UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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254th GENERAL MEETING OF THE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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Nuclear Regulatory Commission
Room 1046
1717 H Street, N.W.
Washington, D. C.

Friday, June 5, 1981

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

BEFORE:

J. C. MARK, Committee Chairman
P. SHEWMON
M. PLESSET
C. P. SEISS
D. MOELLER
M. BENDER
W. KERR
M. CARBON
W. M. MATHIS
D. WARD
J. C. EBERSOLE
D. OKRENT
H. LEWIS
J. J. RAY

DESIGNATED FEDERAL EMPLOYEE:

E. G. IGNE

8106090229

P R O C E E D I N G S

(8:30 a.m.)

MR. MARK: The meeting will come to order.

This is a continuation of the 254th meeting of the Advisory Committee on Reactor Safeguards. During today's meeting the Committee will hear reports on and discuss primary coolant system piping failure criteria, requirements for qualification of nuclear power plant equipment, the integrity of reactor pressure vessels and other nuclear power plant components. The Committee will discuss the NRS safety research program and other topics and hear reports from various Subcommittees.

The Committee also plans to meet with the NRC Commissioners to discuss planning guidance for FY '83 and the ACRS budget and staffing for FY '81 and '82.

Mr. E. G. Igne is the Designated Federal Employee for this portion of the meeting. We have not received any written statements or requests to make oral statements from members of the public regarding today's sessions.

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 will now proceed with the meeting. The first item will be a report from the staff on proposed changes for pipe failure criteria and some discussion of the leak before break.

MR. JOHNSTON: Mr. Chairman, I am William Johnston. This morning in this particular connection Dr. Pawlicki will speak for the staff. I would say at the beginning that this morning we are only planning to give you a progress report or status report. Our review is not complete of these reports that are referenced here.

The SER is under review and Dr. Pawlicki will tell us -- will tell you where we stand in those connections. And the we will, of course, be interested in hearing the comments and questions that the Committee has.

MR. MARK: Thank you. I was also about to say that what we shall hear is a status report, not at this point an officially adopted position of NRR, but there is an SER in preparation so that it is a status report on the staff's position.

I cal. ,on Dr. Pawlicki.

MR. PAWLICKI: My name is Stefan Pawlicki. I am Chief of the Materials Engineering Branch. The topic of my presentation today, which as Dr. Johnston said will be brief, we will primarily discuss this report, on the evaluation of the blowdown loads on PWR primary systems.

In 1975, the NRC staff was informed of some newly defined asymmetric loads. Now, the loads were -- the internal loads result from depressurization of the coolant and the external loads from the pressurizing -- overpressurization in the reactor vessel an the surrounding shield.

Both the internal loads and the external loads can cause severe stresses in the internal components of the reactor vessel and the fuel elements, as well as in the external supports like reactor vessel support itself. As a matter of fact, under these postulated conditions core geometry could be so impaired that core meltdown could result despite functioning of the ECCS.

In 1978 we had asked the PWR owners to evaluate their plants for us. Their responses were submitted in July 1980 and the results of these plant analyses indicate that some plants will require extensive modifications.

At this point I would like to briefly summarize what modifications we are really talking about, what modifications would be involved. The design modifications being proposed by several licensees illustrate the difficulties in achieving a balanced approach to mitigation of asymmetric loads.

The remedies proposed by the applicants as necessary to take care of these loads would involve certain pipe restraints both with the inspection of the primary system, maintenance of the plant, and in many instances would actually compromise the integrity of the primary piping by restraining it from normal motion under thermal stresses.

MR. OKRENT: I am trying to understand from what you are saying whether this discussion is restricted to the question of asymmetric loads inside the reactor vessel, or is it a general discussion of primary system piping?

MR. PAWLICKI: This discussion is limited to the asymmetric load.

MR. OKRENT: Thank you.

MR. PAWLICKI: The safety evaluation report that we are preparing, we are still reviewing, and it handles or deals only with the asymmetric blowdown load as such. So basically, the remedial measures to cope with these blowdown loads are not necessarily a conservative approach, but may result in the lowering of the safety of the plant due to the limitation on maintenance, inspection, and limitation on the normal expansion of the primary system.

Now, how to cope with the asymmetric loads. Some owners have given us potential modifications that may be required. They engaged Westinghouse to make a mechanistic fracture evaluation. They assumed the double-ended pipe rupture is not credible, not a credible event for PWR primary pipes.

We have received a report from Westinghouse, WCAT-9558 and WCAT-9787. The analysis performed by Westinghouse in WCAT-9558 was performed to demonstrate on a deterministic basis that the potential for double-ended failure of the stainless steel primary piping -- and again, this is limited only to PWR piping, primary piping, and stainless steel of this type -- the report demonstrates or intends to demonstrate that the probability of double-ended break need not be considered -- I am sorry. The probability of a double-ended break is so low that it does not have to be considered as a design basis for designing structural loads or resolving unresolved safety issue A2.

MR. PLESSET: Let me ask you a question. This report 9558 was issued in '79, August. I wonder why it is, as far as I know I have never seen it until today.

MR. MARK: Dr. Plesset observes that this report was issued in '79 and it would have been of interest to him, and he is wondering why it only comes into his hands today.

MR. PAWLICKI: I may have problems answering some of your questions, primarily because of the fact that the report was reviewed by one of my men who has resigned his position since that time, and he will know the story much better than I do.

Nevertheless, the fact is that the report was submitted in a draft form about two years ago. It has been discussed by our people, our experts, and who recommended certain recommendations and revisions. So when I am talking about January 1981, I am talking about the final version of the Westinghouse report, 9558.

Revision 1 was submitted last year and some drafts two years ago.

MR. MARK: Yes.

MR. MacINERNY: John MacInerny of Westinghouse.

Revision 2 of that report is currently being printed and will be submitted to the staff in a week or two. This Revision 2 should be the final revision. It was intended to incorporate additional staff comments that we had not received up until Revision 1.

MR. MARK: And the work on this final revision has included work throughout the last several months?

MR. MacINERNY: Yes.

MR. PAWLICKI: This will be Revision 3, right?

MR. MacINERNY: 2, I believe 2.

MR. PAWLICKI: Okay.

MR. PLESSET: Have you made essential changes in the report?

MR. MacINERNY: No, there have been no basic changes in the methodology.

MR. PLESSET: Or the results?

MR. MacINERNY: Or the results. It just addresses certain areas in a little more detail and addressed additional questions that had been raised by the staff.

MR. BENDER: Could you clarify the matter of the distribution? Westinghouse prepared the report. Has it been submitted to the regulatory staff for review?

MR. MacINERNY: Yes. Generally what we have done, the history of this report, is we have throughout the revision, we have given the staff various draft revisions of the report. That was always followed up by a formal submittal to the staff, both proprietary versions and nonproprietary versions of the report.

Now, we address the report only to the staff. I believe the staff has some kind of internal distribution for reports of that nature.

MR. MARK: Proceed.

MR. PAWLICKI: Okay. Now, as I mentioned, I don't know if I made it clear --

MR. BENDER: I do not know how I have gotten my opportunities to see it, to be honest about it, but I presume that there is a channel that distributes these things, unless the regulatory staff has held it as an incomplete kind of report.

MR. FRALEY: Mr. Chairman, we have received copies of this report along the way and we have distributed to it in accordance with our selected distribution list, which means that some members may have seen it and others may have not. But if there is a particular interest in it, we can see that everybody gets copies.

MR. MARK: Well, I think at this point that has just been covered.

MR. FRALEY: I did not want to leave the impression that people were overlooked. Dr. Plesset might not have been on the right list. We will correct that.

MR. MARK: Well --

MR. PLESSET: I just wondered why I was not on that right list.

(Laughter.)

MR. PLESSET: I get a lot of report that I do not care about. I think this is not a problem for Pawlicki.

MR. MARK: Why don't you proceed.

(Laughter.)

MR. PAWLICKI: Now, repeating my last statement, so the next one follows from it, that WCAP-9558 addresses only the problem of asymmetric blowdown loads and has no connection with programs like -- assumptions like double-ended pipe break or other aspects of large break LOCA's involving issues such as containment sizing, radiological release or ECCS design. And I think this distinction is important.

You realize we are talking about a limited application of this report to show that the potential for double-ended pipe break is very small.

MR. WARD: Could you explain why it does not apply to -- over here. Could you explain why it does not apply more generally?

MR. PAWLICKI: I was thinking about it myself and I have not discussed this with the gentlemen who prepared the SER, but we are going to. But my interpretation is the assumption of the double-ended pipe break, it is still conservative. We all knew about it. Therefore, the intention was to make it conservative.

The asymmetric blowdown loads result from a very fast double-ended pipe break under certain conditions and they impose enormous loads on the structures, which to cope with would make it less safe actually than if we make somewhat more realistic assumptions. On the other hand, a slow leak rather than instantaneous double-ended pipe break can also pressurize the containment and also release radioactivity into containment. And the ECCS design should be able to cope with the large LOCA.

So the reason we are limiting it to this application is primarily we feel for this application it is not unrealistic -- not only overly conservative, but unrealistic, unreal[illegible] loads, which require modifications in the plant, [illegible] can interfere with the inspection and maintenanc[illegible] and safety.

MR. WARD: So it is a question of the rate of failure.

MR. PAWLICKI: Yes. Now, our evaluation of these twp Westinghouse reports includes the definition of general criteria that could be used to evaluate the integrity of piping for large postulated loads. But for the time being, we are only talking, again, about PWR's, stainless steel piping, and design of the system.

Based on our review and evaluation, we have tentatively concluded that sufficient technical information has been presented to demonstrate that large margins against unstable conditions exist for stainless steel PWR pipir , postulated for large flaws and subjected to safe shutdown earthquake and other plant loads.

MR. OKRENT: What does that statement mean now? What do you mean by "large margin "? And quantify it probabilistically for me. What is the estimated likelihood of some kind of failure at some rate occurring, with what confidence, or you know, something of this sort?

MR. PAWLICKI: I used the word probability and I was incorrect. The study is based on deterministic -- based on deterministic bases. Probability did not enter into the study at all. Probability of double-ended pipe break -- I should have said potential for double-ended pipe break.

MR. OKRENT: Let me offer a comment, then, now, before you complete action c this. And I am not unsympathetic or sympathetic to any particular approach with regard to asymmetric loads. If I had to guess, I guess I would say the chance of it occurring right there so fast is rather small, is my guess, okay.

Now, this is my intuition, but I think there is building up a body of information about flaws and chances of finding flaws and not finding flaws, and there is some kind of knowledge about what is the likelihood of an SSE, for example, and therefore what is the likelihood of something larger than an SSE. And there may even be some estimates on what the likelihood of other things that might be a source of unusual forces.

In this case, I do not know if water hammer is important, but if you look elsewhere in the system, as you implied, in this system that could be important. And I think, rather than saying what you call just a deterministic basis, at least you should see what probabilistic analysis tells you. It may tell you the uncertainties are 10^{-3} to 10^{-11}, which is what we saw recently as an estimate in another study. If that is the case, all right, that is the case.

Or it may say, in fact for what you are looking at, although the uncertainties are large, they all -- the whole band lies at 10^{-5} or less, which would be interesting, you know. I think a look of that kind would help place this in a less subjective -- it will still be subjective, plenty subjective, but at least you would have taken advantage of what little information there is and what techniques there are.

MR. PAWLICKI: Let me -- I am sorry.

MR. JOHNSTON: Mr. Chairman, if I could comment on what Dr. Okrent said. Outside of the scope today, there have indeed been programs of the sort that you were inquiring about sponsored by the research office. Programs as I understand it have been conducted at Lawrence Livermore which did look at it from a probabilistic point of view; also, another program that was conducted at Battelle Columbus, which is more deterministic in nature.

They are in a sense outside of the scope of what we are trying to do today, which is to talk about our review of this particular input that Westinghouse has asked to have reviewed. So indeed, some of the things that you are talking about are going on in the background and there is extensive other work that we are really not talking about today.

MR. OKRENT: I do not understand. You are telling me there is research going on, but we are nog going to use it? Is that what you are saying?

MR. JOHNSTON: No.

MR. OKRENT: Then all right, what are you telling me?

MR. PAWLICKI: I can offer my own point. It may not be final or even correct. I am not an expert in this area. But when you take a deterministic base, if you compare say stress intensity factor in some plate or member, if you know the toughness, the toughness of the material is three times as high, then deterministically you can conclude that the plate or component will not fail.

Now, of course if there is certain uncertainty about fracture toughness of the loads, you could perform some kind of probabilistic studies saying what is the probability that if, generally speaking, that this component would fail, given that there is one-third of the load that is calculated and tested. And this is the kind of thing we are doing here.

MR. OKRENT: Well again, there is the statement in this WCAT on page 8 that says, the overall conclusion that under the worst combination of loadings, including the effects of safe shutdown earthquake, a realistic postulated flaw will not propagate around the plates in question. This may not be important if the toughness is -- if the toughness is enough, as you have indicated.

On the other hand, it does not say right here what the chance of the toughness not being enough is. I certainly know that the probability of the SSE is not so small that I want to not consider something larger if it could lead to a PWR one or two category release.

Now, if all it is going to do is lead to a LOCA or something, that is a very different story.

MR. PAWLICKI: Yes, I understand.

MR. OKRENT: Okay.

MR. PAWLICKI: Now, as far as -- I do not have the whole report here, the SER that is being prepared. But I have some excerpts in here and the aspect of selection of the flaw size is covered in the report, and basically what it says is that for pressurized water reactor piping there is a good experience that cracking is not very likely an there will be very few service problems for the primary system tubes.

Our criteria for acceptance of these analyses indicate that the flow lengths through the wall should be at least twice as long as the thickness of the pipe. Now this is somewhat arbitrary, at least the way I see it now, but also it should be of such lengths that the leakage from the flaw should be larger than 10 gpm, which is the limit on the allowed leakage.

What I am saying is, the flaw should be large enough that it would be definitely detected by the detection system. When I say definite, there is a question again. It could be argued that there is some probability that it could not be detected.

MR. OKRENT: Well, I think in BWR's we have flaws of the size you just talked about. I do not know whether you should expect it in PWR's. I am willing to be convinced of the fact that they will not occur. But I think just to state, you know, deterministically, I do not know what that means.

MR. PAWLICKI: As far as you mentioned, Dr. Okrent, in boiling water reactors the conclusion section, to which I am coming slowly -- we have one statement which maybe I should read now, that although the safety evaluation has been written exclusively for the primary system piping at the PWR facilities, some of the report is concerned with the generic application. And it says, piping systems other than PWR primary systems have some service history of observed cracking. For these systems consideration should be given to assuming flaw sizes different from those specified for the pressurized water reactor primary system, depending on the history of observed service cracking, the potential for cracking and leak detection capabilities.

So we are only again limiting it to pressurized water reactor systems.

Now, I call the conclusions we have reached so far tentative, from the point of view that so far they have been reviewed by only the other member of the SER, who is no longer with us. It has not been subjected to peer review yet or approval by our manager.

In view of the situation, review is scheduled by June 20. Maybe I am optimistic, but in two weeks time or so we should have it commented upon by people that participated in this study and understand it a little better than say I would at this time.

I would anticipate that the SER would be issued by mid-July 1981 or around this time. We should get management concurrence and it should be published some time in July.

Now, after this overall summary, maybe I would like to describe very briefly what WCAT-9558 includes and what factors can be considered, analyzed, described and so on. Now, WCAT-9558 includes a definition of the plant specific primary piping loadings, includes analysis to define the potential for fracture from rupture and unstable flaw extension. It describes materials tests to define the material toughness and tensile properties and production of leak rate from flaws that are postulated to exist in PWR primary system piping.

As far as the loads are concerned, the loads acting on the reactor coolant pressure boundary piping during various plant conditions includes the weight of the piping and its contents, system pressure, restraint of thermal expansion, operating transients, in addition to startup and shutdown and postulated seismic events, specifically safe shutdown earthquake.

In the design of this piping, the limiting loading combination must be determined, and Westinghouse has prepared a table that -- I have a copy of it and unfortunately I cannot show it to you today. But anyway, it includes a list of 12 plants that were included in the owners group considerations on this topic. It includes location for maximum load, and in 11 cases out of 12 it is the reactor pressure vessel nozzle.

It includes axial loads calculated for those plants and bending loads. The bending loads for these plants are for the axial tension of 1800 kps and for bending moment 1400 kps. These bending loads have been used in the fracture analysis performed by Westinghouse and then evaluated by us.

WCAT-9558 was performed to demonstrate large margins against double-ended pipe break would be maintained for PWR stainless steel primary piping that contains a large postulated load and is subjected to large postulated loadings. If the postulated flaw would grow larger on the application of the load, if any additional crack growth that might occur might be stable and not result in a complete circumferential break -- I think point two is of most greatest importance here. Even if the crack does grow, will it be a stable growth or will it progress in a nonstable manner.

The analysis was performed of the axial and bending loads, the upper bounds of the loads, also in the facilities listed in WCAT-9558. Now, in order -- I am not going to go into details of fracture mechanics analysis, but basically to decide whether the crack would grow at all under the postulated loads, the method was used to check how it compares with the critical load to decide whether the crack, if it grew, whether it would grow in a stable manner or unstable.

The approach that was used was what is called the stability concept developed by Westinghouse and described in NUREG-0311, which assumes of course that the mechanistic flow extension -- in the last analysis -- in the last analysis of the tearing modulus concept, a factor of three margin of safety between tearing modulus as the proper material -- the tearing modulus as calculated from pipe loads. It is comparable to either stress intensity K13 -- basically, it is a margin of safety of K13.

MR. SHEWMON: J_1 or K_1? Are you expressing our margin of safety in terms of stress intensity or what?

MR. PAWLICKI: It is on the Ttearing modulus itself. So basically the tearing modulus of material is based on tests of the fracture toughness properties of the materials would be three times as high as the tearing modulus calculated from the imposed loading. And I can see that Dr. Johnston is there. Maybe he can help me.

Now, I would mention that the criteria that we used to determine the postulated flaw size, I will repeat again that we, based on the service experience and somewhat on our judgment of course, that the flaw should be -- the length of the flaw should be at least twice the pipe thickness and should be long enough to have a calculated leak rate of ten gallons per minute under normal operating conditions. So it would be detectable, instantly detectable, even before the crack.

MR. OKRENT: Now again, you have had experience with flaws much longer than that which were not detectable, as you know.

MR. PAWLICKI: In PWR's.

MR. OKRENT: In reactors. I find it at this stage not for me defensible to assume that because a particular kind of flaw that occurred in the BWR and did not show by leaking, occurred in the BWR, that it is not applicable to a PWR. In other words, if -- if a large flaw in a BWR in fact did not show by leak -- and we have had cases like that -- at the moment I do not know why, if the flaw occurred in the PWR, it also might not show by leak. Because as far as I understand it, that is something related to how the flaw exists in the pipe and whether it is held together tightly and so forth and so on.

Whether or not you expect a flaw size and so forth you may say differs among the two, but that is a different question. But I have a problem with some of your bases. That is one of them.

Let me give you an example of other kinds of questions. Again, in looking at the Westinghouse conclusion, it says they used identification of actual minimum material properties based on the research of Westinghouse quality assurance files. I think that is a good beginning. But I think you have to ask yourself, what goal am I looking for with regard to safety, what assurance do I want to have, and are there ways in which by some anomalous path in fact I have reached material properties far worse?

And of course we have the example in pressure vessels where some of the welds are far more susceptible to radiation damage than other welds, and it could be that one could have had the experience in pressure vessels, he looked at a certain group of vessels and did not have any that was and somebody else did. Do you understand what I am saying?

You are in fact interested, I believe, in a rather high reliability or a very low probability that what you assume will not occur really does not occur. And I cannot tell that in fact somebody is asking are there ways in which the assumptions here might have been defeated. Maybe you have done it, but I have not heard it.

And then you give me the kind of statements you have. And, you know, in looking -- I have also looked at these reports from time to time, and I think you might be able to well make the case probabilistically, but if you do not do it and somebody has not thought about are there things that are weak spots, you may end up vulnerable and maybe even being wrong.

MR. EBERSOLE. May I ask -- right over here, right this way. Is is possible to have a crack which is so small from side to side that virtually no leakage when one considers the contamination by crud or precipitated boron or any other mechanism that you can think of, in fact it does not leak enough that you can see it until it really gets big or, for that matter, to have a crack which never does penetrate the surface?

That sounds impossible to me. I think Dave was thinking about Nine Mile Point, the case where the break was almost, I think, 50 percent circumferential before they found it by a drip. And certainly I think I can take a cracked pipe, and if I carefully cracked it, I can put it in a vise and squeeze it together in a vise so tightly it will not leak.

I do not know whether cracks generate that way or not. Because there is a structural crack does not mean there is a physical leak of fluid.

MR. PAWLICKI: Yes. Basically, what you are saying is you are assuming, say, a 7-inch -- Westinghouse is assuming -- we are inclined to agree with them -- a 7-inch through-the-wall crack, and that would leak at a certain rate. But you are saying that the crack, 7-inch, not through the wall.

MR. EBERSOLE: Or else it was through the wall -- that is right. I think I can take any crack generated, and by appropriate stressing I can close it so it will not leak.

MR. SHEWMON: Not during the cycling of a plant, you cannot; the first part of the cycle, at any rate.

MR. JOHNSON: My name is Richard Johnson. I am in the Generic Issues Branch, Division of Safety Technology.

I would like to address the issue of fracture and fracture prevention, which I believe Dr. Okrent is raising. And I apologize, I am not sure that I followed with precision the line of reasoning. So if I -- perhaps we should have a dialogue rather than me trying to stand up and answer a specific question.

But I think that the issue here centers about the fact that we are talking about pipes with flaws, and we recognize that these flaws may reach sizeable proportions in service, flaws in pipes under accident conditions. One of the things you asked, Dr. Okrent, was can we have a large flaw that does not leak. And I think we are ready to stipulate "Yes."

That is our experience, and that is why we are postulating that we go into the accident with a nonleaking large flaw. Now, large has to be -- it gets to be a little arm-waving in here -- but I think when you see what the staff has done in reviewing the Westinghouse report, I think that the criterion that is set up for the size of the flaw is large. It is certainly within the realm of things that are ordinarily detectable and that the issue is what happens under these large loads, accident loads.

The thing that we are trying to guard against --

MR. OKRENT: What do you mean by "accident loads"?

MR. JOHNSON: Seismic, for example.

MR. OKRENT: Okay.

MR. JOHNSON: For the maximum design loads, is that the proper terminology?

MR. OKRENT: Seismic should not produce an accident load.

MR. JOHNSON: Well, okay. What the ASME criterion would call Level C and Level D.

MR. OKRENT: Excuse me. Seismic should not produce Level D by itself; should it?

MR. JOHNSON: I do not know. I cannot -- it is not proper for me to address that. I am a metallurgical engineer and not a stress analyst.

(Laughter.)

So I will stay out of the field where I have no business entering.

But the issue then is going into a condition with large loads and a relatively large flaw, what is going to happen and what we are concluding is that for the given materials and given conditions, the flaw will grow in a slow and stable fashion. It will indeed .ecome a leak rather than a catastrophic failure. Now, that is the point of all this.

And is there something that still remains that we ought to discuss, Dr. Okrent, within that frame?

MR. OKRENT: All right, I will try to state it again simply. There is some chance -- it may be quite small -- of a flaw larger than whatever it is you have assumed in here. Now, it may be that the results are not sensitive, in other words, that you would get the same result, in fact, if you had a flaw four times as long as what you have said.

If so, you could say that, and that would say the probability of our conclusion being right is not affected by that. It may be that the material properties are worse than you have assumed here. There is some probability of this. You may be able to decide in fact that there is some lower limit on material properties that you see no way of getting -- you know, violating; even at that limit you might be okay.

All right, then, you can say it, or you can say that -- you can argue on whatever reason the probability of getting material property so that is acceptably low based on some experience and so forth.

I would say the argument that at the SSE level you are okay is not a very good one probabilistically because the other parts of the staff are estimating the likelihood of the SSE for most plants being in the vicinity of, let's say, one in a couple of thousand per year.

We have a lot of plants -- that is a round number; sometimes it is smaller, sometimes it is larger. But that is not, by itself, a number that I or, I think, the staff considers a very small likelihood if it automatically leads to an uncontrollable event which has severe consequences.

So, I think, in fact a good case can be made, my intuition tells me. But to avoid looking at this, in fact, may leave something important out. When I say I think a good case can be made, I think it can be made if you say we are not generalizing it to all pipes. But now you have just that much higher probability of something going wrong. You talk about a selected number of pipes, pipes that are looked at frequently, whatever, or something. And that is my guess.

MR. JOHNSON: As far as one aspect that you brought up, the material properties, I think that some tests that have been conducted say that indeed one is using not average but a lower-limit fracture resistance. As far as the flaw --

MR. OKRENT: Excuse me. What does the term "lower limit" mean?

MR. JOHNSON: Within the bounds of the tests conducted that the fracture --

MR. OKRENT: Yes, but -- okay, just so we understand, you do not test every piece of material actually; do you?

MR. JOHNSON: Sampling.

MR. OKRENT: Sampling for each and every pipe?

MR. JOHNSON: Sampling from representative heats. There is a sampling problem that can be addressed.

MR. OKRENT: Again, maybe you can make a good case both for the welds and for the material. You know, there is no reason to assume -- in fact, the probability is zero that the material can have lesser properties. I do not know if you can make that case. If you can, that helps your argument.

MR. JOHNSON: Well, then, you are talking about the material properties and trying to look at them in a probabilistic way. Then one can say, surely, since we have not sampled every pipe that is in every reactor, there may be something lower.

But one does a sensitivity study and says what if the material were somewhat lower than what we have, seeing a scatterband of material properties. Let us assume that, for example -- and rational engineers do this all the time, as we all know -- let us assume that although we think we have a lower limit, there is a material with, say, 10 percent less resistance, what does that do for our analysis?

And when you go through this particular analysis for the piping, with the postulated flaw you find that a little bit less fracture resistance does not really significantly affect your conclusions; namely, that the postulated flaw, if the material had a little less fracture resistance, would still be leak-before-break.

And that gets back to what Mr. Pawlicki was talking about when he said there is a margin in the tearing modulus, which is a term, a measure of the materials fracture resistance or stability. There is also the possibility that there may be a larger flaw than what was postulated, and the same thing is done with a sensitivity analysis.

Supposing there were a larger flaw and the conclusion is not going to be significantly affected by that. Eventually, of course, with respect to the flaw size, the largest flaw you postulate is where the flaw has already come apart and you have to stop somewhere short of that because you are assuming that the plant is operating when you are going into the accident. So the pipe is performing its function prior to the application of accident-level loads. So there is a reational basis for how large a flaw can be -- should be postulated.

And I am not really sure, speaking of that, that I fully understand what you said when you alluded to there being very large flaws that were not detected. The largest ones that I know of in boiling water reactors were the flaws in the recirc lines at Duane Arnold.

MR. OKRENT: You are not analyzing flaws equivalent to that here, are you?

MR. JOHNSON: No, sir.

MR. OKRENT: The plant was running with those, was it not?

MR. JOHNSON: And the plant was leaking like a sieve.

MR. OKRENT: It was shut down at some point, and those flaws were only, you know, somewhat smaller before it was shut down. So you say it was leaking like a sieve, you know, they have to shut down at about 5 gpm. And so I assume they did not grow in a day -- I hope. I am not suggesting that you need to use -- I am saying you might be able to make a good case why in a PWR at this particular site that is unlikely.

In fact, those flaws did exist, and you cannot just wave them away. That is the point I am trying to make.

MR. JOHNSON: If one performed the analysis --

MR. OKRENT: Is there some other mechanism by which you could lead to the same thing here? If there is not, then the probability is low for what reason?

MR. JOHNSON: The Duane Arnold cracks occurred -- one can go through an analysis for the Duane Arnold type of flaw and still come up with the same conclusion; that is, that we would get leak before break.

MR. OKRENT: If you can do that -- in other words, you say your conclusion depends on flaw size?

MR. JOHNSON: Not quite, but almost.

MR. OKRENT: You do not have to hinge it to the flaw size. That is okay if you can do it that way. But you have not -- what I have heard here --

MR. SHEWMON: Dave, this is not a complete discussion of the topic. There have been a lot of reports on that. We have had people here before, a month ago, or years ago is more like it, talking about Duane Arnold and how many degrees around and how deep it could have been and whether it would have leaked before it broke.

I think you are doing them a mild disservice to want an education more than you can absorb and more than they are prepared to give all in four minutes or even 40.

MR. OKRENT: I am, in fact -- I am not sure they have looked at this probabilistically. I have not heard that they have looked at it probabilistically.

MR. SHEWMON: No, they have not often, although I think they have done more than they are prepared to give. There has been the study at Battelle on the cold-line break and that got expressed probabilistically. I think, in effect, you are asking them where the codes are done deterministically to come back and express it in a different language, and it makes them ill at ease.

Now, we can go out to Livermore someday, and they are reinventing the ASME code expressed all in terms of probabilistic analyses. And it is not clear whether they are doing a better job or a worse job. It is just clear that we are spending a lot of money for them to do much the same thing on certain subprojects.

MR. OKRENT: I have a reason for suggesting that they look at it probabilistically, just to provide, let us say, another, oh, mechanism for seeing that they have not overlooked some path. And you sort of ask yourself, you know, have I covered all the ways, whatever it is that I can get 10-5 chance of this occurring, and you ask what other things that can do this, and so forth, and you satisfy yourself there. I think you are just one step further along.

MR. SHEWMON: My point is you can solve some problems with these functions and you can do the same thing a lot of other different ways. And what bothers me a little bit, are we asking have you done things in terms of these functions as opposed to those functions?

MR. OKRENT: The chance of large flaws being in a light-water reactor is not zero.

MR. SHEWMON: Nobody here argues with that. And they have a formal way of going through that and coping with it in their way. But it is not expressed in terms of, "That gives us a probability of 10-8 per reactor-year plus or minus 2."

MR. PAWLICKI: I also feel that is the best basis for any probabilistic analysis is past experience.

MR. SHEWMON: Let me ask a couple of different questions on this. There have been concerns on the part of the staff about what are called "impertinences," I guess, flow deflectors inside pipes, and so on. This may not be part of it here, but it again has to do with what LOCA loads may do. Is there any code that you know of which addresses these things, or is that something the staff reviews or is it something which is left to the AE's good judgment?

MR. PAWLICKI: I am not familiar with this aspect but maybe someone --

MR. SHEWMON: You are standing.

MR. LEVIN: I was ready to discuss another question before you got on that issue, just trying to set the record straight as far as the history of the staff's look at probabilistic evaluation. If you are ready to hear that, I would be glad to --

MR. SHEWMON: Tell us who you are, and go ahead. We will get on to the question.

MR. LEVIN: My name is Howard Levin. I am with the Division of Engineering.

Back in '76 Combustion Engineering owners group and B&W owners group submitted topical reports which were completed by SAIs. And the staff completed its review of them, if I recall, in 1978, and concluded that those reports could not form the basis for walking away, so to speak, from the asymmetric loads.

The reason, as I understand them, was although the methodology was acceptable from a probabilistic point of view, the data just did not support it. In other words, there were not adequate data from a probabilistic point of view, materials properties from a probabilistic point of view.

So, in a nutshell, the conclusion was that that data could not, although the approach was nice and provided insights, could not form the basis for a licensing decision to walk away from asymmetric loads. Therefore, in '78, the staff sent a letter to the PWR owners, requesting them to address asymmetric loads. In parallel to that effort, Westinghouse presented this study.

I do not know if that gives you a little bit of the history, but I think it is just basically a conclusion that that was not an adequate basis at this time as a licensing basis.

MR. OKRENT: I remember that study, in fact, and I guess I do not want to hear of it today, but I think at some point I would like to understand if in fact the data, you know, is insufficient to satisfy the probabilistic analysis.

I have trouble translating it into how you arrive at the deterministic position, because in the deterministic position either you are assuming you can handle any flaw size, in which case the previous problem on inadequacy of flaw-size data is unimportant, or you are saying the flaw size will not be larger than a certain amount. And you will say, "I am relying on engineering judgment," but you would -- probabilistically, it seems to me the same kind of thinking enters.

Okay, I remember that study quite well.

MR. MARK: Mike, could you --

MR. BENDER: Why don't we answer Paul's question?

MR. MARK: Fine.

MR. SHEWMON: My question has to do with there are flow deflectors which are inside piping, and these come loose, if not regularly, several times a year. And so the probabilities are so high that I can grasp them easily. And the question has to do with what design, what consideration, if any, this has gotten in here?

And, if none, as a question of information, whether there is any design code that has to do with these or whether that is all left to the AE and the owner as to how often they want to pick these things out of the steam generator or the pump or wherever they collect.

MR. JOHNSTON: I cannot give you a direct answer to that, Dr. Shewmon. I just do not know.

My understanding from what I have seen of this report would -- that would not be included in the report.

Dick Johnson may have a different answer.

MR. JOHNSON: No, no, no, I do not have a different answer. I only want to remind everyone here that the scope of this problem has to do with violation of the primary pressure boundary, and I believe the things you are talking about are within and do not violate the primary pressure boundary. And although I will not say it is an unimportant problem, I think you are going beyond the scope of the review.

MR. SHEWMON: Well, I could bring them into the steam generator out of the feedwater, I suspect, if I had to; and that would violate primary to secondary. Now, that may not be the primary boundary, and it may be.

Let me ask a different question then --

MR. MARK: Would you picture how that would make a double-ended pipe break in the pressure vessel and an asymmetric blowdown load?

MR. SHEWMON: No, I would not. I am asking another question. Let me come back a little bit closer to this. As a result of going to asymmetric loads or considering the question, I understand that what we did to cope with this hypothetical accident was to increase the strength of hold-down bolts substantially and increase the load which the torque could take.

And also, that is not part of your presentation today, but it will be a part of a presentation next month, I am told. And I would like to have some of this firepower we have here in fracture mechanics and what have you here next month, I guess. And if you have not read of that party, why, by all means, invite yourselves.

(Laughter.)

MR. BENDER: I have some sympathy with the viewpoints that have been expressed here, both Dr. Okrent's and Dr. Shewmon's. I find myself faced with the dilemma of how to bring together these two approaches to addressing the problem. The Livermore people did in fact do an analysis of the piping loading question and came up with some very, very low probabilities of the crack propagating, except they left out a rather important element; namely, the uncertainties associated with mistakes in design errors, detection techniques, and that sort of thing.

I think that is the point Dr. Okrent is making, and it is legitimate to say, "Well, if you are going to make this argument, how much is it dependent upon things being done right?"

Now, can you make that probabilistically? That is the question I want to know, because I think that is the one he is trying to get at.

MR. PAWLICKI: I do not know. I personally feel that it could be expanded to include consideration of probability of either the material fracture-resistance being lower or the load being higher or the flaw being longer than we are assuming. And so judgmental factors would enter: How large or long a flaw do you assume?

MR. BENDER: That is a matter of how sure you are, the basis on which you are making the judgment is the right basis.

MR. JOHNSON: I would like to comment. I think we have gotten off a little bit from what the nature of this review is and what the assumptions are that are behind it. What, as I understand, we are reviewing is a proposal by Westinghouse, is we assume we have a crack that is large neough that it is leaking at least ten gallons per minute. And if we have a crack that large and we then apply certain loads to it as caused by earthquakes or whatever, will that crack grow in a catastrophic manner or will it grow slowly an in a stable manner?

What Westinghouse is trying to show us is that if the loads are less than a certain number, which is on the order of 40,000 kps, and what we are trying to do is review that proposition to see whether we agree with it or not, it already assumes a large crack. It is not a tiny little crack; it is a great big one. It is already leaking at a rate which is detectable, because it is ten gallons per minute. And that is above the tech spec limits. The question is: Will that thing propagate in an unstable manner? Given the kinds of loads that you might get under that circumstance and given the material properties which they have made some tests and offered us a lower limit.

So I think our problem is to evaluate that proposition. It is not a question to us at this point of probabilities and things like that. It is already assumed to be of a certain size. And the question is will it grow?

MR. BENDER: No, I think that is not the whole issue. That is part of the issue. I think the first question is whether we have established the basis that we know the leak will occur; and secondly, that we will detect. There are some other issues involved; namely, having to do with whether the crack would grow in a certain way and the rate at which it would grow.

Those may be deterministic. As a matter of fact, I think they are. But we have to start first with the premise that we accept the idea that the leak is there and that we can can find it at some point. And I do now know.

I have heard the argument that we know that yet; that is a postulate.

MR. PAWLICKI: The only thing I can answer to this -- it is not very clear to me either at this time, but I am limited only to the item I had. Nevertheless, to assure that adequate leak detection systems are in place, the facilities listed in the WCAT report -- the detection capabilities should are listed in the report. I agree with you that some, or many of us, have doubts whether the leak detection systems that do exist are adequate.

MR. BENDER: I am just trying to separate the thing into pieces. If you want to just listen to the fracture mechanics argument, which is what I think you are trying to present to us, then we have to set aside all the probabilistic things and just say how willing are we to accept the fracture mechanics? That is what you want us to hear, and not be concerned about whether we understand the asymmetric load thing. Okay, we will understand it in that context.

MR. JOHNSON: It will have to be evaluated in the larger context; that is correct. What we have received from industry is a proposal that is attacking one portion of it, and we are trying to evaluate that particular portion. But in order to resolve the total A2 issue, the total picture that I have been hearing from the committee this morning has to be included. I did not mean to say that it was not.

MR. BENDER: I am not sure what you are trying to resolve this morning.

MR. JOHNSON: We are not trying to resolve any of it this morning.

MR. BENDER: I will stop.

MR. SHEWMON: Let me bring up one other thing which is on at least my mind and, I think, several others'. And that is, we have with this asymmetric load gotten ourselves into the situation where there is a good chance we are decreasing the safety of the plant, perhaps in nonquantifiable loads, but with bolts that come loose and pipes that cannot be inspected or things that hang up so thermal expansion regularly overloads the pipes. And the probability of that is nonquantified.

But there is still the suspicion, I think, that it is reasonably commonly believed that maybe if we could get rid of some of these things, we would be ahead, whether we can nail it down in a firm quantified manner or not. I do not what we could explicitly give that, but I think it is there.

MR. BENDER: I support that view strongly. But I suspect we are not going to get all that today. Next month I expect to hear a lot more on that too.

MR. MARK: Go ahead, Mr. Pawlicki.

MR. PAWLICKI: Actually, basically, this completes my presentation. I just wanted to maybe recapitulate the conclusion we have reached, tentative conclusion as of today. One, of course, is that based on compliance with our acceptance criteria, we conclude that full double-ended pipe break need not be considered as a design basis to resolve generic problems. This applies only to asymmetric loads in the reactor vessel cavity. This applies, of course, only to PWRs with stainless steel piping. We are not even extending this argument to ferritic piping. Maybe we will, but at this time it is a very limited scope of this conclusion.

Now, the second conclusion is maybe of sort of minor importance. Two domestic utilities have not performed yet their seismic analysis. They are scheduled to complete them by end of 1981. And as long as the maximum bending moment will no exceed 42,000, again we feel that they would be within the group of plants that are acceptable.

Finally, the conclusion that I would also mention to you, just a few minutes ago, is we do require that the leakage detection system in these plants complies with Regulatory Guide 1.45 in order to make reasonably certain, let us say, that large leaks would be detected.

Now, my personal opinion, I have some reservations about the leak detection systems. But it is something that I may be wrong. And we will have to look at it.

Finally, as I already mentioned also before is that the safety evaluation has been written exclusively for the primary system piping at PWR facilities and stainless-steel piping and for piping systems other than PWR primary systems which do have some service history of observed cracking — and this is putting it mildly.

Then, for those systems, consideration should be given to considering flaw sizes different than those specified for the PWR systems, depending on the potential for cracking and leak detection capabilities. This is also different for different plants. We do not have a uniform type or quality or sensitivity of leak detection systems in all the plants we are talking away.

Now, in a way coming to Dr. Okrent's argument, which is definitely valid, but in order to establish either the probability or likelihood of the thign happening, you have to have to some data base, in the first place. And by limiting this to PWR experience, primary piping, and stainless steel, we have formed a set which is pretty well known and can sa- with reasonable confidence that there is no major leakage or cracking in the PWR and the primary system.

MR. OKRENT: How many, I suppose you would call it, discontinuity years, or whatever is the right term, of experience do you have that is applicable in this pipe size and so forth? Do you have, you know, a million discontinuity years, so that you can say a flaw like this has not occurred, or is it a thousand?

Because, you know, until a couple of years ago we did not have cracking in turbines in PWRs; right? So you have to be a little bit careful about what your data base is. You may in fact have quite a few discontinuity years, or whatever measure you want to use, to argue that this is the case. But you have to be a little bit careful when you say it has not occurred so far.

MR. PAWLICKI: I remember that was the case. We were told we had no cracking in the primary system and two weeks later we were told we did. So that is a valid argument. So I do not know how many discontinuity years or whatever we had, but it could be estimated a thousand. Of course, I think, as we all very well know, that the probabilities calculated will depend on the size of the data base. If it is small -- it could also be high. 10^{-4} were based on 10,000 known years of operation.

The final -- just what Dr. Okrent was mentioning, it is not a conclusion but sort of a comment, which says -- and I will read it -- that the parameters chosen by the staff for our evaluation criteria are sufficient conditions and are believed to be conservative. However, a quantitative estimate of the degree of conservatism cannot be defined without additional experimental data. It is likely that experimental data would show that bending moments higher than 42,000 kps would be allowed.

Experiments now being planned by the Office of Research, NRC and industrial organizations such as EPRI should help to clarify this matter in the future. These additional data are not necessary to complete this review.

MR. OKRENT: If I can make one last comment. If I try to translate the thinking from WASH-1400 to your problem, I would ask for a common mode. At the moment the most likely common mode that I can think of is that there is a deficiency in the welding such that at one and the same time you have material that is inclined to run if you have a big enough crack and also it is inclined to develop a big crack.

So that in fact these are not now random events that you multiply the probability of, but in this particular weld the one is connected with the other. And so the question is in my mind, is there a chance that you could have weld material of that kind? I have to rely on Shewmon or someone else to tell me there is a chance or there is no chance. I can raise the question. I cannot answer it.

MR. SHEWMON: Do you want an answer today?

MR. OKRENT: Any time you want.

(Laughter.)

MR. MARK: Can you put it in one word, like yes or no?

MR. SHEWMON: Do we have any handout on this presentation?

MR. PAWLICKI: No, not at this time.

MR. SHEWMON: In particular, I would like to see a written version of what you said your principal conclusions were about what could be used in the cavity for blowdown loads. You read something off.

MR. MARK: There is an SER in the course of preparation and that I think will cover the points.

MR. SHEWMON: That is the official document?

MR. JOHNSTON: That is the official document. It should be available before your next meeting. And in fact, if the Commission is interested we would be perfectly willing to come down and give a more detailed technical discussion. We are simply not ready at this time to do it. We could meet with the Subcommittee or whatever.

MR. SHEWMON: But what he read about the conclusion, I would like to see the words, and I can wait until next month.

MR. MARK: At this moment it is sort of a draft, unofficially unauthorized conclusion.

Thank you. I think this will probably be looked at with interest and also receive more discussion.

Shall we go on to our next topic, which is requirements for qualification of nuclear power plant equipment, and it is a report from members of the staff on proposed requirements for seismic qualification of equipment in operating plants.

MR. BURNS: Gentlemen, my name is Jack Burns. I am presently on detail to the Generic Issues Branch of the Division of Safety Technology. My home base is the Division of Engineering Technology over in Research.

Today I want to present to you a preliminary draft of a task action plan related to the unresolved safety issue A46.

(Slide.)

And that is related to the seismic qualification of equipment in operating plants. What we are concerned with here today is to present this preliminary draft to you and look for your comments and suggestions on how to improve this program as it stands at the present time.

You can see here on the first vugraph, the lead organization is the Division of Safety Technology. I am currently the task manager of this program. It is applicable to all light water operating plants, and we expect the program to be about a three-year effort.

Before we go any further, I want to say that we want to split the definition of qualification into two areas: one is the structural qualification of the component, that is, to see that it does not break, it does not distort beyond possible limits, that it does not become unstable nor break loose from its basic foundation.

The second category would be operational qualification or in-plant operability. To some degree these will overlap with the structural qualification. An exact breakdown in the definition of the two may be hard to come by. But for qualification we are concerned about fluid flow conditions, things like circuit breaker set points, multi-control operation and processing signals and the like. Obviously, with circuit breakers you could say that this is a distortion causing the break. It could be structural or it could be classified as strictly operability. So that is one of the overlapping difficult points.

(Slide.)

The objective of this task is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants.

MR. EBERSOLE: May I ask a question? Is it within your scope, since you are taking a broad view of the problem here, to examine the plant mechanical and electrical equipment and structural equipment, for that matter, to validate that that equipment which is not seismically qualified does not fall down and destroy that which is?

MR. BURNS: Not specifically, because we are not going to be getting into plant-specific areas. What we are going to be doing here primarily is evaluating equipment for qualification.

MR. OKRENT: Can I ask, is the problem thought to be generic in the sense that it is about the same level of possible contribution to risk in all plants, or are there some plants where it has been estimated that the problem has been an order of magnitude more severe, either because of the likelihood, the higher likelihood of the large earthquake, or because of a much lesser degree of confidence in the seismic qualification or some of this kind of stuff?

MR. BURNS: At the present time I cannot answer that. After task one of the program, which is the initial task of digging into seeing what the problem areas really are, we may be able to answer it in greater detail. However, this would not be a risk assessment type of program.

MR. OKRENT: All right. Then let me put the question this way: Would the advent of this particular task A46, does that mean that for all plants the matter is deferred until resolution of this item? Or are specific plants being looked at to see whether they violate some threshold?

MR. BURNS: Plants are right now being specifically looked at, primarily through the SEP program.

MR. ANDERSON: Let me address that. You asked, Dr. Okrent, in what sense it is generic. I think we see it as generic in the sense that it would be a set of generic criteria or guidelines, if you will, for the assessment of the seismic adequacy of equipment in operating plants. As a number of the old plants, of course, and equipment was qualified many years ago to various types of standards, there was concern that the techniques that were used to qualify the equipment originally may not have been adequate.

The whole thrust of this program is to develop the guidelines that would be used to assess the equipment. Now, as far as waiting until all of these guidelines are developed before we do any implementation on the plan, that is not going to happen. The Environmental Qualification Branch in the Division of Engineering has a plan, program plan, which is in final formulation at this point to address qualification of equipment in the plants.

MR. OKRENT: I do not understand what you just told me. You are developing guidelines for -- to get a program in place or something. I missed something.

MR. LEVIN: Maybe I could clear that up for you, Dr. Okrent. I think -- okay, this action plan that Newton is referring to is in the final stages of concurrence within NRR. And it is going to come out in the form of a Commission paper.

But there are certain activities, and I would characterize A46 as a subset of that activity. Let me give you an example of the activities that are going to go on in parallel, so to speak. Clearly, the first thing you have to do in this area is find out what people did in the early days, and that is like activity number one in the action plan as far as the seismic goes. So there is basically going to be a survey, an assessment of what the status of qualification is and what did people do in the late 1950's and early 60's.

I think the work that is being done under A46 will try to address the question, given older standards in light of new criteria, how do we resolve that and what do we do to -- what do we have to do to determine that that older plant is qualified? And that will come out in the form of generic guidelines for assessment.

So these things can go on in parallel, and it is not like we are going to do it in series, we are going to wait until this is done and then we are going to go ask that person a question on operating plants. Some of those details I think on the action plan we can discuss later on.

MR. OKRENT: Let me ask two questions. The first is with regard to the older plants and the evaluation program, if you found that there is equipment important to safety, if that is the right word, that is not seismically qualified or maybe partly qualified, is there some process by which you arrive at the conclusion whether or not you need to do something, you know, and what is the process?

MR. LEVIN: First of all, I would like to characterize the work we have done, and it is I think -- part of the genesis of A46, after having reviewed these 11 older plants, people learned something about equipment qualification and certain suspicions were created. Therefore, the recommendation was made to study the problem a little bit further across the board.

As far as the existing evaluations, SEP is not really making a conclusive finding on the functional capability of the equipment. I think the judgment that was made there was we can only take it so far in that program, tha a larger evaluation was needed. The structural adequacy of anchorage and hold-down was addressed directly, some of the mechanical considerations. But as far as functional ability, it was not addressed in the detail that we think is necessary.

And so what I would like to say is that some judgments -- I would characterize them as engineering judgments -- were made as to the functional capabilities of those equipments, trying to make some argument -- but it was not a systematic, as well documented as we think is necessary. Therefore the SEP plants do not have a writeoff, so to speak, on the functionability.

We have judgments of experts, we have judgments of some staff, and a little bit of data. And really what we need is a prescription, so to speak, coming out of A46 that will tell us how to address functionability more directly.

MR. OKRENT: All right. Now I will ask Professor

Kerr's question: What was the basis for your judgment? Did you have some kind of acceptance criterion that said, we are willing to take a risk of something, whatever it is per year, of core melt due to earthquake, or how did you do it?

MR. LEVIN: To make that kind of assessment, one needs a lot of data and one of the things that we did not have was a lot of data. And quite frankly, we were happy to get our hands on anything that we could that would even suggest that equipment had certain capabilities.

MR. OKRENT: How did you decide was it okay or not okay for any specific piece of equipment?

MR. LEVIN: To be quite frank, okay, there was no -- well, I would have to characterize it as more ad hoc. There was no well-documented logical rational criterion. That is what we are asking A46 to come up with.

MR. OKRENT: Let me ask this question. Were the utilities in each of these cases asked to make the case for why their equipment was good enough, then?

MR. LEVIN: As you recall from previous discussions, plants fall into two groups. Some of the early facilities have that burden. They were sent letters requesting them to do this and they were involved in their own programs and they have to do that.

Some of the later facilities, the licensees really only got involved in providing -- accumulating as much data as they could. And it was determined that the staff was to perform that review.

MR. OKRENT: Now I will get to my final question. Why aren't the utilities being asked to show that their equipment is good enough, and why is the staff developing these guidelines? This is where I was leading.

MR. LEVIN: Okay. They are -- and this is certainly very tentative, because the program plan has not been approved yet. But at least now there are plans for requests of that nature to go out some time at the end of July or August, that would say, tell us what you did, tell us why you are okay.

But this program is directed toward developing criteria to take a look at that and say, well, is that a convincing enough argument or not. Now certainly the best of all worlds would be that we had the criteria, here it is, evaluate your plant. But that would be working the problem -- there is some difficulty in a procedure like that because the program would take much longer. I think it is going to take longer than we all feel is adequate. So we are trying to work the problem as much in parallel as possible.

MR. OKRENT: I don't mind you workig in parallel. I am just trying to understand. You feel this -- I am trying to see whether this is needed or you can just ask the utilities --

MR. LEVIN: We can ask them what they did, Dr. Okrent, but --

MR. OKRENT: Not only what they did, but why is it good enough. Are you going to ask them why it is good enough or just what they did?

MR. LEVIN: I would assume we are going to ask that question. That is the issue. But we have to develop some basis for accepting that argument, and I think we are at a point now where we need more study to do that. We have to resolve the differences between the old criteria and the new criteria.

MR. EBERSOLE: Mr. Chairman, may I ask a question?

What do you do about this rather severe problem -- let me put it this way. A seismic event is the only event I know which affects the whole plant simultaneously at one time. There is very good reason that you have to postulate a zero failure consequence for a seismic event, because if you invoke a thing like the single failure criterion in the presence of a common influence like that, then you are wide open to have the next failure and only two will do you in.

So the staff has pretty much always said that under a seismic event they will not accept any failure whatsoever of any seismically designed equipment, realizing if they did someone would immediately invoke the second failure and the thing would be done. This implies a great deal of conservatism for the seismic event in particular, since that is the only particular event that I know of that simultaneously affects the whole plant design, and the plant has millions of pieces.

What is your general rationale for guaranteeing zero failure in a seismic event, other than extreme conservatism?

MR. JOHNSON: I will attempt to answer the question. At present plants are reviewed against the IEEE 1971 to 1974 -- against, I believe it is, 1971-1974 criteria, depending upon the time at which the CP was issued. And that is the basis for the present plan.

Industry already has a whole series of testing programs under way which are being developed under their funding, not ours, and which they have gotten together various groups and so forth and they are funding the work. The sequence for testing instrumentation, for example, calls for a test in which it seems the various power runs, it is aged, it has undergone the seismic-type shaking on a shaking table, which is done by the industry, which we have to review.

That is the kind of program that is presently going on. That is already ongoing. We have some interim acceptance criteria which will be discussed -- which is the subject of a Commission paper, and I believe there is something to come to you in the next month in this connection, with our hope of issuing these criteria by the end of this year. So you will be hearing details of acceptance criteria for these type of tests, both for seismic and dynamic loads.

All the instruments have to undergo and pass these -- this test sequence, including large pieces of equipment, which are either cut out of the system and put on the table or which transducers are put onto it and shaken according to particular frequencies and frequency distributions, and so forth. That is all a part of the program plan which the Equipment Qualifications Branch has under way.

There is a meeting the week of July 7 with all of the industry to discuss this aspect of things, along with a number of other areas of qualifications. It will be open discussion of these interim criteria.

The final answer I think is yes, it is due -- as a conservative basis, they are required to pass the test, and if they pass the test they are deemed acceptable. That is 100 percent, of course.

MR. BURNS: I have a vugraph up on the board now describing the program outline, again strictly for the A46 program.

(Slide.)

These are three consecutive tasks. The first one is associated with the evaluation of equipment -- equipment seismic qualification methods. This is a general task to get us on our feet more or less, to find the pitfalls, the errors, the need for improvements and so forth, in not only the qualification of components as they have been qualified, but also possible methods of requalifying components.

There are a number of programs that will fit into this as complementary programs, including all those that will be associated with the equipment qualification plan. One of these in particular is a program which I will show in just a second, which is related to a review of seismic qualification methods in general. This is a research program. This is currently being conducted at the Southwest Research Institute.

The second one is the SEP structures program. As was indicated a few minutes ago, this would be aimed primarily at the structural components. The area which we will be concerned with in this task and reviewing ourself in the task will be primarily geared to methods of requalification and merely using the others as supplemental programs to provide the necessary information in that regard.

What we are concerned about, number one, is the fact that we have many kinds of mechanical and electrical equipment to be concerned with and we cannot possibly do it on a piece by piece basis. So in Task 1.A we are saying we are going to develop a category of mechanical and electrical components. Now, this category, I do not know how many it will be yet. We have not completely formulated it. I imagine it would be a category something like those presented in the SSMRP program and the fragility calculations in that program, although they may vary depending on what we get into in this investigation in Task 1.A.

As I say, in Task 1.B it will be the evaluation of methods used to seismically qualify the components themselves, both the methods that have been used to qualify the components as they now stand in plants and the requalification of components.

In Task 1.C it is -- we would draw initial conclusions and preliminary guidelines. It will be exceedingly preliminary at that particular point. The conclusions will be trying to point out the areas for need for further development or further investigation of qualification methods. Primarily, as I see it, because of the complementary programs in research and such, we will be concerned with primarily the requalification of components.

At the end of Task 1, which will be about a six-month program, we hope to be able to get down and really be able to define Task 2 at that particular time. Before I go into discussing Task 2, though, let me show a couple of other vugraphs for a second, mainly on the research program and the SEP program.

(Slide.)

This is the research program. It is an ongoing program started as of June 1 at the Southwest Research Institute. It is a general program on seismic qualification of nuclear plant mechanical and electrical equipment. It is not restricted at all to the review of methods used in current operating plants, but also looking at the present methods for the new plants.

There are about four or five tasks associated with this program. The first two tasks will be used to feed information into our particular task action plan. Task 1 is concerned with the evaluation of methods of seismic qualification of components, and here we are basically going to be developing the advantages, disadvantages and anomalies and so forth associated with the methods of both past and present.

Task No. 2 will be the correlation of the seismic qualification test methods as generated from Task 1, how does one method compare to the other. If you had tested a component or are planning on testing a component by one method, how does it compare with today's criteria or how would it compare with other methods and so forth in a ranking tpe of order?

The correlation functions we have not decided as yet. They will probably be related back to some type of damage function. The other functions in this task are associated with fragility methods and so forth, again related to the overall equipment qualification program, not specifically to this particular program.

The second program which we will definitely feed information into this will be the SEP program. As indicated before, there are two phases in this. The first phase consisted of a five-plant review. It was review of existing seismic documents and limited re-evaluation of those documents.

The five plants -- I understand the results of four of these studies have been released now in NUREG reports. The fifth one is to be released very shortly.

Phase 2, the current phase, and this involves six operating plants, the licensees are required to reanalyze their facilities effectively and update if necessary their seismic designs. This is primarily -- the second phase is primarily concentrating on the structural aspect, not on the operability aspect of qualification.

MR. OKRENT: Why is that? I am trying to understand. It seems to me I read everywhere that the licensee is responsible for the safety of the plant, we want to improve management to know about the safety of the plant and so forth. Why would you not want the licensee to look at his plant and examine it and judge that it is okay and tell you why, or in fact judge that maybe here or there he needs to do something?

MR. BURNS: I cannot answer specifically why the SEP group does not look at the qualification. In essence, they are relying on other organizations, possibly the Equipment Qualification Branch and so on, to do that aspect of it. I have not been into it deep enough to understand right now as to why they have not. I imagine it is asociated with manpower or other needs or something along those lines.

MR. OKRENT: I did not think the NRC staff had excessive manpower.

MR. BURNS: In this particular branch I am talking about.

MR. OKRENT: It is not clear to me you are taking the Licensee's job here. I am trying to understa.d the philosophicsl --

MR. LEVIN: Dr. Okrent, the SEP owners group has initiated an effort. Together they are addressing this concern, and in fact, they are ahead of the rest of the operating reactors in the sense that they are getting together, trying to develop a data base of information that might be useful from one plant to another. That is step number one that they have undertaken, and the burden has been appropriately placed on them. They are working on it, and they have got programs in place, generic cable tray shaking program, etc. They have got -- they have been trying to develop a data base for control centers and various categories of equipment, and then the subsequent steps will be to look at the specific plants and make judgments as to primarily similarity arguments, why their equipment can be qualified at looking at the data base of information. And that is generally the steps that have been taken.

MR. OKRENT: That sounds, I mean, good, but I am trying to understand the NRC approach at the moment, to ask the Licensees to look at the structural but not the performance function ability.

MR. LEVIN: They have been asked, and that is why they have instituted this program.

MR. OKRENT: They have been asked?

MR. LEWIS: Yes. There were 50.54F letters that went out in, I cannot recall, January 1979 or something like that. Don't quote me on the date, but certainly they went out a while ago.

MR. MOELLER: You listed several items of input. What about foreign studies and research?

MR. BURNS: We will be following the HDR program through our research efforts. We have a program in research. We will definitely follow that aspect. I believe the same emphasis will be given to the Japanese programs. Information that comes out of those two programs will definitely feed back into this, especially for things like the HTGR where they are actually putting explosives in their plant as a method of -- which can be interpreted as a methods of requalification.

Not judging adequacy or guidelines for requalification, these will definitely be an input. We will be following those as best we can.

MR. MOELLER: Is there any coordination with them or are you just following?

MR. BURNS: In research we have a coordination contract. John O'Brien is the fellow responsible for it. I believe somebody from AMCO is over in Germany right now through our funding seeing what is going on. I do not believe we have anybody over in Japan right now. I could be wrong in that regard. I just do not know.

MR. KERR: Is anyone within NRS looking to see, given what I assume is the case, that a Licensee cannot do all of the things that he is now being required to do simultaneously, what is the most important, and what should be given priority, or is it assumed that he can carry out simultaneously all of the requirements that are being placed on Licensees?

MR. BURNS: I think I could only say from our guidelines -- from the guidelines standpoint, we would probably put emphasis on the important areas and so on. From the actual licensing standpoint, I am not involved with that area. I cannot answer your question, the licensing procedure.

MR. KERR: Who could I ask, short of the Commissioners, who might know the answer to that question.

MR. ANDERSON: I do not know if I can give you a very satisfying answer or not, but of course, when we put too many requirements on the utilities, they are not hesitant about telling us about it. The clearinghouse for all of this would be the Division of Licensing, of course. They are the people who are in principal contact with the Licensees and the applicants. I am aware that in a number of instances the Licensees have screamed because they do have so many requirements, and they may very well have a good point. But we do not have any real control over that at the level we are working this problem. We certainly intend to coordinate anything that we do through the project managers on the plants, and if we get into the business of dealing with the plants or with the Division of Licensing.

But I am afraid that we are not in a position to do a lot about it right at this stage.

MR. SHEWMAN: On the top, if you just took off in an effort to get through your presentation, I am interested in the seismic qualification of supports for steam generators or pumps, something whose toughness is of concern on a different part of the forest at this point in time.

In what you are doing here, what you would do is an elastic analysis, I presume, if elastically these things were strong enough to cope with all the oscillations that somebody might come with an earthquake, why, that would certify them as being strong enough to keep the steam generators and pumps in place.

MR. BURNS: The applicant is being asked to reanalyze and submit analyses. I would think that today's capabilities would probably be an elastic analysis, and undoubtedly it will get down to considering the supports and such. I think that this would have to be part of this analysis to be acceptable from the SEP people, that standpoint. Definitely, our guidelines definitely, supports will be considered in our specific program, whether it comes out of the structural end there or we come up with it on our own. From my aspect, supports are an ultimate factor. If you have support failure, the component is effectively useless to a large degree. So we would have to consider those.

MR. SHEWMAN: Thank you.

MR. BURNS: Let me go back now to the Task 2.

(Slide)

MR. BURNS: This is our A46 program. Here we say we want to develop methods for qualifying equipment in operating plants. Now, this is a limited type of program. Remember, we do have the overall encompassing program of equipment qualification, and this will in turn fund a number of research and development areas and such to complement this. What I anticipate that we will be concentrating on in this task, it will be on the requalification methods. Our current program is to go on the outside with a funding. I don't know as yet if we have defined what the program really will be after we finish Task 1, before we say whether we want to go to a national laboratory or go out on bid or so forth, but we will go in the direction of what seems most appropriate to getting the best solutions to the problems as we identify at the end of Task 1 and the start of Task 2.

But as I see it right now, since the other programs like the research programs and such will be concentrating on the basic methods of qualification, this program here will probably concentrate more -- mainly on the requalification from the operability standpoint for equipment in operating plants.

MR. BENDER: I may be treading on ground that has been covered maybe one way before, but let me try anyhow.

A great deal of what you can do in terms of qualification has to do with being sure that the mounting arrangement for the equipment is representative of circumstances in the plant, and furthermore, you understand the transmission of the forces.

MR. BURNS: Right.

MR. BENDER: Through the plant.

In trying to develop this task, how is it that you plan to get such circumstances defined? If I wanted to go buy a piece of equipment and somebody says, well, it is seismically qualified according to Category 1 or Category 2 or whatever those things are, what is it that they have told you? Have you thought about what you were going to get?

MR. BURNS: To me, if I were buying a piece of equipment and somebody said it was qualified to Category 1 or 2, I would look to see what qualification, what that really consisted of.

MR. BENDER: What is going to come out of this when you catalogue all this equipment that need to be seismically catalogued? I have got to ask well, now that I have made the list, how do I convert the list into something that represents a physical set of requirements, a circuit breaker, for example? What is the qualification requirement for a circuit breaker? Is it going to be that the breaker has to open and close while the seisuic event occurs or what?

I am just using that as an illustration.

MR. BURNS: I think operabiity is to assure that in a seismic event the component, whether it be electrical or mechanical, will perform its function during and after the seismic event.

MR. BENDER: That is easy to say, but I am trying to say now -- I have said it, and now somebody is going to show me that it will do that. What would you expect?

MR. ANDERSON: Let me try to answer that. I think we do recognize the problem. I see the problem that you are getting at. In two years maybe we can give you the answer.

The problem is the guidelines that are developed will have to include both how you interpret the particular design basis at the plant as well as the mounting of equipment, the function of the equipment. How successful we will be in coming up with a set of generic guidelines for this remains to be seen. That is the problem that we are approachin. We recognize it is a difficult problem, but it is something that we feel is needed. You cannot take a piece of equipment that has a statement tha it is seismically qualified and move it from plant to plant and expect it to have the same degree of assurance that it is going to function.

MR. BENDER: That is where the concerns are arising I think in the industrial element. Everybody says we have to do it better, we have to do it different. But -- and so we are going to start this task action plan, and we are gong to do it different and better. but when you get down to asking, well, how is it going to be different and better, all I can get from people is that it is going to be. It does not seem to me like there has been much analysis of what would have to be done to make it different, better or both.

MR. ANDERSON: Another facet we see is to determine how well we do it. Then the answer may very well be not very well.

MR. BURNS: I think we will be in a better position to answer your question after we have finished Task 1. I think we will have a much better handle as to where we have to go when we get into Task 2.

Task 2 I would not attempt to define what we will be doing in detail in that particular task in the way of a work statement for a potential coutractor.

MR. BENDER: Do you have a schedule for Task 1?

MR. BURNS: I am hoping to see it finished by October, November, sometime in that time period.

MR. BENDER: Who is going to be doing it?

MR. BURNS: Basically in-house i conjunction with the other programs, in specific, the SEP program.

One advantage for the research program, I am the task manager on the research program also.

MR. LEWIS: Could I just follow up one thing that Mike's question reminded me of, using circuit breakers as an example? One of my memories of World War II is that when you shot a gun, the circuit breakers opened on the ship. That was simply a routine event, and they functioned just fine. They did what they were supposed to do, which was open, and then they would also well after the shot because you would close them again. An obvious solution to that is make them more resistant to opening, tighten up the springs. But that produces safety problems of a different kind.

Now, somewhere, somebody must be thinking through these things, and if I understood the sense of what Mike's question was, it was laying on reasonable criteria that are directed toward the purpose of the instrument, and the circuit breakers were a good example.

MR. BENDER: They happen to be a troublesome one, too.

MR. LEWIS: That is my memory. That is my memory.

MR. BURNS: That would be beyond the scope of this program here.

MR. LEWIS: I see.

(General laughter.)

MR. LEWIS: That is an extremely interesting answer.

MR. ANDERSON: That is not really beyond the scope. It is not really beyond the scope of this program, part of the implementation. It is really part of the implementation.

One of the key things you bring up here is the time that it is required to function, and whether it can live through an event and function later, or what its function is supposed to be. A relay, for instance, may very well open up in the chatter, but as long as it is there and will function when required, it is also important. It is one of the important facets of this program dealing with the operability question.

Of course, the operability has to address the time of function and the importance of the function and the consequences of not functioning, are there alternate ways of performing the same function? There are a number of facets to it, and I think that although these may not all show up in the guidelines, we are certainly aware of them and we will consider them in the development of the guidelines.

MR. LEWIS: That is precisely the collection of things that were raised by Professor Kerr a few moments ago becauss in order to do that, you can set your priority list in such a way that these requirements then mesh with all the other requirements on the safety of the plant. There has to be ultimately a safety philosophy that tells you what you are going to emphasize, what you are going to regard as more important, and as you make changes in requirements, how they will have an impact on other requirements.

And I must say, in the efforts I have made to find that philosophy within NRC, because I am defective, I have been unble to determine it.

MR. BENDER: Let me add a minor point maybe to what you just said. Some members of this committee are proponents of seismic scram. One of the bases on which seismic scram will make sense is if it will get the plant into a safe state so you don't have to worry about doing it at the time when the worst shaking occurs. That is an essential element of this evaluation.

If a plant can get itself into a state where the seismic event cannot cause any worse problem than you anticipate, then what the seismic qualification is may be moot.

I do not want to press people to shake equipment whose function has already taken place before the event has reached the situation of -- where shaking is important, and somehow or other you have to get to that logic. So far we have not succeeded to do it, and I would hope this effort would deal with that timing question as much as anything else.

MR. BURNS: As I mentioned when I first came up, one of my purposes here is to get your suggestions and comments, and I have been writing like mad, and we will go over the transcript and definitely want to incorporate or at least consider as many of your comments as we can to improve this program.

MR. RAY: What has been our experience in the world? Do you have any information where there have been rather severe earthquakes and nuclear power plants have ridden through them, we know the pipes have not fallen down, we know the structures have not failed? Has there been a widespread malfunctioning of circuit breakers afterwards?

MR. BURNS: I do not know, but I can imagine there would be some that may have tripped. They may have come back on again with the design.

MR. LEVIN: All I can say is that a very limited data base of experience with particular power plants -- I guess you would have to characterize the experience as being that there are no known failures of electrical control systems or the ability to plan in total to get the plant into a safe state.

On an isolated basis the information is sketchy. We have had difficulty learning what happened at Fukoshima after the earthquake on a component basis. We have other pieces of data at fossil units and things like that. It is hard -- all you can really conclude is the plant in toto met its function. At nuclear plants, you know, the way this conversation has gone, in fact a circuit breaker may trip; that may or may or not be a problem. You may have times to go reset it, but the data base of experience is just so small that I do not really think it can be addressed.

MR. RAY: It might be a good idea to set something up so that the data base can start to be accumulated.

MR. LEVIN: We need a couple more earthquakes.

MR. RAY: They are happening.

MR. EBERSOLE: May I ask a question about Item 1A, please? It says there catalogue of mechanical and electrical equipment, and I can see a huge basket with zillions of items in it that you are going to have.

I would like to ask, if you break that problem 1A down into a system by system basis, for instance, the most important systems at the plant under a seismic condition are those which enable a successful shutdown without the presence of any loss of coolant accident or whatever, and that is a different set of systems than you have for mitigation of LOCA. As a matter of fact, you could argue, since you have to proclaim that seismic resistance is perfect because you cannot have one failure because then you will have two that you indeed will not have a LOCA. So why do we spend all this money on having seismically qualified LOCA mitigation equipment which we do?

The presence of seismically qualified LOCA mitigation equipment is contrary to the fact that we are not supposed to have any damage to that very good primary loop. But there is a differential requirement in the matter of safe shutdown equipment. It ought to be a great deal better than the LOCA mitigation equipment, and the seismic safety margin ought to reflect that fact in particular, that we will not have any failures in that system, since to have one implies a failure of more than one.

MR. BURNS: Very good point.

Yes, in the breakdown -- in the categories, we will consider definitely the functional aspect, again relating back to the systems and so forth. We will be looking at the size of equipment and the methods of qualification, the methods that have been used to qualify equipment. The reason for sizing it in, some valves are small enough you can test it on a machine. Others are too big. You have to relate it back to a combination of analyses and equipment testing. I understand what you were saying. We will have to keep that in mind.

All right, in Task 3, that is the terminal stage, and this will be the establishment of the actual guidelines. We will base our guidelines on the results of the Task 1 and 2 study. We will base them on the results of the overall equipment qualification program, results of those studies, including research study. We will base these on the results of the SEP program.

We expect to be generating these guidelines in final form in two or two and a half years from now. We will release these guidelines, and in terms of a final NUREG report, to have to go through the process for review and so on, sending it ot and so on, and upon completion of this we will also try to endeavor to incorporate or at least initiate changes, any necessary changes, any potential for General Design Criteria Rule changes, and Standard Review

Plan changes and so forth because of this.

So that is the overall program we have.

I wanted to present just one more viewgraph, and I will give you a rough idea of how these programs will flow together.

(Slide.)

MR. BURNS: Essentially what I am showing here in the center block is this generic issue program, this Task A46. Again, we show Task 1 as being roughly a six month phase. Task 2, we believe it will run for about a year or so. And Task 3 will be about, we figure, a total of a year to get it completed.

The object there, by the time we do all the reviews and so on, it takes time. I would estimate that the report, the preliminary NUREG report, will be finished sometime in the summer of '83.

This also reflects back on the research program as the bottom line below the dashed line showing the phases 1 and 2 and how these will reflect in to support the Task 1 and 2 of our program, and also the SEP creeping in.

In addition to this, obviously we have the overall equipment qualification program. Things go on there which will basically feed in continuously. We have a good handle -- we will definitely have good coordination between the research program and the NRR safety program, the generic issues, because I am handling both of them. I am the task manager on each.

That is about my presentation. I can answer any more questions or receive any more comments or suggestions. I would be happy to.

MR. MARK: I guess you have had a number of questions. Thank you, Mr. Burns.

MR. BURNS: Thank you.

MR. MARK: I guess we will have a ten minute break.

(Whereupon, at 10:43 o'clock a.m., a brief recess was taken.)

MR. MARK: We are going to hear a report of the DOE safety assessment of their own reactors. We will hear from Colin Heath.

MR. HEATH: Thank you very much.

As Dr. Mark says, my name is Colin Heath. I am with the Department of Energy. And I am here to give you a report today on the activities at the Department of Energy which have paralleled the investigations post-Three Mile Island to assess the lessons learned from Three Mile Island and to apply them to the Department of Energy-owned and -operated nuclear reactors.

Let me start off by saying that I am normally employed in the area of nuclear waste management. I have addressed you before concerning radioactive waste disposal. I am on temporary assignment, reporting to the undersecretary of the department to spearhead a task force that is working on implementing corrective actions that have been identified as a result of the review conducted by the department.

Let me say that in the audience I have here today Mr. Jack Crawford, who was the chairman of the committee, the nuclear facilities personnel qualification training committee. I will not say that again; I will just call it the "committee," the "Crawford committee," since we have the Kemeny Commission and the Rogovin Report. Mr. Crawford was the chairman of the group that wrote the report, and he has agreed that if we get into very detailed discussions of some of the individual assessments involved, he will assist me in answering the questions, the detailed questions that might come up.

(Slide.)

I think that you all have copies of the vugraphs that I am using, but I will flash them on the screen simultaneously.

First of all, I am just summarizing, of course, the sequence of events beginning with the accident at Three Mile Island on March 28, 1979. Now, let me say that because of the amount of time in this presentation, the things I will be describing to you today are what I would call the "official" investigations and the "official" activities, and in order to simplify the presentation, we will be focusing on those things that are official.

I am sure that, as you are aware, that just as individual utilities and the Nuclear Regulatory Commission have had an ongoing series of reviews and activities following the lessons learned from Three Mile Island, I want you to know that that has also been the case at the Department of Energy and that from -- as soon as the information started to become available, individual field office managers and responsible reactor operators were, of course, conducting ongoing activities and evaluations.

And what we are focusing on today are those official things that were done by the department.

The other thing, I would like to begin with the premise you will observe that the focus in this investigation, the studies are on the defects and deficiencies. And I must say that as I have gotten involved in this, one tends to get an impression that, "My God, there is nothing going right. Everything is wrong." But I think that is a result of the fact that the report is an assessment whose job it is to find out those things that are deficiencies. And I just wanted to give that as a caveat at the beginning, because the conclusion of the report, as you will see, is not that everything is terrible, but the focus is o. those things that do need to be corrected.

Referring just for a moment to the summary of events, there was this committee established in October of '79, shortly after the issuance and the availability of the Rogovin Report. And this committee, its original charter and the reasonfor the name derived from the fact that the initial impression was that the problems at Three Mile Island were caused by inadequate operator training. And so the committee started out bein he nuclear facilities personnel qualification and training committee.

Now, shortly after the committee was formed, the

Kemeny Commission report was issued, and it became apparent that the problems were deeper than just personnel training. And so the scope of the committee was expanded to include all of the elements of the Kemeny Commission report and not just personnel qualification training.

The committee prepared a deatiled plan of action which was subsequently improved and, as you can see, spent in excess of a year doing detailed evaluations and submitted a report to the undersecretary of the department on March 10th of this year.

The undersecretary immediately responded to that report and identified the necessity to respond to the report and created a group to prepare an action plan to respond to the findings of the report. And as you will see as I get into the presentation, you will see that this was high top-management people who were given the direct responsibility to take whatever corrective action was required.

This action plan was submitted to the undersecretary on the 14th of May and was subsequently approved on the 20th of May, and the Secretary of Energy was briefed at the time so that he knew what was going on in terms of corrective action.

And the task force that I am heading was created on that day, and I am here to tell you about where we are after about two weeks since this task force was created.

(Slide.)

Let me just mention in passing, the membership of the committee that did the evaluation. The chairman was Mr. Crawford, as shown here. The point that I want to illustrate here is that high-level people within the department were selected, all at the deputy assistant secretary level. The two people who have the asterisks on, they left the department with the change in in the administration around the end of January, early February, but they were participants in the early work of the report.

(Slide.)

Just to refresh your memory, since I have said that the Kemeny Commission subjects were the subjects of the report, you may recall that in the Kemeny Commission report in their recommendations sections, they had these following topics that were identified and the Crawford committee then developed equivalent headings.

And the only changes that were necessary, of course, is that where the Kemeny Commission was looking at the functions of the Nuclear Regulatory Commission, the Crawford committee was looking at the functions of the department, safety overview functions; and where the Commission was looking at the utility and its suppliers, the equivalent entity within the department are those programs that have the responsibility, the line programs within the department, for example, the assistant secretary for nuclear energy.

The other topics that are addressed are identical to those covered by the Commission.

(Slide.)

Now, what are we talking about here in the scope of what has to be addressed? How many reactors does the Department of Energy own? It turns out the Department of Energy owns and operates 84 reactors. Of these 84, 67 are operable and 17 are on standby condition, many of which do not even have fuel loaded into them.

Of the 67 operable reactors, they break down as follows: 8 naval prototype; 22 test reactors -- this varies from the FFTF down to the Argonne fast-source reactor which is a very low power level. We have some production reactors, two basic types: the NPR at Richland, with the power level shown; and three reactors operating at Savannah River, C, P, and K. The rest of the reactors out of the 67 are small critical research or zero-power reactors or transient test reactors.

Now, when the committee addressed the scope and the number of reactors the department had, it elected to select from those a representative set for more detailed evaluation. And they selected this set using these types of criteria:

(Slide.)

They focused on those with the higher power levels, on which would therefore represent the highest inventory of fission products. They used the potential off-site risk as a criterion, and they tried to maximize the number of organizations which would be subject to the review, the number of program organizations that were dealing with reactors. Using these criteria, the committee identified 13 reactors owned by the department which were considered representative for the purposes of the survey.

Now, for comparison purposes, we put the key parameters of the Three Mile Island reactor at the bottom of the chart. You will see in this list in which the reactors are listed in order of descending power level, the NPR has a thermal power of 3800 megawatts thermal down to the bulk shielding reactor with a power of 2 megawatts thermal.

I would like to call your attention to the conditions that are indicated here by comparison to the conditions at Three Mile Island. With the exception of the LOFT facility, which, of course, is intended to duplicate conditions in a commercial reactor; and then, of course, the NPR, which has as one of its products steam as the generation for electricity.

With the exception of those, the primary pressures are significantly lower, and the average primary loop temperatures are significantly lower than those that are normally found in commercial reactors, with the obvious exception, of course, of the liquid metal reactors, the FFTF and EBR2. So these were the representative reactors listed. And we also listed their location for completeness.

(Slide.)

Now, as part of the review done by the committee of the 13 reactors, in addition to extensive review of the documentation and the procedures and the incident reports and the like, there were also conducted detailed on-site review of four specific reactors. And the four reactors that were subjected to this detailed review are the four shown here: K production reactor at Savannah River, ATR in Idaho; HFIR; and the HFBR.

Now, the obvious question is: Why was not the N reactor analyzed? It had been planned to analyze the N reactor, but during the time of the committee's primary activity, there was a labor stoppage going on in the Hanford area and it was just not possible to be able to get in there and conduct an assessment during the time of the labor problems.

However, the assessment of the N reactor is being conducted and has been conducted by the field office operations. And it was certainly not our intent to neglect that. But the specific committee headed by Mr. Crawford looked to these four.

In addition to extensive preparations, the field reviews were conducted by nine to twelve people. Each review team was headed by a member of the committee. It included qualified consultants both within the Department of Energy and from the private sector.

Nine reactors that were not visited, their assessment by the committee was limited to document reviews and interviews with key personnel.

(Slide.)

Turning to the major findings, first of all, of the Crawford committee, I have reproduced them exactly here with quotes. These are exactly the texts in the committee report.

The first finding was that, "There was no evidence that any of the DOE-owned reactors are being operated in an unsafe manner or that any of these should be shut down immediately." However, they did find a number of significant deficiencies existing in the management activities and both on-site and at the headquarters.

Now, the obvious question is: In the light of these deficiencies, why did they find it acceptable not to order the shutdown of the reactors? I think that in the same sense of the reviews done of Three Mile Island did not lead to a recommendation to shut down all operating reactors, I think you have a similar situation here. The report stated that they felt that the continued operations were justified as long as there were expeditious corrections of the defects that were identified in the report.

They also took note of the fact that the majority of the DOE-owned reactors have significantly greater safety margins because, first of all, many of them are very remote locations in the middle of large government reservations. Some of them have infrequent operation and, of course, the properties of pressure and temperature that were previously identified.

The other third major finding was there a need to strengthen substantially both the technical and managerial capabilities within the department, and that at the time of the review many of the TMI "lessons learned" had not at that time been adequately addressed or applied in the programs.

(Slide.)

Now, let me just point out -- I think I am going to go out of order a little bit from what you have -- there were some recommendations made by the committee that fell into a specific category, and that had to do with the organization.

Now, the subtask of the subcommittee of the Crawford committee that wrote these recommendations acknowledged that the recommendations did not necessarily represent the only feasible solutions and acknowledged that the reviewers were not in all cases not in the best position to understand all the implications of the recommendation. And so they characterized the recommendation as one of several alternatives which could be considered to improve a significant safety area.

Now, they have specific recommendations that it was necessary to ensure continuous attention at the level above the assistant secretary level in the Department of Energy. As you will see shortly, the changes that are being made in the department's organization will ensure that the undersecretary receieves information on a frequent basis and is kept aware of the safety situation.

But the organization responsible for this will not directly report to the undersecretary, rather, the assistant secretary for -- and this is a a long title -- environmental protection, safety, and emergency preparedness, which we call "EP," for short; that this assistant secretary is a secretarial officer who does report to the undersecretary, and it would be his responsibilty to ensure that top management was kept aware of the safety activities in the department.

There was another recommendation: to establish an independent safety overview group reporting directly to the undersecretary. And as you will see, it has been decided to place the group within the ASEP organization. That has the necessary skills and qualifications and is operating at a level where they will be able to put their primary emphasis on nuclear reactor safety alone.

And finally, the recommendation to establish a group of experts external to the department to advise the secretary. Now, in this regard, it was noted that this committee, the ACRS, is available to work with the department in the design of and review of new reactors, as I understand has been done with the FFTF, for example. And so it was not felt necessary to create, if you will, an ACRS for the Department of Energy.

Now, then, I would like now to talk about the ways in which the department has responded to the committee report. And what I will be doing is focusing on the findings of the committee as opposed to the recommendations, trying to show you how the specific findings of the committee were respond * to by the department.

Now, as I said, rather than going through each of the individual recommendations and comparing them to the alternates that in some cases were selected and in some cases the recommendations were accepted, I am going to go back to the basic findings and compare those to the actions taken by the department.

Let me just say, however, that as you can see here, the report was submitted on the 10th of March, and the undersecretary directed these officials of the department, who were the assistant secretaries who were then there, as you will see, they are practically all acting. but I think this is because of the fact that as with many agencies in the government permanent officials have not yet been appointed and confirmed in many cases. So the people who are acting in the role of assistant secretaries, these people, of course, are the defense programs and the nuclear energy and the director of the office of energy research. These represent the heads of these program organizations that have responsibility for operating nuclear reactors.

This is the very long title for the acting assistant secretary for environmental protection, et al., that I mentioned, and then the deputy director of administration in the department.

As this group began to review the Crawford report, many of them, because of their line responsibilities, were instructing people in their organizations to respond to specific findings and to take corrective action on a routine basis. However, the collective work of the assessment of this group was represented in an action plan submitted to the undersecretary, as shown, which after review and consideration was approved. And at that time I was appointed to head this task force for those things that were called for to be done within headquarters.

Now, some of the things that needed to be done needed to be done by the field office. And the undersecretary the following day directed the field offices in a written order to proceed with those things called for in the action plan. That was their responsibility.

(Slide.)

Now, what I want to go into now is -- hopefully, I will not get bogged down too terribly much in detail -- but what I have done here is I have tried to describe the basic findings that the committee found in each of the areas that correspond to the chapter headings of the Kemeny Commission report and characterize the type of findings that were made.

So, starting with the first function, which would be equivalent to the NRC function in the Kemeny Commission report but in this case it is the DOE safety overview, there were six findings found in this category.

Now, the first one has to do with the direct personal involvement in the program of the top management. And the committee found that because the top management in the department has so many other priority things that it is responsible for, that it really did not have adequate focus on nuclear reactor safety. And the committee called for a higher level of attention by the top management.

The next one, referring to the independent reactor safety overview and as illustrated in their main recommendation, the finding was that this was not technically an administratively as strong as it should be and corrections were needed. And they also noted, as said in the next finding there, that the organization level in the department was too low. The reactor safety overview function was at a level that, in their opinion, was too low in the organization for top management visibility and should be elevated.

The next finding speaks to the technical capability within the DOE safety organization, a finding reflected in the main finding that I have already shown you.

The next one had to do with the degree to which the department's management directives, or what we call the "DOE orders," was specific in their requirements. And there was a finding that too many of these DOE orders were too general, that the guidelines were too general, and that what was needed was more special and specific requirements.

And finally, they observed that there was some confusion in some cases in the line management authority with regard to safety. Now, in some cases, for example, defense programs, there is a very clear line of authority to the assistant secretary for defense programs.

But in other cases where a field office operations officer was reporting to the undersecretary and so was the program assistant secr·tary, there was some clarity needed in the organization. And they felt it was imperative that that be straightened out.

Now, in response to this category of findings, the department is proceeding with the following activities:

(Slide.)

First of all, we are clearly establishing a direct line of authority in areas of safety to eliminate any confusion in this regard.

(Slide.)

And as this diagram shows, the line of responsibility for nuclear safety is being very specifically and firmly designated to the operations programs, to the programs that have the operational responsibility. And in every case, the responsibility in safety will flow up from the operations office managers through the program secretarial officers to the undersecretary.

(Slide.)

Secondly, then, in terms of strengthening and clarifying the safety overview responsibilities, the first thing was to establish clearly that the departmentwide overview has the responsibility of the assistant secretary for environmental protection, safety, and emergency preparedness, and that this official will have the lead responsibility for clarifying and improving the required policy and directives and conducting the independent assessments.

And then an office of nuclear safety has been established. And, as you can see in this diagram here, this is an office that reports to the deputy assistant secretary for environment, safety, and health.

(Slide.)

An? this particular office previously was at a division level. It is being elevated to a full office level. And while I have the diagram up, just let me say that later on we will discuss the fact that there has also been created an office of quality assurance and standards.

(Slide.)

And finally, in response to the first findings, we do have an ongoing activity to review and update the safety directives. Within the last two weeks, we have had a team of about 30 people, field specialists in. We have reached the point where we have a first draft on clarified orders, and we anticipate that we will have revised final orders with clear delineation of safety responsibilities and requirements before the end of this month.

(Slide.)

The next general area, as covered by the Kemeny Commission, had to do with the utility, whereas in the case of the department this would be the program line management. And there were a number of findings made here, 12 in all. Let me just try to summarize them for you.

First of all, the first one had to do with the uniformity of the quality assurance guidance. And it was found that different sites and their quality assurance guidance in many cases were making reference to different basic DOE documents. There was a necessity to restore some uniformity.

The second finding had to do with how the quality assurance policy was being applied. They found that in some cases quality assurance directives from headquarters were merely handed on to the contractors without a detailed analysis of what the specific application should be to the unique characteristics of the site at which they would be applied.

The next one had to do with the scope of the QA program. There was a finding that just as the Kemeny Commission faulted the Nuclear Regulatory Commission for focusing only on those pieces of equipment which were categorized as safety-related, there was a finding that the quality assurance program had been tending to focus too much on those things related to safety without giving equal attention to things like maintainability and reliability and the like.

The next finding had to do with the degree to which organizational independence for quality assurance personnel was being applied at the contractor level. Observations were made that in some cases people were performing quality assurance on a part-time basis in addition to other duties that they had.

The next finding had to do with the documentation. The citation was that there was considerable diversity in the quality of the documentation for various contractors.

The next one had to do with operating procedures. The finding here was that in many cases the operating procedures were not specific enough in requiring second-party verification of things like valve lineup. There was inadequate requirement for independent inspections of systems after maintenance, and the like.

The next finding had to do with the adequacy of the shift relief procedures and -- excuse me, I skipped the operating procedures.

In the operating procedures the finding was that they were not specific enough, that too often there were citations of the operator was instructed to check the water level or to check the operability of something without the specific direction of what it was that that person was supposed to be checking for.

Shift relief procedures, nonuniformity in requirements to ensure that there was a conscientious checking of the conditions at the plant during the time of the shift relief.

The adequacy of control room procedures just has to do with control of access to the control room, permission to enter the control room, and the degree to which activities could take place in the control room which might distract from the main function of the control room.

The next finding had to do with the adequacy of incident reporting, a finding that that is was uneven, that in some cases incident reports were being faultily prepared and not adequately distributed.

The next finding had to do with, as I said before, at the time of the review the "lessons learned" at TMI on operating procedures had not been fully implemented and looked at by the department people.

And finally, this goes back to the comment of the Kemeny Commission about the use of words like "safety-related" and "systems important to safety" and just a finding that it was going to be necessary to clarify this and use a common set of definitions within the -- all the department programs.

(Slide.)

Now, the responses that the department is making to this -- I think, as you sense, most of these findings are the kinds of things that have to be corrected by clarifying the specific instructions or regulations within the department, if you will.

However, in addition to that, the responsibility for the coordination of quality assurance in the department is being moved to this office. I identified for you before an office of quality assurance and standards, and there will be -- this office will have the primary responsibility for issuing clear standards and guidelines.

However, in the meantime, as part of this task force activity, we are conducting a rewrite of the basic quality assurance order, addressing those standards which should be uniformly applied. And a serious activity in this office will further identify standards which will be made mandatory for across-the-board use in the department.

In terms of the operating procedures, the operations officers are to report to the program secretarial officers and to the assistant secretary for the environment by the end of this month on an assessment of the operating procedures. And there will be an assessment of these reactor operating procedures set up on a continuing basis.

The uniform reporting system will be established under the assistant secretary for environmental protection, and we will be establishing a uniform DOEwide system to ensure that these reports are prepared, that they are analyzed for their impact and distributed with the associated analysis to ensure that incidents that occur at one location, that if they show some trend or some "lessons to be learned," that they are recognized at other locations.

And finally, an organizational introduct... is to establish a safety council which will ensure a continuing management overview and to allow the exchange of experience and key data in the safety area, probably on a six-months' basis.

(Slide.)

The next area then had to do with training, a number of findings here, I think, if I can summarize them. There was a call for a more uniform set of providing unambiguous requirements for selection, training, and qualification; a need to have a departmentwide definition of minimum requirements; and a call to have a specific office at headquarters for that function. And this, of course, will require the revision and the upgrading of the management directive in this area.

There was a finding that it was not clear that there was a comparability between the personnel requirements on operators of DOE reactors to those imposed on the operators of licensed reactors, and that a finding about the way in which operator performance assessments were conducted, a concern that too often this was left to the operator's immediate supervisor and the next line of management above; and a call that more independent review and assessments needed to be conducted; a finding that the knowledge in many cases of the basic fundamentals needed by reactor operators was in many cases was not what it should be.

One particular finding that I found particularly interesting was a finding that there was inadequate knowledge of what the meaning was of the operating limits that had been set on the reactor and what the implications would be if the operating limits were exceeded.

Finally, a recommendation, a finding about casualty drill training, that in terms of radiological emergencies, many of the drills that were conducted were not exercised on those in which significant accidents had taken place and a need to do that.

And finally, a finding about the use of written and oral examinations and a strong finding that written examinations should be prepared anew each time and not just use the same examination each time a group of operators is qualified.

(Slide.)

The response of the Department, here again, in many of these cases you will see, the response is to improve the Department regulations. And although that sounds kind of trite, obviously that is the first place you start to make sure that the requirements are clear and specific. The Office of Nuclear Safety will have the program and overview responsibility. The operations officers will be provided the capability and expertise to appraise the contractor's performance. And the -- as I say, the Department of Energy standards will be developed on an expedited schedule, and they are being addressed by the task force that I am directing.

(Slide.)

In the area of technical assessment, these findings were in a slightly different category. The first one was similar to ones previously made, and that is at the time of the review these technical assessments by the individual contractors and by the individual field officers were not yet complete and in many cases had not been got to. And the committee found that this was the case and urged that these be made promptly.

The next one had to do with the fact that a number of DOE reactors are located in ventined confinement buildings, rather than containment buildings. And the committee observed that in the case of a large uncontrolled release that occurred at TMI, that such a release could lead to significant doses possibly, although not necessarily exceeding emergency limits.

And the finding said that that needed to be looked at in more detail. They did point out that in some cases of release of radioactivity there may be a possibility of difficulty of access to the required operating stations. However, while making that finding, the committee also observed that because of the nature of many of the DOE reactors, the power level for example, the inherent heat capacity, that in many cases it may be possible to provide walk-away type protection and that what was needed was a detailed review of the decay heat removal system to see if indeed this could be provided.

The next finding had to do with the fact that one of the phenomenon at TMI that of course caused a problem with gas was the zirconium-water reaction, and the Committee found that most of the DOE reactors are aluminum fuel and they pointed out that the temperatures required for an aluminum-water reaction exceeded the melting point of the aluminum fuel, and so that they felt that a more detailed analysis of what the gas generation potential might possibly be in DOE reactors needed to be conducted.

And then finally, a finding that the human factors, which of course was a significant problem at TMI, needed to be more systematically evaluated for the DOE reactors.

(Slide.)

The response in this area is primarily in terms of conducting these assessments.

MR. KERR: Excuse me. May I ask a question, please.

MR. HEATH: Beg your pardon. Yes, sir.

MR. KERR: The impression one gets is that many of these evaluations did not exist or were not not available, for example walk-away type protection features, evaluation of aluminum-water reactions. They simply had not been done for many of the reactors. One did not find the evidence that these things had been investigated, is that the implication?

MR. HEATH: My understanding -- and perhaps Mr. Crawford could help me here -- my understanding was that in some cases they had been done, but in some cases it was felt that perhaps a more rigorous analysis might be called for than the analysis which was available.

Would you like to amplify on that, Jack?

MR. CRAWFORD: Many of them have been started, well advanced in some cases, but not brought to a sufficient state of completion or not done with sufficient thoroughness, at least in our view, that one could draw a definitive conclusion from it that they ought to be pursued with more rigorous completeness.

MR. KERR: Thank you.

MR. HEATH: I think that, Dr. Kerr, you are focusing on something which I think I have realized in the preparation of this material and reading the report. If I have not done so, I think it is important to stress that if the finding -- if the finding of non-uniformity for one or two reactors did not seem to be up to snuff, you know, a finding would be made that, okay, it is not comparable. And it is very possible that there are some reactors where there exists an excellent review and an excellent calculation.

And the key findings seem to be that, okay, there are clearly some cases where excellent reviews have been done, and so this demonstrates that they can be done. And the point was, however, that the Department was not requiring that this level of review be conducted at all its facilities.

MR. CRAWFORD: One important aspect is our feeling in many cases, by doing a more systematic and orderly analysis and taking full advantages of developments that had occurred since these current reactors were first put into operation, that in point of fact our reactors, many of them would show a better advantage, the status of safety would appear to be better than it is based on present documentation.

MR. KERR: Well, I asked because I am not quite certain what I am supposed to conclude from this report. It is rather general. I do not know whether I am supposed to conclude, for example, does a finding that all of the reactors need further investigation or that some of the reactors need further investigation, that things are in reasonably good shape or that things might be in reasonably good shape if one investigated further.

I am looking for something that will permit me to draw some sort of conclusion. Can you help me?

MR. CRAWFORD: I think I can offer this solution for you, Dr. Kerr. I would suggest that perhaps the thing to do would be to provide Dr. Kerr or anyone else who is interested with the detailed backup report in the specific area of technical assessment. I have a copy with me.

MR. KERR: I think I am being told that I really should not draw any conclusions from this report, then.

MR. CRAWFORD: You can draw general ones. But I think if you are looking for more detailed specifics on the individual reactors, that you could find them in the backup report. I mean, I think it would be very -- it would be impossible today here to discuss --

MR. HEATH: Maybe I can answer directly the conclusion I would hope you would draw from the briefing. Given the length of time of the briefing, clearly it is not -- I mean, I am already going -- as you can see, there are many, many different findings. It is just not possible to go into a great degree of detail.

I think what I would like you to draw from the briefing is, first of all, that the Department at the highest management level takes the issue of reactor safety very seriously, and that they think that the report has served a useful purpose in that it has identified a number of areas in which the safety program needs to be strengthened, and that we are moving vigorously to strengthen that program as is needed.

Now, that is really the overall impression that I think, within the time of the briefing, that I would like to give you.

MR. OKRENT: Can I ask, does DOE have some kind of a quantitative safety goal?

MR. HEATH: I am not aware of one that has a number assigned to it. But perhaps I am not understanding your question.

MR. OKRENT: No, I think you understood the question. I am just trying to see how DOE is going to judge that an acceptable level of safety exists in relation to -- how some measure -- the question was whether there is any intent of applying probabilistic risk analysis to any other reactors.

MR. HEATH: Let me answer the first part of the question, and that is that there is a Department policy that says that in terms of NRC regulations, that it is the Department's policy to apply an equivalent requirement to comparable reactors that the Department is operating. And so one of the things specifically that we are going to establish in the Office of Nuclear Safety is a function in which the specific NRC regulations will be analyzed in detail for their applicability to the DOE-owned reactors.

And if it is found that the basis for the NRC requirement is common to some feature of a DOE-owned reactor, it is the Department's policy to apply an equivalent requirement to its own reactors.

Now, in terms of a probabilistic safety assessment, I am personally not aware of any such activity. But let me hasten to add that I have been working on this management task force for about two weeks now. So I do not know if any of the other DOE people here are in a position to advise me on it, but I am not aware of any such activity. Are you, Jack?

MR. CRAWFORD: I am under the impression -- I am under the impression that -- in fact, I know in some cases that for their relative value the proper -- the probabilistic techniques are used by individual contractors, but certainly as a Department policy I am not aware of any such thing.

MR. EBERSOLE: May I ask a question? Your report appears to step off from the TMI-2 incident and the Kemeny Report and so forth, and to be oriented toward doing good things that can be done by improvement of operations and generally the managerial aspects of improving reactor safety.

Prior to TMI-2, did you have a standard base from which to proceed such as our GDC's, our general design criteria, and our other regulatory requirements as you know, the GDC's and the regulatory guides, et cetera, et cetera, as a basis for starting and both considering the operational as well as the design aspects of your machine?

MR. HEATH: Well, I think the answer to that is yes, that there were and always have been a series of management directives. The old ERDA manual chapters were the basis for providing these basic safety directives, and so we are not starting from a clean slate, if you will.

If I can characterize the findings, the findings seem to be that the specific requirements were not being adequately and rigorously applied, and in some cases this was because, as orders have been redrafted, that perhaps in some places we at one time said --

MR. SHAO: People tended to stick in there --

MR. HEATH: And it was a matter of making sure that these things are followed up on.

The answer is yes, I believe there was a basis for that, but what we are addressing now is whether or not that basis is being adequately carried through.

MR. EBERSOLE: Do you have, for instance, a set of general design criteria?

MR. HEATH: I will have to defer to one of the safety people here, and either we have a set of general design criteria equivalent to -- what are there, 67 in the NRC?

MR. EBERSOLE: 80, I believe, 87.

MR. HEATH: This is Andrew Pressesky, Nuclear Energy Group.

MR. PRESSESKY: Yes, in the -- at least for the reactors for which the Assistant Secretary for Nuclear Energy is responsible, we do have general design criteria for at least all the large and more recent reactors, like the advanced test reactor and the FFTF and the EBR-II.

What we did in this case is we took the NRC general design criteria and asked our contractors to develop the analogous set, because obviously the general design criteria for light water reactors might not be entirely applicable to liquid metal reactors, for instance.

MR. EBERSOLE: NRC found it necessary to expand those originally in a form called safety guides, and subsequently into regulatory guides, which are now great stacks of things, and then imposed other regulatory requirements by other documentation. Do you have such an operation going on, too?

MR. PRESSESKY: To a certain extent, we do, perhaps not to the legal detail that NRC does. But the general design criteria are then reflected in system design descriptions for specific reactors. Since our reactors do not have much in common with each other, it is difficult to develop a library of requirements to apply to all reactors. There are some common elements.

MR. EBERSOLE: As a case in point, can your reactors sustain a rather long sustained loss of all AC power without damage?

MR. PRESSESKY: Again, speaking for the more recent reactors I am familiar with, like the FFTF, yes, that is correct. Yes, we have analyzed that.

MR. EBERSOLE: What about the production reactors?

MR. PRESSESKY: I am not responsible for the production reactors, so I really cannot tell you.

MR. SHEWMON: Isn't DOE responsible for production reactors? What are we talking about here today?

MR. PRESSESKY: Yes. I represent a rather narrow aspect.

MR. HEATH: I think the answer to that question is we do not have a person from the Defense Programs Office here today, and I think we owe you an answer on that. I personally do not have personal knowledge of that and Mr. Perseski does not, either.

Yes, the Department is responsible, and I think what we should agree to do is to supply you with an answer on that because I do not know the answer.

MR. PRESSESKY: I suspect the N-Reactor does have this capability, since it was evaluated by the NRC and by yourselves, I imagine. So I think that question is probably --

MR. EBERSOLE: However, I do not think our analyses have brought an answer to that question very well on our own reactors. I know your own concepts of power reliability are quite different from ours, and your design methods are entirely different and maybe a lot better. I am not sure about that.

MR. HEATH: I just do not know the answer to that question on production reactors. I think we owe you a response.

MR. CARBON: This safety group that you said you were going to set up to make some sort of comparability to

NRC safety standards and DOE safety standards, how large a group would that be?

MR. HEATH: Well, one of the tasks that is faced by my task force is to identify exactly what the scope of work needs to be for these groups, what the required skill levels are, and finally to go out and recruit the people. One of the -- I think one way to encapsulate some of the findings of Mr. Crawford's report is that many of the problems that they observed have to do with the capabilities of the people.

And one of the things that has happened historically is that the Department of Energy has evolved from an agency that was primarily a nuclear agency, and many of the specialists in the nuclear area have now found themselves working in other energy technologies. And one of the things that I am tasked to do is to identify precisely what our requirements are in terms of skills and numbers and the like, and then go out and recruit them. So I do not have that answer yet, but I hope to have one in about two weeks.

MR. CARBON: Do you have a feeling in general for the approximate magnitude? Are you talking a group of five people or 50 people, or do you have any feeling?

MR. HEATH: The working groups that went home yesterday, that gave us some preliminary estimates, If I remember correctly they had recommended that the Office of Nuclear Safety have about 30 people in it at headquarters. But whether or not that is a valid number or, you know, whether it is inflated or it is too small, I am not in a position to answer that because I have not yet been able to analyze it in detail.

MR. CARSON: Thank you.

MR. BENDER: The last time you came in here was to discuss the waste management program. I want to congratulate you on attracting these more, what, controversial type of assignments.

(Laughter.)

But I wanted to ask a question. Dr. Kerr made earlier reference to the analysis of various kinds of accidents. It is true that in many cases the analyses that were done, and reviewed by this Committee, incidentally, were based on premises that have changed since the TMI event. How is that question being addressed?

That is, for example, the idea of hydrogen generation is not a new subject, but the consequences that came out of the TMI accident showed that hydrogen occurred after the reactor was shut down. Is somebody looking at those contingencies?

MR. HEATH: The answer is yes. The people, obviously, who have the most expertise, are the most familiar with the specific design details, are the people at the operations offices and the reactor operators. And yes, as part of the slide I just took down, one of the findings was that at the time the committee went out in the field -- that was last summer -- they felt that these assessments had not been carried through fully.

And one of the assignments is to ensure that these assessments are done according to -- when you say according to TMI lessons learned, yes, it is in these categories. So there is a requirement now being imposed at the field office level for the reactor operators, the contractors, for example, to perform those detailed assessments. And then they will then report back to the headquarters function and provide those assessments, and then an independent judgment will be made as to whether those have been adequately applied.

So yes, the Committee found that at the time they went out into the field, that people really had not got going too well on that, and they really needed to get a kick, if you will. And so, yes, there is a specific requirement to assess those reactors according to those lessons learned.

MR. BENDER: Thank you.

MR. SHEWMON: Let me ask a different question. Some of these reactors are rather venerable. I am not familiar with the dates of all of them, but certainly some are 20 years old, some older. Is there any program to do a once a decade review on some of these to see which particular corrosion problems should be inspected for or what kinds of basic changes in design philosophies, or how long you want to keep them, how you want to try to update these things to current criteria?

MR. HEATH: Well, I think the answer to that is there is nothing like an automatic once a decade. There is a requirement, and one which we intend to follow up on more rigorously, to have a continuing assessment on the technical capability of these reactors.

Now, the issue about the age and whether you continue to keep another one running or you go into Congress to give you the money to build another one, I think that is a continuing assessment that takes place and is really one of the prime functions of the management of the agency, and that is the responsibility that is continuing.

MR. SHEWMON: I guess my concern is not so much that it be an economic one as that it be an engineering one, for certain kinds of degradation that occur and change the safety criteria or the ability to meet them that was there when the plant was new or the component was new.

MR. HEATH: Let me take that as a piece of good advice. Until such time as I am successful in recruiting somebody, I am going to be the Acting Director of the Office of Nuclear Safety, and I will take that as a piece of good advice.

(Slide.)

If I can move on, if there are no other questions for a while, the next category in the Kemeny Commission had to do with worker and public health and safety. The first finding here again addressed the adequacy of the headquarters function and support in this area. Basically this comes down to a finding about the number of people, professional health physicists, for example, and the scope of work they had, and I think the finding was that this was an area that needed to be beefed up.

The next finding had to do with control of radiation and radioactivity, and I think this is a finding that paralleled that in the Kemeny Commission, that there needs to be more emphasis on the control of radiation and radioactivity at times other than during emergencies.

And the next finding addresses the same area, and that is the need to put more emphasis on ALARA, as low as reasonably achievable, during the routine operations, not just during the emergency things. And in that finding specifically there was a finding for a more precise requirement on minimum standards for dosimetry practices.

The next finding had to do with the ALARA program. Here again it was a case of uniformity. In other words, one case found where significant reductions in personnel exposure had been achieved over the past few years and another case where the personnel exposure was really very unchanged. So the finding was that there should be a more rigorous application of methods to reduce personnel exposure.

A finding about the requirement for drills for radiological control people, the need to be more of them; and here again the emphasis on what I would call more day to day situations. There was a phrasing which I had difficulty with which talked about day to day emergencies, which I think is kind of a contradiction, but I think what was intended there was if you have a minor spill or, for example, there needs to be more drill on that type of activity.

And then finally, a finding that the contractor's own internal audit programs were non-uniform, that in some cases the schedules were too infrequent.

MR. MOELLER: Are you familiar with and do you plan to utilize the health physics appraisals of the commercial nuclear power plants which have recently been implemented?

MR. HEATH: I am not personally familiar with them, but that was one citation that was made in the action plan report, that there are -- there is one at the Albuquerque operations office which is felt to be quite good, and there was reference made to these commercial ones. And a recommendation was made that those might very well serve as the model for a standard DOE one.

(Slide.)

So in terms of, you know, what are we doing about all that, I think here again you have to -- the focus obviously is on making sure that the standards are there and that these things are required and that they are followed through on. There is an ALARA guide that has been used on a trial basis, and it is our intent to issue that as the standard and to impose that and more emphasis on the field office auditing the contractors' programs to make sure that their assessments are more frequent and more rigorous.

(Slide.)

In the area of emergency planning and response, the basic finding here was that all of the reactors have emergency plans. But what was found was there was here again a non-uniformity. There had not been at the time of the review the preparation of a DOE-wide plan for response to a nuclear emergency.

As a result, the levels of emergency varied from site to site, some confusions as to who would have what role, what the appropriate role would be for the headquarters people versus the field office people, who the appropriate spokesman would be; in the case of other federal agencies, particularly FEMA, a need to establish in the case of an accident at a nuclear facility who would be the spokesman and who would be in charge of the interface with the local communities; and finally, a finding that there were places in which improvements were made in the emergency preparedness area.

In response to that, let me say that that has been a specific topic of one of our working groups in the task force.

(Slide.)

It is generally found that the emergency plan that exists at the Hanford site is excellent and that has been used as a model for developing a standard requirement for emergency plans for all the other sites, and that will be translated into a directive.

We do intend to proceed with a modified memorandum of understanding with FEMA to make it clear as to who will play what role in the case of an accident, and then finally we will impose as a requirement that we will have at least annual drills in cooperation with state and local governments on emergencies at Department of Energy reactor sites. Some of these drills have taken place, but we see a need to impose it as a requirement on a regular basis.

MR. MOELLER: In this subject area, how many DOE facility emergency plans has FEMA reviewed?

MR. HEATH: I do not know the answer to that. Do you know the answer to that, Jack?

MR. CRAWFORD: No.

MR. HEATH: I guess that is another one we owe you an answer on. I do not know.

MR. MOELLER: And it is also mentioned in the comments on the Crawford report that DOE will assist the states in emergency planning. Do you have a formal program there?

MR. HEATH: We have -- we have a formal agreement with FEMA. One of the issues that is being addressed now is whether or not we have provided sufficient resources to the field offices to allow them to follow up on that agreement. The field offices in the various regions have the responsibility, so that if a state is seeking assistance, for example, the State of New York would have to work with the Chicago operations office. There is an issue in the Department right now as to whether or not adequate resources have been supplied to the field office. And as part of our review of the organizational requirements, this is an area that is being addressed.

MR. MOELLER: You mention in your review the objectives of emergency planning. I wondered if you might send us the list of DOE's objectives in emergency planning, as well as this Richland report which serves as your example, the Richland emergency plan.

MR. HEATH: Fine. What I think I would also like to agree to do is, I really do anticipate a uniform DOE plan being approved within the month and I will send you that.

MR. MOELLER: And you talked about the problem of FEMA and public affairs. Could you elaborate on what that was?

MR. HEATH: Well, yes. I think one of the lessons from the Three Mile Island incident was some confusion as to who the spokesmen were and what the access to the press were, and the finding of the committee was that when they visited the particular site and they said, well okay, when you have an emergency who is going to be the principal spokesman, it was not always very clear who that was going to be.

MR. MOELLER: Thank you.

MR. EBERSOLE: Concerning emergencies, I note that one of your big plants has the capacity to be able to be operated, safely shut down from a point ten miles away. I think that is a fascinating idea and certainly one that we do not have now.

Does that mean that if you have a large disastrous fire in the control room in such a plant, that you have an independent capacity ten miles away to shut it down?

MR. HEATH: I am not sure which plant you are referring to, but that is my understanding.

MR. EBERSOLE: Thank you.

MR. HEATH: Which plant would that be?

MR. EBERSOLE: Savannah River.

MR. HEATH: Yes, that is the intent.

(Slide.)

The final chapter or the final heading of the Kemeny report was, of course, the public's right to information. And I think I've already touched on this. It mostly has to do with whether or not the public affairs people had adequate planning.

One of the key things was if you have an incident, say at Richland or at Savannah River, who is going to be the responsible person at headquarters? We do have an emergency operations center at headquarters. It is clear that we need to establish more standard procedures.

One of the things, for example, that is going to be put into the new orders is that we are going to establish emergency plans for each of the centers and have control copies of them, and to ensure that control copies of the emergency -- of the current emergency plans are available in the central emergency response headquarters in Germantown, so that in the case of an emergency the officials in

Washington are looking at the same plans as the officials in the field.

(Slide.)

So these are the kinds of things that are under way to promulgate specific policy responsibilities and procedures. And then finally, as part of the annual meetings, a review of emergency procedures, to get together with state and local officials, and also cover the public affairs aspect.

(Slide.)

Now one final thing that had to do with the task force, but of course it was not a chapter in the Kemeny Commission, is this is all great but what about the resources. And many of the people in the field offices are very concerned that if the Department is going to undertake to do these things, will we be sure to have the resources available, the right number of people, the right number of personnel slots, if you will, and the money.

And that is a major concern expressed both in Mr. Crawford's cover letter to the Under Secretary and also in the action plan, and the agency has agreed that as part of the output of my task force we will identify specifically what those requirements will be and that we will make those requirements known to the Assistant Secretaries and the Under Secretary. And the Assistant Secretary for Management and Administration and the Under Secretary have agreed that within this fiscal year they will review those requirements and reprogram people and funds if that is required.

So I think that is a summary. I see I have already run over my prescribed time. But Mr. Crawford and I are available for further questions.

MR. MARK: You probably may not have gotten to this yet, but I believe you indicated that the training and perhaps some harmonization of requirements for perhaps reactor operators or people doing the same thing for one reactor or another would be reviewed and possibly in some cases upgraded.

I think you called attention to someone not knowing what the technical operating limits applied to were, even. Do you see a need in connection with operators to bring them to some level of knowledge of the technical subjects that are involved in the operator -- in the reactor behavior?

MR. HEATH: Yes, I do. Quite frankly, this has been the subject of quite a bit of debate. I think there is -- on the one hand, there is a firm recognition that you cannot qualify an operator like a commercial pilot and he can therefore go and run any reactor. In DOE reactors particularly, they vary immensely from one to another, and they are just basically different characteristics, the K

Reactor versus EBR-II, for example.

So there is a clear recognition that you need a degree of certification and qualification that is specific to individual reactors. However, there is also a belief that we need to establish a minimum set of fundamental requirements, basic heat transfer knowledge, for example, fluid mechanics, reactor physics, whatever. And one of the things that the Office of Nuclear Safety is going to address is to what those minimum fundamental requirements should be, and then we are going to address how those should be applied.

And we are looking at the possibilities of having some kind of approval of curriculum, some kind of a standard course. Those are various things that are being looked at.

MR. MARK: So one would think that you would expect a person to be at least a high school graduate so far as his general knowledge of reading and writing was concerned, and then above that courses which would fringe on some particular items, as what we might refer to as college level of information.

MR. HEATH: Yes, that is correct. It would be our intent to define a clear baseline, you know, across the board. But obviously there is another degree that is --

MR. MARK: That will bring itself in sight in half a year or something like that.

MR. HEATH: That is probably a reasonable estimate. I am not -- there has been some work done in the past. There was a contract with a university looking at that and I am not intimately familiar with all the details. There is some previous work that has been done that is going to be carried forward.

It is clearly our intent to establish a baseline minimum set of fundamental requirements.

MR. MARK: Thank you.

MR. MOELLER: You mentioned your unusual -- you did not, but your report did -- the unusual occurrence reports filed by the DOE reactor operations group. Do you know if those are provided to the NRC or whether the NRC makes use of those? Could someone tell me?

MR. HEATH: I do not know. Do you know, Jack?

MR. MOELLER: We can--

MR. HEATH: First of all, let me say that it is my understanding that the ones that will be distributed throughout the Department will be available, and so it seems to me that it is most reasonable that they will be supplied to the NRC.

MR. MOELLER: It is something -- you mention in here analyses that DOE conducts of the LER's from commercial nuclear power plants. Can anyone tell us, either now or later, what sort of studies you have done and what you found out?

MR. HEATH: Well, I think I am not aware as to whether or not this has been very rigorously done in the past. What I am saying is that we intend to have a function in headquarters that will systematically, on a routine basis when these LER's are available, we will review them and make a finding as to whether or not they are relevant to the Department of Energy operations. I do not know what has been done so far.

MR. CRAWFORD: I would like to make two comments on that: One, what we found in too many cases was the frequency of reporting unusual occurrence reports -- in one case over a year's time only two would come out of a site when there were many more than that.

Secondly, in the years since the separation of AEC into ERDA and NRC, there has been a gradual but nevertheless perceptible -- perception that the various elements within DOE have kept track of ongoing developments in NRC and in the utility community.

The purpose for which we recommended this establishment of a central activity would be to effect a greater transfusion both ways between our experiences over into the NRC domain and conversely, particularly with emphasis to our desire to get that experience and feed it out for its potential applicability for our sites. And that would be the focus of this group to which Mr. Heath referred.

MR. MOELLER: One last question. You mentioned that you did not intend to establish within DOE or set up by DOE an external review group, since there does exist the ACRS. In what ways might you possibly call upon this Committee?

MR. HEATH: Well, I think you may be in a better position to know this than I. It is my understanding, for example, during the design of the FFTF that there was a review by the NRC staff of the design and some recommendations made which were incorporated. Now, I do not know whether the FFTF was the subject of an ACRS meeting or not. But I suspect that it very likely was.

And that is the intention, that when we go forward with a new reactor, as FFTF, that we would take advantage of that.

MR. MOELLER: At previous times we used to every couple of years review, for example, the Savannah River reactors and so forth. And this has not been done in recent years, but it used to be done.

MR. HEATH: I see. Are there further questions?

MR. CRAWFORD: A comment I would like to make. With respect to that question that you just raised in proposing the establishment of a group of experts external to DOE, but reporting to an advisor to the Secretary of Energy, it was our thought, the committee's thought, that we needed a group reporting to the Secretary whose reviews of the whole DOE safety situation would permit them to apprise the Secretary of how things were going, basically. And we expected them to attend not only to design, relevant research and development that was done, but a whole spectrum of activities that really add up to nuclear safety. I mean, how the construction efforts were being carried out, fabrication, testing, maintenance, a look, a comprehensive look at the whole activity and an independent view available to the Secretary to say, you are doing well in these regards, you need beefing up in these regards. That was our purpose.

And the second point I would like to make is that I hope you will not infer from what has been said here today that I have changed the position or any member of the committee has changed the position we put forward, the need for this committee to which I have just referred and also for the need for this oversight group within DOE, the continuing one of which Colin is now head, to report to the Under Secretary.

MR. HEATH: Let me just say as part of the clarification that at the present time as the head of the task force I am reporting to the Under Secretary. But it is the intent that after we have completed the task force phase of the work, that the organization will permanently reside and be responsible to the Assistant Secretary for Environmental Protection and --

MR. BENDER: I am just inclined to ask a question that we often ask of NRC applicants, and that is: What kind of criteria is the NRC laying on its own administration in terms of the capabilities -- excuse me, DOE, to deal with the safety issues? If the Assistant Secretary for Environment and Health has a responsibility, what level of knowledge is implied by that?

MR. HEATH: Well, the Assistant Secretary is a presidential appointee and the one -- I think one of the reasons for putting the emphasis on the director of the office of nuclear safety is the fact that that is where the specific requirements will be directed.

Now one of my tasks, quite frankly, is to find somebody so that I do not end up being the head of this office. And I will be the first to admit that I have a Ph.D. in nuclear engineering, but I do not have many years of reactor operating experience. Now one of my jobs is to find that kind of a person, who combines the academic credentials with hopefully the many years of reactor operating experience.

And one of the things that the agency -- that we are empowered to do is w are going to try to define precisely the kind of person we want for this job and then go find them. Now, I have a crosswalk, if you will, of everybody in the Department of Energy with nuclear engineering backgrounds, and we are going to search through that. And if we are unable to find somebody that we think meets the requirements, then we will have an intensive recruiting activity, and hopefully we can find somebody who is willing to come in at the civil service pay level and who walks on water and performs that function.

(Laughter.)

MR. CRAWFORD: Could I make one comment on that? I would like to make one comment on that. The committee's report pointed cut the possibility at least that you would not be able to -- you would not -- the agency would not find it possible to recruit -- restore the technical management capability within the civil service rules.

We were very mindful of the fact of what we were able to do under the civil service -- under the personnel rules under which the AEC was allowed to operate, under which we believe the NRC currently operates. The type of people that we are talking about really have to be -- have the qualifications I think you have in mind, Mike, and it will be tough to get them, and some extraordinary measures may have to be taken.

MR. HEATH: Let me just say, privately if individual members of this Committee are in a position to make recommendations of some sterling individual for this job, I would be delighted to hear them.

MR. MARK: You mean is somebody about to resign here?

(Laughter.)

I am afraid I am going to have to switch toward our next agenda item. I would like to thank the people from DOE, who really gave a very interesting picture of their taking this matter up. Thank you, Mr. Heath.

(Whereupon, at 2:50 p.m. the Committee was recessed.)

MR. MARK: The meeting will continue. We will take up the topic of the integrity of reactor pressure vessels and other matters. Paul?

MR. SHEWMON: There is under tab 13 a summary of the meeting that the Subcommittee on Metal Components had the other day, and there was an interesting variety of items that came up. The meeting was set off by a request from the AIF group that had been wrestling with the question of supports for steam generators and reactor coolant pumps and such things.

You may remember that this started with North Anna when they realized, with LOCA loads and such things, the supports under some of these critical components in the primary system might collapse. And the staff came back then with a ruling saying, if you go to appreciably tougher steels or exclude all those that might be brittle, then everything will be all right.

Once they got to looking at this more carefully, they decided that that is a rather awkward way for them to skin the cat. So there really was not a very tidy meeting with regard to things we could get hold of, because the staff in the next month or so will come out with what their revised set of requests would be. And there was palpable dis-ease on the part of the utilities because it was not focused as well as it might have been.

I guess I got some interest in this because it slowly sank through my head that one could indeed probably have a LOCA or something very awkward if the support of one of these major components did collapse during say an SSE. And so I guess on that one there is the A12 group. We are waiting for the next installment and there is active interest.

A different item, which I guess we will hear about in more detail, sort of came up -- focused for the first time at that meeting, and this had to do with bolts. We often have asked about the specifications which apply to hold-down bolts that are there to assist in the safe shutdown earthquake and other things, and it seems to me we have gotten answers before, it is not clear that they were valid answers, but one of the interesting -- what happened, apparently, historically, is that when asymmetric loads came in there was a certain generation of plants that already had concrete poured and support thing put on the steam generators and pumps for the bolts. And after the asymmetric loads came in -- or let me postulate this is the way it is. I think it is accurate.

Somebody decided that they needed a lot more hold-down force, so they ordered stronger bolts. And one can certainly buy stronger bolts. There are some very strong materials developed in later years.

But these bolts have an interesting characteristic, and that is that if you torque them up to actually a lot above the yield, when you get up to the high-strength levels and put them in a most atmosphere, they will break. And it may take months, but cracks develop and grow. And we had one or two of these.

And so this is something which has people's attention. And on that one the staff said they would like to come back next time because -- and that means next month. It is partly a matter of, can we get rid of these very strong bolts, and the other is, do we have to tension them on to seven-tenths of the yield or seven-tenths of the code applicable situation, which has been the situation for relatively lower-strength bolts, I think, where you have -- you cannot develop near these kind of loads in the bolts. So that we will hear about next time.

The third one that came up and was touched on had to do with a different set of bolts, this time holding the -- in the primary boundary, and these can be either holding onto the pumps or I guess manholes on steam generators sometimes. And people have been doing inspections on some of these and find that they have appreciably thinned. And so -- I do not know.

One story I heard was that if you put a borated water solution on and then dry it off, and then moisten it up again, you can get this problem. I have not heard a good explanation of why. But there will be more inspection of this.

This is not an inspection where you can put an electronic transducer on it. And so the only way to inspect them with certainty that is available now is to go in and look. And so that was another interesting item that came up, sort of as a second installment, or first installment of something that people will be looking at again.

The third -- the fourth item, and the thing we will take up today in more detail, or the staff will, has to do with thermal shock and repressurization. Thermal shock of the pressure vessel has been a source of some concern for a long time and I guess it has been post-TMI-2 or actually, better, post-Rancho Seco, maybe, that it has dawned on people that one can get significant cooling.

And post-TMI-2 there was an order that if the ECCS comes on in a small break LOCA that the operator would put on his high-pressure pumps and leave them on for 20 minutes or some appropriate time like that. And if you are cooling down the inside of a pressure vessel, it could in the future, as pressure vessels get older, I believe is the conclusion now, go below its ductile-brittle transition region, which means with repressurization you could get thermal shock.

If the pressure vessel is old enough and has enough copper in it, it could go all the way through. And so some of the scenarios that are coming out, the calculations being done at Oak Ridge, the HSST program, is that it looks okay for this year and next year, but the year after that we are not so sure. And I think that those people are trying to decide what they want to cope with next. I hope we hear more about it today.

MR. CARBON: Paul, are those very, very conservative calculations or are they somewhat realistic?

MR. SHEWMON: Well, you can ask the staff that. After Rancho Seco, they are credible at least for B&W plants. As I understand it, they have after Rancho Seco put some sort of an approximation on it, that kind of pressure excursion and with repressurization, then it would crack and the crack could in that environment radiate and go far enough to do some sort of serious damage. So you would not know what would happen when it came to the other side.

MR. BENDER: I do not know that I understand everything I ought to know about this business, but when we say that under the Rancho Seco conditions if we irradiated a couple of more years the vessel would have cracked, that is a pretty strong statement. It seems to me that there is still a lot of materials questions that are being addressed in a very conservative way that influence that answer.

For example, when they establish the radiation dose to the vessel they assume that the dose that it gets on the inside surface, that it goes throughout the vessel, and that is not obvious to me.

MR. SHEWMON: No.

MR. BENDER: No.

MR. SHEWMON: They assume it is the same all the way around. And you might say it is where the core is closest to the vessel. But there is an increase in the critical stress intensity.

MR. BENDER: They have in fact determined that the toughness of the vessel is known for the condition that exists, recognizing a range of influence that exists across the wall thickness --

MR. SHAO: (Inaudible.)

MR. BENDER: I know they calculate that. I guess my question is, do they know the behavior of the material or are we assuming something?

MR. SHEWMON: We will get into that in more detail. Let me just say that, whereas before I think it was felt that this -- you could say that under the extreme transients it was incredible, and now I suspect it is going to have to be more qualifications on making it incredible.

MR. BENDER: I guess my point is, I think the transient -- I do not have to think the transient is credible from the standpoint of, somebody has measured the temperature. That is something that has happened once. You have to say it is credible again unless there is a change in design.

But I am not clear that we know that much about the materials properties. I hope we will hear more about that, because I still think we are using very conservative assumptions concerning the materials properties.

Warren is nodding, either because he understands me or because he agrees, I am not sure which.

MR. SHEWMON: Are there any other -- that covers what I wanted to say by way of introduction. If anybody who was there has other comments or questions?

(No response.)

I have not seen an agenda for what the staff is going to present.

MR. MURLEY: Well, we had not planned to go into the kind of detail -- what I thought I would do is go through the calculations that led us to where we are today, and then summarize what we are doing and the responses we have gotten from the industry groups, and then respond to questions, because I know there are some, like Dr. Bender has. Okay.

By way of background, I will just mention that I think this Committee knows that for years we focused on the large loss of coolant accidents for thermal shock and even through analyses and even some tests at Oak Ridge we concluded that a crack cannot propagate through the vessel in a large loss of coolant accident. There have been, though, concerns about other transients, primarily the steam line break where, due to the overcooling nature of the coolant that rushes out of the steam generator, it cools the primary system down quite rapidly. And that has been a concern; and also with the small loss of coolant accident, where the high-pressure injection can overcool the system.

We have, as a result of the Three Mile Island accident, one of the action plan items in the TMI action plan, II.k.2.13, we have asked the BWR licensees to address that issue and they are doing that. With this background of pressurized thermal shock, we have a program in the budget -- in the research program to look at pressurized thermal shock and run a test.

In fact, there was money included in the fiscal 1980 budget to start such a test at Oak Ridge. In February of 1980, when I was still in the Office of Research, one of my staff, Demetrios Basdekas, came to me and one of his concerns at that time, and still is, was the control system failures and their impact. And he postulated what would happen if there were a failure of the main feedwater control system whereby the main feedwater were to keep on, let's say, after the reactor tripped. Then you would cool down the primary system and cool down the vessel quite substantially.

So we talked about it, my staff and I, and at that time I decided to -- let's do a detailed analysis of the thermal hydraulics and fracture mechanics. I thought at the time that we had the capability and the codes to do that. So we asked Brookhaven to do a calculation for the Oconee plant.

This was meant to be a bounding calculation and there were deliberate conservatisms in that calculation, on the basis that if we could show the vessel was not threatened by such a bounding calculation we would have some degree of confidence that we had time to work on the problem.

So the transient used by Brookhaven was a turbine trip where the feedwater flow was kept in, in fact kept on longer than even realistic for the Oconee plant. This led to a 450-degree Fahrenheit cooldown in 20 minutes, with a subsequent repressurization. That is, the high pressure injection system came on and it was allowed to repressurize the vessel.

These thermal conditions and metal temperatures were then given to Oak Ridge, where our experts are on fracture mechanics. They then carried out a fracture mechanics calculation. Again, this was meant to be a bounding calculation. They used end of life vessel properties for that calculation. Keep in mind, this was done about a year ago now, 1980.

The analyses showed that a vessel failure was likely under those conditions. And this, I have high confidence in the capability of the Oak Ridge group to be able to predict vessel failure. They have done this on many occasions. They failed many vessels and I think they know where the conservatisms are, and if their analyses show that vessel failure is likely then I tend to believe that.

So we asked ourselves, well, we had better -- we were not successful in showing that a bounding calculation showed no problem, therefore we had to look at more realistic calculations. So the staff then took the actual pressure and temperature conditions for the Rancho Seco transient of March 20, 1978. This was a 310 degree Fahrenheit cooldown in 60 minutes, with subsequent repressurization.

If you recall, in this transient there was a failure of the non-nuclear instrumentation system which led to this behavior. It also caused the operator to be blinded in the control room because some of his instruments were out.

With those actual pressure and temperature conditions, then, Oak Ridge did a calculation, which at the time they reported that vessel failure was likely after about ten effective full power years. That was based on critical crack depth curves generated at 10^{19} neutrons per square centimeter, 10^{19} fluents, which was assigned approximately a value of ten effective full power years.

I think they would modify that now to be perhaps 12 effective full power years for B&W plants. I do not think that would change the nature of the conclusions substantially.

MR. SHEWMON: That was probably for the highest copper welds.

MR. MURLEY: No, they used the Oconee vessel.

MR. SHAO: .3 percent copper content. That is very high.

MR. MURLEY: Excuse me, I think I misspoke. That was for the actual Rancho Seco weld content, was it not?

MR. SHEWMON: The report I saw did high and low calculations.

MR. MURLEY: My understanding of this from talking with Jack Strosnyder -- perhaps Warren or someone can help me -- is one finds a wide variation in copper content even in the same weld, and even around the vessel. So that there is not a unique --

MR. SHAO: Rancho Seco is about .35.

MR. HURLEY: Okay, Rancho Seco is about .35. So that is high copper content.

And these results came -- were found about February of this year. We reported it to Harold Denton and suggested that we might want to talk with the industry and find out their response of it. So he called a meeting of the BWR regulatory response group on March 31 of this year. We had a meeting with them, told them of our concerns, and asked them, do you have the same concerns and what is your analysis of the problem.

And we had a progress briefing on April 21 with them. And about a couple of weeks ago we had submittals from each owners group. And I will talk about their findings in just a moment.

We talked with the Subcommittee about the overcooling transients. In summary, I think we can say that the severe overcooling transients have occurred. Rancho Seco is an example. And they seem to be more likely in B&W plants.

The staff's estimate -- and this is the research staff as well as my own staff -- conclude that the probability for B&W-designed plants that they will experience a severe overcooling transient similar or greater to the magnitude of that of Rancho Seco is about 10^{-3} per reactor year. We believe that the probability for such transients in CE or Westinghouse-designed reactors is lower, perhaps by an order of magnitude, than for B&W-designed reactors.

MR. BENDER: Excuse me, Tom. I did not mean to interrupt you, but I would like to understand the temperature value a little better. Does the 310 degrees apply to the temperature of the vessel, the temperature of the vessel wall, the temperature of the coolant?

MR. MURLEY: The temperature of the coolant and therefore of the vessel, the inside vessel wall.

MR. BENDER: Vessel surface.

MR. MURLEY: Yes.

MR. BENDER: What does that mean to the vessel itself, do we know that?

MR. MURLEY: Yes. It means a fairly steep gradient. The gradient through the wall of course changes with time.

MR. BENDER: We are talking about an hour or so, as I understand it.

MR. MURLEY: Yes, yes. And so that the magnitude of the stresses changes with time. Initially, of course, you have high tensile stresses on the inside of the wall due to the thermal gradient. That moves into the wall with time.

The repressurization, of course, adds a hoop stress and that is enough in the calculation --

MR. EBERSOLE: Do you have any data as to, in a parametric sense as to how much hoop stress you have to get off before you rub out the problem?

MR. MURLEY: I do not think so, no.

MR. EBERSOLE: What is suggested here is you should not repressurize when you are cold.

(Laughter.)

MR. EBERSOLE: And there are various ways of not doing that, you know --

MR. MURLEY: I must say that when I talked with the materials people they say, send out a procedure to the operator, don't let him repressurize. But the same operator has gotten a procedure that tells him that under indications of loss of coolant he is supposed to turn that high-pressure injection on and leave it on.

MR. EBERSOLE: That is right.

MR. MURLEY: So it is not an easy matter. I will get to the operator in a minute.

Okay, so to sum up then --

MR. EBERSOLE: Maybe you need two operators, then, one to follow that procedure and --

(Laughter.)

MR. MURLEY: Yes. Severe overcooling transients are -- have occurred. They have some probability of occurrence, and they seem to be more likely in B&W plants.

There have been control improvements in B&W plants.

MR. MARK: You say they have occurred and you refer to the one at Rancho Seco, which indeed is a proof of such a staterent. Are there other members of this class?

MR. MURLEY: Yes.

MR. MARK: Like roughly how many?

MR. MURLEY: There have been about a dozen.

MR. MARK: Okay.

MR. MURLEY: Primarily in B&W plants. We are searching now through the LER records for Westinghouse and Combustion plant transients, but we are sure they are much less frequent and none at least come to mind that the staff about. Because of the thermal inertia of the steam generators in Westinghouse and Combustion plants, they will be less subject to this.

Now we, as I said, discussed with the owners groups what they thought about it and what their conclusions were. The Combustion owners group sent in their responses as the following. They say that for their plants the steam line break transient is the bounding transient, that there are numerous conservative assumptions in the calculation of that transient, and that the vessel with the longest service life and high residual element weld metal has experienced about five effective full power years as of today.

They claim that all vessels in CE plants will maintain their integrity during this bounding stream line break transient without an operator action for at least ten effective full power years, which is to say they assert that they have a margin of five effective full power years even in their worst vessel.

MR. OKRENT: What is the confidence that they make that statement with, or you made your estimate of the likelihood of an overcooling transient? Is that a best es' ate in their case and in your case, or what?

MR. MURLEY: Okay. You have asked two different questions there. The confidence in our estimate of the frequency of the transient, is that one of your questions?

MR. OKRENT: Yes.

MR. MURLEY: That is based really on historical data, with some credit given for improvements in B&W plant control systems that we have mandated. Now, I personally called the resident inspector at every B&W plant and asked our resident inspector, have there been improvements in the reliability of the non-nuclear instrumentation power supply. And in every case I was assured, yes, there have been some improvements.

And furthermore, there have been improvements in most, if not all of the B&W plants with regard to redundancy of signals coming into the control room. So that we have some confidence, I would say relatively high confidence, that there have been improvements in the control system that will prevent or lessen the probability of the kind of transient at Rancho Seco and at Crystal River; and that furthermore, that even if that were to occur the operator would have better information than they had in those two instances.

So taking credit for those two, that led us to the 10^{-3} estimate. I would not place a real high confidence on this. I do not know just how to quantify that.

MR. OKRENT: Well, I have read the, you know, memoranda, so --

MR. MURLEY: Okay.

MR. OKRENT: I will ask the next question. What confidence do you think the regulatory staff has to have in its estimate of the probability that an overcooling transient is not a problem this year? In other words, I heard you say earlier, you know, you take 10 full power years or 12 full power years or in some cases it may be one or two full power years. But what confidence -- when you make a statement like that, with what confdidence do you have to make it for it to be used as a basis of decisionmaking?

MR. MURLEY: Yes. There are two aspects to that, to that answer. The first part is, if the probability of this transient is about 10^{-3} for a B&W plant and there are about ten plants out there, chances are we are not going to see, over the next year, we are not going to see a severe transient like Rancho Seco or worse. So that is one. And even less likely for Westinghouse and CE plants.

The other aspect of our confidence is that even if such a transient were to occur, the staff does not believe that the vessel would fail, even the worst vessel.

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MR. OKRENT: That is a zero probability of failure, given the worst case.

MR. MURLEY: Not zero probability. I do not -- we are in a new area here. We do not have a probabilistic model for vessel failure under these conditions.

MR. OKRENT: The reason I am pressing you is the fact that this is not a trivial issue. The accident that would follow a failure of the vessel could be one of the larger-release scenarios, as you well know.

MR. MURLEY: It would be dramatic, yes.

MR. OKRENT: So, presumably, if in fact -- I have to assume that you want to achieve some probability of this scenario, which is like what is in WASH-1400, where it is the same kind of scenario. And in WASH-1400 it was, you know, a few times 10-6. Of course, that was not a -- the upper -- that was their median value.

But in other words, what I am getting at is you do not look for the same number as for, I suppose, what you might think of as core melt, because you do not know with very high confidence that your containment is going to be useful.

MR. MURLEY: It is these kinds of considerations that lead us to the conclusion that something will have to be done. But we think that we can look at it more carefully over the next year.

MR. SHEWMON: One other question, Bill Kerr: What sort of failure do we get? Is this the vessel falling in little pieces on the floor, or does a crack develop out of which water will leak, or does anybody know?

MR. SHEWMON: That is a good question and one that is currently, or up to now, has been unthinkable and, I think, still is. But my feeling is we are now beginning to look at it.

If you wanted to be somewhat optimistic, you could say, "Well, you have got a weld which opens up and runs until it bumps into tougher metal." That could be out of the core, it can be into a different course. It can be into a different part of the weld which has less copper and has not embrittled as much. It is not falling in little pieces, but there is a fair-sized hole in the side of it.

MR. MURLEY: I asked my staff the same question and essentially got the same answer: that it could range from a relative -- if it cracks at a higher temperature, let us say, it could be a relatively benign opening that relieves pressure, and one might even concede that you could maintain water in the vessel. If the crack were to occur at very low temperatures where it is very brittle, one can imagine missiles being generated and thrown about.

So there is a wide range. We just do not know. It depends on the scenario.

MR. BENDER: The question -- the postulate was the worst condition was a full steam line break.

MR. MURLEY: Yes.

MR. BENDER: Is there a B&W postulate?

MR. MURLEY: Yes, I will get to that.

MR. BENDER: Okay, I will wait then.

MR. EBERSOLE: Did the Combustion postulate assume main feedwater run-on?

MR. MURLEY: I do not know. It is a Chapter 15 steam line break.

MR. EBERSOLE: That is with main feedwater run-on.

MR. MURLEY: I do not know what the assumptions are.

MR. EBERSOLE: A main feedwater run-on makes it worse. It will make it worse.

MR. MURLEY: Yes. Westinghouse --

MR. OKRENT: They did not get a confidence estimate, did they?

MR. MURLEY: No. As a matter of fact, we understand that these are relatively old calculations, Westinghouse, it is about a 1975 analysis. And they have not -- CE. And so they are still relying on that. In fact, they showed a much greater mileage. They showed about 20 effective full-power year back then. And as a result of the Maine Yankee recent inspection where they found that their fluence calculations may have been off by a factor of 2, they said even if it is off by a factor of 2 we have still got ten effective full-power years.

So I am taking that into consideration when I mention it. We will be looking at every one of these vessels over the next year or so. I will give you the story with Westinghouse and then Babcock & Wilcox.

Westinghouse owners group says that a specific vessel may be limited in lifetime by one of three transients. One is a large LOCA. It could be limited by a small LOCA or a large steam line break; it depends on the plant, whether it is two-loop, three-loop, four-loop. Of course, they have either a Combustion vessel or a B&W vessel or, I think, perhaps they even have some Rotterdam vessels.

They assert that all of the vessels in the Westinghouse-designed plants can sustain the Rancho Seco transient for a minimum of at least three more effective full-power years. They cautioned they do not expect that they can get a Rancho Seco-type transient because their system is not as susceptible to that. But nevertheless, they did the calculation giving the pressure temperature input for the Rancho Seco transient, and even for that transient they say they have at least three more effective full-power years.

They also say that all domestic Westinghouse vessels can sustain the most limiting severe thermal shock at least through 1982. Now, as near as I have been able to understand, we have to get back to the owenrs groups. They apparently do depend on operator action to limit repressurization during this most limiting transient.

Now, with B&W, their bounding case is --

MR. BENDER: I missed the transient again. What is it?

MR. MURLEY: For Westinghouse, it depends on the type of Westinghouse design. But it could be either a large LOCA; that would be the worst transient. A small LOCA with high pressure -- cold ECC water; that would be the most limiting transient. Or a large steam line break.

MR. EBERSOLE: A large steam line break, though, without repressurization. You just said they depend on operator action.

MR. MURLEY: It is not clear in the report to us whether they depend on operator action. My reading of it is they do depend on operator action.

MR. BENDER: We need to get it down.

MR. MURLEY: Now, with B&W, it is quite clear that they depend on operator action to prevent repressurization. Let me read their bounding case.

They assumed a small loss-of-coolant accident where natural circulation stops at ten minutes into the transient. The downcomer fluid temperature ramps down to 90 degrees Fahrenheit in 60 seconds. So this is a cooldown rate of 406 degrees Fahrenheit per minute.

They assume no repressurization. The operator throttles the high-pressure injection to maintain the core outlet less than 100 degrees Fahrenheit subcooled.

They have also looked at the large steam line rupture, but it is less severe for a B&W plant because of the smaller inventory in their steam generator than the Westinghouse or Combustion.

MR. EBERSOLE: What about main feedwater run-on? That makes it compounded worse. They are flooding a dry boiler.

MR. HURLEY: They claim that the bounding case that I gave you, that the small-LOCA case that I gave you, is a bounding transient.

MR. EBERSOLE: That is because they did not look at main feedwater run-on. Main feedwater run-on into a normally dry boiler will really do it in.

MR. HURLEY: As I said, we did that calculation also as a bounding calculation at Oconee, but we used end-of-life vessel properties. So we do not have enough information to back up what you say, but we will certainly look into it.

MR. SHEWMON: Jesse, you are talking now about a

CE or Westinghouse?

MR. EBERSOLE: B&W.

MR. SHEWMON: Can you not get water or the pumps continuing to run in the Westinghouse or CE plants?

MR. EBERSOLE: Yes. It does not matter as much for them as it does for a boiler, because it is normally much less full than the others. The others are completely full, and they evaporate at the top. This is just normally just a cylindrical boiler that evaporates on the way up, and when you have continued run-on it becomes a tremendous chilling agent.

MR. MURLEY: I will amplify that. I think the B&W steam generators are normally only about one-third filled, I think. And if you have, let us say, a turbine trip and the main feedwater pumps keep running, then you have two effects: You can go from a heat transfer -- you triple the heat transfer area in the steam generator. Also, I understand, the feedwater preheating is lost, so the temperature of the feedwater goes down.

Those two effects can severely overcool the primary system.

MR. EBERSOLE: Right.

MR. MURLEY: Okay. So that is then what we have received from the owners groups. We have subsequently gotten in material from each one of our licensees, PWR licensees. They generally just support the owners group response.

We have not gotten a lot of additional detail from the licensees themselves, although there are some cases like, for example, Davis Besse says that they have no longitudinal valves in their reactor pressure vessel. That is just one example of the variations that we have to look at. We will be looking at every vessel by itself.

Okay. The NRR staff would not expect the vessel with the worst material properties to fail today even in the event of an overcooling transient as severe as the Rancho Seco event.

MR. OKRENT: You said that several times, but you have not told us whether you have looked at, you know, what it takes for somewhat more severe overcooling as Jesse Ebersole has just identified, like the Rancho Seco one that has occurred. It is not impossible, though it may be improbable. Rancho Seco, I do not think, was a limiting overcooling event. Am I correct?

MR. MURLEY: I believe that is correct. It is the worst we have seen. But one can postulate worse transients, that is what I am saying.

MR. OKRENT: I do not know what it means when you make a statement -- I mean I know what it means when you are saying something in terms of the Rancho Seco event, but I do not know how to translate this to regulatory positions if I cannot say, "Well, but could there be something, could it be an hour faster, whatever? Is that tolerable, intolerable?"

MR. BENDER: We need to know more than just whether it is tolerable, intolerable, likely, unlikely. If we are going to deal with this thing probabilistically, then we ought to try to put some probabilities on the combinations of events that have to be dealt with.

MR. MURLEY: Okay. I guess I will have to --

MR. BENDER: I doubt you can answer that.

MR. MURLEY: But the research staff is developing a probabilistic model similar to the Octavia model you recall that they developed several years ago to deal with the low-temperature repressurization problem. This one is much more complex. One has to deal with variations of material properties, variations in fluent through the thickness and probabilities like that as variations around the vessel.

It is quite complex, yes. But they will do that. And I think such a model is absolutely necessary if we are ever going to understand margins. And we hope to have that over the next several months, at least preliminary versions.

MR. EBERSOLE: A comment. They are supposed to be safety-grade protection, instantaneous run-on of the water pumps in order to preotect the containment sump. But if the hypothetical break is outside of containment, I do not think those things work. In other words, there is a safety-grade system for cutting off feedwater flow, but I think it works only when it discharges into the containment.

MR. SHEWMON: Yes. I can promise you will hear about this problem again.

MR. EBERSOLE: Okay.

(Laughter.)

MR. MURLEY: Okay. During this next year the staff will be looking at each PWR pressure vessel in the sense that we have asked each of the licensees to submit reports dealing with the capability of their pressure vessel to withstand thermal shock. We will have the licensees continue to analyze the problem. We will try to get an assessment of the margins for each vessel. And we will be looking at potential improvements because I think we can say with some confidence that relying on operator action is not going to be a long-term solution.

That is not a firm regulatory position. I guess you can say that is my view. But also I think it is shared by Steve Hanauer and quite a few others. Relying on the operator not to repressurize the vessel is not a long-term solution.

Some of the fixes that we will be looking at are --

MR. KERR: Is it a short-term solution?

MR. MURLEY: Well, in the sense that -- first, let me say my understanding is that the B&W has notified their utilities, their own B&W plants, their customers, that they should throttle back on HPI on indication of 100 degrees subcooling.

Now, one has to ask what is the probability that the operator will make a mistake and not do that? There is some number; I do not know what it is. But it is clearly too high to rely on for a number of plants over their lifetime.

MR. JOHNSON: You asked whether it is long-term or short-term.

MR. KERR: I asked if it were a short-term solution.

MR. JOHNSON: Okay. In a sense, it is a good short-term solution when you consider the irradiation effect. The status of the most highly irradiated vessels today and you consider the gradient of toughness -- let me try this on you.

MR. KERR: You are giving me a lot of information which I do not know what to do with.

MR. JOHNSON: I will tell you in a second. The point being that the thermal shock will, for today's vessels, not propagate a crack clean through the wall. Therefore, after the thermal shock, you still have a vessel which will hold water, and if the operator will throttle back you can keep the core covered and the core cooled.

MR. KERR: My question really had to do with whether you were going to put the operator in jail if he did not follow his instructions, which are to continue to let the high-pressure injection system run. That was the only question I had.

MR. MURLEY: He answered a different question, I think. He said that if the operator does not repressurize, then you are okay. The question was, I think, can you rely on him?

MR. KERR: Under the present rules, is he permitted not to repressurize?

MR. MURLEY: Yes.

MR. KERR: I asked this same question last month, and the answer was: Under present rules, he must repressurize.

MR. MURLEY: He must maintain 50 degrees subcooling.

MR. KERR: And if he has to repressurize to do it, he repressurizes?

MR. MURLEY: Yes. But not --

MR. KERR: So he has to be fairly skillful.

MR. MURLEY: My understanding is that he --

MR. KERR: It may not be bad, because it means he has to be really on his toes. You cannot do it with one hand.

MR. MURLEY: If he understands exactly the transient he is dealing with, yes. That is the key. I think you are talking about a --

MR. BENDER: We are needling you excessively here. Let us let Tom get on with it.

MR. KERR: I am simply trying to find out what the present status of things is, and I think I have discovered that the opertor has some fairly difficult choices to make.

MR. EBERSOLE: In the case where we have (inaudible).

MR. MURLEY: No, the Rancho Seco vessel was not inspected, but they asked the licensee to do a careful analysis before they were allowed to start up. And, of course, it was quite a fresh vessel in 1978, and the staff is fairly confident that there was no damage.

MR. OKRENT: At least there had not been any through-cracks showing a leak yet.

(Laughter.)

MR. MURLEY: There are four, at least four, items that we will be looking at over the next year for ways to improve this, because, as I say, I think no one that I know of is claiming that these vessels can uniformly go 40 years without some improvements. Perhaps some vessels can, I do not know. But I do not think anyone is claiming that they all can. And as I said, Westinghouse says they are only confident through, at this time, 1982.

Okay, so we are looking at the possibility of removing fuel from the outer rows of the core. This can cut down the fluents to the vessel by a factor of perhaps 2 to 3. That means quite a bit in the lifetime.

We are looking at, or will be looking at, perhaps, control systems or protection systems that could prevent repressurization.

Another option that has been talked about is raising the temperature of the water in the borated water storage tank so that the thermal shock is not quite as great.

Also, there is a program at EPRI on in-place annealing if all other things fail, annealing at somewhere around 650 or 750 degrees Fahrenheit to gain back the toughness of the vessel.

None of these are sufficiently attractive or are either attractive or will solve the problem that it is a clear-cut decision what should be done. The research program will be continuing over the next year to do independent analyses and will attempt to assess the margins, as I said, develop a probabilistic model that will help us at least assess the margins for vessel failure.

Also, they plan on conducting a thermal shock test, pressurized thermal shock test, at Oak Ridge next year which will allow us to test our models under such conditions.

So that sums up what I had to say.

MR. BENDER: One other variant that I did not hear. It may be impossible, but I will ask it anyhow. My interpretation of what I have been told, it seems to me it is conceivable to say we will sleeve the vessel or isolate the vessel in that particular location by putting a metal membrane over that surface, just so it does not see the temperatures. Is that out of the question?

MR. SHAO: (Inaudible.) I think this is a good point. You can put some local shielding, just as Mike said. Maybe there are two things you can do: either isolate against thermal shock; or put on something like shielding, local shielding, to reduce the fluents.

MR. BENDER: Anything to divert the cold water away from it would probably solve the problem.

MR. SHAO: The core, like to receive the cold water, the reactor vessel could not take cold water. So essentially, you want this specific area to isolate.

MR. MURLEY: As a matter of fact, we have -- not now, but we had -- paper going through the mill which will probably come down to you in about a month, which will then go to the Commission, recommending it be an unresolved safety issue. So, to me, that is a logical way to deal with it.

MR. SHEWMON: A different question: The finns in the Westinghouse, so-called, had a problem something like this, and they took out 10 percent of the fuel, especially that closest to the pressure vessel, and thought that they had decreased the rate of radiation and transition temperature change. Have you heard of any discussions of that?

MR. HURLEY: The report that I have is confidential, and I do not know how much is public knowledge and how much is not. If you say it, why -- yes, they use that method for cutting down the fluents, they say, by a factor of 3 in the vessel. They also heated up their high-pressure injection water.

MR. SHEWMON: I was just telling Professor Plesset, on my left, that we can then increase the power out of the remaining fuel and transform it into an ECCS question.

MR. HURLEY: Right.

(Laughter.)

MR. SHEWMON: Okay, are there any other questions?

MR. OKRENT: You said we will hear about it again. I guess I am interested in knowing how the staff is going to decide how it sets its priorities for this problem and the time scale for resolution and the rationale for resolution.

And I would like not to wait half a year to hear them. And while I think, in fact, you probably will have it as an unresolved safety issue, I do not think that will take a year.

MR. MURLEY: I will reiterate that there are two bases, there are two parts of the bases, for our actions and the timing of our actions. One is our assessment of the probability that a transient as severe as Rancho Seco or worse is about 10-3 for a B&W plant but substantially less for Westinghouse and Combustion plants. That, coupled with the fact that even if a Rancho Seco event were to occur we would not expect the worst type of -- worst vessel to fail today, it gives us confidence that we can deal with this problem.

We have at least a year to deal with this problem. I believe personally there are more conservatisms in the calculation, and we are going to try to quantify those and find out better what the margins are over the next six months, let us say. If it turns out that the margins are not as great as we thought, then we will maybe have to take some actions.

MR. OKRENT: Well, actually what you gave was part of a rationale. I think if you went back and looked at the transcript, you would not accept it from a licensee as an adequate basis.

(Laughter.)

Let me leave it at that. I am not saying that I know what time there is to resolve this. But what I am saying, it warrants some thought early on as to just what you know and what your conservatisms are and on what basis you are making a judgment about the time as best you can now. And it is not clear to me necessarily, based on what you said today, that you know you have a year in terms of some particular safety goal. I think that safety goal has to be pretty small, you know, the probability to --

MR. SHEWMON: Jesse, did you have a question?

MR. EBERSOLE: Would it help very much to trip the main coolant pumps?

MR. MURLEY: Oh, I do not think we have investigated that in detail, Jesse. But at first blush, it makes it -- first, because with the pumps running you at least get some mixing.

MR. EBERSOLE: That is what you do not want.

MR. MURLEY: No. Thermal mixing of outlet --

MR. EBERSOLE: I was thinking of keeping the cold water from creeping in.

MR. MURLEY: It depends. It depends on the transient. If it is a loss-of-coolant accident where you are injecting cold water, you would like to mix it. If it is a steam line break -- yes?

MR. BENDER: I think I have never understood the way in which cooling is treated in these analyses. Was there a vessel temperature measurement at Rancho Seco?

MR. MURLEY: No, I think they used the cold-leg temperature.

MR. BENDER: So they had to do some analysis based on how the water was channeling down around the vessel.

MR. MURLEY: Yes. But, you see, this was an hour-long event, and we are talking about bulk coolant temperatures. So I think the question of mixing and all that stuff is not --

MR. BENDER: It has to do with how fast the vessel is being cooled. And I think that has to do with turbulence of the water at the surface of the vessel and things like that that I do think are heat-transfer phenomena. They have to be understood better than we understand them now.

MR. MURLEY: If we are talking about an injection of ECC water, I would agree. Here we are talking about bulk primary coolant temperatures, and I do not think that you are going to see variations like you would in a loss-of-coolant accident.

MR. BENDER: I know the heat-transfer rate is very high, but the heat is being drawn out of the vessel at some rate.

MR. SHEWMON: Drawn out by the vessel, you mean.

MR. BENDER: In this case, being drawn out of the vessel.

MR. MURLEY: Okay.

MR. BENDER: And the rate at which that temperature is changing is a function of the heat transfer at the surface.

MR. MURLEY: Yes.

MR. BENDER: And it is not just bulk temperature; it is the turbulence, all of those things that go with it.

MR. SHAO: It is the regular heat transfer --

MR. BENDER: I do not know if it is or not.

MR. SHAO: (Inaudible.)

MR. SHEWMON: Fine.

You had a question, Milt?

MR. PLESSET: I was going to ask, Tom, is there some thought being given to ways of getting the operator information so that he can fairly quickly tell the difference between a small break on the primary side and a cooldown from the transient on the secondary side? That is pretty important. Is there a way for him to get this information fairly quickly? Any thought to how this might be made available to him? Because otherwise, initially, he does not know.

MR. MURLEY: I cannot give you a definitive answer. The information is available to him right now in the control room, clearly, to dope out what is happening. The trouble is you have too much information.

MR. PLESSET: That is right; too much to make a decision fairly quickly.

MR. MURLEY: Is your question, "Is there a computer or a logic that analyzes it for him?" Then I think not.

MR. PLESSET: "A small break on the primary side," or "This is a secondary-side transient which is dropping the pressure in the mary loop"?

MR. MURLEY: No. As I said, I think the information is available if he analyzes it correctly. But there are many things going on, many alarms going on, and that is what has the staff that I talked with concerned with relying on the operator to dope it out properly.

MR. PLESSET: And to do it fairly quickly.

MR. MURLEY: Yes.

MR. PLESSET: It seems to me you need something like that; don't you agree?

MR. MURLEY: It could be. One of the things, as I said, we would be looking at is a protection system that could analyze this, and one could take a combination of measured temperatures and pressures and prevent repressurization, allow the operator to maintain the high-pressure injection system on to get 50 degrees subcooling but not take it, say, beyond 100 degrees subcooling or something like that.

That could be built into the protection system. It is not a favorite of mine, because I do not like the idea. I do not think it is a good idea generally to go in and monkey with those.

MR. SHAO: (Inaudible.) The repressurization essentially is to drive the crack all the way through the wall. But thermal shock mainly damage vessel very badly.

MR. SHEWMON: Any other questions?

(No response.)

MR. SHEWMON: Fine. Thank you very much.

(Whereupon, at 5:30 p.m., the committee was adjourned.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

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In the matter of: ACRS/254th General Meeting

Date of Proceeding: June 5, 1981

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)

OFFICIAL INVESTIGATIONS OF REACTOR OPERATIONS, POST-TMI

Summary of Events

• March 28, 1979	Accident at Three Mile Island, Unit 2
• July, 1979	Issuance of Rogovin Report
• October 24, 1979	NFPOT Committee Established by DOE Under Secretary
• October 30, 1979	Issuance of Kemeny Commission Report
• November 6, 1979	NFPOT Committee Scope Expanded to all Elements of Kemeny Commission Report
• February 1, 1980	Committee Plan of Action Approved
• March 10, 1981	Committee Report Submitted to DOE Under Secretary
• March 12, 1981	Under Secretary Directs Creation of Action Plan to Respond to Report Findings
• May 14, 1981	Action Plan Submitted to Under Secretary
• May 20, 1981	Action Plan Responding to Committee Report Approved by DOE Under Secretary

NUCLEAR FACILITIES PERSONNEL QUALIFICATION AND TRAINING COMMITTEE

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* RESIGNED DOE FEBRUARY 1981.

TOPICS ADDRESSED BY KEMENY COMMISSION	EQUIVALENT HEADINGS IN NFPQT REPORT
The Nuclear Regulatory Commission	DOE Safety Overview Function
The Utility and Its Suppliers	DOE Programmatic Function
Training of Operating Personnel	Training of Operating Personnel
Technical Assessment	Technical Assessment
Worker & Public Health & Safety	Worker & Public Health & Safety
Emergency Planning and Response	Emergency Planning and Response
Public's Right to Information	Public's Right to Information

Scope of Reactor Assessment

- 84 DOE-OWNED REACTORS:
  - 67 OPERABLE
  - 17 STANDBY
- OF THE 67 OPERABLE REACTORS:
  - 8 NAVAL PROTOTYPE REACTORS
  - 22 TEST REACTORS

    E.G., FFTF (400 MWT), EBR-2 (62.5 MWT), AFSR (.001 MWT)
  - PRODUCTION REACTORS

    NPR (3800 MWT) AT RICHLAND, WA

    C, P, K (2200 MWT) AT SAVANNAH RIVER
  - REMAINING ARE SMALL CRITICAL RESEARCH FACILITIES OR TRANSIENT TEST REACTORS OF NEGLIGIBLE POWER.

13 REPRESENTATIVE REACTORS ASSESSED

<u>SELECTION CRITERIA</u>

1. POWER LEVEL.
2. POTENTIAL OFF-SITE RISK.
3. PROGRAM PARTICIPATION (MAXIMIZE NUMBER OF ORGANIZATIONS ASSESSED).

13 Representative Reactors Assessed

Name		Power (MWt)	Primary Pressure (PSIG)	Average Primary Temperature (°F)	Location
1	NPR	3800	1730	475	Richland, WA
2	SRP-K	2200	5	154	Savannah R., SC
3	FFTF	400	Atmospheric	940	HEDL, WA
4	ATR	250	355	146	INEL, ID
5	ETR	175	200	124	INEL, ID
6	HFIR	100	650	140	ORNL, TN
7	EBR-II	62.5	Atmospheric	900	INEL, ID
8	LOFT	55	2250	576	INEL, ID
9	HFBR	40	250	140	BNL, NY
10	ORR	30	38	126	ORNL, TN
11	PBF	28	Atmospheric	200	INEL, ID
12	OWR	8	12.5	112	LASL, NM
13	BSR	2	Atmospheric	107	ORNL, TN

For Comparison

TMI-2	2772	2185	582	TMI, PA

On-Site Reactor Assessments*

1. K
2. ATR
3. HFIR
4. HFBR

- Extensive preparation preceded field reviews.
- Field Review Team - 9 to 12 persons; headed by DOE Committee member; included qualified consultants.

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* Basis for assessing nine reactors not visited - document reviews, personnel interviews.

NEPOT COMMITTEE MAJOR FINDINGS

1. "NO EVIDENCE WAS FOUND THAT ANY OF THE DOE-OWNED REACTORS ARE BEING OPERATED IN AN UNSAFE MANNER OR THAT ANY OF THESE REACTORS SHOULD BE SHUT DOWN."

2. "A NUMBER OF SIGNIFICANT DEFICIENCIES EXIST IN DOE'S REACTOR SAFETY MANAGEMENT ACTIVITIES, AS REVEALED BY THE COMMITTEE'S ON-SITE REVIEWS AND BY ASSESSMENTS OF SITE AND HEADQUARTERS DOCUMENTS."

3. "THERE IS A NEED TO STRENGTHEN SUBSTANTIALLY THE TECHNICAL AND MANAGERIAL CAPABILITIES OF DOE HEADQUARTERS AND FIELD ORGANIZATIONS WHICH HAVE REACTOR SAFETY RESPONSIBILITIES."

4. "MANY OF THE 'TMI LESSONS LEARNED' HAVE NOT BEEN ADEQUATELY ADDRESSED OR APPLIED IN DOE REACTOR PROGRAMS."

DOE RESPONSE TO NFPQT COMMITTEE REPORT

- MARCH 10, 1981 - SUBMISSION OF REPORT ON A SAFETY ASSESSMENT OF DOE NUCLEAR REACTORS TO THE UNDER SECRETARY (US)
- MARCH 12, 1981 - US DIRECTS CREATION OF ACTION PLAN TO RESPOND TO REPORT FINDINGS
  - ACTING ASSISTANT SECRETARY FOR DEFENSE PROGRAMS
  - ACTING ASSISTANT SECRETARY FOR NUCLEAR ENERGY
  - ACTING ASSISTANT SECRETARY FOR ENVIRONMENTAL PROTECTION, SAFETY, AND EMERGENCY PREPAREDNESS
  - ACTING DIRECTOR OF OFFICE OF ENERGY RESEARCH
  - DEPUTY DIRECTOR ADMINISTRATION
- MAY 14, 1981 - ACTION PLAN SUBMITTED TO US
- MAY 20, 1981 - US APPROVED ACTION PLAN
  - APPOINTS C. A. HEATH TO HEAD IMPLEMENTATION TASK FORCE FOR HEADQUARTERS ACTION
- MAY 21, 1981 - US DIRECTS FIELD OFFICES TO COMPLETE ITMES IN ACTION PLAN ASSIGNED TO FIELD OFFICES

ORGANIZATIONAL RECOMMENDATIONS

- ENSURE CONTINUOUS ATTENTION AT LEVEL ABOVE ASSISTANT SECRETARIES.
- ESTABLISH INDEPENDENT SAFETY OVERVIEW GROUP REPORTING TO UNDER SECRETARY.
- ESTABLISH GROUP OF EXPERTS EXTERNAL TO DOE TO ADVISE SECRETARY.

A. DOE SAFETY OVERVIEW FUNCTION - NFPOT FINDINGS

A-1: DOE TOP MANAGEMENT INVOLVEMENT.

A-2: THE INDEPENDENT REACTOR SAFETY OVERVIEW CAPABILITY.

A-3: THE HEADQUARTERS REACTOR SAFETY OVERVIEW ORGANIZATION LEVEL.

A-4: STRENGTH OF TECHNICAL CAPABILITY WITHIN DOE SAFETY ORGANIZATIONS.

A-5: BROAD GUIDELINES VS. SPECIFIC REQUIREMENTS.

A-6: CLARITY OF LINE MANAGEMENT AUTHORITY.

A. DOE SAFETY OVERVIEW FUNCTION - DOE RESPONSE

CLARIFICATION OF LINE OF RESPONSIBILITY FOR SAFETY

- FIELD OFFICE MANAGERS ARE RESPONSIBLE TO PROGRAM SECRETARIAL OFFICERS FOR SAFE CONDUCT OF THEIR PROGRAMS.

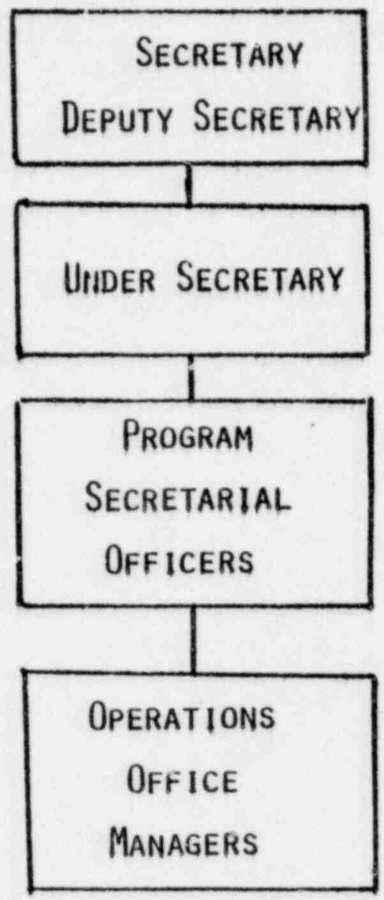
STRENGTHEN AND CLARIFY NUCLEAR SAFETY OVERVIEW RESPONSIBILITIES AND FUNCTIONS

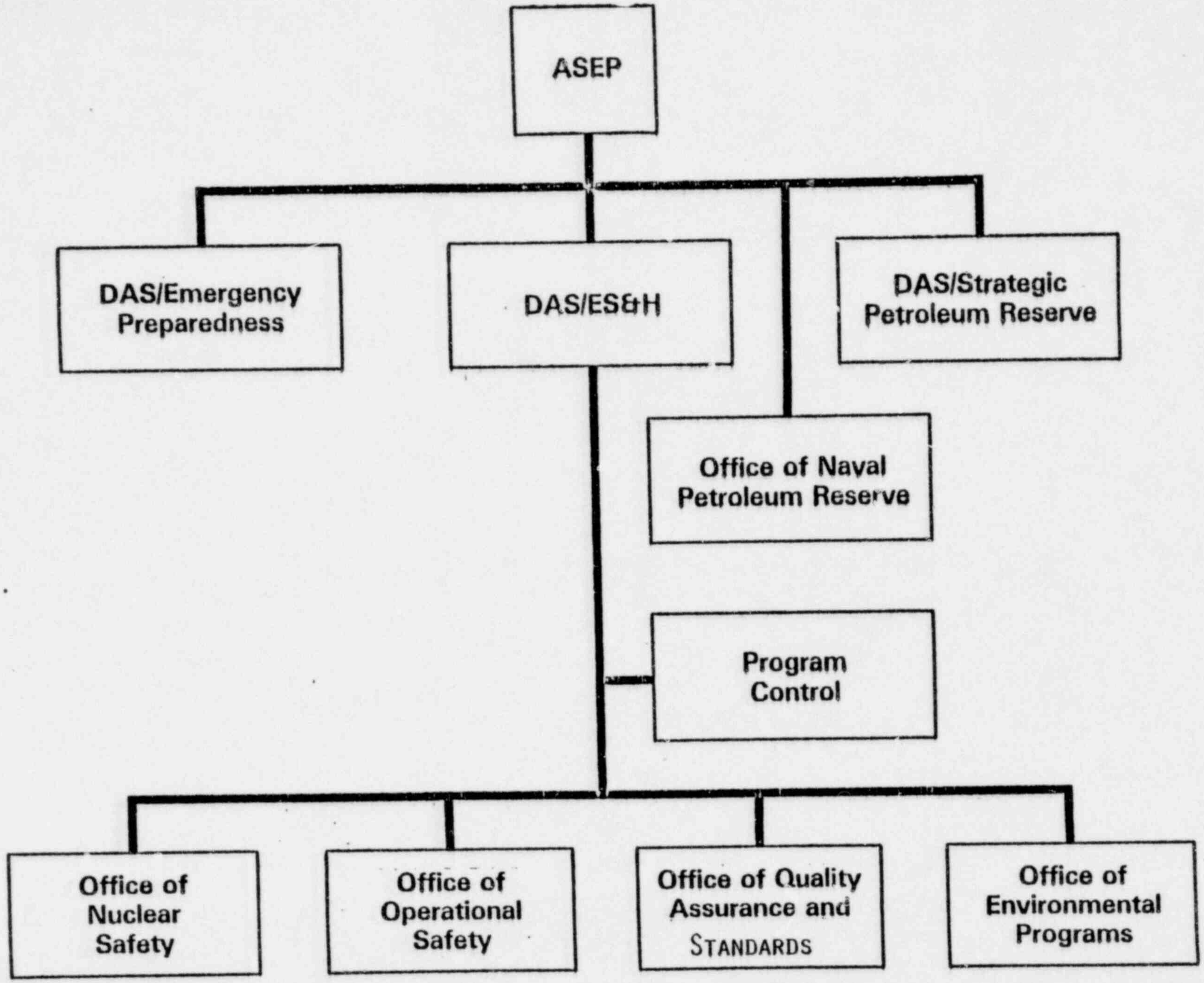
- ASEP HAS DEPARTMENT-WIDE OVERVIEW; LEAD RESPONSIBILITY FOR POLICY AND DIRECTIVES, CONDUCTING ASSESSMENTS, AND ASSEMBLING AND DISSEMINATING INFORMATION.

- ESTABLISH AN OFFICE OF NUCLEAR SAFETY (ONS) REPORTING TO THE DAS FOR ES&H.

- ONGOING ACTIVITY TO REVIEW AND UPGRADE SAFETY DIRECTIVES, STANDARDS AND GUIDES.

LINE OF DOE RESPONSIBILITY OR NUCLEAR SAFETY





ASEP
DAS/Emergency Preparedness
DAS/ES&H
DAS/Strategic Petroleum Reserve
Office of Naval Petroleum Reserve
Program Control
Office of Nuclear Safety
Office of Operational Safety
Office of Quality Assurance and Standards
Office of Environmental Programs

B. DOE PROGRAM FUNCTION - NFPQT FINDINGS

B-1: Uniform Quality Assurance (QA) guidance.

B-2: Application of QA policy and instructions to specific sites.

B-3: QA program scope.

B-4: Organizational independence for QA personnel.

B-5: Documentation of DOE QA programs.

B-6: Independent verification in operating procedures.

B-7: Adequacy of operating procedures.

B-8: Adequacy of shift relief procedures.

B-9: Adequacy of control room procedures.

B-10: Adequacy and use of incident reporting.

B-11: Application of the TMI lessons learned to operating procedures.

B-12: Common nuclear safety terminology.

B. DOE PROGRAM FUNCTION - DOE RESPONSE

STRENGTHEN QUALITY ASSURANCE (QA) FUNCTIONS

- MOVE RESPONSIBILITY FOR COORDINATION FROM US TO EP.
- ESTABLISH AN OFFICE OF QA AND STANDARDS UNDER DAS FOR ES&H.
- QA STANDARDS AND GUIDELINES.
- ONGOING ACTIVITY TO PROVIDE UNIFORM QA.

REVIEW OF CONTRACTORS OPERATING PROCEDURES FOR CLARITY AND COMPLETENESS AND TAKE CORRECTIVE ACTION

- OPERATIONS OFFICES TO REPORT TO PROGRAM SECRETARIAL OFFICERS AND ASEP.
- OPERATIONS OFFICES TO COMPLETE ASSESSMENT OF REACTOR OPERATING PROCEDURES.

ASEP TO ESTABLISH INFORMATION EXCHANGE SYSTEM

- ACTIONS UNDERWAY TO ESTABLISH DOE-WIDE SYSTEM TO OBTAIN, ANALYZE AND DISTRIBUTE OCCURRENCE AND INCIDENT REPORTS RELATED TO NUCLEAR SAFETY.

ASEP TO ESTABLISH A DOE SAFETY COUNCIL

- MANAGEMENT OVERVIEW.
- EXCHANGE KEY DATA.

C. TRAINING OF OPERATING PERSONNEL - NFPQT FINDINGS

C-1: Requirements for selection, training, and qualification of operating personnel.

C-2: DOE Headquarters office for operating personnel training.

C-3: Revision of DOE Order on reactor personnel training.

C-4: Comparability of DOE reactor personnel training programs to those for licensed reactors.

C-5: Detailed training reviews and operator performance assessments.

C-6: The knowledge of reactor operators and reactor supervisors.

C-7: Reactor plant casualty drill training.

C-8: Independence of certification process for operating personnel.

C-9 Administration of written and oral examinations.

C. TRAINING OF OPERATING PERSONNEL - DOE RESPONSE

Establish Strong Policy and Program for Training and Qualification of Reactor Operations Personnel

- DOE Order on Safety (5480.1) to be revised.
- Office of Nuclear Safety to have program and overview responsibility.
- Operations Offices to have expertise to appraise contractor performance and compliance.
- Operations Offices to perform short-term assessments.
- Expedited development of DOE standards.

D. TECHNICAL ASSESSMENT - NFPOT FINDINGS

D-1: Post-TMI DOE technical reviews.

D-2: Impact of releases in vented confinement buildings.

D-3: Access to operating stations in event of radioactivity release.

D-4: "Walkaway-type" protection features at many of the DOE reactors.

D-5: Evaluation of aluminum-water reactor compared to zirconium-water reaction.

D-6: Technical evaluation of "human factor" issues.

D. TECHNICAL ASSESSMENT - DOE RESPONSE

<u>DEVELOP AND IMPLEMENT PLAN FOR COMPREHENSIVE REVIEW AND ASSESSMENT OF DOE-OWNED NUCLEAR FACILITIES</u>

- CONDUCT ON-SITE REVIEW OF REMAINING NINE REACTORS LISTED IN NFPQT COMMITTEE REPORT.

- ONGOING DEVELOPMENT OF PLANS FOR ASSESSMENTS OF OTHER DOE-OWNED NUCLEAR FACILITIES.

E. WORKER AND PUBLIC HEALTH AND SAFETY - NFPQT FINDINGS

E-1: DOE Headquarters support of the radiological protection function.

E-2: Control of radiation and radioactivity during normal operations.

E-3: Completeness of appraisals of radiological controls program.

E-4: Implementation of ALARA program.

E-5: Drills for radiological control personnel.

E-6: Contractors' internal audit programs.

E. WORKER AND PUBLIC HEALTH AND SAFETY - DOE RESPONSE

DEVELOPMENT OF MINIMUM REQUIREMENTS FOR HEALTH PHYSICS ASSESSMENTS

- ONGOING REVISION OF APPROPRIATE ORDERS.
- FORMAL ISSUANCE OF DOE ALARA GUIDE AS A REQUIRED STANDARD.
- FIELD OFFICE AUDIT OF CONTRACTOR PROGRAMS.

F. EMERGENCY PLANNING AND RESPONSE - NFFOT FINDINGS

F-1: OVERALL DOE POLICY DIRECTIVES FOR EMERGENCY RESPONSE.

F-2: DOE ORGANIZATIONAL ROLES IN AN EMERGENCY.

F-3: THE ROLE OF OTHER FEDERAL AGENCIES IN REVIEW OF STATE AND LOCAL EMERGENCY PLANS FOR THOSE JURISDICTIONS SURROUNDING DOE FACILITIES.

F-4: IMPROVEMENTS TO EMERGENCY PREPAREDNESS AT DOE REACTOR FACILITIES.

F. EMERGENCY PLANNING AND RESPONSE - DOE RESPONSE

<u>IMPROVE CAPABILITY FOR RESPONDING TO A NUCLEAR ACCIDENT</u>

- ONGOING ACTIVITY TO PREPARE DIRECTIVE ON RESPONDING TO TO A NUCLEAR EMERGENCY AT A DOE SITE.
- CLARIFICATION OF THE RESPECTIVE ROLES OF FEMA AND DOE FOR EMERGENCY RESPONSES AT DOE SITES.
- EXPAND PERIODIC EXERCISES AT DOE SITES TO TEST EMERGENCY RESPONSE PLANS.

G. PUBLIC'S RIGHT TO INFORMATION - NFPGT FINDINGS

G-1: PUBLIC AFFAIRS PLANNING FOR EMERGENCIES BY HEADQUARTERS.

G-2: IMPORTANCE OF PUBLIC AFFAIRS EMERGENCY PLANNING.

G-3: DOE POLICY GUIDANCE TO GOVERN PUBLIC AFFAIRS PROGRAMS.

G-4: THE LESSONS OF THREE MILE ISLAND RELATING TO PUBLIC INFORMATION.

G. PUBLIC'S RIGHT TO INFORMATION - DOE RESPONSE

<u>Improve Public Communications during a Nuclear Accident</u>

- Promulgate policy responsibilities and procedures for public affairs actions during a nuclear emergency.
- Prepare plans for annual meetings with State and local officials.

APPLICATION OF APPROPRIATE DOE RESOURCES TO NUCLEAR SAFETY

Action Plan

Major concern expressed that adequate resources be applied to implement plan.

DOE Response

Review underway by task force.

Requirements being identified by ASEP and by Field Offices.

ASMA and US to assess needs for reprogramming as required.