with these matters.

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MR. GEORGE: Okay. Your question is whether some short-term action is being planned for --

MR. EBERSOLE: Yes, exactly. For instance, I am now wondering, in fact, whether the mitigation systems for ATLAS are all that good. I had some comfort in the fact that, according to GE, Louche and others, we never were going to see it, but it looks like we might.

MR. GEORGE: I can't answer that as far as any of the short-term actions. I don't know if anyone else is familiar with that, short-term actions as far as procedures.

MR. PLESSETT: Jesse, I never thought that Louche had convinced you.

MR. EBERSOLE: Not really.

MR. PLESSETT: Oh, okay. I feel better.

(Laughter.)

MR. OKRENT: But what he is saying is that while we are trying to figure out whether it is probable enough to include it in the licensing process, maybe it would be useful if they had procedures for these things in case they do occur, or when they do occur, depending on your point of view.

MR. EBERSOLE: Right. I think you will find they are pretty bare now. It is hard to believe they will work.

MR. GEORGE: Also under control room design, there is a fair amount of related research studies that the information from those research studies will be factored in to whatever new requirements are developed for control room. It involves things like operator process communication, current use of lights, alarms and enunciators, how to improve that so that it provides the most useful information to the operator without providing unnecessary information.

MR. KERR: I had gotten the impression from some of the earlier NRC staff comments that once you got the operators trained and qualified sufficiently, that they could cope with that sort of confusion, and one might not have to do much with the control room. But there is apparently going to be a more balanced approach.

MR. GEORGE: That is correct. This control room design is a pretty large effort.

MR. KERR: Okay.

MR. GEORGE: There is also going to be some research in some related areas, disturbance analysis systems and improving some process monitoring instrumentation.

MR. SHEWMON: Before you leave that, you agree there are long-term research requirements involved in this, yet that is the basis for not putting something in the action of --

MR. GEORGE: Well, not simply because it is long-term research. We screened out things. It was long-term, confirmatory or exploratory, whether or not --

MR. SHEWMON: We ought to be doing both of those on control room design, I thought.

MR. GEORGE: Well, to a certain extent as exploration, but there are specific issues that are identified, things like how to provide instrumentation that is easier for the operator to read. So it is looking at different instrumentation design, developing things that are really beyond the state of the art right now. There is some research effort in that area.

There are a number of things that get picked up under research mainly because research dollars are being used for it, but it is to investigate certain specific concerns or specific improvements. And those are part of this issue.

MR. SHEWMON: About the middle of the page under the description on that in SECY 80-25, you have got as a first step the staff will, including site visits to establish existing control room design capabilities. Can you tell me how you evaluate the design capability of a control room, or do I miss something?

MR. GEORGE: I think the intent -- those are words out of the Action Plan, and we used them, but the intent, I believe is really taking a look at the existing design and trying to get familiar with what the current designs and layouts are, the range, the scope of what needs to be considered, in addition to gathering whatever useful information you can on the size of the room and the extent to which you can make changes within the room. I am not sure that it means anything more than that.

MR. SHEWMON: Okay. As a second step you develop final control room design requirements, standards, reg guides, and improve control room instrumentation research. What do you have in mind with regard to retrofit? Or is this all with regard to new plants? Is that part of the Action Plan?

MR. GEORGE: Well, I am not sure. Maybe someone else would be able to address that. My understanding was that there would be some retrofit involved. The extent of it, of course, is going to depend on the Action Plan, and once the final criteria are developed, then someone has to make a decision as to how much difference you have between what you would like to have and the criterion which you actually have, and how much in the way of changes do we need to make.

MR. SHEWMON: It seems to me honest men could differ about whether if you redo an operator's control room to fit your ideas, it would improve or decrease safety if the same operators now have to come back in and cope and you rewired everything.

MR. EBERSOLE: Mr. Chairman, earlier today we were talking about Reg Guide 1.97, instrumentation to follow the course of an accident. We find that that accident has the connotation of really being a process system accident of some sort like a LOCA or small LOCA or steam tube failure or something. It does not embody consideration of failure of instruments themselves being the accident.

So what you are doing now in control room design must include an assessment of the quality level of the available readout information in the context of how good and reliable are the enunciators and recorders and indicators, and are there potential accidents that will scrub these, and I think you will immediately find that we have a tremendous backlog of such systems like that that are not at all seismically competent and not 1E. These are eyeball instruments. Because it was not thought prior to TMI-2 that eyeballs are very important, that plants would live with automatic circuitry. Now we know better.

MR. SHEWMON: I take it eyeball doesn't have to do with the height the instrument is at.

MR. EBERSOLE: No, it has to do with receiving it in your head and doing something with your hands. That was not a high order of business prior to TMI-2. So you are going to find in the old plants, unqualified, non-1E, unseismically qualified indicators, recorders and enunciation systems. I think in looking at instrumentation to follow the course of an accident, and taking as a case in point, maybe, a seismic event, although you can take anything else, like a fire or whatever, you have to consider that that accident might, in fact, be loss of a substantial portion of instrumentation, which renders you virtually blind as to know what to do.

So you have an interface with Reg Guide 1.97 in the control room.

MR. KERR: Speaking of Reg Guide 1.97, I find under coolant control room design a reference to post-accident monitoring instrumentation research. Does that have to do with the sort of instrumentation that is specified in Reg Guide 1.97, that this anticipates further research on that instrumentation?

MR. GEORGE: I am not sure that that was the intent of this one. I think a lot of this post-accident monitoring was really considering instrumentation that is beyond 1.97.

MR. KERR: What sort of instrumentation is that?

MR. GEORGE: I guess it could be things like radiation levels in various levels of the plant, not so much that you are following the accident, but afterwards, in which you want to get in and do some --

MR. KERR: Reg Guide 1.97 in its present version does consider post-accident monitoring, I believe.

MR. GEORGE: It does include that?

MR. KERR: Is there an interface here?

MR. GEORGE: I'm afraid I can't answer that. Tom?

MR. COX: I can maybe help here. The control room instrumentation research task that is part of this proposed unresolved safety issue is broader than post-accident instrument monitoring.

MR. KERR: I recognize that, but it does refer to post-accident instrument, and I wondered if this were the same instrumentation that is referred to in 1.97.

MR. COX: I think our understanding of this has to be yes, it could. In the end, if changes were required or indicated over what is today 1.97, those would be recommended. If recommended changes to that guide came out of this research, we would certainly consider changing the guide at that point.

MR. KERR: This unresolved safety issue, then, is just a maybe issue. You don't know whether there is a problem or not but there might be, so it is a good idea to look at it. Is that it?

MR. COX: We think as far as the instrumentation goes, there certainly are improvements that can be made long-term.

MR. EBERSOLE: Of course, looking at Rancho Seco and Crystal River and realizing we have got instruments that are vulnerable to cascade failure -- see, I get the impression that you are going to look at the geometry, the human factor and all that.

MR. GEORGE: I think the major part of this effort is the human factors. The other issue that you are raising is identified as a separate issue in the Task Action Plan as classification of electrical instrumentation, electrical equipment.

MR. EBERSOLE: But they are sure intermixed.

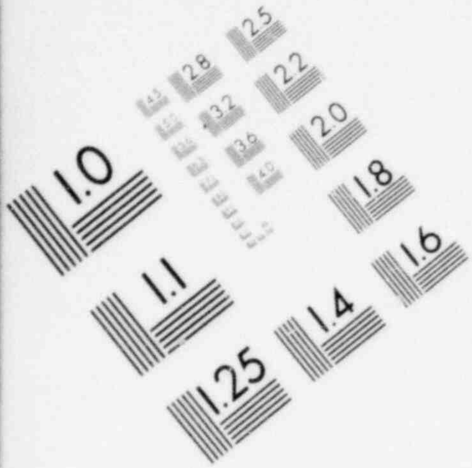
MR. GEORGE: Yes, they would certainly have to interface. No question. There is an effort that is ongoing to identify just what instrumentation needs to fall within that umbrella.

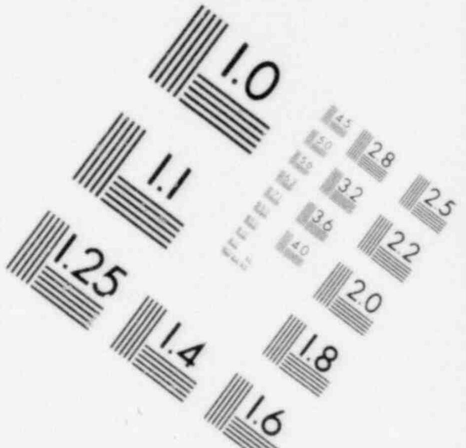
MR. OKRENT: Is that an unresolved safety issue, the one you were just talking about?

MR. GEORGE: No, it is not identified as one.

MR. OKRENT: Could you tell me why? It is not on your list for study, either.

MR. GEORGE: No, it's not. That is one that we had deleted out, or I guess recommended not including in there. I am not sure if someone else can address this better than I can, but as I recall, the essential reasons were that the instrumentation that would be affected was not going to be that much; that on a few plants, such as Crystal River and Rancho Seco, in the original design basis there





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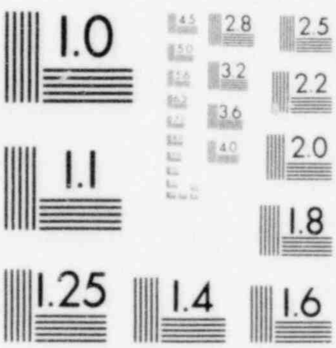

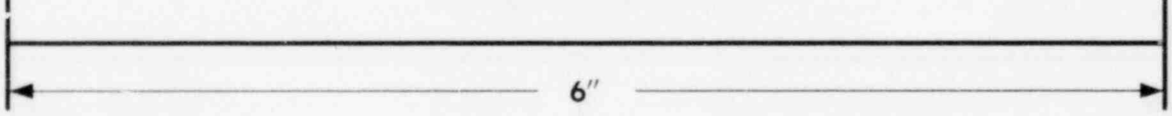

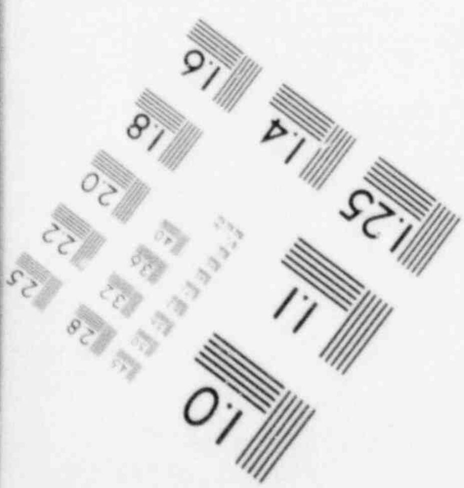

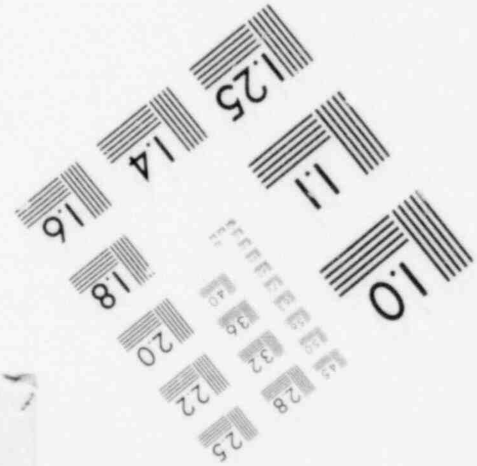

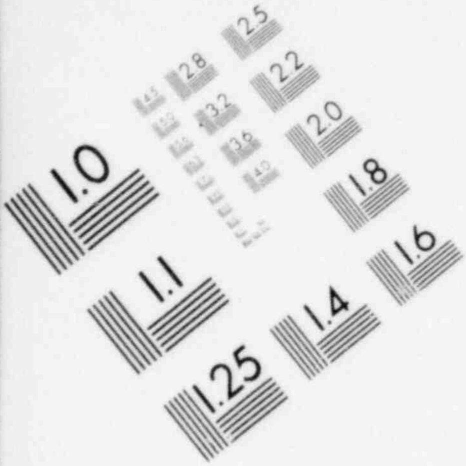

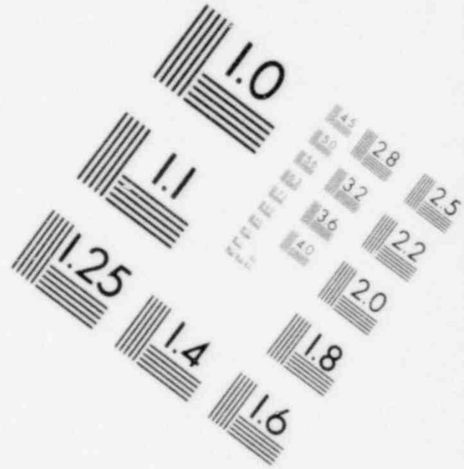

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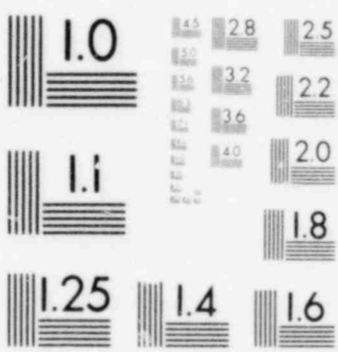

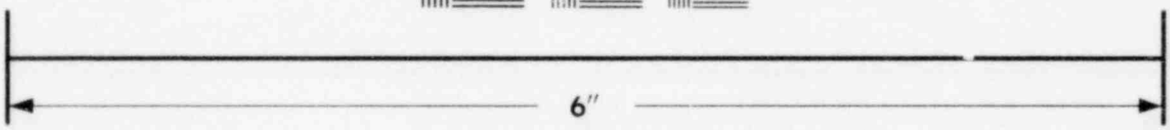

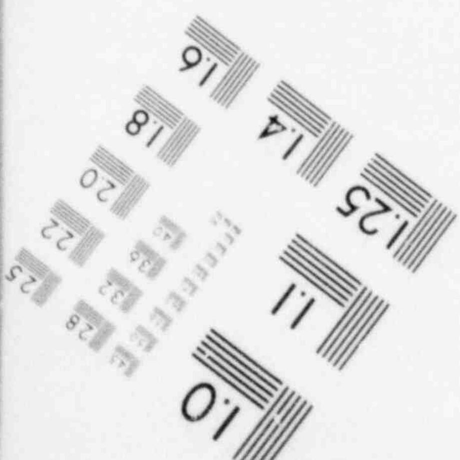

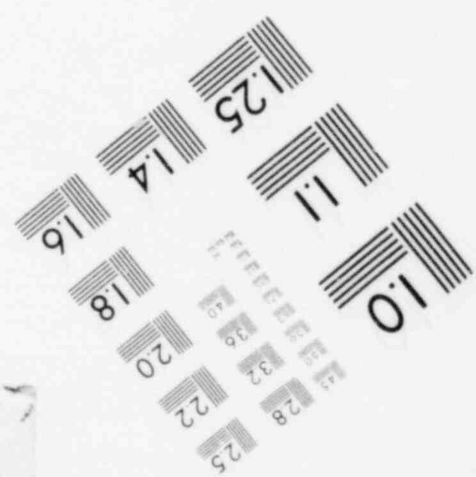

was not the requirement to have safety grade monitoring instrumentation, the only instrumentation providing input to things like reactor protection system, engineer safety features, but that it was a limited number of plants that were affected.

On most plants the criteria was that you had some channels that not only provided input but provided monitoring in the control room with safety grade all the way up. It is a current requirement, also.

MR. EBERSOLE: Does that include enunciation and indication and recording equipment?

MR. GEORGE: That I'm not sure of.

MR. EBERSOLE: Well, that's what feeds your eyes.

MR. GEORGE: Well, the instrumentation that I was referring to would be things like dials, gauges --

MR. EBERSOLE: Yes, that's what I meant.

MR. GEORGE: It does cover that. I'm not sure if the current requirements also cover the enunciation lights overhead. That I don't know.

MR. OKRENT: The slightly broadened version of this, which you certainly could find in some ACRS letters, raises the question of how one has looked at control systems and their role in safety. I would think that arises out of Rancho Seco and Crystal River, as well. It is not only the particular failures that occur there. And I don't find that quite enveloped in the item you just discussed, and I didn't realize you had surveyed all the plants even with regard to the item you just discussed to know that everything is at hand.

MR. GEORGE: Well, I will have to clarify that. We didn't survey all the plants, but it is based on discussing it with individuals in that area who are familiar with what the design criteria were for the different vintage plants. It was felt that the impact of this reclassification was going to be a limited number of plants in limited areas. This is as far as what this current standard is that is being prepared.

The other question that you are raising concerning some ACRS concerns that were raised on effects of control systems, I think we felt that there were a couple of issues that were in the Action Plan that were covering that, in addition to Task A17 on systems interaction. I guess our feeling in the Generic Issues Branch was that A17 really should be broad enough to cover some of the concerns that are brought up there, things like jet impingement from steam line break on control systems and making the event worse than what was analyzed. A17 should be picking that up.

MR. EBERSOLE: That is sort of a different question, isn't it? I mean the problem at Rancho Seco sort of was you had a control system that could lead to a loss of information, a loss of feedwater, and maybe auxiliary feedwater. I'm sort of curious -- and what I've heard, the first IREP look at Crystal River didn't consider the Crystal River event. It's an area that has not been looked at very deeply, to my knowledge, at least not in anything I have seen in writing.

MR. SCHROEDER: I think that is correct. Again, what we are dealing with here is decisions as to whether to apply the label of unresolved safety issues to some of these activities. While I am not as intimately familiar with the Action Plan, that subject is treated in the Action Plan. There are intentions on the part of the staff to take a harder look at the whole subject of safety-related, nonsafety-related, control versus protection, and so on.

It was the judgment of this group, admittedly subjective judgment, which is one of the reasons we appreciate your comments, that in trying to apply the admittedly fuzzy standards of what constitutes an unresolved safety issue, they judged that the contribution to be expected to reducing risk from that activity didn't make the test.

Other people might feel differently about that. It is not a matter that we don't intend to pursue that issue on the staff; it is a matter that it didn't make our definition of an unresolved safety issue.

MR. OKRENT: But there is implicit in this a question of when, because it is not on your --

MR. SCHROEDER: I don't recall, Dave, what the Action Plan says for schedule on that. I suspect it is a very long-term one. The staff is mandated as committed to follow the Action Plan. That is independent of whether these things are categorized as unresolved safety issues.

MR. OKRENT: Let me say I don't get any comfort out of hearing words such as staff is committed to follow the Action Plan. It is a vague kind of document, and things are or not in there.

MR. SCHROEDER: Admittedly, that is one advantage to designating something out of the plan as an unresolved safety issue, but the mechanism we set up to handle those unresolved safety issues demands an early identification of just what the scope and bounds of the schedule are.

MR. OKRENT: Mr. Lewis is here and I wanted to ask him: doesn't one of the agencies involved in airplane safety use a couple of categories? They have something that they say is a safety question or something, meaning it may be important, we better look at it, or whatever it is, but it is not in the next higher category, which probably means it is an action item and they are going to have to fix it in a certain time or so forth.

SPEAKER: It is the DC-10 --

MR. OKRENT: Am I right?

MR. LEWIS: You may well be right. I don't know.

MR. OKRENT: No.? Oh, my consultant let me down.

MR. LEWIS: Well, I'm sorry. I can tell you that there are some rules on what sort of accident you have to report to the FAA if it happens to you and you survive, and the usual things involving the loss of the airplane and all the passengers; but also, any accident, however minor, that involves a fire in flight is a reportable occurence, things like that. But I don't know of any A, B system of --

MR. EBERSOLE: Dave, you are referring to the case of the DC-10 baggage door, where it was given a lower priority for a fix than it should have been. One of these is a flight service directive, isn't it called, and the other is simply a recommendation to the manufacturer.

MR. LEWIS: Oh, if that is what you are talking about, these aren't categories of importance but categories of enforcement. There is what is called an air worthiness directive, an AD, which is mandatory on the people who receive it, either by the next flight or the next year or the next century, and then there is an advisory. An advisory circular is optional, but the air worthiness directive is a directive, if that is what you meant.

MR. OKRENT: I was really trying to see if they didn't have a way of categorizing safety questions. In other words, these things that are called unresolved safety issues in effect are things that introduce core safety questions and -- well, let me drop it.

MR. LEWIS: No, they may; and I'm sorry to have let you down, but it was bound to happen.

(Laughter.)

MR. SHEWMON: Before he goes on, would you tell me what you think the item is that you have been discussing for the last few minutes? I was not sure what it was, often as not.

MR. OKRENT: At Rancho Seco there was a very interesting transient on which there is a memorandum written by Bernero and Bausum to somebody, which indicate that this might have been quite serious, in fact. I think it raises the question of a class of events which are not, in fact, included in the WASH-1400 main contributors, which people continually use then to look at the next plant and the next plant and so forth. This is one that occurred that was, in my mind, a fairly close call, whatever that means, and Crystal River reiterated some of this.

I guess it is not clear to me that although the staff has tried to take some specific actions following Crystal River, that this constitutes a handle on the broader question. I would have expected to see this at least on the list for study; if not, in the resolved issues. I think it is an unresolved issue which OC

MR. KERR: Is that clear, Mr. Shemon?

MR. OKRENT: It is control system initiated transients. Not all of those, but there are certain of these which can be quite severe.

MR. LEWIS: Well, you know, I, of course, agree with that completely, with your comments. In fact, at some point it would be nice to know where we are on Crystal River itself in coping with what we learned six months ago.

MR. PLESSETT: Jesse, you have a wise word or two.

MR. EBERSOLE: I have feeling Crystal River and Rancho Seco really represent the bulk of our plants and the design of these systems, and to consolidate what happened to them, one could say quite easily that a single-channel failure lost instrumentation critical to shutdwon. So we are riding on a single-channel failure of critical instrumentation, so when we look at things like control rooms or instrumentation to follow the course of an accident, we should remember that these old plants -- and I don't think they are all that old, I think they would embody Sequoyah and probably North Anna -- that the bulk of the instrumentation is not -- and I am talking about indicating, recording and enunciation instrumentation -- not qualified to the usual safety standards that we demand of automatic circuitry.

Does anyone want to challenge that?

VOICE: Mr. Chairman, (inaudible) and I wish to make a brief comment.

VOICE: You are not being picked up by the recorder. Switch that on. It is kind of intermittent.

VOICE: I think the question of what is safety issues is important, and what we use to describe it appears to be (inaudible). It is used to inform Congress and the public what our problems are and what we do about them. I think it is not a problem for an agency to have what boils down to two sets of books on problems, one that would tell the Congress and the public about them, and the other for workmen.

So this problem has been, as you very well know, the subject of discussions for quite some time. The last year, January 8, 1979, specifically, at the request of the Commission, a number commented on the subject. At that time the points I made to my management as well as to the Commission itself were that to the question of unresolved issues, the improper use of probabilistic methods of risk assessment and improper use of WASH-1400, in particular, are very closely related.

As a matter of fact, it would be very appropriate to have the improper use of WASH-1400 listed as an unresolved safety issue. But it is not. Nonetheless, some proper and some improper use of WASH-1400 has been utilized in the definition in arrival at a list of so-called unresolved safety issues.

So the problem is not only semantics but substance in terms of having a realistic reflection of what our problems are and what we are doing about them, as well as having a program that will reflect the resolution.

Now, what I would like to request is to introduce into the record, with permission, if I may, that correspondence to my management and to the Commission that discussed the whole problem of definitions that I believe were arrived at somewhat arbitrarily, I believe, and which I do not believe reflect the congressional intent in the Energy Reorganization Act of 1974.

So, without getting into specific issues that I believe should appear on the list, and there are several I believe should appear on the list with the 133 or so issues to be kept in one set of books and in this formal official set of books, I wish to leave you with the request that you do read that discussion because I believe it to be very pertinent.

Thank you.

MR. PLESSETT: We will certainly read it if you supply it. Thank you.

Let me ask Paul -- Shewmon, that is. I am turning to you for guidance. I find the discussion very enriching, but there is only a certain amount of time left for me to be enriched in. I thought our problem was to see if we think that this list is an acceptable one; if not, what would we delete, what would we add. So, could you help us out, Paul? I don't think we need to have you discuss the other three. At that rate we would be here --

MR. SHEWMON: I was urging you to stay with the agenda earlier, and the agenda that I have says conclusions and recommendations by M. Plessett, and I was looking forward to that.

MR. PLESSETT: What I was going to say, with your permission, Paul, is that we don't have to have him read those things to us, the last three, because as he has gone through each one, we get into these tangential and centrifugal areas. If that is agreeable with you, we can go to the conclusions and recommendations.

MR. SHEWMON: Well, I would --

MR. PLESSETT: Do you want to go through those?

MR. SHEWMON: Well, it is probable that we have gone through these enough to raise the tangential questions that should have been raised, or issues, so let's go on. I think I would like to have the committee observe a moment of silence, maybe, while they read the seven that didn't make the list. D.C. power system reliability is one that sort of jumps out. I am not quite sure what reliance on ECCS means, but I am almost afraid to ask, in view of your saying you didn't want to get into tangential questions.

I guess there are seven things that weren't there for study. Of the ones I know something about, like the PWR pipe cracking or radiation effects on reactor vessel supports, I am not bothered by their being there. But it seems to me one of the things that should or could come out of this would be other things that we feel should be on the list, and the question of instrumentation or control system initiated transients is one of the things we might want to seriously think about.

If the rest would look at this and feel that whatever is in second place ought to stay there, then I think we could go on.

MR. PLESSETT: I don't know quite what reliance on ECCS means either. I was afraid to ask.

MR. SHEWMON: Well, why don't you so we all will know.

(Laughter.)

MR. PLESSETT: I don't know. Do you want to tell us in a word what that means?

MR. GEORGE: I think I can give it to you briefly. It is directly out of the Action Plan, 2A61. It is a concern that there have been a number of actuations of

ECCS for non-LOCA events. The concern is that if you have a large number of these and that if you consider a wear factor in the equipment, that perhaps you are degrading the reliability of ECCS and it may not be as reliable as you thought when you really do need it.

MR. PLESSETT: Well, that was your doing, this set-up where you are calling on ECCS all the time, not --

MR. SHEWMON: When you really get at that, also look at this checking weekly or whatever it is the diesels from cold standing start and having to be up to full power. That just puts my teeth on edge each time, that separate detail.

MR. PLESSETT: Well, you have told me what you meant by it, and I appreciate that, but --

MR. SHEWMON: D.C. power system reliability will wait till next year, is that it?

MR. EBERSOLE: No, I hear that next week we are going to get a report.

MR. RAY: I think I would like to hear why they think it is not an unresolved issue.

MR. EBERSOLE: It is an issue requiring further study, which meant to me that they didn't know enough to make it an issue or not.

MR. GEORGE: This one, D.C. power system reliability, is under further study because mainly the question was the study that has been performed by the contractor assumed a model D.C. power system. That model system was one that met what they interpreted as the minimum staff criteria for D.C. system. There was no comparison at the time of that system with existing installations or new installations coming down the line. So we are questioning whether you could apply the conclusions of the study, or at least the draft that was out, to existing systems.

Since then they have gone back and taken a look at it and determined that -- I believe there were about six operating plants that were compared to this model system, or vice-versa. It was determined that there were a number of them that were very close to the model system and therefore you could probably apply the conclusions.

At the time that we put this on further study, this further information wasn't available. I think at this time what we would probably say is that it looks like there probably is a significant impact on risk due to D.C. power reliability, or unreliability.

MR. PLESSETT: Jesse, do you concur with that?

MR. EBERSOLE: I have been following this every month.

MR. PLESSETT: Yes, I am aware.

MR. EBERSOLE: Originally the report was going to come out in June. I was told yesterday that it was due in about a week, isn't it?

MR. GEORGE: The draft that is being put together, I believe it is closer to two weeks, and the plan is to have a draft to be presented to the ACRS in September.

MR. EBERSOLE: Yes. See, we are that close. So I am just holding back waiting for that.

MR. PLESSETT: But do you think that we can consider it resolved safety issue?

MR. EBERSOLE: Oh, no. I consider we are at the stage now whether we can do an identification of whether it is or not.

MR. PLESSETT: I thought we could identify it as a --

MR. EBERSOLE: Oh, I would call it unresolved up to now.

MR. PLESSETT: Beg your pardon?

MR. EBERSOLE: I would call it an unresolved safety issue myself.

MR. PLESSETT: Yes, that was my impression. But why is it not on the list, just because a report is coming out?

MR. EBERSOLE: Well, I think the reason is the reliability, the probabilistic studies tended to indicate that it was all right; but my argument was that the input data was wrong and you ought to take another look at that, I think.

MR. GEORGE: I believe that is correct. My understanding was that the reliability data that is in there now indicates something to the effect that D.C. power reliability or unreliability is a contributor to risk for those events that rely on DC system.

MR. EBERSOLE: That is virtually everything.

MR. GEORGE: Yes.

MR. PLESSETT: You are not helping yourself with me, anyway, with that.

MR. GEORGE: Okay. But then the question we had was did that system really model the existing system. I think the information we are tending to get now is that it probably is. But let me add one other thing that I didn't before. The report that you are going to be seeing in September, the staff has included or is planning to include in that report the specific fixes that we feel need to be made. It is not a report that says yes, it is a problem, we need to do something about it. It is pretty clear from the information that is in the report what needs to be done, and I think it probably falls more in the category of one where you could probably consider it resolved within a couple of months if everyone agrees on the fixes.

MR. EBERSOLE: Yes. As a matter of fact, I think that is an ideal way to have it. I wish they were all that way.

MR. PLESSETT: Paul.

MR. SHEWMON: In that vein, the last issue on that previous list, seismic qualification of equipment and operatinag plant.

MR. PLESSETT: I'm sorry, Paul. I --

MR. SHEWMON: Seismic qualification of the equipment and operating plant. It seems to me that is one so close to knowing what needs to be done that you could alm st be accused of taking easy problems so you could get them off your list next year or something. Why is that one there?

MR. GEORGE: To my understanding, it is because it is -- well, it is obviously a problem with the electrical and mechanical components in the plant. The issue is how to develop methods to assess the adequacy of that equipment when you don't have very good data on the original qualification of the equipment, and can you develop analytical techniques that will do it for you or test methods that will do it for you. That is basically what the issue is, developing those methods of testing.

MR. SHEWMON: And that is one of the highest risk items you can think of this year, is that right?

MR. EBERSOLE: Wasn't that spiked by the finding that about a half-dozen plants didn't have a recognized means of shutting down because they couldn't do bleed/feed and they didn't have an aux feed system? Is that not that?

MR. GEORGE: No, this is a different issue.

MR. OKRENT: A lot of the old plants -- well, some of them were just designed to use DC, and some of the next group may not have had their electrical systems all that well qualified, and then certain mechanical system aspects.

MR. EBERSOLE: So it is broader.

MR. OKRENT: Well, where do you take on the question of the adequacy of the single-failure criterion, in your opinion? Is it a nonresolved safety issue or what?

MR. GEORGE: The way we addressed that issue is that in the TMI action plan, that question is listed as one of the questions that needs to be addressed once we get through IREP and to a certain extent NREP and we look at the plants and we decide just how we need to define the criteria. Do we need to have some reliability goals established, or multiple failures for certain systems.

It is a question that needs to be addressed after IREP, and what we considered was that IREP was really an investigatory type program that may identify outliers, identify problems, and then those concerns may or may not be unresolved safety issues, but IREP itself is -- we didn't feel it was an unresolved safety issue.

MR. OKRENT: I didn't say IREP. I said the single -- the adequacy of the single failure criteria. Do you consider that to be an unresolved safety issue? Was it on your list?

MR. PLESSET: In our paper, we didn't consider it as an unresolved safety issue.

MR. OKRENT: Okay.

MR. PLESSET: Do you think it should be?

MR. OKRENT: Yes.

MR. PLESSET: Okay, that is -- I like something specific.

MR. OKRENT: A different question. I notice that you considered at some point that the ACRS has made recommendations that the staff look more at design errors and whether they were -- had an adequate system, let's say, with -- not that you can catch them all, but to improve the situation in that regard, and that didn't make your list of things to be studied, and somewhere I think it was concluded that this is not probably a big contributor to risk. I think I read that somewhere in these documents.

MR. GEORGE: Perhaps the one you are referring to was design, check, and audit of balance of plant equipment, and I think the way the ACRS recommendation was written was one along the lines of to verify that the equipment satisfies the design -- the equipment as installed satisfies the design intent. Is that the one you are referring to?

MR. OKRENT: That is a way -- it appears in more than one place. You referenced the LER report. It is also in the review of the regulatory process or whatever it is that came out in December. There was a recommendation about design errors, and I am trying to understand on what basis this was screened out. Was it -- in hand, or it is not important to risk, or it is a policy decision, or what?

MR. GEORGE: As I recall, that one was -- it wasn't screened out in the initial screening. It was considered an issue as potentially -- it was generic, and potentially should be considered. When we evaluated it, I guess the collective judgment was that it would not be a significant improvement in risk.

MR. OKRENT: Now, it says, for example, the issues not recommended for designation as an unresolved -- transients or safety system challenges result more frequently from operator errors and random component failures rather than from deviations from the intended plant design.

Now, I must say that doesn't tell me that design errors are not a possible contributor to risk, and let's take the one that occurred at Trojan, where there was a problem wi h, if I recall orr tly, how the control room was connected for earthquakes. That would never show as initiating a transient, you know. It just can't compete with almost any other thing you can name in the reactor, but nevertheless, given an earthquake, it could have a very significant effect on risk.

MR. GEORGE: Yes, I think that is a different issue than what we are trying to address here. That is a much broader one. You are talking about there going back and taking a look at even safety related structures, components, systems, and --

MR. OKRENT: Well, let's look at the first word. For verification of balance of plan as built, configuration satisfies the design intent. It seems to me that would fall right in those words.

MR. GEORGE: Well, the balance of plant that we are referring to there would be what are nominally referred to as the non-safety systems, powered conversion systems.

MR. OKRENT: But you have a general recommendation to ACRS to look at design errors. I mean, you take a limited look and then say, because I have limited the look, there is no risk, and therefore -- you know, and rule out the design error. So you haven't addressed the question. So I am still trying to understand whether the design error question is one that you have ruled out because there is no important risk associated, or what.

Let me read one other thing. You say, "Additionally, other ongoing studies, IREP and systems interactions will identify potential potential adverse impacts from balance of plant equipment."

Well, I agree. There are certain kinds of design errors you will pick up that way, but not the type at Trojan, again.

MR. GEORGE: It was not the intent I don't think in the issue we are discussing here to cover that concern, you know, the concern of all design deficiencies in safety systems. We were addressing this one specific issue that was out of the LER NUREG report.

MR. OKRENT: Yes.

MR. GEORGE: That is where this one is, and that is what these words really apply to, is strictly that concern. You are saying again --

MR. OKRENT: Well, then, where did you dispose of the one that was in the review of the regulatory process and so forth?

MR. GEORGE: I guess I don't recall that particular recommendation offhand. I will see if I can locate it.

MR. OKRENT: There were a lot of plants shut down for many, many months just because of design error. We have to assume you considered it -- I think the -- I don't know what -- well, it was even stronger than an unresolved safety issue. It was strong enough that the plants have to be shut down while they were being checked for design errors.

MR. EBERSOLE: There were a lot of them. The -- design errors in the Brown's Ferry recent case was a case in point.

MR. PLESSET: Well, that made me feel very good about Dave bringing up the single failure criterion as well.

MR. OKRENT: I think we will let Dr. Moeller make a comment. He has been waiting.

MR. MOELLER: In just looking at this in an overall way and discussing items that -- other items that presumably have been considered, it seems to me that any matter that is under rulemaking would be in essence an unresolved safety issue.

MR. OKRENT: If it involves a safety question.

MR. MOELLER: Right, if it involves a safety question.

MR. SCHROEDER: I guess one of my difficulties is that I am not quite sure that if I were sitting down to write an action plan for that issue, what it was that we would do. Yes, we ought to do better in uncovering design errors and in preventing them, but I am not quite sure what you do with the unresolved safety issue on that.

MR. OKRENT: No, you are leading me right to a suspicion I had, namely, that these particular unresolved safety issues are things where you saw an approach -- well, in one case I am not sure they see resolution, but --

MR. SHEWMON: But let's go back to, say, it seems to me control system initiated transients is something that has come up here from a couple of different members. I am not sure how you would write an action plan for that, but would you argue that you shouldn't look at it because we shouldn't call it our safety issue, because we don't see our way through it?

You don't really want to do that, do you?

MR. SCHROEDER: No, but I can see writing a plan for that one a little more clearly. What you would have in mind is what new requirements should we establish both in the terms of requirements on design and requirements on our review that would get at the problem of addressing control system initiated transients and accidents.

MR. SHEWMON: Whereas in the other one you are looking for things nobody has ever found before, and you aren't sure how to do that?

MR. SCHROEDER: Yes, how do you --

MR. SHEWMON: I sympathize.

MR. OKRENT: It is not an easy question, the question of looking at design errors, but if you don't give it priority --

MR. SHEWMON: Professor Okrent doesn't like to work on easy problems. That is one of his redeeming virtues.

MR. OKRENT: I think you are unable to say it is unimportant, so it meets that test. In other words, there have been significant failures, and if you ask around, I think you will find a lot more from the past, and Mr. Ebersole is able to recount, you know, 20 at the drop of a hat.

MR. PLESSETT: Let me get back to Paul's point that he raised a little while ago about the seven issues that didn't make the list. I don't feel too bad about leaving those off, and Jesse has kind of reinforced me on Number 4.

MR. OKRENT: Excuse me. In my opinion --

MR. PLESSETT: You don't feel that way?

MR. OKRENT: -- the DC power is unresolved.

MR. PLESSETT: Okay. I thought Jesse was satisfied.

MR. EBERSOLE: Well, I am saying it appears that next month I will be able to make some determination from what you have done as to what it is. I don't know --

MR. OKRENT: Well, if you want to keep that in, I misunderstood. I thought you were more satisfied --

MR. EBERSOLE: There is only a month. There is only a month here, and I am not nervous for a month, having seen it for 15 years.

MR. PLESSET: Well, I was just going on that sentiment, as I read it.

MR. OKRENT: I think -- to me, that meets their criteria quite well, and I don't understand the term "requiring further study," and having seen that resolved in 1973, I don't find, myself, the statement that there will be a report coming out in two or four weeks in any way constituting resolution.

MR. PLESSETT: Well, I would feel not at all distressed tdo have it icluded. I think it is important, and if that is the sentiment of the committee, I would say fine. But I think none of the others bother me particularly as not being on there now.

MR. EBERSOLE: Let me explain a little bit about --

MR. EBERSOLE: Jesse, do you want to explain your stand, because --

MR. EBERSOLE: Yes, I want to explain my stand.

MR. PLESSETT: Okay.

MR. EBERSOLE: The DC power system reliability was sort of an open issue in the background of the thesis that statistical probabilistic analyses showed by and large that it looked like it was in the lower class of probability of events. The consequences were indeterminant, if you really lost it.

There were some statements to the effect that a mode of recovery was possible from this sort of event which didn't hold any water, and there were reasons to believe that the input to the probabilistic analysis were not as good as they should have been or had to be, so it merely required that one take a new and harder look in greater detail at this, and at the prospects of non-recovery if you got into this condition, and if you found that to be non-acceptable, the fix for it is almost evident.

It is not a hard thing to do. It is no big R&D program. You just upgraded the system in a straightforward fashion.

Now, I really don't think that in is the nature of an unresolved safety issue so much as a -- as a question of inadequate conservatism, or something like this, in a known process.

MR. PLESSETT: You convince Okrent, and I am with you. Otherwise, I think it gets added to the list.

MR. EBERSOLE: Right now, I would have to say it is an unresolved safety issue at this point, when I know no more. In another month from now, I don't know.

MR. PLESSETT: I understand your point. Dave is a little more pessimistic than you are, I think, about what is going to happen in the next month.

MR. EBERSOLE: It is a little bit more manageable than most of those things.

MR. SHEWMON: Do you agree, Dave, that the fix is well defined and easy enough to implement if we decide indeed it is needed, which seems to be the way to summarize what Jesse said?

MR. PLESSETT: Right. That is a good question for Dave.

MR. OKRENT: My guess is that it is one of the easier things to fix. That would be my guess. But as you heard earlier, I don't really accept the staff's criteria that if we could decide it in six months, therefore that is an adequate reason for not putting it on our list, because "could" -- you know --

MR. PLESSETT: Doesn't mean that it will.

MR. OKRENT: If wishes were horses, or something.

MR. PLESSETT: Yes, okay. Well, I certainly don't feel strongly about not putting it on the list.

MR. EBERSOLE: A month from now, I may want to put it on the list for sure.

MR. PLESSETT: All right. Well, but we are talking about relatively immediate.

MR. OKRENT: It is either this month or a year from now, is the way I understand it.

MR. PLESSETT: That's right. Paul, you are the subcommittee chairman. How do you feel about this? I would just as soon see it on the list myself.

MR. SHEWMON: Well, we are going to write a letter? Is that --

MR. PLESSETT: Yes. We have to send a communication up from this meeting as to things that we would like to really see on the list.

MR. SHEWMON: Well, it seems to me part of the things that the letter will say in its first draft --

MR. PLESSETT: Which you are writing, I understand.

MR. SHEWMON: Yes. That is why I say, what we say in the first draft. I can't say what it will say later. Is, it will sort of suggest --

MR. PLESSETT: Nobody can.

MR. SHEWMON: -- that maybe the staff have not faced the hardest problems they can find on the block, but it seems to me that that one is -- gee, that is going to turn around and say, here is one your batting average can be great, because there is one you can knock off for sure before next year.

MR. PLESSETT: But Dave's point is that they might not. Being able to do it might make it something they will just put aside.

MR. SHEWMON: No challenge to it.

MR. PLESSETT: Yes, that's right, and he would just as soon see them knock it off for good.

MR. SHEWMON: I will put a sentence in, and we will see what happens to it when the committee decides --

MR. PLESSETT: I think Dave had a very good point about the single failure criterion. I think we would all agree with that. Is there anybody who is reluctant to add that to Paul's letter?

MR. EBERSOLE: I am not reluctant. I am enthusiastic.

MR. SHEWMON: Do we want to say anything about siting?

MR. PLESSETT: Well, let's get this one out of the way first.

MR. KERR: Well, I wish I know more than I know

now about what one accomplishes by moving something and making it an unresolved safety issue. If I thought that would hel solve it I would feel differently, but it isn't clear to me that --

MR. PLESSETT: Well, I think we have to believe, Bill, we have to have some belief that it does do some extra good to promote it.

MR. SHEWMON: Bill, what it does is to put it out in the public and the staff has to explain a year from now, when they write their next letter to Congress what it is they have done or why they haven't, and that tends to establish the priorities of what they will work on.

MR. GEORGE: Can I clarify one thing? This list they were putting together now and reporting to Congress is to satisfy reporting requirements for the 1979 annual report. It was obviously not completed at the time of the annual report, so we included a statement in there that we would be doing it at a later date, and the next list will be in the 1980 annual report, which, you know, within the next couple of months we have to start working on that.

MR. EBERSOLE: It seems to me there has been a 180-degree switch here from the characterization of what is an unresolved safety issue as it was and what it is now. It used to be, if it were categorized as an unresolved safety issue, that would guarantee that it would get no attention indefinitely.

(General laughter.)

MR. EBERSOLE: Now, the table is turned.

MR. SHEWMON: Don't answer that.

MR. PLESSETT: Don't say any more. I think we were making progress until you helped us out.

(General laughter.)

MR. SHEWMON: Bill, as I have inquired around, that seems to be one of the virtues, and I sincerely believe it is, you know, it will be one of their high priority items now because they would be embarrassed if it doesn't move. And so partly it is a matter of, do we think that this should be their highest priority.

MR. KERR: I think it is an important issue. I have not looked at all of the items in the action plan in detail, for example, so that I can categorize this. I don't object to putting it on the list, but I have an uneasy feeling, and I am not quite sure what I am doing to it when I put it on that list.

MR. EBERSOLE: I guess we need a statement of the relative rapidity of action and solution, depending on whether it is on the unresolved list or on the action plan, and I don't know what that is. I don't know which is the fastest or the slowest. I can't quite tell.

MR. PLESSET: Well, maybe somebody can.

VOICE: It should be both places.

MR. SCHROEDER: I am not sure there is a single answer to that.

MR. PLESSETT: There is your answer. There is no answer.

MR. SCHROEDER: The action plan, the agency has said, we are going to follow that action plan. It has schedules in for resolution of these items. If we take one of those items and also designate it an unresolved safety issue, there will be perhaps more detailed schedules developed because of its character.

MR. PLESSETT: It gets kind of a C-A category somewhat.

MR. SCHROEDER: If you give a dedicated task manager to it, which may accomplish more than just having it the responsibility of one of the branches -- it may not; the branch might do the same thing -- it does get, as has been pointed out, the highlight of having to report our progress to Congress.

MR. OKRENT: So it is better.

MR. PLESSETT: Yes. Well, I think it is getting clearer to me what is going on, and Paul, I think, has a pretty good idea of what is going to go into this draft, and I would like to suggest that if anybody has any really important thought, that he communicate it to Paul, and if he likes it he might accept it.

And with that, can we -- Yes, Dave? One last --

MR. OKRENT: It seems to me the Commission needs some method of designating problems so that we don't have to have it in the action plan, or it doesn't have to be specified as an unresolved safety issue in order for it to receive resources and priority, and there can be important problems or potentially important problems where you are not able to say yes, it does represent a significant risk, or yes, there is a significant improvement to be made here.

Maybe that is going to turn out to be the case, and you had better find out if it is the case, and therefore it requires a priority, and that right now that seems to be a kind of question that could well fall by the wayside except by chance.

MR. PLESSETT: Well, I think that some sentiments along that line are significantly valuable, and should be somehow -- and Paul is indicating agreement. Jesse, did you --

MR. EBERSOLE: Do I understand now the single failure criterion is going to get on there?

MR. PLESSETT: Yes.

MR. EBERSOLE: Great. Okay.

MR. PLESSETT: As I understand it.

Well, with that --

VOICE: Do you mean the 1979 list or the 1980 list?

MR. PLESSETT: I don't care. It is the list we are fussing with. 1979, I guess.

MR. SHEWMON: We are responding on what we think of what he refers to as the 1979 list.

MR. PLESSETT: Yes. It is the one that is going in --

MR. SHEWMON: It was really deferred in 1979, and is clearly coming out in 1980, within --

MR. PLESSETT: And it is late already.

Well, can we have a well-earned ten-minute recess?

(Whereupon, a brief recess was taken.)

MR. PLESSETT: I think we're going to have some information from the NRC staff. First with regard to the Trojan Nuclear Plant. Right? Would you want to take over?

MR. CLARK: My name is Robert Clark. I'm Chief of the Operating Reactor Branch Number 3. The project manager for Trojan is unable to be with us today since he's taking some annual leave. But we're here on a goodwill mission. We'd like to share with you some of the information we have achieved and acquired during the last few months in the continuation of the Trojan plant saga and we've prepared a two-part presentation. I would suggest that perhaps if you could hold your questions till the individual has completed his presentation it might provide us the opportunity to give you the information that is available and would expedite the presentation somewhat.

We will first venture into the area of volcanism and Mr. Harold LeFever from the Geoscience Technical Branch will make the presentation in this area.

MR. LEFEVER: I have several handouts here I'd like to distribute first and then we can go from there. I have about five viewgraphs I would like to show you here. They all relate to the Trojan Nuclear Plant and the Mt. St. Helens vicinity. I'll wait until the distribution has been made of the handout before I start.

MR. MUELLER: I might ask if the Cascades you're talking about here are the same ones that Mr. Ebersole has been referring to.

MR. LEFEVER: As you can notice on the viewgraph I have here, Mt. St. Helens, which is one of a number of volcanos in Washington State and Oregon State extending down to California. Mt. St. Helens is approximately 33 miles to the northeast of the Trojan Nuclear Plant. It's the closest of the Cascade volcanos to the Trojan Plant. And see we have to the northeast Mt. Rainier, to the east Mt. Adams, and to the southeast, Mt. Hood.

There have been four eruptions generated from Mt. St. Helens since May 18. These are the May 18 event, the May 25, June 12 and July 22. There have been some intermediate events which consisted essentially of very minor ash emission. The four larger events that I mentioned have resulted in varying degrees of ash deposition throughout Washington State into Oregon -- for one of the smaller events. And, of course, the May 18 event is the one that precipitated the volcanic ash and that, of course, covered a much wider area, including fallout in the east and it is continuing in the atmosphere.

Now as far as the Trojan Nuclear Plant is concerned two events -- the May 18 and the May 25 events -- have resulted in some type of volcanic phenomena at least reaching the Trojan Plant. Ashfall has occurred on April 29. There was simply a trace of ash. On May 25 there was approximately a 16th to 1/8th inch of ash fallen in the form of a muddy rain. It happened to be raining -- a heavy mist -- at the time. It's very difficult to measure, but at any rate it was quite mild.

As far as the other geologic or volcanologic phenomena, there was a large mud flow associated with the May 18 event that traveled -- let me show you another slide here that will give you the general route of the mud flow. Here's Mt. St. Helens. Here's Trojan. As I mentioned before approximately 33 miles between the Trojan Plant and Mt. St. Helens.

The darkened area is the blast area resulting from the May 18 eruption. As you can see, it essentially involved the northern half of Mt. St. Helens. There are actually three areas here. I'll show you another slide. It will give you an idea of the area of devastation, shall we say. We have the main area which is essentially a mud flow, debris flow -- trees and various assorted types of materials which were carried down the North Fork of the Toutle River to the Cowlitz River, which is the west and then emerging in the Columbia, which is approximately -- the Cowlitz-Columbia juncture is five to six miles from the downstream of the Trojan Plant.

As you can see here, the area of devastation or affected by the blast has essentially a radius of fifteen miles, confined essentially in the northern area extending a bit to the east as well as to the west.

MR. MARK: Could you say what's the mechanism for killing trees without knocking them over?

MR. LE FEVER: Heat. But not enough heat to char the trees. But it removed much of the vegetation and the leaves without knocking them down. The air blast flattened the area here, as you can see, but here was just the burning and heat associated with the blast.

MR. LEWIS: Very often trees recover from that, I assume.

MR. MARK: Well, there could have been mud around the roots and I was wondering which it was.

MR. LE FEVER: There is some vegetation reappearing, by the way, within this area. Some plants, flowers and things were not completely covered with muddy ash and apparently they have survived to some degree. So there is some life in the area. But generally it's an area of complete devastation.

The events of May or June 11, and July 22 was no ashfall or other phenomena affecting the Trojan Plant. All the -- the two phenomena that have affected the plant -- or at least reached the plant, shall we say -- were both considered during our review of the plant. The last occasion was 1978, when we wrote an affidavit dealing with volcanologic effects on the Trojan Nuclear Plant. As I say, the ashfall has been minimal -- an eighth of an inch. Mud flow did reach the plant. It resulted in a number of feet of deposition in the Columbia River channel. We have other staff members here who can dwell or discuss with you the mud flow itself as far as it relates to the operation of Trojan. We also have other staff members who can address the ash fall and what it means.

MR. SHEWMON: My reaction is that it's nice it did not go in the southwest quadrant. Given that it's woken up, can you now be reasonably sure that within the next forty years it's not likely to take off in another direction? Or is the probability of another significant blast in a different direction more or less probable, given what we've seen in the last year?

MR. LE FEVER: It's probably less likely, but it still could occur at any time.

MR. EBERSOLE: Isn't the issue mainly not the direction of the blast but which way the wind's blowing?

MR. LE FEVER: The wind certainly has much to do with the ashfall.

MR. SHEWMON: If arbitrarily wind didn't have anything to do with --

MR. EBERSOLE: But that's only fifteen miles and he's thirty. But if I take this case -- if I take the worst ashfall and I orient the wind in the most pessimistic way, toward Trojan, how would it have fared?

MR. LE FEVER: In the case of the May 18 event, which this is a depiction of, at a distance comparable to what the Trojan Plant is -- which is 33 miles. The plume of this, by the way, blew to the east, northeast.

If one would rotate that plume and put it directly over the Trojan Plant the ash fallout would have been on the order of less than a half an inch.

MR. EBERSOLE: Are the diesel intakes rigged for that?

MR. LE FEVER: They might. We have others here who can address that. Bob Clark, I believe, can respond to you on that matter.

MR. CLARK: If you want to go into that area, I can develop a little bit of the work done the staff to assure ourselves that conditions at the plant were accommodating the hazard which might be posed by the volanc. We visited the plant as a group in June and not only toured the site but, as one of the members of the committee know, we took a helicopter flight to the mud flows and some of the other areas of devastation.

One of the things that was important to us was the intake -- air intakes -- and what provisions we were making there and we had Jack Donahue of the staff spend the whole day at the site looking at the air intakes and the filters that they were putting in place as they received warning of ashfall.

Now for the most part these filters were installed only in what we would call "vital areas" such as the diesel generator air intake, which has a two-fold purpose. It provides combustion air for the diesel and also is room cooling. And it's located in a very favorable position from the standpoint that they have a large bay that the air comes into and it makes a couple of right angle turns before it reaches the filter area. But they did have roughing filters and other filters in place and inside the room was reasonably clean.

Trojan is not the cleanest plant which we ever visit, but comparatively speaking, from the ash standpoint, it was clean.

MR. EBERSOLE: Bob, what I'm really asking is if you extrapolate this accident to the worst meteorology, and then add some factor to the ash release from the explosion, would Trojan have made it?

MR. CLARK: From everything that we have seen we believe Trojan would have -- if it had been operating. It was shut down, of course, which was fortunate in itself. It gave more leisure to consider the consequences. But from our visit and the precautions that were being taken by the licensee, I believe that the plant could have continued to operate and certainly could have shut down safely.

MR. LEWIS: Just following up and asking a question in a more ignorant way than my learned colleague on my right, if it were to shut down the principal danger to the plant would be from fly ash, ash in the air, or from being gradually buried? Where do you begin to run into trouble?

MR. CLARK: Well certainly being buried is not desirable. I think the principal concern relates to rotating machinery and what effect the ash might have on this, the effect that ash might have on personnel who are required to remain at their stations to operate the plant. And seemingly the provisions that we have seen certainly would adequately protect them from that standpoint.

The other concern we had related to the supply of water to the service water system -- that they would be able to remove the decay heat successfully. There's quite a margin related to this, even under these conditions. The intake structure was not endangered as far as losing section, and they have a large capacity basic for their cooling tower which would also supply water some time if they had to take water out for other cooling purposes.

MR. EBERSOLE: Would the ash have got in the water under a more severe case and clogged the filters and journals and seals?

MR. CLARK: Well, because the Columbia River has quite a bit of sediment in it, this plant has already taken precautions. They have some hydroclones which remove sediment from the Columbia River water to protect their bearings and so probably they're in a good position from that standpoint.

MR. EBERSOLE: You said that they would find it easier to shut down than to run. It might have thought it would be easier to keep running because it's only after they shut down -- say with a loss of power because of insulation failures -- that they demand that the diesels work, which is probably the most vulnerable part of the whole complex to dust, isn't it?

MR. CLARK: Well, I don't know that we have determined what's most vulnerable. All these things need protection, Mr. Ebersole,and I think that it's a matter of judgment as to whether you want to continue to maintain the amount of heat that you have to discharge under power operation or whether you want to go to a position where you're under decay heat conditions.

But certainly in either case, the vulnerability of the transmission system was examined carefully. The licensee had established on a nearby mountainside early in the history of this event mock-up transmission lines that they were studying the effect of ash as they determined that. And the also had consultants and were gathering information not only from the transmission systems nearby but were investigating the industries nearby and the effect that the ash was having in their rotating equipment. And so they gathered quite a bit of information.

They have established a task force which assembles when they have early warning from either the Forest Services, the Weather Bureau, the USGS, that an event is going to happen or has happened. And these people then make the judgments which are necessary to protect the plant.

MR. EBERSOLE: Did they conclude they will lose high lines from dust?

MR. CLARK: Seemingly the biggest part is under conditions where you have ash and rain. There's some flashover on the insulators from the mud that accumulates on the insulators, but they did not lose any transmission during this.

MR. EBERSOLE: I know they didn't, but I'm speaking about the hypothetically worst case.

MR. MATHIS: They did lose some lines on low voltage, but that was minor. But high voltage transmission lines withstood them with absolutely no problem.

MR. CLARK: Generally the geometry -- the angle of the insulator -- has a big effect and evidently on the high lines the insulators have a more favorable pitch than the low voltage lines.

MR. EBERSOLE: Does recovery under these circumstances involve a repeated changeout of filters during the course of the event? Is the duration time in which they have to go in and stop machinery and load in new filters? Or is it a wet scrubber or what?

MR. CLARK: Well the filters that they were putting in place were roughing filters. They're certainly not HEPAs. They're basically what you would see as removing particles ten microns and larger with high efficiency. The smaller particles with somewhat less efficient.

They do have, in many cases, changes of direction which precipitate the particles out by settling. They did not have any problem as it related to overloading the filters. Now we did suggest to them that they examine an installation in the filters of differential pressure measurement devices so they could monitor the condition of the filters.

In a couple of other instances they have them located with quite of a bit of exposure to the weather and they are looking into providing a more permanent type installation with some sheltering for these filters.

MR. CARSON: You gave us lots of assurance here that they had filters and precautionary measures and they could have withstood a ninety or a 180 degree turn. Were all of those filters and precautionary measures in place before the first eruption, or are you talking about some things that have been added since that first?

MR. CLARK: Well, that's certainly an appropriate question and Bob Jackson, who is here from the Geotechnical Branch began his monitoring of this in March, I believe. The licensee also began their research and their sensitivity in March, so there were precautions -- such things they had been prepared for because of the -- I guess the most recent hearing on the spent fuel pool where this type of event was a significant issue and they had to address the things that they would do.

They had, for instance, snowblowers on the site to blow ash off the roof if necessary. They had the filters to put in place. So it was not a matter of -- as soon as they had the warning they went into their protective mode.

MR. CARSON: So I guess the answer to my question is they had all these things installed or taken care of ahead of time.

MR. CLARK: They had them available at the site and when the incidents became of significance they did install them. Yes. They are not filters which they normally operate with in place at all times.

MR. EBERSOLE: Did the station air system have any problems?

MR. CLARK: Well, as Harold has indicated to you, the effects of the volcano on Trojan have been minimal.

MR. EBERSOLE: I guess I should have said, would the station air system have had problems without prior arrangement?

MR. CLARK: You mean habitable --

MR. EBERSOLE: No. I'm talking about the uptake of the station control air systems. That's a highly filtered system.

MR. CLARK: Oh, I see. You mean the control air systems. I'm not sure that we investigated that in any depth. We looked at the areas where these systems take their air from and all of them have roughing filters on. But whether or not where were small particles that would interfere with the pneumatic control system we'd have to look into that in more depth.

MR. MATHIS: Well, Bob, we went down through the room that had the air compressers in it. I don't remember just the configuration, but it was nice and clean -- just like the auxiliary feedwater setup and the emergency diesels. I don't remember. We roamed around. Inside it was nice and clean. Of course it was dirty as the devil on the roof.

MR. LEWIS: Of course, Jesse's question is what would it have been had the thing turned 180 degrees?

MR. CLARK: Certainly there was no effect from the volcano under the existing conditions. What they would have been under a 180 degree reversal of path -- I guess that we'd have to examine that now.

One thing that we should advise the Committee that we have asked some additional questions to be responded to by the licensee and one of the questions is directed toward examining the facility and what consequences would have resulted from having this event directed toward the plant.

MR. LEWIS: Excuse me, I thought you first answer to Mr. Ebersole's question was that if the event had gone 180 degrees in a different direction there would have been no problem. And I understand you now saying you're not sure of that.

MR. CLARK: We want to document that. But from what we have seen, we don't believe there would be any problem from a half-inch of ashfall.

MR. EBERSOLE: Well, do you now intend to establich new criteria for ashfall rate and duration thereof?

MR. CLARK: That, of course, is not one of the responsibilities of the Operating Reactor Branch, but certainly it's something that we'll have to give consideration and consultation with the technical branches. Bob Jackson can comment a little bit on that if you'd like to --

MR. EBERSOLE: There's some other plants in this category out there or some new ones coming on like this, aren't there?

MR. CLARK: I'm sorry, I didn't hear you.

MR. EBERSOLE: There's some new plants coming on that are going to have to face the same sort of problem.

MR. JACKSON: Yes, the Trojan Plant was the first plant which addressed the ashfall problem. That actually becomes worse for Pebble Springs and the Hanford facility, which has four or five plants scheduled to be operating there at some time in the future. It is in the area of the heaviest ashfall and reports were that there were problems in operating throw rigs and things like that that were shut down within a day.

I just wanted to mention at the Trojan facility there was no specific design ashfall -- design basis ashfall -- was described that we know of. But the USGS reviewed the site almost in total at the original CP stage and the USGS specified specific eruption events. And the event was -- the assumed design event for Mt. St. Helens was a Mazama-type of explosion, which is the Crater Lake explosion, superimposed on Mt. St. Helens and assumed to occur there.

Therefore, they had to take in specific considerations of ashfall. I don't believe any specific amounts were assigned in terms of how many inches it's designed for. Strong weight was given at the time in this fuel pool hearing to prevailing wind direction and other geologic arguments. For instances, the U.S. Geological Survey issued a report by Mollenau and Crandall on previous eruptions at the Mt. St. Helens site. In fact, they predicted -- estimated I guess would be a better word -- that the major ash flows and mud flows would be down the Toutle River. In fact if you read the report you'd see that that was where the hazardous area would be.

The also indicated that the area to the north -- Spirit Lake area -- was an area of previous ash flows and mud flows during the previous 4,500 years. So really, in essence -- although they didn't come out and state it -- they were saying the higher probability of your danger is to the north.

Coupled with that is the prevailing wind direction, which I think over the 45 year geologic history they were able to interpret blew to the east 95 percent of the time. I have a copy of their report if you're interested.

MR. EBERSOLE: As I recall Pebble Springs does have an ashfall rate and --

MR. JACKSON: It does. I don't recall what it is. I remember it's a very large amount -- like 24 inches.

MR. LA FEVER: I can verify that. It's 8-1/2 inches in a 24-hour period.

MR. EBERSOLE: Is that still valid? Would you show where Pebble Springs and (inaudible) are?

MR. LA FEVER: Pebble Springs would be down in this area. As you can see, none of the ash has fallen there -- at least as far as this preliminary plot goes.

MR. EBERSOLE: Well, none of it has. I guess it could fall.

MR. LA FEVER: Yes, it could. This could just as well have shifted down here.

MR. EBERSOLE: Where did you say Hanford was?

MR. LA FEVER: Hanford is in this -- here's Richland and Hanford -- the Wolfs plants are in this area. As you can see -- these contours are in millimeters -- this is a ten millimeter contour and this is a trace. And there are inconsistencies even in this. In other words, you can have ten millimeters here and you could have none here and you could have none here. It's not a uniform blanket of ash by any means.

MR. SHEWMON: Why is there that bullseye in the eastern half of the state?

MR. LA FEVER: That's Rittsville.

MR. SHEWMON: Why is it?

MR. LA FEVER: This is a concentration of ash. In other words it was thickest here as far as this particular plume is --

MR. SHEWMON: My question was why? It was just sort of the most probable trajectory of what got ejected, or what?

MR. LA FEVER: Meteorological conditions dictated that velocity of the wind and particle size all combined to deposit the particular material here. You can see we also have a plume back in this area which would be a coarser type of material. And of course the wind would carry the fines a bit further and if conditions -- meteorological -- were favorable at this particular time to drop a good portion of the plume in the vicinity of Rittsville.

MR. SHEWMON: It just is striking. I would have expected it to just sort of trickle off more.

MR. LA FEVER: No, it's very sporadic.

MR. SHEWMON: How much was there at Rittsville?

MR. LA FEVER: Well, this indicates 70 millimeters, but I'm not sure that this is certainly a maximum. I think I've seen a -- this, by the way, is a map that has been provided to us by the licensee. I have seen another map prepared by the Washington State Geological

Survey and they show two to three inches in the Ritsville area. There are other areas, I'm sure that might have three or four. It's far from complete. But it does give you a general picture of the direction and gross amounts of ash involved as far as thickness goes.

MR. EBERSOLE: In your fallout estimates for the other plants is there a rate figure as well as a total depth?

MR. LA FEVER: I was not involved with those plants. I'm not sure what we do have. We are currently reviewing what to do and this will certainly be factored into.

MR. EBERSOLE: Well, it would seems like there would have to be a rate number if you're going to change filters.

MR. LA FEVER: Yes. But as I say I think from what we've learned on this particular event the numbers that we've used as far as Pebble Springs go would be very conservative -- certainly as far as this event. And as far as Mt. St. Helens goes, there have been comparable events with this, some of them two to three times as large but I'm certain that the eight inches -- 8-1/2 inches -- of dry ash that we have considered at Pebble Springs would be more than adequate.

By the way, there is something on the order of one cubic kilometer of material involved in this particular eruption, but how much of it ended up in the mud flow and how much ended up as ash the U.S. Geological Survey is still working on those particular numbers.

MR. MARK: Could you help me? My Greek is either nonexistent or very rusty. What is that word "Tephra"? And why is it found advantageous to use it instead of something comprehensible?

MR. LA FEVER: Well, I don't know where the name originated, but it's a term that covers the airborne volcanic debris.

MR. MARK: And it refers to the debris and not to the event?

MR. LA FEVER: That's correct -- the ash and the larger size materials that would be erupted.

MR. MARK: Pac? It's the same etomology as pachyderm.

MR. SHEWMON: I thought you were going to ask about isopack at the other end. Well, I could almost mate that one.

MR. CLARK: Mr. Chairman, if it's your pleasure, we would proceed on to the second presentation.

MR. LEWIS: May I just ask one question? And that is that, having decided that if there were any threat to the plant it would be from airborne stuff. The first of two questions. One is it is a fact, isn't it, that under normal conditions no air enters the plant except through a filter?

MR. CLARK: In sensitive areas that's correct.

MR. LEWIS: And sensitive areas are defined?

MR. CLARK: Well, generally defined -- if you take the control room, for instance, it's defined to maintain habitability under conditions of chlorine release, for instance. And also radioactivity.

In other areas of the plant they are designed to produce an air which is free of dust. And protec the rotating machinery.

MR. LEWIS: Okay, fine. So normally the answer is yes.

Okay, then, the other question is that the filters involved. One envisages other kinds of accidents beside volcanos. An oil truck ov turns and produces sooty smoke for twelve hours right next to the air intake -- that sort of thing. These filters are good enough to deal with that kind of junk?

MR. CLARK: Certainly that kind of particulate matters would be usually filtered by these filters, yes. Generally speaking, that type of smoke is sort of chained and the particles are rather large.

MR. LEWIS: The reason for this line of concern, which is not real concern, I'm convinced, is that there was an accident in Ohio not involving a nuclear power plant five, six, eight, ten years ago -- I've forgotten exactly when -- that involved a fire at a factory that was making carbon composites and everything electrical in the town stopped at the time of the fire and I just worry about that kind of contamination of the atmosphere in this context. You tell me I shouldn't.

MR. CLARK: I tell you that in relationship to the Trojan Nuclear Power plant this is not a site-specific concern.

MR. LEWIS: No, the question wasn't site-specific. Ohio is nowhere near Trojan.

MR. CLARK: I guess what I'm trying to say diplomatically is that I'd prefer to address questions related to Trojan.

MR. LEWIS: I understood.

MR. CLARK: Our next presentation will deal with a structural problem sometimes referred to as masonry walls and Ken Herring from the Operating Reactor Assessment Branch will make the presentation and will provide you the information and updating the status of not only the masonry walls but the recent hearing that involved one wall of the control building and its deficiency as related to seismic design.

MR. OKRENT: Before he begins, I might note that we at some point in the past year or so asked the staff about this question of carbon deposits and how they've been looked at generically and I think one of the ACRS staff engineers would remember how we forwarded the question to the staff. I think it was a question that came to mind from something the British had written down in a report. And we were advised at that time that this had been looked at and it was not supposed to be a problem.

Now that's something they need to look at again.

MR. CLARK: Well, I'll have to take the refuge. I think I was perhaps more concerned about security and safeguards when what was developing and I'm not familiar with it -- I'm sorry.

MR. HERRING: Today I'll spend most of the discussion on Trojan specific and also present two slides on an IE bulletin -- 80-11 -- which was issued on May 8 of 1980 addressing this problem on a generic basis for operating plants. Ed Jordan, who was going to give these slides unfortunately.

At Trojan essentially over the past couple of years there have really been three different problems relative to the walls. The first one started back in May of 1976 and it affected only the walls in the control/auxiliary/fuel building complex. These three buildings are interconnected by boards and slab diaphrams.

The second is the wall problem which started presenting itself in October and was first documented by way of LER 79-15 in November '78. This was relative to all walls in the plant. And most recently, in May, there were a few problems identified with walls connecting with the slabs at the top of the walls.

This is just the general layout of the plan. This is what is commonly referred to as the complex -- the turbine building, the shake space, and containment.

There really are three basic types of masonry construction at Trojan -- the single wythe wall -- these are looking down at the cross-section of a wall -- in which you have either A-shaped block or your normal cinder block with grouted cells.

Second are the mortared double wythe, in which you have two walls essentially side-by-side with a mortared collar joint connecting the two. And in the Trojan walls there are ties going between wythes. However, they're very nominal. They're only number three bars four feet on center in both the horizontal and vertical direction. They would be going through the walls holding the two wythes together.

MR. SHEWMON: Is a number three bar a piece of corrugated steel laid in between the two by grouping?

MR. HERRING: No, it's just a regular rebar -- 3/8 inch in diameter.

MR. SHEWMON: Thank you. With a washer on each end and it holds?

MR. HERRING: No, it's C-shaped. And the third type are what are called composite walls in which you have two wythes but rather than mortaring you have concrete and it's anywhere between 4 and 48 inches thick and there is always steel present in the block, no matter which type you're looking at. It's number four and five bars, two feet on center vertically, typically, and around four feet on center horizontally going down through the wall.

And there may or may not be steel in the concrete core in this case. In the control building most of the walls have steel as you go through the -- the complex contains it in the majority of the walls. There are only two single-wide walls in containment and one composite shield wall in the bottom. But as you get into the auxiliary building and the fuel building a couple walls have steel, but most of them don't have steel in the core.

MR. EBERSOLE: What sort of reinforcing did you say was in the single-wide wall?

MR. HERRING: Some as little as number four bars two feet on center vertically and about the same number four, number five bars four feet on center horizontally.

MR. EBERSOLE: It doesn't use this hard wire every other course -- that type of thing?

MR. HERRING: Well the ties that I referred to before --

MR. EBERSOLE: No, I'm not talking about ties -- longitudinally -- vertical?

MR. HERRING: Okay. Vertical it's just using regular reinforcing steel that's placed in the --

MR. EBERSOLE: No, I'm talking about the single walls up-top.

MR. HERRING: Same thing. It's typical construction. The masonry wythes have the same type of steel in them. Okay? And it's the core that may or may not have it, but it's as low as number four bars four feet on center -- two feet on center vertically and about number fours or number fives four feet on center horizontally.

This just gives a very brief run-down of item 1 I showed you on the first slide -- the initial 1978 control building design deficiencies and it was relative mostly to in-plane strength of the shear walls. There were really three errors that led to the walls not being as strong as they should have, and the first is that there was a bad number pulled out of the ACI code, which is the design document for the wall. And too much in capacity was alloted to the concrete alone.

The shear capacity of a wall is the summation of the concrete contribution plus the steel contribution.

Secondly there was an arithmetic error in which this value was multiplied by a load factor which it should not have been, so that further compounded the walls and led to -- depending on how you would design it -- either walls that were too thin or they should have put more steel in the walls. So you either had too thin a wall or not enough steel.

It was further compounded that the composite walls in the complex are the major shear-carrying elements and in the control building end of it there's a steel encased frame that was put up to carry vertical floor loads, to give cover because of the weather out there so they could construct the walls year-round and not be set back in schedule and that frame was generally in the core. And wherever the reinforcing steel in the core contacted the frame rather than being welded to it or hooked or something it was just a straight bar. So it was generally discontinuous and therefore could not be relied upon to carry the load it should have.

That was the subject of an order in which we determined that the building had the capacity to resist a .25g OSSE. However, it didn't have the required margin in the initial design criteria and further that it no longer could resist the .15 g OBE at the site and that the OBE was reset down to .08g until the modifications required by that order could be performed to the complex to substantially restore its resistance to the SSE and bring back its OBE qualification. This was the subject of an ASLB hearing. They were shut down in May of 1978. The hearing went on and interim operation was ordered by the ASLB order in December of '79.

Modifications were looked at an final approved by the ASLB in July of 1980 to perform the modifications as far as the in-plane shear resistance of the complex. There are no walls that were affected outside the complex.

MR. EBERSOLE: Your whole discussion has been based on seismic loads, I believe.

MR. HERRING: That's the predominant load. The other lateral load would be tornado, but that was one-quarter of the seismic load.

MR. EBERSOLE: Are there any hydroloads due to any pipe breaks or anything like that that you have to deal with?

MR. HERRING: No. Not with this particular problem. The control building design deficiency is primarily an in-plane problem with overall shear resistance of the three buildings.

Okay, now I'll get to the wall problem, which started springing up in October of 1979. Essentially the problem is that there was insufficient design criteria for masonry walls of the types of loads that you find in a nuclear plant structure. And this one was primarily concerned with the out-of-plane loading since we had been addressing the in-plane loading of masonry walls with the other proceeding.

So the primary concern with this one was the out-of-plane resistance of the wall. The way it was discovered was they were conducting investigations pursuant to the base plate bulletin -- that should be 79-02. The base plate bulletin and the as-built configuration of piping systems bulletin and they found a support that had some problems with the anchor bolts and loads going way up and they went back and found the wall was somewhat under design to resist the pipe support reactions that were attached to it.

It was found in the initial that it was designed only to resist its own lateral load and not the 60 supports that were attached to it. And digging into that further, that's the general design deficiency that is inherent in masonry design criteria.

MR. EBERSOLE: Was that seismic piping that was fastened to it?

MR. HERRING: Yes. The particular wall was buried -- it was used for train separation and carried RHR, safety injection, containment spray, boron injection and so on.

MR. EBERSOLE: Well, when you found out that the walls would fall down, did you have to go back and do a redesign on the piping analysis?

MR. HERRING: Part of the modifications for that particular wall was to offload many of the supports that wee on it and support them in some other manner and where appropriate redo --

MR. EBERSOLE: Redo the seismic design of the pipes?

MR. HERRING: Right. Okay, really there were seven major areas of concern that cropped up in the investigations and masonry design criteria. The first are was how do you determine the flexibility of the wall, account for the appropriate material properties in cracking and such that were dynamic load -- mainly earthquake -- in controlling load on most walls. So that you go to the floor response vector and pick off the appropriate seismic load.

The second concern was how does in-plane and out-of-plane load interact to contribute to stiffness degradation of the wall. The third area was the capability of the multiple wythe walls to behave compositely. For out-of-plane resistance they were assumed to behave compositely. They only had the number 3 ties going through four feet on center, which weren't enough to tie t e two together to give it the appropriate stiffness.

So that led to concern over could you develop these appropriate shear stresses required to be interfaced to resist lateral loading of the wall. And the third was the concrete expansion anchor bolts were anchored predominantly in one wythe of a multiple wythe wall. If you have significant tension on a support, say, that you would have to transmit load -- tensile forces -- across interfaces in order to invoke the resistance of both players of the wall and how they interact.

Now another area of concern was resistance to local loads from missiles, piping, equipment supports, general concrete expansion, anchor bolt integrity-- since they are anchored in the outer face of this walls and bond is the predominant thing that you need to ensure that they're going to behave together.

Sixth was inaccurate mortaring of the collar joint, which is that mortar joint on a double-wide walls. Some problems were found that the collar joint was not filled appropriately during construction at Trojan. So relying on bond that's a notoriously bad joint -- not too strong really -- and it was further compounded by some construction difficulty.

And seven was how were loads due to interstory displacements considered, since on many walls they were considered like a secondary type load. In other words, they were not considered only the inertial load from the earthquake.

Right now the status of Trojan is that we initially resolved the problem based on what we knew at that time -- December 31, 1979. One of the problems with masonry is that there is really a lack of good test data and verified criteria. It's about 25 years behind concrete design.

Further tests and investigations were performed and additional concerns were raised in A[illegible] of 1980 -- by the way, Trojan was shut down until December 31, until this wall problem was resolved. They were shut down for it.

Finally, in June of this year we developed a "final criteria" that resolved these additional concerns and it was essentially two-levels -- one which must be satisfied for them to start operating. They were down for refueling in April and were kept down beyond their schedule until the middle of July when they performed additional modifications to go back to power. But the more conservative criteria that's part of this that was developed could be satisfied by October 31.

And through all this we've only been focusing in on the SSE qualifications. The criteria is based upon really application of existing code criteria and test data, where appropriate. Where not appropriate we felt ranges of properties to consider in looking and knowing it was between two points. Look at those two points and be satisfied in-between.

There was some limited in situ testing done of the Trojan walls. Part of it was proving the competency of bond of the interface of composite walls with the concrete core were done quite well. However, they did some shear tests of the collar joint and it did not work out as well as we expected initially.

There was a program done to grout any voids found in collar joints in walls by actually drilling holes and checking for voids. And the third part of it was ensuring system safety function. Some walls, while not meeting the letter of keeping stresses below yield, displacements were limited to such a level that if there were a system attached to it that the check was made to see that the extra displacement that you would get above the normally designed structures could be tolerated without impairing systems.

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In October through July, essentially, there were about 150 modifications performed at the facility; 48 cases where supports were through-bolted. As I said, one concern was transmittal of the transient across the interface, and on double wythe walls there was really no capability for that assumed, and they just through-bolted the support to mobilize both wythes, and there were 100 cases interspersed between wall strengthening by putting added steel to it on the outside, exterior steel, to stiffen it and brace it, putting post-tension bolts through the two wides to lock them together tightly, and modifications to supports themselves.

What is remaining on Trojan, there are some additional mods that may have to be performed by October 31 to meet a more stringent criteria. Their response to the Bulletin 80-11 -- The bulletin was not sent to them. It was sent for information only, since they were far ahead of what the bulletin required for action.

However, they need to complete some confirmatory testing of samples, laboratory samples that would lead to a quantification of the margins inherent in their design criteria at Trojan.

The third problem I mentioned was the wall connection deficiency. Essentially, one wall, almost the entire south wall of the auxiliary building was found not to be connected to the floor slab by dowels or any other steel. The wall just dead-ended at the slab. The problem was just caused by, there was no explicit detail supplied at that location, so they built it as they were supposed to.

This led to further investigations, drawing reviews, -- inspections, ripping siding off, some constructive examination to check the interfaces on all walls in the plant. They found six other discrepancies that were of safety significance. There were some other minor ones, but these were some major ones, where some grout was missing at the top of the wall. Some dowels were exposed, and a top layer of block wasn't added to one wythe of a multiple-wythed wall in one case; missing steel support angles to provide for support, lateral support of the wall in one case; and there were missing lintels.

Appropriate modifications have already been taken, so, -- or evaluations to show that no modification were required.

Now, the status -- essentially I&E Bulletin 80-11 was sent to all facilities with operating licenses except Trojan, Sequoyah 1, North Ana 2, Salem 2; information to all CP holders, plus the ones that did not get it for action; and separate letters were prepared for these other facilities.

Essentially, the bulletin requires that within 60 days from the date of issuance, which was May 6th of this year, they identify the safety related masonry walls in plants and establish a program for their re-evaluation; within 180 days, perform the re-evaluation, provide justification for the criteria used, and construction practices employed such that you are assured that the masonry is constructed well.

Also in here would be establishing any kind of testing program that would be required in order to confirm what is not able to be substantiated at this point in time.

MR. OKRENT: It sounds like they could establish a testing program that I remember from my childhood. It is called the Three Little Pigs.

(General laughter.)

MR. HERRING: Some of that has been done. That is about it.

(General laughter.)

MR. HERRING: At Berkeley, that is the extent of it. They constructed a few little houses for NSF and shook them.

MR. MARK: One of the things that you had on the previous slide -- I don't remember which slide -- looked like a list of things that I&E ought to have found seven years ago or something.

MR. HERRING: Well, many of these were --

MR. MARK: Missing dowels, et cetera.

MR. HERRING: Well, exposed dowels. Essentially, there was an interference with the steel beam because of the case crank, and nobody bothered to get rid of the interference, so the dowels were bent over and were hanging out of the wall. The fix in that case was just to weld the dowels to the beam, and put some grout over it.

In this particular case --

MR. MARK: That doesn't change the implication of the question.

MR. HERRING: -- I looked at it myself. It was a hidden spot where somebody had come up with the wrong sequence for construction, and built the wall on one side of a beam, and the missing block just could not be put in there, not even with a midget.

So, you know -- and the guy, he just put a cover over it, and the only way to get it was really get up there on a ladder, lean your head against the wall, and look real hard, and if you knew it was there, you found it.

I mean, they were -- some of this stuff was hidden by steel siding.

Not too many of them turned out to be particularly significant.

MR. KERR: That was a safety significant discrepancy in the sense that it decreased the strength of the wall significantly?

MR. HERRING: The overall strength, none of these really affected significantly. They were all -- There were a couple that were fairly minor sheer walls, providing for lateral resistance for the plant. I think the south wall of the auxiliary building, at the elevation it was not connected, I don't recall any very severe safety systems hanging on it, and it is not to say that it just would have flown out under an earthquake, but they were relying upon it being blocked up and bottomed, so it might have failed in that mode.

It was a sheer wall, but there was a thick sheer wall near it that would have picked up the load that it could not get transmitted into it due to the lack of reinforcement.

MR. MOELLER: You have provided us with the source of several of the problems, one being the drawings being inaccurate.

MR. HERRING: Yes.

MR. MOELLER: What were basically the sources of the major problems? And could you answer Mr. Mark's question why I&E didn't discover these discrepancies?

MR. HERRING: Well, Trojan has several hundred walls, over 300. I don't know the exact number.

MR. MOELLER: Well, can you give us a ballpark?

How much was due to inaccurate drawings? How much was due to slipshod construction? How much of it should have been readily uncovered by I&E?

MR. HERRING: On the wall problem alone, or do you want all three problems thrown in?

MR. MOELLER: Look at all of them as a group.

MR. HERRING: Okay.

In this case, there were three basic deficiencies. It turned out that the person who had performed the calculations, because these are purely calculational errors, performed the calculations in 1969, I believe, and early 1970. The calculations were not checked until 1974, and in that time, the person had been promoted to a supervisory position, and checked his own calculations.

(General laughter.)

MR. MOELLER: For whom did he work?

MR. HERRING: Bechtel is the architect-engineer.

MR. MOELLER: So it is the AE.

MR. HERRING: Yes.

MR. MOELLER: Okay.

MR. HERRING: This particular -- I will call it an error for lack of anything better, was caused by, Number One, the unique construction of Trojan, with the steel-encased frame inside the composite walls, and it was just failure to provide adequate -- for the reinforcement.

Now, as to why it was not picked up, I can only speculate. There are I&E inspection reports that were done that there were some deficiencies, but the QA/QC procedures in effect in Bechtel at that time, which are not present nowadays, there was an -- I&E did an independent inspection of Bechtel at the time of this deficiency and also during the wall problem. So, there have been some inspections done. Essentially it is a QA/QC problem.

MR. CLARK: Dr. Moeller, we did expect to have the representative from I&E here with us today, but unfortunately, he was subpoenaed and called to a trial. I think it is on the Quad Cities case. But he would have been here otherwise. Mr. Jordan.

MR. MOELLER: Well, I notice that NRC has fined other government agencies such as TVA for errors. I wonder if anyone fines I&E.

(General laughter.)

MR. LEWIS: That is a serious question. Obviously, not in the sense of a fine, but what is the consequence of this kind of error, other than saying, oh, well, these things happen?

MR. HERRING: Well, I think at that time the function of the AEC in those days was not to go in and do a detailed design review. That is being done nowadays, but it just was not done at that time. It should have been.

Something like this would show up. But it was not done.

MR. EBERSOLE: It is on the checklist now, isn't it?

MR. HERRING: Yes.

(General laughter.)

MR. CLARK: Mr. Noonan is here with us, and he would like to address the subject of what we are doing about these modifications to assure that these same errors aren't perpetuated. Vince, if you would like to address that subject.

MR. NOONAN: Vince Noonan of the Division of Engineering.

Under the new reorganization, we now have a QA/QC branch under my ADship. Having been branch chief when this problem occurred, in the engineering branch, one of my first tasks with this QA branch is to form them up to go out to Trojan and look into their QA/QC procedures, along with the I&E people, and make sure that in these modifications we don't run into the same kind of problems we had when this plant was built, and that will be done very shortly.

MR. HERRING: Well, I just want to finish answering the question. The wall problem itself, the essential problem there was just a lack of knowledge about the behavior of masonry. Even today, there are really a handful of people in the country that really know about the behavior of masonry, and it was something that precipitated itself.

The older plants, very few -- very little masonry used. In the newer plants, it started being used because it is easy to put in as a backfit. People didn't intend it to support piping. However, the piping support people go through, see a wall there, and walls are supposed to stay there, so he hung his supports off it, and it just perpetuated itself

So, the wall connection deficiency, they are minor discrepancies. I would list them as minor compared to everything else. They were in obscure areas because the plant had been walked down previously at the discovery of the initial control building deficiency; however, no destructive examination was done in many cases, or going up with mirrors behind small obstacles. So, it wasn't that great in that particular case.

(Slide.)

MR. HERRING: The last slide I have is just the status of the bulletin, and these are just number of masonry walls, which is really all we would require in the 60-day response, that are safety related, and the number of facilities that have varying amounts of walls.

I just learned yesterday that this particular number is down to four, and that particular plant lies in about the 50 masonry wall category.

MR. EBERSOLE: In your discussion, "masonry" is a very general term, isn't it? It can start with adobe, I guess, and go on up to better things.

MR. HERRING: Yes. Well, essentially what is used in the nuclear plants is concrete masonry. There are cases --

MR. EBERSOLE: Does that mean that these are true concrete blocks?

MR. HERRING: Yes, they are made of concrete. They are concrete blocks.

MR. EBERSOLE: They are sand-mixed concrete blocks?

MR. HERRING: Yes.

MR. EBERSOLE: They are not lightweight aggregate or cinder?

MR. HERRING: No, they are not cinderblocks. There are some what are called standard weight blocks, and then there are heavyweight blocks with much higher density. The standard weights are a little bit less dense than concrete.

MR. EBERSOLE: Well, when you bring that stuff into the plant, do you have any manufacturing curing program to make sure that you are not getting just a weak mix bunch of blocks?

MR. HERRING: Yes. They are subject essentially to the types of things that are done for concrete to determine the block compressive strength, and it is controlled.

MR. EBERSOLE: You do -- All right. You have --

MR. HERRING: Yes, there are specified design minimums --

MR. EBERSOLE: For the block.

MR. HERRING: -- and then they do checks of the batches to make sure that they are getting good quality block. There are cases of solid block also in some plants. A large use of it is for shielding purposes.

MR. EBERSOLE: Anyway, they are held to requirements on the recipe, and so forth?

MR. HERRING: Yes.

MR. PLESSET: Well, thank you very much. We appreciate that.

Now, what I would like --

MR. OKRENT: Mr. Chairman, we did hear what the staff is going to do to see that they repaired Trojan correctly, but I just wanted to know, you may recall there was an earlier discussion about were they giving enough attention to catching design errors, and I would submit that we haven't heard from the staff any meaningful program to at least begin to develop some new technique, assuming there is one, of reducing the number they get through.

MR. PLESSET: I don't think you can answer that right now, but it is noted, and I think that is a significant point that Dr. Okrent wanted to make. I don't know of any really good way to make audits except a lot of tramping around.

MR. KERR: I thought that we were assured that much of your design errors were now being made than was the case ten years ago, so the slope of the curve must be in the right direction.

MR. HERRING: Yes. Well, as I said, I&E has done an inspection of Bechtel a couple of times to see if there are problems today that would cause something like what happened in the past. And they have found that they are in conformance with what is done today, and furthermore, on the wall problem especially, all along we have been doing -- taking a look at the calculational methodology through sample calculations, plus actual design calculations also.

MR. KERR: I think we need to have a QA lecture prepared for Dr. Okrent. He probably doesn't know about QA.

MR. OKRENT: That is probably my problem.

MR. PLESSET: We will not accuse you of lacking.

MR. EBERSOLE: Before these fellows leave, Mr. Chairman, concerning QA, I can recall a case, a particular case where it was found that if one invoked an ordinary, legitimate pipe break, that the situation was such that you were in a confined gallery with very good doors, and the throughput of water through the break with the pump running on literally took the building down, and unfortunately, the building supported some critical switchboards.

Do you all look at things like that?

MR. HERRING: Pipe break has been addressed by Trojan, and we did look at those types of things under this wall problem for lateral load.

MR. EBERSOLE: The internal pressure that you might --

MR. HERRING: As it turned out, there were no compartments that were subject to internal pressurization.

MR. EBERSOLE: Okay. So it is a ritual for you to look at that, then?

MR. HERRING: We did look at it, yes.

MR. EBERSOLE: No, I said, is it a standard ritual?

MR. HERRING: Yes.

MR. EBERSOLE: Okay.

MR. PLESSET: Well, thank you again.

MR. HERRING: You are welcome.

MR. PLESSET: And let me ask the committee to do an item very fast, because we have an additional thing to do, which is also short. That is regarding our future schedule. Tab 521 and 522. Now, I am not going to go over that in detail. I think if you will just look at it, you will get the general picture that -- I am getting some additional information. Just a moment.

(Pause.)

MR. PLESSET: Mark is calling my attention to, in 521, Item 1, the last one, "Propose NRC plan to compare operating nuclear plants with current regulatory criteria," which is a requirement now. And what he is asking is a subcommittee assignment to do this.

I would think that -- It is not just the operating plant subcommittee.

MR. ZECH: Let me give you a little more background on it perhaps, and then we can decide or you can discuss who you might want to assign the subcommittee action to. In the appropriation bill for the Commission this past year, there was a section, Section 170, just for reference purposes, that requires that a certain amount of our funding be allocated to what is called "a systematic safety evaluation of all currently operating utilization facilities required to be licensed under the sections applied to the power reactors."

The plan that they are requiring that we pursue would be to identify each current rule and regulation that is issued by the Commission with particular significance to the protection of the health and safety of the public.

The effort which Bob Baer's branch will be initiating, its first step will be to do just that, to attempt to identify and define what that word is, "significant rule and regulation." The next step then would be to determine the extent of compliance to those rules and regulations that are identified.

There is a plan that has been formulated. I understand that it will be made available within a matter of a few days, and what I propose to do is to provide a draft, these draft copies to whatever subcommittee you should decide should look at it.

We would ask that during the meeting next month, that we discuss this plan with you, and to have you input, because we have a rather short time period in which we have to get back to both the Commissioners and to Congress. There is a 90-day clock that commenced this past month, and we have essentially until the first part of October in which to provide a plan of attack in which we will ultimately determine the extent of compliance to those rules and regulations which are considered to be significant.

Those are just two parts of a four-part requirement. The other two parts deal with NUREG-0410, which is a list of generic issues, and we are being required to go through this list of generic issues and identify those issues which have been resolved -- quote, end quote -- and then to determine which ones should perhaps be included in the rules and regulations of the Commissioners.

That is a second effort that is really not the effort which we are asking a subcommittee to consider. We feel the first two items, that is, the significant rules and regulations, and the degree of compliance, are the major ones. The legislation references the SEP program, and suggests or recommends that a program similar to that be established to carry out these requirements.

MR. PLESSET: Thank you. When would you be ready to come to a subcommittee, once we have chosen the right one?

MR. ZECH: We would surely request that it be made or scheduled in the near future. We are suggesting the first part of September.

MR. PLESSET: It sounds to me -- Yes, Bill?

MR. KERR: I presume, and maybe I shouldn't, but do rules and regulations include regulatory guides?

MR. ZECH: No, they do not, but the legislation indicates that for those rules and regulations that are identified, that the Reg. Guides, standard review plans, and so forth should be used to determine or assist in determining the compliance with those rules and regulations.

MR. PLESSET: I would expect that it would be the Operating Reactor Subcommittee that would take on this task.

MR. MATHIS: I was afraid you were going to say that.

MR. PLESSET: I know. I have been looking at you pretty intently. Bill, do you have a counter suggestion?

MR. KERR: I thought that was a very statesmanlike decision.

MR. PLESSET: Yes, I thought so, too. I think there are 13 in favor --

(General laughter.)

MR. PLESSET: Is that agreeable?

MR. OKRENT: I am with Bill 100 percent.

(General laughter.)

MR. MATHIS: It sounds to me like it is very similar to IREP.

(General laughter.)

MR. MOELLER: There should be -- I guess there should be a lot of interplay with regulatory activities.

MR. PLESSET: Right, I agree, and I think that Mathis will take care of that by getting the right participation. I think we can leave it to his wise choice. Do you want to make a suggestion, Dave?

MR. OKRENT: No, if this topic is done, I want to --

MR. PLESSET: Okay. Yes, it is finished, I think.

MR. MATHIS: Thanks a lot.

MR. OKRENT: With regard to the September meeting, you currently under Item 2 of the memo by Fraley note that there is supposed to be a report on the quantitative risk criteria effort. It is our expectation, although I can't say it with 99.9 percent confidence, that we will have for the full Committee to look at a set of three documents which are a package that they might consider forwarding to the Commissioners as what Bill Kerr once talked about as the first part of an interim process.

I think if we are to meet October, which we are sort of committed to in a variety of ways, if we can, it would be well to try to do it in September, to see why we can't, so that we can make it in October.

MR. PLESSET: All right.

MR. OKRENT: So I would like to suggest that you treat it sort of like a case in September.

MR. PLESSET: All right. Now, let me point out another item that you might note. This last item on the first page of this sheet on report of ACRS subcommittees, I would call Dade's attention to that.

MR. MOELLER: Mr. Mathis is chairman of the LER subcommittee? I did want to comment on that.

MR. PLESSET: Well, if you comment enough, you might inherit a consequence.

MR. MOELLER: Let me just make one remark. In terms of information and supporting documents on efforts to improve the LER reporting system, I would like the letter which I wrote to Dr. Bates about a month ago to be included, because at that time I enumerated a number of problems which we are having with the LER system.

MR. PLESSET: Fine. There is one more item.

MR. MATHIS: Before you leave that, Mr. Chairman, we are not ignoring the LER situation.

MR. PLESSET: Right.

MR. MATHIS: Andy and I have had a couple of discussions trying to decide where we go next. We also ran into this a little bit earlier this week. The LER system needs a lot of attention, because there is too much chaff and very few kernels, and we are going to see what we can do about it, but I don't have any firm answer right now.

MR. PLESSET: Thank you.

Any other comments?

MR. MOELLER: I think we do have to keep -- or certainly the Committee is interested in keeping close tabs on Carl Michaelson's program, because I think it is -- I agree with Bill. It is of the utmost importance that some of these problems be corrected.

MR. PLESSET: Well, if I may, this is not on the agenda, but you may recall that I made a remark about our Sequoyah letter this morning, which I presume we will mention again to the Commissioners tomorrow, and to which we will return at our next meeting in more detail, but right now, both TDA and the staff wish to make a statement, and I very generously allowed five minutes to each one of them.

MR. TEDESCO: Yes, Mr. Chairman, thank you.

The subject that we would like to discuss with the Committee at this hour deals with the pressurizer relief line well repair on the Sequoyah Unit 1 plant. This subject was discussed at the July ACRS meeting in terms of resolution of a minority viewpoint of a staff member, Mr. Joseph Halapats.

Subsequently, we have learned of new information concerning a second mock-up of a test that was performed by TVA. We first learned of the second mock-up test during our peer review by research somewhere in the period of the 17th and 18th of July of this year, which was after the July 11th ACRS meeting.

Some members of the staff did become aware of the second mock-up test subsequent to the July meeting. The report of the peer group and the NRR disposition are given in the report that was transmitted this morning. It involved the report that was dated July 25th by Richard Volmer to Harold Denton of the staff.

We have an opportunity to look back upon where we were and where we have come from today. Certainly in our judgment it would have been prudent for us to have notified the ACRS upon first learning of the second mock-up. We would like to note at this time to the Committee that we have asked I&E to make inquiries of the entire matter for us, just to determine the whole chronology of what did happen to bring this about.

At this point, before we go further into our staff summary, I think it is appropriate that we ask TVA to add their statement at this time. Larry Mills.

MR. MILLS: I am Larry Milles, manager of nuclear regulations and safety for the Tennessee Valley Authority.

We and the NRC staff responded in the July meeting by giving a history of the weld problem, including Mr. Halapats' concerns. We discussed all information that had been supplied to the NRC on the Sequoyah docket and used in making the decision that the weld in question was satisfactory.

In addition to the information on the docket, our laboratory staff proceeded with additional testing on welds. Now, this additional testing was not performed, nor was it used for the purpose of supporting our justification that the weld in question was satisfactory.

Since the results of the additional test were in support of TVA's position, had not been submitted to the NRC staff, had not been reviewed by Mr. Halapats, and were not considered by TVA or the NRC staff in reaching a conclusion, I considered it very inappropriate to introduce a discussion of those tests at the July meeting.

Certainly if the test had indicated anything other than supportive type information, TVA would have been obligated to report the results to the NRC and the ACRS. This was not the case. There is nothing derogatory in those tests. It was merely supportive information.

Now, subsequent to the July Committee meeting, NRC staff members visited our laboratory, and the additional tests were discussed with them. We are prepared, if the ACRS would like any detailed discussion of that or anything else, we are certainly prepared to discuss it.

I would like to make it very clear, though, that we did not think it appropriate to bring in additional information which was anything but supportive type information.

MR. PLESSET: Well, I think that your definition of "suitable information" is a little bit on the narrow side.

MR. MILLS: Dr. Plesset, I didn't say -- I am sorry. What did you say, first?

MR. PLESSET: I said that your judgment of "suitable information" and your definition of it, I think, rather limited, a very narrow one.

MR. MILLS: I didn't say "suitable," sir. I said "supportive."

MR. PLESSET: Well, but I think that the

Committee, as you will recall, spent a lot of time trying to find out if there was a section of pipe that was the same lot as the pipe under question, and it appeared that there wasn't, as far as the Committee knew, or there was great difficulty in getting such a section of pipe and performing such a test. That was the impression I think the Committee had.

MR. MILLS: Yes, sir. Dr. Plesset, may I ask some of our design engineers to -- I don't think I took five minutes. Can I let them take an additional two minutes to explain that situation, please?

MR. PLESSET: Well, if it is an explanation that helps, sure.

MR. MILLS: I will ask John Ralston to address that.

MR. RALSTON: The second mock-up was not of the same lot of pipe as the original -- the pressurizer relief line. The second mock-up is of 316 stainless steel, which is the same material that the pressurizer relief line as installed is made from.

We felt and still do that the original mock-up, which was made out of 304 stainless steel, supported our arguments that the weld was safe and proper. In fact, the 304 stainless steel is probably a more conservative material to use from the standpoint of stress corrosion cracking.

MR. SHEWMON: The question -- The tests you did had to do with sensitization, though. You didn't run stress corrosion cracking tests on them, did you?

MR. RALSTON: Which tests are you referring to?

MR. SHEWMON: The ones you just got -- you said you thought the 304 was more prone to stress corrosion cracking. It seems to me that is irrelevant. I thought the question had to do with whether the weld was sensitized to stainless steel. Was part of the test you presented last time stress corrosion cracking test results? Was that how you justified the weld?

MR. RALLSTON: No.

MR. PLESSET: I don't think we even heard that, Paul.

MR. SHEWMON: I don't think you did either, so I don't quite see what his point is.

MR. MERRICK: My name is Ed Merrick, and the tests we presented last time were the A262 practice E tests, which in our estimation says that we don't have a susceptible material for intergranular corrosion.

MR. SHEWMON: But things that have passed that test have given stress corrosion cracking, as you must realize if you follow the nuclear literature.

MR. MERRICK: That is true, in highly oxygenated environments.

MR. SHEWMON: Well, "highly?" But then, go ahead.

MR. MERRICK: Eight parts per billion. We are talking now in the range of .126 parts per billion.

MR. SHEWMON: Let's hope, up at the top of that stagnant pipe.

MR. MERRICK: I would like to mention we are going to be taking oxygen measurements in that pipe.

MR. PLESSET: Do you have any other point?

Yes, Bill?

MR. KERR: Mr. Chairman, I wanted to make sure that I understood another part of this statement. It seemed to me, if my memory is correct, that we were asking last time if piping from the same run, whatever that means, was available, and we were told it was not probably available. Now, you seem to say that indeed pipe from that run was not available, and that the test you did on 316 stainless was not from the same run, but only the same material. Is that right?

MR. MILLS: That is correct, Dr. Kerr.

MR. KERR: Okay.

MR. CARBON: But by the same token, we requested that if material from the same lot were not available, that a mock-up with 316 be run, if I remember correctly. We were asking that that be done. You sat here, knew it had been done, and didn't make any comment on it. I don't understand that.

MR. MILLS: Dr. Carbon, perhaps we were in error there. I will say that the work that had been done at that time, although some of the tests that had been done, these fellows were still trying to put a report together on it. We felt like we had completely justified our case, and we had nothing but supportive type information. We had been through this, not before ACRS but priort to coming to ACRS, on about two steps already.

Certainly if the ACRS wanted such a thing, you know, we would finalize our reports and get it in.

MR. CARBON: Well, we said we wanted it. You heard us say it. You knew it had been done. You didn't mention it. I don't understand.

MR. MERRICK: The tests, as I understood it, in that meeting were not defined appropriately. I think the test was some nebulous test in an oxygenated environment that has no acceptance standards established in the industry. We had not planned to run that type of test. And the question that we were asked was, how much would that test cost if we were to run it, as I remember, and we had no idea.

(Pause.)

MR. PLESSET: Well, I think we will set aside some time at our next meeting on this point. I think we don't have enough time to pursue this, and you will be prepared, too, and we will come back to it.

MR. EBERSOLE: May I ask a question, just to show my own ignorance, if nothing else?

MR. PLESSET: Yes.

MR. EBERSOLE: Downstream from this repair, a valve has been welded into this pipe, hasn't it?

MR. SHEWMON: Yes.

MR. EBERSOLE: Is that not sensitized?

MR. SHEWMON: I asked the questions about how were the welds at either end of this pipe different from what was done there, and the response I got was that they had been done in one pass, and this one had the benefit of having been ground out and rewelded again, which certainly would have increased the amount of sensitization.

MR. EBERSOLE: So it is a matter of degree. This is sensitized worse than those.

MR. SHEWMON: Presumably.

MR. MERRICK: May I say -- This is Ed Merrick.

MR. SHEWMON: Yes.

MR. MERRICK: There are a number of welds in that line, and every other line that we have in our plants, that have seen as many as -- oh, I would like to say three, but I think I am being underly conservative of there of repair cycles, so that weld is certainly not any worse than other welds in the system.

MR. SHEWMON: You are saying that where that pipe came into the pressurizer, for example, you went around in three passes to complete the weld?

MR. MERRICK: No, this is a multi-pass weld. We are talking about three-quarters of an inch material, and the -- by repair cycles, I am saying that I know of at least three welds in that system that have seen almost through-wall repairs which to date I know of no regulation that prohibits us from doing that, for the purposes of grinding out defects and things of that nature.

MR. SHEWMON: I don't understand what you are saying yet. Let's say it twice more, and maybe I will get it. My question was, was the weld made between the six-inch pipe and the pressurizer in several passes?

MR. MERRICK: That is true.

MR. SHEWMON: You are talking about repairing something some place in some system.

MR. MERRICK: Well, Dr. Shewmon, what I am talking about is the fact that in that same line --

MR. SHEWMON: Let's talk just about the weld I asked about. Then we can talk about what you want to talk about. Okay?

MR. MERRICK: I don't have specific data on the weld that you are asking about. I suspect that it has -- it has been made by a multi-pass welding procedure, because of the thickness of the pipe. I would say it is probably a TIG root and shield metal arc fillup, so we've probably got as many as nine passes in it with some repair cycles.

MR. SHEWMON: Now, you are saying that in general the repair procedures which are code approved for this material do allow you to make as many repair passes as you want without ever calling this out as giving undue sensitization. Is that right?

MR. MERRICK: That is true, sir. There are no regulations prohibiting that that I know of, and I am not --

MR. SHEWMON: Either in BWR's or PWR's?

MR. MERRICK: I think that is true. I think that there is some interest in establishing this type of program to study it --

MR. SHEWMON: There sure as hell ought to be.

MR. MERRICK: Well, we are doing some work in this area on our own. So, we have recognized the problem, and we are not letting it sit still.

MR. SHEWMON: I am sorry. Is that responsive to your question?

MR. EBERSOLE: Well, I think we just agreed that we have other sensitized places. Having sensitized something, a hydro test doesn't tell you anything about the quality of it. You have got to wait for time and corrosive effects. So, I am just trying to get this thing in perspective.

MR. MERRICK: That is true. We have committed to doing augmented ISI, as I understand, on this weld.

MR. PLESSET: Are you? Did we ask for that in our letter? We didn't, as I recall. But you are committed to it?

MR. MERRICK: Yes, sir.

MR. PLESSET: We can add that to our letter?

MR. MERRICK: Yes, sir. It is -- I belive it is in the SER.

MR. EBERSOLE: Maybe this is not the worst case. Maybe we ought to be thinking of something else.

MR. PLESSET: Yes?

MR. OKRENT: Mr. Chairman, I am not sure whether you are trying to resolve an issue now, in which case we ought to --

MR. PLESSET: Not really. They wanted an opportunity to make a presentation, a brief one.

MR. OKRENT: It is not clear to me, in other words, where this matter was thought to stand at 6:15, when we began it, whenever it was that we began it, and where we think we are now, or where we are going today.

MR. PLESSET: Well, I agreed to allow them to make brief presentations, and I was proposing to set some time aside at our next meeting.

MR. OKRENT: Isn't Sequoyah going to be in for other reasons next month as well?

MR. TEDESCO: May I speak for a moment?

MR. PLESSET: Yes, go ahead.

MR. TEDESCO: At this stage of our review, we are planning to go ahead with our licensing action for full power operation. We will be appearing before the Commission in the latter part of this month. The staff has satisfied itself techncally on the resolution of this matter, and if the Committee so desires, we can indeed come back next month and talk further about it, but we would at this point move that we go ahead and complete our SER and move forward with the plant.

We are now able to give you a five-minute overview, if you wish, and tell you where we are, and to state our position that we are putting into our SER, but that is the staff's present approach.

MR. OKRENT: Has the staff reviewed the reports done by RDA on the Sequoyah containment and so forth, and does that --

MR. TEDESCO: Right. We had two reports from RDA, and these matters are being resolved now, and there is a Commission briefing next Wednesday on this whole matter, on the matter of hydrogen as it applies in an ice condenser that order, would like to make comments.

MR. CARBON: Mr. Chairman, could we discuss this more Saturday? Is there time?

MR. PLESSET: Yes, I think we have time.

MR. CARBON: I would so recommend.

MR. PLESSET: Yes. Carson?

MR. MARK: I have been a little disturbed that there has been a suggestion that information has been withheld. I think it would not have been out of place to have mentioned the existence of a 316 mock-up the last time we talked of it, but we really were talking about stuff from the same heat, and there is no suggestion that that can be found, or it was suggested that it could not.

The tests which we talked of were certainly not, I think, the tests which they had subjected their supplementary item to. We wanted a through-wall metallographic examination. They have not done that, as far as I know.

While I think it would have been favorable to have it mentioned, I don't think that unless it had shown bad signs, that the suggestion, which I have heard made several times, that information was withheld and this was really very reprehensible, that there is really any basis for that feeling.

MR. CARBON: I may be quite wrong, but it is my impression that they had already run exactly the same tests that we were asking that they go out and run. If I am wrong, I am wrong, but that is my impression.

MR. PLESSET: Yes, Jerry?

MR. RAY: Just a quicky that can be answered yes or no. Do we know yet whether or not Mr. Halapats' concerns have been resolved by this second mock-up?

MR. MILLS: Dr. Plesset, we would address Dr. Carbon's question first, please, sir.

MR. PLESSET: All right.

MR. MILLS: The second mock-up that we did was subjected to the ASTM 262 practice -- no, that is what I was getting to. It was subjected to the ASTM A262 Practice A and E tests, and was not, as we understood it, the tests that were asked for or suggested at the last ACRS meeting. I don't believe we left the ACRS meeting, as I think Ed had said earlier, with any clear understanding of exactly what tests were intended, and this -- the tests that were done on this second mock-up were the ones that we would normally do in furtherance of our work on this weld technique.

MR. CARBON: If I remember correctly, you had a representative here during our letter-writing session or discussion last Saturday, and there are also I guess probably the transcripts, but in any case one of your people was here, and we put a comment in our letter on the kind of mock-up we thought you ought to go ahead and do. Again, I confess that I may be under --

MR. KERR: Well, Max, if you could understand what we wanted from that letter -- I could not interpret the tests that we were asking for in the letter.

MR. MILLS: Dr. Carbon, I did have a representative here for that session that you are talking about. The engineer that was here worked directly for me. He is a licensing engineer. And I will have to admit to you that, you know, you can sit down and lay it out in minute detail for me, and I still wouldn't know what you are talking about.

Now, the fellows I have with me today here are involved in that testing, and maybe they should have been here, and I am sorry that we were not, but the fellow that was up here that day is a licensing engineer who works directly for me.

MR. CARBON: Well, it is my remembrance that you had run a test with 304 stainless, and much of the discussion was centered around the fact that 304 is of course different from 316, and 316 is the actual material, and unless I am mistaken, our letter suggested, urged that you set up a mock-up using 316 instead of 304 and see what results you came out with, duplicating what you had done before.

It is also my understanding that that is what you had already done. You had taken 316 and run the same test, and it was the same thing as we were asking you to do.

MR. MILLS: I think that is basically correct. I think, however, during the ACRS meeting, I don't believe there was any specific discussion of what test it was that came out in the letter. In other words, the letter to us was a little bit more explicit as to your desires than the discussion we had in the ACRS meeting.

MR. PLESSET: Well, I think we should let Mr. Halapats -- Are you going to be brief, Mr. Halapats?

MR. HALAPATS: Very brief. With a wife and three daughters, you learn to keep your mouth shut. They tell you when to talk.

MR. PLESSET: That's good. A lot of us need that lesson.

MR. HALAPATS: Okay. Now, I am quoting from the ACRS letter to the Commission. The statement is made that, "The evidence available is inconclusive on this point, and more specifically relevant information could be obtained without serious difficulty. This could be done by constructing a mock-up of the weld in question using material and procedures as similar as possible to those which apply in the actual case and subjecting the mock-up to a through-wall metallographic examination. The results of this examination could then be compared," et cetera, et cetera.

These are the words from the Commission letter. However, there is another point which perhaps should be clarified. The mock-up that was examined originally as the qualification mock-up is identified as a 316 stainless steel mock-up. This is in accordance with a TVA letter evaluation of draw beat mock-up 316 stainless steel for sensitization.

The memo is dated January 14, 1980. Subsequently, presumably the same mock-up was identified in the March 13th meeting at Bethesda as being 304. So, there were no commitments in the ACRS letter that you perform any intergranular corrosion tests in the heat service environment.

The only thing you were requested to do was to build a mock-up, perform through-wall metallography, mock-up of as similar material as you could get.

Incidentally, I have reviewed the findings of the peer review group. The review was done by members of the Office of Nuclear Research. I concurred with the findings and the resolution proposed. The resolution identified -- the findings identified that the mock-up, the second mock-up was in fact sensitized, and likely to be subject to intergranular stress corrosion cracking in the service environment.

The fix that was proposed was to perform augmented in-service inspection in accordance with the guidelines provided in NUREG 313, Revision 1, which required that service sensitive non-conforming lines be given augmented in-service inspection at certain stated periods of time.

I concurred in the resolution.

MR. PLESSET: Thank you. I appreciate that. And I gather that you are going to require this.

MR. TEDESCO: Yes, that will be included in our technical specifications.

MR. STAHL: Carl Stahl, the project manager on this project, which I have lived with since Day 1.

On this particular issue, I do have the safety evaluation report from the staff. It concurs in the recommendation of a peer group. In particular, of course, is the technical specifications that we will be adding as a requirement for in-service inspection, an augmented in-service inspection.

We feel, I think, as I think all the staff, technical staff and supporting groups, including I&E and the inspector and so forth, that this matter is technically resolved. We are now satisfied, fully satisfied with the manner and approach that we are now taking.

I believe this is now satisfactorily closed out and resolved, and this will be reflected. I do have that safety evaluation report, and I am incorporating into the overall SER Supplement Number 2 that covers all of the full power requirements, which will be issued shortly.

MR. PLESSET: Yes, Al?

MR. LEWIS: Mr. Chairman, just one question very quickly, and then one comment.

The question. My own ignorance, because I wasn't present during the discussion. This particular pipe, the issue is that it would be a non-isolable LOCA if it were to yield?

MR. PLESSET: That is the question.

MR. LEWIS: Thank you. I wasn't sure where it was.

Comment. This is a very important issue, as nearly as I can tell, and there seems to be some difference of opinion, if I understand it, about whether it was reasonable for our friends from TVA to have understood what my cryptic friends for ACRS were asking at the last meeting well enough to have volunteered information about the 316 mock-up.

It would be very helpful to see the transcript.

MR. PLESSET: We will get that, Al.

MR. LEWIS: Will we get it soon?

MR. PLESSET: We will have it tomorrow.

MR. LEWIS: Okay. That would be soon enough.

MR. PLESSET: It is available, I think, and we will get it for you.

MR. LEWIS: Because some comments that do imply withholding of information have appeared. If they are correct, they are very serious. If they are incorrect, we owe these guys an apology.

MR. PLESSET: Never that.

MR. LEWIS: Well, we owe them something strong on one or the other side of the coin, and it would be nice to resolve it, just speaking for myself.

MR. PLESSET: Okay. We will get that for you for tomorrow.

I thank all of you gentlemen for being so patient and staying so late. Yes, Dave?

MR. OKRENT: Since TVA is here, last month, when we were talking about ignition systems and filtered vented containment, they quoted a number of 800 or 900 rem, the dose in low population zone if one used a filtered venting system, and they said they would supply the backup to that calculation, what assumptions went into it.

I haven't seen that yet. I remain interested. When I bounced that number off some people who do this kind of calculation, they said it sounded high, quite high, and so I would like to see just what did go into that.

I assume we can get it soon.

MR. MILLS: Dr. Okrent, I recall you request. I recall also telling you that we would get that to you. I will have to admit, I will have to check and see when I get home, but yes, we will get it to you soon. I assume we haven't yet.

MR. PLESSET: Well, I think we will recess until 8:30 tomorrow.

(Whereupon, at 6:50 p.m., the meeting was recessed, to reconvene at 8:30 a.m. of the following day.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

in the matter of: 244th MEETING

Date of Proceeding: August 7, 1980

Docket Number:

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

David S. Parker

Official Reporter (Typed)

(SIGNATURE OF REPORTER)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

in the matter of: 244th MEETING

Date of Proceeding: August 7, 1980

Docket Number: ______

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Suzanne R. Babineau

Official Reporter (Typed)

Suzanne Babineau

Official Reporter (Signature)

R.G. 1.97 AND ANS 4.5

POINT 1 - THE POINTS OF AGREEMENT ARE TOO FEW IN NUMBER AND IN CONTENT

POINT 2 - THE AREAS OF DIFFERENCE HAVE NOT NARROWED SINCE 12-79; AN UNEXPECTED RESULT

POINT 3 - ANS 4.5 HAS A BROAD BASE OF INDUSTRY SUPPORT FOR ACCIDENT MONITORING VARIABLES AND REQUIREMENTS

POINT 4 - MAJOR OVERHAUL OF R.G. 1.97 IS NEEDED FOR:

(A) SCOPE, AUDIENCE, PURPOSE

(B) REQUIREMENTS TIED TO OBJECTIVES AND FUNCTIONS

(C) FORMAT, CLARITY, UNAMBIGUITY

(D) REASONABLENESS

LS
8/4/80

ANS 4.5 APPROACH

1. Defined Accident Phases
2. Defined Functional Requirements
3. Defined Process For Variable Selection
4. Defined Criteria To Be Applied To Variables (Based On Functional Requirements)
5. Defined Minimum Variable Set
6. Designer Selects Variables/Performance Requirements By Applying Criteria/Designer Analysis

M O N I T O R I N G   F U N C T I O N S

TYPE A - PREPLANNED MANUAL ACTION

TYPE B - CRITICAL SAFETY FUNCTIONS

REACTIVITY CONTROL

CORE COOLING

REACTOR COOLANT SYSTEM INTEGRITY

PRIMARY CONTAINMENT INTEGRITY

RADIOACTIVE EFFLUENT CONTROL

TYPE C - BARRIER INTEGRITY

FUEL FAILURE

REACTOR COOLANT SYSTEM BREACH

PRIMARY CONTAINMENT BREACH

POTENTIAL FOR PRIMARY CONTAINMENT BREACH

10.

ANS 4.5 PROGRESS

7-31-79	WRITING GROUP FORMED
10-15-79	ANS-4 BALLOT ON DRAFT 3 COMPLETED
2-29-80	NUPPSCO BALLOT ON DRAFT 5 COMPLETED
4-2-80	DRAFT 6A DISTRIBUTED
6-17-80	NUPPSCO LETTERS RECEIVED
7-14-80	FINAL CHANGES TO DRAFT 6A SUBMITTED
8-31-80(E)	NUPPSCO RECONSIDERATION PERIOD ENDS
9-1-80(E)	SUBMITTAL TO STANDARDS STEERING COMMITTEE
10-1-80(E)	SUBMITTAL TO ANSI
12-1-80(E)	ANSI APPROVAL
2-1-81(E)	PRINTED COPY DISTRIBUTION

LS
8/4/80

SIGNIFICANT DIFFERENCES

	R.G. 1.97	ANS 4.5
PURPOSE	ASSESS PLANT AND ENVIRONS CONDITIONS DURING/AFTER ACCIDENT	CRITERIA FOR AMI FUNCTIONS, VARIABLES, REQUIREMENTS.
AUDIENCE	OPERATING ORGANIZATION	CONTROL ROOM OPERATOR
SCOPE	AMI, STATUS, E-PLAN SUPPORT, SAFETY PARAMETER DISPLAY, TECH. SUPPORT CENTER, EMERG. OPERATIONS FACILITY, NUCLEAR DATA LINK	ACCIDENT MONITORING.
	ACCIDENTS AND ANCIP. OPER. OCCURRENCES	ACCIDENTS
VARIABLE TYPES	A, B, C, D, E	A, B, C
SPECIFIC TECHNICAL REQUIREMENTS	ORGANIZED BY TABLE 1 QUAL. CRITERIA	ORGANIZED BY FUNCTION AND VARIABLE

LS
8/4/80

SIGNIFICANT DIFFERENCES (CONTINUED)

	R.G. 1.97	ANS 4.5
TYPE B VARIABLES (PWR)		
REACTOR CONTROL	4	1
CORE COOLING	5	1 TO 4
RCS INTEGRITY	4	3
CONT. INTEGRITY	3	2
RADIOACTIVE EFF. CONTROL	0	1
TYPE C VARIABLES (PWR)		
FUEL CLAD BARRIER	2	1 TO 2
RCPB BARRIER	3	3
CONT. BARRIER	2	2 TO 4
TYPE D VARIABLES	~30	NONE
TYPE E VARIABLES	~19	NONE
TABLE NOTES	TABLE 1 - 17 TABLE 2 - 21 TABLE 3 - 20	6

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8/4/80

RECOMMENDED STEPS

1. SPLIT RG 1.97 CONTENT INTO TOPICAL SECTIONS OF VARIOUS REGULATORY GUIDES

   AMI INTO RG 1.97
   SAFETY SYSTEM STATUS INTO RG 1.47
   EFFLUENT DISCHARGE PATH MON INTO RG 1.21
   NUREG 0696 COMMUNICATION NEEDS IN NEW RG

2. MAKE EACH TOPICAL SECTION SELF-SUFFICIENT

   SPECIFY CRITERIA
   SPECIFY REQUIREMENTS
   SPECIFY VARIABLES

3. REALLY ENDORSE ANS 4.5 FOR AMI PORTION

4. ELIMINATE CONFUSION INTRODUCED BY TABLE 1 QUALIFICATION CRITERIA CATEGORIES

   FUNCTION-SPECIFIC REQUIREMENTS
   VARIABLE-SPECIFIC REQUIREMENTS

5. EMPHASIZE CLARITY IN COMMUNICATING REQUIREMENTS

   ONE INDIVIDUAL RE-WRITE

6. ENCOURAGE SOLUTION FLEXIBILITY

   CRT GRAPHICS
   TRADE-OFFS

HUMAN FACTORS CONSIDERATION MISSING

- HUMAN FACTORS ENHANCEMENT IN THE CONTROL ROOM AND AMI AT CROSS PURPOSES
- NECESSARY/SUFFICIENT CRITERIA MUST BE APPLIED
- REG. GUIDE 1.97 HAS SUBSTANTIAL IMPACT
  - INFORMATION OVERLOAD
  - BACKFIT ANOMOLIES
- HUMAN FACTORS PLAY SIGNIFICANT PART IF AMI TO BE SAFETY IMPROVEMENT

AMI IMPACT

	ANS 4.5	REG. GUIDE 1.97
TOTAL CLASS IE DISPLAYS	34	53
• ADDITIONAL	20-30	29-41
• UPGRADED	0-8	8-16
• EXISTING	4-6	4-8
TOTAL "CLASS 2E" DISPLAYS	NONE	110
• ADDITIONAL		71
• UPGRADED		39
TREND RECORDER POINTS	34	95
POWER UPGRADE ON NON-1E DISPLAYS	NONE	172
TOTAL ADDITIONAL INSTRUMENT CHANNELS	20-30	163-175

SYSTEMATIC APPROACH MISSING

- BASIC APPROACH OF ANS 4.5 ABANDONED BY GUIDE
- REG. GUIDE 1.97 DOES NOT EVOLVE FROM BASIC FUNCTIONAL CRITERIA/ANALYSES
- TABLES MANDATE UNJUSTIFIED DIVERSITY REQUIREMENTS ON FUNCTIONAL LEVEL

SCOPE EXPENSION UNSUPPORTED

- ANS 4.5 CONTROL ROOM OPERATOR ORIENTED
- REG. GUIDE 1.97 EXTENDS SCOPE TO ENTIRE OPERATING ORGANIZATION
- NO BASE DOCUMENT REFERENCE OR FUNCTIONAL REQUIREMENTS ARE IDENTIFIED IN GUIDE FOR INCREASED SCOPE
- INCREASED SCOPE IS JUST NOW BEING FUNCTIONALLY DEFINED

SCOPE EXPANSION BLURS AMI FOCUS

- PRIOR AMI DEFINITION RECOGNIZED SCOPE LIMITATION (REG. GUIDE 1.97, REV 1)
- TYPE D & E VARIABLES NOT FUNCTIONALLY ESSENTIAL
- TYPE D VARIABLES WOULD MORE APPROPRIATELY BE ADDRESSED AS PART OF A STANDARD ON SAFETY SYSTEM REQUIREMENTS
- TYPE E VARIABLES FOR "DEFENSE-IN-DEPTH AND DIAGNOSIS" ARE INAPPROPRIATE

REQUIREMENTS OVERLY PRESCRIPTIVE

- SINCE REG. GUIDE 1.97 IS NOT BASED ON FUNCTIONAL REQUIREMENTS AND A PLANT ANALYSIS, BLIND COMPLIANCE IS REQUIRED
- COMPLEXITY IS NOT NECESSARILY A VIRTUE
- DETAIL DESIGN REQUIREMENTS OFTEN UNJUSTIFIED AND/OR BEYOND THE EXISTING STATE-OF-THE-ART
- APPROACH COUNTER TO KEMENY COMMISSION ADMONITION
- REG. GUIDE 1.97 SHOULD ADDRESS AMI FUNCTIONAL REQUIREMENTS NOT DESIGN THE SYSTEM

5 Aug 80

R.G. 1.97 REASONS FOR CORE EXIT TEMPERATURE FOR BWRs

- INDICATE POTENTIAL FOR OR ACTUAL FUEL CLAD BREACH.
- MEASURE EXTENT AND TREND OF CORE DAMAGE
    - 5-10% CORE BLOCKAGE WITH NO ECCS

CDS-1
8/6/80

CURRENT VARIABLES WHICH INDICATE CLADDING BREACH

- CLADDING BREACH OCCURS WHEN:
  - HIGH CLADDING TEMPERATURE/HIGH HOOP STRESS
    - BWR RUPTURE TEMPERATURE 2200F
  - CLADDING OXIDATION
- VARIABLES INDICATIVE OF BREACH
  - HIGH HYDROGEN LEVELS
  - HIGH STEAM LINE RADIATION
  - FISSION PRODUCTS IN REACTOR COOLANT/ CONTAINMENT AIR/SUPPRESSION POOL WATER
  - OFFGAS RADIATION LEVELS
  - LOW WATER LEVEL
  - LOSS OF MAKEUP
- CURRENTLY MEASURED VARIABLES
  - PROVIDE DIVERSITY
  - UNAMBIGUOUS INDICATION
  - QUALIFIED AND TESTED

MANY CURRENTLY MEASURED VARIABLES ALREADY PROVIDE INFORMATION ABOUT CLADDING BREACH

EVALUATION OF SITUATIONS WHERE TC's MIGHT BE USED

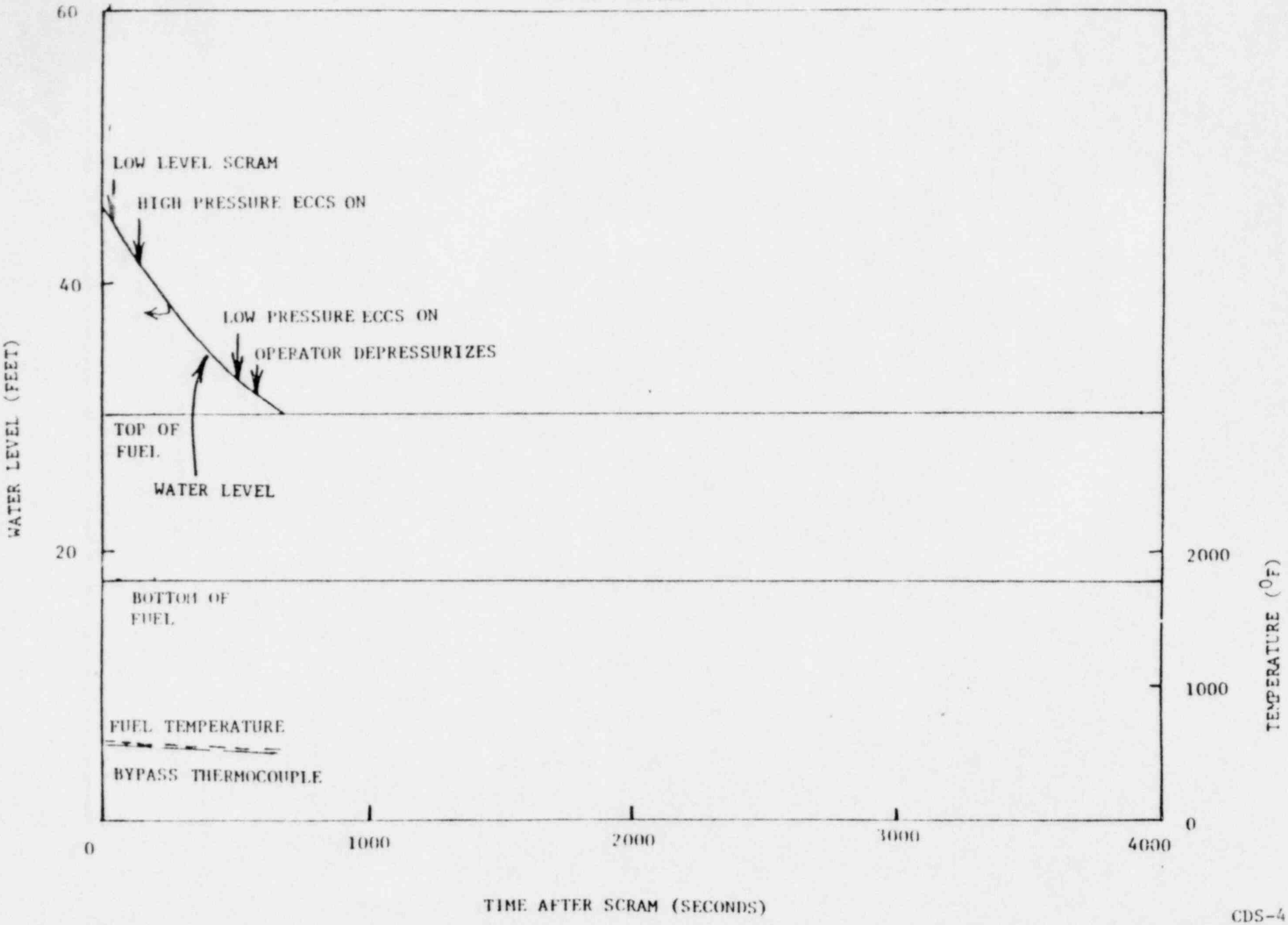
- PRIOR TO ~~CLADDING BREACH~~ core uncovery
  - BWR OPERATES SATURATED
  - WATER LEVEL KEY VARIABLE THAT DETERMINES ECCS INITIATION AND OPERATOR ACTION
- DURING CORE HEATUP
  - ONLY USEFUL IF
    - WATER LEVEL BELOW TOP OF FUEL AND NO MAKEUP
  - NOT USEFUL WHEN
    - CORE SPRAYS OPERATING
    - TWO PHASE MIXTURE IN UPPER PLENUM (CCFL)
    - WATER LEVEL ABOVE CORE
- DURING RECOVERY PHASE
  - NATURAL CIRCULATION NOT A CONCERN
  - OPERATOR REQUIRED TO DEPRESSURIZE AND MAINTAIN LEVEL
  - NO CORE DAMAGE PROPAGATION WHEN CORE COVERED
    - NUMEROUS PATHS FOR FLOW PER BUNDLE
  - TC's WILL NOT INDICATE ABOVE SATURATED

TC's ONLY USEFUL WHEN CORE AND UPPER PLENUM COMPLETELY EMPTY

CDS-3
8/6/80

BYPASS THERMOCOUPLE RESPONSE FOR AN EXTREMELY DEGRADED CASE

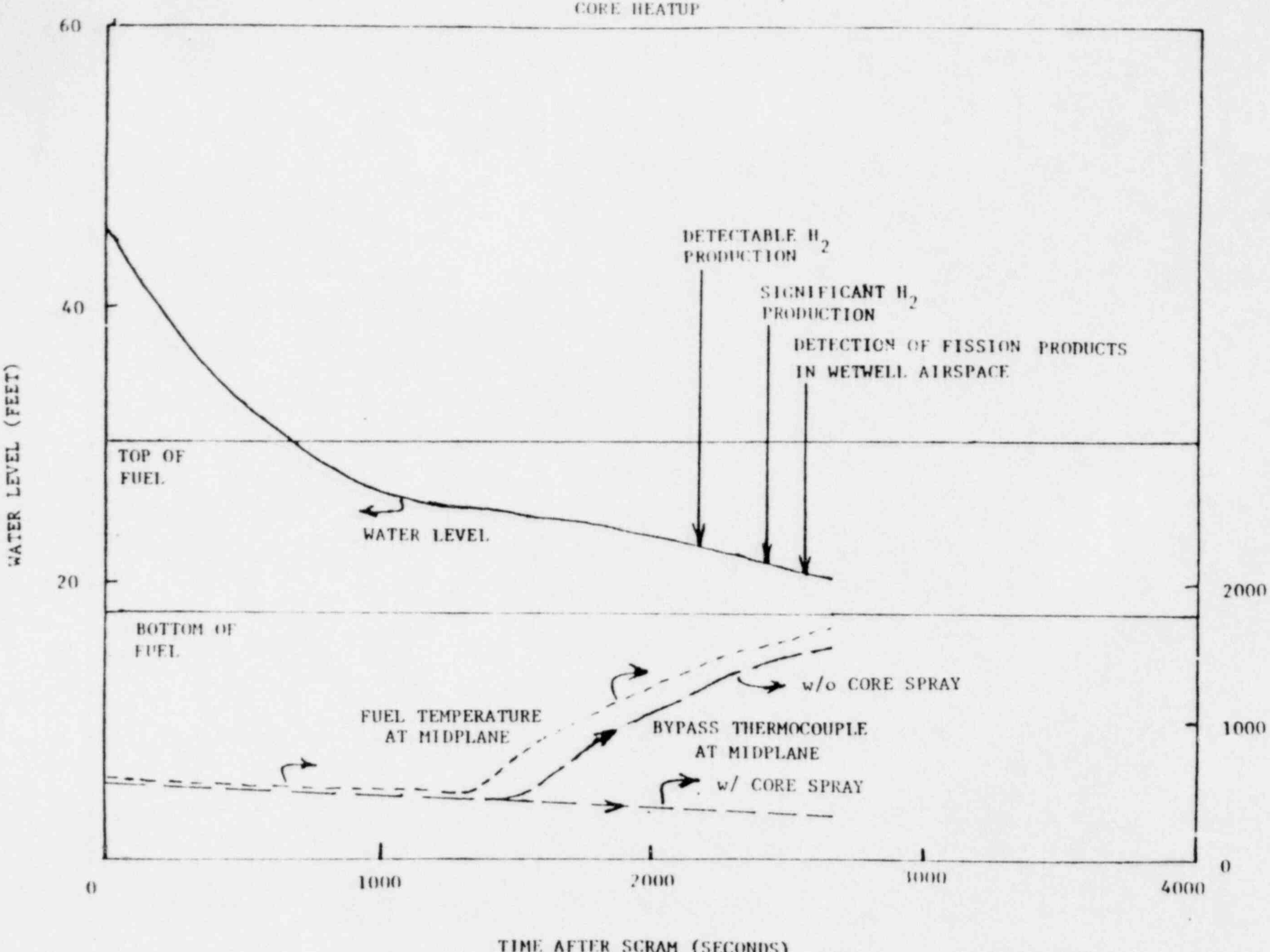
CORE COVERED



CDS-4
8/6/80

BYPASS THERMOCOUPLE RESPONSE FOR AN EXTREMELY DEGRADED CASE

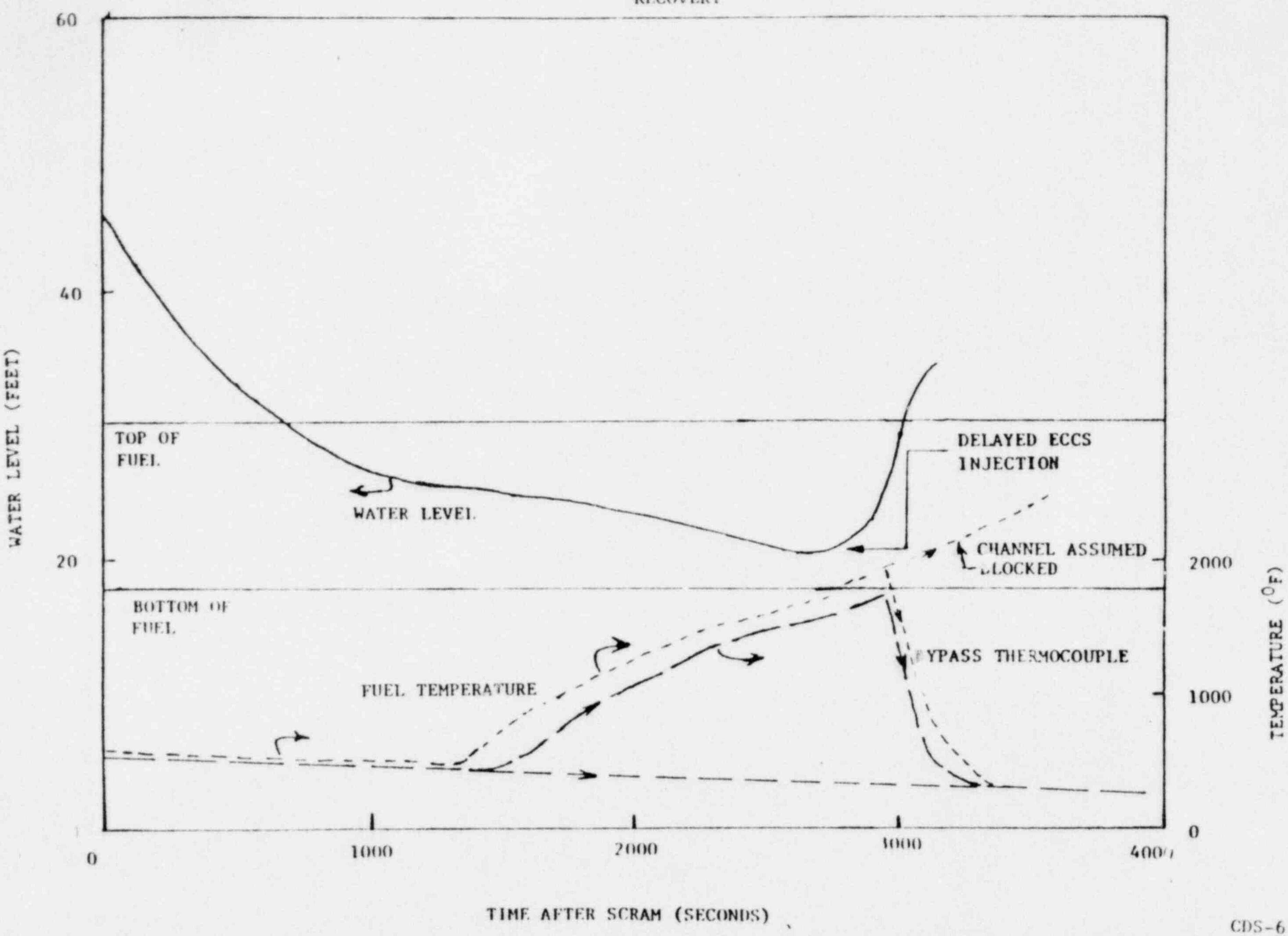
CORE HEATUP



CDS-5
8/6/80

BYPASS THERMOCOUPLE RESPONSE FOR AN EXTREMELY DEGRADED CASE

RECOVERY



CDS-6
8/6/80

CONCLUSIONS

- CONSIDERABLE NUMBER OF VARIABLES ALREADY MEASURED WHICH INDICATE (POTENTIAL FOR) CLADDING BREACH.
- NO CORE DAMAGE PROPAGATION EXPECTED.
- TEMPERATURE MEASUREMENT NOT A RELIABLE INDICATOR OR EXTENT AND TREND OF CORE DAMAGE.

REG. GUIDE 1.97 (DRAFT 2 - REV. 2)

BWR COMMENTS

PURPOSE

- o PROVIDE BWR COMMENTS ON DRAFT 2
- o DISCUSS TECHNICAL ASPECTS OF CORE EXIT TEMPERATURE MEASUREMENT REQUIREMENT

CONCERNS

- o REVISION TO REG. GUIDE 1.97 INAPPROPRIATE NOW
- o CORE EXIT MEASUREMENT NOT NECESSARY FOR BWR
- o ADDITIONAL SPECIFIC COMMENTS IN POSITION PAPER

8/1/80

SUMMARY OF RECOMMENDED REG. GUIDE 1.97 CHANGES

- o REFLECT UNIQUE BWR FEATURES
- o PROVIDE VARIABLE SELECTION CRITERIA
  - INTEGRATE WITH PROCEDURE GUIDELINES
  - INTEGRATE WITH NUREG-0696
  - FOCUS ON KEY VARIABLES
- o ELIMINATE MARGINAL VARIABLES (NOTABLY <u>CORE</u> <u>EXIT</u> <u>TEMPERATURE</u> MEASUREMENT FOR THE BWR)

8/1/80

IMPACT OF CORE DISCHARGE TEMPERATURE MEASUREMENT

O BASIS: TC IN PRM ASSEMBLY

O COST: DOLLARS

- $400K/PLANT - FORWARD FIT

  TOTAL FOR 33 PLANTS: $13 MILLION

- $600K/PLANT - BACKFIT

  TOTAL FOR 25 PLANTS: $15 MILLION

  AGGREGATE FOR 58 PLANTS: **$28 MILLION**

O COST: DOSE

- MAINTENANCE = 8 MAN REM/YR/PLANT - ALL PLANTS;

  58 PLANTS X 40 YRS X 8 = 18,500 MAN REM FOR TOTAL PLANT LIFE

- INSTALLATION - 100 MAN REM/PLANT - BACKFIT

  25 PLANTS X 100 = 2500 MAN REM TOTAL

  GRAND TOTAL: **21,000 MAN REM**

COST HIGH FOR VERY MARGINAL BENEFIT

8/1/80

GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125

August 4, 1980

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

Attention: Sam Duraiswamy

SUBJECT: REGULATORY GUIDE 1.97 (DRAFT 2 OF REVISION 2) - BWR COMMENTS

References:

1) GE letter Buchholz, R. H. to Secretary of the Commission, "Comments on the Draft of Revision 2 to Regulatory Guide 1.97", dated March 28, 1980.

2) GE letter Sherwood, G. G. to Roger J. Mattson, "Draft Regulatory Guide 1.97, Revision 2", dated May 30, 1980.

Gentlemen:

This letter is written to provide the ACRS comments from the BWR perspective on Draft 2 of Revision 2 to Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following and Accident". Specifically, Attachment 1 provides the technical basis for not requiring core exit temperature measurements in the BWR; Attachment 2 provides several additional recommendations for changes to Draft 2 necessitated by recognition of specific BWR design features. Marked-up copies of Tables 3 & 3A in Draft 2 are provided in Attachment 3 to show suggested modifications. Attachment 1 has been reviewed by and is submitted on behalf of the BWR Owners Group.

General Electric has provided the Staff comments on all aspects of Revision 2 in References 1 and 2. The Attachments in this submittal are focused on those proposed requirements which are particularly inappropriate for the BWR. Emphasis is placed on the core exit temperature measurement because of its extreme cost from both a personnel exposure and a dollar viewpoint, compared to its value. General Electric has systematically reviewed use of core exit temperature measurement and concluded that only in the case of core uncovery with no normal, emergency, or alternate water make-up systems available to replenish vessel inventory would unambiguous and definitive information be provided the operator. As discussed in Attachment 1 even in this case there are several other indications available which provide ample indication of propagating core damage.

General Electric and the BWR Owners Group recognize the need to take all necessary actions identified as the result of post TMI evaluations. However, as discussed in Attachment 1 the requirement for core exit temperature measurment is unnecessary when BWR design features and capabilities are systematically considered. We look forward to further discussion of this matter with the ACRS during the August 6, 1980 meeting.

Very truly yours,

R H Buchholz

R. H. Buchholz, Manager
BWR Systems Licensing

Attachments

cc: T. D. Keenan
D. Waters
BWR Owners Group
L. S. Gifford
P. W. Marriott
R. J. Mattson
A. S. Hintze
V. Benanoya

ATTACHMENT 1

GENERAL ELECTRIC AND BWR OWNER'S GROUP COMMENTS TO THE ACRS

ON CORE EXIT TEMPERATURE MEASUREMENT PER

DRAFT 2 OF REVISION 2, REGULATORY GUIDE 1.97

1. Technical Basis

1.1 Background

The reason cited in R.G. 1.97 for requiring core exit temperature measurement for accident monitoring is to indicate the potential for, or actual occurrence of, fuel cladding breach. The NRC staff has also indicated that they desire to identify local hot areas and the propagation of core damage; they have suggested that approximately 50 thermocouples should be utilized. This quantity is felt by the staff to be sufficient to detect blockage of 5-10% of the core with no core spray (or other ECCS) at a high confidence level and with a sufficient allowance for attrition.

1.2 Detection of High Core Temperature

In assessing the plant safety improvement resulting from core exit temperature measurements, several periods during the course of an event must be evaluated. The first period is prior to core uncovery. The BWR operates under saturated conditions with very strong natural circulation inside the reactor pressure vessel. Studies (Reference 1) have shown that, as long as the core remains covered with water, adequate core cooling is assured. Therefore, for there to be a cladding breach, there must first be a challenge threatening to uncover the core. Thus, reactor water level is a key parameter on which both automatic and operator actions are based. Water level is also the primary measure of accomplishment of the core cooling safety function during accident situations. The BWR provides multiple and redundant water level instrumentation for these purposes. During this time period, core exit thermocouples would be indicating, at most, saturation temperature corresponding to the reactor vessel pressure. Core exit thermocouple readings would probably be erratically indicating lower temperatures due to the subcooling effect of ECCS (core spray and LPCI). The use of core exit thermocouples would not provide useful additional information for the plant operator and the erratic readings may be confusing.

The second time period when knowledge of core exit temperatures might be useful is during fuel heatup following core uncovery. It is during this time that the potential for cladding breach exists, and, depending on the duration and amount of core uncovery, the potential exists for creating local flow blockage as a result of core damage. Reactor vessel water level provides the ability to detect core uncovery and, thus, by itself, indicates the potential for cladding perforations. Automatic and operator manual actions would already be underway to restore water level to cover the core. Continued monitoring of reactor water level and water makeup system performance parameters provides the capability for monitoring this critical safety function.

There are many other parameters available to the operator that are reliable indicators of actual fuel clad breach. These include high steam line radiation, high offgas radiation levels, high area radiation levels in the containment, high hydrogen concentration in the containment, and high radioactivity in reactor or suppression pool water. Details of these current provisions are discussed in Section 1.3.

Core exit temperature measurement will not provide an unambiguous indication of either the potential for or actual clad damage. This results since the BWR's multiple, safety-grade core spray systems would continue to supply water spray over the top of the core even though the core may be uncovered in a bulk sense. Even if there is only one core spray system functioning (out of two provided), the core exit temperature, whether measured locally or in bulk, will not be superheated. The core sprays need only provide 300 gallons per minute of their total typical design flow rate of 12,000 gpm to remove any superheating in the steam. In the BWR 5 and 6 designs, the Low Pressure Coolant Injection (LPCI) system directly floods the core bypass region, providing further subcooling. The Staff contends that these ECCS functions not be considered when determining the merits of core exit temperature measurement; that contention is unreasonable.

During fuel heat-up following core uncovery, there is only one condition for the BWR that a core exit temperature measurement would provide unambiguous and definitive information useful to the operator. This occurs in the highly unlikely event that, following a loss of water invertory, no normal, emergency, or alternate water makeup systems are available to replenish coolant inventory to the pressure vessel. During this situation the core is cooled by water and steam flow for a considerable period of time until the water in the core region is boiled off. Under such conditions, measurement of steam superheat anywhere above the core region would indicate core heatup and a low water level. However, should this condition occur the operator would be taking all appropriate actions to restore water level above the core based only on knowledge that water level is low and no injection is available.

The third time period, called the recovery phase, covers the interval after the operator has restored the water level in the core region. If there were no significant core damage, core exit temperature measurement would not provide any relevant information. The possibility of thermocouples providing useful information for operator actions has been raised by the Staff for the situation when 5-10% of the core is damaged. The Staff contends that high core exit temperature readings would indicate localized propagating core damage and guide the operator in long term decision making.

This position is unreasonable because: (a) once water level is restored in the core, core damage will not propagate to the rest of the core from the postulated 5-10% damaged core, and (b) temperature readings would not provide relevant information. A detailed discussion of both these points follows.

Core damage propagation, when the core is covered, has been discussed in a Licensing topical report (Reference 2). Because each bundle in the

BWR core is surrounded by a flow channel, cross-flow between bundles is eliminated and any thermal-hydraulic effects of localized core damage remain localized. Each channel forms an essentially independent flow path connecting the upper and lower plena and the core bypass region. To assure no damage to an undamaged fuel assembly, less than one gallon of coolant per minute must be provided. Since there are three independent flow paths into each fuel assembly (the top and bottom of the fuel bundle, and the flow paths between the bundle and bypass), any core damage propagation must start by almost complete blockage of all these paths. Calculations have been performed which show that all three paths have to be greater than 99% blocked for any damage to result. Even if almost total flow blockage of the bundles were postulated, this situation would not be likely to persist for long. Localized heating of the cladding would result in molten cladding coming in contact with the channel wall. Such localized heating of the channel would eventually form a hole in the channel, thus opening another flow path for the coolant from the bypass region to enter and cool the fuel rods.

Calculations have also been performed for the situation with 5-10% core damage and with an uncoolable geometry postulated to determine if superheated steam can be detected in the region around the damaged portion of the core. The calculations were done assuming the available instruments were those directly adjacent to the bundles in the damaged core region. The analyses show that the heat generation (decay heat and heat from metal water reaction) in the post-recovery phase are so low that, under all situations analyzed, nucleate boiling would be maintained and no superheat would be measured in the bypass region surrounding the damaged core.

It has been suggested by the NRC staff that if a temperature sensor was located adjacent to the assumed local blockage and if it were postulated that it could indicate some superheat, the operator could restart recirculation pumps. This would then force coolant through the partially blocked flow paths. However, as indicated above superheat would not be observed and the operator would have no knowledge that this action is necessary. In addition, because of the strong inherent natural circulation in the BWR, this action would be likely to be helpful for only a very limited situation where greater than 99% but less than 100% of all available flow paths were blocked. Therefore, operator actions would be no different: the principal emphasis would still be only on maintaining reactor water inventory. The addition of 50 thermocouple data readouts may, indeed, add to operator confusion such that the reliability of operator action is reduced.

The most practical location to install thermocouples in a BWR is in the in-core power range monitor (PRM) instrument assemblies. All other locations (see Section 2) would require additional penetrations and major redesign of the vessel internals and/or the fuel bundles. A review of the temperature response of a thermocouple in the PRM assembly indicates that it would only provide an indication of gross core discharge superheat conditions in the highly unlikely event that no water makeup systems were operating for an extended period. But for such a situation, as discussed above, a single thermocouple anywhere above the core would provide comparably useful information as to the existence of a bulk superheat condition. Figure 1 shows the response of the various variables already available to the operator to guide his actions during a core uncovery event. It also shows the expected temperature response of thermocouples in the PRM tubes if there should be no normal, emergency or alternate water makeup systems of any kind in operation. The comparisons show that the operator already has multiple and unambiguous indications to guide his actions during the core heat-up time period.

1.3 Detection of Propagating Core Damage

For the worst-case assumptions (i.e. uncovered core and no make-up) for which the NRC staff proposes that thermocouple indication would be useful, alternate means are available to provide trend information relating to the possible propagation of core damage (PCD). Those means which were previously available or are presently required by R.G. 1.97 and NUREG 0578 and provide direct indication of PCD, with or without ECCS functional, include: (1) reactor and suppression pool water/containment air sampling and analysis for radioactive material, (2) containment gross gamma monitoring, and (3) containment hydrogen monitoring. Other measured variables required in R.G. 1.97 could also be used to infer PCD.

Analysis of reactor water samples would measure fission product activity and the concentration of dissolved hydrogen in the reactor water. The fission product activity from the gap/plenum would be released within several minutes after the onset of fuel clad perforations. It is expected that the reactor water sampling system will be sufficiently sensitive to detect the hydrogen concentration resulting from the reaction of as little as four pounds of zirconium. This is equivalent to a metal-water reaction involving about 3% of the cladding of a single fuel bundle.

For the dry-core case, vessel depressurization is expected. It will occur naturally if the event is initiated by a primary system break of sufficient size. It will occur by automatic or manual actuation for the no-break or small-break case because of safety/relief valve (S/RV) actuation. Thus, for the entire spectrum of initiating events, indication of core damage will be provided by various instruments in the containment. These include the suppression pool water/containment air sampler system, gross containment gamma monitor, and the containment hydrogen monitor. The gross gamma monitor would detect fuel clad gap/plenum activity release within several minutes from the onset of clad perforation.

Activity due to noble gases alone should provide sufficient indication of PCD. For the relatively straightforward case involving blockage of a single fuel assembly during normal plant operation, analysis (Reference 2) shows that within 9 seconds, fuel element melting would be detected by the steam line radiation monitor; scram and steamline closure would follow within 4 seconds. The off-gas radiation monitor would alarm within two minutes.

The more complex case involving main steam isolation valve (MSIV) closure for reasons other than high steam line radiation has also been investigated. For this case, the safety relief valves (S/RV) open within seconds to relieve vessel pressure, and noble gases are transported via the S/RV discharge piping to the suppression pool water, then released to the containment free volume. The results of this analysis are illustrated in Figure 1 for the situation in which all reactor water makeup systems (normal, emergency and alternate) are postulated to remain inoperative for an extended period. Eventually the water level is reduced such that the readings on all thermocouples would increase with a distribution related to the core power distribution. For the situation in which the bulk water level has been significantly reduced there would be little or no correlation between thermocouple readings and core area cross sectional blockage. In this case the insufficient reactor water inventory would affect all fuel assemblies independent of whether or not blockage exists. The extent to which actual fuel failures occur could only be assessed by monitoring fission gas release to the primary system or the containment. Gross gamma monitoring should provide a more rapid indication of PCD for purposes of operator action. Confirming indications of the rate of PCD will be provided by the suppression pool water/containment air sampler system, as well as the containment hydrogen monitor. The containment hydrogen monitor is expected to be sufficiently sensitive to detect PCD as low as 2 to 3% core-wide, metal-water reaction per day.

2. Design and Operational Considerations

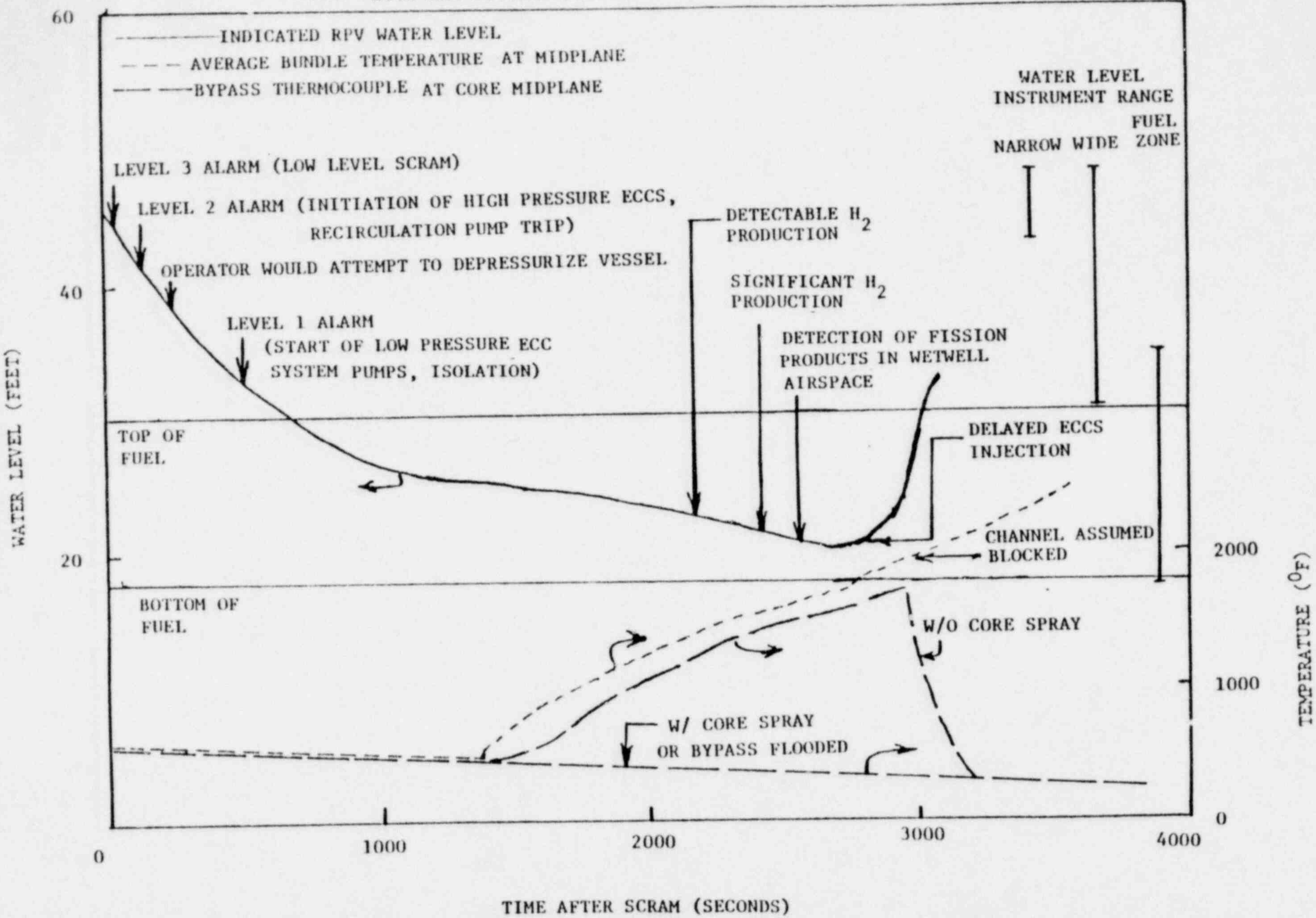
There are three possible locations for thermocouples within a BWR. These are: within or on the fuel assembly; on the shroud head with leads projecting downward to near the fuel assembly discharge; and in the PRM assemblies. While detailed design investigations have not been performed, the first alternate is considered unacceptable since it would create localized flow disturbances and cladding stress concentrations with the potential for initiating fuel damage. Both the first and second alternatives are also considered unacceptable due to the interference created between the thermocouple lead supports and the ECCS function - specifically core spray. They create an extremely difficult vessel and vessel internal design problem because of the multiple penetrations required in order to route the thermocouple leads. These alternatives could significantly impact the duration of each refueling outage. For both, the number of thermocouples required could be large, since the BWR utilizes a channeled fuel design which, as previously discussed, prevents propagating core damage.

Only placement in the PRM assemblies is technically feasible without extensive plant redesign. The PRM assembly is inserted into the reactor vessel from above the core with the vessel head and separator and dryer assemblies removed in earlier BWR designs, and from below the core in the BWR 6 design.

In both BWR/6 and pre-BWR/6 designs, the PRM assemblies are secured to the top grid within the vessel. The top of the PRM latches approximately 10 inches below the top of the channel of the fuel assembly. The PRM latching mechanism design precludes locating the thermocouple higher than approximately 13 inches below the top of the fuel channel.

FIGURE 1

COMPARISON OF BYPASS THERMOCOUP. RESPONSE TO SIGNIFICANT VARIABLES CURRENTLY AVAILABLE TO THE OPERATOR FOR A DEGRADED LOSS OF INVENTORY EVENT WITH ALL SYSTEMS FAILED



To withstand the 200-day, post-accident drywell environment of radiation, spray, and immersion for BWR/6, requires metal-sheathed cabling with waterproof connectors from the vessel through the containment penetration. Based on preliminary design considerations, a minimum of two connectors -- one located at the bottom of and the other about one or two feet below the in-core housing flange--would be required for each PRM to permit its replacement. Difficulties are expected during both maintenance and installation.

Making, breaking, and testing for leak tightness of the thermocouple connectors is estimated to require 10 minutes each (with allowance for occasional stripping of threads and lead breakage) during maintenance. An appropriate means for leak testing each connector has not been developed, and end-to-end testing of the metal-sheathed cable may be required. It is therefore estimated that 40 minutes extra (due to thermocouple addition) would be required each time a PRM assembly was replaced. For pre-BWR/6 units, average PRM assembly replacement is expected to be 25% per year; for BWR/6, PRM assembly replacement is estimated as 15%/year (limited by life of the thermocouple or structural deterioration of the assembly). For an 1100 MWe plant utilizing 41 PRM assemblies, the manhour exposure for a crew of three would be:

Pre BWR/6 - 3×0.67 Hr. $\times .25 \times 41 \cong 20.0$ manhours
BWR/6 - 3×0.67 Hr. $\times .15 \times 41 \cong 12.0$ manhours

Actual dose rates under the vessel vary from plant to plant; from 40 mr/hr to over 300 mr/hr have been observed. Thus, the plant annual personnel exposure would be expected to increase by ~0.8 to 6 man-rems/year for pre-BWR/6 plants and ~0.5 to 4.0 man-rems/year for BWR/6 plants.

Also, additional personnel exposure can be expected as a result of increased control rod drive (CRD) removal complexity. The presence of the thermocouple leads would further restrict personnel space availability and increase the possibility of damage to the cable leads and connectors during drive removal and replacement. Detailed studies and field experience would be required for a complete assessment, but some increase (perhaps 10 minutes) in CRD servicing time can certainly be expected. Such an increase would result in an exposure time increase for a crew of four of 40 minutes per drive, or a total increase of $0.25 \times 180 \times .67 = 30$ manhours per year. The annual plant personnel exposure increase would be in the range of 1.2 to 9 man-rems/year.

The total annual plant personnel exposure increase due to PRM, thermocouple and control rod drive maintenance would be in the range of 2 to 15 man-rems/year for pre-BWR/6 plants and 1.7 to 14 man-rems/year for BWR/6 plants.

For installation, thermocouple leads would require routing from under the vessel in four separate arrays of about ten leads each, with the thermocouple leads distributed inside the pedestal in such a manner that each bundle would contain leads from the thermocouples located in each core quadrant. Complete isolation of these leads from the consequences of a specific accident is not feasible in operating plants, and is also thought to be unfeasible for plants under construction and design. Each of the four bundles of thermocouple leads is assumed to be routed through the containment in a structural housing to provide some protection during the accident (e.g. jet impingement). Assuming two penetrations can be made available through which the thermocouple leads could be brought through containment, the installation of the leads in the containment is expected to take about 2,000 installation manhours. It should be noted that spare penetrations may not be available on operating plants considering other current NRC requirements. Including installation, modification engineering, and field engineering, the cost is approximately $300,000* per plant.

Installation outside the drywell is assumed to be in a two-bundle configuration, with Division I power to one bundle and Division II power to the other bundle. Four multi-point recorders in the control room are assumed, although this is uncertain considering that the readings may be significantly delayed and illegible (due to similarity of readout).

On this basis, total installation cost is estimated to average $600,000* on operating plants and $400,000* on plants in construction. Exposures to installation personnel in each operating plant is estimated to be 100* man-rems assuming a 50 mr/hr general radiation field.

Excluding prototype testing, it is estimated that initial shipments of PRMs including thermocouples could begin 18 months after design initiation.

Note, application of the single-failure criterion of Table 1, Item 2 of R.G. 1.97 would eliminate readings from 50% of the thermocouples and accident consequence criteria could eliminate readings from another 25%. This presumed loss of installed thermocouples is of little consequence, since as previously discussed, exit thermocouples will be of little use in detecting local fuel temperature. Only 25% of the thermocouples (assuming 50 total) would still indicate bulk core uncovery with no water makeup. Even this function is of little value, but at least in this sense, it is concluded that the single failure criterion can be met.

* These estimates are approximate. Precise definition would require plant by plant assessment. Probable accuracy: ±50%

3. Regulatory Requirements

Because of their marginal usefulness and associated design and installation problems, core exit thermocouples, if required, should have a Design and Qualification category no more severe than 4 (Reference R.G. 1.97, Table 1) since,

a) As previously discussed, thermocouples cannot provide an effective indication of core cooling and would not provide a reliable additional basis for operator action.

b) Any meaningful thermocouple reading would occur long after other core damage indications have become evident. On-demand scanning of the thermocouples should be more than adequate. Continuous readout of thermocouple data could further confuse the operator as to true core status.

c) It is not possible to meet all the Category 1, 2 or 3 criteria assuming that one thermocouple per PRM assembly is required.

d) It is unrealistic to postulate the occurrence of an SSE level earthquake simultaneous with an event in which all ECCS are also presumed inoperative.

Finally, if core exit thermocouples are to be required, they should not be specified via a Regulatory Guide. The requirement should be deleted from Regulatory Guide 1.97 and the issue added to the scheduled core damage rulemaking.

4. Conclusion

It has been determined that core exit thermocouples provide only marginally useful additional information to the operator. Moreover, the only practical location for their installation in any plant (operating or in design) would result in no significant enhancement of the operator's ability to protect the plant or public.

The combination of existing or planned (as a result of R.G. 1.97) instrumentation is sufficient to detect not only the presence of PCD, but also its rate and trend without core exit thermocouples. This is true for all possible loss of primary system coolant events independent of ECCS operational combinations. Detection is expected to occur within several minutes following initial clad perforations with PCD trend detection capability extending beyond 100 days.

The introduction of thermocouples in the PRM assemblies constitutes not only a significant design problem, but also subjects plant personnel to increased radiation exposure.

For all of these reasons, core exit thermocouples should not be required in boiling water reactors in operation or design.

REFERENCES

Reference 1. NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," August 1979

Reference 2. Licensing Topical Report: Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor NEDO-10174, October 1977

ATTACHMENT II

GENERAL ELECTRIC COMMENTS TO THE ACRS

ON OTHER PROVISIONS OF DRAFT 2 OF REVISION 2 TO REGULATORY GUIDE 1.97

1. Reactivity Control by Neutron Flux

Tables 3 and 3A in NRC Regulatory Guide 1.97 (Draft 2, Rev. 2) would require that neutron flux be monitored as a means of providing the plant operating staff with an indication of the reactivity control status of the core. The guide also would require that instrumentation meeting design and qualification Category 1 be provided to monitor neutron flux over a very wide range from one count per second (cps) to 1% (rated) power corresponding to a core average neutron flux measurement range from 10^{-11} to 10^{-2} of full reactor power. ANS Standard 4.5 requires flux measurement over the range from 10^{-8} to 10^{-3} of full reactor power. The neutron flux range requirements of R.G. 1.97 exceed that which is available for a fixed position detector.

The intent of these requirements appears to be the assurance that core average thermal neutron flux can be reliably defined as decreasing, constant or increasing, over a range extending from a significant power level to somewhat below the minimum neutron flux at initial criticality. The rate of flux change would allow the calculation of reactor period, and the absolute count rate can be compared to that of previous reference values at various conditions to infer roughly whether the core is fully shutdown. It is assumed that 1) basic core geometry is maintained, 2) the bottom head and core volume are flooded at least to the level corresponding to the top of the jet pumps, and 3) localized core region voiding does not occur near the detector.

Experience has shown that the full-in SRM count rate at initial criticality due to decay of spontaneous neutron sources is approximately 1000 cps at 200 days corresponding to 10^{-8} of rated core average thermal flux. With a six decade SRM range capability, the full-in SRM reading would be off-scale at approximately 10^{-5} of rated core average thermal neutron flux. Withdrawal of the SRM at initial criticality to reduce the count rate by a factor of 10^{-3} would allow a full scale reading at approximately 10^{-2} of rated core average thermal neutron flux. Although detailed analyses have not been performed, the neutron flux at this positon is expected to increase by several orders of magnitude at rated power and could significantly shorten detector life.

An approach which is considered to meet the intent of R.G. 1.97 is to power the SRM's in core Quadrants 1 and 3 by Division I power, and those in core Quadrants 2 and 4 by Division II power. One each of the Division I and II power SRM's will be inserted such that full scale (10^{6} cps) corresponds to approximately 10^{-2} rated power. One each of the Division 1 and 2 SRM's will be withdrawn somewhat further such that full scale corresponds to approximately 10^{5} cps at 1% power. Since the SRM drive mechanisms are not seismically or environmentally qualified to operate in the accident environment and are not powered by a Class IE power supply, the SRM position would remain fixed (locked in) during all modes of plant operation, except startup and possibly refueling. The more fully inserted SRM's can measure as low as 10^{-8} of core average flux. The more fully withdrawn SRM's should have sufficiently long life to assure that period can be assessed down to approximately 10^{-7} of rated core power.

This approach meets the intent of R.G. 1.97, including the effect of a single active failure in the power supply system. However, it will not meet the R.G. 1.97 requirements for those hypothesized specific accidents for which the resulting mechanical consequences could disable one of the two signal cables of the single power division remaining after the postulated single active failure. This approach will not necessarily satisfy the requirement for one cps minimum sensitivity because: (1) burn-up of the inserted detectors will reduce their sensitivity, (2) neutron flux at the location of the withdrawn detector may be too low, and (3) neutron flux will decay at 200 days to a very low level.

Sufficient analyses have not as yet been performed to assess fission chamber neutron flux for various core lifetimes and rod patterns. Preliminary indications are that the proposed approach could reduce the life of the two more fully inserted fission chambers to approximately one year, while the life of the other two would be approximately five years (versus the present 10-year or more life.) This would represent a significant increase in SRM replacement expense.

The BWR employs four SRM chambers, one located in each quadrant of the core. The sensor, electronics and mechanical/structural support portions of the assemblies do not now meet R.G. 1.97 Category 1 requirements. Specifically, the following changes would be required:

1. Seismic redesign and qualification,

2. Signal cable and connector upgrade to meet long-term high radiation and water immersion service,

3. Connection to on-site emergency power. (Two SRM's each on Division I and II power.)

The feasibility of seismic upgrade and emergency power provision is uncertain. Approximately six months will be required to assess seismic feasibility and, if qualification is not possible, a substantial redesign may be required. The other design changes appear to be technically achievable.

Therefore, it is recommended that the following changes be made to R.G. 1.97:

1. The range of neutron flux measurement should be revised to reflect a range of approximately 10^{-8} to 10^{-2} of core average thermal neutron flux at rated thermal power. This change more properly reflects a measurable design criteria and clarifies that SRM's need not be calibrated to core power.

2. Add a note to Tables 3 and 3A identifying that a reduced range of five decades is acceptable for the low probability condition in which the specified event could disable one neutron flux measurement channel. Since this would occur only when the single active failure was hypothesized to disable two of the neutron flux channels, no significant impact on plant safety would result. In addition, even one SRM channel is sufficient to eventually detect whether any part of the core is supercritical. Operator action (eg., to initiate boron injection) still could be accomplished prior to the generation of sufficient core power to produce core damage.

2. <u>Main Steam Line Flow</u>

Regulatory Guide 1.97, Table 3, requires that main steam line flow be monitored to provide an indication that the core cooling function is being performed. For a BWR, there is no relationship between the accomplisnment of core cooling and the presence or absence of steam flow in the main steam lines since 1) MSIV closure will occur, and 2) steam flow is independent of inventory provided there is water in the vessel.

Main steam flow recording is provided in the BWR control room; however, the design and qualification criteria for the control room readout are roughly equivalent to R.G. 1.97, Category 5, rather than Category 1 as is required.

Since there is no known relationship between steam flow and core cooling in the BWR, the main steam flow requirement of R.G. 1.97 is unwarranted and should be deleted for the BWR.

3. <u>Core Coolant Level in the Reactor</u>

The GE design provides a Category 1 water level measurement and indication to approximately the top of active fuel rather than bottom of the core support plate to assure initiation of all necessary safety functions and provide appropriate operator information. R.G. 1.97 should be revised to make a less stringent criteria category apply to the range of water level from the bottom of the core support plate to top of active fuel. Full range redundant indication is available when offsite power is available. Even if water level indication below the top of the fuel was not available, low water level indication on the Category I instruments (i.e. those for water level above the core) would be unequivocal indication that full ECCS should be maintained. Operator action would be no different even if he knew from an instrument reading that the water level was below the top of the core. It is excessively conservative to impose the higher water level measurement requirements for the low probability occurrence of the Design Basis Accident simultaneous with loss of offsite power and ECCS functions.

4. <u>Primary Containment Pressure (Drywell)</u>

For the Mark III plant, the drywell is not the primary containment; hence, the word "Drywell" should be deleted. However, for Mark I and II plants, the drywell is part of primary containment and hence the word "Drywell" is appropriate.

5. Drywell Temperature for Mark III

There is no relationship between drywell temperature and the maintenance of containment integrity. This variable should be deleted from the Type B and (if for some reason desired) inserted under Type D as a Category 4 variable.

6. Containment High Range Area Radiation

The requirement for the 10^7 R/hr range to apply to the secondary containment portion of the reactor building is unjustified. A range of 10^3 to 10^4 R/hr is more than adequate for the secondary containment.

7. Emergency Ventilation Damper Positions

The dampers significant to safety are those in the openings between secondary containment and the environs. Add "between secondary containment and the environs" to that variable description in Table 3.

8. Effluent Radioactivity - Noble Gases

The words, "release points" should be added after the phrase, "reactor building or secondary containment," in order to make it consistent with that which follows, and to make it more explicit as to what is to be monitored.

9. Post-Accident Sampling Capability in Sumps

The suppression chamber is the collection point to which all drains in the post-accident mode would eventually collect and, hence, the suppression chamber would be the only meaningful measurement. Tables 3 and 3A should be revised to read, "Suppression Chamber Water" in lieu of "Sumps."

10. Type D

Accurate measurement of zero or low flow in any of the lines is virtually impossible. Note #1 should be revised to add "The accuracy should be $\pm$5% of design flow."

11. Steam Flow to RCIC

This variable is not a measure of the performance of the RCIC system and should be deleted.

12. Containment and Drywell Hydrogen Concentration

Present commercially available equipment is designed to sense hydrogen concentrations ranging from 0.1 to 10 volume percent hydrogen (dry), rather than the range of 0 to 30% specified in Table 3 of R.G. 1.97. The current range is considered acceptable since it adequately covers the range over which hydrogen is of practical importance for all planned operator actions. The range is consistent with the requirement to monitor the accomplishment of critical safety functions. Monitoring for the event for which hydrogen could be postulated to exceed 10 v/o is adequately provided for by the Containment Air H_2 grab sampling requirements specified elsewhere in Table 3 of R.G. 1.97.

Tables 3 and 3A should be revised to require a range of 0.1 to 10 v/o (dry) for this variable.

Since the containment and drywell communicate freely through vacuum breakers for all pressure suppression plants and for BWR 6 plants mixing between the two volumes is assured by the drywell mixer system, it is adequate to monitor the two volumes sequentially but not simultaneously. Simultaneous measurement would double the number of sensors required to meet the single failure criteria. A note should be added to the variable to read as follows: "Simultaneous sampling of each volume is not required. Sampling transfer from one volume to the other with a maximum sampling interruption of 30 min. is satisfactory."

This variable in Table 3A should be revised to read <u>Suppression Chamber</u> and drywell hydrogen concentration since both volumes are primary containment.

13. <u>Containment and Drywell Oxygen Concentration</u> (for those plants with inerted containments)

The recent staff decision (Facility Operating License Nos. DPR 33, DPR-52 and DP2-63 for Browns Ferry Units Nos. 1, 2 and 3 - June 22, 1978) supports the position that the control of combustable concentrations in inerted containments can be adequately accomplished by monitoring the hydrogen concentration. This is a technically appropriate position. If, for some reason oxygen monitoring is desired, it is adequately provided for by the Containment Air O_2 grab sampling requirements specified elsewhere in Table 3 of R.G. 1.97.

Tables 3 and 3A should be revised to delete this variable.

ATTACHMENT 3

GENERAL ELECTRIC'S SUGGESTED REVISIONS TO

TABLES 3 AND 3A OF DRAFT 2 OF

REVISION 2 TO REGULATORY GUIDE 1.97

TABLE 3

BWR VARIABLES

Purpose & Variables	Range	Design & Qualification Criteria Category (See Table 1)
TYPE A - Variables Which Indicate Necessity for Pre-planned Manual Actions	Plant specific	1
TYPE B - Variables Which Indicate Accomplishment of Critical Safety Functions		
Reactivity Control		
Control Rod Position	Full in or not full in	5 (for 2 hrs minimum)
Neutron Flux	10^{-8} TO 10^{-2} FULL ~~1 c/s to 1%~~ power	1
Core Cooling		
Coolant Level in the Reactor	APPROXIMATELY 2/3 TOP OF ~~Bottom~~ of core HEIGHT ~~support plate~~ to above top of discharge plenum	1
	BOTTOM OF CORE SUPPORT PLATE TO 2/3 CORE HEIGHT	4
~~Main Steamline Flow~~	~~[illegible] flow[a]~~	
	TOP OF [illegible]	
Maintaining Reactor Cooling System Integrity		
RCS Pressure	15 psia to 1500 psig	1[3]
Main Steamline Isolation Valves' Leakage Control System Pressure	0 to 15" of water 0 to 5 psid	1
Primary System Safety Relief Valve Positions, including ADS or Flow Through or Pressure in Valve Lines	Closed-not closed or 0 to 50 psig	1

[a]Design flow - the maximum flow anticipated in normal operation. THE ACCURACY SHOULD BE ± 5% OF DESIGN FLOW

TABLE 3 - (continued)

Purpose & Variables	Range	Category
TYPE B - (continued)		
Maintaining Containment Integrity		
Primary Containment Pressure ~~(Drywell)~~	10 psia pressure to 3 times design pressure[1] for concrete; 4 times design pressure for steel	2[21]
Containment and Drywell Hydrogen Concentration	0 to ~~30%~~ 10 (capability of operating from 12 psia to maximum design pressure[1])	1
~~Containment and Drywell Oxygen Concentration (for those plants with inerted containments)~~	~~0 to 20% (capability of operating from 12 psia to design pressure[1])~~	~~1~~
Primary Containment Isolation Valve Position (excluding check valves)	Closed-not closed	1
Suppression Chamber Air Temperature	30°F to 230°F	1
Drywell Temperature	40°F to 440°F	4 ~~1~~

DELETE FROM CATEGORY B. MOVE TO CATEGORY D.

TYPE C - Variables Which Indicate Breach or Potential for Breach of Barriers to Fission Product Release

Purpose & Variables	Range	Category
Fuel Cladding		
~~Core Exit Temperature~~	~~150°F to 2300°F~~	~~1~~
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	Normal to 10 Ci/gm	5[19]

[1]Design pressure - that value corresponding to ASME code values that are obtained at or below code-allowable material design stress values.

TABLE 3 - (continued)

Purpose & Variables	Range	Category
TYPE C - (continued)		
Reactor Coolant Pressure Boundary		
PRIMARY Containment High-Range Area Radiation	1 to 10^7 R/hr ~~(60 [illegible] photons [illegible] accuracy for [illegible] of 0.1 to 3 MeV)~~	$2^{5,18}$ ~~(for Mark III containments, two redundant monitors are required for primary containment & reactor building)~~
REACTOR BUILDING HIGH RANGE AREA RADIATION	1 TO 10^4 R/HR	$2^{5,18}$
Drywell Drain Sumps Level (Identified and Unidentified Leakage)	Bottom to top	1
Containment		
Standby Gas Treatment System Vent	10^{-6} to 10^5 µCi/cc	$4^{16,17}$
Effluent Radioactivity - Noble Gases	10^{-6} to 10^5 µCi/cc ~~(Xe-133 calibration)~~	$4^{6,16,17}$
Environs Radioactivity - Exposure Rate	10^{-6} to 10^4 R/hr ~~(60 keV to 3 MeV)~~	$4^{7,18}$
TYPE D - Variables Which Indicate Operation of Individual Safety Systems		
Power Conversion Systems		
Main Feedwater Flow	0 to 110% design flow[1]	5
Condensate Storage Tank Level	Bottom to top	5

TABLE 3 - (continued)

Purpose & Variables	Range	Category
TYPE D - (continued)		
Containment Systems		
Containment Spray Flow	0 to 110% design flow[1]	3
Drywell Pressure FOR MARK III CONTAINMENTS	12 psia to 3 psig 0 to 110% design pressure[2]	3
Suppression Cham. Water Level	Top of vent to top of weir well	3
Suppression Chamber Water Temperature	30°F to 230°F	3
Auxiliary Systems		
~~Steam Flow to RCIC~~	~~0 to 110% design flow[1]~~	~~3~~
HPCI Flow	0 to 110% design flow[1]	3
RCIC Flow	0 to 110% design flow[1]	3
Core Spray Flow	0 to 110% design flow[1]	3
RHR System Flow (LPCI)	0 to 110% design flow[1]	3
RHR Heat Exchanger Outlet Temperature (LPCI)	32°F to 350°F	3
Service Cooling Water Temperature	32°F to 200°F	3
Service Cooling Water Flow	0 to 110% design flow[1]	3
Flow in Ultimate Heat Sink Loop	0 to 110% design flow[1]	3
Temperature in Ultimate Heat Sink Loop	30°F to 150°F	3
Ultimate Heat Sink Level	Plant specific	3

TABLE 3 - (continued)

Purpose & Variables	Range	Category
TYPE D - (continued)		
Auxiliary Systems (continued)		
SLCS Storage Tank Level	Bottom to top	5
Sump Level in Spaces of Equipment Required for Safety	To corresponding level of safety equipment failure	5
SLCS Flow	0 to 110% design flow[1]	5
RADWASTE SYSTEMS		
High Radioactivity Liquid Tank Level	Top to bottom	5
VENTILATION SYSTEMS		
Emergency Ventilation Damper Position BETWEEN SECONDARY CONTAINMENT AND THE ENVIRONS	Open-closed status	3
Temperature of Space in Vicinity of Equipment Required for Safety	30°F to 180°F	5
POWER SUPPLIES		
Status of Class 1E Power ~~[illegible]~~ and Systems Sources	Voltages currents pressures	3[8]
Status of Non-Class 1E Power ~~[illegible]~~ and Systems Sources	Voltages currents pressures	5[9]

TABLE 3 - (continued)

Purpose & Variables	Range	Category
TYPE E - Variables Which Indicate Magnitude and Direction of Dispersion of Released Radioactive Materials		
RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY-RELATED EQUIPMENT		
Radiation Exposure Rates	R/hr 10^{-1} to 10^{4} R/hr ~~[illegible]~~	4[18]
AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT		
Effluent Radioactivity - Noble Gases		
Reactor Bldg or Secondary Containment RELEASE POINTS	10^{-6} to 10^{4} μCi/cc	4[16 17]
.Other Release Points (including fuel handling building, auxiliary building, and turbine building)	10^{-6} to 10^{3} μCi/cc ~~[illegible]~~	4[16 17]
Effluent Radioactivity - ~~[illegible]~~ Radiohalogens and Particulates	10^{-3} to 10^{2} μCi/cc	4[10]
Environs Radioactivity - Radiohalogens and Particulates	10^{-9} to 10^{-3} μCi/cc for both radiohalogens and particulates	4[11]
Plant and Environs Radioactivity & Radiation (portable instruments)	High Range 0.1 to 10^{4} R/hr photons 0.1 to 10^{4} rads/hr betas and low-energy photons	6[12]
	multi-channel gamma-ray spectrometer	6

Purpose & Variables	Range	Category
Type E - (continued)		
POSTACCIDENT SAMPLING CAPABILITY (Analysis Capability Onsite)		
Primary Coolant	Grab Sample	5[13]
Gross Activity	10 µCi/ml to 10 Ci/ml	
Gamma Spectrum	(Isotopic Analysis)	
Boron Content	0 to 1%	
~~Sumps~~ SUPPRESSION CHAMBER WATER	Grab Sample	5[13,14,20]
Gross Activity	10 µCi/ml to 10 Ci/ml	
Gamma Spectrum	(Isotopic Analysis)	
pH	2 to 12	
Containment Air	Grab Sample	5[13,20]
H_2		
O_2		
Gamma Spectrum	(Noble gas analysis)	
METEOROLOGY[15]		
Wind Direction	0 to 360° (±5° accuracy with a deflection of 15°. Starting speed 0.45 mps (1 mph))	5
Wind Speed	0 to 30 mps (67 mph) (±0.22 mps (0.5 mph) accuracy for wind speeds less that 11 mps (25 mph), with a starting threshold of less than 0.45 mps (1 mph))	5
Temperature	-60°F to 120°F (±0.9°F accuracy)	5
Vertical Temperature Difference	-9°F to +9°F (±0.3°F accuracy per 164-foot intervals)	5
Precipitation	Recording rain gage with range sufficient to ensure accuracy of total accumulation within 10% of recorded value - 0.01" resolution	5

TABLE 3

BWR VARIABLES

Purpose & Variables	Range	Design & Qualification Criteria Category (See Table 1A)
TYPE E - (continued)		
METEOROLOGY[15] - continued		
Dew Point Temperature	-60°F to 120°F (±2.7°F accuracy for temperature range, -22°F to 68°F when relative humidity is greater than 60%)	5

NOTES continued -

[3]The maximum value may be revised upward to satisfy ATWS requirements.

[4]Approximately 50 thermocouples should be available, the exact number needed will depend on thermocouple location and other characteristics. In the absence of core spray the thermocouples should detect 5 to 10% core area cross sectional blockage, with high confidence. Sufficient numbers should be installed to account for attrition.

[5]Minimum of two monitors at widely separated locations.

[6]Provisions should be made to monitor all identified pathways for release of gaseous radioactive materials to the environs in conformance with General Design Criterion 64. Monitoring of individual effluent streams only is required where such streams are released directly to the environment. If two or more streams are combined prior to release from a common discharge point, monitoring of the combined stream is considered to meet the intent of this guide provided such monitoring has a range adequate to measure worst-case releases.

[7]For estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors) - continuous readout capability. (Approximately 16 to 20 locations - site dependent.)

[8]Status indication of all Class 1E A-C buses, D-C buses, inverter output buses and pneumatic supplies.

[9]Status indication of all non-Class 1E inverter output buses, D-C buses and pneumatic supplies.

[10]To provide information regarding release of radioactive halogens and particulates. Continuous collection of representative samples followed by onsite laboratory measurements of samples for radiohalogens and particulates. The design envelope for shielding, handling, and analytical purposes should assume 30 minutes of integrated sampling time at sampler design flow, an average concentration of 10^2 μCi/cc of radioiodine in gaseous or vapor form, an average concentration of 10^2 μCi/cc of particulate radioiodines and particulates other than radioiodines, and an average gamma photon energy of 0.5 Mev per disintegration.

[11]For estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors). Continous collection of representative samples followed by laboratory measurements of the samples (Approximately 16 to 20 locations - site dependent.)

[12]To monitor radiation and airborne radioactivity concentrations in many areas throughout the facility and the site environs where it is impractical to install stationary monitors capable of covering both normal and accident levels.

[13]To provide means for safe and convenient sampling. These provisions should include:
1. Shielding to maintain radiation doses ALARA,
2. Sample containers with container-sampling port connector compatability,
3. Capability of sampling under primary system pressure and negative pressure,
4. Handling and transport capability, and
5. Pre-arrangement for analysis and interpretation.

TABLE 3 (continued)

NOTES continued -

14 An installed capability should be provided for obtaining ~~containment sump~~ SUPPRESSION CHAMBER WATER, ECCS pump room sumps, and other similar auxiliary building sump liquid samples.

15 Meteorological measurements should conform to the provisions of the forthcoming revision to Regulatory Guide 1.23, "Onsite Meteorological Programs".

16 Monitors should be capable of detecting and measuring radioactive gaseous effluent concentrations with compositions ranging from fresh equilibrium noble gas fission product mixtures to 10-day old mixtures, with overall system accuracies of ± 1/2 decade. Calibration should be performed using radiation sources representative of both low and high energy portions of the emission spectrum. For low-energy gamma photon calibration, source emission energies should fall within the range of approximately 60 keV to 150 keV (examples - Am-241, Cd-109, Tm-171, and Co-57). For high-energy gamma photon calibration, source emission energies should fall within the range of approximately 500 keV to 1.5 MeV (examples - Cs-137, Mn-54, and Co-60). Effluent concentrations may be expressed in terms of Xe-133 equivalents or in terms of the equivalent of any noble gas nuclide(s).

17 It is not expected that a single monitoring device will have sufficient range to encompass the entire range provided in this guide and that multiple components or systems will be needed. Existing equipment may be utilized to monitor any portion of the stated range within the equipment design rating. Additional extended range instrumentation should overlap the range of existing instrumentation by at least a factor of 2.

18 Detectors should respond to gamma radiation photons within any energy range from 60 keV to 3 MeV with an accuracy of ±20% at any specific photon energy from 0.1 MeV to 3 MeV. Overall system accuracy should be within ±1/2 decade over the entire range.

19 Measurement should be made of the gross gamma radiation emanating from circulating primary coolant, with instrument calibration permitting conversion of readout to radioactivity concentrations in terms of either curies/gram or curies/unit-volume. System accuracy should be ±1/2 order of magnitude. The point of measurement should be external to a circulating primary coolant line or loop, such as a hot leg, and should not be a line or loop subject to isolation, e.g., PWR letdown line or BWR main steam line. While such an instrument may not be currently available off-the-shelf, the staff considers that the necessary components are available commercially and have been employed and demonstrated under adverse environmental conditions in high-level hot cell operations for many years.

20 Sampling or monitoring of radioactive liquids and gases should be performed in a manner which assures procurement of representative samples. For gases, the criteria of ANSI N13.1 should be applied. For liquids, provisions should be made for sampling from well-mixed turbulent zones and sampling lines should be designed to minimize plateout or deposition.

21 FOR MARK I AND MARK II CONTAINMENTS THE MEASUREMENT SHOULD BE OF DRYWELL PRESSURE.

TABLE 3 A

BWR VARIABLES

Purpose & Variables	Range	Design & Qualification Criteria Category (See Table 1)
TYPE A - Variables Which Indicate Necessity for Pre-planned Manual Actions	Plant specific	1
TYPE B - Variables Which Indicate Accomplishment of Critical Safety Functions		
Reactivity Control		
Control Rod Position	Full in or not full in	5 (for 2 hrs minimum)
Neutron Flux	10^{-8} TO 10^{-2} FULL ~~1-ε/s to 100~~ power	1
Core Cooling		
Coolant Level in the Reactor	APPROXIMATELY 2/3 T of Bottom of core HEIGHT ~~support piece~~ to above top of discharge plenum BOTTOM OF CORE SUPPORT PLATE TO 2/3 CORE HEIGHT	1^{21}
~~Main Steamline Flow~~	~~0 to 110% design flow~~	~~1~~
Maintaining Reactor Cooling System Integrity		
RCS Pressure	15 psia to 1500 psig	1^{3}
Main Steamline Isolation Valves' Leakage Control System Pressure	0 to 15" of water 0 to 5 psid	1
Primary System Safety Relief Valve Positions, including ADS or Flow Through or Pressure in Valve Lines	Closed-not closed or 0 to 50 psig	1^{21}

*Design flow - the maximum flow anticipated in normal operation. THE ACCURACY SHOULD BE ± 5% OF DESIGN FLOW.

TABLE 3A- (continued)

Purpose & Variables	Range	Category
TYPE B - (continued)		
<u>Maintaining Containment Integrity</u>		
Primary Containment Pressure ~~(Drywell)~~	10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel	2[21, 22]
Containment and Drywell Hydrogen Concentration	0 to ~~30%~~ 10% (capability of operating from 12 psia to maximum design pressure[2])	1[21]
~~Containment and Drywell Oxygen Concentration (for those plants with inerted containments)~~	~~0 to 20% (capability of operating from 12 psia to design pressure[2])~~	~~1~~
Primary Containment Isolation Valve Position (excluding check valves)	Closed-not closed	1
Suppression Chamber Air Temperature	30°F to 230°F	1
Drywell Temperature	40°F to 440°F	4 — DELETE FROM CATEGORY B. MOVE TO CATEGORY D.
TYPE C - Variables Which Indicate Breach or Potential for Breach of Barriers to Fission Product Release		
<u>Fuel Cladding</u>		
~~Core Exit Temperature~~	~~[illegible]~~	~~1[4]~~
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	Normal to 10 Ci/gm	5[19]

[2]Design pressure - that value corresponding to ASME code values that are obtained at or below code-allowable material design stress values.

TABLE 3A- (continued)

Purpose & Variables	Range	Category
TYPE D - (continued)		
Containment Systems		
Containment Spray Flow	0 to 110% design flow[1]	3
Suppression Chamber Water Temperature	30°F to 230°F	3
Auxiliary Systems		
~~Steam Flow to RCIC~~	~~0 to 110% design flow[1]~~	~~3~~
HPCI Flow	0 to 110% design flow[1]	3
RCIC Flow	0 to 110% design flow[1]	3
RHR System Flow (LPCI)	0 to 110% design flow[1]	3
RHR Heat Exchanger Outlet Temperature (LPCI)	32°F to 350°F	3
Service Cooling Water Temperature	32°F to 200°F	3
Service Cooling Water Flow	0 to 110% design flow[1]	3
Flow in Ultimate Heat Sink Loop	0 to 110% design flow[1]	3
Temperature in Ultimate Heat Sink Loop	30°F to 150°F	3
Ultimate Heat Sink Level	Plant specific	3

TABLE 3A- (continued)

Purpose & Variables	Range	Category
TYPE C - (continued)		
Reactor Coolant Pressure Boundary		
PRIMARY Containment High-Range Area Radiation	1 to 10^7 R/hr ~~[illegible]~~	2[5, 18, 21] ~~(for Mark III containments, two redundant monitors are required for primary containment & reactor building)~~
REACTOR BUILDING HIGH RANGE AREA RADIATION	1 TO 10^4 R/HR	2[5, 18, 21]
Containment Water Level	0 to 5 feet above normal water level	1[21]
Containment		
Standby Gas Treatment System Vent	10^{-6} to 10^5 μCi/cc	4[16, 17, 21]
Effluent Radioactivity - Noble Gases	10^{-6} to 10^5 μCi/cc ~~(Xe 133 calibration)~~	4[6, 16, 17, 21]
Environs Radioactivity - Exposure Rate	10^{-6} to 10^4 R/hr ~~(60 keV to 3 MeV)~~	4[7, 18]
TYPE D - Variables Which Indicate Operation of Individual Safety Systems		
Power Conversion Systems		
Main Feedwater Flow	0 to 110% design flow[1]	5
Condensate Storage Tank Level	Bottom to top	5

Purpose & Variables	Range	Category
I. & D - (continued)		
Auxiliary Systems (continued)		
SLCS Storage Tank Level	Bottom to top	5
Sump Level in Spaces of Equipment Required for Safety	To corresponding level of safety equipment failure	5
SLCS Flow	0 to 110% design flow[1]	5
RADWASTE SYSTEMS		
High Radioactivity Liquid Tank Level	Top to bottom	5
VENTILATION SYSTEMS		
Emergency Ventilation Damper Position BETWEEN SECONDARY CONTAINMENT AND THE ENVIRONS	Open-closed status	3
Temperature of Space in Vicinity of Equipment Required for Safety	30°F to 180°F	5
POWER SUPPLIES		
Status of Class 1E Power ~~[illegible]~~ and Systems Sources	Voltages currents pressures	3[8]
Status of Non-Class 1E Power ~~[illegible]~~ and Systems Sources	Voltages currents pressures	5[9]

Purpose & Variables	Range	Category
Type E - Variables Which Indicate Magnitude and Direction of Dispersion of Released Radioactive Materials		
~~RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY-RELATED EQUIPMENT~~		
Radiation Exposure Rates	R/hr 10^{-1} to 10^{4} R/hr ~~[illegible]~~	4[18]
AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT		
Effluent Radioactivity - Noble Gases		
Reactor Bldg or Secondary Containment RELEASE POINTS	10^{-6} to 10^{4} µCi/cc	4[16,17]
.Other Release Points (including fuel handling building, auxiliary building, and turbine building)	10^{-6} to 10^{3} µCi/cc ~~[illegible]~~	4[16,17,21]
Effluent Radioactivity - ~~[illegible]~~ Radiohalogens and Particulates	10^{-3} to 10^{2} µCi/cc	4[10]
Environs Radioactivity - Radiohalogens and Particulates	10^{-9} to 10^{-3} µCi/cc for both radiohalogens and particulates	4[11]
Plant and Environs Radioactivity & Radiation (portable instruments)	High Range 0.1 to 10^{4} R/hr photons 0.1 to 10^{4} rads/hr betas and low-energy photons	6[12]
	multi-channel gamma-ray spectrometer	6

Purpose & Variables	Range	Category
TYPE E - (continued)		
<u>POSTACCIDENT SAMPLING CAPABILITY (Analysis Capability Onsite)</u>		
Primary Coolant	Grab Sample	5[13, 21, 22]
Gross Activity	10 μCi/ml to 10 Ci/ml	
Gamma Spectrum	(Isotopic Analysis)	
Boron Content	0 to 1%	
~~Sumps~~ SUPPRESSION CHAMBER WATER	Grab Sample	5[13, 14, 20, 22]
Gross Activity	10 μCi/ml to 10 Ci/ml	
Gamma Spectrum	(Isotopic Analysis)	
pH	2 to 12	
Containment Air	Grab Sample	5[13, 20, 22]
H_2		
O_2		
Gamma Spectrum	(Noble gas analysis)	
<u>METEOROLOGY</u>[15]		
Wind Direction	0 to 360° (±5° accuracy with a deflection of 15°. Starting speed 0.45 mps (1 mph))	5
Wind Speed	0 to 30 mps (67 mph) (±0.22 mps (0.5 mph) accuracy for wind speeds less that 11 mps (25 mph), with a starting threshold of less than 0.45 mps (1 mph))	5
Temperature	-60°F to 120°F (±0.9°F accuracy)	5
Vertical Temperature Difference	-9°F to -9°F (±0.3°F accuracy per 164-foot intervals)	5
Precipitation	Recording rain gage with range sufficient to ensure accuracy of total accumulation within 10% of recorded value - 0.01" resolution	5

TABLE 3A

BWR VARIABLES

Purpose & Variables	Range	Design & Qualification Criteria Category (See Table 1A)
.YPE E - (continued)		
METEOROLOGY[15] - continued		
Dew Point Temperature	-60°F to 120°F (±2.7°F accuracy for temperature range, -22°F to 68°F when relative humidity is greater than 60%)	5

TABLE 3 A (continued)

NOTES continued -

[3]The maximum value may be revised upward to satisfy ATWS requirements.

[4]Approximately 50 thermocouples should be available, the exact number needed will depend on thermocouple location and other characteristics. In the absence of core spray the thermocouples should detect 5 to 10% core area cross sectional blockage, with high confidence. Sufficient numbers should be installed to account for attrition.

[5]Minimum of two monitors at widely separated locations.

[6]Provisions should be made to monitor all identified pathways for release of gaseous radioactive materials to the environs in conformance with General Design Criterion 64. Monitoring of individual effluent streams only is required where such streams are released directly to the environment. If two or more streams are combined prior to release from a common discharge point, monitoring of the combined stream is considered to meet the intent of this guide provided such monitoring has a range adequate to measure worst-case releases.

[7]For estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors) - continuous readout capability. (Approximately 16 to 20 locations - site dependent.)

[8]Status indication of all Class 1E A-C buses, D-C buses, inverter output buses and pneumatic supplies.

[9]Status indication of all non-Class 1E inverter output buses, D-C buses and pneumatic supplies.

[10]To provide information regarding release of radioactive halogens and particulates. Continuous collection of representative samples followed by onsite laboratory measurements of samples for radiohalogens and particulates. The design envelope for shielding, handling, and analytical purposes should assume 30 minutes of integrated sampling time at sampler design flow, an average concentration of 10^2 µCi/cc of radioiodine in gaseous or vapor form, an average concentration of 10^2 µCi/cc of particulate radioiodines and particulates other than radioiodines, and an average gamma photon energy of 0.5 Mev per disintegration.

[11]For estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors). Continous collection of representative samples followed by laboratory measurements of the samples (Approximately 16 to 20 locations - site dependent.)

[12]To monitor radiation and airborne radioactivity concentrations in many areas throughout the facility and the site environs where it is impractical to install stationary monitors capable of covering both normal and accident levels.

[13]To provide means for safe and convenient sampling. These provisions should include:

1. Shielding to maintain radiation doses ALARA,
2. Sample containers with container-sampling port connector compatability,
3. Capability of sampling under primary system pressure and negative pressure,
4. Handling and transport capability, and
5. Pre-arrangement for analysis and interpretation.

NOTES continued -

14 An installed capability should be provided for obtaining containment sump, ECCS pump room sumps, and other similar auxiliary building sump liquid samples.

15 Meteorological measurements should conform to the provisions of the forthcoming revision to Regulatory Guide 1.23, "Onsite Meteorological Programs".

16 Monitors should be capable of detecting and measuring radioactive gaseous effluent concentrations with compositions ranging from fresh equilibrium noble gas fission product mixtures to 10-day old mixtures, with overall system accuracies of ± 1/2 decade. Calibration should be performed using radiation sources representative of both low and high energy portions of the emission spectrum. Fow low-energy gamma photon calibration, source emission energies should fall within the range of approximately 60 keV to 150 keV (examples - Am-241, Cd-109, Tm-171, and Co-57). For high-energy gamma photon calibration, source emission energies should fall within the range of approximately 500 keV to 1.5 MeV (examples - Cs-137, Mn-54, and Co-60). Effluent concentrations may be expressed in terms of Xe-133 equivalents or in terms of the equivalent of any noble gas nuclide(s).

17 It is not expected that a single monitoring device will have sufficient range to encompass the entire range provided in this guide and that multiple components or systems will be needed. Existing equipment may be utilized to monitor any portion of the stated range within the equipment design rating. Additional extended range instrumentation should overlap the range of existing instrumentation by at least a factor of 2.

18 Detectors should respond to gamma radiation photons within any energy range from 60 keV to 3 MeV with an accuracy of ±20% at any specific photon energy from 0.1 MeV to 3 MeV. Overall system accuracy should be within ±1/2 decade over the entire range.

19 Measurement should be made of the gross gamma radiation emanating from circulating primary coolant, with instrument calibration permitting conversion of readout to radioactivity concentrations in terms of either curies/gram or curies/unit-volume. System accuracy should be ±1/2 order of magnitude. The point of measurement should be external to a circulating primary coolant line or loop, such as a hot leg, and should not be a line or loop subject to isolation, e.g., PWR letdown line or BWR main steam line. While such an instrument may not be currently available off-the-shelf, the staff considers that the necessary components are available commercially and have been employed and demonstrated under adverse environmental conditions in high-level hot cell operations for many years.

20 Sampling or monitoring of radioactive liquids and gases should be performed in a manner which assures procurement of representative samples. For gases, the criteria of ANSI N13.1 should be applied. For liquids, provisions should be made for sampling from well-mixed turbulent zones and sampling lines should be designed to minimize plateout or deposition.

21 The provisions of this table are consistent with NUREG 0578 and NRR letters dated September 13, and October 30, 1979, for this variable.

22 FOR MARK I AND MARK II CONTAINMENTS THE MEASUREMENT SHOULD BE OF DRYWELL PRESSURE.

Presentation by William Coley
on behalf of the AIF Subcommittee on
Safety Parameter Integration
ACRS Subcommittee on Regulatory Activities
August 6, 1980

My name is William Coley. I am Manager of Engineering Services Steam Production Department at Duke Power Company and I am here today representing the AIF Subcommittee on Safety Parameter Integration. I am also chairman of the AIF Subcommittee on Control Room Considerations.

The purpose of my presentation today is to offer a way to allow the proposals for emergency facilities to be resolved and implemented in the most timely and safety effective way and at the same time provide a vehicle for resolving the controversy surrounding the proposed R.G. 1.97 instrumentation list and requirements. This proposed approach is an outgrowth of intensive interactions between our Subcommittee on Safety Parameter Integration and NRC technical management concerning development of an integrated approach for defining the requirements for SPDS, TSC, EOF and other facilities to support crisis management. This effort has involved a series of meetings over the last three months with many experts who have contributed to our approach. In presenting this approach we intend to proceed in three distinct steps:

(1) To explain the rationale behind our approach.
(2) To propose an example list of parameters that should be given first precedence and is the first step in development of subsequent lists and requirements resulting in an integrated data display system.
(3) To underscore the serious problems with the currently proposed R.G. 1.97 requirements and the subsequent implications of these problems on emergency facilities.

At the time work was initiated on Regulatory Guide 1.97, industry did not have in place structured crisis management plans and organizations to address fully a major site emergency. Further, emergency facilities such as the SPDS, TSC, and EOF, which support the crisis management plan were not defined. Thus, the selection of variables in R.G. 1.97 was not related to their use in these emergency facilities. Consequently, the requirements of R.G. 1.97 are not in concert with industry and NRC efforts on these facilities.

This disconnect is particularly important since the NRC is now tying the instrumentation requirements for these facilities to R.G. 1.97.

Additionally, R.G. 1.97 does not recognize the current industry efforts and evolving NRC requirements to improve the operator interface; in fact, it has not addressed human engineering factors which validate the usefulness and help to the operator of the parameters selected.

In our efforts with NRC Staff to define the functional requirements of emergency facilities, we have embarked on a systematic approach to establishing the data requirements for emergency facilities. This approach in contrast to R.G. 1.97, integrates the consideration of human factors engineering, the need for and importance of the information, and the function for which the information is going to be used. Implementation of R.G. 1.97 in its present form would preempt this timely and more safety effective approach.

Through sequential application of this methodology to first meet the requirements for the SPDS and other emergency facilities, a set of accident parameters can be defined which are generic to the detection and mitigation of any site accident. Further, the application of the methodology should allow us to implement more quickly in operating plants those factors which have potential for the greatest improvement in safety.

We are now in the process of several parallel efforts. One is to review the functional requirements of the emergency facilities. Another is to do a human factors review of the Control Room. We see a logical evolution of the intent and original spirit of R.G. 1.97 through the progressive development of the emergency facilities, the human factors control room review, and then consideration of what requirements remain to be addressed in other regulatory guides. Accordingly, it is our judgement that the instrumentation requirements in R. G. 1.97 should not be implemented until such time as the appropriateness of these requirements can be verified through this progressive development. As discussed above, implementation of R. G. 1.97 in its present form at this point in time will preempt this timely and more safety effective approach.

Our first step in this approach has been to develop a minimum parameter set for localized display in the control room. We have selected those SPDS parameters that we feel are essential for focused attention of the operator. Dave Cain of NSAC will provide the methodology for selecting these parameters and the resulting list of parameters for PWRs. Ellery Hammond representing the BWR Owners Group will also give a presentation on SPDS instrument selection from a BWR perspective. To further illustrate some of the specific problems with R.G. 1.97, we have submitted to you our previous comments on it. Xavier Polanski will highlight our general concerns.

STRUCTURED PARAMETER SELECTION PROCESS

- FUNCTIONAL REQUIREMENTS
- SELECTION CRITERIA
- DECISION LOGIC

PARAMETER SELECTION CRITERIA (SPDS)

LEADING INDICATOR

PLANT SAFETY FUNCTION

RADIOACTIVE BARRIER

DETECTION

DIRECT MEASUREMENT

RELIABILITY

DIVERSE PLANT CONDITIONS

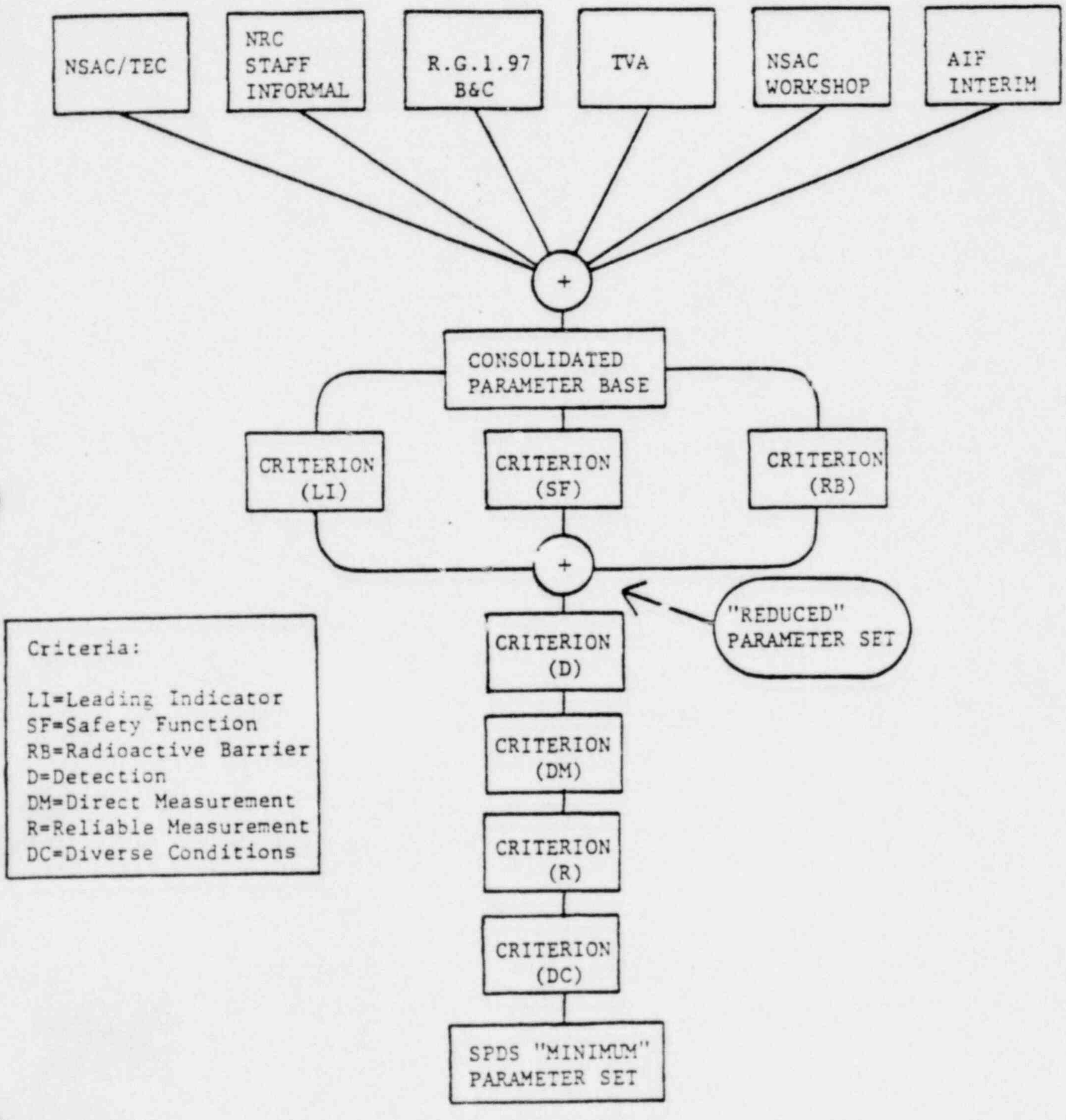
THE OPTIMAL PARAMETER SET

MORE IS BETTER

VS.

EFFICIENCY IN DE^IGN

PARAMETER SELECTION PROCEDURE: SPDS



SELECTION MATRIX

Consolidated Parameter Base	D	DM	R	DC	LI	SF	RB	COMMENTS
T hot*	x	x	x	x	x	x		Proposed as an alternative to Core Exit Tc's.
T cold*	x	x	x	x	x	x		
S/G Level*	x	x	x	x	x	x		
S/G Pressure*	x	x	x	x	x	x		
Cont. Rad. Mon.*	x	x	x	x	x		x	
Core Exit Tc's	x	x	x	x	x	x		
Vessel Level	x	x		x	x	x		State of the art precludes reliable, unambiguous level measurement at this time.
RHR Flow		x	x		x			
Aux. FW Flow*	x	x	x	x	x	x		Should be augmented by normal feedwater for normal operations.
CST Level		x	x	x	x			
RCS Flow Rate		x			x	x		
S/G RV Pos.		x	x	x	x			
RCS Rad. Mon.	x	x	x	x			x	Installation of high range rad. monitoring instruments under present requirements would be sufficient to meet present selection criteria.
Cond. A/E Mon.*	x	x	x	x			x	
CR Pos.	x	x		x	x	x		Control rod position not considered reliable, nor practical, given number of variables to be monitored by SPDS.
Main Fac. Exh. Mon.*	x	x	x	x			x	
RHR Rad. Mon.		x	x				x	
Pzr. Level*	x	x	x	x	x	x		
RCS Press*	x	x	x	x	x	x		
Cont. Sump Level*	x	x	x	x	x	x		
Drain TK Level		x	x	x	x			
BWST Level		x	x	x				
SRV & PORV Pos.		x	x	x	x			
Boric Acid Chg. Flow		x	x		x			
Boron Conc.	x			x				Boron conc. req'd after TMI(2); methods are unreliable and do not account for concentration in core during boil-off.

SELECTION MATRIX

Consolidated Parameter Base	D	DM	R	DC	LI	SF	RB	COMMENTS
CVCS Tank Level		x	x	x	x			
Neutron Flux*	x	x	x	x	x	x		
Letdown Flow		x	x	x	x			
Coolant Subcooling	x		x	x	x	x		Subcooling and/or superheat may be computed internal to SPDS.
Cont. Press.*	x	x	x	x	x	x		
Cont. H_2 Conc.	x					x		On-line H_2 monitoring presently considered unreliable.
Cont. Iso. Valve Pos.		x	x	x				
Cont. Temp.	x	x		x	x	x		
Heat Removal--Cont. Fan Cool.			x		x			

*AIF Minimum SPDS Parameter Set for PWR

MINIMUM SPDS PARAMETER SET FOR PWR

I. REACTIVITY CONTROL

1. NEUTRON FLUX (<1% POWER)

II. REACTOR CORE COOLING

1. CORE HEAT REMOVAL AND RCS INVENTORY CONTROL

 - RCS COLD LEG TEMP
 - RCS HOT LEG TEMP OR CORE EXIT TEMP
 - RCS PRESSURE
 - PRESSURIZER WATER LEVEL

2. HEAT TRANSFER PATHS

 - STEAM GENERATOR WATER LEVEL
 - STEAM GENERATOR PRESSURE
 - AUXILIARY FEEDWATER FLOW
 - MAIN FEEDWATER FLOW

III. REACTOR COOLING SYSTEM INTEGRITY

1. RCS PRESSURE
2. CONTAINMENT PRESSURE
3. RCS TEMPERATURE (HOT LEG OR CORE EXIT)
4. CONTAINMENT HIGH-RANGE AREA RADIATION
5. CONTAINMENT SUMP WATER LEVEL
6. SECONDARY SIDE RADIATION (AIR EJECTOR OFF-GAS)
7. PRESSURIZER WATER LEVEL

IV. CONTAINMENT INTEGRITY

1. CONTAINMENT PRESSURE

V. RADIOACTIVITY RELEASE (FINAL RELEASE POINT MONITORS)

1. STACK RADIOACTIVITY NOBLE GASES
2. AIR EJECTOR RADIOACTIVITY NOBLE GASES

METHODOLOGY FOR

BWR SPDS

ACRS
8-6-80
EH-1

BWR SAFETY PARAMETER DEVELOPMENT

BASED ON EMERGENCY GUIDELINES

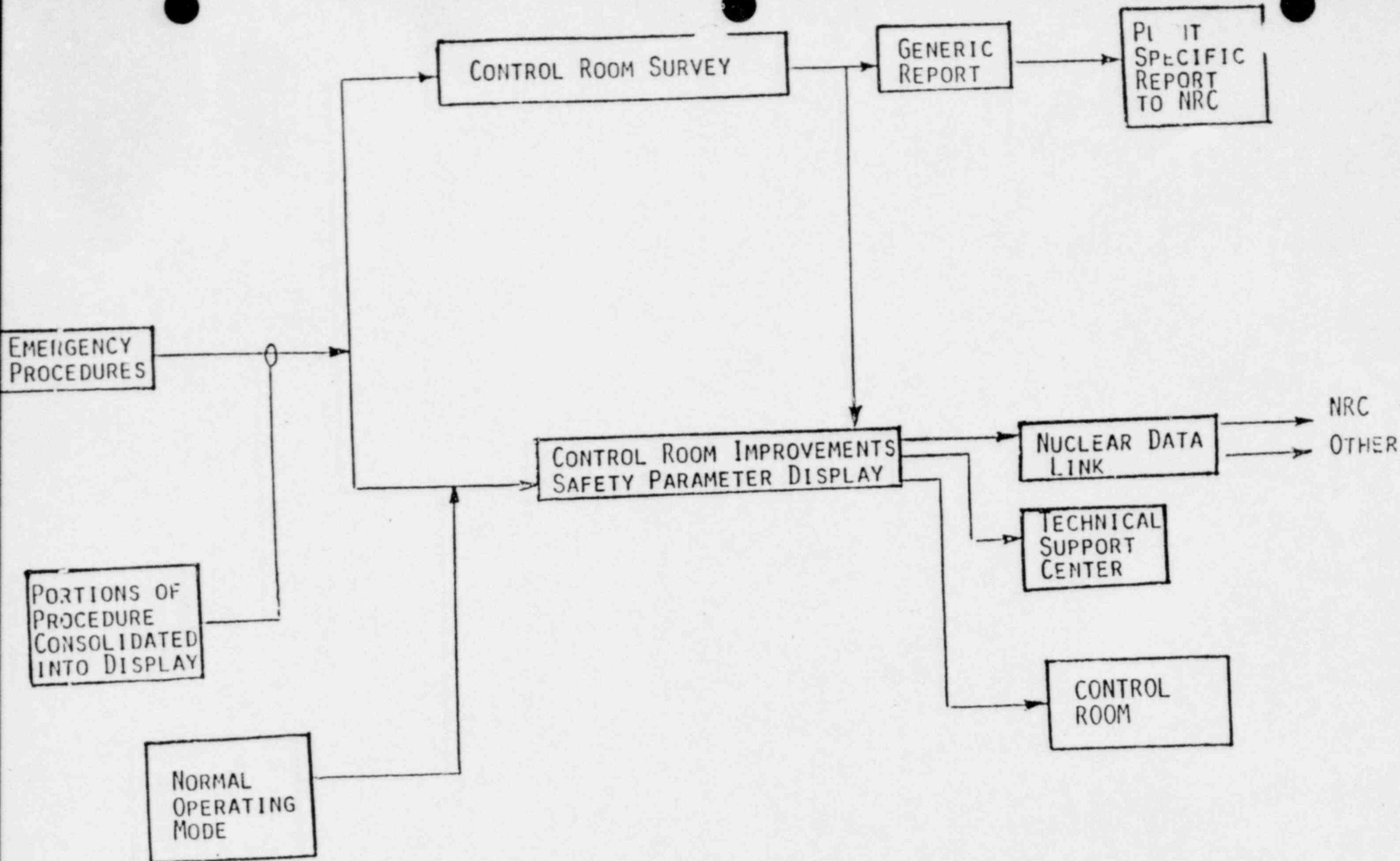
- o DEVELOPED BY OWNER CONSENSUS
- o SYMPTOM BASED
- o COVER MULTIPLE FAILURES

DEVELOPED FROM KEY GUIDELINE FUNCTIONS

- o LEVEL CONTROL
- o CONTAINMENT CONTROL
- o SHUTDOWN
- o CONTINGENCIES

PRIMARY VARIABLES WERE DETERMINED

- o INDICATE STATUS/VALUE OF SAFETY PARAMETER
- o SUPPORTED BY SECONDARY VARIABLES



NORMAL AND TRANSIENT
SAFETY PARAMETER DISPLAY

ACRS
8-6-80
EH-3

TABLE B

EMERGENCY PROCEDURE GUIDELINES

CONTAINMENT CONTROL GUIDELINE

Operator Action	Action Ident #	Supporting Parameters	Where is Information Available & How Shown
Entry any of,			
• High Suppression Pool Temp.	1	Suppression pool temp.	CR
• High Drywell Temp.	2	Drywell temp. (C 9)	CR
• High Drywell Pressure	3	Drywell pressure	CR
• High Suppression Pool Water level	4	Suppression pool water level (C 12)	CR
		NOTE 1: Display suggested	
Monitor and control all entry conditions concurrently	5	(4 entry conditions (C12))	All in CR
Close any SRV within (2 minutes), or Scram Reactor	5-1 5-2	SRV positions Rod positions	CR/Ind. Lights CR/Ind. Lights
Operate available Suppression pool cooling when pool temp. exceeds normal operating limit.	5A	Suppression pool temp. (C 7, 8, 18, 20)	CR
If Suppression pool temp. reaches scram limit, scram the reactor or verify scrammed.	5A-1	Suppression pool temp. Control rod position Neutron flux (SRM) NOTE 2	CR CR (P680) and lights CR (P680), meter/recorder
Control suppression pool temp. and/or RPV pressure below the heat capacity limit.	5A-2	(C 16, 19, 20) Suppression pool temp. RPV pressure	 CR CR

ACRS
8-6-80
EH-4

<u>Minimum SPDS Parameter Set for BWR</u>

I. <u>Reactivity Control</u>

1. SRM Period/Neutron Flux

II. <u>Reactor Core Cooling</u>

1. Core Heat Removal

   Reactor Water Level

2. Heat Transfer Paths

   Suppression Pool Water Level

   RHR Water Temperature

   RHR SW Exit Temperature

III. <u>Reactor Cooling System Integrity</u>

1. RCS Pressure
2. Drywell Sump Collection Rate
3. Drywell Pressure

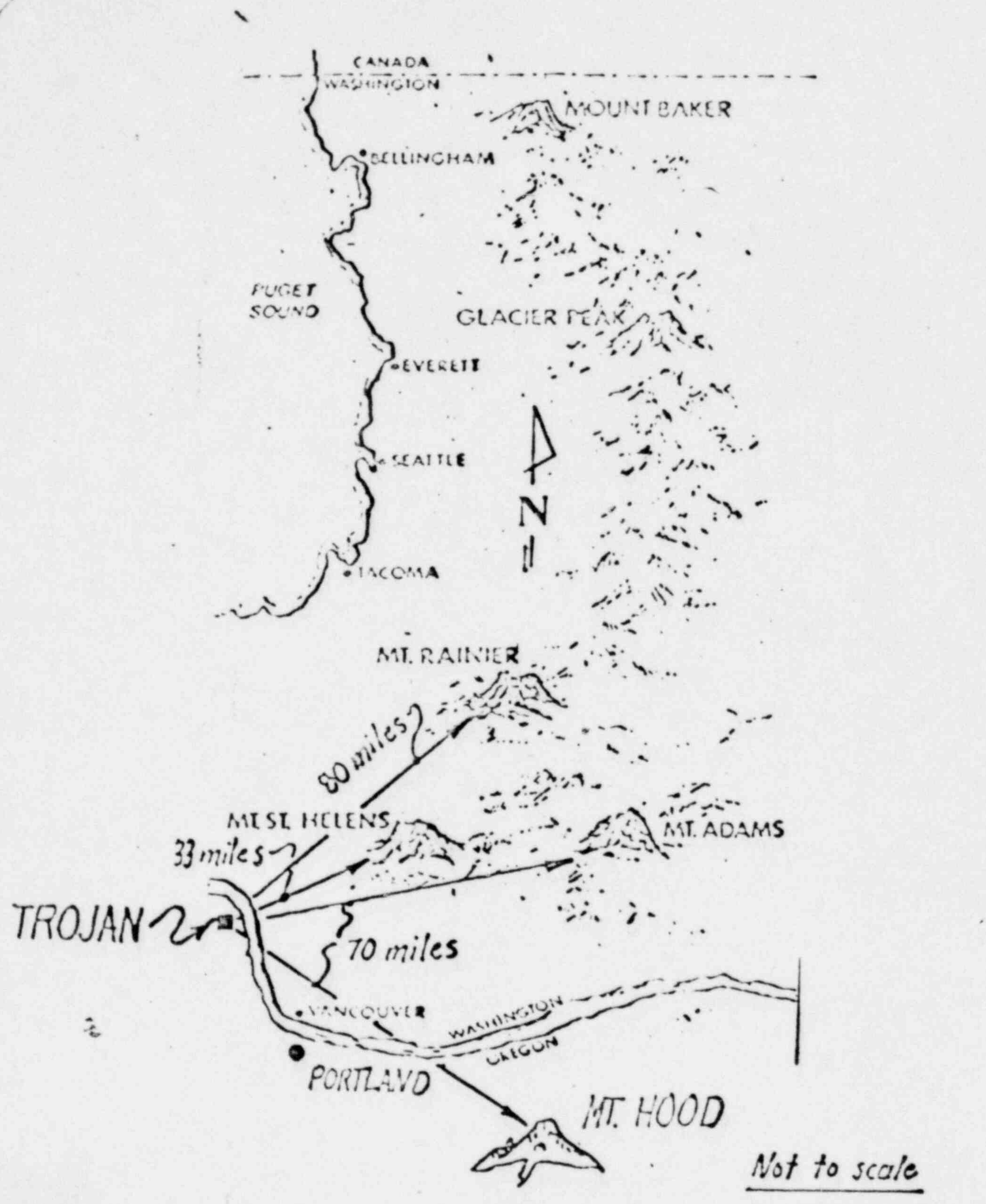
IV. <u>Containment Integrity</u>

1. Suppression Pool Water Temperature
2. Suppression Pool Pressure
3. Drywell Atmosphere Temperature
4. Drywell Pressure
5. Suppression Pool Water Level

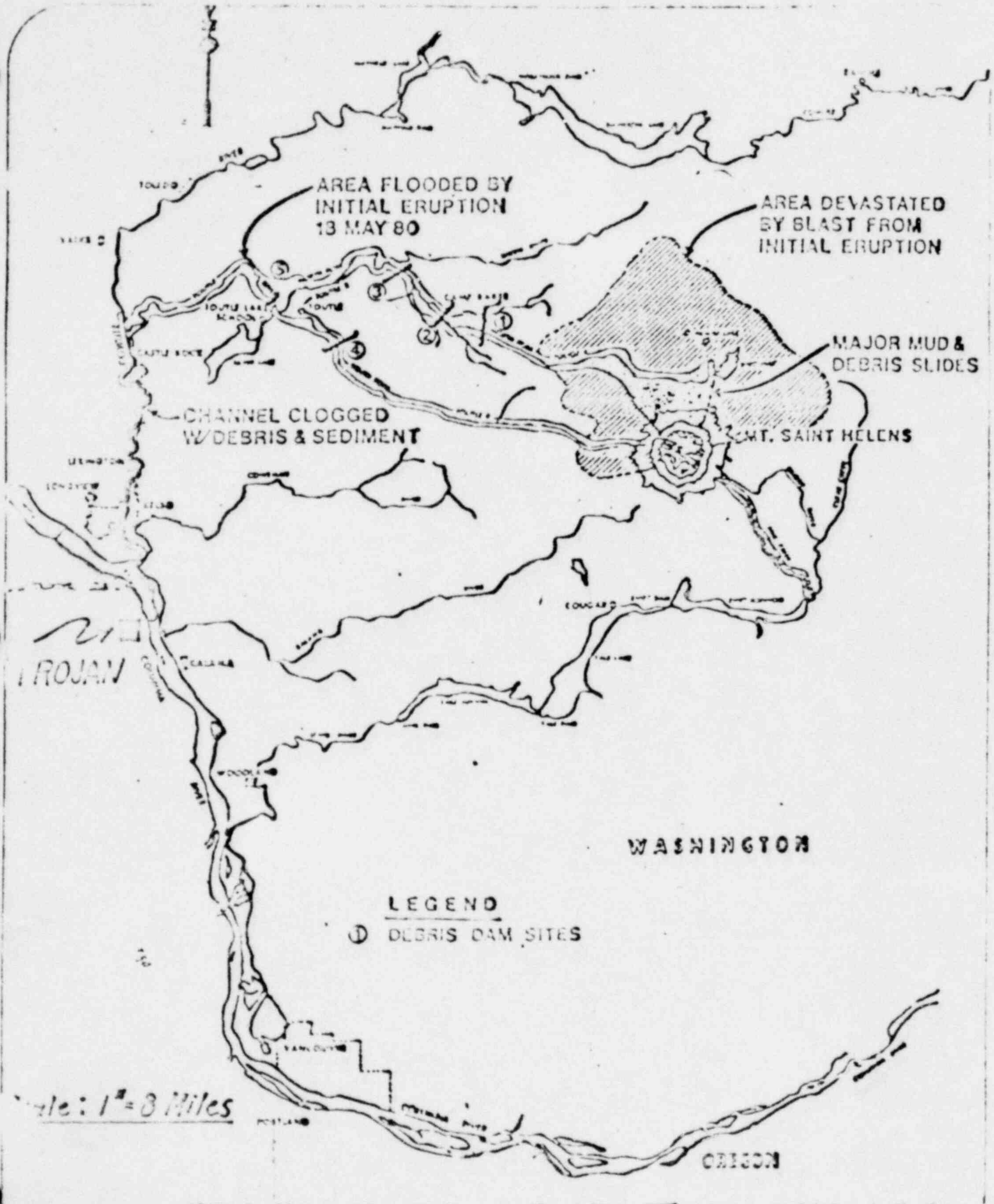
V. <u>Radioactivity Release</u>

1. Reactor Building Exhaust Ventilation Radioactivity
2. Standby Gas Treatment System Radioactivity
3. Off-Gas Stack Radioactivity
4. Process Liquid Radioactivity

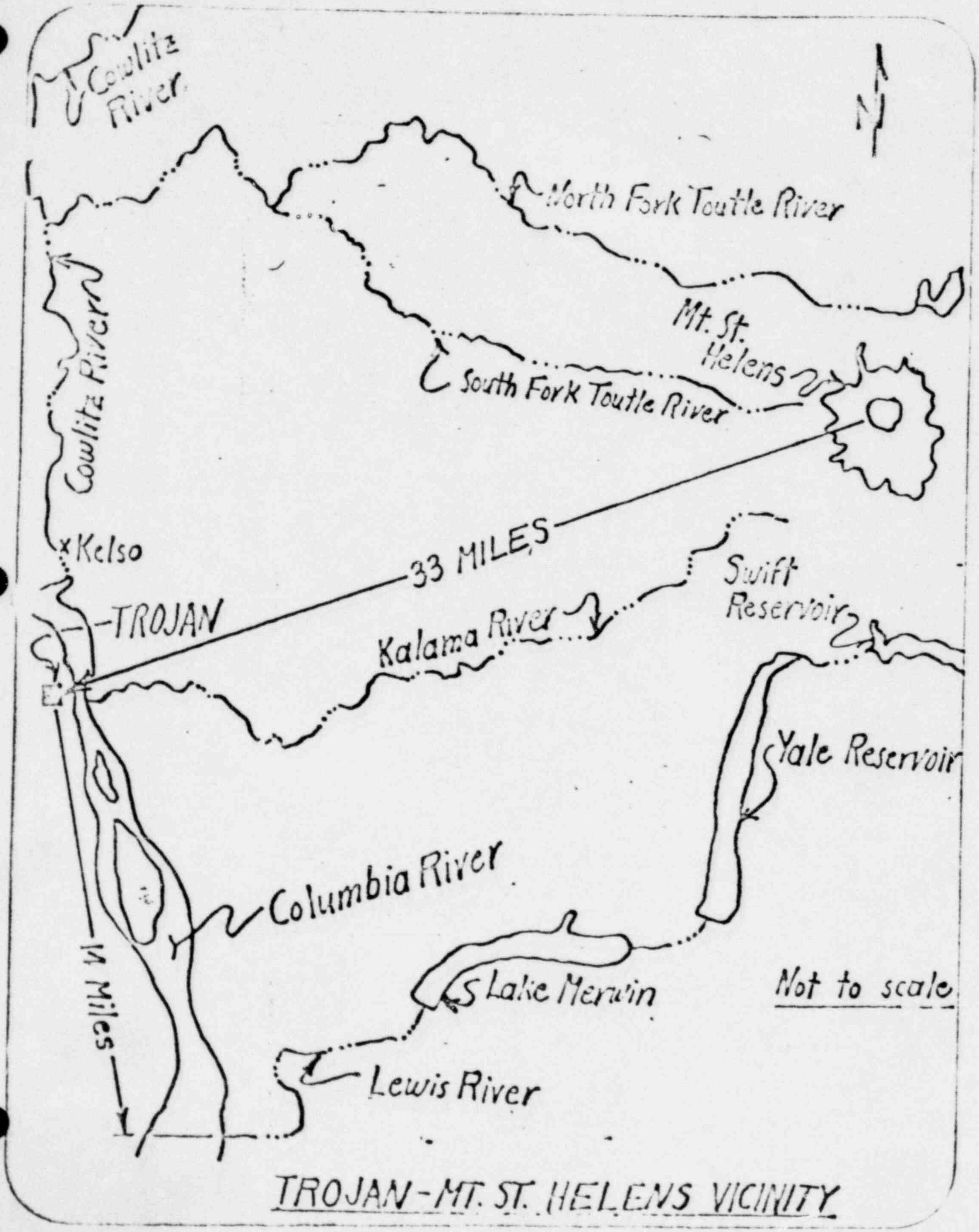
ACRS
3-6-80
EH-5



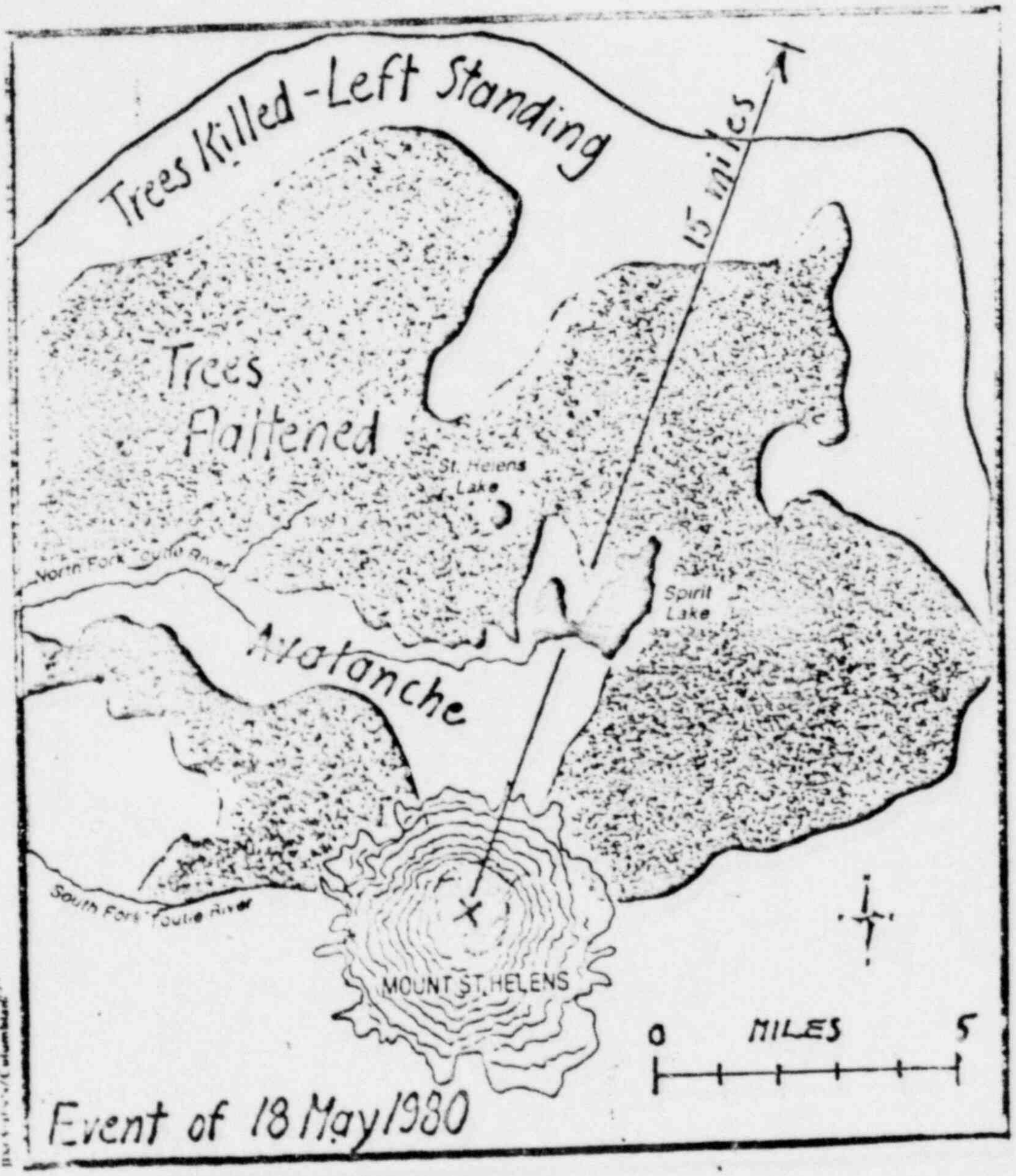
TROJAN-CASCADE VOLCANO RELATIONSHIP

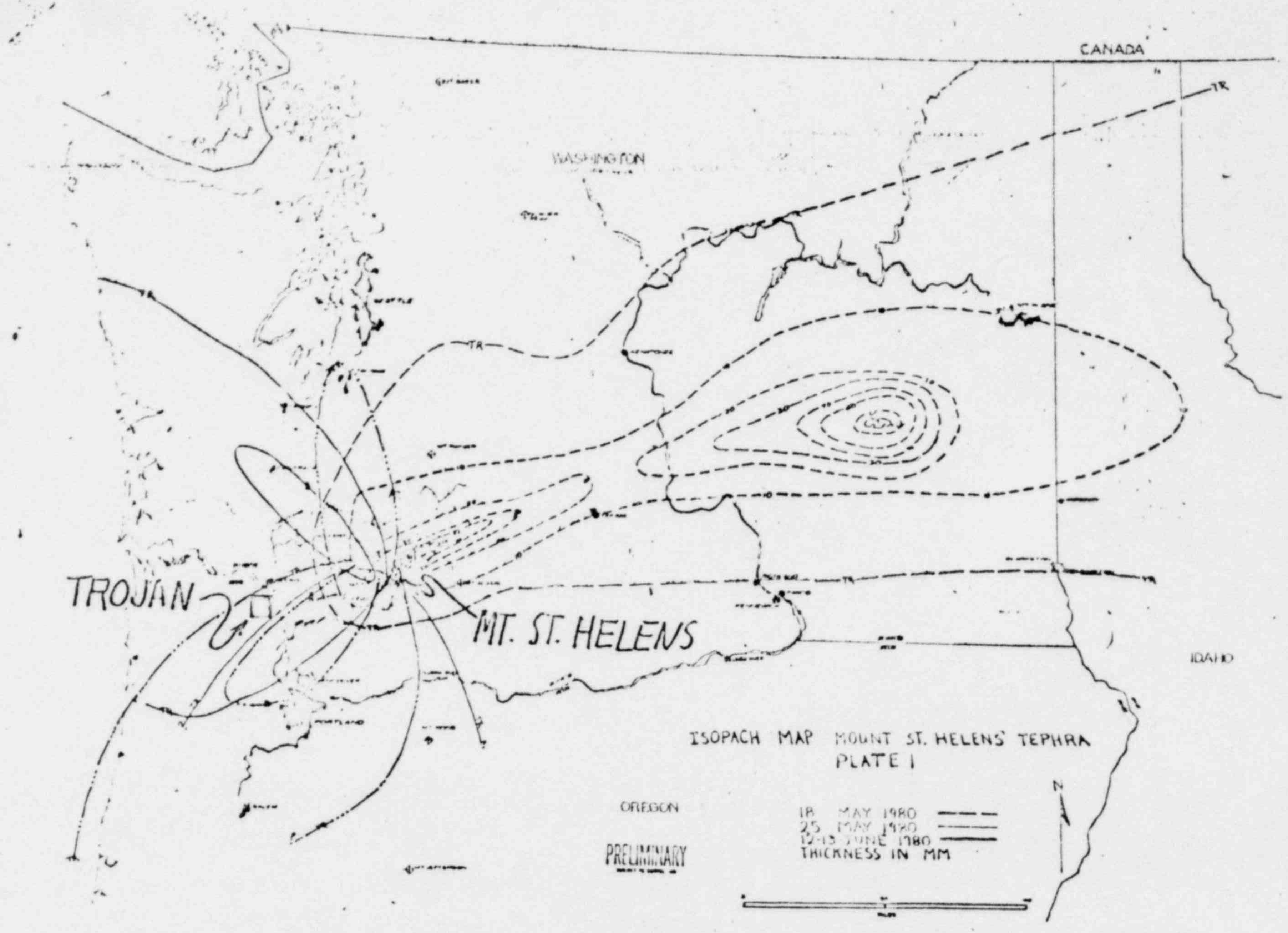


TROJAN-PORTLAND-MT. ST. HELENS AREA



TROJAN - MT. ST. HELENS VICINITY


Trees Killed - Left Standing
15 miles
Trees Flattened
St. Helens Lake
North Fork Toutle River
Spirit Lake
Avalanche
South Fork Toutle River
MOUNT ST. HELENS
0 MILES 5
Event of 18 May 1980


CANADA
WASHINGTON
IDAHO
OREGON
TROJAN
MT. ST. HELENS
TR
ISOPACH MAP MOUNT ST. HELENS TEPHRA
PLATE I
18 MAY 1980
25 MAY 1980
12-13 JUNE 1980
THICKNESS IN MM
N
PRELIMINARY

IE BULLETIN 80-11

MASONRY WALL DESIGN

ISSUED MAY 8, 1980

ACTION: FACILITIES WITH AN OPERATING LICENSE (EXCEPT TROJAN, SEQUOYAH UNIT NO. 1, NORTH ANNA UNIT NO. 2, SALEM UNIT NO. 2).

INFORMATION: CONSTRUCTION PERMIT HOLDERS (PLUS TROJAN, SEQUOYAH UNIT NO. 1, NORTH ANNA UNIT NO. 2, SALEM UNIT NO. 2)

SEPARATE LETTERS WERE PREPARED FOR THESE FACILITIES

ACTION TO BE TAKEN:

60 DAYS - IDENTIFY THE SAFETY-RELATED MASONRY WALLS

- ESTABLISH A PROGRAM FOR THE RE-EVALUATION OF THE MASONRY WALLS

180 DAYS - PERFORM THE RE-EVALUATION

- PROVIDE JUSTIFICATION OF THE ACCEPTANCE CRITERIA USED, INCLUDING THE CONSTRUCTION PRACTICES AND FUNCTION OF THE MASONRY WALLS

NUMBER OF MASONRY WALLS	NUMBER OF OPERATING FACILITIES
0 - 50	35
51 - 100	14
101 - 150	7
151 - 200	2
> 200	3
REQUIRE ADDITIONAL INFORMATION	5

TROJAN NUCLEAR PLANT

MASONRY WALL DESIGN DEFICIENCIES

I. CONTROL BUILDING DESIGN DEFICIENCIES

- LER 78-13, MAY 1978
- AFFECTED ONLY WALLS IN THE CONTROL/AUXILIARY/FUEL BUILDING COMPLEX

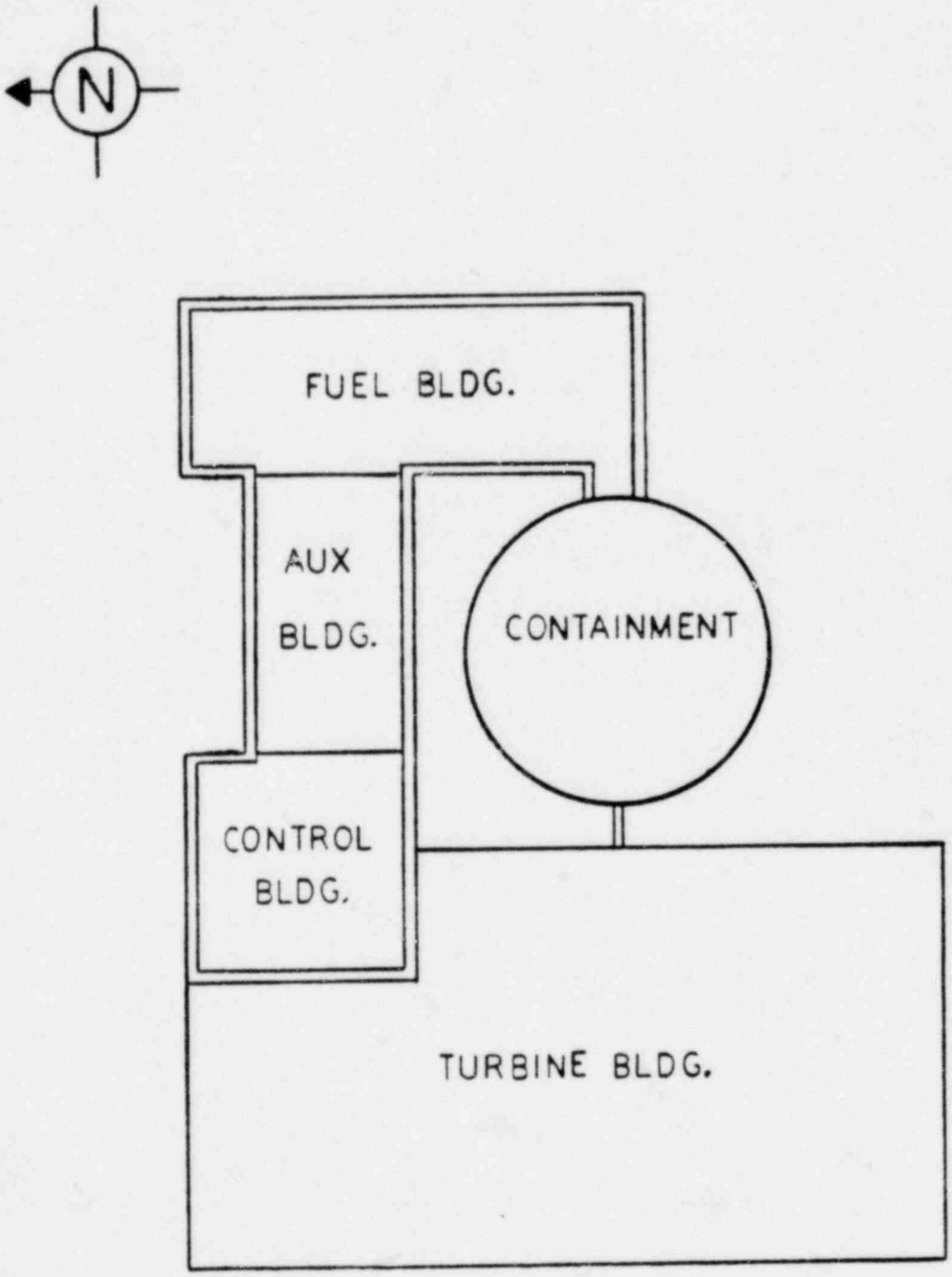
II. WALL PROBLEM

- LER 79-15, NOVEMBER 1979
- AFFECTED ALL WALLS IN PLANT

III. WALL CONNECTION DEFICIENCIES

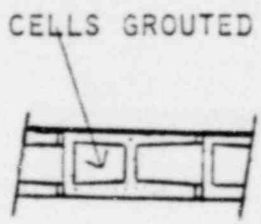
- LER 80-7, MAY 1980
- ALL WALLS REVIEWED BUT ONLY SEVEN (7) WERE SIGNIFICANTLY AFFECTED

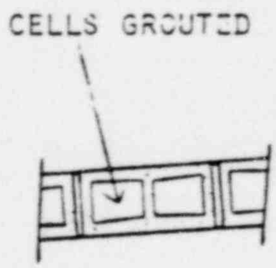
BASIC LAYOUT OF TROJAN PLANT STRUCTURES:



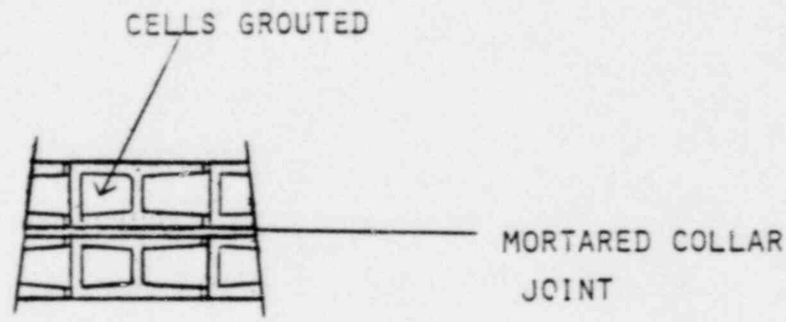
THREE BASIC TYPES OF MASONRY WALL CONSTRUCTION AT TROJAN

A. SINGLE WYTHYE

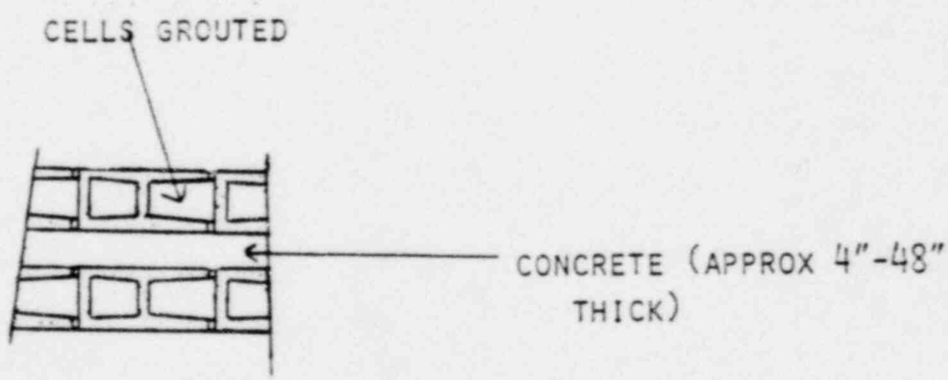




B. MORTARED DOUBLE WYTHE



C. COMPOSITE



I. CONTROL BUILDING DESIGN DEFICIENCIES

A. THREE DEFICIENCIES RELATIVE PRIMARILY TO IN-PLANE WALL STRENGTH

1. TOO MUCH CAPACITY ALLOTTED TO THE CONCRETE OF THE COMPOSITE WALLS

2. CAPACITY ALLOTTED TO THE CONCRETE WAS INCREASED FURTHER BY THE GOVERNING OBE LOAD FACTOR DUE TO AN ARITHMATIC ERROR

3. STEEL WHICH WAS PLACED IN THE CONCRETE CORES OF COMPOSITE WALLS WAS GENERALLY DISCONTINUOUS

B. MODIFICATIONS TO THE COMPLEX ORDERED BY NRC IN MAY 1978

- INTERIM OPERATION OF TROJAN ORDERED BY ASLB - DECEMBER 1979
- MODIFICATIONS APPROVED BY ASLB - JULY 1980

II. WALL PROBLEM

A. Insufficient Design Criteria For Masonry Walls For The Types of Loadings For Nuclear Plant Structures. Primarily Concerned With Out-Of-Plane Wall Strength.

B. Discovered While Conducting Investigations Pursuant To IE Bulletins 79-07 And 79-14

- Wall Carrying Approximately 60 Pipe Supports Was Found To Have Been Designed To Withstand Only Its Own Load During An Earthquake

- Further Investigations Led To The Identification Of General Deficiencies In Masonry Wall Design Criteria

II. WALL PROBLEM (CONTD)

C. MAJOR AREAS OF CONCERN

1. DETERMINATION OF WALL FLEXIBILITY INCLUDING CRACKING CONSIDERATION, AND THUS THE APPRORPIATE DYNAMIC LOAD RESULTING FROM SUCH PHENOMENA AS EARTHQUAKE AND TORNADO

2. EFFECTS OF THE INTERACTION OF IN-PLANE AND OUT-OF-PLANE LOADS ON STIFFNESS AND STRENGTH

3. CAPABILITY OF THE MULTIPLE WYTHE WALLS TO BEHAVE COMPOSITELY

   - RESISTANCE TO SHEAR STRESSES AT WYTHE INTERFACES
   - RESISTANCE TO TENSION STRESSES AT WYTHE INTERFACES INDUCED BY CONCRETE EXPANSION ANCHORS

4. RESISTANCE TO LOCAL LOADS FROM MISSILES, AND PIPING AND EQUIPMENT SUPPORTS

5. CONCRETE EXAPNSION ANCHOR BOLT INTEGRITY

6. INADEQUATE MORTARING OF COLLAR JOINT DURING CONSTRUCTION

7. CONSIDERATION OF LOADS DUE TO INTERSTORY DISPLACEMENTS

II. WALL PROBLEM (CONTD)

D. Status of Resolution

1 - Initial Resolution Was Reached on December 31, 1980

- Further Testing and Investigations Raised Added Concerns in April 1980

2 - Final Reevaluation Criteria Agreed Upon In June 1980 Consisted of Two Levels

- Criteria Which Must Be Satisfied For Operation

- More Conservative Criteria To Be Satisfied By October 31, 1980

3. Criteria Based Upon

- Application of Existing Code Criteria and Test Data

- Development of Appropriate Ranges of Behavior and Strength Where Existing Criteria is Lacking

- Limited In-Situ Testing of The Trojan Walls

- Filling of Significant Collar Joint Voids With Grout

- Insuring Safety System Function

WALL PROBLEM (CONTD)

D. STATUS OF RESOLUTION

4. MODIFICATIONS TO SAFETY RELATED WALLS NEEDED TO SATISFY CRITERIA FOR OPERATION CONSISTED OF

   - 48 SUPPORTS THROUGHBOLTED

   - 100 CASES OF WALL STRENGTHENING OR STIFFENING, AND SUPPORT REMOVAL OR MODIFICATION

5. REMAINING ACTIONS

   - PERFORMANCE OF ADDITIONAL MODIFICATIONS BY OCTOBER 31, 1980

   - COMPLETION OF CONFIRMATORY TESTING

   - QUANTIFICATION OF MARGINS INHERENT IN MASONRY WALL DESIGN CRITERIA FOR TROJAN