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NUCLEAR REGULATORY COMMISSION

In the Matter of: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

267TH GENERAL MEETING

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

267TH GENERAL MEETING

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

Friday, July 9, 1982

The Committee convened, pursuant to recess, at 8:30 a.m.

PRESENT FOR THE ACRS:

PAUL G. SHEWMON, Chairman
JEREMIAH J. RAY, Member
J. CARSON MARK, Member
MILTON S. PLESSET, Member
CHESTER P. SIESS, Member
ROBERT C. AXTMANN, Member
MAX W. CARBON, Member
WILLIAM M. MATHIS, Member
DAVID A. WARD, Member
JESSE C. EBERSOLE, Member
DAVID OKRENT, Member

DESIGNATED FEDERAL EMPLOYEE:

RAYMOND FRALEY

ALSO PRESENT:

MR. CHECK
MR. MORRIS
MR. LONGENECKER
MR. PIPER
MR. HULMAN
MR. CLARE
MR. KNIGHT
MR. LEE
MR. ABRAHAM
MR. STRAND
MR. GOODWIN
MR. THOMAS
MR. STARK
MR. RUMBLE
MR. GOESER
MR. KNIEL
MR. MARCHESE
MR. BERRY

P R O C E E D I N G S

(8:30 a.m.)

MR. SHEWMON: Could we begin?

This is the second day of the 267th meeting of the ACRS. During today's meeting the Committee will hear reports and discuss the following: discussion of the NRC report on the site suitability of for the Clinch River Breeder Reactor and ACRS Subcommittee report on the proposed NRC rule on high level waste disposal; discussion of the NRC Task Action Plan 845; evaluation of alternate decay heat removal systems; continued discussion on the proposed Committee report on the fiscal year 1984-85 NRC research budget; and miscellaneous subcommittee reports and activities.

The items scheduled for discussion on Saturday are listed in the schedule for the meeting which is posted on the bulletin board in the back of the room.

The meeting is being conducted in accordance with the provisions of the Federal Advisory Committee and Government in the Sunshine Act. Mr. Paul Boenert on my right is the designated Federal employee for the meeting.

A transcript is being kept and we would appreciate your help in allowing the transcriptress to hear you. We have received no written statements or requests to make oral statements from members of the general public for today's meeting.

The first item has to do with the Clinch River Breeder Reactor, and Dr. Max Carbon with handle that.

MR. CARBON: Thank you, Mr. Chairman.

We have got a rather full agenda, so I will give only a brief report and then open it for questions.

I would like to begin by stating what we are to cover this morning and what we will not from the standpoint that the Staff is drawing some distinctions in here which seem worthwhile.

This is a meeting to review NUREG-0786, which is the site suitability report in the matter of the Clinch River Breeder Reactor plant, and to consider the suitability of the CRBR site for such a plant. You will recall that the Full Committee decided several weeks ago that we wanted to be involved in the consideration of the CRBR site. And further, I am sure that we will hear from the Staff this morning that they do want us to write a letter on this topic.

At this point I would try and get into a distinction here that the Staff is making that is a rather important one for our discussion this morning. The Staff has discussed this with the ASLB and is asking us to review the proposed site as "a suitability location for a reactor of the general size and type as CRBR." And that is the end of the quotation.

The Staff concluded in 1977 and has reconfirmed that it does indeed consider the site suitable for the general size and type as the CRBR. It has no concluded at this time that the CRBR plant as it is currently designed can be located there. In fact, the Staff won't complete its safety evaluation and issue an SER until next March.

I guess in theory that leaves open the question of whether the CRBR could ever be placed there, although in practice the Staff has stated its belief that if the present CRBR design doesn't fit for some reason, that the design could be modified in a reasonable fashion such that the modified plant could be placed there. That is their belief at this time.

Examples there are the present design might have to be beefed up to handle greater seismic loads of higher energetics or some such thing. I am just grabbing something out of the air. But the Staff believes that if modifications are necessary along those lines or any other, they could be done in some fashion.

So the Staff is asking us today to review the site for the CRBR-type plant, and it specifically is asking us not to review the safety aspects of the CRBR as it is currently designed.

Some of the major topics that will come up in today's meeting include the following. One is certainly what is the Staff's understanding of the general hazards of a plant of this size and type, what is the basis for that understanding. The second topic is how does one compare the safety of an LMFBR with that of an LWR. The third topic is what is the basis for selecting the site suitability source term, which is equivalent to that of an LWR, with an addition. The term for an LWR, as you recall, is 100 percent of the noble gases, effectively 25 percent of the halogens and 1 percent of the fission products.

Then for the CRBR they add 1 percent of the transuranics. This is sometimes stated as 1 percent of the plutonium, but it is quite sure it is 1 percent of the transuranics. As part of this question, the basis for selecting the site suitability source term and associated question of what role, if any, does a potential for core disruptive accidents play in the definition of the source term will be taken up.

A fourth topic concerns the assumptions made in estimating the effects of an upstream dam failure. Specifically, what is the basis for assuming a partial failure rather than a total collapse, and what is the significance of it?

A fifth topic has to do with the level of the SSE that is appropriate. The value actually chosen for the CRBR is .25 g acceleration. Dr. Trifunac, who is a consultant on our Subcommittee, has estimated that there is something like one chance in 30 of exceeding that value over a 50-year lifetime. This gives a return period, then, of 1640 years.

The Applicant sent us a letter a day or two ago which you all have, and if I interpret it correctly, it estimates a return period of 5000 to 10,000 years, which if my arithmetic is correct, is six to ten times as long. I believe the Staff will concentrate on what is a reasonable level for an SSE value rather than for the design of the CRBR for a .25 g.

This meeting today follows the site suitability meeting held June 24, and it was attended by several people, if I remember correctly, Mr. Bender, Mr. Ebersole, Kerr, Mark, Moeller, Okrent and Ray. This review that we are undertaking of site suitability is unusual but not unique. Paul Boenert has pointed out that the ACRS has reviewed a half-dozen or so sites in the past. And further, if the U.S. were to establish a bank of pre-approved sites, as has been sometimes suggested, we might do a lot more such reviewing in the future.

I would add for background information that several safety review meetings have also been held on topics such as HCDA containment and so on. I would like to turn from the site suitability discussion as such to the topic of the letter from Dr. Cochran of the Natural Resources Defense Council which arrived the day befoe yesterday, of which you have copies, I am sure.

There are two or three points I would like to inform you of. First, I called Cochran yesterday morning simply to discuss the letter. He was out to a meeting. I left a message for him to call me but he has not done so yet.

A second point. I am somewhat puzzled by the letter. Dr. Cochran is fully aware, I am sure, from his personal experience from Federal Register notices that both written and oral statements may be made by members of the public at our meetings, and yet, although he says he has attented four such meetings, we have had no communication from him of any kind that I am aware of except now when it is really quite late in our schedule to factor his concerns into our evaluation.

He presumably, surely is aware that it has not been our policy to seek out comment from potential public groups, and yet he has not come forth pr_or to this time.

With regard to his specific points, I would state that the Subcommittee was not aware of any of the NRDC contentions. He states to the contrary in citing a reference in one of the transcripts, but all that transcript reference actually says is that there are intervenors. It doesn't say who they are or anything like that. Actually, we had heard that NRDC was an intervenor, but we had no specific knowledge.

On the second point, that Staff and Applicant have made strikingly different presentations in the ACRS than to the Licensing Board, I cannot offer any comment now. Paul Boenert is assembling the transcripts and depositions, but it seems to come to something like 1000 pages, I guess, and it will take some time to assemble and some time to scan. So we have no information.

I guess I would say that since the letter was really addressed to the Full Committee rather than the Subcommittee and not me as chairman, I will stop there. I am sure, though, that the Staff will be prepared to present more information on the NRDC contentions this morning if you would like to hear them.

On this point -- let me go back a minute -- that the Staff and the Applicant made different presentations to us than to the Licensing Board, I was informed this morning that an appeal was being made on that. I am not much more acquainted with or aware of exactly what is going on further than that.

Let me at this time ask other members who were present at the June 24 meeting for comments which they would like to add, and then perhaps we could have any questions that anyone might wish to raise.

Carson?

MR. MARK: I have nothing to add.

MR. CARBON: Dave, do you have anything to add?

MR. OKRENT: Nothing.

MR. CARBON: Jerry, were you there?

MR. RAY: I wasn't there.

MR. CARBON: You were at one of the other meetings.

Are there any questions that anyone would like to raise before we follow the agenda in turning to the Staff?

MR. MARK: When you said that the Staff is not prepared to say yet that the site is suitable for what is is presently designed for, this is separate from the fact that there might be features in the design which might suggest further thought. This tangles, surely, head on with the request for an LWA-1, which if at all, will be acted on before the summer is over, or before the fall is over.

MR. CARBON: I think the Staff would say that any activities undertaken by the Applicant prior to issuance of the construction permit is strictly and solely at their risk. I expect they will address that.

If there are no further questions, then let me go ahead and turn to Mr. Check of the Staff.

Paul.

MR. STARK: Paul, before you start, maybe I would like to make a minor adjustment -- this is Richard Stark, by the way -- to the agenda. In addition to what is on the agenda, we have a consultant, Ed Rumble, who will assist Mr. Morris on II.C, Comparability to LWRS. In addition to that, under Geology and Seismology, Bob Rothman will be available to assist Jim Knight if required.

I would like to make a recommendation that after we complete Item II.C, Comparability to LWRs, that we then discuss source term. I think they follow each other. In addition, we have a slide that addresses the admitted contentions. If the Full Committee would like to address that, we could attempt to do that.

With that, I would like to turn the meeting over to Paul Check, who is the Director of the Clinch River Breeder Reactor Program. Thank you.

MR. CHECK: Thank you.

As you see, Richard Stark, Project Manager, is directing traffic for the Staff. He will continue to do so.

There isn't too much that I can add to what Dr. Carbon has said and to what Mr. Stark has said by way of introduction. But because there are some important distinctions to be made, it is probably worth repeating things just a bit.

We are here today to consider the suitability of the Clinch River site, not the acceptability of the Clinch River reactor. The distinction is important and, judging by the Subcommittee meeting, apparently somewhat elusive. I ask that you bear the distinction in mind, however, as we proceed through this morning's agenda.

You will hear more in a moment from Cecil Thomas regarding what specifically in our view is germane to site suitability.

The reason we are considering the site in this manner is because the Applicants have exercised the right they have in law to request a limited work authorization, LWA. That is, they seek to begin certain site preparation activities prior to receipt of a construction permit. A construction permit, as you know, would authorize them to construct the plant proper as well as to undertake any site preparation.

I want to stress at this point that the limited work that would be authorized does not include any safety-related systems or structures. While limited work authorizations are not uncommon, approximately 30 of them have been granted. They are usually granted at a point in the licensing process well beyond that at which the ACRS has reviewed the case. Thus, since the ACRS review had already been completed for each of those cases, there was never a need for the Committee to distinguish between the site and the plant.

MR. MARK: When you use the word "safety-related" in the statement you just made, is that the same use of the word we have meant before where some valves and pumps and pipes aren't safety-related?

MR. CHECK: It's a standard definition.

MR. MARK: So they could really build the aux building and quite a few things that aren't safety-related.

MR. CHECK: That may in fact be true, but that test is not before us. What is being requested is less than anything that we could agree is the plant proper. The law, that is, the NRC regulations contemplate such request for partial permission and prescribes in 10 CFR Part 50 conditions to be met. Briefly, they are that the ASLB find: one, that the NEPA or environmental review has been completed with a salutary conclusion; and they must find also that the proposed site is a suitable location for a reactor of the general size and type proposed.

Notice that the traditional finding of reasonable assurance on safety of the reactor is absent. That conclusion is a natural end point for the safety review, which in this case is not scheduled for completion until next year.

Now, as Dr. Carbon has mentioned, the Atomic Safety and Licensing Board has defined the issue for the limited work authorization hearing and has promulgated a schedule for its accomplishment. Our delineation of the site suitability issues, some of which are to be discussed here today more than others, is consistent with the Board's definition of the scope of the LWA hearing.

Okay, that concludes my orientation. And as I say, some of it is repetition, but I hope it is helpful.

Before I turn, then, to Cecil Thomas for a more careful description of the Staff's site suitability process, let me say again, as Dr. Carbon has said, we are requesting that the Committee give its opinion on the matter of the suitability of the site, the Clinch

River site, in a letter in order that we might proceed to hearing in August.

Unless there are questions, I would invite Cecil Thomas to come up and discuss the process. As I said, following Cecil, Bill Morris and our consultant, Ed Rumble, who has already been noticed to you, will discuss the matter of comparability which arose in the Subcommittee meeting.

MR. OKRENT: Excuse me. Where in the agenda are you going to tell us about the contentions, and also where will you comment on the statement that was made in the letter by Dr. Cochran that there was a difference in what was presented to the ACRS and to other groups?

MR. CHECK: I can do some now and some under Cecil Thomas. Cecil Thomas is prepared to deal with the contentions. I hadn't expected that we were going to be asked to respond to the letter impropriety by impropriety, but I will.

It is, of course, difficult -- without having Tom Cochran here -- to know precisely what that second point of his really meant. I think at the top of that paragraph he says something about the Staff has not been candid with the ACRS regarding how it has been dealing with the Board, or vice-versa. In any case, I think that is an error because if we looked at the transcript of the Subcommittee meeting, I think you would find that I was rather explicit in stating that the delineation of issues that we had before us with the Subcommittee and the full Committee was the outgrowth of discussions we had had with the ASLB.

So I believe in terms of scope of discussions, scope of issues, definition of what we think needs to be accomplished, we are telling the same story to both groups.

I believe toward the end of that paragraph he says something more on this point, and I would only offer as a possible explanation that Mr. Cochran sees that the entire spectrum of discussions we have had with the Committee, the Subcommittee, primarily, are fit and proper things and within the scope of what the Board has decided is the LWA hearing. I would say that that is not true.

What we have been talking about with the Subcommittee is much broader. It includes the safety matters. Most of our discussion has been on safety matters. Only two meetings, today and the previous Subcommittee meeting, had to do with the question of site suitability per se, and it is the content of those two meetings that would be what would map on the conversations or discussions we have had with the ASLB.

Now, as I say, the question of contentions will be taken up by Cecil Thomas in his discussion.

MR. CARBON: Will you or will someone else be saying anything more about the schedule? And particularly I would welcome --

MR. CHECK: I could show the slide I showed at the Subcommittee? We have a rather packed agenda --

MR. CARBON: Would you address whether or not you consider it important that we try to act at this meeting?

MR. CHECK: I'm sorry. I mentioned that the hearing is scheduled for August. A letter in July leaves the Staff with its traditional time to respond to that. We have that analog of an SER out there, which is the site suitability report, in this case. The Committee, as you know, write letters and then the Staff supplements its report prior to going to hearing. So some time is needed, and I think the time we have asked for is not undue. So a letter in July conforms to reasonable and traditional assumptions about scheduling.

MR. CARBON: If we were to delay for some period of time, would that cause any sort of delay farther down the road in the other aspects?

MR. CHECK: Well, there is no specific requirement for the letter in law that I am aware of, so we could proceed without it. I would have expected that the letter would have been the natural interim end point for this part of the Committee's deliberation. The Committee did ask to review the site suitability matter, and we have been assuming that they meant to unburden themselves of an opinion on that question.

Much of what we have said to the Committee has been interesting and stimulating, there has been a lot of good discussion. It is somewhat of an open record now, an open book, a book without a conclusion, a symphony without a final movement. Without the letter, it would give us a very interesting hearing to face without being able to point to something that represented some consensus view of the Committee.

MR. CARBON: If we delayed a month, would that delay the opening of the ASLB hearing?

MR. CHECK: It would be awfully close. I can't promise that it would, but it would, as I say, leave us virtually no time to amend our basic testimony document.

MR. CARBON: Fine.

MR. OKRENT: One more question. Can you tell me what the status or significance is of the criteria, the things called Clinch River Breeder Reactor plant design criteria given in Appendix A of NUREG-0786?

MR. CHECK: I will start by observing that in the preface, we have a paragraph from which I will quote. "Because the staff has not completed its safety review of CRBR, a process which may lead to changes in design or design criteria, descriptions of specific CRBR design features in this report are presented only as representative of a facility of the general size and type as CRBRP. Similarly, CRBRP design criteria in Appendix A are included only as representative design criteria for such a facility."

MR. OKRENT: What does that statement mean?

MR. CHECK: It means that we are not prepared today to say that they are the design acceptance criteria for the Clinch River reactor. I recall that in an early subcommittee meeting we went over the question of the development of design criteria at some length. Dr. Morris, who is here with us, can review some of that. We can if there is time in the schedule.

MR. OKRENT: I don't want to go over the criteria blow by blow today.

MR. CHECK: I am talking about the process.

MR. OKRENT: I am trying to see whether the staff treats these as plausible design criteria, acceptable design criteria. You use this word "representative", which suggests some degree of applicability in the staff's mind.

MR. CHECK: There is. I simply want to reserve, we are not finished. The scheme that we have outlined to the subcommittee regarding the development and establishment of design criteria is one that proceeds in parallel with the safety review itself, and that is why the conclusion of the safety review will be the time when we will be able to announce, these are the criteria. These are the criteria we would be prepared to defend as applicable, suitable and applicable for this plant, and then, of course, measure the plant against them.

So it is our plan that we proceed with the safety review, bearing in mind and testing continually the design against these initial statements, initial derivations from the general design criteria, but that we remain flexible and prepared if necessary to change the design criteria.

MR. OKRENT: Well, I couldn't tell my students what you meant, if I can use a common calibrater, but let's let it go at that.

MR. CHECK: Cecil Thomas.

MR. OKRENT: I might note in passing while he is getting ready, I don't feel that I am willing to accept many of the design criteria written in that appendix. That is why I raised the question. I don't know about the rest of the subcommittee.

MR. THOMAS: Good morning. My name is Cecil Thomas. I am section leader of the licensing section for the Clinch River Breeder Reactor project office.

(Slide.)

MR. THOMAS: This morning, I will give you an overview of the staff's site suitability review. I will briefly review some of the important features of an LWA-1 I will give you a list of the site preparation activities that the applicant proposes to conduct under the LWA-1. I will describe our approach to the site suitability review, and I will conclude with a few words on our site suitability report.

(Slide.)

MR. THOMAS: LWA-1's are governed by 10 CFR 50.10E. An LWA-1 authorizes the conduct of non-safety related site preparation activities. An LWA-1 requires the completion of the environmental and site suitability reviews, and the completion of public hearings on both environmental and site suitability matters. Any activities undertaken by the applicant that are authorized by the LWA-1 are undertaken solely at the applicant's own risk. The issuance of an LWA-1 has no bearing on any subsequent issuance of a construction permit.

The finding that must be made before an LWA-1 is issued is that based upon the available information and review to date, there is reasonable assurance that the proposed site is a suitable location for a reactor of the general size and type proposed from the standpoint of radiological health and safety considerations.

And finally, we have obtained a more accurate account since Paul indicated, we have now issued 27 LWA-1's since they were first establised in 1974.

(Slide.)

MR. THOMAS: This is the first of two slides that show in detail or at least list the site preparation activities that are proposed by the applicants should they get an LWA-1. I will not burden you by reading the specific activities. They are in your handouts. They basically fall into four general categories, the first of which is general site clearing and grading. The second, excavation. Thirdly, installation of temporary construction facilities.

(Slide.)

MR. THOMAS: And fourthly, other miscellaneous activities. We believe these activities are allowable under 10 CFR 50.10E, and we believe that none of these activities are safety related.

(Slide.)

MR. THOMAS: I have attempted to depict pictorally in this slide our approach towards site suitability review. We begin by defining the facility of the general size and type proposed. We define this by a limited set of characteristics that are relevant to a determination of site suitability. These characteristics are based on proposed CRBR plant design features, our experience with LWR plants, and our experience with other types of plants.

These bases also provide us with reasonable assurance that the characteristics and parameters assumed for a facility of the general size and type proposed are feasible, and finally, we assess the compatibility of the characteristics of the parameters of the facility of a general size and type with those of the proposed site. The characteristics of the proposed site are those normally found in Chapter 2 of the safety evaluation report or the PSAR. They generally conist of the ologies, geology, demography, foundation engineering, and consideration of emergency planning.

So, as Dr. Carbon pointed out this morning, once we have defined a facility of the general size and type, we in effect conduct an early site review, as you are all familiar with, under the standardization policy. We compare the characteristics of the facility with those of the proposed site, and make a conclusion as to a facility of the general size and type's suitability.

(Slide.)

MR. THOMAS: Our site suitability report for the Clinch River Breeder Reactor plant is NUREG-0786. That report documents the results of the staff's evaluation of the suitability of the Clinch River site for a facility of the general size and type as the proposed Clinch River Breeder Reactor plant. We conclude in that report that based upon the available information and review to date, there is reasonable assurance that the Clinch River site is a suitable location for a facility of the general size and type of the proposed Clinch River Breeder Reactor plant from the standpoint of radiological health and safety considerations.

This concludes my presentation on the site suitability review. If the Committee would like, I would be happy to address the contentions that are admitted into the proceeding, and specifically those contentions that we feel are related to the question of site suitability.

MR. CARBON: Yes, please do.

MR. MARK: A question. We know better some 1,300 megawatt electric plants, more about its design than you do at this moment let's say about the final details of the CRBR, so would this be a site suitable for such a plant?

MR. THOMAS: If those characteristics of the 1,300 megawatt plant --

MR. MARK: I am just referring to any standard type plant.

MR. THOMAS: I am, too. I am will try to be general in my answer. If the characteristics that are relevant to the site suitability match up with the parameters of the site in a way that we have looked at site suitability in the past in our acceptance criteria for site suitability, yes.

MR. MARK: "If" they did.

MR. THOMAS: Yes. We would have to evaluate the specific characteristics.

MR. MARK: Well, the ground is firm enough to hold the plant. The water is generous enough. And the only thing that might come into the way would be what should the SSE value be, and how does it stand with respect to floods.

MR. THOMAS: By and large my answer is yes, we would have to look at the details just to assure ourselves; but I see no reason, based on our review, there is no reason why you couldn't site any plant.

MR. SHEWMON: Is that the way you normally answer questions, or are you just being evasive? I cannot really tell.

MR. THOMAS: Dr. Mark asked a fairly general question. I tried to answer --

MR. SHEWMON: You are giving very general and vague answers, depending on your viewpoint. I take it the answer was yes. Is that roughly correct?

MR. MARK: Look, there are a couple of things that have to be looked at. The demography probably is okay. The soil is okay, probably. Water.

MR. THOMAS: There is no overriding consideration that would preclude licensing a larger plant at that site that we are aware of.

MR. AXTMANN: What actually happened when you updated the NUREG-0786 for the site suitability report of 1977?

MR. THOMAS: The changes in that report from the earlier report are indicated by lines in the margin. Basically, we updated it to reflect additional information that was provided by the applicants over the years, modifications to the design, for example, changing a homogeneous core to a heterogeneous core. It was updated to reflect later meteorology, which accounts for more data. It is more statistically significant, if you would. To note, for example, that FFTF is now in operation, and we have begun to receive operating experience from that, things of that nature.

The changes were not overwhelming. There were a limited number of them, and it was clearly characterized as an update.

MR. SHEWMON: Get on with the contentions, if you would.

(Slide.)

MR. THOMAS: You have been given a copy of the complete set of contentions that have been admitted to the proceeding. These contentions cover both the site suitability portion of the hearing and the construction permit portion of the hearing. The Board, as I am sure you have heard by now, has ruled on which of these contentions or which parts of the contentions are admissible for the site suitability part of the hearing. I have not attempted to mark those, because it is not easy. I would say that the Board has parsed those contentions, if you would, both vertically and horizontally.

Certain of the contentions have been ruled appropriate for the CP part, but others only portions of the contentions have been ruled appropriate for the LWA or site suitability portion of the hearing. For example, Contentions 1, 2, and 3, as I have indicated here, deal with inclusion of the core disruptive accident and the design basis accident spectrum, and hence -- the Board has sliced these three horizontally, and have limited consideration of those contentions to the feasibility of designing the plant in such a way that the CDA's, the probability of the core disruptive accidents could be made so low that they could be excluded from the design basis accident spectrum.

So, to give you the complete set of contentions that have been admitted to date, you have that package. This slide pulls out from that list those contentions or the portions of those contentions that we believe are relevant to site suitability. Those are 1(a), 2, 3(b) and (d), which, as I indicated, have been limited as to the extent that the Board will allow inquiry at the hearing. The general subject matter is whether the CDA should be included in the spectrum of design basis accidents and hence in the site suitability source term.

Contention 5(a) I have included in here for conservatism, if you would. It is really more related to the NEPA review of alternative sites. It has to do with the adequacy of the meteorology and population density at the Clinch River site versus the alternate sites.

MR. SHEWMON: Would you read from the vu-graph back there so that we can, too?

MR. THOMAS: Contention 5(b) relates to the effects of long-term evacuation of nearby facilities, specifically, the facilities at Oak Ridge, X10 Y12, K25 and the proposed synfuels plant. We have discussed this matter in some of the subcommittee meetings. Item 11(d)(1) has to do with 10 CFR 100.11, organ dose equivalent limits.

Without attempting to characterize the various positions of the parties at the hearing, I have attempted to be as objective as I could in summarizing what our understanding of the issues in controversy are.

MR. OKRENT: On 5(b), what is the staff position, that this will not occur, or that even if it occurred it wouldn't be an unacceptable loss to the nation? Or just how do you respond to that?

MR. THOMAS: To begin with, we normally do not look at the effects of evacuation of nearby industrial facilities during the course of our review specifically. Let me elaborate on that. We do a review to assess the radiological health and safety consequences for various distances out from the site. We make a judgment as to the acceptability of those consequences.

During the operating license phase of our review, and to some extent during the construction permit stage, we work with the applicant in developing an emergency plan that is an integrated plan that considers a number of things, not only --

MR. OKRENT: You are really asking a different question, because that is not the question raised in the contention.

MR. THOMAS: I am getting -- let me get to that.

MR. OKRENT: Get directly to it.

MR. THOMAS: The Department of Energy is responsible for those facilities. They know the activities that go on at those facilities, both in terms of national defense and energy. It is really ultimately their responsibility to make the decision whether they want to site a nuclear plant near those facilities.

During the OL stage, we will be sure that there is an emergency plan, but that falls within the Department of Energy.

MR. OKRENT: So the decision with regard to Contention 5(b) is that it is not offering an opinion? Is that what you are saying?

MR. THOMAS: The bottom line is that, yes. Any further questions?

(No response.)

MR. THOMAS: Thank you very much.

MR. AXTMANN: On that point, is the applicant the best judge of such matters? It sounds like this is a --

MR. THOMAS: The applicant is the Department of Energy.

MR. AXTMANN: Yes, but it is the Department of Energy who makes the statement is the section that we are discussing, 11(3)(b).

MR. CARBON: 5(b).

MR. AXTMANN: 5(b).

MR. THOMAS: These are the contentions of the intervenor. These are the intervenors' contentions.

MR. AXTMANN: Based on what you have said, I find that they have merit.

MR. THOMAS: I didn't mean to imply that at all.

MR. AXTMANN: I know.

MR. THOMAS: You see, these contentions, there are three parties to the proceeding, the applicant, the staff, and the intervenors. The staff and the applicant at this point in the proce.d'g have no contentions. It is only the intervenor. T. intervenor is saying, hey, look, there may be a problem here. It is the applicant, specifically the DOE portion of the applicant, that has to make the decision as to whether or not it is prudent to locate the Clinch River Breeder Reactor plant within the proximity of those other facilities.

MR. OKRENT: With regard to Contentions 3(b0, (d), you will address those when you address the site suitability source term?

MR. THOMAS: We could go into that more.

MR. OKRENT: Would you include that in your presentation, fit it in at the appropriate time so that it is not just --

MR. THOMAS: For clarification, are you referring to the general subject of whether the CDA's can be made -- there is reasonable assurance that they can be made sufficiently improbable that they don't have to be included in the spectrum of design basis accidents?

MR. OKRENT: That is one of them. I think you should comment on it. If you have no comment, you can say you are not going to address it or whatever it is. Similarly, under (c), (b), (d), and so forth, if you would just include that, it would be helpful.

MR. THOMAS: I would just note that in the many meetings we have been having with the subcommittees, and I think once before the full committee, we have made it clear that it is our position that the core disruptive accidents are sufficiently improbable or could be made sufficiently improbable that they do not have to be considered in the spectrum of design basis accidents since the site suitability source term bounds the sources of that spectrum of accident so that it is not appropriate to include.

MR. CARBON: Why don't you tie that in more with the discussion at the time?

MR. THOMAS: We will do that during the source term discussion.

MR. CHECK: Let me address that briefly while Bill Morris is moving up. I believe Dr. Mark asked the question, could we put down a 1,000 megawatt E light-water reactor on this site, and would it be a suitable location for such a reactor. I think if you will accept the short answer that I would give is "probably."

MR. MARK: I recognize there are questions you would have to go into.

MR. CHECK: What keeps us from making a fuller answer and a more affirmative positive answer is that we know we probably have to put cooling towers down there. There are the environmental considerations, and raising the power by three probably means the exclusion distance may be a little tight; we might have to go out a little. But I am almost certain that it could be made suitable by a combination of things.

MR. SHEWMON: Mr. Longenecker, do you wish to make a statement at this time?

MR. LONGENECKER: Yes, Mr. Chairman.

MR. SHEWMON: I really don't want to get into the Natural Resources Defense Council letter. If that is what you are mainly there for, I wish you would make it extremely brief. What is it you wish to --

MR. LONGENECKER: Mr. Chairman, I have a few remarks to make. I would like to discuss the presentation we are going to make today. I would like to make a brief, as in about ten seconds, statement on our position on the NRDC contention, and I would like to make a statement --

MR. SHEWMON: Fine. If you take more than ten seconds we will cut you off.

MR. LONGENECKER: My name is John Longenecker, from the Department of Energy. We are the applicants. We will be presenting today information on the site and clean up some of the issues that were left over from the June 24th meeting. I would like to introduce the presenters for you, but before that, I would like to thank Dr. Carbon for the opportunity to present information on the CRBR project over the last six months.

With regard to the action that the full committee is considering here today, I would like to state that we believe, based on the information that we have submitted, that we have adequately assessed in all regards the potential environmental impacts of the CRBR site. The request made by Paul Check previously that the full committee take a position and report favorably on their report, we support that.

With regard to the NRDC letter, which speaks to the appropriateness and consistency of our presentations to the ASLB and the ACRS subcommittee today, I would just like to say that in our opinion our presentations have been totally consistent, both technically and factually, and a key difference, as I believe was pointed out in Cecil Thomas's presentation, is in the scope of the contentions and the detail at this point in the process with which the Atomic Safety and Licensing Board has ruled that we should review.

In other words, we have presented exactly the same information bottom line with the information that is contained in the Staff site suitability report, which is representative of that we have presented. We will as a point of final clarification be presenting information at the hearings which will begin in August on each of those contentions, and the intervenors will also be providing their testimony, and each of those that the Board has admitted will be discussed and litigated to the extent the Board has allowed.

So as a final point to introduce the people who will be presenting, today we have an assortment of people both from the Department of Energy, from the Project Management Corporation and the Tennessee Valley Authority. We are co-applicants with our utility partners in this matter.

For the first presentation on the overall description of the site will be Henry Piper, who is from the Project Office Licensing Branch. Following that on the site suitability source term George Clare who is from the Westinghouse Licensing Division will review both that and he will be discussing a comparison of source terms with other countries, principally focusing on source terms used by the U.K. and the Japanese. George will also discuss the open item from the last meeting, and that is the assessment of the consequences of a release of non-radioactive sodium on site and other potential impact of sodium aerosols with LMFBR reactor system design.

On the last agenda item, hydrology, we have some people here from TVA. Namely, Ray Lee, who is an expert hydrologist with TVA, will discuss the analysis of Clinch River floods which could impact the site. He will be followed by Tom Abraham, who is the head civil engineer of the hydrostructural design for TVA, and he will discuss the Norris dam failure.

For the last discussion, Henry Piper will summarize the effects and potential impacts on groundwater due to the hypothetical core melt.

In addition, we have with us from TVA Joe Hunt, principal engineer of the geotechnical and earthquake engineering staff of TVA; and and Jim Domer, the TVA supervisor for BWR licensing. They will address any issues any of you may refer to.

Unless there are further questions, I would turn it over to Bill Morris, who will continue. We will be prepared to do the site suitability in the order that you discuss.

MR. MORRIS: At the recent joint meetings of the CRBR Siting Subcommittees, a question arose with regard to the Staff statement in its site suitability report about our objective that the risks at Clinch River would be comparable to those for light-water reactors and what that meant, and a more general question, I believe, arose with regard to a general characterization of the hazards that could be related to a plant such as Clinch River or a general LMFBR.

I am going to try to clarify some of these points today and just point out to you that when we talk about comparability of risk of Clinch River to LWRs, we are talking about the current generation of LWRs undergoing licensing today. That is modern LWRs.

We are not talking about an average LWR, I think. We are talking about risk in terms of both the probabilities and frequencies of accidents. More, we have that risk as a product of frequency in mind. However, I want to make it clear that we believe it would be too restrictive a concept to think of this only in numerical terms. We believe that the capabilities of current probabilistic risk assessment methodology are such that the uncertainties are still too large to try to make a numerical comparison of this type and to rely only on that as a basis for making this judgment.

In addition to this, we have a number of more deterministic criteria that we apply in making judgments about the comparability of the different kinds of reactors. Of course, the CRBR will be expected to meet all the applicable LWR criteria, and there are a number of these, from the Standard Review Plan, the Code of Federal Reguations, the ASME and IEEE code standards that would apply to systems such as, for instance, protection systems, which we believe will ensure comparability between the light-water reactor and those for Clinch River.

There are, of course, specific examples where there must be special criteria developed for the sodium-cooled reactor. Those will be applied also. But there in general we are talking about deterministic criteria and the application of those to assure low risk.

Another point is the consequences of design basis accident, including a bounding site suitability source term to adequately bound all the design basis accidents, will be required to meet the 10 CFR Part 100 guidelines.

Now, remember that the site suitabilty source term we are using here for Clinch River is a non-mechanistic postulated release of radioisotopes into containment followed by design basis leakage from containment but with no containment failure. That source term includes some contribution from core melting that could only realistically be considered to be associated with some core disruption. But the source term does not include -- is not based on an attempt to bound all postulated CDAs that one might consider. And in the non-mechanistic way that it is treated, we don't anticipate that it really is closely related to CDA analysis.

We have required in previous communications with the Applicant that certain design measures will be imposed on the Clinch River design to assure that severe accidents such as CDAs will be improbable and hence are beyond the design basis spectrum.

MR. OKRENT: Are you able to quantify for me "very improbable"?

MR. MORRIS: We don't have a specific threshold that we apply to decide whether an accident is within or beyond the design basis. We can characterize the probability and make estimates of what the probabilities may be, but I would not want you to believe that that was being used as a threshold discriminator for design basis versus nondesign basis accidents.

So I think that I could not give you a numerical value for what this probability threshold might be. We do have a judgment that there are sufficient design measures that can be imposed to assure that the accidents will be sufficiently low in frequency that they need not be considered as part of the design basis.

MR. SHEWMON: When you talk about a CDA, is that defined as when the core might disrupt or when the accident might get out of the pressure vessel?

MR. MORRIS: The core might disrupt. We are talking about a core disruption.

MR. SHEWMON: So it has nothing to do with the energetics of what that would be or the strength of the head or anything that would be there to contain it?

MR. MORRIS: Once you postulate that there might be core disruption, you have to consider the possibility that there would possibly be some damage done to the primary system, either mechanical or thermal.

MR. SHEWMON: I am aware that such things are sometimes done with my present position. That was why I asked the question. The question has to do with when you talk about a probability, is it the initial core disruption or are you including in that some of these other considerations?

MR. MORRIS: When we talk about a judgment that the probability of a CDA is sufficiently low to exclude it from the design basis, we are talking about the initiation and the core disruption but not the subsequent failures of the primary system and/or containment that might occur.

MR. SHEWMON: Thank you.

MR. OKRENT: Excuse me. Suppose before the time that you get to the point where a hearing board is reviewing a construction permit and arriving at some decision with regard to CRBR, the Commission arrives at some position with regard to severe accident rulemaking for LWRs and says that certain measures need to be dealt with for light-water reactors.

It already has said that you have to deal with substantial amounts of hydrogen. Would that in some way affect what you are telling us now? I am trying to understand how you relate what you have just said about CRBR to this halfway position that the Commission is already in, and in fact it is sort of a five-eighths position with regard to near-term CPs, which this is not quite.

Do you understand my question?

MR. MORRIS: I think I understand. Even before Three Mile Island and the current interest in degraded cores. the Staff and the Applicant were considering how to go about accommodating severe accidents.

Two points. First, you take measures to assure that the accidents are very improbable. But then the Staff further said in its letter of May 6, 1976 that the design should be capable of accommodating severe accident. The word "accommodate" here should be distinguished from "mitigate." We use "accommodate" to mean that there should be in the design sufficient measures to assure that containment will survive for a long enough time that the consequences will be acceptably low.

What that translates to in terms of design features and what has been proposed by the Applicant are measures to cool the steel between the annulus and the outer building subsequent to a core melt, measures to vent through filters and to purge the system to control hydrogen subsequent to a core melt, and in general I would say there is already in the Clinch River design and in our licensing proceedings measures taken to take into consideration the consequences of severe accidents. We don't just say we want to make it very low. We go beyond that to say we want to make sure they won't result in severe consequences.

We are looking at a design that will accommodate core melt-throughs and the pressures and temperatures that could occur in containment and assure that they do not cause early containment failure.

MR. OKRENT: Why don't you go on.

MR. MORRIS: Again, one point that wasn't clear. This does include a consideration of hydrogen that could be generated from sodium-concrete reaction, and we take that into account.

Well, all these more deterministic criteria have been discussed. We have gone recently through an exercise to come up with a preliminary evaluation of the risk that could be associated with the CRBR. This is in relationship to the work to issue an update to the final environmental statement. It is similar to the analyses performed for other recent environmental statements in conformance with the policy statement made by the Commission to take into account the consequences of a Class 9 accident.

We do not propose to you that this is a PRA. This is not anything but a preliminary scoping analysis of accidents and their consequences that might occur. Subsequently, however, to further confirm that the risks from CRBR will be low and will be somewhat comparable to light-water reactors, there will be a probabilistic risk assessment performed by the Applicant to confirm that the safety goal will be met for Clinch River.

MR. CARBON: Do you believe that that last one truly will ever be met?

MR. MORRIS: Do I believe the safety goal will be met?

MR. CARBON: No, that the PRA will confirm that the CRBR meets the safety goal.

MR. MORRIS: To the extent we understand the way the implementation of the safety goal may proceed, this is, again, somewhat preliminary because the actual implementation plans are just being discussed and developed. But we believe the PRA will be a large part of how we deal with the implementation of the safety goal. We believe that to the extent the PRA can be used for dealing with the safety goal for light-water reactors, that it can also be applied to Clinch River.

It is in that sense that we make this statement about how this PRA will be used. I have to qualify it to say that only to the extent we have become confident that PRA methodology is sufficient for that purpose would I believe that we could do it.

MR. OKRENT: I guess if I were a member of the public and said what did he just tell me by the statement that the Staff is going to try to make the risks comparable to those of a light-water reactor, I would say, well, he didn't tell me how low a probability accident had to be before I didn't consider it in the design basis. He didn't tell me how low in probability the core melt accident needed to be. He did tell me he didn't know how to quantify the risk from light-water reactors or from LMFBRs. So what did he tell me when he said he was going to try to make them comparable? I guess I couldn't even explain it to my children. Am I wrong?

MR. MORRIS: I think I said that there are a number of deterministic criteria that, when applied, will assure us that the risk will be acceptably low. We can do probabilistic assessments to try to come up with numerical values for what those risks might be, but because of the uncertainties inherent in risk methodology, I think one would have to be cautious about interpreting those values. So that is the reason I don't want to be very specific about the use of numerical values in making these decisions. I don't think it has been generally accepted that the methodology is yet sufficient for its use in that way.

MR. OKRENT: Well, let me suggest you could have a non-acceptance limit, if the core melt probability was 1 in 10 per year, that that should be a design basis. If it were 1 in 100 per year, I would expect you would have to find that it was included in the design basis.

MR. MORRIS: If I found that, it wouldn't be an acceptable design.

MR. OKRENT: Well, they might have a way of dealing with it, that it was so good that the reactor was still safe. So whatever. I said core melt, but you take your choice. A leak in the hot leg. Okay? There is some probability when it becomes a design basis, and there is some when it is not. There is a threshold in your mind.

MR. SHEWMON: You made the point. Let's go on. Are you through?

MR. MORRIS: I now want to introduce Ed Rumble, who is Corporate Vice President for SAI. We have asked Dr. Rumble to assist us in evaluating the hazards and risks of Clinch River and in performing this analysis for incorporation into our final environmental statement. He has had considerable experience in probabilistic risk assessment, and will be involved in our review of the applicants PRA.

The most significant experience he has, to our way of thinking, is the work that he has done in evaluating the risks for the SNR-300 reactor, which is a reactor very similar to the Clinch River design in concept, and he has been involved in evaluating certain significant accident sequences all the way from inception through the release fractions.

MR. CARBON: One question before you leave. If he is going to talk primarily about PRA, you have used the word "deterministic," but doesn't it really come down to it is going to be your engineering judgment?

MR. MORRIS: I think it is a consensus judgment based on the continued acceptability of these deterministic criteria. There are some specific cases where we will have to look at very detailed acceptance criteria related to those that we will have to make a judgment that they have been met, ultimately.

MR. CARBON: Well, in the final analysis it is going to be your engineering judgment.

MR. MORRIS: I think it would evolve down to a judgment, but it is not one individual's judgment but it is a concensus judgment about what is an accepted practice for designing a reactor.

MR. OKRENT: Before you take that microphone off, one of the recent light-water reactor PRAs, Zion, and I guess again Indian Point, makes the claim that with their design for certain families of core melts, they don't have any loss of containment integrity. It is not just that there is a delayed loss. They say they will not have any loss.

They also argue that the probability of a loss of containment integrity, again, not a delayed loss, is very low and you integrate it over the family of more probable core melts.

Now, have you considered whether that should be the kind of criterion you should have for a CRBR rather than the one which you expressed in the May 19 something --

MR. MORRIS: 1976. .

MR. OKRENT: -- '1976, was it, letter, where you, if I understand it correctly, requested that there be an ability to maintain containment integrity for an extended period of time like 24 hours? But if I recall correctly after that, sort of the building could fall down and everything could get out, according to the letter.

MR. MORRIS: We have recently discussed with the Subcommittee a new set of criteria that we think will prefer to that 24-hour criteria. It relates to the fact that in this design there will be a possible venting of the material inside containment subsequent to a core melt accident. We believe such venting should not result in consequences greater than 10 CFR Part 100 even though those guidelines are not designed for that purpose. It is to assure that venting will not be a severe health hazard compared to, say, the subsequent failure.

We believe we want to see a high probability that the containment will survive and that there will not be a precipitous failure. That is, it will not fall apart. So we think that is too narrow a criterion and we are changing to some slightly broader criterion.

MR. OKRENT: At the Subcommittee meeting when I asked this question, the only statement I was given at that time, and the committee meeting continued after I left, but you just said, look at the May 6th or whatever it is letter, and you did not qualify it.

MR. MORRIS: I think it was the inadequacy of that response that prompted us to come back today and give you a better discussion, but that was what we had said at that time.

MR. OKRENT: Is there something in writing that modifies the position taken in that letter?

MR. MORRIS: We have said in the site suitability report that that is under review, that 24-hour criterion is under review, and in the subcommittee meeting on containment when we discussed core melt accidents, I presented the new criteria we would hope to impose. We are still evaluating those criteria to determine whether we find them acceptable or not.

MR. OKRENT: Are you telling us you are not prepared to tell the ACRS today what your criterion in this regard is? You are evaluating it?

MR. MORRIS: To recapitulate what I said before at that earlier meeting, if that is what you would like, if there is time to do that --

MR. OKRENT: It is what you would like. I am trying to understand just what the Staff position is. I just heard of a change which is different from what I heard at the Subcommittee meeting which was held, I think, only last week or the week before.

MR. MORRIS: I believe in the Subcommittee meeting I was trying to say without I think referring specifically to the 24 hours in my discussion, that we would like for containment to be capable of retaining radioisotopes for a sufficiently long period of time subsequent to a core melt that the risk would be acceptably low. We believe if that is done it will be comparable with light-water reactors.

The specific time is what I think is the problem here. The 24-hour number that was chosen and was published in the May 6th letter of 1976 was based on a judgment coming from WASH-1400 studies in which containment failures occurred both before and after 24 hours and that was taken as a mean value to be used as a target.

I think that there is somewhat of an insensitivity of the consequences to the exact time of a release provided the release is held up for a sufficiently long period of time.

MR. SHEWMON: Could we leave it at that?

MR. MORRIS: So we have a criterion, but it is not a very specific one such as 24 hours anymore.

MR. OKRENT: Well, Mr. Chairman, I am confused. I think it is somewhat relevant to know what their criterion in this regard is.

MR. SHEWMON: You may ask him to send it to you. I am not sure asking him several more times today is going to help.

MR. OKRENT: Well, you may find it hard to write a letter when what we really heard is it is under review.

MR. MORRIS: I would just point out that it would be in the transcript of the Subcommittee meeting on containment in which we discussed the core melt accidents. Those criteria were spelled out there.

MR. CARBON: Mr. Chairman, I think it would be worthwhile to ask him once more to try to be more specific. Can't you be a little more precise in answering Dr. Okrent's question? Forget what you said in the past.

MR. MORRIS: Okay. Let me see if I can -- What I will have to do is recapitulate what those criteria that we are proposing to use would be.

MR. OKRENT: Look, if you want to sit down and tell us in an hour, that would be perfectly fine as far as I am concerned. Collect your thoughts.

MR. SHEWMON: If you have got it written someplace, that would be a little more authoritative than you repeating what you think you said while on your feet.

MR. MORRIS: I will address it later.

MR. SHEWMON: Now we get to discuss PRA? Is that right? Mr. Stark, you are responsible for the agenda here and to be a traffic cop, I was told. We have now increased by 50 percent the number of people who are on this agenda and the time is going up appropriately. I hope you are a good traffic cop before the day is over.

MR. STARK: Well, we have some provisions for shortening some of the sessions later on if necessary.

MR. RUMBLE: My name is Ed Rumble, and in the context of risk comparability, I am going to briefly review a quick scoping analysis that was done, a short scoping analysis that was done for input to the final environmental statement regarding quantifying frequency and consequences of some accident sequences for CRBRP.

There are a couple of things you should keep in mind before I get started with the presentation. It will be pretty short. I am talking about the CRBRP design. There are some assumptions regarding procedures, human interactions, things like that that are comparable with LWR procedures and human interactions -- These things are not available right now -- and also that the plant is built the way it is supposed to be built and maintained and operated the way it is supposed to be.

On the other hand, in this analysis I did note that there is a wealth of information, I guess a wealth of analysis as background material that is available. The accidemt delineation study at Sandia. There was a PRA that was done earlier. CRBRP-1 is the name of the report. There are a number of topical studies that have been done. So there is quite a lot of information available on this facility to start with.

(Slide.)

Basically, performing this analysis for the FES. We had to start somewhere. Some of the basic assumptions are on the first slide. The basic considerations. First of all, as in an LWR, the dominant risk associated with the facility comes from the core. Secondly, the core inventories are comparable on a megawatt basis. Plutonium is roughly a factor of 3 higher, but otherwise, it's roughly comparable -- the core inventories are roughly comparable.

Thirdly, the starting point for looking at core disruptive accidents -- the severe accidents are the ones we're going to be talking about today -- is, again, the heat imbalance problem, the heat generation versus heat removal.

One can start from that point and fairly logically deduce initiators that can get you into these conditions and then try to quantify the frequency of such initiators.

The types of accidents that could occur at this facility are broken into three categories; internal plant failures, external forces and sabotages. This is a fairly important here that I am going to be discussing in internal plant failures, and I have not done any work directly, in a PRA-oriented vein, on external forces or sabotage at this point. This is just the internal plant failures.

There are really three phases to the analysis. One is the initiation phase, then one looks at the primary system and how that can be challenged; and thirdly, one looks at the containment and how that can be challenged.

In the initiation phase, as I mentioned before, if you start with the heat imbalance and logically proceed from there to look at the ways you can get this heat imbalance, you come up with several classes of accident initiators. On the second slide I have these classes listed.

(Slide.)

They are fairly familiar. LOCA, flow blockage, et cetera. The transients include here -- these transients mean that the scram system is demanded, and in the transient category we include cases where the scram systems do not work.

When you look at the internal plant failures and you try and analyze their impact, you have to look at the safety systems engineered in the plant and see how they respond, such as decay heat removal systems and the scram system. And this was done.

In addition, one other important point I forgot to mention and a very large part of any kind of analysis like this is you want to look at common connections between initiators and containment failures and primary system failures, things that can cause all three to fail at one time. That is also part of the thinking here.

Going on to the primary coolant system, once we have the initiation of a core disruptive accident, then we look at the ways the primary system can fail. There are thermal failures, mechanical failures that are considered. In the case of CRBRP there is a potential for head releases after energetic CDAs, and also for the bottom vessel head to fail from a melt-through type situation.

There are some accident sequences that could end at the point of a head release and no thermal vessel failure if the vessel could retain the degraded core. If the vessel does fail, then we have both a potential head release and the dumping of the sodium, a million pounds or more of sodium, plus the core inventory and steel into the reactor cavity.

The next part of the analysis is the containment response. The containment response is both the thermal dynamic response and the integrity response.

(Slide.)

When you compare containment of the CRBRP to the LWR, there are important differences. First of all, in the design of the CRBRP, of course, we have sodium which is many, many degrees below its boiling point. We do not have the blowdown forces associated with LWRs initially.

Additionally, we have the potential for a relatively large amount of sodium aerosols to be in the containment environment, which can play an important role in fission product behavior. These aspects have to be taken into account when looking at containment behavior.

As far as containment failure modes are concerned, there are the typical containment systems, as are in LWRs. There's a containment isolation system, for example. In addition, it has a filtered venting system specifically for the severe accidents in which the containment environment can be scrubbed and filtered and vented out of the containment to maintain containment integrity.

There is also the potential for prompt failures in the containment due to, at this point, hypothetically postulated type instances of very large energetics or a very large sodium spray fire. These are the aspects of the scoping analysis that were considered.

As I mentioned before, you have to look at things that connect all three of these aspects together that could all of them to fail at once.

(Slide.)

An important one that one has to consider is, for example, loss of all off-site electric power. In this case, electric power could supply power to both the containment systems and to the engineering safeguard systems, and this has to be looked at and it was looked at in this analysis.

The point here is that there are a number of systems that are available for decay heat removal, which is a primary concern in this case. However, they all require electrical power, and what may seem to be totally diverse systems may not turn out to be when one looks at the service systems needed to power these systems. So in the case of the loss of all off-site electrical power, electrical power plays a predominant role at the site in this accident sequence.

So, one has to analyze the emergency power supplies, the diesel generators and the batteries to come up with frequencies for this type of an event.

Basically, these considerations were taken into account and we came up with a set of accident sequences which were then supplied for analysis so far as consequence goes.

(Slide.)

I should mention the work I was involved with was to develop frequency estimates and release fraction estimates for typical accident sequences that could occur at the site. This matrix sort of slide here shows four types of accident sequences that were analyzed using the COMIX code. It was considering both small and large head releases and the various type of containment failures one could have.

I guess at this point, if there are no questions or further discussion, I want to turn it over to Mr. Hulman who will discuss the risk numbers that we got for these accident sequences.

MR. OKRENT: Just one easy question. What are the weak points, would you say, in what you have done? In other words, where do you feel that you may have made poorly-based assumptions or you ignored things of necessity, or whatever?

MR. RUMBLE: First of all, the completeness question is certainly a weak point. This was a scoping study to look at typical accident sequences. This is not a fullblown PRA. Therefore, I have no confidence that this is a complete study, although I did look at it.

For example, we looked for accident sequences such as an interfacing system LOCA type of accident that could occur in an LWR. A situation where you could bypass the containment.

We spent some amount of time looking for it with some information. There could be more time spent in looking for these types of sequences. So the completeness point is one area. Again, we're not considering any external events or sabotage, but certainly it should be pointed out as a very important part of any PRA analysis and that was not done at this point.

The human interaction area, of course, is another area. Because we are looking at a paper plant at this point, we don't know specifically how the human is going to interface with this system.

MR. EBERSOLE: May I ask a question?

MR. RUMBLE: Those are some of the weaknesses. I could go on.

MR. EBERSOLE: For the numerical values you used, this containment uses a design which envisions large flow ventilation and large purge valves hypothesized to close under such pressure pulses and release rates as one might get during an accident. What reliability did you use to estimate the closure of the containment in the mechanical context?

MR. RUMBLE: Yes. The containment isolation system used a frequency of 10^{-2} per demand. We think that is achievable for a containment isolation system to be designed and operated and maintained at 10^{-2} per demand or less. Typical LWR values I think are in the 10^{-3} to 10^{-2} range, 3×10^{-3} comes to mind as a number for WASH-1400. And I think, reviewing the system quickly, the number of valves 24 inch inlet and outlet on that system, and the redundancy in the electronics and things, I think that's achievable at 10^{-2} or less for that system under accident conditions in the environment for that system.

MR. SHEWMON: Thank you.

MR. HULMAN: Good morning, my name is Jerry Hulman, I am Chief of the Accident Analysis Branch in NRR. I want to talk about four interrelated subjects. In the interest of brevity, I am going to do it quickly.

The first thing I'm going to talk about is the risk of a beyond design basis accident. Secondly, I want to talk about site suitability source term. Thirdly, dose guidelines for site acceptability. And last, the design basis accident enveloping event that we have used for site suitability.

Ed Rumble has just told you about the accident sequences and the consideration of probability and release fractions. We have used that information in the same manner that we would use it for a lightwater reactor to evaluate the consequences and risks of severe accidents for beyond design basis events, for environmental impact statements.

Our conclusions are that the risks are generally comparable. They are not only comparable for a lightwater reactor of similar size, but they are also comparable for a contemporary reactor of 1000 or 1200 megawatts.

With regard to site suitability source term, we have presented to the subcommittee on two occasions our conclusion that we can use a non-mechanistic event that is analogous to what is used for lightwater reactors and postulate the release of 100 percent of the noble gases, 50 percent of the halogens, 1 percent of the solids and 1 percent of plutonium in a design basis, limiting kind of accident for site suitability.

The only difference between this array of activity and what we use for lightwater reactors is the addition of plutonium. Plutonium beig a significant potential dose contributor.

Questions were raised at the subcommittee meetings about whether plutonium is the only actinide that we've considered. I will try to address that quickly by saying that we've considered all the actinides and as it turns out, plutonium is the dominant dose contributor. So we have used plutonium all by itself and all the isotopes of plutonium in our site suitability accident analysis.

With respect to dose guidelines for site acceptability, for Part 100, we have dose guidelines for site suitability for thyroid and whole body; 300 rem thyroid and 25 rem whole body, respectively. For site suitability purposes for the breeder, because we have the possibility of releasing different kinds of activity, we have added other organs and changed somewhat the dose levels that we are using. We have added lung, bone surfaces, red bone marrow and liver, and have made dose equivalents to the lightwater reactor dose guidelines in Part 100.

Basically, what we have tried to do is say we do not want the risks from the breeder for design basis accidents to exceed the risks from the lightwater reactor. We have developed criteria for that. We have evaluated the site suitability source term using engineered safety features of the type being proposed for the breeder, and have found that the resulting doses we would get are a small fraction of the guideline.

That is brief and to the point. If anybody would like any details beyond that, I am happy to present them. Yes, sir?

MR. SHEWMON: There is appreciable evidence that the source term used for LWRs, there is little physical resemblance to what the fission products are in the gas in such an accident. This can be partly plate out, agglomeration of particulates, conversion of iodine to cesium, a variety of different things. Do we know enough about how fission products are likely to come out of the core or the pressure vessel of an LMFBR to say that there is any comparable sorts of conservatism? Or if we don't, is that largely irrelevant?

MR. HULMAN: The question is not irrelevant. I think it is right to the point. Let me point out that the staff has been considering the question of source terms ever since Part 100 was formulated. Recently, we have published two NUREGs on the subject for lightwater reactors, NUREG-0771 and 0772.

In those NUREGs we found that there is a possibility that the source term we postulated for site suitability could be conservative for some accident sequences. We have not been able to conclude to date that it is conservative for all possible accident sequences for lightwater reactors.

We have tried to consider very briefly the same question with respect to the breeder. My understanding of the staff judgment is the source term we have postulated in terms of its contribution to potential doses is representative of some kinds of beyond design basis accidents and is not as conservative as we might get for other events. It is not a bounding source term for all possible breeder events. That is my understanding of the staff judgment.

MR. SHEWMON: Okay. Any other questions?

MR. MARK: You said, I believe, plutonium is the dominant contributor to dose. Does that mean of the heavy elements only plutonium is dominant?

MR. HULMAN: Of the actinides, that is correct.

MR. MARK: The fission fragments are much larger.

MR. SHEWMON: The statement was it was dominant for the actinides, not dominant --

MR. MARK: Right. How do you get the curium out of the picture? There is ten times as much activity as plutonium.

MR. HULMAN: We have made a computation of the release of all of the actinides, taking them through decay in the containment structures and through the annulus filtration system, out into the environment with the normal atmospheric diffusion and have found that their dose contribution is small with respect to --

MR. MARK: I am just unable to detach my thoughts from the fact that there are ten times as many alpha particles per second released by curium as from plutonium. In the end-of-fuel cycle stuff in the fast reactor, I can understand the curium may not be dominant because you might have some reason for saying it is not as volatile or something. I'm asking how you get rid of it.

MR. HULMAN: We release it. I would like to ask one of my staff who did the calculation to discuss his consideration of the actinides, and it may shed some light.

MR. BELL: What we did was, we ran the case with all the actinides and we ran the case with strictly the plutonium isotopes. What we found was that the plutonium in the critical organs that we are considering at the present time, it contributed 97 percent of the total dose.

MR. HULMAN: Larry, I think the question was how much curium did you release?

MR. BELL: The same amount as we did for plutonium, 1 percent.

MR. MARK: One percent?

MR. BELL: One percent.

MR. MARK: Then it has ten times the alpha activity, and so it must go somewhere else. Maybe it does not go to the organs, or --

MR. BELL: Maybe not the ones we have been told are the critical organs. I don't know. But it's not included in these calculations; it just doesn't show up. It may be that RAB of the --

MR. HULMAN: Is it the combination of decay, Larry, and the diffusion conditions that you have assumed that could result in curium not being the major dose contributor?

MR. BELL: No, I wouldn't think -- well, it may be decay. I would have to compare the decay scheme--

MR. MARK: It's quite good enough for a short-term dose.

MR. BELL: For a short-term dose, no, because the filters presumably would act on all the elements the same because we assume that they were particulate, but maybe Walt Pashack will speak to that.

MR. MARK: I am just curious as to where it goes.

MR. PASHACK: I am Walt Pashack, I am section leader of the Radiological Assessment Branch. The doses we're calculating here are 50-year dose commitments; something that decays very quickly. Although the initial absorbed dose rate by the body is very high, it falls off very quickly.

And, of course, the other thing you have to factor into these calculations is the effective half-life within the body. So I don't know what the numbers are specifically, but --

MR. MARK: Well, curium -- plutonium lasts forever. It lasts for 30-odd-thousand days.

MR. PASHACK: But the body also has removable mechanisms, and I don't know what the difference between them is.

MR. HULMAN: Dr. Mark, I don't think that we have given you a satisfactory answer to your question. I propose to provide it to you.

MR. MARK: It could be that with your 50-year assumption the curium goes away and the plutonium, after five years, is dominant; for the first five years curium is dominant, and that may be long enough to be bad for the lungs.

MR. HULMAN: It very well may be, but I don't believe we've provided you with a satisfactory answer and I propose to give you one separately if that's acceptable.

MR. MARK: There are things in the literature where it is called to attention that there is a big dose factor. An article in 1975 in Health Reactor Safety, something like that. Certainly, the big radioactive thing for the first several years has a very small maximum permissible body burden and so on.

MR. HULMAN: Yes.

MR. MARK: So I would like to be clear on this, because I am just giving some numbers which may not be the important ones.

MR. HULMAN: We will provide that information to you.

MR. OKRENT: I have a question about a combination of things we have just heard. I just want to understand with regard to contentions 1A, 2, 3B-D, all of which are limited to feasibility of designing CRBR plants to make CDAs sufficiently improbable that they can be excluded from the DBA spectrum, according to the handout from the staff.

When you meet with the ASLB, how will you treat this term "sufficiently improbable"? Are you going to have a numerical number, or are you going to wave your hands, or what?

MR. MORRIS: This is Bill Morris of the NRC staff. We will not present a numerical value as a discriminator for that. We will base it upon the deterministic criteria and the feasibility of achieving a high reliability for those systems that are supposed to prevent severe accidents.

MR. AXTMANN: In the original historical doses proposed for thyroid, whole body for the LWR, most assumed a source term of some kind. Did that involve any chemistry involving water or were these assumed to be all of these -- the iodine and the plutonium -- to be simply airborne?

MR. HULMAN: Let me see if I can try and briefly answer your question by saying that the LWR source term history goes back to a footnote in Part 100 that refers to a technical information document, 14844, that did consider the chemistry as they understood it at that time.

But we wanted to try and envelope the type of activity that could be released. There was a substantial melting of the core.

Now, my understanding is they did not define what they meant by a substantial melting of the core. They did not define their considerations of the chemistry, nor the water that would be available in a wet containment. But there were considerations and, in fact, we believe that the iodine that we're using to represent the halogens was a surrogate of those elements whose volatilities were generally similar, and was the basis for selecting iodine at 50 percent.

MR. AXTMANN: Well, I am suggesting that simply moving the LWR source term over to the CRBR might violate assumptions on which the original LWR source term was generated in all ignorance of what we seem to be finding out the real source term is.

MR. HULMAN: Well, it may very well violate it, but if I also consider what is in NUREG-0771 and 0772, which indicate that the present lightwater reactor source term is probably conservative but we don't know how much, and if I consider what Ed Rumble has come up with for release fractions from his analysis of accident sequences for beyond design basis events, I think we still have an analogous source term. It may be conservative; it may not be, but it is analogous.

MR. AXTMANN: Well, it might be quite different if the original source term did involve water reactions. It didn't involve cesium, iodide and those sort of things.

MR. HULMAN: Yes, and it didn't involve the effects of tellurium and ribidium, that's correct.

MR. AXTMANN: I should think that would be worth looking into.

MR. SHEWMON: I think my personal feeling is that the water reactor source term, using that source term for water reactors is extremely conservative. On the other hand, if you go back and say look, we have 20 percent of these things coming out past sodium, all the noble gases are part of the particulate -- that seems to bound things. You can't really increase iodine much more than a factor of 5, and there's a very good chance that the sodium iodide is semi-stable, at least.

MR. AXTMANN: You may be quite right, unless they did take credit for the water atmosphere.

MR. SHEWMON: They don't. That was the earlier contention in this 701 and 702, or whatever the NUREGs were that he talked about, that often the water does -- or the cesium iodide would tie it up, that water would wash it down.

But there is good evidence that in a lot of accidents, there's a great deal of conservatism in that set of assumptions.

MR. AXTMANN: Right. Brand new chemistry was dimly perceived after TMI, but I am just wondering if other chemistry --

MR. OKRENT: Could I make a comment? It seems to me there are two kinds of source terms. There is the source term that is used in defining the requirements for the impact on containment leak rate, so forth and so on, and that is what is in 10 CFR Part 100, and that's what I understand was this source term. It is not related to a specific accident scenario, it is not related to what is functioning and what is not functioning and so forth.

Then there is -- I don't know what the term conservative means in connection with the source term. It was developed a long time ago with a different kind of reactor situation in mind. Small reactors in large containments, in effect.

MR. AXTMANN: That is what I am objecting to. Simply going from the lefthand column to the righthand column.

MR. OKRENT: This discussion of conservatism and the TMI accident and so forth, now you have to ask yourself what happens in various accident scenarios if you are going to follow them mechanistically. Then you have to follow the full spectrum, and you treat each event. Well, if you want to follow that route as accurately as you can with the uncertainties and presumably that is related to what is your estimate of the risk from the plant. You are not assuming the containment holds all the time now. If the containment is going to fail, it is going to fail, and some stuff is going to get out. If it holds, it will be a different situation. It is another set of analyses.

So, I must say, I would be inclined not to mix this NUREG-0771 one and 0772 with the discussion of the particular source term you are talking about. They need to have a source term in order to meet Part 100. There is a requirement they do a site suitability calculation, and frankly, I am somewhat more interested in the question of what is going to be the containment capability for a spectrum of accidents.

MR. SHEWMON: We each have our interests.

MR. OKRENT: No. But here you are examining what I have sometimes in writing called a ritual that Part 100 requires.

MR. SHEWMON: One other physical chemical fact. When these things come out, they are very likely to come out through a crevice, a floor, a small hole or something, and what gets taken out by aerosoles is very poorly treated these days, and is probably not at all conservatively treated, so that's something else in your back pocket, but we are not smart enough to handle it, so we don't, is my impression of it.

MR. HULMAN: I would agree with that.

MR. SHEWMON: Are there other questions?

MR. HULMAN: One question was raised at the subcommittee meeting on what is the foreign experience wih site suitability source terms. I believe the applicant is prepared to answer that one.

MR. CLARE: Good morning. I am George Clare. I am going to try to address two questions that were left over from our meeting with the subcommittee two weeks ago. First is a comparison of the Clinch River source term with that used in foreign countries. Second was a somewhat unrelated question relating to the off-site effects of sodium reaction products.

(Slide.)

MR. CLARE: Very briefly, what I have done is given you a vu-graph that compares the site suitability source term used in the United States for CRBRP with that used in the United Kingdom for the CDFR reactor and in Japan for the Monju reactor. You will note that the source terms that are released to the containment of these other two reactors in their site suitability studies are identical to that used in the United States, with the exception of Japan, where they have decreased the fraction of the halogens that are assumed to be in the containment.

I haven't indicated for the U.K. or Japan what might be considered airborne as compared to what might be considered to be plated out in the containment. We just do not know that information at this point in time.. The 50 percent used in the U.K. may be all airborne. There may be some plated out. In Japan, this 10 percent number may be all airborne, or may be in some part plated out.

For France and Germany, we are unaware of any equivalent of our site suitability source term. We do not believe the French use any major accident, hypothetical or otherwise, source term in their siting studies in any specific sense. In Germany, we know they do deterministic type studies similar to our studies of hypothetical core disruptive accidents. They have applie? some ietermination of acceptability of the risks from those accidents, but that evaluation is not done in terms of a site suitability evaluation the way these site suitability source terms are used in the United States, the U.K., and Japan.

Our overall conclusion from this is that the site suitability source term specified for the staff for CRBRP is comparable, favorably comparable to that used in siting foreign LMFBR's.

The other subject I want to address is the non-radiological effect of sodium reaction product aerosols. First, to briefly review what are the sodium reaction product aerosols that we are interested in. When sodium burns with oxygen, one gets sodium oxides of various sorts, NAO2, NA2O, various products. Those products can and do in fact combine with water, specifically water vapor and air very quickly, in a matter of seconds, to become sodium hydroxide. The sodium hydroxide can itself react with other elements in the air, other constituents of the air, specifically carbon dioxide, and in a time frame ranging from say a half minute to a few minutes, sodium hydroxide will convert to sodium carbonate, NA2 CO3, released in water.

Over a much longer period of time, the sodium carbonate can also be converted to sodium bicarbonate, taking again a hydrogen atom up from the water in the air.

Now, I want to emphasize that the effects of the sodium reaction product aerosols, radiological or otherwise, have been considered in the design of the plant, specifically in our environmental qualification program and in our control room habitability studies.

We are not asking you to review these details today. However, I bring them up here because it is important, and in these considerations we have reached a conclusion that it is necessary, it is prudent to provide aerosol medication features on the plant. Specifically, on our steam generator building where we can have significant quantities of sodium, non-radioactive sodium, in our intermediate heat transport system piping, and the pumps, et cetera, where, if we had a fire, it would leak from that pipe and a subsequent sodium fire, one can release a significant amount of aerosols to the environment.

We have specifically provided features, dampers, protectors, appropriate features to minimize the release of those aerosols while still allowing the pressure relief that is necessary to relieve the pressure that can be built up as a result of the sodium.

Also, although I failed to note it in writing on the vu-graph, we are considering the non-radiological effects of sodium fires, sodium reaction products in our emergency planning. That was done for FFTF. They have identified in the FFTF emergency procedures, the appropriate points in time at which various actions of the emergency plan would be taken.

Now, just to give you a feeling for the effects off-site, what I will call site boundary effects, of a major fire, I have tried to provide on the next vu-graph an outline of a fairly simple calculation that could be done to estimate that.

(Slide.)

MR. CLARE: First of all, if we assume that a design basis leak in our intermediate heat transport system piping were to take place in the steam generator building, it's a spray fire some 1000 gallons per minute of sodium burning in air, assume that 100 percent of the reaction products from the sodium oxidation would become airborne as sodium oxide, and then of course converting, as I indicated on the earlier vu-graph, and that the mitigation that takes place would only take place to the extent that we have designed our engineered safety feature mitigation system in the plant.

We evaluate that using the HAA3 code, which is a well validated code, for the calculation of the depletion of the fallout, the plateout of sodium oxide aerosols. It has been validated against tests at the containment test facility at the Hanford facility. It is a very large facility that shows very good comparison between the experimental evidence in the code.

What one finds is that we would release 440 pounds of sodium oxides in five minutes, over a period of five minutes. Combine that with site measured meteorology, which gives one a chi over Q of 1×10^{-3} seconds per meter cubed, taking into account the fact that there would be some settling of the aerosol over the roughly half-mile transit from the plant site to the site boundary, the fact that that would take several minutes to take place, we don't know how significant conversion of the sodium hydroxide to sodium carbonate, which is not a great irritant to the human body, one finds that we would have a site boundary concentration for this five-minute period of roughly seven milligrams of sodium hydroxide per cubic meter.

Our conclusion then is that the off-site concentration of sodium reaction product aerosols will be low. Seven milligrams per cubic meter is low in the sense that we have known various instances, for example in our experimental facilities, where there have been either intentional or accidental fires of sodium and other liquid metals, and the workers there have been exposed to concentrations as high as 50 to 100 milligrams per cubic meter, no effects other than some irritation of the eyes, the nose, and the throat, et cetera. When you do get up around 50 milligrams per cubic meter, the symptoms are such that the workers do want to lay down their tools and get out of the room, but there is certainly no permanent damage of any sort at concentrations as high as 50 milligrams per cubic meter. There have been tests done on animals by Battelle Laboratories, and they find that there is no tissue damage for concentrations as high as roughly 50 milligrams per cubic meter.

MR. EBERSOLE: Can these materials act as carriers of fission products?

MR. CLARE: We assume that whatever fission products might be distributed in sodium in the coolant can be carried by the sodium aerosols either by the aerosols or perhaps even separately, but in the same types of calculations that we do here, and in fact we calculate the off-site doses for all accidents where we could potentially release radioactivity with sodium in pipe leaks and that type of accident.

MR. OKRENT: Is the steam generator building seismic Class 1?

MR. CLARE: Yes, it is.

MR. OKRENT: And the whole secondary line, and the steam generators are seismic Class 1?

MR. CLARE: That is correct.

MR. SHEWMON: While he is thinking of another question, let me inquire what the project's policy is about having any water lines within containment.

MR. CLARE: We do have water lines within containment. We have strict criteria about how we separate those water lines from any cells where sodium is present. The water lines we have are there for equipment decontamination. We have a decontamination facility there. We would like to wash some of the radioactive sodium off of equipment. We also have some water for cooling.

MR. SHEWMON: In any of the accidents you are postulating or Mr. Rumble is postulating, the water lines have to be in the top half of the building and the sodium will all run into the bottom half of the building? I am looking more for criteria of that sort.

MR. CLARE: We don't have criteria that say that the water lines have to be in the top half or the bottom half. The criteria are along the line that one must have at least three barriers between sodium and water, and there are various subcriteria as to what those barriers can consist of. For example, one typically has two passive barriers and one active barrier.

MR. SHEWMON: Well, at a later time I would like to get into just whether, if we're getting into a core melt or containment or something, they would cope with that, which takes a pretty good material. Yet if you start having water and sodium together, you are going to transform the containment problems, as you realize at least as well as I do.

MR. CLARE: Yes. We have addressed those things, and we can go over that in the future.

MR. OKRENT: If we postulated that somehow a steam generator disintegrated and the building went with it, could you get lethal doses of sodium at the site boundary? I assume there is some quantity of sodium.

MR. CLARE: I have never studied what would happen if the steam generator disintegrated and the steam generator building and what effect that would have, so I can't give you a very firm answer to that. I think one of the major conservatisms in our analysis here are assumptions about the spray nature of the fire and the fact that we would release --

MR. OKRENT: I just wanted to know whether at the limit there are risks, let's say, of either death or very, very severe injury or not from sodium. You say you haven't studied it?

MR. CLARE: We haven't studied the type of accident specifically that you mentioned.

MR. OKRENT: You don't need an accident. You only have to postulate a release. I don't know how many milligrams per cubic meters it takes. Presumably it is something much greater than 50, but you haven't told me. Is it 500? Can you reach 500? Is it 5,000? Can you reach 5,000? It is just a benchmark which is of some interest when you are thinking about lower probability events.

MR. CLARE: Frankly, there has been so little problem with sodium, sodium fires, et cetera, say, OSHA or any of the typical bodies that deal with chemical hazards haven't specified, haven't come up with any kind of information that would help us define that.

MR. OKRENT: Do you have information on the event that occurred in Russia, how much sodium got into the air and how far it went?

MR. CLARE: No.

MR. RAY: Okay, we'll take a ten-minute break at this point.

(Whereupon, a brief recess was taken.)

MR. SHEWMON: Gentlemen, as is our wont, we have come up with things that have interested us and stretched out the early part of the agenda. To try to shorten this up, what I propose is that we treat the site suitability or the description of the site very briefly, since it is in the blue book, that we handle the seismology by questions, and that we get the full treatment on the hydrology, since the questions of dams and other such things is a question which has not been settled satisfactorily at the subcommittee meeting.

MR. STARK: This is Richard Stark. I was asked to summarize very quickly the findings of the staff on population and site location. I apologize to Henry Piper for trying to do his work for him. I point out that in the document of interest, the site suitability report, NUREG-0786, the population distribution is shown on Page III-I, and I will quote one sentence:

"Approximately 4,400 people reside within five miles of the Clinch River site in 1980." I will point out one other thing, that the site itself is located 25 miles away from Knoxville, which is, I guess, the closest large city. In addition to that, the document, Chapter 3 of the document describes the population and the distribution, and I will just leave it at that.

MR. SHEWMON: How far is it from Oak Ridge Labs?

MR. PIPER: It is four and one-half miler.

MR. SHEWMON: I guess I look at that as more of a plus, because you have a fair population who have heard of nuclear energy before, and might be of help in an accident. It was in that sense that I was interested.

MR. SHEWMON: We are ready now for the seismology and geology. Who is the speaker? An eminent seismologies.

MR. KNIGHT: This is Jim Knight from the staff. You had mentioned that we would do this by question. Might I suggest further, particularly after yesterday's session, when we had a similar discussion on Perry, the suggestion was made that perhaps this is a matter suitable to be brought before the Committee in and of itself.

Dr. Rockman is here. We are prepared to discuss what we think are the germane items that we could handle in 20 minutes, a discussion fully of the subject. I see eight to fifteen staff members for a day and a half or more to get into this type of thing.

MR. SHEWMON: Yes, I think that recurrence time and such things do come into the staff policy and is something we would like to have, but I think it would be best done at a subcommittee meeting instead of trying to have it now. Why don't we just handle this by questions, if we could? At least Dave said he had some questions he could start with.

MR. OKRENT: Well, as you are aware, there will be differences regarding the estimates of the probability of SSE excedence that will probably range somewhere between one in 2,000 and one in 10,000 per year type things. If we accept that this is likely to be the situation, and in fact it is from what I have already seen, how do you propose to assure yourself that the combination of SSE design and design basis and the actual design and the qualification of various things and so forth gives you an adequate degree of protection for this plant, which is not like an LWR in many ways, so you cannot draw fully on LWR experience on margins.

MR. KNIGHT: Yes, and the only answer that I can give you on behalf of the staff is that in our review to date, we do not see features of the plant that are, let's call them exquisitely sensitive. We have no structured program at this point that would call for some increase in seismic value, given some trigger level. The presumption, and I feel it is a sound one at this point, is that matters of structures, systems, basic components in the plants, we don't have a situation different from light-water reactors. We have specific design problems, and as the plant design evolves, we may well see something that requires attention and will have to be handled on an ad hoc basis.

That is the posture we are in.

MR. OKRENT: Well, you see, you are saying you are finding that the site is suitable for a reactor of this type, but I am not sure if you know the seismic characteristics of a reactor of this type, if you know, for example, is there a margin with regard to the lining on the concrete rooms or a margin on the guard vessels or whatever that is sufficiently large so that you have considerable structural capability to take an earthquake of lower probability but a higher degree of shaking than the proposed design basis, or that other things that may be vital to whatever equipment you need to run for a shutdown heat removal given an earthquake, that it will be such that there is the appropriate margin.

In fact, I do not even know that you expect to look to see whether there is additional margin. In fact, a member of the staff yesterday, in talking about Perry, I think indicated he did not see any point to looking at an iota beyond the SSE.

MR. SHEWMON: What is the question?

MR. OKRENT: They are making a finding on a reactor site suitability of a certain type, and I am trying to find out -- they won't give a number, but on the other hand we do have this range of estimates for the SSE which are like one in 2,000, one in 10,000 per year. Are you saying that is a sufficiently good number with regard to a major release? I do not think you are saying that. Even though you will not give a number, I am sure you will say that that number is not a good number. Therefore, you have to have some margin, and yet you can't --

MR. SHEWMON: The margin doesn't have to be the occurrence of the SSE.

MR. OKRENT: Exactly, but it is not clear that they have in mind to assure themselves of this margin or whether they in fact know that the reactor of this type is capable of providing the margin. I am not saying that this is undoable, but I haven't heard the staff say they are going to do it.

MR. SHEWMON: Jim, you have been over the hoops a couple of times before this committee on would you go back and look more closely at that particular plant and come back and see what has the smallest margin. You do this with certain codes and procedures. To what extent are there comparable codes, and are you sure that at least the applicant has designed to acceptable procedures so that you have a reasonable confidence that they are a comparable margin in the plant we are proposing putting here.

MR. KNIGHT: There are two facets to the answer to that question, the first being that yes, we believe that we have a totally analogous system from the standpoint of the codes that are in place, the engineering methodologies that are in place, and as I said earlier, at this juncture we are at a rather advanced concept, and looking at it from the standpoint of very probably a site suitability that there is every reason to believe that we will have the same type margin, the reinforced concrete structures. You have linings very analogous to what you have in fuel pools, the piping systems, and all of the usual type of things.

MR. SHEWMON: They are not very thick walled.

MR. KNIGHT: I beg your pardon?

MR. SHEWMON: The piping systems are not very thick walled in that area.

MR. KNIGHT: They are not thick walled, but there are design codes for them that must be adhered to.

MR. SIESS: The pressure doesn't dominate the piping load as much as it would for pressure piping.

MR. KNIGHT: Absolutely true.

MR. SIESS: Would the ratio of seismic stress to the total stress be significantly different?

MR. KNIGHT: I would have to believe it would be, yes. But again, I don't believe it is so different that you are thrown into a situation where you have an unknown.

MR. SHEWMON: That was the first question. You are dividing your answer into two parts.

MR. KNIGHT: Yes. The other is, and I think it is directly related to that, yes, there is a system in place, a recognized system by which you do these things, so that when you are done you know what those margins are. You can find them if you want. You can report them. You can understand them. I feel that there is a tendency to say, well, before the fact, let's significantly increase the seismic input and then go away and be satisfied that we have added lots of margin. I think that that is not really the way to go. If a distinct margin is desired and necessary, it ought to be, first of all, let's see what we have, because I believe the necessary premise for that is that we do have particularly in the major structures to start with very analogous situations, and see what we have, and if there are areas that are exquisitely sensitive, if there are places where the margins are too close for comfort, examine those on that basis, rather than an overall interest.

MR. OKRENT: I just wanted the subject discussed briefly to save time, and I think that is the issue that was most relevant.

MR. SHEWMON: Are there other seismic and geologic questions?

MR. SIESS: Jim, when I looked at the Zion PRA, I got some kind of a feel for the seismic margins. Do we have a PRA for the CRBR that would be comparable?

The question is, do you have a PRA that includes the seismic effects for the CRBR.

MR. MORRIS: A PRA will be performed.

MR. SIESS: With seismic effects?

MR. MORRIS: With seismic effects included. I don't know. I cannot state now that it will be a design PRA in that sense, but it will include seismic margin evaluation.

MR. SIESS: I guess Dr. Okrent's question could be stated -- I guess the question couldn't be stated, but he is asking, do you expect the results to look like the Zion results. Do you have the same kinds of margins you have in this that you would have had in an LWR, or am I misquoting you?

MR. OKRENT: I don't want to endorse the specific Zion results. That is my only reservation.

They claim some pretty seismic resistant systems that the staff's consultants have questioned, for example.

MR. SIESS: And we all reserve the right not to endorse the CRBR results.

MR. SHEWMON: But my impression was that what Jim said, he had every reason to believe from what the codes and procedures involved said that there would be comparable margins.

MR. SIESS: That's what he said, and I am saying a PRA with earthquakes in it would give another basis for judging that the margins are comparable in the two types of plants, and it would be a different kind of basis, because it would be systems related rather than stress or code related.

MR. SHEWMON: Yes.

MR. SIESS: When will you have that PRA? This will be performed by the project?

MR. MORRIS: Yes. I believe the PRA is scheduled to be completed somewhere between the issuance of the CP but prior to the final OL issuance. Therefore, it would be possible before granting an operating license to have evaluated this question of the margins for severe earthquakes and to determine what the situation would be at that time.

MR. CARBON: Bill, has the applicant committed to do the phase of the PRA that includes the siesmic aspect?

MR. MORRIS: Yes. We have a letter from the applicant in which he explains the way he will perform the PRA, and the sequence in which that will be performed, and those phases will be performed on the schedule appropriate for the evaluation between the CP and the OL.

MR. SHEWMON: Mr. Goeser, do you have any other comments?

MR. GOESER: We have indeed provided the plan. We have committed to do the seismic evaluation and the current schedule for completion of it is December of '84.

MR. SHEWMON: Thank you.

MR. GOESER: That is completion of the entire study.

MR. SHEWMON: Okay. Are we ready to go on to hydrology now?

MR. OKRENT: I wonder if the staff is ready to provide that answer they were going to take a half an hour or an hour to think about. Do we have a short answer? They said they had modified their position.

MR. MORRIS: Bill Morris, NRC staff. I have provided the excerpts from the previous subcommittee meeting, and I think that is what you have got in your hand there.

MR. OKRENT: That is the staff position?

MR. MORRIS: That is the interim staff position to replace the 24-hour criterion, but I must say that we are still evaluating that position. We are still looking at that.

MR. SHEWMON: Okay. Mr. Lee?

My agenda says Mr. Lee. This is the guy who stood up when I asked. Is my agenda wrong?

MR. LEE: I am Raymond Lee from TVA. I will give a quick review of what went into the determination of the design basis flood level at the CRBR.

We looked at two types of events; one, a rainfall flood, and then two, seismic-caused floods. For the sake of time, since the rainfall flood and the rainfall flood we used, -- and since the CRBRP is a nuclear plant, we used the probable maximum flood in determining the rainfall flood elevation -- and since the maximum elevation was 778 at the site, which is 26 feet below the controlling event or 27 feet below plant elevation 815, unless there are any questions I won't go into how we determined that.

But I would rather go on to the seismic-caused event which produced a controlling elevation.

In the seismic-caused flood, we looked at two types of conditions. One is an operational basis earthquake, coincident with a one-half PMF, and a safe shutdown earthquake, SSE, coincident with a 25-year flood. We looked at all dams upstream of the CRBR site, both individually and in groups, to determine the controlling situation.

It was found that the controlling situation in both cases was the failure of Norris Dam. The controlling elevation is given by the OBE failure of Norris.

MR. SHEWMON: Would you compare the one-half PMF and 25-year flood for me? Could you quantify that somehow?

(Slide.)

MR. LEE: Yes. The PMP or probable maximum precipitation is the rainfall that produces the probable maximum flood. At the CRBR site, it was a nine-day storm consisting of a three-day main storm amount of 17.2 inches, and a three-day antecedent storm, which is 40 percent of the three-day main storm, of 6.8 inches, for a total of 24 inches in 9 days in the watershed above Watts Bar Dam.

The drainage area above Watts Bar is 17,000 plus. So half the PMF would be half that value.

MR. SHEWMON: Of rainfall?

MR. LEE: Correct. So it is significantly greater than the 25-year flood.

MR. SIESS: Half the PMF is what? Is that the standard project flood?

MR. LEE: That is comparable to the standard project flood.

MR. SIESS: It was just arbitrarily taken as half as being comparable?

MR. LEE: We take half because as specified in Reg Guide 1.59, those are the two conditions we have to look at.

MR. SIESS: The Reg Guide calls the standard flood one-half the PMF?

MR. LEE: Yes.

MR. SHEWMON: I do happen to know that the 25-year maximum flood is appreciably different than that, but would you tell me what the 25-year flood is in inches?

MR. LEE: You can't relate rainfall necessarily to a frequency of flood, but I can say that a 25-year flood would occur on the average of once every 25 years, or have a chance of one.

MR. SHEWMON: Thank you. I will ask you the question again in a few minutes when you get to working with different units. Go ahead.

(Slide.)

MR. LEE: Looking at the elevations obtained from the three events, as I said, the PMF is non-controlling. The elevation of the PMF at the site upstream into the site is 777.5, the OBE failure of Norris coincident with the one-half PMF flood at the upstream end of the site elevation is 804.3, and the SSE failure with the 25-year flood is 796.3.

Those elevations compare with plant grade elevatio 815.

MR. OKRENT: Suppose you did the SEE failure with average flow conditions; how much different would that be?

MR. LEE: Not significantly different. It would be somewhat lower due to the fact that your 25-year flood has your reservoir levels at a high level. If you use a regular situation it would be lower.

MR. OKRENT: So it's primarily a matter of how much water you have in the reservoir and nothing else?

MR. LEE: Well, that's the biggest factor, right.

MR. OKRENT: Do you have high water in the reservoir for other reasons sometimes, other than because of flood conditions?

MR. LEE: No.

(Slide.)

Okay. To arrive at the controlling elevation, 804.3, which is the OBE failure of Norris, we postulated a failure mode, okay? We postulated that blocks 33 through 4? in the center of the dam -- that is the tall blocks -- we assumed they would fail, and we assumed they would be deposited downstream of the dam as shown.

The anlysis that went into determining this mode of failure will be addressed shortly by Tom Abraham. Using this mode, we had a model study made to study the discharge rating, and it didn't agree very well with our analytical solution but we used the lab model to be sure we had the correct discharge from this configuration.

MR. CARBON: The OBE corresponds to .125g, is that right?

MR. LEE: Yes.

MR. CARBON: You are saying here that this is your best estimate of what would happen to Norris Dam?

MR. LEE: That is our best judgment of what would happen.

MR. CARBON: With that magnitude.

MR. LEE: That is correct.

MR. SHEWMON: Does TVA have a seismic criteria for the design of their dams?

MR. LEE: I will have to defer that question to Mr. Abraham.

MR. ABRAHAM: Originally, it was analyzed for one-tenth, an acceleration equivalent to one-tenth of that. We made a re-analysis to measure for this nuclear plant. Let me say quickly that in the total analysis of Norris Dam, TVA does not feel that Norris Dam would fail in this kind of an earthquake.

We have done a conservative analysis on the information I will give you later, and we stand by the fact that in assessing all of our dams in which Norris came up, that Norris would not fail under the maximum earthquake that could be expected in the region.

However, in the interest of conservatism, in siting this nuclear plant, we took the position that we could not definitely prove it would not. Therefore, we took the most conservative position in postulating the failure. This was originally analyzed for one-tenth g seismic coefficient with no amplification of the base acceleration.

MR. SIESS: By that mean, you mean equivalent static?

MR. ABRAHAM: Yes.

MR. SIESS: And the analysis that leads to that picture is .125g on the same basis?

MR. ABRAHAM: That analysis is .09g.

MR. SIESS: I thought he said it was the OBE, and I thought somebody asked if the OBE was .125 and the answer was yes.

MR. SHEWMON: The OBE is for the nuclear plant, isn't it?

MR. LEE: Yes.

MR. SIESS: Is the OBE for the dam the same as the OBE for the nuclear plant?

MR. LEE: I don't know if we have the OBE for the dam as such.

MR. SIESS: Then I don't guess I understand anything. You're not analyzing the dam for the earthquake at Clinch River, at the site; the dam is somewhere else, isn't it?

MR. LEE: This postulation was from the analysis made for the CRBR. Isn't that right, T.J.?

MR. ABRAHAM: Yes, that is correct.

MR. SIESS: The figure is entitled "Analysis for OBE at one-half PMF assumed condition of dam after failure." Now, you can take it word by word, or somebody can tell me what it means.

MR. SHEWMON: Let me tell you what I think. What happened was the one-half OBE is defined at the plant, and you assumed that that same earthquake would occur at the dam, and that the dam would fail. Is that correct?

MR. HUNT: May name is Joe Hunt, I am with TVA engineering design. Let me clarify this a little bit, if I could. The analysis of Norris Dam for nuclear plants was originally done for Sequoyah and the Buford plant. At that time, the OBE was .09g.

MR. SIESS: The OBE for what, the dam or Sequoyah?

MR. HUNT: For the dam. We assumed the OBE occurs at the dam. Now, in the interim, of course, the OBE has increased. The failure of the dam would not be influenced by the difference between .09 and .125g.

MR. SIESS: That helps a lot.

MR. HUNT: The analysis of the dam considered that the 90g was amplified up through the dam, and Tom will get into that in his presentation.

MR. OKRENT: Do you have another figure that shows what your assumed failure mode is if you have the SSE?

MR. LEE: I don't have a figure but I can tell you what it is. It is the conditional blocks in this link becomes 833 feet instead of the 665 feet.

MR. CARBON: Say that again?

MR. LEE: This is for the OBE case. In the SEE case we have additional blocks that would fail, increasing the length from 665, as shown for the OBE, to 833. We are still with the same debris configuration downstream except it's a longer section of the dam that fails.

MR. CARBON: Simply lengthens.

MR. LEE: Right.

MR. OKRENT: And it doesn't fail to a greater depth. These are upper blocks, if I understand correctly.

MR. LEE: You mean it doesn't --

MR. OKRENT: In other words, --

MR. SIESS: If you look at the cross-section it looks like the whole dam moves out.

MR. LEE: It is the whole dam, isn't it, T.J., the whole dam within that block?

MR. ABRAHAM: It is assumed in that picture that the dam overturned in its entirety and laid back in that position.

Now, for the SSE, the length will increase because it is not as high as it goes up the abutment, and this improves the stress conditions. Therefore, we assumed for the SSE, since it's a higher earthquake, we took one more step up and took a couple more blocks on each side to increase that 665 length.

MR. OKRENT: Okay. So if you had an earthquake beyond the SSE it just would make it somewhat wider, is that what you're saying?

MR. ABRAHAM: Yes.

MR. CARBON: A question here. I sort of got an implication from what you said that when you went to the SSF you added a couple of blocks on each side. I presume what actually happened was that with the SSE level, that that is your best estimate as to what will happen? You didn't simply arbitrarily take a couple more blocks?

MR. ABRAHAM: We didn't make a further analysis. That was the judgment. The foundation doesn't look quite as smooth as the steps in the block, so we merely made a judgment that the SSE is higher stress, so we went up two levels on the estimated block elevation and picked those two blocks as being taken out.

MR. CARBON: Does that represent your best judgment, then, as to what would actually happen with the SSE?

MR. ABRAHAM: Yes. I want to point out that this kind of thing really doesn't lend itself to exact analyses. There are too many unknowns involved. We have done what we consider a very conservative analysis, and will hand you some information pointing out where I think the analysis is very conservative.

To do an exact analysis we feel is not productive and not realistic. We don't think it lends itself to this, so we did a onservative standard stability analysis and arrived at the conclusion of failure. Then from there, we did assume just two more levels going without doing further analysis.

MR. EBERSOLE: Could I ask you a question? The residual structure you've got there which has gone down to 665 feet doesn't strike me as being anywhere near as durable as the original one, yet it must be confronted with enormous hydrodynamic forces. How do you know it wasn't cut down lower than that, or swept further than that? What are the force balances that kept it where you've got it?

MR. ABRAHAM: The foundation itself. We know what the base of the structure is. Tell me again the rest of the question.

MR. EBERSOLE: You said the earthquake was gone; that's the reason it stopped there.

MR. ABRAHAM: Oh, you mean the debris?

MR. SIESS: What puts those blocks where they are, the earthquake or the water after the earthquake?

MR. ABRAHAM: That assumes the blocks overturned, and that is the position of the blocks after overturning.

MR. SIESS: And the overturning force is the lateral inertial force from the earthquake plus the water?

MR. ABRAHAM: Static plus dynamic loads.

MR. EBERSOLE: The water does not move the debris further because of no anchor, et cetera?

MR. ABRAHAM: These are matters of judgment. I don't think that anybody can conclusively say where the debris will collect with any type of failure. We don't have experience of this kind. As far as we know, we know of no dam that has ever overturned from seismic activity of this nature. We do have evidence that some have withstood forces much higher than this. And that, again, is one of the listings of the conservative approach we took to it.

We have taken the position that this is one logical collection of debris. There may be many others, but we take the position that the obstruction at the river will be approximately the same. I don't think mathematically, that could be calculated, so that is a judgment on the knowledge.

.MR. SHEWMON: We will hear from him again. Let's get on with it?

MR. LEE: I am finished, yes, unless you have other questions.

MR. SIESS: Is the last slide somebody else's?

MR. LEE: Yes.

MR. SHEWMON: Okay. Mr. Abraham, do you want to come up again, then?

MR. SIESS: Maybe it would help to know the disciplines of the speakers so that we could ask the right questions to the right people.

MR. SHEWMON: Mr. Lee, your responsibility or degree is in -- ?

MR. LEE: Hydrology.

MR. SIESS: The next one we're going to hear from is what?

MR. SHEWMON: I suspect he is a civil engineer, but I don't know.

MR. ABRAHAM: I am Tom Abraham and I am a civil engineer. I am a structural engineer, and my experience with TVA has been entirely in the design of hydroelectric structures.

MR. LONGENECKER: Mr. Chairman, I repeat again, we have four people from TVA; we have Ray Lee who is a hydrologist, you have Tom who is a civil engineer from the hydrostructural division; we also have available Joe Hunt who is in the geotechnical earthquake engineering staff, and Jim Domer who is Supervisor of BWR Licensing who has been involved in some of the Norris Dam failure calculations.

MR. SHEWMON: Do we also have Mr. Piper?

MR. LONGENECKER: Yes, we have those four people in those disciplines.

MR. SHEWMON: Thank you.

(Slide.)

MR. ABRAHAM: Again, let me start out by saying that our position with regard to dam safety is that Norris Dam would not fail under these conditions of earthquake. We have made the ultimate conservative assumption because, repeating myself, we cannot totally prove that it would not. So the proven course we took in regard to this is we would postulate a failure of Norris Dam.

MR. OKRENT: By "these conditions" you mean OBE 50 percent larger than the SSE? Could you be more specific?

MR. ABRAHAM: In dam safety analysis, we -- not in regard to nuclear plants, but in regard to dam safety itself, we take two earthquakes. One is the OBE and the other is the MCE, maximum credible earthquake. The maximum credible earthquake is .15g. In our judgment, it will withstand .15g.

MR. OKRENT: Okay. So your comment is made in the context of .15g.

MR. ABRAHAM: Yes. To give you a little background information, Norris is a gravity dam. It's approximately 1800 feet long. The approximate height is 265 feet. The dam is basically a solid concrete mass structure. It has an overflow spillway section, sluices and non-overflow sections on each side of the spillway.

The dam was completed in 1936. It was originally designed for an earthquake accelerat'on of 0.1g throughout its height. That means the rock acceleration wasn't amplified in that particular analysis, which is taken constantly throughout the height of the structure.

To ensure the safety of its dam, TVA does have a well-developed inspection and maintenance program with grade inspection at intervals to ensure that the dam is in good condition.

(Slide.)

Now, when the question of flooding at the plant came up, we took another look and made a new standard stability analysis of the structure. The chief difference between this analysis and the analysis that was made originally was the amplification of the basic acceleration.

This is a diagram shown in the spillway section. This is a non-overflow section. You can see from this diagram that actually the acceleration increased from .09 at the rock to as much as .75 g at the top, and .64 g at the non-overflow dam.

The results of the analysis are shown here. This is a base stress diagram for the two. Actually the base stress diagram indicated that the resultant forces would stay inside the base but would not overturn. But they were of such high value that we thought it was prudent to assume that it would fail. So that is another bit of conservatism that is in the analysis.

The stress was 1200 psi compression on the non-overflow section and 500 plus on the spillway section.

MR. OKRENT: How was this dam fixed to the rock, and is that a possible point of failure in an earthquake, in your opinion? I am talking now not about .09 g but .25 g or .35 g or earthquakes beyond your

.15 g credible earthquake.

MR. ABRAHAM: Sliding is always an important factor to review. This dam was built on competent rock foundation. It was highly treated during its construction to take care of cavities and imperfections within the foundation. This was well keyed in to the foundation and the foundation is rather rugged. So in order to get the sliding, it would be very difficult. It is almost like a keying action into the foundation in this case.

MR. OKRENT: So you would guess, if I can put words in your mouth, that this is a much less likely failure mode than overturning?

MR. ABRAHAM: We did consider it in that regard and that is the conclusion we arrived at.

MR. OKRENT: For the more severe earthquake as well as for the one you analyzed?

MR. ABRAHAM: Yes.

[Slide.]

I thought I would point out some conservatisms in this analysis. This analysis of using an instantaneous picture, if you will, of the dam at some point in which acceleration is maximum, you photograph the dam and apply these forces as a static stability normally referred to as a pseudo-static method of analysis, was used. Actually it does not work that way.

This is not a sustained load, it is an oscillating force and load. Consequently, it is not sustained long enough to develop the stresses that I just showed you.

The second area of conservatism in the analysis, in amplifying the rock acceleration, we went through actually a simplified dynamic analysis to arrive at the moments and shears in the amplification of the load. Now, if you remember the diagram, these amplifications of all portions of the structure did not occur simultaneously, they occurred at different periods during the earthquake, but we put the maximums all together to arrive at this base acceleration.

We assume the concrete had no tensile ability, and we think that we used conservative judgment in assessing the stress analysis after the final stability analysis was made. Some could have assumed that since the resultant forces did not fall outside the base, it could have been asssumed to be all right. In fact, some Federal agencies now have that criteria.

I mentioned to you that our dam safety assessment presumes that Norris would not fail under as much as .15 g acceleration. I think an important factor here that enters into our judgment on all gravity dams is we do not know of a gravity dam that has failed because of earthquake. We do know of at least one that was subject to an earthquake much higher than the design earthquake here at Norris, a dam in India called Konya dam. It cracked but it did not fail. Its accelerations were at least twice as much as we did here at Norris.

Now, to take the analysis a bit further, we took the Konya the dam, we made an analysis with Konya using our conservative approach, our conservative analysis. That indicated that the Konya dam under those conditions would have been in much worse shape than Norris would have been in, but nonetheless it still didn't fail.

MR. SIESS: Why is it conservative to assume the spillway gates open?

MR. ABRHAM? I don't think it makes a whole lot of difference whether they are open or not.

MR. SIESS: If they are open, does the water level go up to 861?

MR. ABRAHAM: This condition that was critical was the one-half PMF plus the OBE. At that time the spillgates must be open.

MR. SIESS: Would you go back to that previous figure?

[Slide]

MR. ABRAHAM: The stability?

MR. SIESS: Yes. I guess I don't understand why, if the water is at 1036, the hydrodynamic force goes up to 1061.

MR. ABRAHAM: This is not the hydrodynamic force. This is the mass of the concrete that appears in between the spillway stress.

MR. SIESS: Then the label is wrong, right? That figure is labeled "Hydrodynamic Force."

MR. ABRAHAM: Well, you do have some on the pier. Excuse me.

MR. SIESS: You said in the --

MR. ABRAHAM: A further clarification. Let me clarify that a bit more. This is the design of the mass acceleration. From the crest on up there are piers in between that take this part of the acceleration. Now, the dynamic force of the water resulting from being accelerated into the reservoir itself is this diagram, and then this is the static (indicating).

MR. SIESS: Okay. You said under your conservatism that there was not sufficient energy to overturn it. When I look at that slide, it doesn't look to me like there is enough force to overturn it. You have a factor of safety of overturning of 103 on the overturn section? On your assumption, it wouldn't overturn?

MR. ABRAHAM: That is correct.

MR. SIESS: What would happen if you sheared off the toe?

MR. ABRAHAM: We made an analysis of the shears on the planes at the toe in both cases. In those cases the shear was high, and that led us to further assuming that the dam would fail rather than stay there under this analysis. Remember, now, that these blocks were assumed to fail.

MR. SIESS: Is that figure on here, vertical plane 8A, 535 psi? Over on the bottom right of the boxes. Is that the shear we are talking about?

MR. ABRAHAM: That's correct.

MR. SIESS: You assume that you take off the shear and that is how it would overturn?

MR. ABRAHAM: Yes. Again, that is a rather conservative assumption. Some of the Federal agencies have a criteria for this type of analysis under these conditions that if the resultant shear falls within the base itself --

MR. SIESS: Is that a pure shear?

MR. ABRAHAM: Yes, sir.

(Slide)

I will flash this one back up. This is a presumption of failure at grade level. I think we have covered it in the approach we have taken, unless there are some other questions about it.

MR. OKRENT: The forces of the water rushing by, are they insufficient to move debris of this type in whatever form it might be given, that it failed in an earthquake or what?

MR. ABRAHAM: My judgment would be that it would be relative to the amount or size of the debris itself. I would judge that if it stayed in blocks as large as this, you would not develop enough force to move it significantly downstream. Of course, you cannot prove that it won't break up into other pieces. We have addressed that and we don't think it is possible to really tell where the debris is. We do feel there will be obstruction in the channel, and perhaps it would be about the same amount of obstruction.

Now, we did go a little bit further in making some sensitivity tests, if you will, in which other amounts of debris were taken out. In other words, if the crests were lower or if more of the blocks were totally taken out and we did some flood routing based on that. Raymond might want to address that for a minute.

I think the conclusion, Raymond, was that it would still not get over plant grade. Is that so?

MR. EBERSOLE: If you brought 970 down foot by foot by foot, at what point would you cover plant grade?

MR. ABRAHAM: I don't think we made that analysis, Jess. I think he flashed up that if you evaporated the dam, if you figure on that, if you totally evaported the dam. Did you give that figure?

MR. LEE: Yes. We looked at so-called sensitivity runs. I don't like to call it that, but if you can think of anything better, we will. Where we had problems was that we instantly vanished the entire dam. At the upstream end of the project we have elevation 818 at mile 18. At mile 16 downstream of the site, elevation was 811. All other runs gave us elevations lower than plant elevation 815.

MR. OKRENT: This is if you instantly removed the dam and there is no obstruction from it?

MR. LEE: No debris. It just evaporates.

MR. SIESS: I think it would be worth putting up. It was the slide that was handed out to us, gentlemen.

MR. SHEWMON: Why don't you tell us what it is?

MR. SIESS: It says "Sensitivity Runs, Postulated Failure Mode." It gives an elevation of water to the site for four different assumptions regarding --

MR. SHEWMON: It is in the first handout we had on hydrology.

MR. SIESS: Yes.

Now, on that slide it says OBE conditions with one-half PMF. Would it be any different if it was a PMF? There couldn't be any more water behind the dam, would there?

MR. LEE: It would be different if it was a PMF. However, that would be a combination that would be incredible and we don't deal with that.

MR. SIESS: Can you explain briefly why the water from the dam is different, whether it came from the PMF or half the PMF?

MR. LEE: Okay. The water --

MR. SIESS: The downstream?

MR. LEE: The elevation at the dam for PMF would be higher than it would for the one-half PMF because your control volumes are higher.

MR. SIESS: The elevation behind the dam is at the spillway level.

MR. LEE: The elevation was one-half PMF. I believe it was 1036.

MR. SIESS: Yes, and that is the spillway level, isn't it?

MR. LEE: It is overtop the spillway.

MR. SIESS: Oh, I see. It is some couple of feet over the spillway, isn't it.

MR. LEE: Yes.

MR. SHEWMON: Can we move on on this?

MR. SIESS: If the worst case you can think of with the combination of OBE at half the PMF would give you 3 feet above the plant grade at mile 18, what would it give you at the site?

MR. LEE: Excuse me?

MR. SIESS: The distinction between mile 18 and mile 16 as opposed to what it would be at the site. Does that make any difference?

MR. PIPER: Dr. Siess, my name is Henry Piper. You are looking at the map of the site on the river where the plant is located. The river mile at which we would take our intake, which is the nearest river mile to the plant there, is river mile 18,. River mile 16 is at our discharge point and would have an elevation of 811. Mile 18 would have an elevation of 818. There is a grade there of 7 feet in that space of the river for approximately a couple hours while the peak level of the flood passes the site.

Now, on one side it would be 818, on the other, 811.

MR. SIESS: What would it be at the plant?

MR. PIPER: We didn't make that specific calculation, but you will notice that there is a valley that runs rather at a 45 degree angle of the page to the right there, and that valley is open to the plant grade at 815. So there is not a higher structure than 815 between the river at river mile 18 and the plant itself.

MR. SIESS: So it would tend to flow down the valley, around the plant, into the river on the other side.

MR. PIPER: Yes, because of the gradients there. We may not be able to accomplish that.

MR. SIESS: I assume the plant grade was established before these calculations were made?

MR. PIPER: No, it wasn't. The plant grade was established based on what we consider to be a very conservative analysis of the combination of floods required in 1.59.

MR. SIESS: Thank you.

MR. ABRAHAM: I think with respect to that I will make maybe one other comment. This partial failure with accumulation of debris was also done in connection with other dams in studying the Sequoyah and Watts Bar sites, too.

MR. SHEWMON: Let me bring up a different question. Is it clear whether -- let's say this question does come up within ten feet of grade. The plant is designed as a matter of course to cope well with that sort of thing?

MR. PIPER: I'm not sure I understand your question, Dr. Shewmon.

MR. SHEWMON: We are talking about various credible and incredible floods. Not being a civil engineer and structural and all those good things, I struggle along not quite knowing what happens if we do get up to 85 feet.

MR. PIPER: 85 is fine. Our plant grade is set at 815. That means that all of the essential buildings are protected against the flood up to a level of 815. Above 815 --

MR. SHEWMON: You start worrying about pumps and other things.

MR. PIPER: Yes, and auxiliary-type buildings. The water would not intrude into the containment even if the flood got well above 815.

MR. SHEWMON: Do we have enough information?

MR. SIESS: Would you put this slide back up that shows the chunks of concrete downstream? The calculations made are for essentially overturning of the 665-foot width with the debris of the 945 level. Now, that is a considerable lower level than that sketch shows, am I correct?

MR. ABRAHAM: Yes.

MR. SIESS: That shows 970. That would be a little lower of that. Then you made a calculation with somewhat less width, 370-foot width, with the debris down at the 925 level. That is still lower, right?

MR. ABRAHAM: Raymond, I think you will have to answer that.

MR. SIESS: It is really the height of the debris, isn't it?

MR. ABRAHAM: This was the assessment given to the hydrologist to route the flood. From there he said he didn't like to use the term "sensitivity," but I can't think of a better one right now to look at the debris. He did some changing of this to make it worse. He made these assumptions of first going to a lower level for the crest, assuming rather than it falling in this fashion, t would break up in some fashion, at which this crest is --

MR. SIESS: What I don't see on this sensitivity list is what it would be for the case that is shown on that slide. That is, a 665-foot width with the debris at the 970 level.

MR. LEE: That is the design basis flood

level, elevation 804.3. It is not shown on that slide but it is shown on the --

MR. SIESS: 804.3 at mile 18?

MR. LEE: And mile 17. That is the design basis flood level as based on this postulated failure.

MR. SIESS: 665 by 970. Okay, thank you.

MR. CARBON: I want to ask the Staff, are you completely happy with everything that has been said?

MR. LEAR: I am George Lear, the Chief of the Hydrologic Engineering Branch.

Based upon our earlier review at the beginning of this analysis years ago and also based upon our recent review and analysis by the geotechnical engineers and hydrology engineers and the structural people, the applicant has presented a conservative analysis and we have at this time no further concerns about his approach. We feel it is a logical and acceptable approach at this moment.

MR. SIESS: Mr. Chairman, could I ask one more question?

MR. SHEWMON: Yes.

MR. SIESS: If Norris Dam failed in this manner, and I believe I saw somewhere Watts Bar would also fail and something else, what would be the consequences other than possible damage to CRBR?

MR. SHEWMON: Mr. Abraham, do you want to answer that?

MR. LEE: What do you mean? What kind of damage?

MR. SIESS: To people, or towns.

MR. SHEWMON: Would Chattanooga be under water?

MR. LEE: It would be catastrophic, yes.

MR. SIESS: What would happen to Oak Ridge and some of its plants there that make certain things?

MR. LEE: Oak Ridge. I think we have addressed that. Fringes might be inundated but not significantly.

MR. EBERSOLE: Your analysis assumed that Watts Bar would fail, but I assume that is making no difference to the CRBR whether it failed or not.

MR. LEE: That is correct. It makes no difference to the CRBR.

MR. EBERSOLE: Wouldn't there, in fact, be a cascade failure of Watts Bar and then Chicuamagua, et cetera.?

MR. LEE: Yes.

MR. EBERSOLE: How far down would that go?

MR. LEE: We didn't carry the failure of Norris down that far, but in looking at some of our other nuclear plant sitings, Cascades, the other would cascade on down through Gunnersville, failing the earth embankments as it went.

MR. EBERSOLE: The earth embankments.

MR. SIESS: And hope the Kentucky dam holds.

MR. LEE: We think it will.

MR. ABRAHAM: I think it is important, though, just to make a comment on that, that we are going independently on our dam safety analysis. We are taking a much more conservative approach on nuclear siting. On our dam safety program we have assessed all of our 40 dams, with attention to seismic analysis and so on, and Norris is considered to be safe against normal expected MCE earthquakes.

MR. SIESS: What would Knoxville and the City of Oak Ridge see? What about Knoxville? Somebody mentioned Chattanooga, but Knoxville is closer to the site.

MR. LEE: Knoxville doesn't have the same problem as Chattanooga because it is higher, on high ground. The waterfront property would probably be wiped out.

MR. SIESS: But you wouldn't use the word "catastrophe" for that?

MR. LEE: For a PMF flood of that magnitude I would, yes.

MR. SHEWMON: Do we need to hear about core melt?

MR. STARK: I guess the Applicant can pass out his presentation and you can consider that.

MR. SHEWMON: If that is satisfactory, that would be fine. Okay.

The question is, I guess, can we write a letter at this meeting. Is there anybody who feels they cannot? If so, would they speak up and identify their concern?

[No response.]

Apparently we will write a letter at this meeting.

MR. SHEWMON: Thank you very much.

Why don't we take a couple minute break while they clear the room and then get back.

MR. SHEWMON: Okay. The next item is disposal of high level wastes. Fire when ready.

MR. AXTMANN: In the past 18 months, the ACRS has written three letters on the subject of high level waste and the disposal of same in geologic repositories. The first rather short one in December, 1980, expressed the committee's confidence that a 1,000-year waste package could be evolved, and the geologic barriers could provide tens of thousands of years of isolation.

About a year later, September, 1981, we sent a letter that offered generally favorable comments on the staff's draft of 10 CFR 60, but provided 13 specific comments, including six that have been solicited by the staff. Among the committee's suggestions at that time were the inclusion of the retrievability requirement as a part of the rule rather than as background material, elimination of design and construction criteria from the rule, permitting the licensee to meet an overall safety goal without requiring separate subsystem goals such as the package, the backfill, and so forth, beginning early work on the evaluation and comparison of the various computer models for the reservoir, and relegating the regulation of transuranic waste to a separate document.

I will return to these subjects in a moment.

A third letter on the subject was issued in March of this year. The Commission requested that the subcommittee offer its advice on the staff's choice between only two contractors who had bid on a contract to verify the longevity of the so-called 1,000 year waste containment package. The subcommittee endorsed the staff's choice with what I recall as neither enthusiasm nor unanimity.

The letter itself expressed concerns that the letter we wrote after that endorsement of one of the two contractors, that we expressed concerns that the program was aimed at extraordinarily high standards. In the past six months, the subcommittee has met four times to consider various aspects of the disposal of high level waste. Last December, we considered the domestic and foreign approaches to the overall problem. In January, we met to review the staff's program at that point. In February, we considered the matter of the advice on the verification contract, which I have already mentioned, and earlier this month we considered the latest draft of 10 CFR 60, which I think is -- Mike, is this the only packet?

MR. BELL: That is the only one that has been made available to the ACRS. The Commission paper, the paper to be transmitted to the Commission, is in the

EEO's office, and apparently has not come down yet.

MR. AXTMANN: This is marginally different from what you have in your handout, I believe. The handout has the original -- excuse me. The last version has the changes X'd out so you can still see them in the new verbage typed in. It is that subject which is the subject of today's discussion.

As a result of that meeting, Dade Moeller drafted a letter which has just been passed out. It is the pink sheet. Absent any questions, I don't know, Mr. Chairman. Do we read the letter now, or do you want to postpone that until tomorrow?

MR. SHEWMON: What is your pleasure? Well, we have got some time on the agenda, if that is the way this thing goes. I would just as soon have a first reading, and let's get the reaction that way.

(Whereupon, at 12:20 p.m., the Committee went into executive session.)

AFTERNOON SESSION

(2:00 p.m.)

MR. RAY: The meeting will please come to order.

Paul will be a little bit late getting back.

According to the agenda the next item on the program is the decay heat removal systems.

It's all yours, Dave.

MR. WARD: This afternoon we will hear a report from the staff on their revision to Task Action Plan A-45, which is concerned with requirements to improve the reliability of decay heat removal systems.

We have heard over the past year several reports on drafts of the plans. More recently, earlier this year the plan was fairly extensively revised and simplified in some ways. Parts of it were cut out, and you have had two or three months' delay for one reason or another in hearing a report on this revised plan; but we will hear today that the plan has been officially approved within the staff and is issued and the work is beginning on the plan.

I don't think there's any particular requirement for us to write a letter to endorse the plan, for example. If we have some problems with it, we might want to write a letter. So at the end of the presentation today I will solicit your opinions on what you think we should do about it. We have not held a subcommittee meeting on this scaled down revised plan.

Unless anybody else has any comments, I will turn the meeting over to the staff; and I believe Andy Marchese will do the presentation.

(Slide.)

MR. MARCHESE: Good afternoon, everyone.

For those of you who don't know me, my name is Andrew Marchese, and I am the task manager for ME-45 which is the unresolved safety issue entitled, "Shutdown decay heat removal requirements."

(Slide.)

This is an outline of the topics that I would like to cover today. However, before I start let me just say that I think the most important thing I have to say today is that we finally have gotten the work started on this program after an extensive planning session that has lasted about 15 months; so I am happy to finally be able to come down here and tell you that we've gotten the work started rather than telling you about planning exercises.

MR. RAY: What does "start" mean? Who is starting where?

MR. MARCHESE: I'm going to get into that.

Okay. Some of this now both the subcommittee and the full committee has heard before, but since it's been sometime since we last met, in order to establish some sense of continuity, I want to briefly touch on the overall purpose objective and then get into the background on A-45 and give you an update since last time we met, which I think was last September, and then get into the main elements of the revised plan such that you understand what is in our currently approved plan, and also touch upon what we have taken out of the previous plan and the reasons for that.

(Slide.)

The overall purpose of A-45 is to evaluate the adequacy of current licensing design requirements to ensure that nuclear power plants do not pose an unacceptable risk due to failure to remove shutdown decay heat. That is a very broadly stated purpose.

(Slide.)

The overall objects are shown here. First is to develop a comprehensive and consistent set of decay heat removal requirements for both existing and future lightwater reactors.

Secondly, we are going to evaluate alternative means of decay heat removal, including diverse dedicated systems, to deal with the broad inspection of transient and accident situations.

(Slide.)

Now, to go over some of the history of the program, it is shown here. The Commissioners approved this issue as an unresolved safety issue back in December of -- actually that should be 1980. I was assigned shortly thereafter in February to start developing the Task Action Plan.

These next couple of items show the documents that were published right after the issue was declared a USI. I think the most important one was the draft Task Action Plan was issued in early May and provided a basis of starting the dialogue with the ACRS subcommittee. So we met in the May to December time frame of 1981 I would say four times with the subcommittee and a couple of times with the full committee. We revised the plan in October based on comments we received both from the subcommittee, and also an internal NRC review was conducted through the May to December time period.

This May 22nd we went through a revision of the plan, and I will talk about that in some detail.

(Slide.)

Okay. Task Action Plan Revision 0 was originally approved by the director of the division of safety technology on October 7, 1981. If you remember, this plan authorized a four-year program with a completion date of October 1985. This program was submitted to the director of NRR, and it was not approved, the main reasons being that there was a feeling that the program was too expensive, taking too long of a period, and that there was work described in the program -- in the plan that industry would be in a better position to take on. And so we basically went back to the drawing boards and considered at least a dozen to 18 different options of the October version of the plan.

In about February-March of the 1982 time period we got an agreement on an option that we felt we can go forward with in terms of still meeting our primary goals. So we assessed this program to determine if the primary goals could be realized and with a shorter schedule. We've now come up with a 30-month program that we feel will meet our primary goals. We are now estimating that a draft NUREG report which will contain our proposed recommendations, including any proposed new requirements, along with the supporting technical and cost-benefit basis, will be available by about November of 1984.

(Slide.)

Continuing with the update in terms of how we achieved a reduced schedule from 48 months down to 30 months, the main reasons are stated here. We have deleted most of the work on future plants, although acceptance criteria for decay heat removal systems for future plants will be developed.

If you will remember, in the previous plan we were not only going to look at alternative systems for existing plants, but also we were going to look at future plants. I am sure, as you are aware, the alternatives could be considerably different because the problems you have with existing plants from the standpoint of backfitting are considerable, and the choice of system alternatives could be entirely different for existing ones where you're talking about retrofitting versus future. So we did take out the work concerning alternatives for future plants. However, concerning acceptance criteria we are developing a set of criteria that will be applicable to both existing and future plants.

Quantitative acceptance criteria, our criteria now will be a range that will stem from both qualitative to quantitative. They are criteria that kind of fall in between.

The quantitative criteria now will be based on frequency of core melt due to decay heat removal system failures rather than overall risk. We were originally intending to go forward on an overall risk based approach. We have backed off from that.

To simplify things, we feel that the performance of the decay heat removal systems is more tied to core melt frequency. As you know, when you go from core melt frequency to overall risk, uncertainties propagate considerably in terms of having to get into consequence, in terms of whether the containment remains intact or fails, getting into site meteorology and all those things that we're not now going to get into.

So our goal, at least the quantitative goal, will be based on core melt frequency due to decay heat removal failures.

MR. OKRENT: Excuse me. Does that mean that in setting acceptance criteria for a PWR it won't matter, for example, whether it's an ice condenser or a large dry containment; you'll have the same criteria?

MR. MARCHESE: In terms of the criteria for decay heat removal systems that is true.

MR. OKRENT: You may want to think about the connotations of the question and answer. I don't want to try to solve the problem here but --

MR. RAY: Are the components of the decay heat removal system influencing the degree of failure?

MR. MARCHESE: We are trying to avoid having to include containment heat removal systems unless there is an interaction. The systems we will be concentrating on will be dealing with the frequent events, transient, small break LOCA spectrum. We're not going to get into large break LOCA, and we're trying to avoid having to avoid all of the systems where you're talking about getting into containment heat removal aspects. So we have narrowed the program down in that respect.

MR. RAY: You really don't know what it will show you until you get into this thing I suspect really.

MR. MARCHESE: Yes. I could think of one example like the feed-and-bleed system where there is an interaction with containment. You're dumping heat into the containment, and certainly feed-and-bleed looks like an attractive alternative.

MR. EBERSOLE: Well, the design of the containments, however, will reflect the probability of core melt since some of the scenarios first lead to containment failure and then proceed to a core melt after that. Typical is the boiler which has this internal cooling problem since it can't evaporate to atmosphere in the current models.

MR. MARCHESE: Well, for the BWR suppression pool cooling we're going to focus on that.

MR. EBERSOLE: That's what I really meant.

MR. MARCHESE: That's going to receive a considerable amount of attention.

MR. OKRENT: One other question. Does your schedule include having obtained from industry the plant-specific evaluations of alternative DHRS's indicated in your third bulletin? In other words, will those have occurred by November '84 when you will have your report, or will they follow your report?

MR. MARCHESE: They will follow. We feel it's going to take to about that time period where we are in a position to tell industry which plants are unacceptable and where we know they are improving their decay heat removal systems that will make them acceptable, and we will also know the systems where we feel they are attractive and have a good cost-benefit ratio.

I would also add that we will in our program have an industry peer review, which I'll talk about a little more later, in which we intend to invite the industry in to review our interim reports and also get a dialogue going on what they may be doing in this area.

MR. OKRENT: The evaluations industry might do on a more plant-specific basis would be after November?

MR. MARCHESE: Yes.

MR. WARD: Andy, were those criteria against which you plan to publish in the NUREG?

MR. EBERSOLE: The GE people have proposed that to cope with this intrinsic weakness in their system that they propose to boil to atmosphere out of the suppression pool which brings the price of minor release of radioactivity for the benefit of avoiding core melt.

Will you get that far into extrapolating present designs into future ideas, or are you just going to stop off where they are right now? Are you going to include new concepts?

. MR. MARCHESE: We will include new concepts if they look attractive for existing plants. We will assess them and rank them based on value impact evaluations.

Dennis Barry from Sandia, who is the project manager on A-45, would like to add something.

MR. BARRY: With regard to Dave Okrent's comment, we are planning to do plant-specific evaluations relative to assessing the reliability of existing plant decay heat removal systems by applying screening criteria that are based on either probabilistically-based arguments from work that has been done in that area, phenomenological arguments based on things that have been identified with whether the task can perform the decay heat removal function, and with regard to screening criteria.

In addition to that, we plan for those plants that seem to have probabilistic-based analyses, we plan to use PRA techniques to do some qualitative assessments of realiability of existing systems in those plants that look like they have some weaknesses.

So with regard to this program, we are trying to get a bottom line and make some statement about how plants compare among themselves with regard to the reliability behavior of the systems.

In addition, as part of this program we are going to consider how we can change those vulnerabilities with alternative systems, tie them to specific plants, ones that do look like they have problems, and then form the value impact assessment on those specific plants. So we are doing plant-specific work. We are not waiting.

I believe what Andy was referring to here on the issue of plant-specific evaluations is that we don't plan to have a bottom line for one and only fixes for plants, but we do feel like we have to look at real plants and the alternative for those plants. It will then be up to industry to evaluate our findings and figure out whether or not they can do it better.

MR. EBERSOLE: Andy, when you look at some of the older plants you will recall that the first regulatory guide -- they used to call it something else years ago -- was on NPSH. It was intended to cope with the fact that there have been plants designed -- at least Browns Ferry was and I suspect there are others -- which had the terrible weakness of requiring retention of the atmospheric sufficiently in NPSH to keep the pumps running. If you lost air, you were in trouble.

So going back to these old plants I certainly would suggest you look at how many plants are like this and what the severity of that problem is, since I thought it was rather terrible.

MR. MARCHESE: The last major point we feel here that has reduced our time in this program, and I think also increased the probability that we will pull it off in 30 months, is that we now have recommended that we have one contractor that will have the overall responsibility for project management, the technical direction and integration, including selection and management of subcontractors.

We have made the recommendation, and it has been accepted that Sandia be the lead contractor. Sandia will be responsible for the overall technical lead in the program, and they are going through a subcontractor selection process right now, so they will manage those subcontractors, and we in turn would interface with Sandia all along.

(Slide.)

Steps that have been achieved to start work on the program, we have received the approval of Mr. Denton March 15th; we've received approval from our senior contract review board which was required because we are talking about expenditures in excess of $500,000 per year which requires this board's approval. We have implemented a contract with Sandia as the lead lab. They have begun the work, including preparing a detailed action plan of their own. And we have issued Revision 1 of TAP A-45 on June 2nd, 1982. Copies have been provided to the ACRS. So the latest version of the plan is a June 2nd plan that we're working to.

MR. WARD: Has the Sandia action plan been issued?

MR. MARCHESE: It will be issued next week. We have been having extensive discussions with them on it.

MR. RAY: Andy, sometime ago in the early stages of the first effort we had the impression that you were having trouble with getting staff management approval to assign resources to the project. What is the status of that?

MR. MARCHESE: The status as I know of it -- and maybe Carl Neal can elaborate on this -- it looks like it's becoming easier to get internal support. I think from what I hear, Mr. Denton has given his orders to division directors to start supporting USIs, that the case work load is perhaps decreasing and perhaps now we should start devoting more internal research to USI.

MR. RAY: So this project is still to be funded?

MR. MARCHESE: Well, one of the things I'm doing right now is putting together an internal plan that is assigning branches specific things I would like to have them do along with the schedule of doing that. So we're putting together an internal plan right now.

MR. RAY: And you expect a response then?

MR. MARCHESE: I expect to have a good internal team supporting me on this program because we do not want technical review branches getting involved at the end when we're publishing reports. They should be involved at the beginning. Helping us with the review of those reports should not be at the end of the program because they are the ones who will have to implement them on the plants.

Now, to get into the revised plan --

(Slide.)

-- This chart shows the main elements of the plan. It is broken down into four major tasks along with supporting subtasks. The first one is we are going to be developing acceptance criteria for assessment of decay heat removal systems. We are going to be developing means for improving decay heat removal systems. We're going to be assessing the adequacy of existing systems against our acceptance criteria. And finally, we will be developing a plan for implementation and actually publishing new requirements, if any, on a technical and cost-benefit basis.

This, I think, will show up a little more clearly on the first chart which shows the interrelationship of the tasks in the plan.

(Slide.)

Very early in the program we are starting out with developing both quantitative and qualitative criteria. The quantitative criteria, as I indicated, will be founded on frequency of core melt due to decay heat removal system failures.

What we're planning to do there is to allocate a portion of the overall core melt frequency goal -- for example, the 10^{-4} goal that's been suggested, if it finally winds up as being the Commission's goal -- to decay heat removal system failures, and then from that establishing reliability goals on a per demand basis for the different systems that are involved with the decay heat removal function.

We will also be developing qualitative criteria for those events which we have termed special emergencies. These are events which are difficult to quantify in a probabilistic sense -- such things as fire, flood, sabotage -- although there have been attempts to quantify anything more.

MR. OKRENT: What about seismic?

MR. MARCHESE: That will be included.

MR. OKRENT: How?

MR. MARCHESE: We do not have that fully developed. We are just starting that right now. But we intend to look at the question. If we say we recommend a dedicated system, what should be the seismic design level of that system; should it be beyond the SSE, and what does that gain you; what is the value impact of designing it for a stronger earthquake.

MR. OKRENT: How are you going to assess the current capability with regard to decay heat removal and dealing with small, small LOCAs which I understand is in your ballpark?

MR. MARCHESE: Right.

MR. OKRENT: As you know, the ACRS has several specific cases, including some recent one recommended that the staff consider lower probability earthquakes than the SSE and judging the adequacy in this regard.

Now, do you expect to tackle this problem head on and come up with an assessment of what the current status is of possible alternatives, or just what will you do if you are not going to tackle that option?

MR. MARCHESE: I don't believe -- Dennis can correct me -- I don't believe we're going to tackle that one head on, but I do think in terms of the seismic design criteria we will be looking at the question of whether or not our alternative systems, if we recommend them, should be designed to an earthquake level greater than the SSE, what the value impact of doing that is.

Do you agree with that?

MR. BARRY: First of all, I understand there's another program that is ongoing to look at spectrums of earthquakes beyond the SSE, is that correct, Andy?

MR. MARCHESE: Yes.

MR. BARRY: I don't know what the name of that program is.

MR. MARCHESE: There's one called the Seismic Structural Margins Review Program which we are going to be interfacing with that group to see what they come up with. I think we are going to be looking at, you know, what is the sensitivity of different component seismic design in terms of changes in risk and what the cost-benefit is of changing that. I think that's one of the things they're looking at.

MR. WARD: Does their program have a schedule which is compatible with your program using what they develop?

MR. MARCHESE: That program when I last talked to them was like on a five-year time frame.

MR. OKRENT: Are you talking about the SSMRP program? Don't bet on that one answering your specific question. I don't think you will find that specific question is a part of their program plan, assuming they meet their program plan. I do not know how you will evaluate the benefits of some possible new system with regard to augmented seismic capability if you don't know what the existing capability is.

MR. MARCHESE: They have these fragility analyses going on which I'm sure you're familiar with.

MR. OKRENT: I'm familiar with them, yes.

MR. BARRY: We have been somewhat reluctant to -- one of the aims of this program is not to be generic in nature. We are trying to come up with some assessment on as many plans as we can. Our goal is all plans concerning the reliability of decay heat removal.

One of the many ways that decay heat removal systems can be vulnerable is seismic. Seismic is one part of that. We feel that we cannot go into the level of detail developing fragility curves and determine the seismic signatures of every site and responses of the buildings that would come into play with the fragility curves of the individual components for all the sites.

The approach that we are planning to take is to draw upon the existing regulatory requirements that reflect the current state-of-the-art kind of requirements for seismic design. We plan to draw from those requirements what we consider to be the essence of the requirements, what is necessary to meet current SSE requirements, not going beyond SSE. In a probabilistic way we are not planning to do that.

We also plan to draw on the efforts that have been made in the Zion and Indian Point studies where an attempt has been made to quantify the contribution to core melt. We plan to cull from that those characteristics of the systems that contributed most to those core melt frequencies as a result of an earthquake. With that information we then plan to screen other plants and to look at the characteristics we feel is an indication that they would have seismic weaknesses; but we are not planning to try to quantify those seismic contributions to the core melt at all plants. I think that is beyond the scope of our effort.

Where those plants seem to have seismic weaknesses we will look at whether they meet the current guidelines for seismic events, again not going beyond the current guidelines.

The reason for us doing that is we feel we could for any number of special emergencies call upon this program to reassess the adequacy of the current guidelines for those emergencies. We can do it for fire, we can do it for sabotage, earthquakes, floods, hurricanes, what have you. I think an individual assessment of any one of those could be a program in itself.

MR. OKRENT: Well, I am not trying to talk you into doing anything at the moment. I am trying to ascertain what it is you will do and why it is acceptable, if it is, and whether it will meet your own stated goals.

Now, this vu-graph talks about assessing adequacy of decay heat removal systems in specific plants on a probabilistic basis. Those are not my words; those are your words. I don't see how you can assess the adequacy from the seismic point of view if all you do is look at was it designed to meet the SSE or not and so forth.

MR. BARRY: I agree with you, and that is why we are planning to draw upon the two attempts that have been made to put probabilities on the contribution to core melt from seismic events; but both of those efforts were in themselves extensive efforts. We feel we cannot do that on all plants obviously.

We instead are going to -- we feel those contributions to core melt, whether they be probabilistically assessed or deterministically assessed, those contributions can be assigned to particular plant weaknesses or characteristics such as a nonseismically designed structure or a nonseismically designed system.

But with regard to looking at SSEs or earthquakes that go beyond the SSE with a certain probability and assessing whether or not the probabilities of the systems change with the existing earthquake threat, we are not planning to do that. So the probabilistic assessment that you are talking about here, as I believe Andy has indicated, will be limited to those transient and LOCA events that have been traditionally quantified probabilistically. In addition, we plan to include up to what the state-of-the-art has tried to do with regard to things like fire and earthquake and hurricane where there has been some attempt to do that probabilistically.

But the probablistic assessments we're talking about here are to actually take on the order of a dozen safety analyses that have been done and modify them in a way that reflects the particular features of plants that we don't find to be weak, and to then reassess plants using a current state of the art risk assessment. Where those risk assessments have not probabilistically handled seismic events and other special emergency events, we are not planning to go off and do a site-specific analysis.

MR. OKRENT: Well, I think there is reason to be concerned that by the time you get to the stated end of this study, which is November 1984, that seismic risk studies are somewhat more common than they are now, and I don't know if they will be more believable. That is a separate issue. But if you are not at least having a considerable amount of thinking and effort, going in this direction may present a problem.

Fires, if I understand correctly, have been looked at for many of the existing plants as part of the ongoing reviews, and in some cases people have proposed alternate ways of accomplishing shutdown heat removal because that was simpler than changing the design with regard to mixed cables, et cetera. So you may there have some basis for examining the situation on a large number of plants. I just don't know how many.

MR. BARRY: The fire studies were not quantitative in terms of our probabilistic risk assessment. They were descriptive.

MR. OKRENT: That is true. Of course, again we now have a couple of fire risk estimates, also quite uncertain, maybe even more uncertainty. But there will be more of these that have been done by November '84.

So for those reasons, in the first place it seems to me you may have an incomplete basis on which to judge the adequacy of existing systems or the benefit or whatever, and also because of the fact that by then what looks now to be keeping up with the Joneses may not be keeping up with the Joneses any more. I think you may want to dwell on this aspect.

Let me give you one example that seems to me may warrant some thought. When we ourselves tried to look at specific reactors, casually, briefly, you know, an hour or two in discussion, it seems that there are some aspects of the plant where they have analyzed the structures and so forth and assessed the margins, but there are some aspects of the plants where are they can say is we have qualified it for the SSE, we don't know what it will take.

You can go to somebody's subjective estimate of the fragilities, but I am sure you can get other multiple estimates of fragility that would vary a lot.

MR. MARCHESE: Okay, Dr. Okrent. I think we will reflect on your comment and check on the present status of a number of programs that are kind of touching on the seismic issue and see if we can come up with better answer to your question.

MR. WARD: If there are indeed developments in the area that Dave is suggesting, between now and time for publishing the NUREG is there any particular problem in factoring those into the criteria you have established?

MR. MARCHESE: I think that is on a time scale that is consistent.

MR. WARD: I guess I find it hard to believe that the state of the art is going to be all that advanced in the next two years, but you were cautioning him to try to keep up with it as they are developed over the next two years?

MR. OKRENT: Yes. I am certainly not advising him to take those numbers and put them into their calculations. I think they have to have enough sophistication to be able to use what is being developed as a minimum.

MR. MARCHESE: Along with developing criteria, which is going to be done very early in the program, we are going to start of with these tasks here (indicating), which are aimed at developing improved means of decay heat removal.

This task is divided up into three subtasks. One concentrates on the phenomenological aspect, the second are the engineering aspects, and the third are the operational aspects. The phenomenological aspects will take a good look at all of the thermal hydraulics tests, information that is coming out of LOFT and Semiscale, in which they are looking at modes of heat transfer that involve natural convection and refluxing.

We are going to review that to see to what extent could we extrapolate that information to full-size systems for a range of plant configurations.

We feel that there is a lot to be learned there. We think if we can count on some of these modes of heat transfer, it would simplify the alternative means of decay heat removal considerably.

For example, it would lessen the need to have a large power supply to the alternative system. We will also be doing calculations using existing tools, such modes of heat removal as feed and bleed, which has been discussed quite extensively. This will also be done as part of this 2.1. The 2.2 concentrates on engineering aspects of alternative means of decay heat removal. We will be looking at a number of alternatives that have been described in the plan, and also doing the associated value impact associated with those alternatives.

The third item concentrates on the operational aspects. There is a rather substantial effort going on developing emergency operator guidelines, that is, developing the means for using decay heat removal systems in atypical ways, and we will be doing the procedures there to see if we can count on these in terms of improved means of removing decay heat.

This item will also have thermal hydraulic relationships, and the time for operator action is a critical parameter that needs to be looked at. How long does the operator have to take action? What is the probability of the operator erring in taking corrective action?

MR. EBERSOLE: Although it has been around for many years, it has suddenly come into the limelight and is being worked on intensively. The PTS problem. When you say feed and bleed anymore, you better say feed and bleed at reduced pressures and perhaps with a few other words, too.

MR. MARCHESE: There is an interface we have established with the people doing the pressurized thermal shock issue. That program, fortunately, is advanced more than ours and on schedule. They impose some system restraints.

MR. EBERSOLE: Maybe bigger valves.

MR. MARCHESE: Right. So we are working closely with them.

Now, when we get down into this area, what we are doing here is we are assessing plants against the quantitative criteria and against the criteria for special emergencies. That is these two blocks. In this block here we are trying to group plants to minimize the number of plants that we have to look at.

We are hoping that we can group plants, that is, those plants that can have a PRA for an IREP study performed on them. We are hoping to group all those other plants under those what we call parent plant. A parent plant is one that would have an IREP study performed and looking at other plants in terms of how their system configurations compare with a group of plants.

It is questionable how successful that will be because, as I think I have talked about before, there is considerable variability in decay heat removal systems frpm plant to plant, even with the same vendors. We are finding substantial variations. We have four, three and two-loop plants. We have some of the older plants where they use recirc configurations.

So to the extent we will be successful in doing that will be questionable. Eventually we would like to screen all the plants we look at against those criteria.

And then finally at this stage we will be in a position, having done the value impact assessment on a range of alternatives, we will be in a position to rank alternatives to ensure that they have a favorable value impact. We will know which plants do not meet our criteria. We will know that applying one or more of our criteria that would meet our specifications, and we would basically spell out the requirements along with knowing which plants are in a bad position with respect to decay heat removal.

MR. OKRENT: Is there a task or a subtask which involves ascertaining the decay heat removal system requirements for other countries that are using LWRs?

MR. MARCHESE: Good question.

[Slide]

We have included in this program, because this was also a comment that was in the ACRS letter, at the end of the sentence, back about last fall, in which the Staff as part of this program should take a hard look at the systems in certain countries that have gone to greater lengths, at least I believe, in their systems they employ for decay heat removal.

[Slide]

We intend to do that with the mechanism we are discussing right now. We intend to solicit information first from a number of countries. We are fairly familiar with the hardware that is being used. It is the basis behind the design decisions that we would like to get more of. We are intending to solicit that kind of information from those countries, get back and then arrange for visits to those countries to look at them in detail.

We are also planning on setting up an international seminar where we would invite a number of countries to publish papers and host a seminar in which we would publish the papers on this subject of decay heat removal. So that is what we are toying around with right now.

This shows the overall schedule for the program on each of the main subtasks of the program. I might add these were open areas here. This is where we had started work last year and then it stopped because we ran out of money after the program was not approved. That period represents the internal effort that we went through in terms of reconsidering a number of options that resulted in the revised program that I have just described.

We have started on developing the criteria. That work has started. We have some limited work that is going on grouping the plants. In about a month or two we are expecting to publish a report on the grouping effort.

The other work will get started a little later on because we needed to have some time for Sandia to get the subcontract work started, and the work will be starting a little later on. But there will be a number of interim reports that will be published on this program and we intend to provide those to the committee for your review and comment as well as eliciting comments from the Staff.

The Technical Review Branch. As I mentioned, we hope to get a team lined up to look at those internal reports, and also Sandia is going to be setting up what we call an industry peer review group. That is a group that will have representatives from the vendors, A&Es and utilities that have expertise in this area. They will come in and we can talk about the pros and cons as well as solicit their views.

I do feel that there are probably some efforts going on in industry that are not being widely publicized. I have had a lot of discussions with different people and I get the feeling that the utilities are thinking about this. We have also discussed this plan with EPRI, trying to get them involved in it. In fact, I might just talk about that a minute.

[Slide)

This was a major comment, I think, we got from the full committee, to encourage industry to get involved. We don't want to go down a three or four year program and develop requirements that industry was screaming about. We are trying to get them involved in the beginning.

So we have encouraged industry to get involved in A-45. We have asked them to consider a number of options. Setting up a parallel program in this area probably would be ideal, or perhaps doing specific parts of the Action Plan. This one here was a task we had deleted from the previous plan which had gone into a specific plant and looked at a specific alternative system for that plant. I think industry can do a better job in that area than we can. At the minimum we see setting up an industry peer review group.

In terms of priority for development of conceptual designs for improved decay heat removal systems for a specific plant, this will depend on a core melt frequency due to that plant and on the effectiveness of improvement of decay heat removal systems as a means of reducing that frequency and/or capability for handling special emergency situations.

So this was discussed with EPRI and they are thinking about it. Whether or not they take a lead is questionable.

We also presented this plan to an industry seminar that was hosted by NUS in which there were many utilities, vendors, and even foreign representatives there.

The point I am trying to make is a lot of people are aware of this plan, they know we are working on it, and I am sure they are not going to sit back while it goes on.

There are a number of backup slides in your package. I guess I will leave it up to you to what extent you want to look at it.

David, did you want me to cover anything?

MR. WARD: Well, we have ten more minutes.

MR. OKRENT: Can I ask a question?

MR. WARD: Yes.

MR. SHEWMON: The Chairman will take five if you run out of anything else to do.

MR. OKRENT: If you start using these various existing PRAs, one of the things that arises, what are the stated uncertainties of the PRAs? Then there is a next question: Does one agree with the stated uncertainties of the PRAs or is there some other assessment of them? And then there is a follow-on question: How do I come up with decisions in view of the assessment of the uncertainties, whatever it is?

Now, that is not an easy subject. Anybody working in the area will be, in a sense, breaking ground, but if you don't deal with it at all or deal with it superficially, instead of breaking ground it may be the way people bury you.

I was wondering if you have some kind of approach or is it an identified task or just what?

MR. MARCHESE: You may want to briefly describe the approach you are going to take in putting the system logic on the computer. I think that may help.

MR. BERRY: I agree. Among the uncertainties are regarding completeness and accuracy. There are different numbers that have been used in different PRAs. There are different levels of completeness that have been done in different PRAs. Accident sequences have evolved, some of which appear to be important.

What we plan to do is use completed work not so much off the shelf but instead to draw upon that completed work to define criteria for systems in the power plants for different accident events, and use that existing work to define the components that are called upon to meet the accident situations.

We plan to use this information to define the logic models for how the complete PRAs can be represented in a consistent format and to put all the logic models and the level treatment for PRAs on the same level of detail, in some cases at the expense of eliminating some more detail in other elaborate PRAs and adding accident sequences to things that we think today do not reflect the state of the art understanding.

We then plan to quantify the PRAs to identify what is in the currently adopted data base for frequency of failure of equipment and operator actions. By doing it this way, we will have, we feel, probably a consistent uncertainty that will apply across the spread of PRAs we are talking about. This will apply also in plants we apply to these reference PRAs for further analysis.

We are not planning, at least at this stage, to use the numbers that come out of these things at face value, but instead as a bases of ranking and screening plants to identify ones that are worse and those that don't look so bad.

I would think that in this program it would be a logical approach in that we cannot again go off and assess how much precision or lack of accuracy we have with each plant that we are evaluating. Our approach is to try to identify the good ones and the bad ones. For the bad ones, we plan to look at alternative ways of improving those plants.

There will be uncertainties associated with the way we do our assessment, but we believe that at that stage it would be up to the particular industry representative utility to either attack or not attack what our decisions are. And one way of attacking them, of course, is to draw on uncertainties you may have to attack them.

But otherwise, except perhaps some sensitivity analyses along the lines of looking at uncertainties fo other accident sequences, we are not planning to do a whole lot of numbers with the range of uncertainties. We will have a tool to screen them out.

MR. OKRENT: Well, I certainly wouldn't recommend a whole lot of number punching, but I suppose what you are planning to do is put these PRAs on a similar basis. That wouldn't necessarily make them all right or all wrong. In fact, there can still be differences because there will be a set of assumptions on human error that you put in and it will affect two different plants in a certain way. It may not be alike. It may be just what happens within the human failure handbook or whatever the thing is called.

Now, what you are proposing to do is useful, but it is sort of not directly related to the question, but I will leave it for you to reflect on. I think the NRC needs to begin trying to address the uncertainties and not just do sensitivity studies, that is something quite different, and to try to decide how they are going to incorporate this uncertain state of knowledge into decision making.

There are some things you can do qualitatively, but you are inevitably going to be partly on a quantitative basis if you ever talk about cost-benefit decision-making.

MR. WARD: Andy, did you want to say something about this?

MR. KNIEL: One way of alleviating the uncertainty was in the formulation of the criteria of 10^{-4}, say, for core meltdown or severe core damage, and to say that in that criteria there is quite a substantial fraction to cover the uncertainties. For example, we can't allow 30 percent of that total simply to cover that uncertainty and then allocatè the remaining 70 percent. Those systems that can be quantified without too much uncertainty, where you are dealing only with random errors, and the human operator errors, and then to allow a further component to cover the area where you are pretty uncertain. That is to say, what we call the special circumstances for external hazards, so that is an area where we know we can't quantify it, but be conscious of the fact that it contributes to the overall probability of core meltdown, and therefore has got to be allowed for in any decision-making process.

When you look at it this way, that part is where we hope to do very much with the probabilistic risk assessment. It will be less than one-third of the total target. So that we would be able to have quite a substantial error, I think, in that quantification. In fact, two or three, without being grossly over in the overall target of the analysis.

MR. WARD: Did you want to comment?

MR. OKRENT: It is a possible approach for some people who will say a factor of ten is as good as, you know, sort of in that case, you have a problem, but I suspect that we have a question.

MR. WARD: Andy, on the plan to involve industry, you mentioned three options, or you showed three options, but what sort of schedule are you on to go with one or another of those options? How are you going to influence industry? What do you expect in the way of cooperation from EPRI and other parts of the industry?

MR. MARCHESE: I really can't speak to them at this point in time. At least right now I don't have any --

MR. RAY: Andy, have you contacted them? Are they responsive?

MR. MARHESE: Yes. I have talked to them. Actually, I made a trip out there to discuss this with them at some length. They said they would consider it. They couldn't make any commitments. I guess they felt they were in kind of a precarious situation as being between the utilities and the licensing people. They typically do not like to get in between.

I have talked to the AIF and the other industry people, and really have not gotten any commitments. So I think we will utilize them in the sense of establishing a peer review, but in terms of them actually committing to do anything, we don't have those commitments.

MR. RAY: Don't be too optimistic.

MR. MARCHESE: We won't.

MR. WARD: Is your idea to try to get them to do part of the A45, or to do work broader, beyond the newly defined A45? Part of the reason for cutting down the task was that your management thought that some of the things would rightfully be done by the industry.

MR. MARCHESE: We are too plant-specific.

MR. WARD: How are you going to get those things done by industry?

MR. MARCHESE: Hopefully, we will be in a position at the end of the program to tell them to direc them to do it, because we don't have the basis. Right now it is kind of a negotiating kind of thing.

MR. WARD: So the industry involvement of that type you see as requiring them to evaluate their systems against the criteria that you promulgate in two years?

MR. MARCHESE: Right. But what I mentioned, I would like to see them get involved earlier on. To what extent they will do that is questionable.

MR. BERRY: Our plan is, since we have had difficulty in getting somebody else to carry the ball for industry involvement, our plan is to establish what you might say, consultant agreements or support agreements, in a structural form with people in the industry to which we would pass reports at different milestones within the program to comment how we are doing and to give us suggestions on what we can do better. This was done in other work by Sandia in the sabotage program where a peer review group was established. About 50 members were involved. They involved very senior people from utilities and others.

I have had some people call me to get involved in such a group. In many cases, some people in the industry want to be in on the ground floor of what is being done.

MR. WARD: This is a little bit of an aside, but to give an example of the procedure that was followed in the sabotage program worries me a little bit, because as I recall the meeting we had in March out there at Sandia, we had a presentation for the laboratory people on the sabotag program, and then we had presentations from two of the major vendors on how their advanced designs would deal with the question of sabotage, and neither of those presentations paid any attention as far as I can tell to the work that Sandia had done.

MR. BERRY: Well, there is no way of assuring that the industry representatives will talk among themselves. We are just hoping that we can have some other minds that might be sensitive to concerns we are not concerned about to be interfacing with us in this program. Whether or not the industry people then take that insight back and use it is something we really cannot guarantee. We can guarantee that we will factor their comments and concerns into our evaluations.

MR. RAY: Dave, can we wind it up? Because we are digging into Chet's time, and we have a wall at the end of the day that we should not go beyond.

MR. WARD: Thank you, Andy. Are there any other comments from the Committee members, or is there any point of view that there is something in particular we should be adivsing the Committee or the staff on with regard to this task action plan? Should we write a letter saying something?

(No response.)

MR. WARD: Okay. Thank you, Andy.

(Whereupon, at 3:10 p.m., the Committee

recessed, to go into executive session.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

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In the matter of: ACRS/267th General Meeting

Date of Proceeding: July 9, 1982

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.

Jane N. Beach

Official Reporter (Typed)

Jane N. Beach

Official Reporter (Signature)

CONTENTIONS RELATED TO SITE SUITABILITY REPORT

CONTENTION NO.	SUBJECT
1(A)* 2* 3(B)-(D)*	INCLUSION OF CDAs IN DBA SPECTRUM AND, HENCE, IN SITE SUITABILITY SOURCE TERM
5(A)**	ADEQUACY OF CLINCH RIVER SITE METEOROLOGY AND POPULATION DENSITY.
5(B)	LONG-TERM EVACUATION OF NEARBY FACILITIES
11(D)(1)	10 CFR 100.11 ORGAN DOSE EQUIVALENT LIMITS

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* LIMITED TO FEASIBILITY OF DESIGNING CRBR PLANT TO MAKE CDAs SUFFICIENTLY IMPROBABLE THAT THEY CAN BE EXCLUDED FROM DBA SPECTRUM

** CONTENTION MORE RELATED TO NEPA ALTERNATIVE SITE REVIEW

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July 7, 1982

Dr. Paul Shewmon, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Shewmon:

I understand that the full Advisory Committee on Reactor Safeguards (ACRS) is meeting tomorrow, July 8, 1982, to consider the suitability of the proposed site for the Clinch River Breeder Reactor (CRBR). I also am aware that the ACRS Subcommittee on CRBR has held several meetings this year* to discuss the CRBR licensing approach, core disruptive accidents, and the suitability of the proposed site. I have attended these meetings when possible and have reviewed the transcripts of each meeting.

As you may be aware, the Natural Resources Defense Council, Inc. (NRDC), is a principal intervenor in the CRBR licensing proceedings. Several of NRDC's contentions concern the suitability of the CRBR site and other safety issues under review by the ACRS Subcommittee on CRBR.

I am writing you to express my dismay over the inadequacy of the review to date of the CRBR licensing approach, CRBR design, and the proposed site by the ACRS Subcommittee. First, during eight meetings, the Subcommittee has invited only the Applicants and the NRC Staff to present their respective views on the CRBR safety and site issues. The Subcommittee has ignored completely the Intervenors in this matter. Not a single member of the Subcommittee has directly sought, even informally, the views of the Intervenors' experts regarding CRBR safety and site suitability issues, even though the Subcommittee is aware of at least some of Intervenors'

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*/ Feb. 2-3; March 30-31; May 4-5; May 24-25.

contentions in this case (Transcripts of ACRS CRBR Subcommittee meeting, March 31, 1982, pp. 123-124). Intervenors are in sharp disagreement with both the Staff and the the Applicants on several key issues under review by the ACRS, and the ACRS should be fully aware of all points of controversy before making a decision.

Second, it has become obvious that neither the Staff nor the Applicants are being completely candid with the ACRS CRBR Subcommittee. Neither party has informed the Subcommittee of the severe limitations that have been placed, at their request, upon the scope of the safety and site suitability reviews during the LWA-1 proceedings. I suggest that the Subcommittee and full ACRS review the transcript of the Atomic Safety and Licensing Board Prehearing Conference of April 5-6, 1982, and the depositions of the Staff and Applicants taken by NRDC in June 1982.* You will find the presentations made by the Applicants and Staff to the ACRS strikingly dissimilar to those made to the Licensing Board and the Intervenors.

A third impropriety concerns Dr. William E. Kastenberg, a consultant to the ACRS CRBR Subcommittee. Under contract to the Department of Energy, one of the Applicants in this licensing proceeding, Dr. Kastenberg prepared a report entitled "Anticipated Transients Without Scram for Light Water Reactors: Implications for Liquid Metal Breeder Reactors" (co-authored with Kenneth H. Solomon), RAND Note N-1188-DOE, July 1979. In this report, Dr. Kastenberg draws conclusions about the adequacy of the CRBR design which also bear directly on the suitability of the CRBR site. As a prior consultant to DOE on matters directly related to the CRBR, Dr. Kastenberg should not now be serving as an ACRS consultant on those same issues. I do not know Dr. Kastenberg and make no allegations concerning objectivity; yet I believe he should withdraw from the ACRS CRBR Subcommittee immediately to avoid any appearance of bias or impropriety.

Fourth, at the March 31, 1982, Subcommittee meeting, Dr. Carson Mark , an ACRS member whose opinions I respect but do not necessarily agree with, stated (Transcript, p. 124):

> ... it will be hilarious if the intervenors bring this up -- is [sic] the possibility of interrupting operations at K25, which they obviously would like to interrupt anyway. To raise that contention will really be great fun.

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*/ Staff - May 6, 1982; Applicants - June 16, 21, 1982.

Had the ACRS shown Intervenors the courtesy of inviting our views on our contentions, I might be inclined to dismiss this statement as a little joke in bad taste but of no consequence. The fact that the ACRS continues to thumb its nose at Intervenors while making these remarks reflects a more serious problem; namely, that the ACRS displays a lack of independence and detachment necessary to function as an impartial reviewer of the CRBR.

I would be pleased to hear that you are taking steps to rectify this situation.

Sincerely,

TB Cochran

Thomas B. Cochran, Ph.D.

APPENDIX I

ADMITTED AND RENUMBERED CONTENTIONS

1. The envelope of DBAs should include the CDA.

   a) Neither Applicants nor Staff have demonstrated through reliable data that the probability of anticipated transients without scram or other CDA initiators is sufficiently low to enable CDAs to be excluded from the envelope of DBAs.

   b) Neither Applicants nor Staff have established that Applicants' "reliability program" even if implemented is capable of eliminating CDAs as DBAs.

      (1) The methodology described in the PSAR places reliance upon fault tree and event tree analysis. Applicants have not established that it is possible to obtain sufficient failure mode data pertinent to CRBR systems to validly employ these techniques in predicting the probability of CDAs.

      (2) Applicants' projected data base to be used in the reliability program is inadequate. Applicants have not established that the projected data base encompasses all credible failure modes and human elements.

      (3) Even if all of the data described in Applicants' projected data base is obtained, Applicants have not establishd that CDAs have a sufficiently low

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probability that they may be excluded from the CRBR design bases.

(4) Applicants have not established that the test program used for their reliability program will be completed prior to Applicants' projected date for completion of construction of the CRBR.

2. The analyses of CDAs and their consequences by Applicants and Staff are inadequate for purposes of licensing the CRBR, performing the NEPA cost/benefit analysis, or demonstrating that the radiological source term for CRBRP would result in potential hazards not exceeded by those from any accident considered credible, as required by 10 CFR §100.11(a), fn. 1.

a) The radiological source term analysis used in CRBRP site suitability should be derived through a mechanistic analysis. Neither Applicants nor Staff have based the radiological source term on such an analysis.

b) The radiological source term analysis should be based on the assumption that CDAs (failure to scram with substantial core disruption) are credible accidents within the DBA envelope, should place an upper bound on the explosive potential of a CDA, and should then derive a conservative estimate of the fission product release from such an accident. Neither Applicants nor Staff have performed such an analysis.

c) The radiological source term analysis has not adequately considered either the release of fission products and core materials, e.g. halogens, iodine and plutonium, or the environmental conditions in the reactor containment building created by the release of substantial quantities of sodium. Neither Applicants nor Staff have established the maximum credible sodium release following a CDA or included the environmental conditions caused by such a sodium release as part of the radiological source term pathway analysis.

d) Neither Applicants nor Staff have demonstrated that the design of the containment is adequate to reduce calculated offsite doses to an acceptable level.

e) As set forth in Contention 8(d), neither Applicants nor Staff have adequately calculated the guideline values for radiation doses from postulated CRBRP releases.

f) Applicants have not established that the computer models (including computer codes) referenced in Applicants' CDA safety analysis reports, including the PSAR, and referenced in the Staff CDA safety analyses are valid. The models and computer codes used in the PSAR and the Staff safety analyses of CDAs and their consequences have not been adequately documented, verified or validated by comparison with applicable experimental data. Applicants' and Staff's safety analyses do not establish that the models accurately represent the physical phenomena and principles which control the response of CRBR to CDAs.

g) Neither Applicants nor Staff have established that the input data and assumptions for the computer models and codes are adequately documented or verified.

h) Since neither Applicants nor Staff have established that the models, computer codes, input data and assumptions are adequately documented, verified and validated, they have also been unable to establish the energetics of a CDA and thus have also not established the adequacy of the containment of the source term for post accident radiological analysis.

3. Neither Applicants nor Staff have given sufficient attention to CRBR accidents other than the DBAs for the following reasons:

a) Neither Applicants nor Staff have done an adequate, comprehensive analysis comparable to the Reactor Safety Study ("Rasmussen Report") that could identify other CRBR accident possibilities of greater frequency or consequence than the accident scenarios analyzed by Applicants and Staff.

b) Neither Applicants' nor Staff's analyses of potential accident initiators, sequences, and events are sufficiently comprehensive to assure that analysis of the DBAs will envelop the entire spectrum of credible accident initiators, sequences, and events.

c) Accidents associated with core meltthrough following loss of core geometry and sodium-concrete interactions have not been adequately analyzed.

d) Neither Applicants nor Staff have adequately identified and analyzed the ways in which human error can initiate, exacerbate, or interfere with the mitigation of CRBR accidents.

4. Neither Applicants nor Staff adequately analyze the health and safety consequences of acts of sabotage, terrorism or theft directed against the CRBR or supporting facilities nor do they adequately analyze the programs to prevent such acts or disadvantages of any measures to be used to prevent such acts.

a) Small quantities of plutonium can be converted into a nuclear bomb or plutonium dispersion device which if used could cause widespread death and destruction.

b) Plutonium in an easily usable form will be available in substantial quantities at the CRBR and at supporting fuel cycle facilities.

c) Analyses conducted by the Federal Government of the potential threat from terrorists, saboteurs and thieves demonstrate several credible scenarios which could result in plutonium diversion or releases of radiation (both purposeful and accidental) and against which no adequate safeguards have been proposed by Applicants or Staff.

d) Acts of sabotage or terrorism could be the initiating cause for CDAs or other severe CRBR accidents and the probability of such acts occurring has not been analyzed in predicting the probability of a CDA.

5. Neither Applicants nor Staff have established that the site selected for the CRBR provides adequate protection for public health and safety, the environment, national security, and national energy supplies; and an alternative site would be preferable for the following reasons:

a) The site meteorology and population density are less favorable than most sites used for LWRs.

(1) The wind speed and inversion conditions at the Clinch River site are less favorable than most sites used for light-water reactors.

(2) The population density of the CRBR site is less favorable than that of several alternative sites.

(3) Alternative sites with more favorable meteorology and population characteristics have not been adequately identified and analyzed by Applicants and Staff. The analysis of alternative sites in the ER and the Staff Site Suitability Report gave insufficient weight to the meteorological and population disadvantages of the Clinch River site and did not attempt to identify a site or sites with more favorable characteristics.

b) Since the gaseous diffusion plant, other proposed energy fuel cycle facilities, the Y-12 plant and the Oak Ridge National Laboratory are in close proximity to the site an accident at the CRBR could result in the long term evacuation of those facilities. Long term evacuation of those facilities would result in unacceptable risks to the national security and the national energy supply.

6. The ER and FES do not include an adequate analysis of the environmental impact of the fuel cycle associated with the CRBR for the following reasons:

   a) The ER and FES estimate the environmental impacts of the fuel cycle based upon a scale-down of analyses presented in the LMFBR Program Environmental Statement and Supplement for a model LMFBR and fuel cycle. The analyses of the environmental impacts of the model LMFBR and fuel cycle in the LMFBR Program Statement and Supplement are based upon a series of faulty assumptions.

   b) The impacts of the actual fuel cycle associated with CRBR will differ from the model LMFBR and fuel cycle analyzed in the LMFBR Program Environmental Statement and Supplement. The analysis of fuel cycle impacts must be done for the particular circumstances applicable to the CRBR. The analyses of fuel cycle impacts in the ER and FES are inadequate since:

      (1) The impact of reprocessing of spent fuel and plutonium separation required for the CRBR is not included or is inadequately assessed;

(2) The impact of transportation of plutonium required for the CRBR is not included, or is inadequately assessed;

(3) The impact of disposal of wastes from the CRBR spent fuel is not included, or is inadequately assessed;

(4) The impact of an act of sabotage, terrorism or theft directed against the plutonium in the CRBR fuel cycle, including the plant, is not included or is inadequately assessed, nor is the impact of various measures intended to be used to prevent sabotage, theft or diversion.

7. Neither Applicants nor Staff have adequately analyzed the alternatives to the CRBR for the following reasons:

a) Neither Applicants nor Staff have adequately demonstrated that the CRBR as now planned will achieve the objectives established for it in the LMFBR Program Impact Statement and Supplement.

(1) It has not been established how the CRBR will achieve the objectives there listed in a timely fashion.

(2) In order to do this it must be shown that the specific design of the CRBR, particularly core design and engineering safety features, is sufficiently similar to a practical commercial size LMFBR that building and operating the CRBR will

demonstrate anything relevant with respect to an economic, reliable and licensable LMFBR.

(3) The CRBR is not reasonably likely to demonstrate the reliability, maintainability, economic feasibility, technical performance, environmental acceptability or safety of a relevant commercial LMFBR central station electric plant.

b) No adequate analysis has been made by Applicants or Staff to determine whether the informational requirements of the LMFBR program or of a demonstration-scale facility might be substantially better satisfied by alternative design features such as are embodied in certain foreign breeder reactors.

c) Alternative sites with more favorable environmental and safety features were not analyzed adequately and insufficient weight was given to environmental and safety values in site selection.

(1) Alternatives which were inadequately analyzed include Hanford Reservation, Idaho Reservation (INEL), Nevada Test Site, the TVA Hartsville and Yellow Creek sites, co-location with an LMFBR fuel reprocessing plant (e.g., the Development Reprocessing Plant), an LMFBR fuel fabricating plant, and underground sites.

8. The unavoidable adverse environmental effects associate with the decommissioning of the CRBR have not been adequately analyzed, and the costs (both internalized economic costs and external social costs) associated with the decommissioned CRBR are not adequately assessed in the NEPA benefit-cost balancing of the CRBR.

   a) There is no analysis of decommissioning in the Applicants' Environmental Report;

   b) Environmental Impact Statements (EIS) related to LWRs prepared by NRC have been inadequate due in part to recently discovered omissions (see below), and the FES for the CRBR is no different;

   c) A recent report "Decommissioning Nuclear Reactors" by S. Harwood; May, K.; Resnikoff, M.; Schlenger, B.; and ames, P. (New York Public Interest Research Group (N.Y. PIRG), unpublished, January, 1976) indicates that (with the exception of the Elk River reactor) the isolation period following decommissioning of power reactors has been based on the time required for Co-60 to decay to safe levels. Harwood, et al. (p. 2) believe the previous analyses are in error because they have underestimated the significance of radionuclide, Ni-59. The time period for Ni-59 to decay to safe levels is estimated by Harwood, et al. (p. 2) fo. LWR to be at least 1.5 million years. The economic and societal implications of this 1.5 million year decay period are at present unknown.

d) Petitioner believes the NRC must systematically analyze all neutron activation products that may be produced in the proposed CRBR to determine the potential isolation period, following decommissioning, and then provide a comprehensive analysis of the costs (both economic and societal) of decommissioning.

9. Neither Applicants nor Staff have demonstrated that Applicants' plans for coping with emergencies are adequate to meet NRC requirements.

a) The PSAR contains insufficient information regarding Applicants' ability to identify the seriousness and potential scope of radiological consequences of emergency situations within and outside the site boundary, including capabilities for dose projection using real-time meteorological information and for dispatch of radiological monitoring teams within the Emergency Planning Zones.

b) Applicants and Staff have failed to account properly for local emergency response needs and capabilities in establishing boundaries for the plume exposure pathway and ingestion pathway EPZs for the CRBR.

c) The PSAR contains insufficient analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, nor does it note major impediments to the evacuation or taking of protective actions.

d) The PSAR contains insufficient information to ensure the compatibility of proposed emergency plans for both onsite areas and the EPZs, with facility design features, site layout, and site location.

e) The PSAR contains insufficient information concerning the procedures by which protective actions will be carried out, including authorization, notification, and instruction procedures for evacuations.

f) Applicants' proposed emergency plans fail to take into account the special measures necessary to cope with a CDA, including the need for increased protective, evacuation and monitoring measures, reduced response time and special protective action levels.

g) Applicants and Staff have failed to provide adequate assurance that the proposed emergency plans will meet the requirements and standards of 10 CFR §50.47(b).

10. Neither Applicants nor Staff have demonstrated that the facility will be provided with systems necessary to establish and maintain safe cold shutdown and maintain containment integrity that are capable of performing their functions during and after being exposed to the environmental conditions

    a) associated with postulated accidents, as required by General Design Criterion 4, 10 CFR Part 50, Appendix A; or

    b) created by sodium fires or the burning (or local detonation) of hydrogen.

11. The health and safety consequences to the public and plant employees which may occur if the CRBR merely complies with current NRC standards for radiation protection of the public health and safety have not been adequately analyzed by Applicants or Staff.

    a) Neither Applicants nor Staff have shown that exposures to the public and plant employees will be as low as practicable (reasonably achievable).

    b) Neither Applicants nor Staff have adequately assessed the genetic effects from radiation exposure including genetic effects to the general population from plant employee exposure.

c) Neither Applicants nor Staff have adequately assessed the induction of cancer from the exposure of plant employees and the public.

d) Guideline values for permissible organ doses used by Applicants and Staff have not been shown to have a valid basis.

(1) The approach utilized by Applicants and Staff in establishing 10 CFR §100.11 organ dose equivalent limits corresponding to a whole body dose of 25 rems is inappropriate because it fails to consider important organs, e.g., the liver, and because it fails to consider new knowledge, e.g., recommendations of the ICRP in Reports 26 and 30.

(2) Neither Applicants nor Staff have given adequate consideration to the plutonium "hot particle" hypothesis advanced by Arthur R. Tamplin and Thomas B. Cochran, or to the Karl Z. Morgan hypothesis described in "Suggested Reduction of Permissible Exposure to Plutonium and Other Transuranium Elements," Journal of American Industrial Hygiene (August 1975).

CRBR PLANT SITE SUITABILITY REVIEW

- LWA-1s
- PROPOSED SITE PREPARATION ACTIVITIES
- APPROACH TO SITE SUITABILITY REVIEW
- SITE SUITABILITY REPORT

Cecil Thomas
T2

LWA-1s

- ISSUANCE GOVERNED BY 10 CFR 50.10(E)
- AUTHORIZES CONDUCT OF NON-SAFETY-RELATED SITE PREPARATION ACTIVITIES
- REQUIRES COMPLETION OF ENVIRONMENTAL AND SITE SUITABILITY REVIEWS AND PUBLIC HEARINGS THEREON
- ACTIVITIES UNDERTAKEN ENTIRELY AT RISK OF APPLICANTS
- ISSUANCE HAS NO BEARING ON ISSUANCE OF CONSTRUCTION PERMIT
- ISSUANCE REQUIRES FINDING THAT "...BASED UPON THE AVAILABLE INFORMATION AND REVIEW TO DATE, THERE IS REASONABLE ASSURANCE THAT THE PROPOSED SITE IS A SUITABLE LOCATION FOR A REACTOR OF THE GENERAL SIZE AND TYPE PROPOSED FROM THE STANDPOINT OF RADIOLOGICAL HEALTH AND SAFETY CONSIDERATIONS..." (10 CFR 50.10 (E)(2)
- 27 ISSUED SINCE ESTABLISHED IN 1974

PROPOSED SITE PREPARATION ACTIVITIES

- GENERAL SITE CLEARING AND GRADING

  AREAS FOR ACCESS ROADS AND RAILROADS, TEMPORARY CONSTRUCTION FACILITIES, PARKING LOT, MAIN PLANT. COOLING TOWERS, SWITCHYARDS, STORAGE AREAS, ON-SITE QUARRY, RUNOFF TREATMENT PONDS, CONCRETE BATCHING AND MIXING PLANT AND BARGE UNLOADING FACILITY.

- EXCAVATION

  ACCESS ROADS AND RAILROADS, CONCRETE BATCHING AND MIXING PLANT, PARKING LOT, MAIN PLANT, COOLING TOWERS, SWITCHYARDS, STORAGE AREAS, TEMPORARY CONSTRUCTION FACILITIES AND BUILDINGS, RUNOFF TREATMENT PONDS AND QUARRY OPERATIONS.

- INSTALLATION OF TEMPORARY CONSTRUCTION FACILITIES

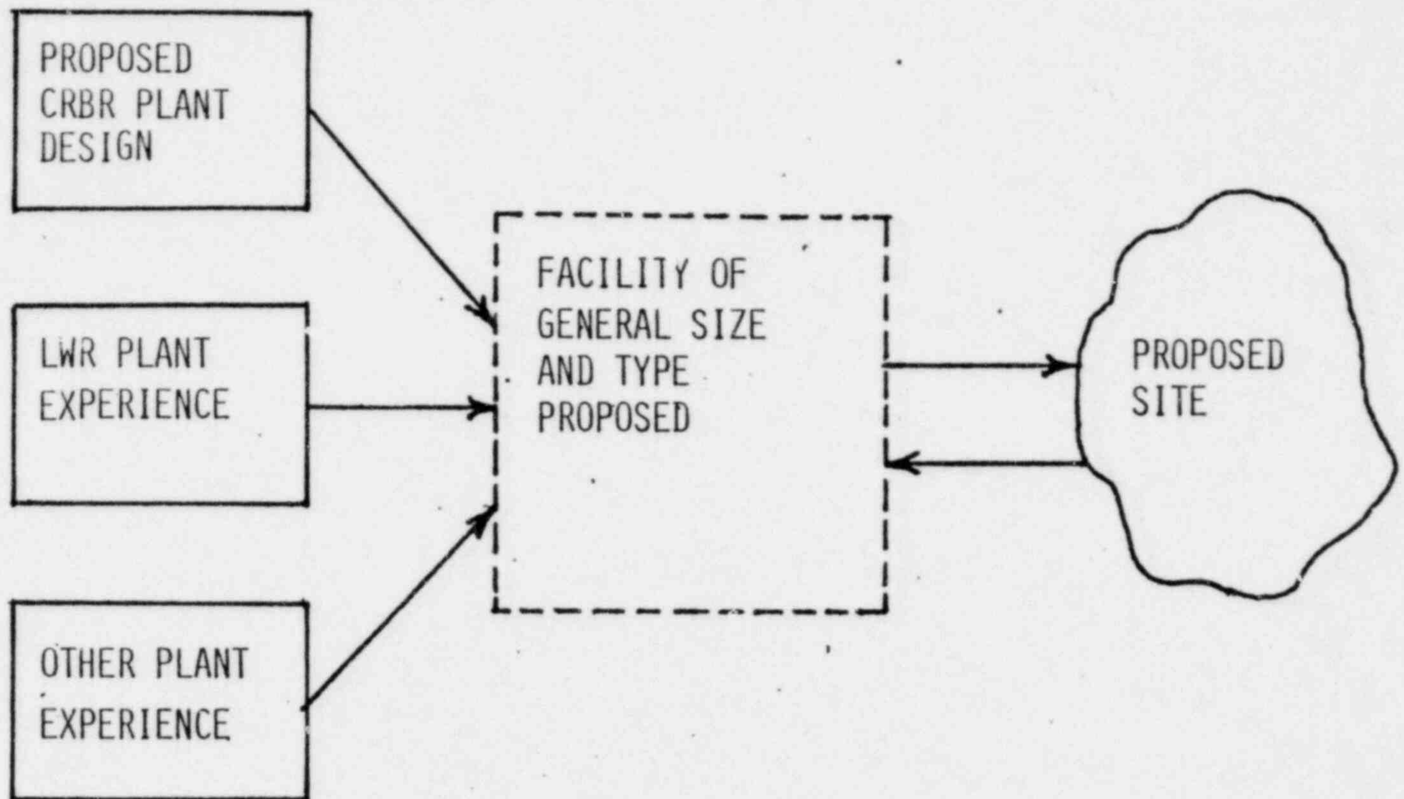
  TEMPORARY ONSITE ROADS, CONSTRUCTION PARKING AREAS, RAILROADS AND RAILROAD SPURS, CONTRACTOR WORK AND STORAGE AREAS, CONSTRUCTION UTILITIES, CONCRETE BATCHING AND MIXING PLANT, ONSITE QUARRY AND CRUSHING FACILITY SEWAGE TREATMENT PLANT AND CRAFT TOILET FACILITY, FIRE PROTECTION SYSTEM,

RUNOFF TREATMENT PONDS, STORM DRAINAGE SYSTEM, BARGE UNLOADING SYSTEM AND CONSTRUCTION BUILDINGS.

- OTHER ACTIVITIES

  PERMANENT ACCESS ROAD, RAILROAD SPUR, CONSTRUCTION PARKING AREA, TEMPORARY ROADS, CONTRACTOR WORK AND STORAGE AREAS, CONSTRUCTION UTILITIES, PERMANENT MAIN SURVEY CONTROL LINES AND BENCHMARKS AND QUARRY AND STOCKPILE AREAS.

APPROACH TO SITE SUITABILITY REVIEW



STEP 1: DEFINE CHARACTERISTICS OF FACILITY OF GENERAL SIZE AND TYPE PROPOSED RELEVANT TO SITE SUITABILITY.

STEP 2: DETERMINE CHARACTERISTICS OF PROPOSED SITE.

STEP 3: ASSESS COMPATIBILITY OF SITE AND FACILITY CHARACTERISTICS.

SITE SUITABILITY REPORT

- o NUREG-0786 (UPDATES MARCH 1977 REPORT)
- o DOCUMENTS RESULTS OF STAFF'S EVALUATION OF SUITABILITY OF CLINCH RIVER SITE FOR FACILITY OF GENERAL SIZE AND TYPE AS PROPOSED CRBR PLANT.
- o CONCLUDES THAT BASED ON AVAILABLE INFORMATION AND REVIEW TO DATE, THERE IS REASONABLE ASSURANCE THAT THE CLINCH RIVER SITE IS A SUITABLE LOCATION FOR A FACILITY OF THE GENERAL SIZE AND TYPE AS THE PROPOSED CRBR PLANT FROM THE STANDPOINT OF RADIOLOGICAL HEALTH AND SAFETY CONSIDERATIONS.

BASIS FOR STAFF's BELIEF THAT CRBR RISK WILL BE COMPARABLE TO LWR RISK

- CRBR WILL MEET ALL APPLICABLE LWR REGULATORY CRITERIA AND ADDITIONAL SPECIAL CRITERIA APPROPRIATE TO LMFBRs.
- CONSEQUENCES OF DBAs AND SSST WILL BE WITHIN 10 CFR 100 GUIDELINES.
- DESIGN MEASURES TO MAKE SEVERE ACCIDENTS (CDAs) VERY IMPROBABLE.
- DESIGN MEASURES TO ACCOMMODATE SEVERE ACCIDENTS (CDAs).
- PRELIMINARY EVALUATION OF ACCIDENT RISKS.
- PERFORMANCE OF PRA TO CONFIRM THAT CRBR MEETS SAFETY GOAL.

Morris

T3

RISK COMPARABILITY

OF

CRBRP DESIGN

WITH LWR'S

SIMILAR SOURCES AND CAUSES

- RISK DOMINANT ACCIDENT SEQUENCES INVOLVE CORE
- CORE INVENTORIES ARE COMPARABLE PER MW (PLUTONIUM LARGER IN CRBRP)
- HEAT GENERATION VS HEAT REMOVAL IMBALANCE FOR FUEL DAMAGE TO OCCUR

SIMILAR ACCIDENT TYPES

- INTERNAL PLANT FAILURES
- EXTERNAL FORCES
- SABOTAGE

Rumble
T4

CORE DISRUPTION

INTERNAL PLANT FAILURE

LOCA

FLOW BLOCKAGE

LOHS

FAILED FUEL PROPAGATION

TRANSIENTS

PRIMARY COOLANT SYSTEM RESPONSE TO CORE DISRUPTION

MECHANICAL FAILURES - HEAD RELEASE

THERMAL FAILURES - RELEASE TO REACTOR CAVITY

CONTAINMENT RESPONSE

- CONTAINMENT ENVIRONMENTAL FACTORS AFFECTING FISSION PRODUCT BEHAVIOR AND EQUIPMENT OPERATION

  PRESSURE

  TEMPERATURE

  AIRBORNE MATERIALS

- CONTAINMENT FAILURE MODES

  FAILURE TO ISOLATE

  EARLY FILTERED VENTING

  OVERPRESSURE FAILURE

  PROMPT FAILURES

LOSS OF ALL OFF-SITE ELECTRIC POWER AT CRBRP

HEAT TRANSPORT SYSTEMS

- STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM
  STEAM DUMPING (SHORT TERM)
  PROTECTED AIR-COOLED CONDENSERS
- DIRECT HEAT REMOVAL SYSTEM

ELECTRICAL POWER

- TURBINE BYPASS SYSTEM
- DIESEL GENERATORS
- BATTERY POWER (SEVERAL HOURS)

CDA SEQUENCE CLASSES FOR SCOPING CRBR RISKS
FROM INTERNAL INITIATORS

INITIATION	PRIMARY SYSTEM FAILURE	CONTAINMENT FAILURE
GENERIC CORE DISRUPTION	SMALL OR LARGE HEAD RELEASE & THERMAL FAILURE	NONE
GENERIC CORE DISRUPTION	SMALL OR LARGE HEAD RELEASE & THERMAL FAILURE	OVERPRESSURE
GENERIC CORE DISRUPTION	SMALL HEAD RELEASE & THERMAL FAILURE	CONTAINMENT ISOLATION
GENERIC CORE DISRUPTION	LARGE HEAD RELEASE & THERMAL FAILURE	CONTAINMENT ISOLATION

A SCOPING COMPARISON OF SEVERE ACCIDENT RISKS DUE TO CRBRP WITH COMPARABLE SIZE LWRs AT CRBRP SITE

- USED CRAC CODE TO PERFORM THE CALCULATIONS TO GAIN A PERSPECTIVE OF RELATIVE RISKS OF CRBRP AND LWRs.

- THE CRBRP ACCIDENT SEQUENCES, PROBABILITIES, AND RELEASE FRACTIONS WERE BASED ON SCOPING ESTIMATES DESCRIBED TO YOU BY ED RUMBLE.

- THE BWR AND THE PWR ACCIDENT SEQUENCES, PROBABILITIES, AND RELEASE FRACTIONS WERE THE SAME AS USED IN OUR ACCIDENT EVALUATIONS FOR ENVIRONMENTAL STATEMENTS. (RSS REBASE-LINE)

- THE CORE INVENTORIES CORRESPONDED TO THE POWER LEVEL OF 1121 MWT. (INCLUDING THE CONSIDERATION OF THE DIFFERENCES IN CRBRP AND LWR CORES).

- FOR THIS COMPARISON WE USED THE CRBRP SITE CHARACTERISTICS (METEOROLOGY, POPULATION DISTRIBUTION, ETC.)

T4

CONCLUSIONS OF THE COMPARISON

- BASED ON THE PRELIMINARY SCOPING ANALYSIS THE STAFF FINDS THAT THE CRBRP RISKS WILL NOT EXCEED THE RISKS FROM COMPARABLE LWRs.

- FURTHER WORK ON A FULL PRA IS IN PROGRESS AND WILL ESTABLISH BETTER ESTIMATES OF PROBABILITIES AND RELEASES AS DISCUSSED BY ED RUMBLE.

CRBR DOSE GUIDELINES

	LWR*		CRBR**	
	CP STAGE DOSE (REM)	OL STAGE DOSE (REM)	CP STAGE DOSE (REM)	OL STAGE DOSE (REM)
THYROID	150	300	150	300
WHOLE BODY	20	25	20	25
BONE SURFACE	-	-	150	300
RED BONE MARROW	-	-	37.5	75
LUNG	-	-	37.5	75
LIVER	-	-	75	150

ADDITIONAL GUIDELINES

Mortality risk equivalent whole body dose from any postulated design basis accident (on a calculated dose basis) should be no greater than the mortality risk equivalent whole body dose value of 10 CFR Part 100 for an LWR (i.e., 34 rem whole body risk equivalent at the O.L. stage, and 24.5 rem whole body risk equivalent at the CP stage).

*BASIS: 10 CFR PART 100

**BASIS: SAME AS LWR FOR THYROID AND WHOLE BODY. THE LUNG AND BONE DOSES ARE BASED ON THE CRITICAL ORGAN CONCEPT.

Site Suitability Source Term Release from Core†

RADIOACTIVE SPECIES	LWR* SOURCE TERM	CRBR** SOURCE TERM
NOBLE GASES	100%	100%
HALOGENS	50%	50%
SOLIDS	1%	1%
PLUTONIUM	-	1%

* BASIS: TID 14844 NON-MECHANISTIC SOURCE TERM (i.e., SEQUENCE OF EVENTS NOT TAKEN INTO ACCOUNT.

** BASIS: SAME BASIS AS FOR LWR SOURCE TERM WITH INCLUSION OF PLUTONIUM

† FISSION PRODUCTS ARE ASSUMED TO BE RELEASED FROM THE CORE TO THE PRIMARY CONTAINMENT. THE ASSUMPTION IS THAT THE SOURCE TERM FISSION PRODUCTS ARE INSTANTANEOUSLY RELEASED TO AND UNIFORMLY DISTRIBUTED THROUGHOUT THE PRIMARY CONTAINMENT AND AVAILABLE FOR RELEASE TO THE ENVIRONMENT (EXCEPT IN THE CASE OF THE IODINES IT IS ASSUMED THAT ONE-HALF OF THE IODINES RELEASED ARE INSTANTANEOUSLY PLATED OUT AND THE REMAINDER IS UNIFORMLY DISTRIBUTED THROUGHOUT THE PRIMARY CONTAINMENT AND AVAILABLE FOR RELEASE TO THE ENVIRONMENT).

SITE SUITABILITY SOURCE TERM ASSUMPTIONS AND DOSE RESULTS

Power Level	1121 MWt
Core Fraction Released to Containment:	
Noble Gases	100%
Iodines	50%
Solid Fission Products	1%
Plutonium	1%
Primary Containment Free Volume	$3.7 \times 10^6 ft^3$
Primary Containment Leak Rate	0.1%/day
Bypass Fraction	0.001%/day
Annulus Filtration System Filter Efficiencies:	
Particulate Iodine, Solids and Plutonium	99%
Elemental and Organic Iodine	95%
Annulus Filtration System Flow Rates, cfm:	
Exhaust	3000
Recirculation	11000
Aerosol Fallout Coefficients in Containment, hr^1	
0-2 hours	.0853
2-8 hours	.0659
8-24 hours	.0571
Minimum Exclusion Area Boundary Distance	670 meters
Low Population Zone	4023 meters
Atmospheric Dispersion Parameters (5% meteorology), sec/m^3	
0-2 hours at exclusion area boundary	1.22×10^{-3}
0-8 hours at LPZ	1.2×10^{-4}
8-24 hours at LPZ	8.4×10^{-5}
24-96 hours at LPZ	3.9×10^{-5}
96-720 hours at LPZ	1.4×10^{-5}

Dose Consequences, rem

	Exclusion Area	Low Population Zone
Thyroid	12	7
Whole Body	0.6	0.3
Lung	0.4	0.4
Bone Surface	10	9
Red Bone Marrow	2.4	2.1
Liver	1.1	1.0
Mortality Risk Equivalent Whole Body	1.7	1.1

EVENTS ANALYZED IN

<u>DESIGN BASIS FLOOD DETERMINATION</u>

PROBABLE MAXIMUM FLOOD

SEISMIC FAILURE

- OBE CONCURRENT WITH ½ PMF
- SSE CONCURRENT WITH 25-YEAR FLOOD

Lee

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PROBABLE MAXIMUM PRECIPITATION

RAINFALL DEPTH (FOR A PARTICULAR SIZE BASIN) THAT APPROACHES THE UPPER LIMIT THAT THE PRESENT CLIMATE CAN PRODUCE.

PMP - CRBR

9 DAY STORM

*3-DAY ANTECEDENT STORM -	6.8 INCHES
*3-DAY DRY PERIOD -	0
*3-DAY MAIN STORM -	17.2 INCHES
*TOTAL	24.0 INCHES

*AVERAGE ON WATERSHED ABOVE WATTS BAR

FLOOD ELEVATIONS

PLANT GRADE ELEVATION = 815

EVENT	CRBR ELEVATION MILE 16	CRBR ELEVATION MILE 18
PMF	776.0	777.5
OBE FAILURE WITH ½ PMF	798.2	804.3
SSE FAILURE WITH 25-YR. FLOOD	790.5	796.3

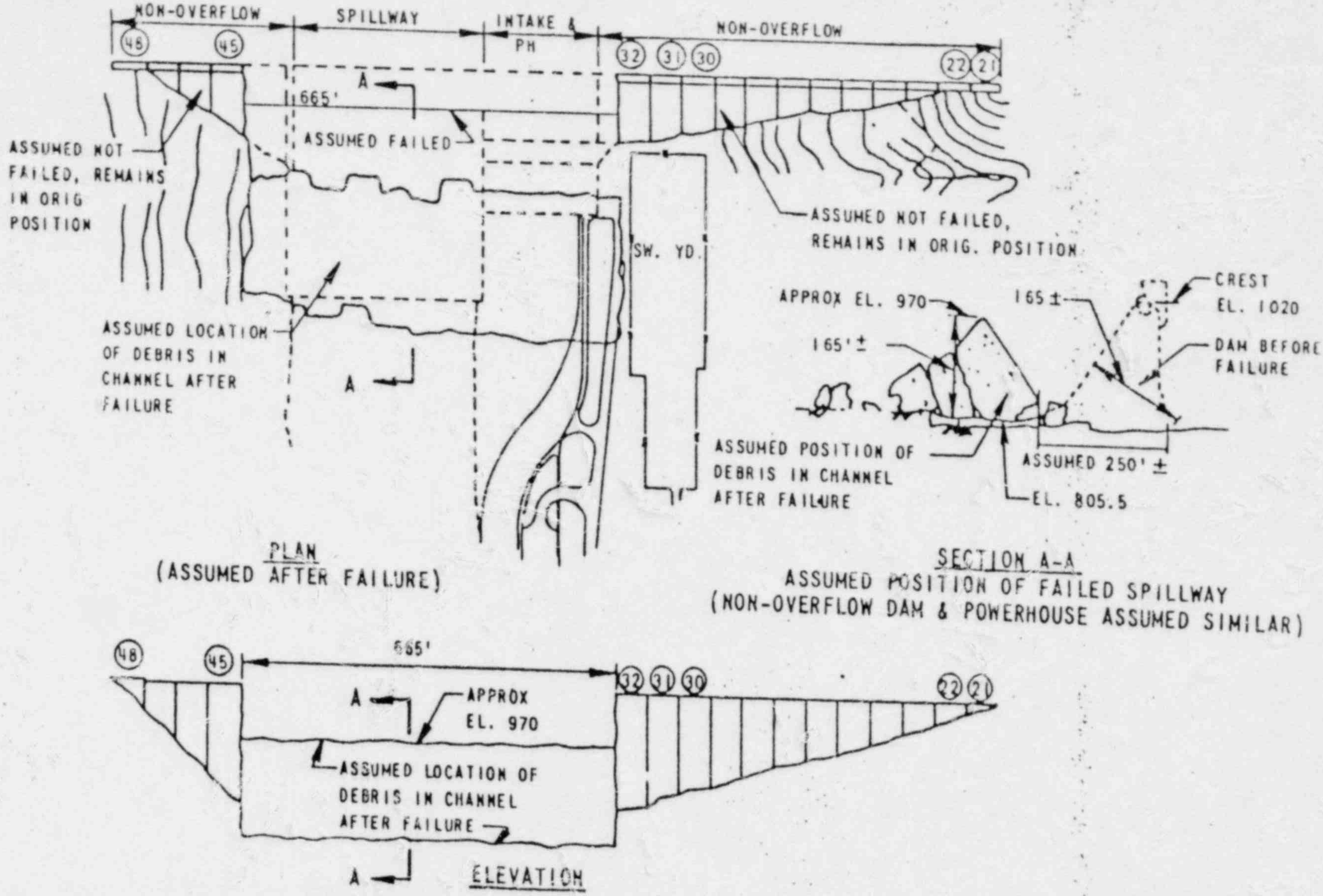

Figure 2.4-30 Norris Dam - Analysis for OBE & One Half PMF Assumed Condition of Dam After Failure

SENSITIVITY RUNS

POSTULATED FAILURE MODE	CRBR ELEVATION MILE 16	CRBR ELEVATION MILE 18
OBE CONDITIONS WITH ½ PMF		
INSTANT VANISHMENT OF ENTIRE DAM (NO DEBRIS)	811.0	818.0
VANISHMENT OF THREE BLOCKS (38-40) TO GROUND LEVEL	802.2	808.4
OVERTURNING OF BLOCKS 33-44 (665-FOOT WIDTH) WITH 945 DEBRIS LEVEL	802.6	808.9
OVERTURNING OF BLOCKS 37-43 (370-FOOT WIDTH) WITH 925 DEBRIS LEVEL	805.3	811.9

CRBRP SITE SUITABILITY

CRBRP PROJECT

BRIEFING FOR:

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

SITE DESCRIPTION

PRESENTED BY:

HENRY B. PIPER
PUBLIC SAFETY
CRBRP PROJECT OFFICE

JULY 9, 1982

T6

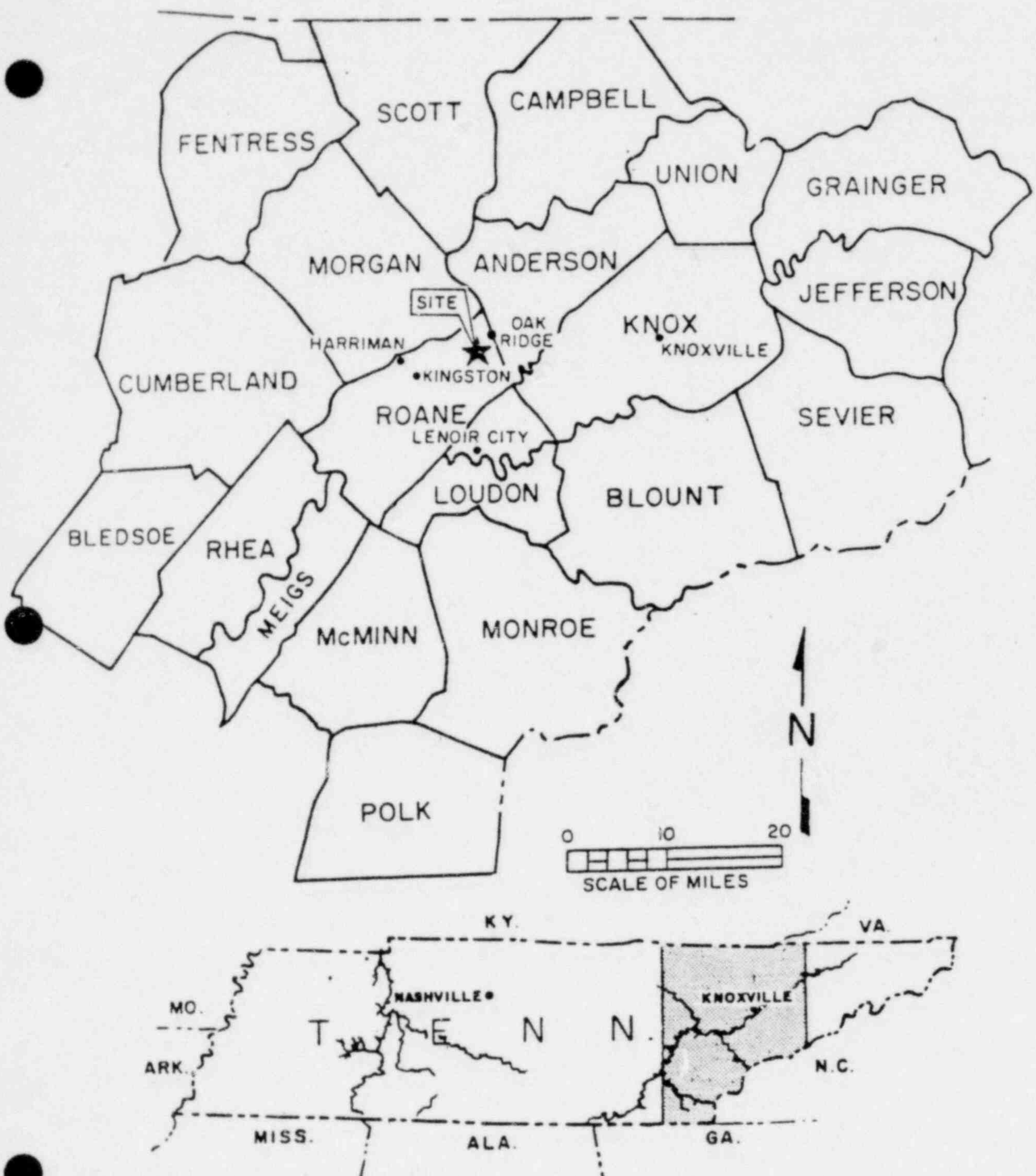

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Figure 2.1-1 LOCATION OF CLINCH RIVER SITE IN RELATION TO COUNTIES AND STATE

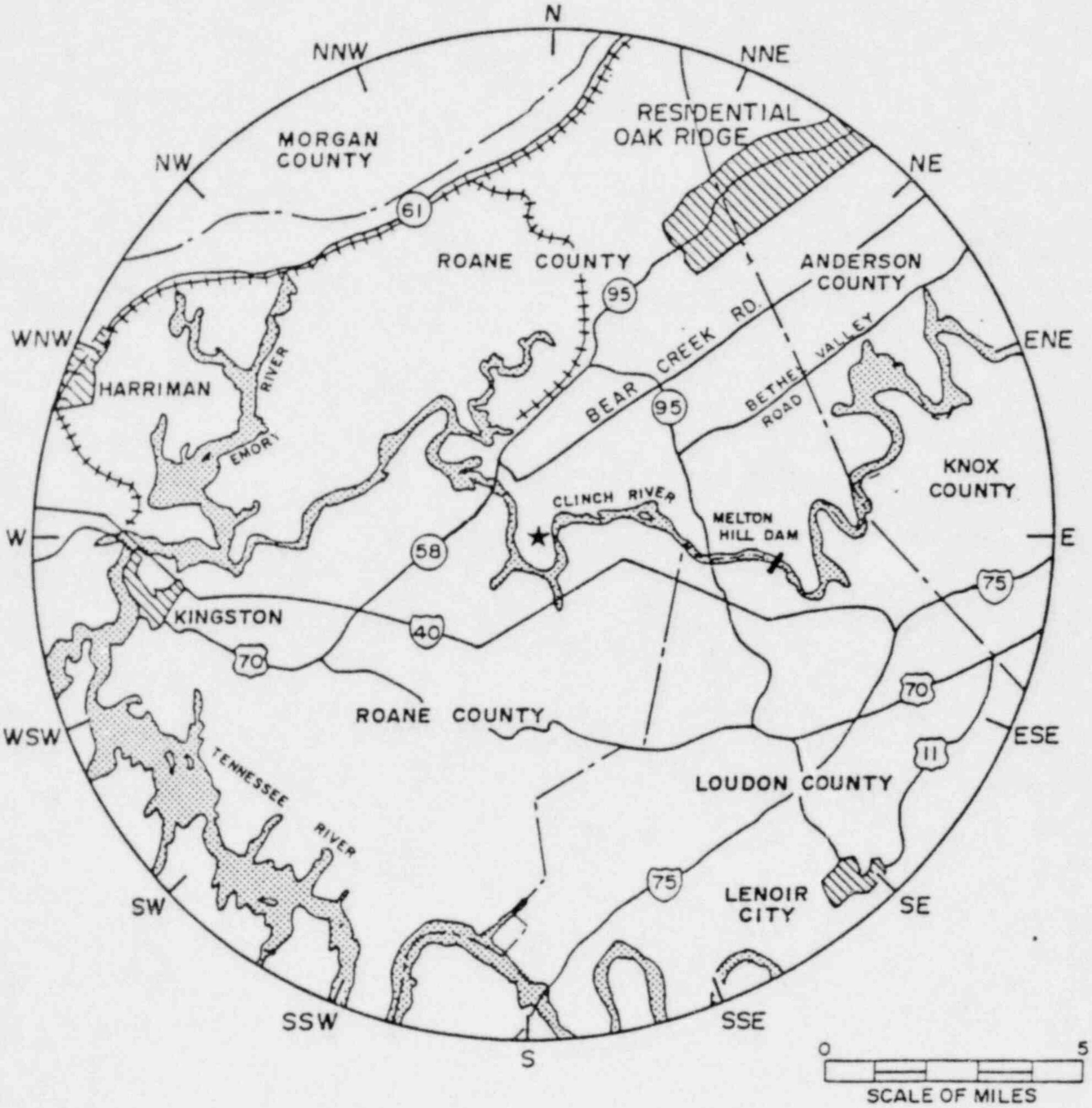

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Figure 2.2-4 URBAN AREAS WITHIN 10 MILES OF THE CRBRP SITE.

Figure 2.1-5 TOPOGRAPHY OF THE CRBRP SITE

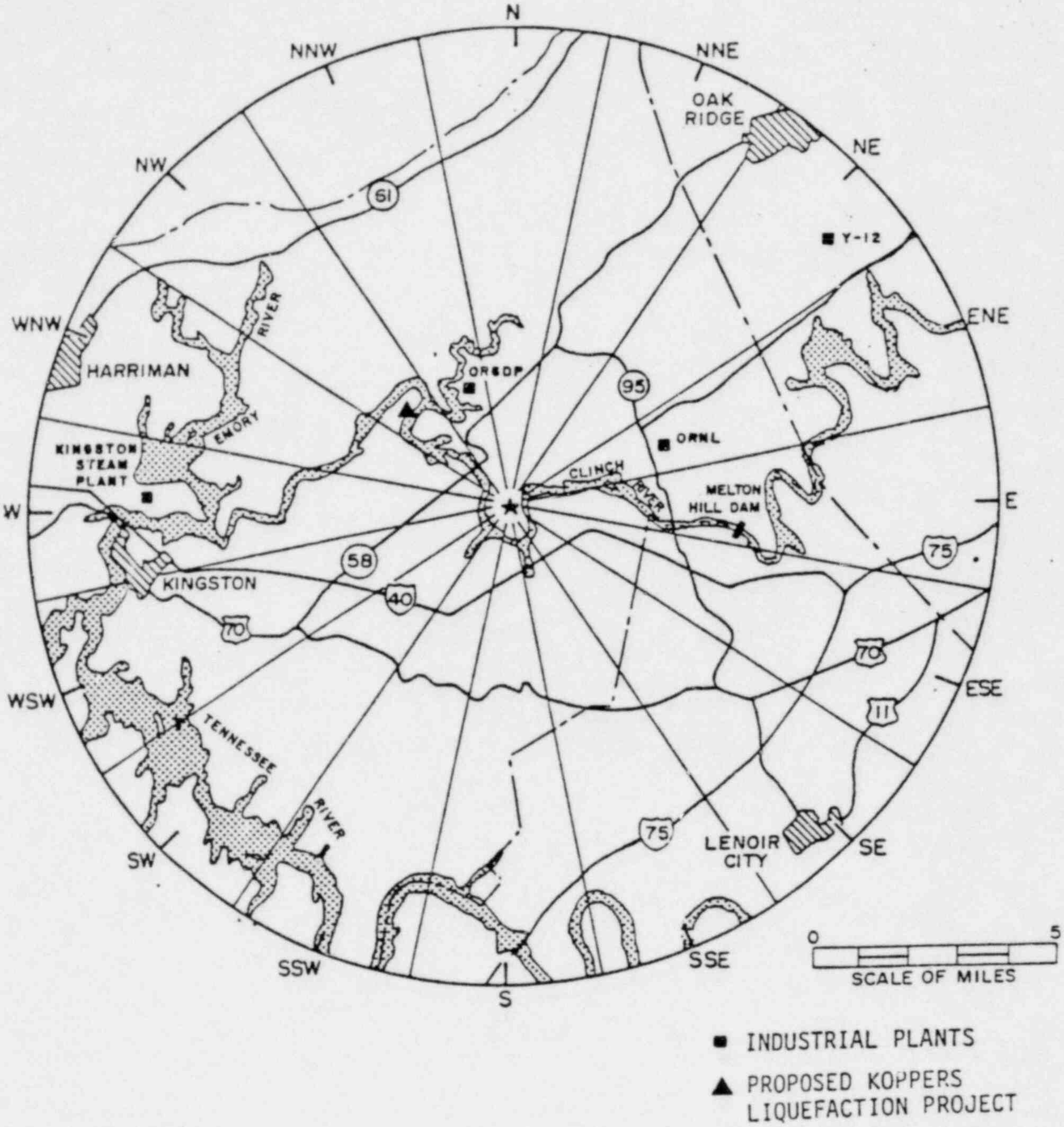

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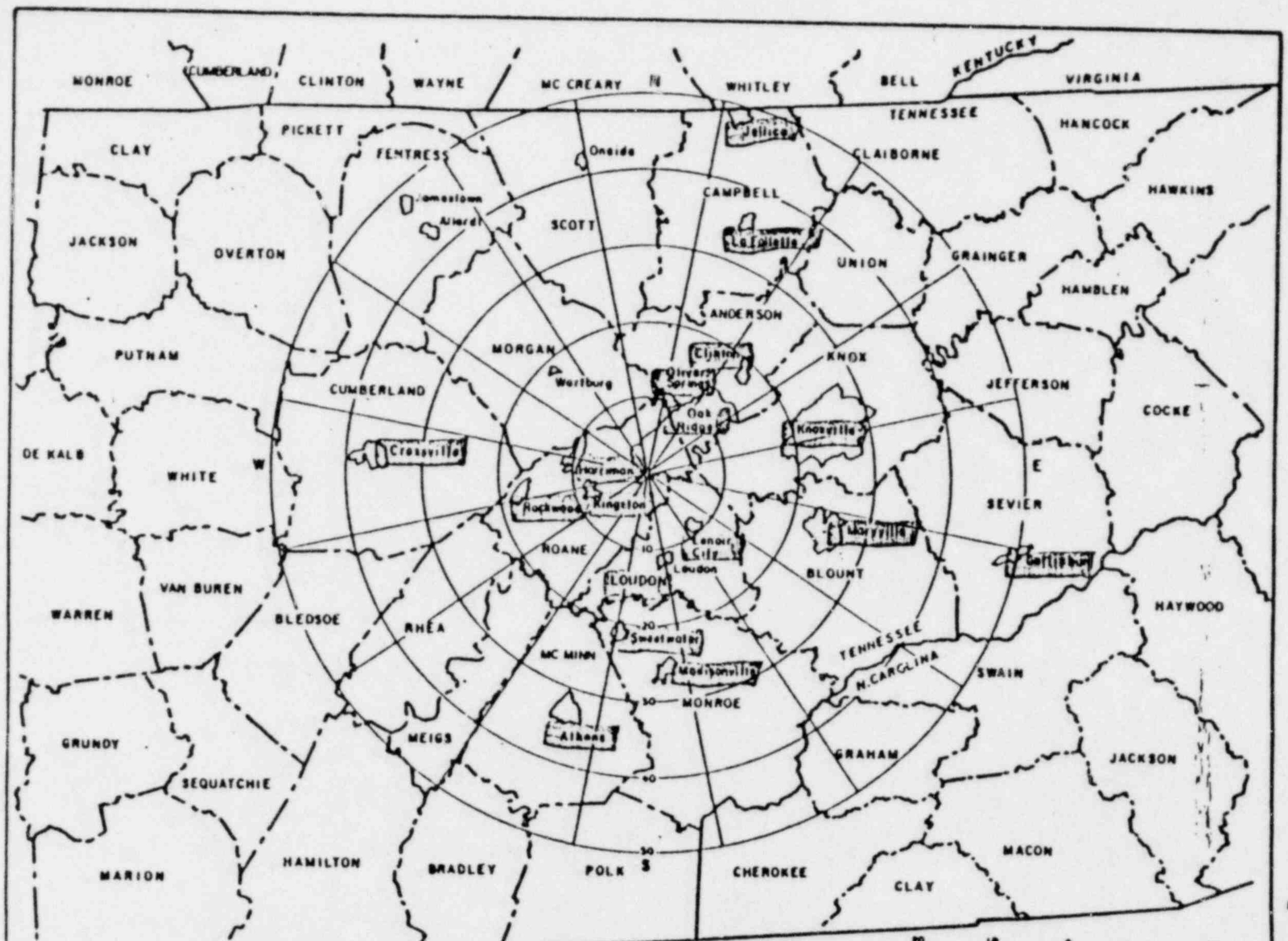
Figure 2.2-10 INDUSTRIAL PLANTS WITHIN 10-MILE RADIUS OF THE CRBRP SITE.

CIRCLE AND SECTOR FIGURE AND STUDY AREA WITHIN 50-MILES OF CRBRP SITE



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1980 RESIDENT POPULATION DISTRIBUTION

0 TO 10 MILES FROM THE CRBRP SITE

Direction	Distance (miles)							
	0 to 1	1 to 2	2 to 3	3 to 4	4 to 5	5 to 10	10-mile Total	
N	0	0	0	0	0	2,000	2,000	
NNE	0	0	0	0	0	4,400	4,400	
NE	0	0	0	0	0	4,500	4,500	
ENE	10	10	0	0	0	3,900	3,920	
E	20	30	50	10	20	4,300	4,430	
ESE	20	30	50	140	120	2,300	2,660	
SE	0	20	50	140	110	7,200	7,520	10
SSE	0	30	40	90	320	2,000	2,480	
S	0	50	50	120	160	1,100	1,480	
SSW	10	30	50	80	90	800	1,060	
SW	20	80	80	110	140	700	1,130	
WSW	20	70	80	140	340	2,800	3,450	
W	0	130	100	110	500	4,400	5,240	
WNW	10	80	170	10	60	4,400	4,730	
NW	30	30	0	10	40	1,700	1,810	
NNW	10	0	0	0	120	1,100	1,230	
Total	150	590	720	960	2,020	47,600	52,040	
Cumulative Total	150	740	1,460	2,420	4,440	52,040		

1980 RESIDENT POPULATION DISTRIBUTION

10 TO 50 MILES FROM THE CRBRP SITE

Direction	Distance (miles)					
	10-mile Total	10 to 20	20 to 30	30 to 40	40 to 50	50-mile Total
N	2,000	4,700	2,200	6,400	7,000	22,300
NNE	4,400	9,100	6,300	17,500	10,300	47,600
NE	4,500	22,100	10,900	6,200	5,100	48,800
ENE	3,920	22,100	100,900	41,800	12,800	181,520
E	4,430	34,400	102,600	34,600	21,300	197,330
ESE	2,660	9,600	43,100	7,000	4,800	67,160
SE	7,520	5,300	6,300	3,700	2,300	25,120
SSE	2,480	5,200	2,400	4,100	6,500	20,680
S	1,480	5,600	7,200	6,200	5,500	25,980
SSW	1,060	3,300	11,200	22,800	9,900	48,260
SW	1,130	2,200	3,600	6,000	10,500	23,430
WSW	3,450	3,400	5,000	6,500	7,200	25,550
W	5,240	12,300	2,600	11,100	4,800	36,040
WNW	4,730	7,800	3,100	5,500	4,500	25,630
NW	1,810	2,400	2,100	3,900	7,900	18,110
NNW	1,230	3,800	1,700	5,600	5,000	17,330
Total	52,040	153,300	311,200	188,900	125,400	830,840
Cumulative Total	52,040	205,340	516,540	705,440	830,840	

10

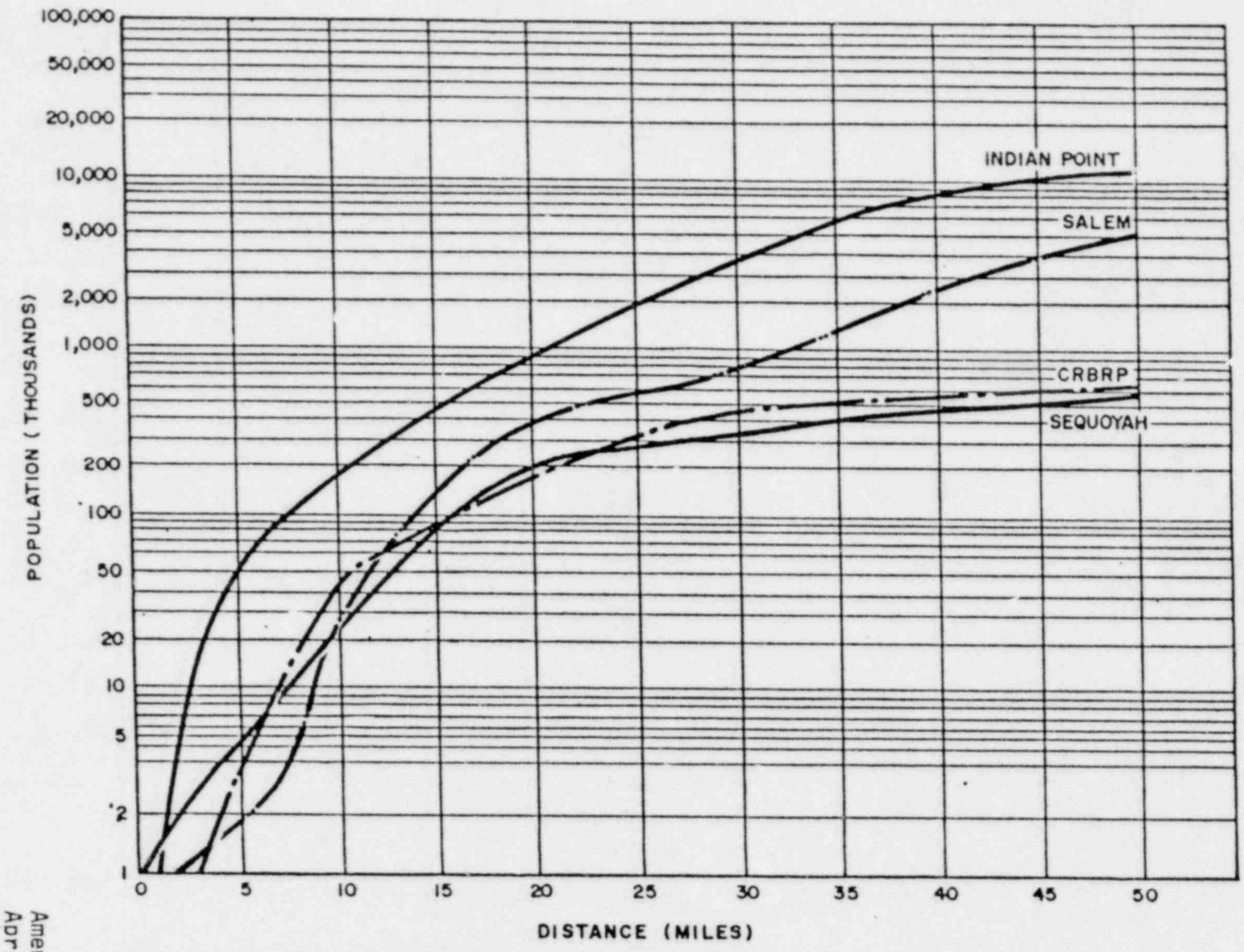

Figure 2.1-6

COMPARATIVE POPULATION DISTRIBUTION SURROUNDING NUCLEAR PLANT SITES

CRBRP SITE SUITABILITY

BRIEFING FOR:

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

SITE SUITABILITY SOURCE TERMS AND NON-RADIOLOGICAL EFFECTS OF SODIUM REACTION PRODUCTS AEROSOLS

PRESENTED BY:
GEORGE H. CLARE
LICENSING MANAGER, CRBRP PROJECT
WESTINGHOUSE
ADVANCED REACTORS DIVISION
JULY 9, 1982

Clare
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THE CRBRP SITE SUITABILITY SOURCE TERM IS COMPARABLE TO THAT USED FOR SITING FOREIGN LMFBRs

	PERCENT RELEASED FROM PRIMARY COOLANT BOUNDARY		
	CRBRP (USA)	CDFR (UK)	MONJU (JAPAN)
• NOBLE GASES	100	100	100
• HALOGENS (AIRBORNE)	50 (25)	50	10
• SOLIDS	1	1	1
• FUEL	1	1	1

NO EQUIVALENT TO THE SSST IS KNOWN TO BE USED IN FRANCE OR GERMANY (FRG).

THE NON-RADIOLOGICAL EFFECTS OF SODIUM REACTION PRODUCT AEROSOLS HAVE BEEN CONSIDERED

- $Na + O_2 \rightarrow NaO_x$
  $NaO_x + H_2O \rightarrow NaOH\ (+\ O_2)$
  $NaOH + CO_2 \rightarrow Na_2\ CO_3\ (+\ H_2O)$
- EFFECTS ON SAFETY RELATED EQUIPMENT ARE ADDRESSED
  - ENVIRONMENTAL QUALIFICATION
  - CONTROL ROOM
  - AEROSOL MITIGATION FEATURES

7-82-2898-6

ANY OFFSITE CONCENTRATION OF SODIUM REACTION PRODUCT AEROSOLS WILL BE LOW

ASSUME:

- STEAM GENERATOR BUILDING DESIGN BASIS LEAK
- 100% OF SPRAY REACTION PRODUCTS AIRBORNE
- ONLY ESF MITIGATION IS EFFECTIVE

EVALUATION:

- DEPLETION IN THE SGB; HAA-3 (440 LB/5 MIN)
- 50% METEOROLOGY; 1×10^{-3} SEC/m^3
- DEPLETION DURING TRANSPORT; 1/100

RESULTS: 7 MILLIGRAMS (NaOH) PER CUBIC METER

7-82-2898-7

CRBRP SITE SUITABILITY

CRBRP PROJECT

BRIEFING FOR:

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

HYDROLOGY

• PROBABLE MAXIMUM FLOOD	R. LEE, TVA
• IMPACT OF NORRIS DAM SITE	T. J. ABRAHAM, TVA
• EFFECT OF CORE MELT ON GROUNDWATER	H. B. PIPER, CRBRP/PO

JULY 9, 1982

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LIQUID PATHWAYS EVALUATION

RADIOLOGICAL CONSEQUENCES ASSOCIATED WITH RELEASE OF RADIOACTIVE MATERIAL TO THE GROUNDWATER FOLLOWING AN HCDA HAVE BEEN EVALUATED IN:

- CRBRP-3, VOLUME 2, "HYPOTHETICAL CORE DISRUPTIVE ACCIDENT CONSIDERATIONS IN CRBRP; ASSESSMENT OF THERMAL MARGIN BEYOND THE DESIGN BASE"
- ER QUESTION/RESPONSE E240.2R

82 2898 2

ANALYSIS OF MELTED-FUEL-MASS LEACH

CRBRP LIQUID PATHWAY ANALYSES SIMILAR TO WASH-1400, WITH THE FOLLOWING EXCEPTIONS:

- CRBRP SITE SPECIFIC FLOW SYSTEM DATA WAS USED
- NO WATER WAS ASSUMED TO BE AVAILABLE FROM THE REACTOR CONTAINMENT VESSEL TO ADD TO GROUNDWATER AT MELT-THROUGH

COMPARISON OF CALCULATED GROUNDWATER EFFLUENT CONCENTRATIONS FOR MOST SIGNIFICANT ISOTOPES AT ENTRANCE TO CLINCH RIVER

	CRBRP		LWR		
NUCLIDE	CONCENT. (μci/cc)	TIME OF PEAK (YRS)	CONCENT. (μci/cc)	TIME OF PEAK (YRS)	MPC (10 CFR 20)
• Sr-90	3.6×10^{-9}	336	7.1×10^{-4}	5.9	3×10^{-7}
• Tc-99	6.8×10^{-8}	45	3.6×10^{-6}	.9	2×10^{-4}
• Pu-239	7.1×10^{-7}	3580	8.0×10^{-7}	535	5×10^{-6}

7.82.2698.4

CRBRP/NRC LIQUID PATHWAY GENERIC STUDY (NUREG-0440) COMPARISON

- CRBRP CONTAINED RADIONUCLIDE SOURCE SIGNIFICANTLY LESS THAN SOURCE USED IN NUREG-0440
  - GENERALLY 2 TO 40 TIMES LESS
- SITE SPECIFIC PARAMETERS ARE SIMILAR.
  - NUREG-0440 USED CLINCH-TENNESSEE-OHIO-MISSISSIPPI RIVER SYSTEM

RADIONUCLIDE SOURCE TERM COMPARISON

ISOTOPE	NUREG-0440 LWR CORE INVENTORY (Ci)	CRBRP CORE INVENTORY END OF CYCLE (Ci)	RATIO $\frac{\text{NUREG VALUE}}{\text{CRBR VALUE}}$
• ^{3}H	5.9×10^4	2.34×10^4	3
• ^{89}Sr	9.2×10^7	1.60×10^7	6
• ^{90}Sr	6.1×10^6	6.79×10^5	9
• ^{90}Y	6.4×10^6	7.11×10^5	9
• ^{91}Y	1.2×10^8	2.04×10^7	6
• ^{95}Nb	1.7×10^8	3.48×10^7	5
• ^{103}Ru	1.4×10^8	5.26×10^7	3
• ^{103m}Rh	1.4×10^8	5.26×10^7	3
• ^{105}Rh	6.7×10^7	3.85×10^7	2
• ^{106}Rh	7.6×10^7	1.96×10^7	4
• ^{106}Ru	5.1×10^7	1.96×10^7	3
• ^{110m}Ag	3.5×10^5	4.33×10^4	8

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RADIONUCLIDE SOURCE TERM COMPARISON

ISOTOPE	NUREG-0440 LWR CORE INVENTORY (Ci)	CRBRP CORE INVENTORY END OF CYCLE (Ci)	RATIO $\frac{\text{NUREG VALUE}}{\text{CRBR VALUE}}$
• ^{111m}Ag	4.3×10^6	2.57×10^6	2
• ^{113m}Cd	1.0×10^3	1.91×10^3	1/2
• ^{115m}Cd	6.2×10^4	3.55×10^4	2
• ^{115}Cd	8.8×10^5	5.46×10^5	2
• ^{123}Sn	9.4×10^5	3.62×10^5	3
• ^{125}Sn	1.5×10^6	7.58×10^5	2
• ^{125}Sb	7.4×10^5	3.96×10^5	2
• ^{125m}Te	2.5×10^5	7.88×10^4	3
• ^{127}Sb	8.3×10^6	3.76×10^6	2
• ^{127m}Te	1.6×10^6	5.40×10^5	3
• ^{127}Te	8.1×10^6	3.69×10^6	2
• ^{129m}Te	6.6×10^6	2.65×10^6	2
• ^{129}Te	3.9×10^7	9.71×10^6	4

RADIONUCLIDE SOURCE TERM COMPARISON

ISOTOPE	NUREG-0440 LWR CORE INVENTORY (Ci)	CRBRP CORE INVENTORY END OF CYCLE (Ci)	RATIO NUREG VALUE / CRBR VALUE
• ^{129}I	2.9	6.7×10^{-1}	4
• ^{131}I	1.0×10^{8}	3.00×10^{7}	3
• ^{132}Te	1.4×10^{8}	4.00×10^{7}	4
• ^{133}I	1.9×10^{8}	5.15×10^{7}	4
• ^{134}Cs	2.1×10^{7}	6.60×10^{5}	32
• ^{136}Cs	5.8×10^{6}	2.65×10^{6}	2
• ^{137}Cs	8.6×10^{6}	1.70×10^{6}	5
• ^{140}Ba	1.8×10^{8}	4.19×10^{7}	4
• ^{140}La	1.8×10^{8}	4.22×10^{7}	4
• ^{141}Ce	1.7×10^{8}	4.29×10^{7}	4
• ^{144}Ce	1.1×10^{8}	2.02×10^{7}	5
• ^{144}Pr	1.1×10^{8}	2.02×10^{7}	5
• ^{238}Pu	2.5×10^{5}	3.29×10^{5}	4/5
• ^{239}Np	2.1×10^{9}	9.48×10^{8}	2

SITE SPECIFIC PARAMETER COMPARISON

PARAMETER	CRBRP SITE SPECIFIC VALUE	NUREG 0440 VALUE
• LENGTH IN FEET FROM CORE BASEMAT MELT POINT TO RIVER.	1600	1500
• AVERAGE SOIL POROSITY	.3	.2
• PERMEABILITY (FLOW VELOCITY)	2000 FT/YR	2446 FT/YR

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CONCLUSION

RADIOLOGICAL CONSEQUENCES ASSOCIATED WITH RELEASE OF RADIOACTIVE MATERIAL TO THE GROUNDWATER FOLLOWING A HCDA ARE:

- LESS THAN THOSE HYPOTHESIZED FOR AN LWR IN NUREG-0440 AND WASH-1400
- COMPARABLE TO 10 CFR 20 EFFLUENT RELEASE LIMITS FOR ROUTINE RELEASES

NORRIS BACKROUND INFORMATION

GRAVITY DAM APPROXIMATELY 1800 FEET WITH A MAXIMUM HEIGHT OF 265 FEET.

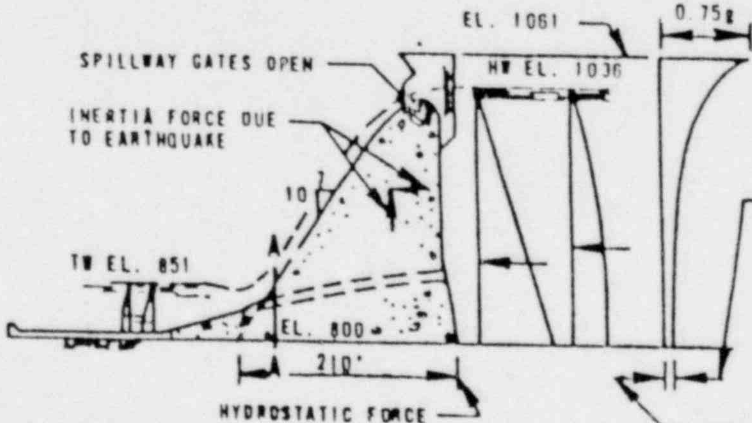
THE DAM IS A SOLID CONCRETE MASS CONCRETE STRUCTURE WITH AN OVERFLOW SPILLWAY, SLUICES AND NONOVERFLOW SECTIONS ON EACH SIDE.

THE DAM WAS COMPLETED IN 1936.

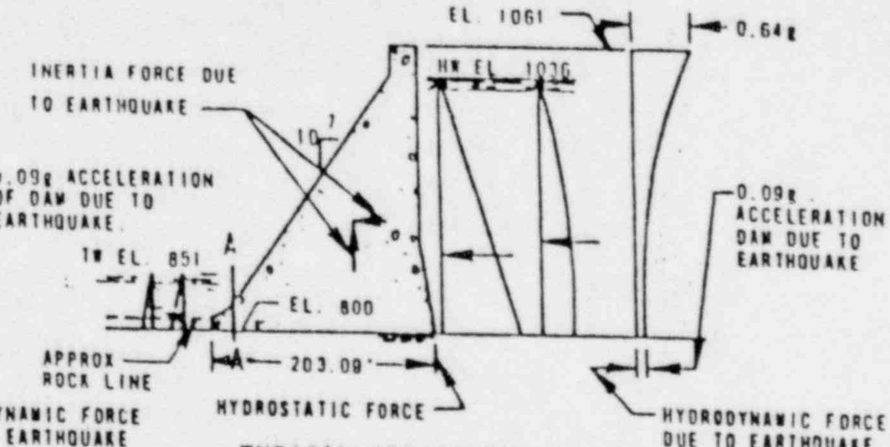
NORRIS DAM WAS ORIGINALLY DESIGNED FOR AN EARTHQUAKE ACCELERATION OF 0.1G THROUGHOUT ITS HEIGHT.

TO ENSURE THE SAFETY OF ITS DAM TVA HAS A WELL DEVELOPED INSPECTION AND MAINTENENCE PROGRAM.

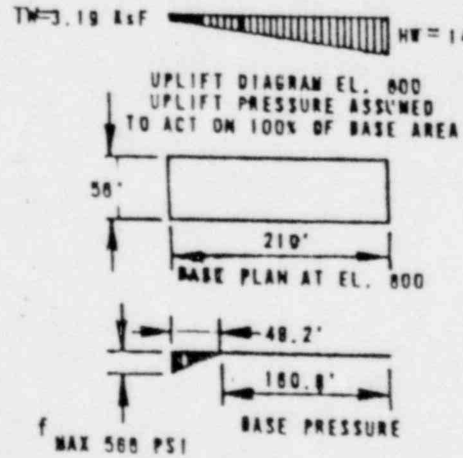
Abraham
T7



TYPICAL SPILLWAY SECTION



TYPICAL NON-OVERFLOW SECTION

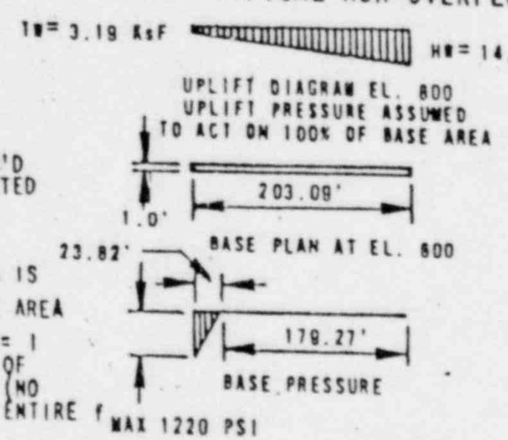


*SHEAR, S, THAT IS REQ'D FOR Q = 1 IS CALCULATED FROM SHEAR-FRICTION FORMULA

$$Q = \frac{0.65\ \Sigma V + SA}{\Sigma H}$$

; A IS ASSUMED TO BE ENTIRE AREA

**SHEAR, S, REQD FOR Q = 1 CONSIDERING PORTION OF BASE IN COMPRESSION (NO TENSION) INSTEAD OF ENTIRE AREA.



NOTES: 1. VERTICAL ACCELERATION OF NON-OVERFLOW AND SPILLWAY AT BASED ASSUMED TO BE 0.06g. BY DYNAMIC ANALYSIS, AMPLIFICATION OF ACCELERATION ABOVE THE BASE WAS DETERMINED TO BE 0.14g AT THE TOP FOR THE NON-OVERFLOW SECTION AND 0.14g AT THE TOP FOR THE SPILLWAY

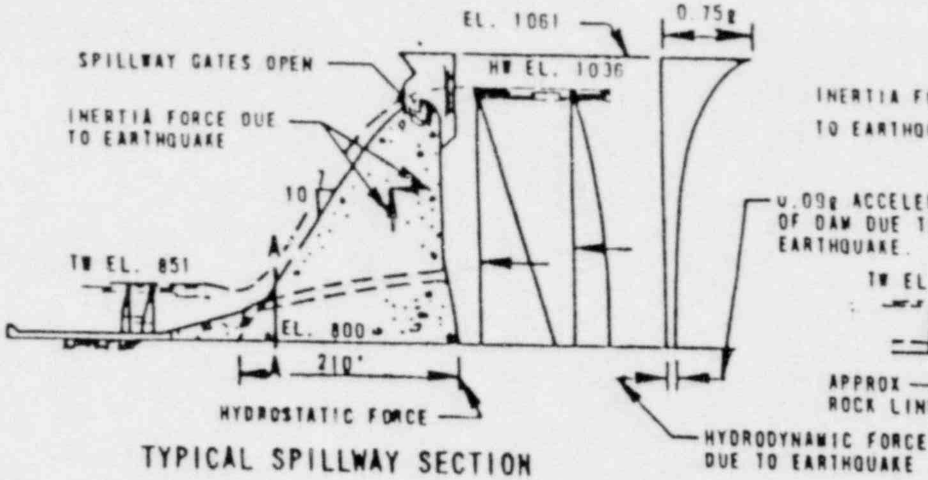
2. HORIZONTAL ACCELERATION OF NON-OVERFLOW AND SPILLWAY AT BASE ASSUMED TO BE 0.09g. BY DYNAMIC ANALYSIS, AMPLIFICATION OF ACCELERATION ABOVE THE BASE WAS DETERMINED TO BE 0.64g AT THE TOP FOR THE NON-OVERFLOW SECTION AND 0.75g AT THE TOP FOR THE SPILLWAY.

3. SPILLWAY GATES ASSUMED OPEN FOR THIS ANALYSIS.

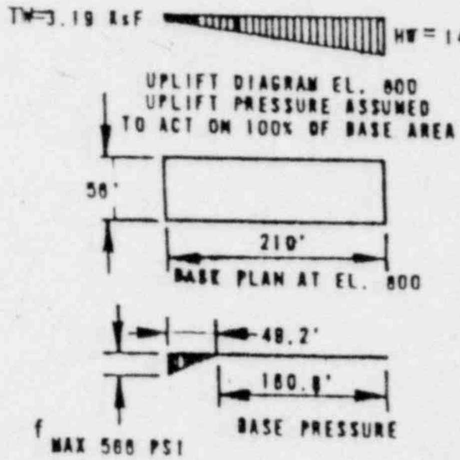
ΣV	ΣH	$\frac{\Sigma H}{\Sigma V}$	AVG SHEAR	S REQD FOR Q=1	f_{MAX}	$FS=\frac{\Sigma MR}{\Sigma MO}$	VERT. SHEAR ON PLANE A-A
112,616K	143,587K	1.28	85psi (ENT. BASE)	49psi* (177 psi)**	566 psi	1.25	247psi

ΣV	ΣH	$\frac{\Sigma H}{\Sigma V}$	AVG SHEAR	S REQD FOR Q=1	f_{MAX}	$FS=\frac{\Sigma MR}{\Sigma MO}$	VERT. SHEAR ON PLANE A-A
2101K	2786K	1.33	85psi (ENT. BASE)	49psi* (415 psi)**	1220 psi	1.03	535psi

Figure 2.4-29. Norris Dam-Spillway & Non-Overflow Results of Analysis for OBE + 1/2 PMF



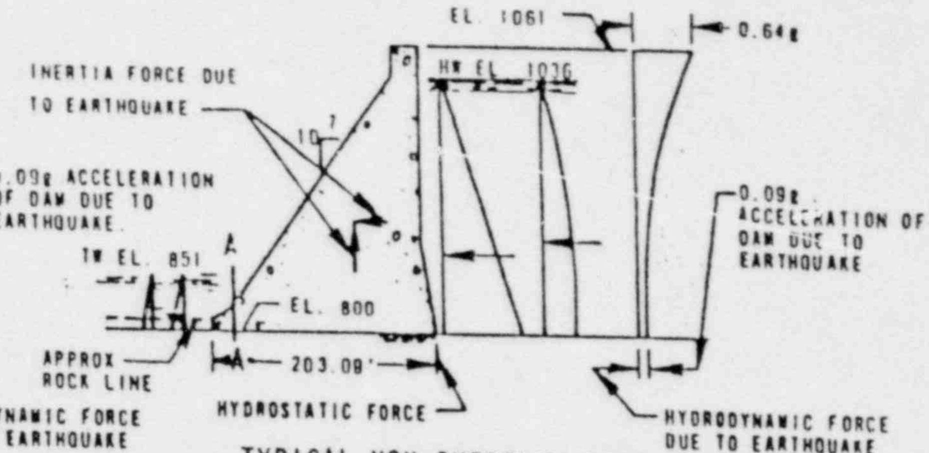
TYPICAL SPILLWAY SECTION



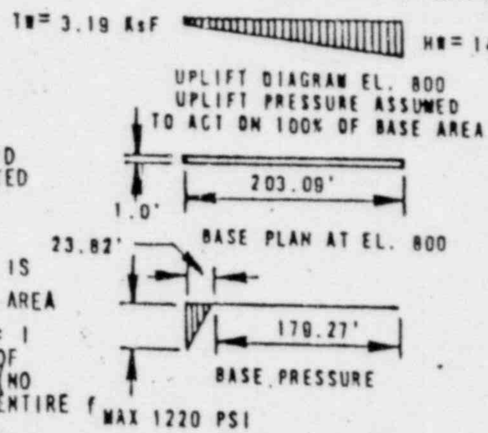
*SHEAR, S, THAT IS REQ'D FOR Q = 1 IS CALCULATED FROM SHEAR-FRICTION FORMULA

$$Q = \frac{0.65\ \Sigma V + SA}{\Sigma H}$$, A IS ASSUMED TO BE ENTIRE AREA

**SHEAR, S, REQD FOR Q = 1 CONSIDERING PORTION OF BASE IN COMPRESSION (NO TENSION) INSTEAD OF ENTIRE AREA.



TYPICAL NON-OVERFLOW SECTION



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2101K	2786K	1.33	95psi (ENT. BASE)	49psi* (415 psi)**	1220 psi	1.03	535psi

Figure 2.4-29. Norris Dam-Spillway & Non-Overflow Results of Analysis for OBE + 1/2 PMF

CONSERVATISMS IN THE SEISMIC ANALYSIS

1. THE CONSERVATIVE PSUEDO STATIC METHOD OF STABILITY ANALYSIS WAS USED. THIS ASSUMES A SUSTAINED RATHER THAN OSCILLATING FORCE.

2. THE AMPLIFICATION OF THE BASE ACCELERATION WAS TAKEN AS THE MAXIMUM FOR ALL PARTS OF STRUCTURE ALTHOUGH THEY ALL DO NOT OCCUR SIMULTATIC ONLY.

3. THE CONCRETE WAS ASSUMED INCAPABLE OF TAKING ANY TENSION.

4. ALTHOUGH THE DAM WAS ASSUMED TO OVERTURN THERE IS INSUFFICIENT ENERGY GENERATED OVER THE SHORT DURATION OF THE LOAD TO OVERTURN THE STRUCTURE.

5. CONSERVATIVE JUDGEMENT WAS USED IN ASSESSMENT OF FAILURE RECOGNIZING NUCLEAR PLANT SITING.

6. TVA'S ASSESSMENT OF NORRIS REGARDING ITS SAFETY PROGRAM IS THAT NORRIS CAN SAFELY WITHSTAND THE MAXIMUM CREDITABLE EARTHQUAKE.

7. OTHER GRAVITY DAMS HAVE BEEN SUBJECTED TO MUCH HIGHER EARTHQUAKE ACCELERATIONS AND HAVE NOT FAILED. FOR EXAMPLE, KONYA DAM IN INDIA. TVA MADE AN ANALYSIS OF KONYA USING THE PSUEDO-STATIC METHOD. RESULTS INDICATED THE DAM TO BE STRESSED MUCH WORSE THAN NORRIS.

NRC STAFF STATUS REPORT

ON UNRESOLVED SAFETY ISSUE (USI), TASK A-45

"SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS"

FOR THE

267TH ACRS MEETING

JULY 9, 1982

ANDREW R. MARCHESE
TASK MANAGER FOR A-45
GENERIC ISSUES BRANCH
DIVISION OF SAFETY TECHNOLOGY, NRR
PHONE: 49-24712

T10

PRESENTATION OUTLINE

- PURPOSE
- OBJECTIVE
- BACKGROUND ON TASK A-45
- UPDATE ON TASK A-45
- MAIN ELEMENTS OF TASK ACTION PLAN A-45

PURPOSE

- THE OVERALL PURPOSE OF TASK A-45 IS TO EVALUATE THE ADEQUACY OF CURRENT LICENSING DESIGN REQUIREMENTS TO ENSURE THAT NUCLEAR POWER PLANTS DO NOT POSE UNACCEPTABLE RISK DUE TO FAILURE TO REMOVE SHUTDOWN DECAY HEAT

OBJECTIVES

- TO DEVELOP A COMPREHENSIVE AND CONSISTENT SET OF DECAY HEAT REMOVAL (DHR) SYSTEM REQUIREMENTS FOR EXISTING AND FUTURE LWRs.
- TO EVALUATE ALTERNATIVE MEANS OF DHR AND OF DIVERSE "DEDICATED" SYSTEMS TO DEAL WITH A BROADER SPECTRUM OF TRANSIENT AND ACCIDENT SITUATIONS

BACKGROUND

- COMMISSIONERS APPROVED SDHR REQUIREMENTS AS AN USI (REF., MEMO, S. J. CHILK TO W. J. DIRCKS, SECY-80-325, DATED DECEMBER 24, 1980
- TASK MANAGER ASSIGNED TO TASK A-45 ON FEBRUARY 17, 1981
- NUREG-0705 (MARCH 1981), "IDENTIFICATION OF NEW USIs RELATING TO NUCLEAR POWER PLANTS - SPECIAL REPORT TO CONGRESS, "PROVIDED AN EXPANDED DISCUSSION OF TASK A-45
- MEMORANDUM, A. R. MARCHESE TO T. E. MURLEY, "ACTIVITIES RELATED TO TASK A-45, "DATED APRIL 8, 1981
- DRAFT TASK ACTION PLAN (TAP) FOR TASK A-45 ISSUED ON MAY 22, 1981
- REVISION 0 OF TAP A-45 (APPROVED BY DST DIRECTOR) ISSUED ON OCTOBER 7, 1981
- REVISION 1 OF TAP A-45 ISSUED ON JUNE 2, 1982

UPDATE ON TASK A-45 SINCE ACRS FULL COMMITTEE MEETING OF SEPTEMBER 10, 1981

- A TASK ACTION PLAN (REV. 0) FOR USI A-45 WAS ORIGINALLY APPROVED BY DIRECTOR, DST, ON OCTOBER 7, 1981
- THIS PLAN, WHICH AUTHORIZED A FOUR-YEAR PROGRAM WITH A COMPLETION DATE OF OCTOBER 1985, WAS NOT APPROVED BY DIRECTOR, NRR
- WE HAVE REASSESSED THIS PROGRAM TO DETERMINE IF THE PRIMARY GOALS COULD BE REALIZED ON A SHORTER SCHEDULE
- WE HAVE NOW DETERMINED THAT OUR PRIMARY OBJECTIVES CAN BE OBTAINED WITH A 30 MONTH PROGRAM
- WE ESTIMATE THAT A DRAFT NUREG REPORT CONTAINING OUR PROPOSED RECOMMENDATIONS INCLUDING ANY PROPOSED NEW REQUIREMENTS, ALONG WITH THE SUPPORTING TECHNICAL AND COST/BENEFIT BASIS, WILL BE AVAILABLE BY NOVEMBER 1984

UPDATE (CONT.)

- REDUCED SCHEDULE OBTAINED BY:
  - DELETING MOST OF WORK ON FUTURE PLANTS, ALTHOUGH ACCEPTANCE CRITIERIA FOR DHRS FOR FUTUTE PLANTS WILL BE DEVELOPED
  - QUANTITATIVE ACCEPTANCE CRITERIA WILL BE BASED ON FREQUENCY OF CORE MELT DUE TO DHRS FAILURES RATHER THAN OVERALL RISK
  - RELYING MORE ON INDUSTRY TO PERFORM MORE PLANT-SPECIFIC EVALUATIONS OF ALTERNATIVE DHRS WHERE THE STAFF CAN SHOW SIGNIFICANT IMPROVEMENTS IN SAFETY
  - HAVING ONE CONTRACTOR WITH OVERALL RESPONSIBILITY FOR PROJECT MANAGEMENT, TECHNICAL DIRECTION AND INTEGRATION, INCLUDING SELECTION AND MANAGEMENT OF SUBCONTRACTORS

UPDATE (CONT.)

- STEPS ACHIEVED TO START WORK ON PROGRAM:
  - RECEIVED APPROVAL BY DIRECTOR, NRR ON MARCH 15, 1982
  - RECEIVED APPROVAL BY SENIOR CONTRACT REVIEW BOARD ON APRIL 9, 1982
  - IMPLEMENTED A CONTRACT ON MAY 3, 1982 WITH SANDIA AS THE LEAD LAB. TO BEGIN WORK & PREPARE A DETAILED PROPOSAL
  - ISSUED REVIEW 1 OF TAP A-45 ON JUNE 2, 1982 THAT IS CONSITENT WITH THE ABOVE

MAIN ELEMENTS OF A-45 TASK ACTION PLAN-REVISION 1

- DEVELOP ACCEPTANCE CRITERIA FOR ASSESSMENT OF DHRS
  - DEVELOP QUANTITATIVE CRITERIA FOR EXISTING PLANTS
  - DEVELOP QUANTITATIVE CRITERIA FOR FUTURE PLANTS
  - DEVELOP QUALITATIVE CRITERIA FOR "SPECIAL EMERGENCIES"
- DEVELOP MEANS FOR IMPROVEMENT OF DHRS
  - PHENOMENOLOGICAL STUDIES
  - CONCEPTUAL DESIGN STUDIES
  - OPERATIONAL ASPECTS OF ALTERNATIVE DHR SYSTEMS
- ASSESS ADEQUACY OF DHRS IN EXISTING LWRs
  - ASSESS ADEQUACY OF DHRS IN SELECTED EXISTING PLANTS ON PROBABILISTIC BASIS
  - ASSESS ADEQUACY OF DHRS IN EXISTING PLANTS ON DETERMINISTIC BASIS
  - GROUP OTHER EXISTING PLANTS FOR ASSESSMENT OF ADEQUACY OF DHRS
- DEVELOP PLAN FOR IMPLEMENTING NEW REQUIREMENTS (E.G., PREPARE NUREG, REG. GUIDE)

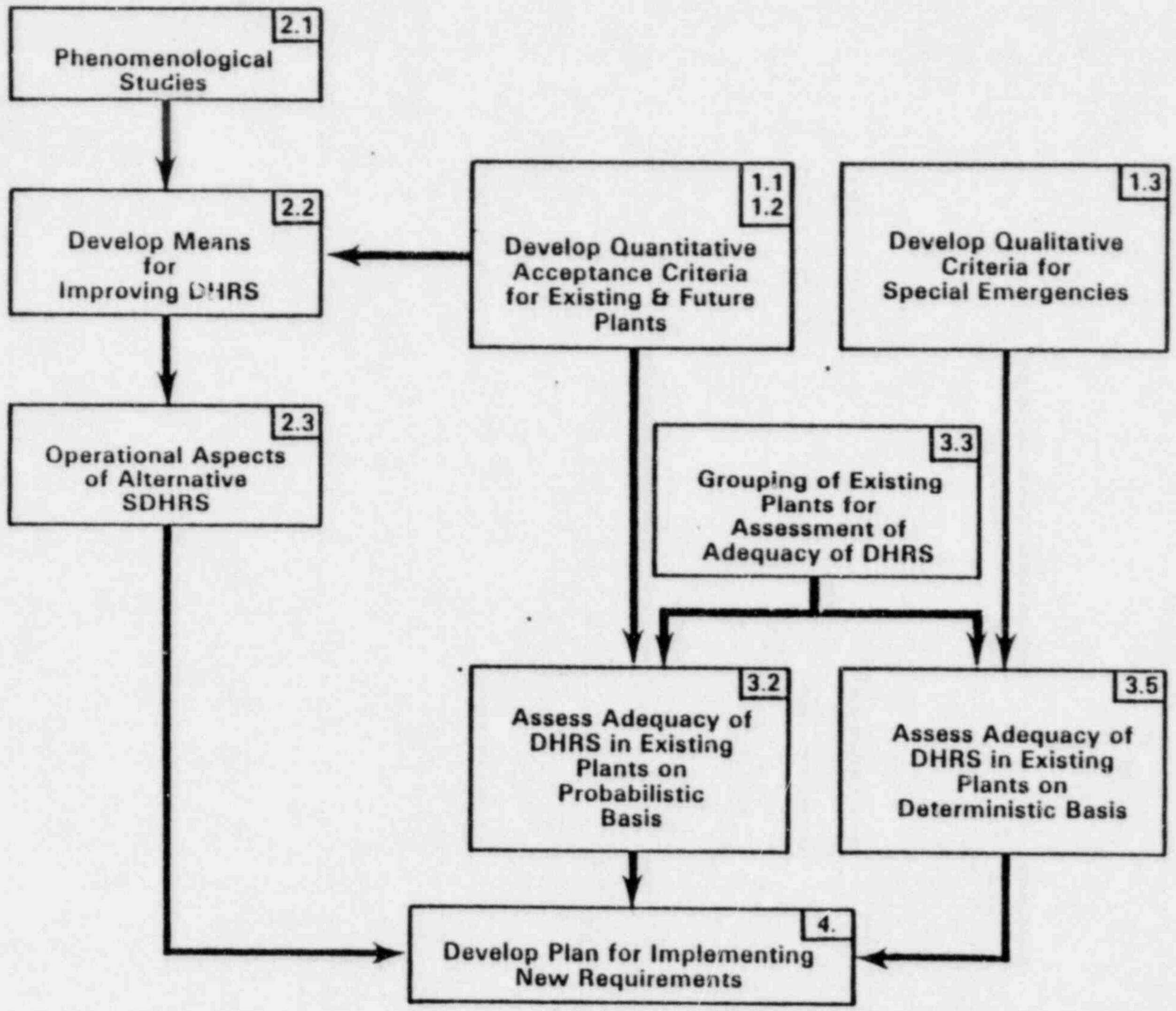
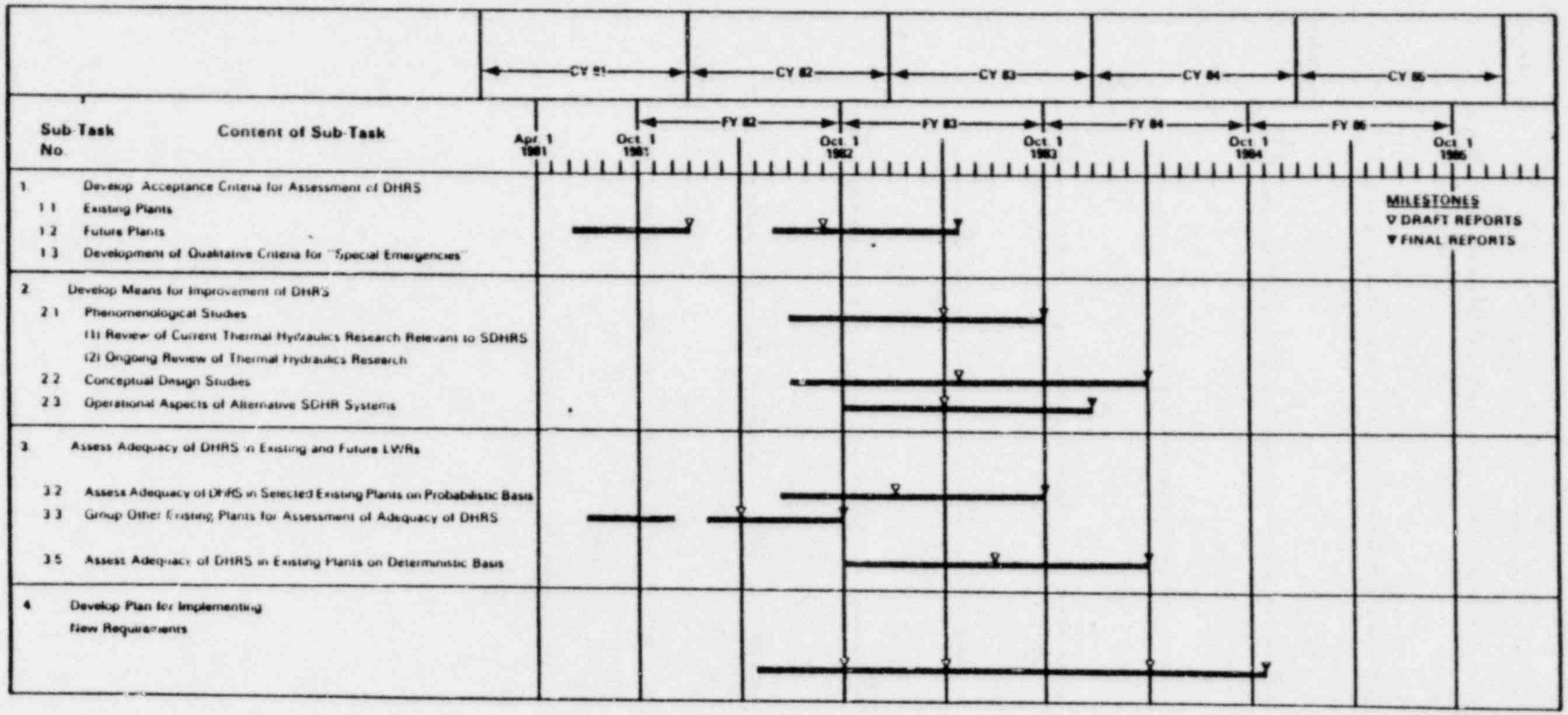


Figure 1. Inter-Relation of Sub-Tasks in Task Action Plan A-45

Figure B-1

DETAILED SCHEDULE FOR TASK A-45, "SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS"



BACK - UP

SLIDES

DEFINITIONS USED IN TASK ACTION PLANT A-45

- REFLOOD PHASE (RFP): THE INITIAL PHASE OF A SEVERE LOCA, WHEN THE OBJECTIVE IS TO REFLOOD THE REACTOR

- SHUTDOWN DECAY HEAT REMOVAL (SDHR) PHASE: THE TRANSITION FROM REACTOR TRIP TO "HOT SHUTDOWN," EXCLUDING THE INITIAL REFLOODING PHASE IN A SEVERE LOCA

- RESIDUAL HEAT REMOVAL (RHR) PHASE: THE TRANSITION FROM "HOT SHUTDOWN" TO "COLD SHUTDOWN" AND MAINTAINING COLD SHUTDOWN CONDITIONS

- DECAY HEAT REMOVAL (DHR) PHASE: SDHR AND RHR PHASES COMBINED

DEFINITION OF DECAY HEAT REMOVAL SYSTEM

IN THE CONTEXT OF TASK A-45, DHR SYSTEM IS DEFINED AS THOSE COMPONENTS AND SYSTEMS REQUIRED TO MAINTAIN PRIMARY AND/OR SECONDARY COOLANT INVENTORY CONTROL AND TO TRANSFER HEAT FROM THE REACTOR COOLANT SYSTEM AND CONTAINMENT BUILDING TO AN ULTIMATE HEAT SINK FOLLOWING SHUTDOWN OF THE REACTOR FOR NORMAL EVENTS, OFF-NORMAL TRANSIENT EVENTS (E.G., LOSS OF OFFSITE POWER, LOSS OF MAIN FEED-WATER) AND SMALL LOCAs (I.E., 1/2" TO 2"). DHR SYSTEM DOES NOT ENCOMPASS THOSE EMERGENCY CORE COOLING COMPONENTS AND SYSTEMS REQUIRED ONLY TO MAINTAIN COOLANT INVENTORY AND DISSIPATE HEAT DURING THE FIRST 10 MINUTES FOLLOWING MEDIUM OR LARGE LOCAs.

MAIN ELEMENTS OF A-45 TASK ACTION PLAN (Oct.'81)

- DEVELOP INTERIM ACCEPTANCE CRITERIA FOR ASSESSMENT OF DHRS
  - EXISTING PLANTS
  - FUTURE PLANTS
  - DEVELOPMENT OF INTERIM QUALITATIVE CRITERIA FOR "SPECIAL EMERGENCIES"

- DEVELOP MEANS FOR IMPROVEMENT OF SDHRS
  - PHENOMENOLOGICAL STUDIES
    - (1) REVIEW OF CURRENT THERMAL-HYDRAULICS RESEARCH RELEVANT TO SDHRS
    - (2) ON-GOING REVIEW OF THERMAL-HYDRAULICS RESEARCH
  - CONCEPTUAL DESIGN STUDIES (GENERIC)
  - OPERATIONAL ASPECTS OF ALTERNATIVE SDHR SYSTEMS

- ASSESS ADEQUACY OF DHRS IN EXISTING AND FUTURE LWRs
  - CATEGORIZE PLANTS AS "EXISTING" OR "FUTURE"
  - ASSESS ADEQUACY OF DHRS IN SELECTED EXISTING PLANTS ON RISK BASIS
  - GROUP OTHER EXISTING PLANTS FOR ASSESSMENT OF ADEQUACY OF DHRS
  - ASSESS ADEQUACY OF DHRS IN SELECTED FUTURE PLANTS
  - ASSESS ADEQUACY OF DHRS IN EXISTING PLANTS ON DETERMINISTIC BASIS

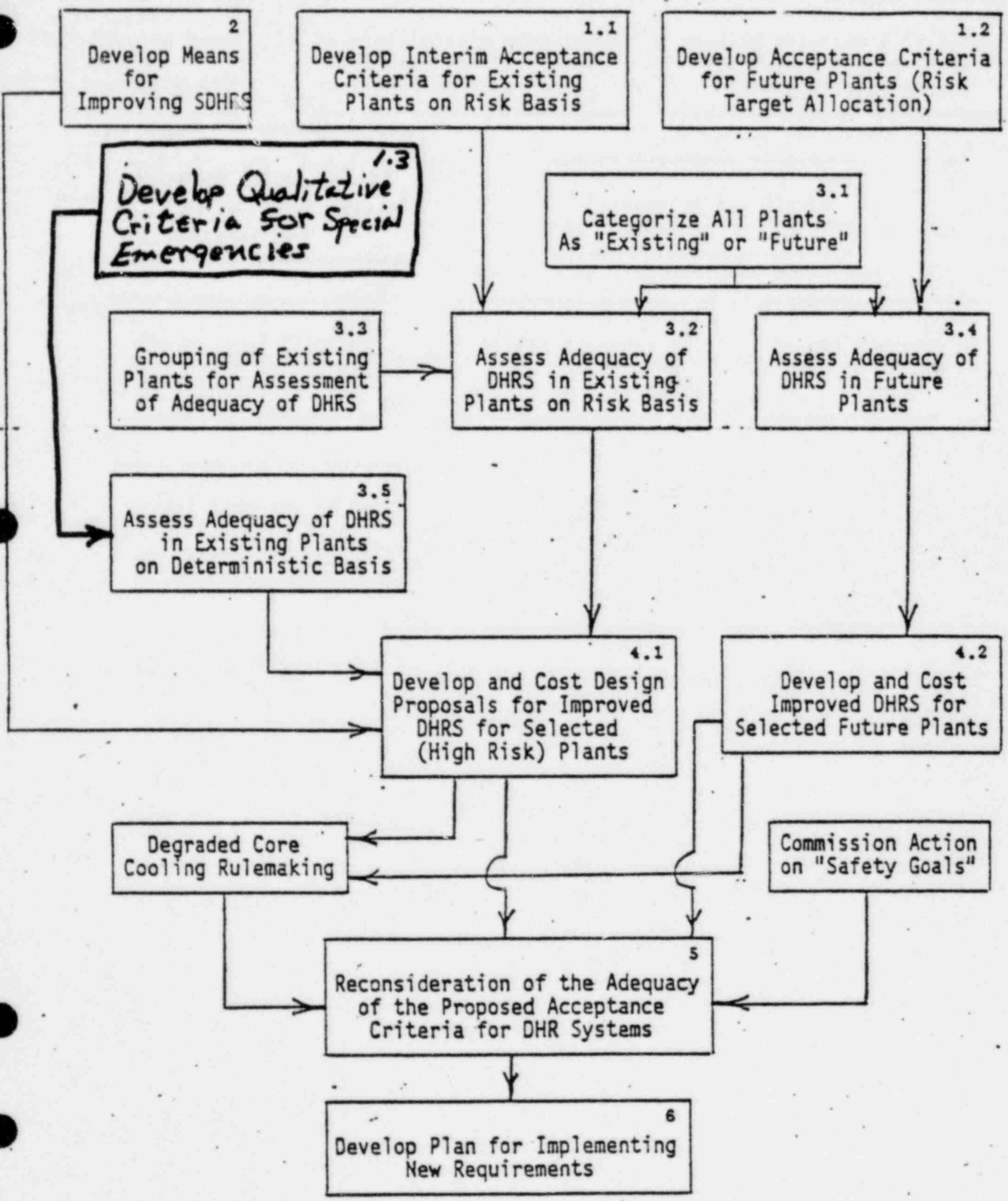
- DEVELOP AND COST IMPROVED DHRS IN SELECTED PLANTS
  - SELECTED EXISTING PLANTS
  - SELECTED FUTURE PLANTS

- RECONSIDER ADEQUACY OF ACCEPTANCE CRITERIA FOR DHRS
  - REVIEW INTERIM DHRS ACCEPTANCE CRITERIA AND TECHNICAL REQUIREMENTS,

    REVISE IF NECESSARY

- DEVELOP PLAN FOR IMPLEMENTING NEW REQUIREMENTS (E.G., PREPARE NUREG, REG. GUIDE)

Figure 1. Inter-Relation of Sub-Tasks in Task Action Plan A-45 - (Oct. 1982)



Legend: X - Identifies Sub-Task Number

MAIN ELEMENTS OF A-45 TASK ACTION PLAN (Feb-82)

- DEVELOP ~~INTERIM~~ ACCEPTANCE CRITERIA FOR ASSESSMENT OF DHRS
  - EXISTING PLANTS
  - FUTURE PLANTS
  - DEVELOPMENT OF ~~INTERIM~~ QUALITATIVE CRITERIA FOR "SPECIAL EMERGENCIES"

- DEVELOP MEANS FOR IMPROVEMENT OF SDHRS
  - PHENOMENOLOGICAL STUDIES
    (1) REVIEW OF CURRENT THERMAL-HYDRAULICS RESEARCH RELEVANT TO SDHRS
    (2) ON-GOING REVIEW OF THERMAL-HYDRAULICS RESEARCH
  - CONCEPTUAL DESIGN STUDIES (GENERIC)
  - OPERATIONAL ASPECTS OF ALTERNATIVE SDHR SYSTEMS

- ASSESS ADEQUACY OF DHRS IN EXISTING ~~AND FUTURE~~ LWRs
  - ~~CATEGORIZE PLANTS AS "EXISTING" OR "FUTURE"~~
  - ASSESS ADEQUACY OF DHRS IN SELECTED EXISTING PLANTS ~~ON RISK BASIS~~
  - GROUP OTHER EXISTING PLANTS FOR ASSESSMENT OF ADEQUACY OF DHRS
  - ~~ASSESS ADEQUACY OF DHRS IN SELECTED FUTURE PLANTS~~
  - ASSESS ADEQUACY OF DHRS IN EXISTING PLANTS ON DETERMINISTIC BASIS

- ~~DEVELOP AND COST IMPROVED DHRS IN SELECTED PLANTS~~
  - ~~SELECTED EXISTING PLANTS~~
  - ~~SELECTED FUTURE PLANTS~~

- ~~RECONSIDER ADEQUACY OF ACCEPTANCE CRITERIA FOR DHRS~~
  - ~~REVIEW INTERIM DHRS ACCEPTANCE CRITERIA AND TECHNICAL REQUIREMENTS;~~
  - ~~REVISE IF NECESSARY~~

- DEVELOP PLAN FOR IMPLEMENTING NEW REQUIREMENTS (E.G., PREPARE NUREG, REG. GUIDE)

DISCUSSION WITH EPRI ON INDUSTRY INVOLVEMENT IN TASK A-45

- ENCOURAGE INDUSTRY COOPERATION AND INVOLVEMENT IN TASK A-45

- OPTIONS TO CONSIDER:

  - INDUSTRY SETS UP ITS OWN PARALLEL PROGRAM, OR

  - INDUSTRY DOES SPECIFIC PARTS OF A-45 ACTION PLAN (E.G., SUB-TASK 4 ON PLANT-SPECIFIC DESIGN OF ALTERNATIVE DHRS)

  - INDUSTRY PEER REVIEW GROUP FOR TASK A-45 MILESTONE REPORTS

- PRIORITY FOR DEVELOPMENT OF CONCEPTUAL DESIGNS FOR IMPROVED DHRS FOR A SPECIFIC PLANT WILL DEPEND ON:

  1. CORE MELT FREQUENCY DUE TO THAT PLANT AND ON THE EFFECTIVENESS OF IMPROVEMENT OF DHRS AS A MEANS OF REDUCING THAT FREQUENCY, AND/OR

  2. CAPABILITY FOR HANDLING "SPECIAL EMERGENCY" SITUATIONS