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NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

DKT/CASE NO. 50-537

TITLE UNITED STATES DEPARTMENT OF ENERGY PROJECT MANAGEMENT CORPORATION - TENNESSEE VALLEY AUTHORITY (Clinch River Breeder Reactor)

PLACE Oak Ridge, Tennessee

DATE December 16, 1982

PAGES 6141 - 6634

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ATOMIC SAFETY AND LICENSING BOARD

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In the Matter of: x

UNITED STATES DEPARTMENT OF ENERGY x

PROJECT MANAGEMENT CORPORATION x

x Docket No. 50-537

TENNESSEE VALLEY AUTHORITY x

(Clinch River Breeder Reactor Plant) x

\- x

Hemlock Room

Executive Seminar Center Building

301 Broadway

Oak Ridge, Tennessee

Thursday, December 16, 1982

The hearing in the above-entitled matter was convened pursuant to adjournment, at 8:00 a.m.

BEFORE:

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GUSTAVE E. LINENBERGER, JR., Member

CADET HAND, Member

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I N D E X

WITNESSES	DIRECT	VOIR DIRE	CROSS	REDIRECT	RECROSS	BOARD EXAM.
THOMAS B. COCHRAN (resumed)						
By Mr. Edgar	6146					
By Mr. Swanson			6171			
By Ms. Finamore				6176		
By Judge Linenberger						6189
JOHN R. LONGENECKER, CARL A. ANDERSON, JR. -and- NARINDER N. KAUSHAL (A Panel)						
By Mr. Edgar	6293					
By Mr. Mizuno			6296			
By Ms. Finamore			6317			
By Judge Linenberger						6380
PAUL H. LEECH, RICHARD A. BECKER -and- JOHN K. LONG (A Panel)						
By Mr. Mizuno	6462					
By Mr. Edgar			6465			
By Ms. Finamore			6468			
By Mr. Mizuno				6491		
By Judge Linenberger						6495

E X H I B I T S

NUMBER	IDENTIFIED	RECEIVED
Applicants':		
55	6150	6289
56	6162	6289
57	6166	6289
58	6293	6406
Staff's:		
20	6172	6289
21	6463	6521
Intervenors':		
22	--	6194
23 and 24	6557	--

ARGUMENT

P R O C E E D I N G S

8:00 a.m.

JUDGE MILLER: Are we ready to resume operations?

Whereupon,

THOMAS B. COCHRAN

the witness on the stand at the time of the evening adjournment, resumed the witness stand and, having been previously duly sworn, was examined and testified further as follows:

CROSS-EXAMINATION (continued)

BY MR. EDGAR:

Q Referring to Page 31 of your testimony and your discussion on Page 31, the Nuclear Safety article, portions of which have been marked for identification at the close of the hearing session yesterday as Applicants' Exhibit 54, I take it you, in citing that in your testimony, you're familiar with the entire contents of the Nuclear Safety article upon which you've relied; is that correct?

A Well, I've read the document.

Q Does the document in question address the frequency of gross containment leakage at values in the range of design basis leakage values?

A Yes, among other things; it addresses all the leakage generically is my reading of the document.

Q May I refer you to Page 619 of Exhibit 54. Does Page 619 of Exhibit 54 -- Is that from the Nuclear Safety article that you relied upon?

A. Well, it looks familiar. It's a short article that I have, if I could check it.

Q Certainly.

A. If you want to supply me with ...

Q Go ahead.

A. Yes. We're looking at the same document.

Q And is it true that the article concerns itself with estimating the frequency of containment leakage, the parameter L_a?

A. Well, I don't recall the precise labeling of the parameter. I would have to go back and refresh my memory, to see if that's a --

Q Let me see if I can help you. Refer to Page 619. In the lefthand column -- and allow me to quote the second sentence, beginning from the top of the page, I quote: "In practice, a value lower than that required to meet the 10 CFR 100 limits is written into the plant's Technical Specifications as the maximum allowable leakage rate (L_a). Any leakage in excess of the maximum allowable rate represents a failure of containment leakage integrity."

Is that a correct quotation?

1-3

A. Yes.

Q. And does the document concern itself with calculation of the failure frequency of containment leakage integrity, as measured by the parameter L_a?

A. Yes. And that's without consideration of the leak rate. I mean, it's a cutoff, anything above that rate is counted; and anything below it is not.

Q. Now that's the Technical Specification rate; is that right?

A. Yes.

Q. And that's lower than the Part 100 rate, or design basis leak rate; is that right?

A. I believe that's correct. I would have to refresh my memory, but I believe that's correct.

Q. All right. And would you agree that that's approximately a factor of ten below the Part 100 leak rate?

A. I don't know the precise factor.

Q. Do you have any reason to believe it's not a factor of ten below it?

A. No, I don't.

Q. Does the article address containment leak rate frequency L_a for both PWRs and BWRs?

A. Yes, as I've indicated in my testimony.

Q. And is it true that PWRs generally show a factor of ten less leakage, in accordance with the tech specs than -- or technical specifications than BWRs?

A. State that again, please.

Q. Is it true that PWRs generally show a technical specification leak rate, which is a factor of ten lower than the corresponding value for BWRs?

A. I don't know.

Q. Would you agree that the CRBR containment concept is similar to that for a PWR?

A. No.

Q. Would you agree that it is more similar to a PWR than to a BWR?

A. Yes, in the broadest sense.

Q. All right. Referring you to Page 34 of your testimony, in particular the paragraph appearing at the top of the page, you make reference in the first three lines to a WASH-1400 estimate of medical capability for supportive treatment.

A. Yes.

Q. Do you know what portion of WASH-1400 you relied upon for that purpose?

A. Well, it's a discussion of the consequences section. I believe it's -- My memory is not terribly good, but I believe it's Volume 6.

But we -- I've got that here.

MR. EDGAR: I have a document that I'd like to have marked for identification as Applicants' Exhibit 55. The document is entitled "Reactor Safety Study: An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants, Appendix VI, Calculation of Reactor Accident Consequences," Nuclear Regulatory Commission, October 1975, consisting of four pages extracted from Appendix VI of WASH-1400.

I'd request that that be marked for identification as Applicants' Exhibit 55.

JUDGE MILLER: It may be marked.

(Applicants' Exhibit No. 55 was marked for identification.)

BY MR. EDGAR:

Q Do you have Exhibit 55 in front of you, Dr. Cochran?

A Yes, I do.

Q Does Exhibit 55 include the material you relied upon in making your statement on Page 34 of your testimony?

A Yes, that's part of the document that I relied on.

Q All right. May I refer you to Page 9-5 of Exhibit 55, in particular the first full paragraph appearing on that page, the third sentence.

Allow me to quote it: "In the event of the worst calculated accident (corresponding to a probability of about 10^{-9} per reactor-year)" --

A. Excuse me. Which line? Where?

Q. The first full paragraph on the page of 9-5, the third sentence.

"In the event" -- I'll quote. "In the event of the worst calculated accident (corresponding to a probability of about 10^{-9} per reactor-year), the number of people receiving a dose in the range of 350 to 550 rads would be about 5000; none would receive a dose above 550 rads. For less severe accidents, these numbers would be smaller, being approximately proportional to the total number of fatalities."

Is that an accurate quote?

A. Yes.

Q. Let me refer you to Page 9-3 of that document. In the fifth paragraph appearing on that page, in the last three sentences of that paragraph -- and I'll quote: "On this basis, it was estimated that 2500 to 5000 people could receive supportive treatment. The advisory group on health effects judged that for such people the $LD_{50/60}$ would be 510 rads. It should be remembered that the supportive treatment is not needed immediately following irradiation but can be started about 20 days later."

Is that an accurate quote?

A. Yes, that's correct.

Q. Now you did not mean to imply in your testimony that the L/D value of 510 rads is associated with any type of accident with a probability approaching those for anticipated transients; is that correct?

A. What reactor are you talking about and what transients and so forth?

Q. Okay. On Page 34 of your testimony you discuss the L/D value of 510 rads; is that correct?

A. That's correct.

Q. And is it true that the 510-rad value is associated with reactor accidents in WASH-1400 with probabilities in the range of 10^{-8} or less per reactor year?

A. I apologize. I'll ask you to state that again.

Q. All right.

Is it true that in WASH-1400, the document you relied upon, that the 510-rad value can be associated with reactor accidents with probabilities of 10^{-8} or less per reactor year?

A. I don't think that I can give a yes or no answer to that. That's -- It's true that those numbers are associated with WASH-1400. Author's estimates of probabilities, however, you must -- You should take into k08

1-8

account the criticisms and even the Commission policy statement that those probabilities aren't reliable.

Q All right.

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Q Let's turn to the question that you address in your testimony of the nuclear explosive.

A. Yes.

Q And let me try to cut through this quickly and just ask you several questions and try to get the record straightened out.

A. Okay.

Q Is it true that the energetic disassembly in a fast reactor would not result in the production of shock waves?

A. I have difficulty with that, answering that with a simple yes or no for the following reasons.

First, there is in my view no theoretical upper limits, sort of, on the energetic or explosive potential --

Q Could I ask you to give me the yes or a no, and then explain later?

A. No, I can't give you a yes or no, and I'm trying to explain why.

JUDGE MILLER: You are asked for a yes or no answer, Dr. Cochran. Your response is that you can't give it.

All right. That's the end of the answer. Go ahead. Next question.

MR. EDGAR: Okay.

2-2

BY MR. EDGAR:

Q You do agree that it would be an error to infer that the energetic disassembly of a fast reactor would result in the production of shock waves; is that correct?

A I can't give a yes or no answer to that for the same reason.

Q All right. Let me refer you to Page 41 of your testimony, the first full paragraph on the page.

A Yes.

Q And the last sentence in that paragraph, and I quote:

> "In any case, my previous testimony at Tr. 2777, 2779, 2785 and 2789 contains an error in inferring that the energetic disassembly of a fast reactor would result in the production of shock waves."

A Yes. The only distinction I'm trying to make is that previously I stated or left the impression that it would occur, or would always occur, and I think that statement was in error.

My hesitation is really associated with two -- in saying it will never occur is associated with simply two factors. One is what happens in the sort of theoretically contemplated range of very, very high energetic events that one would normally associate with extremely low probabilities compared to, let's say, the most likely CDA's, which would be less energetic, or even in the extreme, non-energetic.

And the other sort of caveat I think should be considered is that in the fast reactor community for a number of years, and even still to some degree, there was a debate over whether you could have an energetic fuel coolant interaction analogous to a vapor explosion.

My belief, although I'm not 100 percent positive, is that the rate of transfer of energy in such interaction, if it occurred, could result in shock waves; but I'm not clear.

Also, in some theories, you even had to postulate shock waves to initiate such an event, although that's a weakness in the theory, because people say that sort of situation -- some other experts say that sort of situation is very unlikely to occur.

I think the more recent experimental data would tend to suggest the likelihood of energetic fuel coolant interactions is much less than at least it appeared several years ago when our understanding was less.

Q. All right. Pressures in a chemical high explosive detonation would characteristically build up in a microsecond time scale; is that correct?

A. Repeat that question.

Q. Is it your belief that pressures in a chemical high explosive detonation would build up in a microsecond time scale?

A. In a high explosive chemical detonation, yes; in a low explosive chemical detonation, no.

MR. EDGAR: I move to strike the last part of the answer as non-responsive.

THE WITNESS: There's some difficulty --

JUDGE MILLER: I don't think your question --

MR. EDGAR: The question, Your Honor, with all due respect, I said "chemical high explosive."

The answer came back, "chemical high explosive, yes; low explosive, no."

JUDGE MILLER: Okay. The motion to strike the low explosive portion of the answer will be granted.

BY MR. EDGAR:

Q. And would you agree that the HCDA pressure buildup would be over a millisecond time scale?

A. Certainly, the initial case, with the caveat that if it turned out you had a vapor explosion, the energy transfers which really come subsequent to the nuclear part of the energy production -- or essentially can be separated out from it, that those time scales could be somewhat smaller.

Q. Would you agree that the long-term bubble

2-5

expansion would be the predominant damage mode in an HCDA?

A. Yes.

Q. In terms of physical characteristics, notably time for pressure buildup and peak pressure, is it true that in terms of those characteristics, the explosion or excursion from a CDA is not much like that of an atomic weapon?

A. Certainly not -- well --

Q. In terms of those parameters.

A. Certainly not of an atomic weapon properly designed and functioning as it was intended.

Q. Do you agree that the direct heat removal service system is located in the containment building of CRBR and not in the steam generator building?

A. I don't know the precise location of it.

Q. Would you turn to Page 15 of your testimony, and in particular A.11, the portion of that answer starting in the fifth sentence, which begins with the words, "Unlike an LWR."

A. Yes.

Q. In regard to that opinion expressed in that paragraph, have you taken into account the existence of systems in Clinch River which have design characteristics to accommodate the sodium-water reaction?

A. Well, I think the statement is true, taking

2-6

those systems into account.

Q Did you take them into account when you wrote that portion of the testimony?

A. Well, I was certainly aware of the systems.

Q You say now that you were aware of the systems. Were you familiar with their general design and performance characteristics?

A. No.

Q So you had no knowledge on the system level of the effects of those systems?

A. Well, I wouldn't say no knowledge. I had knowledge, limited knowledge of, for example, the GAO reports that discuss -- the GAO report that I attached in my testimony discusses the fact that there are -- there is a potential problem, potential possibility of failure.

I have some familiarity with CRBRP-1. It discusses the fact that these types of steam generator failures are something that should be considered in sort of the fault tree/event tree analysis in analyzing the over-all failure rate of the system.

MR. EDGAR: I move to strike the answer as non-responsive. The question was knowledge of the systems and we got back GAO.

JUDGE MILLER: Stricken; it is unresponsive.

2-7

BY MR. EDGAR:

Q Dr. Cochran, it is true that you are not familiar with the systems and their general characteristics in Clinch River for accommodating the sodium-water reaction, is it not?

A Not in detail, no.

Q Are you familiar with the general design and performance characteristics?

A I answered that question already.

MR. EDGAR: I demand an answer. I would move to compel an answer.

MS. FINAMORE: Objection. Asked and answered.

JUDGE MILLER: I beg your pardon? What did you say?

MS. FINAMORE: I said objection, asked and answered.

JUDGE MILLER: I don't recall it being answered.

THE WITNESS: It was yesterday.

JUDGE MILLER: Well, my memory doesn't go that far.

THE WITNESS: My answer is the same. No.

JUDGE MILLER: It's easier to say, "No," than to get into this asked and answered business, by the way.

Asked and answered is not a very significant kind of objection, so let's try to save some time.

Go ahead.

BY MR. EDGAR:

Q Dr. Cochran, in regard to the GAO report that's attached as an attachment to your testimony, specifically Attachment 2, is there any specific discussion in the GAO report concerning the question of sodium-water reaction and its effect on the frequency of loss of heat sink in CRBR?

A No.

Q Do you know whether lightwater reactors have a system which is comparable in functional and design characteristics to the direct heat removal service?

A No.

Q Do you have any reason to believe that lightwater reactors have such a system with comparable design and performance characteristics?

A I don't believe they do.

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Q Referring you to Page 35 of your testimony, you refer here in the top of the page to a California Underground Siting Study and an attachment to a letter dated 21 February 1979 from Bryce Johnson, Peter Davis and Hong Lee to the Honorable Morris Udall.

A Yes.

Q I have a copy of a letter, Your Honor.

MR. EDGAR: I would request that this be marked for identification as Applicants Exhibit 56. The letter is dated 21 February 1979. It is addressed to the Honorable Morris Udall. It is signed by Messrs. Bryce Johnson, Peter Davis and Hong Lee.

JUDGE MILLER: It may be marked.

(Applicants Exhibit No. 56 was marked for identification.)

BY MR. EDGAR:

Q Do you have Exhibit 56 in front of you, Dr. Cochran?

A No, I do not.

Q I'm sorry. I will furnish you one.

(Witness handed document.)

BY MR. EDGAR:

Q Do you have Exhibit 56 in front of you?

A Yes, I do.

Q And is Exhibit 56 -- does it include the cover letter transmitting the California Underground Siting Study to the Honorable Morris Udall?

A. Yes.

Q And is that the cover letter to the document which you cite in your testimony?

A. Yes, it is.

Q Referring you to Page 347 of that document, Applicants Exhibit 56, the next to last two paragraphs and I quote:

"The differences between the biological effects listed in WASH-1400 and those used for the California study were not sufficient to change any conclusions of our study."

Paragraph. Continuing the quote:

"In summary, although some variation exists between the data and methodology of the WASH-1400 study and that of the California Study, the differences are considered to be minor. Some are simply due to the availability of more and improved data at the time

3-3

of the California Study. In the portions of WASH-1400 reviewed by our study, we found no errors which were of sufficient magnitude to make an appreciable impact on computed public consequences. The availability of the data and methodology of WASH-1400 provided a considerable amount of information directly relevant to our study, and the existence of such information was of material benefit."

Is that an accurate quotation?

A. Yes.

Q. Is that an accurate copy of the letter transmitting, to the best of your knowledge and belief, thr Underground Siting Study to Representative Udall?

A. Yes.

Q. Does that represent the views of the authors of the California Underground Siting Study?

A. Certainly the views that they expressed in the cover letter, yes.

Q. Do you have any reason to believe that those authors have different views at this time?

A. None.

Q Referring you to Page 35, again, after the discussion of the Udall letter in your testimony, you discuss the Accident Evaluation Code, AEC.

Do you see that reference?

A. Yes.

Q And in that regard, you reference an SAI report, December 19, 1978, Pages 3-6 and 3-8.

Is that correct?

A. Yes.

Q Would you agree that the 350 rem value cited in your testimony for the AEC code is for whole body?

A. Yes. I believe that's correct.

Q And would you agree that the 510 value in the CRAC code is for total bone marrow?

A. Yes, they ought to be the same,approximately.

Q Excuse me.

A. I would infer that the bone marrow and whole body dose ought to be fairly close to one another.

Q Well, the question is, what does the CRAC code use?

Don't they assign the value of 510 to total marrow?

A. Well, just a minute.

MR. EDGAR: I'd like to have marked for identification as Applicants Exhibit 57,a document

3-5

entitled Final Report on -- or portions of a document entitled Final Report On Comparative Calculations For The AEC and CRAC Risk Assessment Codes, Science Applications, Inc., Palo Alto, California.

It is or consists of portions of the document dated December, '78, referenced at Page 35 of Dr. Cochran's testimony.

JUDGE MILLER: It may be marked.

(Applicants Exhibit 57 was marked for identification.)

BY MR. EDGAR:

Q. Do you have Exhibit 57 before you?

A. I do.

Q. And does Exhibit 57 contain Pages 3-6 and 3-8, which are specifically referenced in your testimony on Page 35?

A. Yes, they do.

Q. And are those accurate copies of the pages that you relied upon?

A. Yes, they are.

Q. And does Page 3-6 assign the 510 value to bone marrow for the CRAC code?

A. That's correct.

Q. And is the 350 rem value associated with whole body for the AEC code?

3-6

A. That's correct.

Q. And is it true that the authors of this comparison believe that it would appear appropriate to insert less conservative fatal dose criteria for the calculation of early fatalities in the AEC code?

A. You're referring to the SAI authors?

Q. That's correct.

A. You have misrepresented their conclusion which is on Page 5-2 of the report, and I will read it in full:

> "It would appear appropriate to insert less conservative fatal dose criteria for the calculation of early fatalities in the AEC code provided the assumption of supportive medical treatment for exposed persons is reasonable for the locale considered in the analysis."

Q. Do you agree with that conclusion that you have just cited?

A. Not entirely.

Q. Do you have any specific information that you have developed which indicates that the assumption of supportive medical treatment for exposed persons would be unreasonable for the specific locale of Clinch River?

A. Well, there's no anlysis of that issue given in the environmental impact statement, so one can't draw a reasonable conclusion one way or the other.

Q. Have you done any analysis --

MR. EDGAR: I move to strike that answer as non-responsive.

JUDGE MILLER: It may be stricken.

BY MR. EDGAR:

Q. Have you done any specific analysis of the question of supportive medical treatment in the locale of Oak Ridge?

A. No , I have not.

Q. Have you reviewed any reports by Mr. Harris concerning the issue of pipe rupture probability, other than the report which you cite and attach to your testimony?

A. Yes.

Q. And what report have you reviewed?

A. As I tried to indicate yesterday, there was a earlier version of this analysis or an interim report done approximately a year or perhaps somewhat earlier, specifically for the CRBRP and then this -- this is a rather extensive document by Harris and others on PWR pipe fractures.

Q. Okay.

Can you identify the document, the latter document concerning PWR pipe fracture?

A. I have it with me.

Q. What I'd like to get is an identification of the title and description of the document.

A. I brought portions of two volumes with me.

They are actually two different documents. One NUREG/CR-2301, UCRL-15490-RM, titled Fracture of Mechanics Models Developed for Piping Reliability Assessment in Lightwater Reactors, Piping Reliability Project. Prepared by D.O. Harris, E.Y. Lim, D.D. Dedhia of Science Applications, Inc. and H.H. Woo and C.K. Chow of the Lawrence-Livermore National Laboratory.

This is a Livermore document prepared for the Division of Engineering Technology, Office of Nuclear Regulatory Research, U,S. Nuclear Regulatory Commission.

It's dated June, 1982.

The second document for which I have brought only portions of Volume 5, is NUREG/CR-2189, Vol.5, UCID-18967, Vol. 5-RM, titled Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant.

Prepared by D.O. Harris, E. Y. Lim, E.D. Dedhia, of Science Applications, Inc. Also a Lawrence-Livermore Laboratory document, prepared for the Division of Engineering Technology, Office of Nuclear

Regulatory Research, U.S. Nuclear Regulatory Commission. Date published, August, 1981.

Q. Thank you.

Are you aware of any specific regulatory requirement of NRC which requires completion of a probablistic risk assessment prior to issuance of a construction permit?

A. No, not --

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MR. EDGAR: May I have a moment to confer? I think we may be near --

JUDGE MILLER: Yes.

(Pause.)

MR. EDGAR: No further questions at this time.

JUDGE MILLER: Thank you.

Staff?

MR. SWANSON: Yes, just a few.

CROSS-EXAMINATION

BY MR. SWANSON:

Q. Just a few minutes ago, Dr. Cochran, you were asked by Applicants whether or not you had read any earlier Harris reports, and you mentioned one about a year or so earlier.

Is that the report that's mentioned as Reference 1 at the end of your Attachment 3, a 1977 report?

A. I believe it is. Let me just check to see -- to confirm that.

It may take me longer to find it. If you've got a copy.

Q. Yes, I have a copy.

MR. SWANSON: I would like to have marked as Staff Exhibit 20, while Dr. Cochran is looking over, a document entitled "A Note on the Pipe Rupture Probability Calculations for the Primary Heat Transport System of CRBRP."

It's by D. O. Harris of Science Applications, Incorporated. The date is Octcber 7, 1977.

(Staff Exhibit No. 20 was marked for identification.)

THE WITNESS: Yes. That's the other document that I was referring to.

MR. SWANSON: . would ask the Board to mark that as Staff Exhibit 20.

JUDGE MILLER: It may be marked.

BY MR. SWANSON:

Q. In that earlier report by Mr. Harris, did he reach a conclusion about the absolute pipe rupture probabilities for CRBR?

A. He presents some probabilistic results. I have no basis for knowing whether he believes the absolute numbers have significance, as opposed to the comparative ratio.

Q. Let me refer you to Page 4, the bottom paragraph. I'll read you the first sentence: "Review of Results: The pipe rupture probabilities for CRBR estimated using the above outlined techniques were 10^{-8}/plant-year for the cold leg, and 10^{-7}/plant-year for the hot leg," and he gives a reference.

Did I read that sentence correctly?

A. Well, you didn't read the whole sentence, but --

Q. Okay. I'll continue with the reference completely. "(see Ref. 1, page III-116)."

Now have I read the sentence correctly?

A. Yes. That's CRBRP-1, I take it to mean.

Q. Thank you.

At the top of Page 24 of your testimony, you mention -- actually your discussion begins on the prior page, Page 23, about system interaction. At the top of Page 24 you mention a series of accidents.

A. Yes.

Q. You refer to the TMI-2 accident. Do you know what the leakage path from the TMI-2 reactor was during that accident?

A. Through the pressure relief valve.

Q. The Crystal River LOCA that you mention in 1980, was the leakage path from that reactor -- in that accident the same pathway, through the pressure relief valve?

A. It may have been. I don't recall. These accidents were examined in one of the documents I referred to on Page 23. I would have to go back and -- I have not made a detailed analysis of any of these and would have to go back and refresh my memory on the basis of -- I believe it's the Brookhaven analysis.

Q. So your testimony right now is that you don't know whether or not that was a pressure relief valve accident?

A. No, I don't. I would have to refresh my memory to recall the details of these accidents. These accidents were -- Well, I've already said that.

Q. Do you know if the accident -- I'll just spell it -- B-e-z-n-a-u -- the accident at that reactor, do you know if that was also the result of a stuck open pressure relief valve?

A. That's my recollection, yes.

Q. Do you happen to know whether or not the systems level design right now for Clinch River includes a pressurizer relief valve?

A. I would hope not. No. It doesn't include one.

MR. SWANSON: That's all the questions we have.

JUDGE MILLER: Thank you.

Redirect?

MS. FINAMORE: Yes. I'd like an opportunity to confer with the witness.

4-5

THE WITNESS: I'd like a little longer opportunity.

MS. FINAMORE: May we have a short break?

JUDGE MILLER: All right. Five minutes.

(A short recess was taken.)

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9:00 a.m.

JUDGE MILLER: All right. Are you ready for redirect?

MS. FINAMORE: Yes.

REDIRECT EXAMINATION

BY MS. FINAMORE:

Q. Dr. Cochran, you stated yesterday that you did not have any affirmative evidence that the natural circulation capability proposed by Applicants would in fact be reached.

Can you explain what you meant by "affirmative evidence"?

A. Well, you won't know whether the natural circulation will work as designed until you test the system and you can't test the system until you build the reactor.

I think in a large measure that's probably why the Staff is reluctant to sign off on natural circulation at the early stages of the design process.

Q. You were also asked to read a sentence from your Attachment 2, regarding the opinions of the GAO technical consultant on steam generators.

Did that sentence read constitute a complete opinion in your mind of the technical consultant?

A. No, I don't think that sentence should be

5-2

read without also recognizing the other opinions that the GAO states that the consultants gave the GAO.

For example, on Page 7 at the bottom, it says:

"Our consultant recognizes..." this is the last four lines "...recognizes the potential problems in the areas of structural integrity and the ability of the CRBR steam generators to withstand temperature changes. He also acknowledges that the planned tests will not provide adequate data in these areas."

Well, I'll just leave it at that. I mean, the document speaks for itself.

Q. You were asked yesterday about any subsequent work of Dr. Harris regarding pipe breaks of which you were aware, and you mentioned a subsequent report.

Can you tell me what impact, if any, that report might have on the conclusions in your testimony regarding pipe breaks?

A. I made an effort to determine that by calling the principal author, Dr. Harris, and in our telephone conversation he stated to me -- I asked him if he had read my testimony and he said he had. I asked him --

JUDGE MILLER: Just a moment. I don't hear --

MR. EDGAR: I'm going to --

JUDGE MILLER: -- an objection, but this is obviously not only hearsay, but hearsay of a type which can't be verified or tested by cross-examination.

Is this what you permit --

MR. EDGAR: It's totally unreliable. You know, it's one thing if a document came in. There's some reasonable response permitted, but this is gross hearsay, and totally non-reliable, non-probative evidence to which Applicants and Staff have no ability to respond.

I move to strike it and object to the whole line of questioning.

MR. SWANSON: There's another basis for that, too. I realize that motions to strike are not appropriate yet, but Dr. Cochran has admitted he's not an expert in pipe break analysis; therefore, he's not qualified to analyze or interpret the opinions of others who are experts in that area.

So he certainly can't be permitted leeway in that area, to go into hearsay where experts might otherwise be allowed to do so.

MS. FINAMORE: I would disagree with that statement.

JUDGE MILLER: We will defer a ruling upon the expression of opinions by Dr. Cochran on this or other fields.

We will, however, sustain the objection to the partly reported telephone conversation; not only hearsay, which itself is not necessarily grounds for inadmissibility, but because of the inability to determine its reliability and its not being subject in any way to cross-examination, or testimony arising at this late date not being subject to previous disclosure in prepared written testimony.

The motion will be granted. That portion of the testimony which purports to go into the telephone conversation with someone else will be stricken.

BY MS. FINAMORE:

Q Dr. Cochran, based on your understanding of the subsequent work of Dr. Harris, what effect, if any, would that subsequent work have on the conclusions in your testimony?

A I don't believe it would have any. I think the -- and I think the conclusions one could draw by reading the exhibit of Dr. Harris' analysis would still be correct.

Q Dr. Cochran, you were asked yesterday about a statement in the Harris Report on Page 10 of that report, regarding the failure rate of primary piping in the CRBR, and as 0.121.

In connection with the statement in your testimony on Page 22 that the CRBR pipe break frequency may be

5-5

as much as 12 times higher than that in the PWR, can you explain whether or not these two statements are consistent and why?

MR. EDGAR: Objection. Asked and answered.

JUDGE MILLER: Overruled.

THE WITNESS: I believe there's no inconsistency, and I believe the document speaks for itself. One should simply -- I would simply refer you to the discussion that begins at the top of Page 9:

"The results in Table 1 show a wide range of values varying from .0186 to 11.62," which I rounded off to twelve, "(i.e., three orders of magnitude)."

Then it continues, and beginning with the last sentence on that page:

"With the present state of knowledge, it's not possible to ascertain the controlling parameters."

Then it goes on to say, discarding well length, you would get a different range of parameters, .1 to 1, and that's obvious from the data presented in Table 1 at Page 8.

JUDGE LINENBERGER: Excuse me. The record may be clear, but I'm not.

We've talked about -- Intervenors' Counsel has talked about a Harris document, and we've mentioned several in the course of these discussions.

The witness referred to Pages 8 and 9 of something, and I don't know what "something" is or which Harris document we might be discussing.

THE WITNESS: I'm sorry. I'm referring to the November 13th, 1978, document that's attached to my testimony.

JUDGE LINENBERGER: Thank you.

BY MS. FINAMORE:

Q Dr. Cochran, you were asked today about the microsecond time scale and pressures which you felt occurred in high chemical explosives.

Could you explain what you meant by "high chemical explosives"?

A I think that can probably best be done by referring to a book by Melvin A. Cook, entitled, THE SCIENCE OF HIGH EXPLOSIVES."

JUDGE MILLER: Just a minute. I don't want to get into a lot of books and things now.

You are purporting to be an expert, Dr. Cochran, and we want your views. We are not going to look elsewhere or have testimony that incorporates by reference.

I say this now with applicability to all witnesses, not just you.

Go ahead and answer the question.

THE WITNESS: Well, explosives -- some authors distinguish high explosives from low explosives in terms of the time period over which the product, the chemical reactions take place.

High explosives, which would include explosive substances such as Compound B or PETN that I believe was mentioned in previous testimony, would have reaction rates in the range that would fit in the cross-hatched area represented by the Applicants' Exhibit -- I believe it's 46.

There are other so-called low explosives that would be more akin to gun cotton or black powder, or basically propellants, whose periods would be in the millisecond range and peak pressures would be much lower than those for the high explosives.

Furthermore, they are sometimes referred to as deflagrating explosives, and they don't necessarily produce shock waves.

There was testimony by the Applicants to the effect that chemical --

JUDGE MILLER: Wait a minute. I don't think you were asked to compare anybody else's testimony.

You were asked in what sense you used the term in responding, I think at least, to questions on

5-8

cross-examination.

THE WITNESS: I understand.

Well, I would, I think, in terms of discussing chemical explosives, as I used the term, that covers the whole broad range of both high and low explosives, and that's why I would make the distinction.

BY MS. FINAMORE:

Q What effect, if any, does that distinction have to the answer you gave on cross-examination regarding microsecond time scales and pressures?

A Well, the high explosives operate on a much shorter time period and low explosives operate on -- if I can refer to this text --

JUDGE MILLER: We've asked you not to refer to the text. In fact, I don't understand this entire question, to be frank with you about it.

The cross-examination question was asked. The answer was given. The record is complete, and I don't see that you are entitled to any further embroidery of it, Counsel.

MS. FINAMORE: Well, I'll withdraw the question.

BY MS. FINAMORE:

Q You were asked today about an AEC Code in CRBRP-1 in relation to the CRAC Code used by the Staff, in particular the use by those codes of whole body or bone

5-9

marrow doses.

You stated that bone marrow and whole body doses are fairly close to one another.

Can you explain what you meant by that statement?

A. Well, I don't think you represented my testimony totally.

JUDGE MILLER: We'll sustain your objection.

THE WITNESS: The --

MR. EDGAR: Well, then, I'll object on the grounds that it's improper redirect. Unless there's a predicate or a premise tying it to the cross-examination, it's improper.

JUDGE MILLER: I think the grounds should be it's an improper statement of the record.

MR. EDGAR: In addition, Your Honor, though, if there's no tie into the cross-examination, the question can't be proper.

JUDGE MILLER: Sustained.

BY MS. FINAMORE:

Q. Dr. Cochran, you were asked about the AEC Code and the CRAC Code, in particular their use of whole body and bone marrow doses; is that correct?

A. That's correct.

Q. And is it correct that you mentioned that the

5-10

bone marrow and whole body doses would be fairly close to one another?

MR. EDGAR: Objection. Leading the witness.

JUDGE MILLER: She's entitled to direct his attention to the matter.

Do you recall that line of testimony?

THE WITNESS: Yes, sir.

JUDGE MILLER: All right. Ask your question.

BY MS. FINAMORE:

Q Can you explain what you meant by your statement regarding bone marrow and whole body dose?

A The -- In situations where the exposure is predominantly from gamma or even to a lesser extent, however, high energy betas, one would anticipate the bone marrow dose and the whole body dose to be comparable.

It derives from the fact that the gammas fairly well penetrate the body without much attenuation, much like X-rays do, but even more so.

In the context of accident analysis, such as was conducted in WASH-1400, the early fatalities are due primarily to ground shine, and the ground shine dose, the whole body dose from the ground shine, because it would be predominantly again gammas would be comparable to the -- the whole body and bone marrow doses would be comparable; and, therefore, one should not in my view in that type of

5-11

a situation draw any significant distinction between whether you are measuring the whole body or the bone marrow dose.

I think that one should be cautious, though, in the application of that, because that distinction would not necessarily apply in all situations.

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BY MS. FINAMORE:

Q You were asked whether you were aware of the CRBR systems to accommodate sodium/water reaction; and you responded that you were not aware in detail. Can you explain what you meant by that answer?

MR. EDGAR: Objection. The answer was complete.

JUDGE MILLER: Sustained.

BY MS. FINAMORE:

Q Can you explain what level of detail you were referring to?

MR. EDGAR: Objection.

JUDGE MILLER: Sustained.

BY MS. FINAMORE:

Q You were provided with Applicants' Exhibit 57, a report by Science Applications, Incorporated, and asked whether you agreed with the statement of the authors in Subparagraph 1 on Page 5-2.

You stated that you did not entirely agree. Can you explain what you meant by that answer?

MR. EDGAR: Objection. The answer was complete as given. I'm assuming the witness is under oath, and now we're just saying, "Well, why don't we expand a little bit?"

MS. FINAMORE: No. I believe it's not clear

6-2

the basis for the disagreement with Dr. Cochran from this exhibit of Applicants.

JUDGE MILLER: My recollection is that he was simply asked to verify the wording of the matters contained in Exhibit 57, which, I believe, was a document that Dr. Cochran himself had used, was it not?

MS. FINAMORE: No.

JUDGE MILLER: It was not?

MS. FINAMORE: Excuse me, Your Honor. He was then asked if he agreed --

JUDGE MILLER: First of all, what was the document?

MS. FINAMORE: -- with the statement of the authors.

JUDGE MILLER: Yes. How did that come about? Had not Dr. Cochran said something in his testimony or in his responses to this particular report?

I thought he had, although I'm not positive.

MS. FINAMORE: I don't believe this report was referenced in his testimony.

MR. EDGAR: That's wrong. That's absolutely wrong.

THE WITNESS: It was referenced.

JUDGE MILLER: Page 35.

Okay, it was referenced. Therefore, I think the record is complete as to the questions asked. There again, the cross-examiner has a right to direct his questions as he wishes to a referenced document.

The record is complete on that.

MS. FINAMORE: We have no further questions.

JUDGE MILLER: Does anyone else have any questions?

MR. EDGAR: We have no further questions.

MR. SWANSON: None.

JUDGE MILLER: Dr. Hand?

JUDGE HAND: No.

JUDGE MILLER: Judge Linenberger.

BOARD EXAMINATION

BY JUDGE LINENBERGER:

Q. Dr. Cochran, would you say that the very last sentence of your testimony appearing on Page 44 of Intervenors' Exhibit No. 22 marked for identification really contains -- I'll use Judge Hand's words here -- the message that you would like to convey to the record with respect to this testimony?

A. Yes.

Q. I get myself in trouble sometimes, but I'd like to look at last sentences because they're frequently illuminating.

A. I would also add that our probabilities --

my probabilities in my testimony are not that far removed from the Staff's estimates of the probabilities -- I mean the uncertainties, I should add.

Q. Okay. Now the last sentence of Section J of Staff Exhibit 8 on Page J-25 --

A. Could I get a copy of that in front of me?

Q. Sure.

That last sentence on Page J-25 says that "The Staff's analysis confirms the FES conclusion that the accident risk estimate at Clinch River can be made acceptably low."

Your last sentence indicates that you cannot accept that conclusion.

Folding those two thoughts together, I am tempted to conclude that -- or infer that your position is that accident risk at Clinch River cannot be made acceptably low. Now is that a proper inference that I should draw from your testimony?

A. I think a more correct inference would be that it hasn't been demonstrated that it can be made acceptably low. I think I could go through and show you why, using their own analysis here in Appendix J, why their own data demonstrates that it's not acceptably low.

Now, your question is really, well, if there's another hypothetical -- you know -- suppose you radically changed the design of something and moved the site or whatever, then could you get there.

That's -- I'm not implying that you couldn't get there if you --

Q This is what I was curious about. You're not categorically taking the position that you can't get there from here, I gather from what you've just said, but rather that you're challenging whether the Staff has shown in their exhibit, a plausible way to get there.

A I think their exhibit on the face -- I mean if you set aside my testimony and just look at Appendix J and do the analysis of what Appendix J is telling you, it demonstrates that the CRBR, as designed, is not -- the risks are not acceptably low.

I can take you through that.

Q Well, you have pretty much in various pieces of testimony. But the thing that bothers me here is that one of your -- one of Intervenors' oft-stated -- often-stated positions throughout much of this testimony is that either they have not had access to the design details or that it's premature to discuss design details because they don't exist, and that Intervenors have been -- I believe it has been alleged by Intervenors that they have been handicapped in much of their analysis of what has been done because of a lack of the kind of detailed information they would liked to have had.

And so, therefore, I have a little bit of a problem, based on that backdrop, appreciating the weight that should be given to your conclusion that based on reading the Staff exhibit, you can't get there from here.

A. Well, let me respond to that. First of all, I repeat: I will take Appendix J as their estimate of how low the -- I mean, setting aside the conclusion, but take the analysis that they've presented, and demonstrate to you that the risks are not acceptably low.

Now, as a separate matter, I think we could have put on a better affirmative case had we had access to reliability data and systems interaction data specific to the CRBR and had that access at an early enough period of time where we could have hired consultants in that area to give it a better treatment than I've been able to give it in my testimony.

Q. Well, sir, that very statement assumes an answer that I don't see the evidence to support. What you're saying is that had you been able to probe more deeply, you could certainly have shown that things are worse than they are.

And isn't it just possible that had you been able to probe more deeply, you might have seen that things are not as worse as you think they are -- forgive my English.

But why is it that you assume that had you had more detail, you would have been able to prove a firmer negative --

A. Well --

Q. -- rather than been able to have demonstrated something more nearly to your liking?

A. I certainly can't prove that. That's a hypothesis that would remain to be demonstrated. But there is a possibility that a more detailed analysis would come out the other way.

JUDGE LINENBERGER: I think I'll leave things where they stand now.

Thanks very much.

JUDGE MILLER: Okay. Nothing further, I assume -- Oh, have you offered into evidence your exhibit.

MS. FINAMORE: No. I'd like to offer into evidence --

JUDGE MILLER: You may be excused as a witness, Dr. Cochran. Thank you.

(Witness excused.)

JUDGE MILLER: Is there any objection to Exhibit 22, as modified?

(No response.)

JUDGE MILLER: Hearing none, we will admit into evidence Intervenors' Exhibit 22.

(Intervenors' Exhibit No. 22 was marked for identification and follows.)

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BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION
ATOMIC SAFETY AND LICENSING BOARD

In the Matter of UNITED STATES DEPARTMENT OF ENERGY PROJECT MANAGEMENT CORPORATION TENNESSEE VALLEY AUTHORITY (Clinch River Breeder Reactor Plant)	Docket No. 50-537

TESTIMONY OF DR. THOMAS B. COCHRAN

PART IV

AS SUPPLEMENTED BY NEW INFORMATION

IN CRBR FINAL ENVIRONMENTAL

IMPACT STATEMENT SUPPLEMENT

(Intervenors' Contentions 1, 2, and 3)

DATED: November 12, 1982

Q.1: Please identify yourself and state your qu[illegible]fications to present this testimony.

A.2: My name is Thomas B. Cochran. I reside at 4836 North 30th Street, Arlington, Virginia 22207. I am a Senior Staff Scientist at Natural Resources Defense Council, Inc. My background and qualifications to present this testimony are presented in previous testimony in this proceeding. (Tr. 2870-71, Cochran.)

Q.2: What is the subject matter of the present testimony?

A.2: Part IV of my testimony deals with the potential for severe accidents at CRBR and the adequacy of Applicants' and Staff's analyses of those accidents. These are matters that are raised in Intervenors' Contentions 1, 2, and 3. For purposes of this phase of the proceeding, those Contentions read as follows:

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) [deferred]

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).

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 that 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) [deferred]

   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.

The accident discussion at this phase focuses on Appendix J of the Final Supplement to the FES, NUREG-0139, Supplement No. 1 (henceforth "FSFES").

Q.3: Dr. Cochran, are you familiar with Staff's NEPA analysis of the risks of potential accidents associated with the CRBR?

A.3: Yes.

Q.4: Where is this analysis set forth?

A.4: Primarily in Chapter 7 and Appendix J of the FSFES, although some paragraphs from Chapter 7 of the 1977 FES have been retained, including the conclusions in §7.1.4.

Q.5: Do you have general criticisms of Appendix J?

A.5: Yes. The methodology in Appendix J is crude by today's standards, and the assumptions behind it (and the input data) are not supported by any substantive analysis. While it presents estimates of the absolute probability of CRBR accidents, these estimates are backed up by no calculations and no event tree/fault tree analyses as one finds in risk assessment analyses such as the Reactor

Safety Study (WASH-1400) and CRBRP-1. No operating data are offered in support of its conclusions, and there are no quantified estimates of the uncertainty associated with the probability estimates. It must be remembered that WASH-1400, which contained an incomparably more detailed analysis of accident probabilities for two actual LWRs (and which is, incidentally, the direct progenitor of virtually all nuclear risk assessment work) was severely criticized for making unsupported assumptions, for failing to properly assess uncertainty and for its factual inscrutability. For these reasons, the NRC ultimately repudiated WASH-1400's absolute probability predictions. Yet, compared to Appendix J, WASH-1400 was a model of scientific analysis. Appendix J is not even supported by a plant-specific risk assessment. Its assumptions are not just unsupported by rigorous analysis; for the most part, they are not even presented for evaluation. If WASH-1400's probability estimates were unreliable, as the Commission correctly concluded, then the probability estimates in Appendix J are far more so. There is no reason to accept these on faith, and very little beyond faith is offered.

Moreover, the Staff attempt to quantitatively assess the uncertainty associated with the estimates for various quantitative accident probabilities and consequences presented in Appendix J is a one-sentence conclusory statement (FSFES, p. J-24) which is unsupported in the document by rigorous analysis. Probably the most serious criticism of WASH-1400 from the scientific community was its failure to assess or properly acknowledge the very large uncertainties attached to absolute probability predictions. Those uncertainties, which have been estimated to be as large as a factor of 100 in some cases, must be much greater for predicting CRBR accident probabilities, since the body of relevant operating data for LMFBRs is far less than for LWRs and since, for lack of a plant-specific assessment, the report is almost totally based on conclusory statements that can most charitably be characterized as "engineering judgment." Without some reasonable and scrutable assessment of the uncertainties inherent in these predictions, they are simply arbitrary and meaningless.

Q.6: Do you know whether the NRC Staff performed any calculations, reviewed operating data for other facilities, or did any plant-specific assessment of the reliability of the CRBR systems to back up the probability estimates presented in Appendix J?

A.6: According to the NRC Staff, with only three exceptions (WASH-1400 for PWR auxiliary feedwater reliability and the probability of loss of offsite power, and NUREG-0460 for the frequency of anticipated transients without scram for typical LWRs), they did not. NRDC asked the Staff in discovery to identify the documents relied upon for each of the principal probability assessments in Appendix J. (See Staff Response to NRDC's 27th Set of Interrogatories, Oct. 1, 1982, pp. 53-70.) In almost every case, the Staff responded under oath that it relied on no "specific" documents for any of the conclusions presented, instead relying generally on the "cumulative knowledge" of the Staff and its consultants in general, or a similar response. While "engineering judgment" or "cumulative knowledge" is valuable for many purposes, it is not sufficient to support predictions of the probability of serious accidents in a plant as complex and untested as the CRBR.

Q.7: Have you been limited in your ability to independently assess the probability of accidents beyond the design basis for CRBR?

A.7: Yes, independent assessment has been greatly hindered. The probability of a catastrophic accident in any plant is a function of the plant design, the potential for equipment malfunction and human error, and the reliability of its many complex systems and components. The CRBR is the first plant of its kind. Applicants have done much work in assessing the reliability of the CRBR design, primarily as part of Applicants' Reliability Program (see PSAR, Appendix C). The document known as CRBRP-1 is another prominent example. Applicants have underway a comprehensive probabilistic risk assessment (PRA) of the CRBR and preliminary results have been presented to the ACRS and the Staff (cf., Letter from John R. Longenecker, CRBR Project to Paul S. Check, USNRC, June 21, 1982, subj: Probabilistic Risk Assessment (PRA) Program Plan). However, the scope of this LWA-1 proceeding has been limited to exclude inquiry into what are termed the "details" of the CRBR design. CRBRP-1 has been expressly excluded from consideration. In my judgment, no reliable estimate of CRBR accident probabilities can be made within the present scope of the LWA-1 proceeding and without reviewing the CRBR design in some detail. This has not been possible at this stage.

Q.8: Do you believe that the analysis in Appendix J is realistic and adequate to support Staff's conclusions regarding consequences of Class 9 accidents, namely "that CRBR accident risks would not be significantly different from those of current LWRs..." and that "the accident risks at CRBR can be made acceptably low." (Appendix J, p.

J-25)?

A.8: No.

Q.9: Please proceed to discuss some of the specific probability estimates. To begin, what frequency of occurrence did the NRC staff assign to core degradation due to LOHS (loss of heat sink) events for CRBR and what rationale did the staff give for its estimate?

A.9: Staff assigned a frequency of core degradation due to LOHS events of less than 10^{-4} per reactor year (i.e., one chance in 10,000 per reactor year). Staff cited three principal factors for this result:

1. A "general consideration of typical achievable PWR auxiliary feedwater system reliabilities;"
2. The "potential for common cause failures;"
3. The potential for achieving "high reliability in final design and operation through an effective reliability program." (FSFES, pp. J-3, -4.)

While the three factors above are all listed as the bases for the estimated LOHS probability, only the first -- PWR auxiliary feedwater system reliability -- serves as the basis for Staff's quantified estimate. The role the other two factors play in the choice of the 10^{-4}/year estimate is discussed only in the most general qualitative terms, e.g., "... unavailability estimates for ... heat removal systems have been set high enough to include allowance for potential common mode failures" (Appendix J, p. J-22). The choice of auxiliary feedwater system failure as the controlling failure mode is not justified. In other words, there is no reason to believe that failures in systems other than auxiliary feedwater may not contribute significantly to the LOHS probability. A fault tree analysis is necessary to justify limiting the discussion to auxiliary feedwater reliability.

In order to illustrate the complexity of this issue, consider the generalized fault model for the shutdown heat removal system for CRBR taken from CRBRP-1, Vol. 2, Appendix II, p. 2-14 to 2-22 (attached to my testimony as Exhibit 1). This fault tree, which is developed to the system (or subsystem) level rather than the more detailed component level as in the WASH-1400 case, can be considered applicable to a reactor of the general size and type as CRBR. Clearly, it takes a leap of faith to conclude that the failure rate of the auxiliary feedwater system controls the overall frequency of core degradation due to LOHS events.

Q.10: Setting aside your view that there is no basis for concluding that the failure rate of the auxiliary feedwater system is controlling, do you agree with the

Staff's estimate of the feedwater system reliability? Explain your answer.

A.10: First, I should note that Staff claims that its estimate of the probability of LOHS events was based on independent analyses, primarily by William Morris of the Staff and Staff consultant Edward Rumble of Science Applications Inc., (SAI), each using a different base of information (Deposition of William Morris, Oct. 12, 1982, pp. 24-25).

Dr. Morris claimed his estimate is based on the reliability of auxiliary feedwater systems in PWRs over the years as documented in the Standard Review Plan for LWR feedwater systems (Morris, Deposition of Oct. 12, 1982, pp. 23-24).

Mr. Rumble also claimed his estimate was based on reliability studies of PWR auxiliary heat removal systems, the Accident Delineation Studies (Phases 1 and 2) (NUREG-CR-1407 is Phase 1) prepared by Sandia for NRC-NRR, and the study CRBRP-1 (which is beyond the scope of the LWA-1 proceeding). Mr. Rumble said these estimates were what he believed should be achievable, not necessarily what has been achieved to date (E.R. Rumble, private telephone communication, July 27, 1982, as noted in T.B. Cochran Memo to Files, July 27, 1982).

I do not agree with Staff's estimate or Staff's underlying analysis. First, LOHS fault trees for CRBR developed in CRBRP-1 differ from those of a PWR as developed in WASH-1400, and consequently there is no obvious correlation between PWR system reliabilities and the core degradation frequency due to LOHS accident scenarios in CRBR. This can be seen by comparing the generalized fault models for CRBR shutdown heat removal (see CRBRP-1, Vol. 2, Appendix II) with the fault models for a PWR (see WASH-1400, App. II).

Staff claims that its estimate of 10^{-4}/year is based on "typical achievable PWR auxiliary feedwater system reliabilities" (Appendix J, p. J-4). If this is so, there must be wide variations in achievable feedwater system reliability. For example, the RSSMAP (Reactor Safety Study Methodology Applications Program) report for Calvert Cliffs (NUREG/CR-1569)[34] concluded that, the probability of core melt for Calvert Cliffs was 4×10^{-4} i.e. 1 chance in ~~2400~~ 2500 per reactor year (after improvements), largely due to unreliabilities in the auxiliary feedwater system and failure of backup heat removal methods (p. 6-40). This result is a factor of 4 larger than the Staff's alleged "upper bound" result for CRBR. No justification has been presented for concluding that ~~the~~ the CRBR auxiliary feedwater system will be more reliable than Calvert Cliffs by at least a factor of four. Furthermore, there is a serious question about the comparability of PWR operating data in this area to the CRBR. It should be noted in this connection that the authors of the Applicants' risk assessment work felt that the WASH-1400 data could not be applied to the question of unavailability of decay heat removal systems for CRBR. Instead, a fault tree analysis was conducted to determine the system availability. (CRBRP-1, Vol. 2, at III-3.)

There is no basis for concluding that CRBR's auxiliary feedwater system will be "typical" in its reliability. The conservative assumption to make at this juncture might be to assume that CRBR's auxiliary feedwater system will be no better than Calvert Cliffs' system. Moreover, since CRBR's Decay Heat Removal System (DHRS) is dependent upon AC electrical power, it cannot be assumed to be significantly more reliable than PWR DHRSs; according to Staff (FSFES, pp. J-3,4), a principal unreliability in PWR decay heat removal systems is not in system failures per se but in loss of offsite and onsite AC power and the steam-driven auxiliary feedwater trains. Thus, if Staff is correct, the ability of the CRBR DHRS to operate at "normal" temperature and pressure (whereas PWR DHRSs can operate only at low pressure) should not have a major impact on overall risk.

Q.11: Are there other CRBR heat removal systems that are important in terms of the comparability between the frequencies of core degradation in CRBR and PWRs due to loss of heat sink (LOHS)?

A.11: What I noted above was that one cannot tell the degree of contribution that various component failures have on the overall failure rate without a detailed fault tree analysis. However, it is evident that there are other CRBR heat removal components whose failure rates are not necessarily comparable to PWR systems. The steam generators are an example. There is no discussion whatever in Appendix J of the contribution of steam generator failure to the overall risk of LOHS, nor of the possible mechanisms or modes of failure considered. Unlike an LWR, the steam generators in an LMFBR, such as CRBR, represent a location where significant amounts of sodium and water are in close proximity. CRBR event sequences can be postulated, e.g., propagation of steam generator tube failures, where sufficient water and sodium can be brought together in such a manner as to create a sodium-water reaction coupled with a hydrogen reaction, resulting in loss of the shutdown heat removal function (see generally CRBRP-1, Appendix VIII).

The General Accounting Office in a recent letter to Congress was highly critical of DOE's failure to conduct complete and thorough tests of the steam generators to be used in the CRBR, in spite of the fact that steam generators for LMFBRs have had a history of serious technical problems and the fact that development and demonstration of reliable steam generators have been and still are one of the most significant technical problems facing the CRBR project. (Letter from Charles A. Bowsher, Comptroller General, to Congressman John D. Dingell, May 25, 1982, GAO/EMD-82-75, attached as Exhibit 2).

In sum, because of the inherent differences in the shutdown heat removal systems, e.g., steam generators, between PWRs and LMFBRs introduced by the use of sodium coolant in an LMFBR, it does not directly follow that the frequency of core degradation due to LOHS events in PWRs is directly transferrable to LMFBRs.

Q.12: How did Staff treat the contribution of pipe rupture failure as a contributor to the core disruptive frequency?

A.12: The frequency of large pipe breaks (loss-of-coolant accidents, or "LOCAs") is pivotal to an assessment of the risk of accidents at CRBR or a reactor of the general size and type. A large pipe break in the cold leg (and perhaps the hot leg, as well) would likely lead to core disruption and serious offsite consequences. It is an important determinant in whether the CRBR site is suitable. Staff states:

> Because of the high boiling point of sodium, the CRBRP primary coolant system would

> operate at significantly lower pressures than LWR primary coolant systems. This reduces the frequency of large ruptures in the primary coolant system. To further ensure that large breaks cannot occur and cause core damage, implementation of preservice and inservice inspection of the primary coolant boundary and a leak detection system will be required. In addition, a guard vessel will be included to prevent unacceptable leakage from large portions of the primary coolant system. For these reasons LOCAs are not considered credible (i.e., design-basis) events at CRBRP. The frequency assumed for LOHS adequately bounds the LOCA contributions to core disruption frequency.

(FSFES, p. J.4, emphasis supplied.) When asked to identify every document relied upon by Staff for its conclusion above that "LOCAs are not considered credible ... events at CRBRP," Staff stated:

> The cumulative knowledge of the Staff and its consultants rather than a specific document were relied upon by the Staff for its conclusions in Appendix J regarding whether LOCAs are DBAs for CRBR. This issue was also discussed in the SSR and the Staff's prefiled testimony for the site suitability hearings.

(Staff Response to Interrogatory 33, 27th Set, Oct. 1, 1982, p. 58.) I take this answer to mean that Staff has no documentation or written analysis demonstrating that a LOCA is a low probability event for the CRBR.

In the 1982 SSR, Staff stated:

> It is the staff's opinion, based on the following considerations, that the heat transport system can be designed for a high level of integrity and for continued assurance of this integrity throughout the operating history of the plant. The specifications include stringent

nondestructive examination requirements. The material is characterized by high fracture toughness and corresponding large critical flaw size, a negligible growth rate of postulated defects and the probability of throughwall growth rather than elongation of defects. The system has low stored energy and is monitored by sensitive leak detection instruments. The staff preliminary conclusion is that double ended rupture of the CRBRP primary cold leg piping (an event that could potentially lead to a CDA unless otherwise mitigated) need not be considered a design basis event. This conclusion is conditioned on an acceptable preservice and inservice inspection program, a material surveillance program, continued research and development verifying material degradation processes, and verification of leak detection system performance. The staff considers it feasible to implement programs to satisfy these requirements. The staff intends to continue its review of the sodium cold leg piping to insure that the issues are resolved properly.

Because of its higher operating temperature, the same conclusions have not yet been reached concerning the hot leg piping (995° vs 730° F). The staff has studies underway to evaluate the potential for and consequences of hot leg piping ruptures. Preliminary results obtained so far indicate that this event has more benign consequences with respect to core thermal conditions than the cold leg rupture. For example, a hot leg pipe rupture followed by a scram and a pump trip and normal flow coastdown does not appear to lead to boiling in the core. Analyses of this event are continuing and the results will be factored into any future requirements to assure that hot leg pipe ruptures, like the cold leg case, need not be considered as events that would lead to a CDA.

(1982 SSR, pp. II-8 to II-9.)

Q.13: Do you agree with Staff's assessment, as stated above, of the pipe rupture probability, and, if not, what is the basis for your disagreement?

A.13: I disagree with the Staff assessment. In this regard, it is extremely instructive to compare Staff's analysis with the analyses conducted by D. O. Harris of the Palo Alto office of Science Applications, Inc. (SAI), for the CRBR Project office in the 1977-78 period. SAI was a consultant to the CRBR Project in the development and application of the fault tree/event tree methodology for assessing the reliability of CRBR systems as published in CRBRP-1, March 1977, and continued work for DOE on a variety of CRBR risk assessment issues through early 1979 and perhaps beyond. Staff consultant Rumble is a Vice President of SAI at the same Palo Alto office and has stated to me that he relied in part on CRBRP-1 for his assessment of the core degradation frequency which appears in Appendix J of the DSFES (and therefore the FSFES).

I have not been permitted to address that work in this hearing because, of course, it involves the "details" of the CRBR design. Only the most general conclusions have been presented in Appendix J.

In what appears to be a final risk assessment task report, obtained by NRDC under the Freedom of Information Act, D.O. Harris of the SAI Palo Alto office summarized the result of SAI's assessment of the CRBR pipe rupture probability (Harris, D.O., "Relative Pipe Rupture Probability for the Primary Heat Transport System of CRBRP," Nov. 13, 1978, attached as Exhibit 3 to this testimony).

Harris's analysis appears to be based on the assumption that the primary large pipe failure mechanism is fatigue crack growth due to cyclic stress imposed on defects introduced prior to service, hence other potential sources of failure were not considered. In this respect, Harris's analysis appears similar to that conducted in CRBRP-1 (Vol. 2, App. III, p. III-112). In the Harris analysis, calculated relative probability of pipe rupture in CRBR compared to that of PWRs was primarily a function of

a) probability of having a defect, which in turn was a function of the number and characteristics of the weld joints, Because the appropriate normalization was not known, separate calculations were made using weld volume, weld area, and weld length as the basis of normalization.

b) the initial crack size and depth distribution. Because the appropriate crack distribution was not known, separate calculations were made using four crack distribution expressions.

The differences between Staff's assertions and the SAI ~~anlysis~~ analysis are important. Staff's conclusion that the CRBR cold leg pipe break is incredible (i.e., beyond the design basis) is based in part on the fact that there will be preservice and inservice inspection programs. Such programs have been in place for light water reactors for some time. The SAI analysis assumed equivalent effectiveness for the inspection programs for both CRBR and PWR in each calculation of the relative probability of pipe break failure of the two. This is the appropriate ~~approriate~~ way to treat the subject. Staff offers no evidence that any relative difference in the CRBR and PWR surveillance programs would have a significant effect on the crack distributions in CRBR piping relative to that in PWRs.

SAI found that "[w]ith the present state of knowledge, it is not possible to ascertain the controlling parameters" that govern the relative CRBR/PWR pipe break frequency. SAI found a wide range of values varying from 0.0186 to 11.62 (i.e., three orders of magnitude) in the ratio of CRBR pipe failure to PWR pipe failure depending on the assumptions made. In fully 13 out of 36 cases (36%) analyzed, the probability of CRBR pipe failure exceeded the probability of PWR pipe failure. Furthermore, the probability of PWR failure was found to be strongly design dependent, varying by as much as a factor of 14 among the three PWRs analyzed.

In conclusion, the Staff analysis of the pipe break probability is nothing more than a series of unsupported assumptions that appear to be in conflict with a more rigorous CRBR-specific analysis. The SAI analysis does not support the conclusion that a LOCA is "incredible" for the CRBR. Moreover, as evidenced by the SAI analysis, i.e., the lack of understanding of the controlling factors, the fact that the CRBR pipe break frequency may be as much as 12 times higher than that in a PWR, and the fact that the frequency is a strong function of the number and characteristics of the pipe welds, which are design dependent, the Staff conclusion that a cold (or hot) leg pipe rupture is not credible in a reactor of the general size and type of CRBR is not substantiated by rigorous analysis. It should be rejected.

Q.14: Do you agree with Staff's analysis of common mode failures?

A.14: The one sentence devoted to common cause failure hardly qualifies as "an analysis." LOHS failures due to common causes are but one manifestation of a larger class of failures that fall under the general category of systems interaction (SI). Systems interaction is presently the subject of two unresolved safety issues (USIs) -- namely A-17, "Systems Interaction in Nuclear Power Plants," and

A-47, "Safety Implications of Control Systems." The NRC has sponsored four separate evaluations of systems interaction in an attempt to develop an acceptable methodology for reviewing final designs for adverse systems interactions. These four studies are:

1. NUREG/CR-1321, "Final Report -- Phase I Systems Interaction Methodology Applications Program," G. Boyd, et al., Sandia National Laboratories, April 1980.

2. NUREG/CR-1896, "Review of Systems Interaction Methodologies," P. Cybulskis, et al., Battelle Columbus Laboratories, January 1981.

3. NUREG/CR-1859, "Systems Interaction: State-of-the-Art Review and Methods Evaluation," J.J. Lim, et al., Lawrence Livermore Laboratory, January 1981.

4. NUREG/CR-1901, "Review and Evaluation of System Interactions Methods," A.J. Buslik, et al., Brookhaven National Laboratory, April 1981.

The NRC Staff's evaluation of these four reports is summarized in the periodic "TMI Action Plan Tracking System Report" as follows:

> State-of-the-art review concluded that no single method presently exists in a form that can be used to perform an adequate review for adverse SI.

Thus, it can be fairly concluded that an adequate systems interaction review of CRBR could not have been conducted. Moreover, such a review requires a final design, which is not yet available for CRBR. It should be noted that three of the SI reviews above attempted unsuccessfully to evaluate SI in actual past events involving SI, including the Browns Ferry fire in 1975, the TMI-2 accident in 1979, the Browns Ferry partial scram failure in 1980, the pressurizer relief valve failure at Beznau in 1974, the temporary loss of decay heat removal at Davis-Besse in 1980, the loss of DC control power and diesel generator fire at Zion in 1976, and the Crystal River LOCA in 1980.

In addition, common mode failures and other forms of systems interaction involve more than just hardware failures. Also involved are external events (such as seismic events and hurricanes), human error (including errors of omission and commission, and including not only operations but design, fabrication, installation, maintenance, and testing), and design flaws. The design of the control room and any auxiliary control panels or remote shutdown locations, and actual operating, emergency, maintenance, and test procedures can also impact on systems interactions.

In sum, the effect of potential common mode failures on CRBR accident probabilities involves complex issues that the technical community has been wrestling with for years, thus far without notable success. There is no substantive basis for Staff's broad-brush assertion that "[t]he foregoing estimates of frequencies and risk associated with CRBR have included allowances for uncertainties. For example, unavailability estimates for shutdown and heat removal systems have been set high enough to include allowances for potential common cause failures." (Appendix J, p. J-22.)

Q.15: In estimating the quantitative probability of CRBR accidents, can credit be assigned for an "effective reliability program"?

A.15: In my opinion, it is not possible to assign any particular value to the level of "reliability" to be achieved. No CRBR-specific program has been presented by Staff; no precedent is cited for an "effective reliability program" for any other plant and no criteria are presented.

Finally, such assertions about the achievability of high reliability must be taken in the context of the most recent construction and design experience. This body of experience includes widespread problems at Diablo Canyon, Zimmer, and Midland. This experience is scarcely cause for confidence.

For all the reasons given above, I conclude that the NRC Staff's estimate of the frequency of core degradation due to LOHS events is optimistic, unsupported by rigorous analysis, and fails to properly account for uncertainties.

Q.16: Turning now to other contributors to the probability of core disruption, what assumption did the Staff make with regard to the probability of simultaneous failure of both reactor shutdown systems?

A.16: The Staff assured that "there are sufficient inherent redundancy, diversity, and independence in the overall shutdown system designs to expect an unavailability of less than 10^{-5} per demand," and concluded that "the combined frequency of degraded core accidents initiated by ULOF and UTOP events is less than 10^{-4} per reactor" (FSFES, p. J-4,5).

Q.17: What is the basis for the Staff estimate?

A.17: Beyond the explanation on pages J-4,5 of the FSFES, Staff claimed the value of 10^{-4} per year was a bounding value based primarily on LWR experience as published in NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors." In Vol. 1, Section 4.3 of NUREG-0460, an estimate of $2x10^{-4}$ per year for the frequency of ATWS for typical LWRs was given. Staff also stated, "Because the [CRBR shutdown systems] design and the reliability program are not final they have not been definitive in making the reliability estimate." (Staff Response to Interrogatories 36, 37, 38, 27th Set, Oct. 1, 1982, p. 60.)

Staff Witness Morris claimed that Mr. Rumble of SAI may have had a different basis for arriving at the value of 10^{-4} per year (Deposition of Staff Witness Morris, Oct. 12, 1982, p. 43).

Staff Witness Rumble said the basis for his estimate of the scram reliability of 10^{-5}/demand at DSFES, p. J-4, was based primarily on NUREG-0460; however, several other studies were mentioned as well. Mr. Rumble stated he was not familiar with the Commission's ATWS Policy Statement. (Edward Rumble, private communication, July 27, 1982, as recorded in Memo to files of T.B. Cochran, July 27, 1982.)

Q.18: Do you agree with the Staff conclusion that 10^{-4} per year is a conservative "upper bound" frequency of degraded core accidents initiated by ULOF and UTOP events in CRBR and, if not, what is the basis for your disagreement?

A.18: I do not agree. I believe 10^{-3} per year would be a conservative upper bound based on the Commission's LWR analysis in the Commission's Proposed ATWS rule for LWRs (46 Fed. Reg. 57521, Nov. 24, 1981)(<u>see</u> Tr. 2845, Cochran). While 10^{-4}/year might ultimately be shown to be appropriate, in light of the current absence of the detailed CRBR failure mode and effects analysis for the shutdown systems and consideration of effects of common mode failure, including, for example, seismic induced scram failures, there is at this time no basis for selecting a value smaller ~~larger~~ than 10^{-3} per year.

Q.19: What assumptions did Staff make with regard to the probability of core degradation as a consequence of fuel failure propagation?

A.19: Staff assumed that "the CRBR fuel design will be required to have an inherent capability to prevent rapid propagation of fuel failure from local faults" (FSFES, p. J-4) and that the frequencies attributed to LOHS, UTOP, and ULOF events adequately bound the contribution to core disruption frequency from fuel failure propagation (FSFES, p. J-5).

Q.20: Has Staff provided adequate justification for this assertion, and what is the basis for your conclusion.

A.20: I do not believe there is an adequate basis for this conclusion. Staff has not developed the specific requirements or any associated criteria or confirmatory programs to prevent rapid propagation (details of the systems to prevent propagation of fuel failure are not final at this time), and Staff could cite no documentation for the conclusion that the core disruption frequency due to fuel failure propagation is bounded by 10^{-4} per year (Response to Interrogatory 39, 27th Set, Oct. 1, 1982, pp.

62-63).

Q.21: What assumption did Staff make with regard to the conditional frequency that a CDA once initiated would be energetic?

A.21: Staff developed four categories of primary system failure as a function of the energy associated with disruption (FSFES, p. J-5,6) and assigned a probability of primary system failure by excessive mechanical and/or thermal loads resulting in continuous open venting into the upper containment through failed seals (Category IV) of approximately 0.1 per CDA (FSFES, p. J-6).

Q.22: What basis did Staff give for this assumption?

A.22: In response to interrogatories asking for all documents relied on to support this conclusion, Staff claimed that this estimate was based on "the Staff's general knowledge of and experience with the extensive research on the phenomena that may occur in a core disruptive accident ...", but refused to cite any documents. (Staff Response to Interrogatory 43, 27th Set, Oct. 1, 1982, pp. 66-67.)

Q.23: Do you have any basis for disagreeing with Staff estimate?

A.23: There is inadequate documentation to support the Staff's estimate, which may be correct, incorrect, conservative, or nonconservative.

Q.24: What assumptions did the NRC Staff make regarding containment integrity in its analysis of CDAs?

A.24: Staff assumes that mitigating systems, principally the containment annulus cooling and vent/purge systems, will have an unavailability of less than or equal to 1 in 100 per demand. Staff also assumes that the unavailability of containment isolation will be equal to or less than 1 in 100 per demand. (FSFES, pp. J-6, -7.)

Q.25: Do you agree with these estimates and, if not, why not?

A.25: If Staff is correct that loss of offsite and onsite AC power dominates the failure probability for LOHS events, such a failure could also cause the failure of the mitigating systems. Staff has not accounted for this common failure mode.

Staff Witness Rumble stated that the basis for the 10^{-2} per demand for containment failure was based on estimates of LWR containment failure of $3x10^{-3}$ (Edward Rumble, private telephone communication, July 27, 1982, as summarized in Memo to Files of T.B. Cochran, July 27, 1982). As noted in the Union of Concerned Scientists' comments on the DSFES (letter from Steven C. Sholly to Paul Check, 13 Sept. 1982; FSFES, p. N-50), the operating history of PWRs and BWRs in the United States does not support the assumed unavailability result of 10^{-2} per demand. A review of actual experience through 1980 was reported in Nuclear Safety (Michael B. Weinstein, "Primary Containment Leakage Integrity: Availability and Review of Failure Experience," Nuclear Safety, Vol. 21, No. 5, September-October 1980) and concluded that the overall availability of containment integrity was about 0.85 (i.e., an unavailability of 15 in 100 per demand). This experience base would dramatically affect the Staff's risk analysis of CRBR. Using LWR experience would appear to increase the estimate for containment failure by a factor of 15. Even if the value for PWRs alone is used, the result is only 0.96 (i.e., 4 in 100 per demand unavailability factor). Obviously, if a Category IV CDA (as discussed by Staff) occurs with a breach in containment integrity, a very large release to the environment will occur. Use of actual experience is certainly to be preferred as contrasted with the very soft results obtained from the Staff's "analysis." It has not been shown that there are substantial differences between CRBR and the LWRs that form the present experience base.

In addition, it should be noted that the assumption of the failure of the mitigating systems discussed above (the containment annulus cooling and vent/purge systems) will also dramatically affect source term assumptions for the CRBR plant. Such failures will also increase the failure probability of the primary containment since lack of annulus cooling will cause a more rapid pressure rise and an earlier failure of the primary containment. This allows less time for natural processes to operate to reduce the airborne source term in the containment, and the postulated failure of the vent/purge system will also increase the source term for containment release substantially, especially for particulates and aerosols.

Staff's analysis is inadequate in its failure to address the points noted above and the concomitant large uncertainties inherent in the Staff's assumptions.

Q.26: Turning now to the estimates of the consequences in death and injury of CRBR accidents greater than the design basis, are the Staff's estimates presented in Appendix J likely to be accurate? Explain your answer.

A.26: No, and there are several reasons. First, Staff's assumed radioactivity source terms are not supported by analysis or documentation. When asked the basis for Staff's estimate of the head release fractions selected in Table

J.3 at p. J-10, including all analytical calculations and documentation, Staff stated:

> The head release fractions (Table J.3) were selected on the basis of judgement from consideration of general LMFBR research of energetic CDAs involving a bubble of vaporized fuel material rising against the reactor vessel head, giving consideration also to the relative volatilities of different types of fission products and other materials. The selections were therefore not based on a set of analytical calculations or on any specific documents.

(Staff Response to Interrogatory 53, 27th Set, Oct. 1, 1982, p. 77.)

The release fractions associated with CDAs are highly design dependent. The Staff "judgements," based on no analysis or documentation, represent speculations, and the uncertainties in some of the estimates, e.g., Pu release under Category IV, could be at least a factor of 3 "higher."

Second, the CRAC model utilized by Staff assumes the $LD_{50/60}$ (lethal dose to 50% of the exposed population within 60 days) is 510 rads. In my opinion, this assumption is unrealistic. This dose-response level is associated with a dose-response curve depicted graphically at page 9-4 of Appendix VI of WASH-1400. This dose-response curve, however, assumes that the victims receive "supportive treatment," which includes barrier nursing, copious use of antibiotics, massive transfusions, reverse isolation, and other special sterile procedures. WASH-

1400 estimated that the entire medical capability of the United States could provide such treatment to no more than 2,500-5,000 persons. WASH-1400 failed to address, however, how the victims of the highest exposures would be identified when there will be many others who will be suffering symptoms of radiation sickness (such as prodromal vomiting) from lesser exposures.

There is considerable controversy over the use of the 510 rads $LD_{50/60}$. The Risk Assessment Review Group (NUREG/CR-~~0040~~ 0400, "Risk Assessment Review group Report to the U.S. Nuclear Regulatory Commission," Harold W. Lewis, Chairman, September 1978) concluded that scientific opinion supports a range from 400-600 rads. This range could cause a factor of two change either way in the number of early fatalities. Moreover, the Risk Assessment Review Group concluded with regard to supportive treatment that "the ability to carry out such intervention has not only not been demonstrated, but isn't even well planned at this time" (NUREG/CR-~~0040~~ 0400, p. 19). Changing the $LD_{50/60}$ from 510 rads for "supportive treatment" to the level of "minimal treatment," i.e., 340 rads, could increase the number of fatalities by a factor of two to four (WASH-1400, Appendix VI, p. 13-50; NUREG-0340, pp. 26-28).

Other groups have used more realistic dose-response relationships which are closer to the "minimal treatment" curve used in WASH-1400. The California underground siting study used an $LD_{50/60}$ for minimal treatment of 286 rads and for supportive treatment of 429 rads (Subcommittee on Energy and the Environment, House Committee on Interior and Insular Affairs, "Reactor Safety Study Review," Serial No. 96-3, 1979, p. 366, attachment to letter dated 21 February 1979, from Bryce W. Johnson, Peter R. Davis, and Long Lee to Hon. Morris Udall, p. D-7). In addition, the "Accident Evaluation Code" (AEC) used to calculate health effects in CRBRP-1 utilizes an $LD_{50/60}$ of 350 rems (SAI-078-78-PA, Z.T. Mendoza and R.L. Ritzman, "Final Report on Comparative Calculations for the AEC and CRAC Risk Assessment Codes," Science Applications, Inc., December 1978, p. 3-6 and 3-8).

Third, the CRAC code contains several "hidden" assumptions regarding the cancer risk estimator for latent cancers, including an assumption that the cancer risk at low dose is a function of dose rate. The net effect of these assumptions appears to be to reduce the estimate of latent cancer fatalities (exclusive of thyroid cancers) by a factor of 2 to 2.5 compared to the estimate one would obtain using 135×10^{-6} potential cancer deaths per person-rem, which Staff claims to use for estimating offsite health effects (FSFES, p. 5-13). Furthermore, a number of experts, including Radford, Morgan, Gofman, Stewart, Mancuso, Kneale, and Tamplin, believe the Staff cancer risk estimator, $135/10^6$ person-rem, is low, or probably low. Their own estimates of the cancer risk vary, but range from a factor of 3 (Radford, Edward, Science 213, 602 (7 August 1981), to a factor of 7 (Morgan) to a factor of 28 (Gofman, John W., Radiation and Human Health (Sierra Club Books, San Francisco, 1981), p. 305) times greater than the Staff's estimate of $135/10^6$ person-rem for fatal cancers due to ~~wheid~~ whole body low-LET exposure.

Fourth, the source terms used by the NRC Staff in the CRBR accident consequence calculations appear to ignore any possible common cause failure of the containment annulus cooling and/or filtered venting systems. Certainly both of these systems are dependent upon offsite and onsite power supplies, and both will fail if all power is lost. On this basis, as noted previously, it makes little sense to largely ignore common cause failures involving these systems, as Staff has done. If the containment annulus cooling system fails, this will shorten the time between initiation of a CDA and failure of the primary containment. This affects decay of radionuclides that make up the source term and reduces the time available for natural processes such as gravitational settling and aerosol agglomeration to reduce the source term. Failure of the filtered venting system shortens the time between primary containment failure and secondary containment failure and also increases the source term when the containment fails. In particular, the source term for particulates and radioiodines will be greater if these systems fail. This scenario will result in a larger source term for release to the environment and will result in more serious consequences than predicted by the NRC Staff analysis.

Another consequence of assumption failure of the containment annulus cooling and filtered venting systems is a greater release of Lanthanide group radionuclides, including Pu-239. These long-lived radionuclides will certainly have an impact on cancer fatalities and on land contamination (and related interdiction criteria).

Q.27: What is Staff's position regarding the potential for a nuclear explosion in the CRBR?

A.27: In comments on the DSFES, Ohio Citizens for Responsible Energy (OCRE) asserted that "LMFBRs can suffer criticality accidents that can cause <u>nuclear</u> explosions as shown by <u>The Accident Hazards of Nuclear Power Plants</u> by Dr. Richard E. Webb" (FSFES, p. N-10).*

Q.28: Do you agree with Staff's position? Explain your answer.

A.28: No. Staff is incorrect in this regard as evidence by Staff's and Applicants' own characterizations of CDAs as

*"The Staff's response (FSFES, P. 12-82) was,such a recritically would not result in an energy release over such a short duration that it would be characterized as an explosion."

explosions. In testimony before the Senate Subcommittee on Nuclear Regulation of the Committee on Environment and Public Works, (attached as attachment 4 ~~Exhibit 3~~), DOE and NRC Staff witnesses discussed environmental and safety matters related to the CRBR, including "hypothetical core disruptive accidents (HCDAs)," "core meltdowns and energetic disassembly," and design basis accidents. During the course of this testimony the following exchange took place between Senator Bumpers and Edson G. Case, then Acting Director, Office of Nuclear Regulation at the NRC:

> Senator Bumpers: May I ask one question? What is an energetic disassembly? Is that an explosion?
> Mr. Case: In layman's terms, it would be called an explosion. Yes sir. (~~Exhibit~~ attachment 4, p. 19)

Later in the same hearings the following exchange took place between Senator Bumpers and Eric S. Beckjord, Director of the Division of Reactor Development and Demonstration at ERDA.

> Senator Bumpers: Mr. Beckjord, what are the probabilities by ERDA's estimates of an explosion occurring in a breeder reactor plant?
>
> Mr Beckjord: That would be the same order, 10^{-6} per reactor year. I might add that one of the margins that is to be included in this plant design is the capability to withstand a very sharp explosion. The words "energetic disassembly" came up earlier. Maybe that is overly technical, but we hve been in discussions with the Nuclear Regulatory Commission on the amount of energy, the amount of explosive force that must be

> accomodated within the structure. That matter is not settled yet. (Exhibit 3, p. 29).

These are not isolated references. The energetic disassembly of a fast breeder reactor is commonly referred to as an "explosive disassembly "[see, e.g., Lee J.C. and Pigford, Thomas," Explosive Disassembly of Fast Reactors, "Nuclear Science and Engineering 48, 28-44 (1972)] or "a small nuclear explosion "Hicks, E.P. and Menzies, D.C., Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors," Oct. 11-14, 1965, ANL-7120, pp. 654-670], a "low-efficiency nuclear explosion" [Stratton, W.R., and Engle, L.B., "Reactor Power Excursion Studies," "Engineering of Fast Reactors for Safe and Reliable Operation" (1973 Karlsruhe Conference), pp. 1331-1551].

There is no universally accepted definition of the word "explosion." The Webster's Seventh New Collegiate Dictionary defines "explosion" as "a large-scale, rapid and spectacular expansion, outbreak, or other upheaval." Cook defines an "explosive" as "any substance or device which will produce, upon release of its potential energy, a sudden outburst of gas, thereby exerting high pressures on its surrounding" [Melvin A. Cook, The Science of High Explosives (Robert E. Krieger Publ. Co., Huntington, N.Y.) 1971, p.1] Cook groups explosives under three fundamental types, mechanical, chemical and atomic (or nuclear). Johansson C.H. and P.A. Persson in Detonics of High Explosives (Academic Press, London, 1970) state (at p.6):

> Explosion is basically a rapid expansion of matter into a volume much greater than its original one. The word explosion thus includes the effects following or including rapid combustion or detonation, as well as purely physical processes as to bursting of a cylinder of compressed gas. We have chosen not to limit this rather useful wide definition of the word.

By these definitions an energetic disassembly of an LMFBR Core would constitute an explosion. It would not constitute a detonation which is a specific type of exothermic reaction that is always associated with a shock wave. If, as some authors prefer, an explosion is given a more limited definition such as to require the production of a shock wave, then most energetic disassemblies of LMFBR cores would not fit that definition.

A nuclear explosion is an explosion in which most or all of the explosive energy is derived from nuclear processes, either fission or fusion, or a combination of both.* [See generally, Samuel Glasstone, The Effects of Nuclear Weapons, 1962 Ed. ¶ 1.10]. Thus, an explosion in an LMFBR, that is, an energetic disassembly following a prompt critical excursion, would constitute a nuclear

* Fusion does not apply to the LMFBR for reasons that are obvious.

explosion as opposed to a chemical or mechanical explosion.

In response to a series of questions by Judge Linenberger in earlier testimony, I characterized a nuclear explosion as requiring a sufficient rate of energy deposition to result in the generation of a shock wave. Upon reflection, I do not believe this is the preferred definition. In any case, my previous testimony at Tr. 2777, 2779, 2785 and 2789 contains an error in inferring that the energetic disassembly of a fast reactor would result in the production of shock waves.

For the disassembly to be sufficiently energetic for the mechanical loading to challenge the containment, the nuclear excursion in a large Fast Reactor such as CRBR would have to be characterized by a rapid reactivity insertion and the reactivity exceed prompt critical. This will result in a rapid introduction of energy from the nuclear process, a rapid increase in ~~reactor~~ reactor power, elevated fuel temperature and vapor pressure formation. In such an event the core will begin to expand.*

---

* Core expansion and fuel motion which reduces the material density will produce a negative reactivity feedback. Only a small expansion of the core is required to produce ~~a~~ larage disassembly reactivity. The reactor rapidly becomes sufficiently subcritical that any continued external reactivity insertion mechanism has no appreciable ~~baring~~ bearing on the ultimate consequences. This marks the conclusion of the neutronic excursion and the disassembly of the accident [Waltar, Alan E. (cont. next page)

An energetic disassembly, or nuclear explosion, in an LMFBR differs from a chemical explosion following detonation of a high-explosive in terms of the pressure-time characteristics of the two. Generally mechanical damage from an explosion or pressure transient can be caused by either a shock wave, which is transmitted rapidly to a structure, or the more slowly expanding bubble of reaction products or vaporized material or both. Pressures in a chemical high explosive detonation build up on a microsecond time scale. As a consequence, much of the damage potential of a chemical high explosive to immediate surrounding structures is likely to come from blast or shock wave effects. In an explosion in an LMFBR the build up is over a millisecond time scale and shock waves are generally not produced. Long-term bubble expansion (at least in the absence of a vapor explosion driven by a molten fuel-coolant interaction) would be the predominant damage mode for the slower time scale pressure build up associated with an LMFBR nuclear excursion. (See, generally, Walters and Reynolds, ibid., p. 664.)

Q.29: What is your overall conclusion regarding the Staff analysis in Appendix J?

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and Albert B. Reynolds, Fast Breeder Reactors (Pergamon Press, N.Y.) 1981, p. 619].

A.29: According to Staff Witness Rumble, Appendix J was done hurriedly because of the severe time constraints (Edward Rumble, private telephone conversation, July 27, 1982, as summarized in T.B. Cochran Memo to Files dated July 27, 1982). This is apparent from the depth of the analysis presented.

Staff can correctly point to several conservative assumptions made in Staff's analysis. Nevertheless, Staff's analysis of the CRBR accident probabilities and consequences is inadequate and unreliable. Staff claims "the uncertainty bounds could be well over a factor of 10 and may be as large as a factor of 100, but is not likely to exceed a factor of 100" (FSFES, p. J-24) As noted previously, the uncertainties in the probability estimates are larger than those of WASH-1400 and the Commission's previous conclusion -- that the numerical estimates of accident probabilities in WASH-1400 are unreliable -- applies equally to the Staff Appendix J analysis. Furthermore, the consequences (i.e., health risks) of "Class 9" accidents at CRBR as estimated by the Staff are based on a series of assumptions with large associated uncertainties. One can find uncertainties of at least two orders of magnitude ~~and~~ in consequences. When these uncertainties are considered together (compounded), I believe they result in an uncertainty of at least two or more orders of magnitude in Staff's estimate of the acute and delayed health effects. With these large uncertainties in the probabilities and consequences, Staff's analysis in Appendix J does not support Staff's conclusions in the FSFES, Section J.1.3, at J-25.

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION
ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)
UNITED STATES DEPARTMENT OF ENERGY PROJECT MANAGMENT CORPORATION TENNESSEE VALLEY AUTHORITY	)
(Clinch River Breeder Reactor Plant)	)

AFFIDAVIT OF DR. THOMAS B. COCHRAN

City of Washington)
) ss:
District of Columbia)

DR. THOMAS B. COCHRAN hereby deposes and says:

The foregoing testimony prepared by me and dated November 12, 1982, is true and correct to the best of my knowledge and belief.

Dr. Thomas B. Cochran

Signed and sworn to before me this 12th day of November 1982.

Donna Marie Shelcer
Notary Public

My Commission Expires July 31, 1987

Exhibit 1, TESTIMONY OF COCHRAN, PART IV

6240

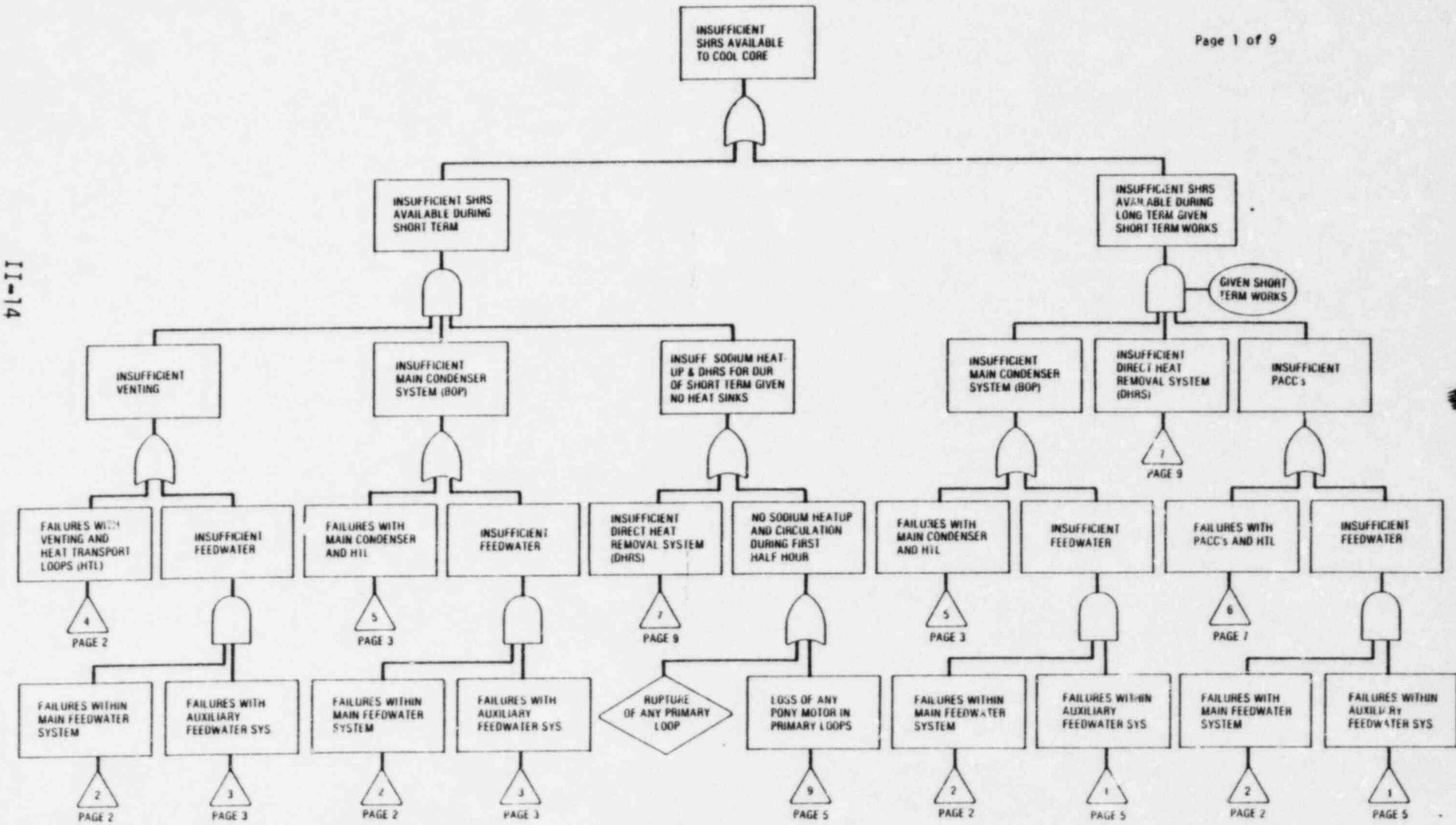
CRBRP-1

CRBRP RISK ASSESSMENT REPORT

MARCH, 1977

VOLUME 2: TECHNICAL APPENDICES

CLINCH RIVER BREEDER REACTOR PLANT



* THE BEGINNING OF LONG TERM IS DEFINED AS WHEN A CLOSED SYSTEM, SUCH AS PACC'S, IS CAPABLE OF REMOVING ALL DECAY HEAT GENERATED WITHIN THE CORE

FIGURE II-1 GENERALIZED FAULT MODEL SHUTDOWN HEAT REMOVAL

6241

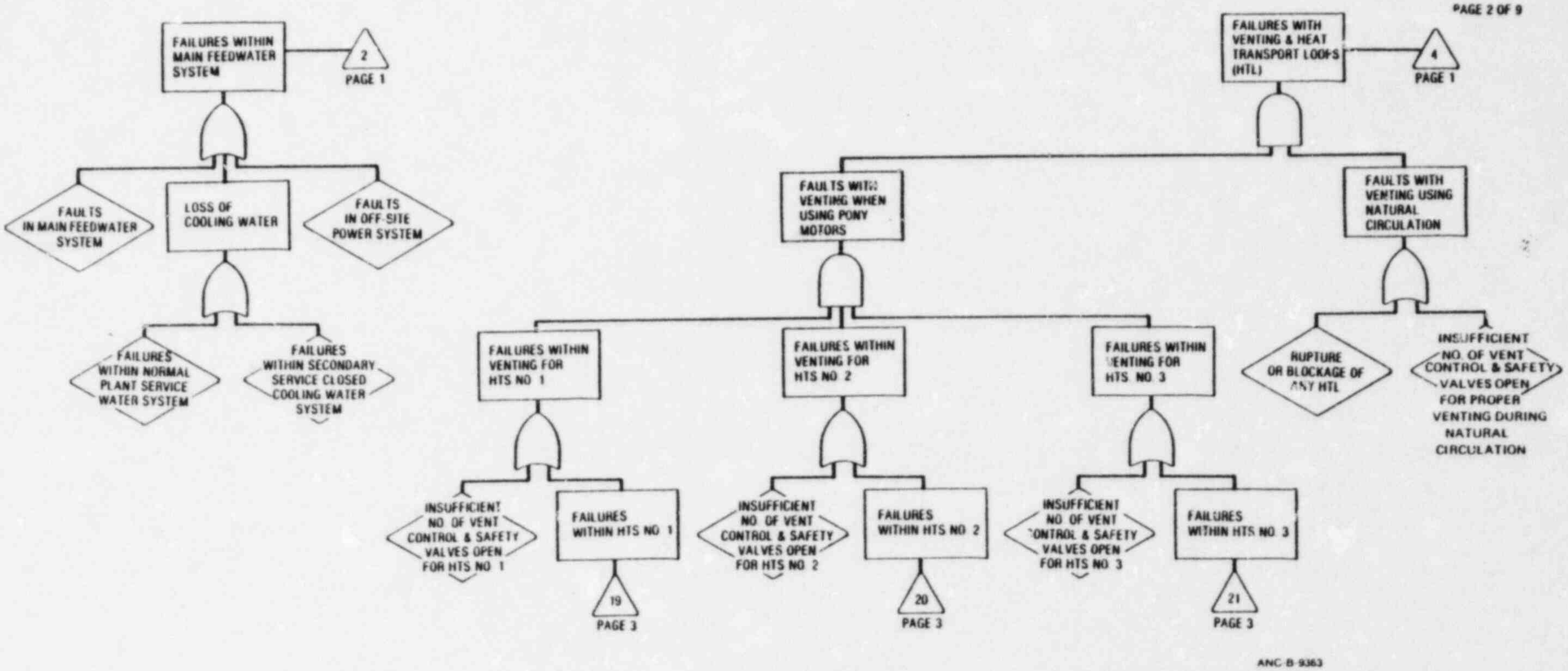
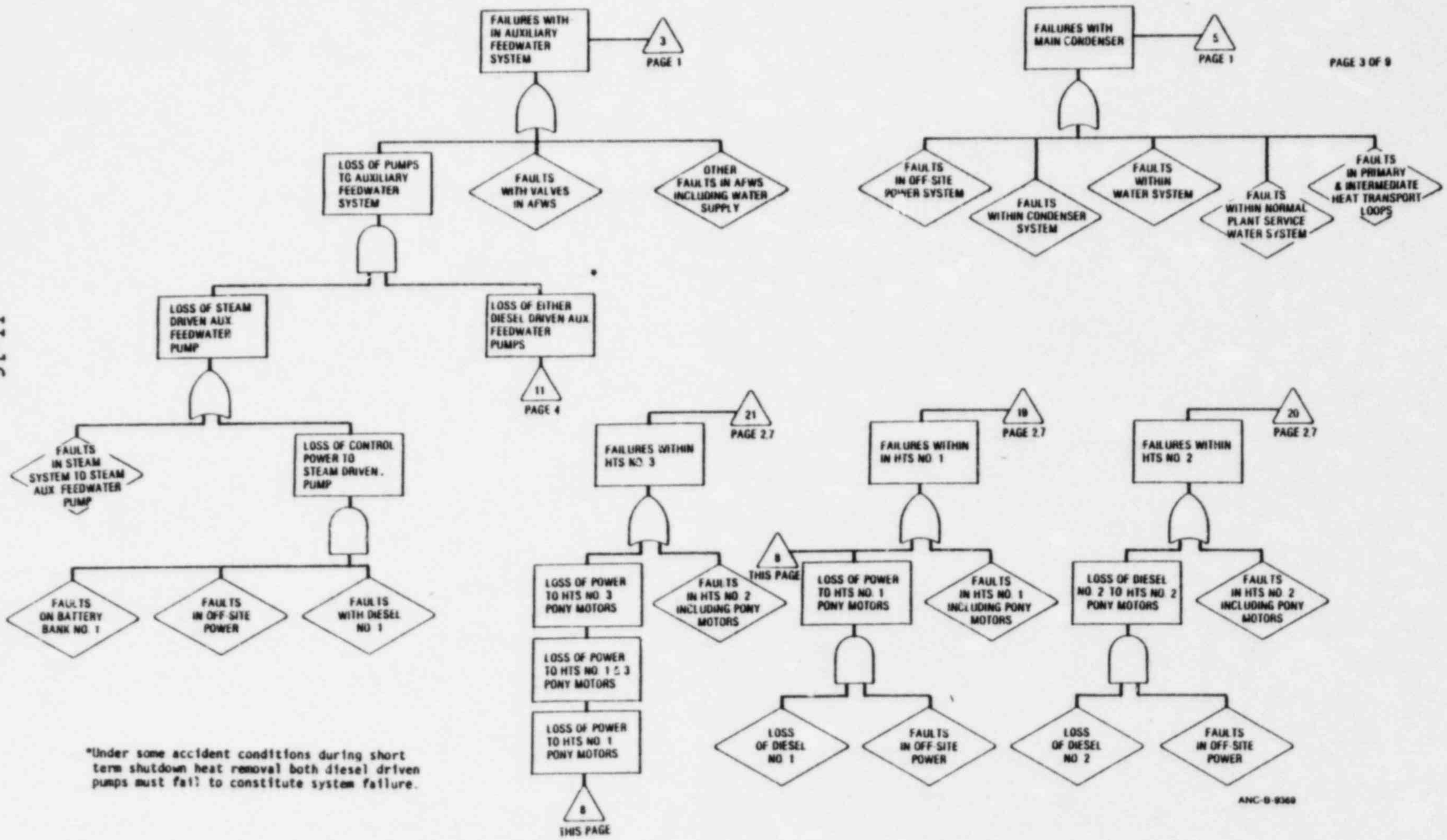


FIGURE II-1 (Con't.) GENERALIZED FAULT MODEL SHUTDOWN HEAT REMOVAL

6242



*Under some accident conditions during short term shutdown heat removal both diesel driven pumps must fail to constitute system failure.

Figure II-1 (Con't) Generalized Fault Model Shutdown Heat Removal

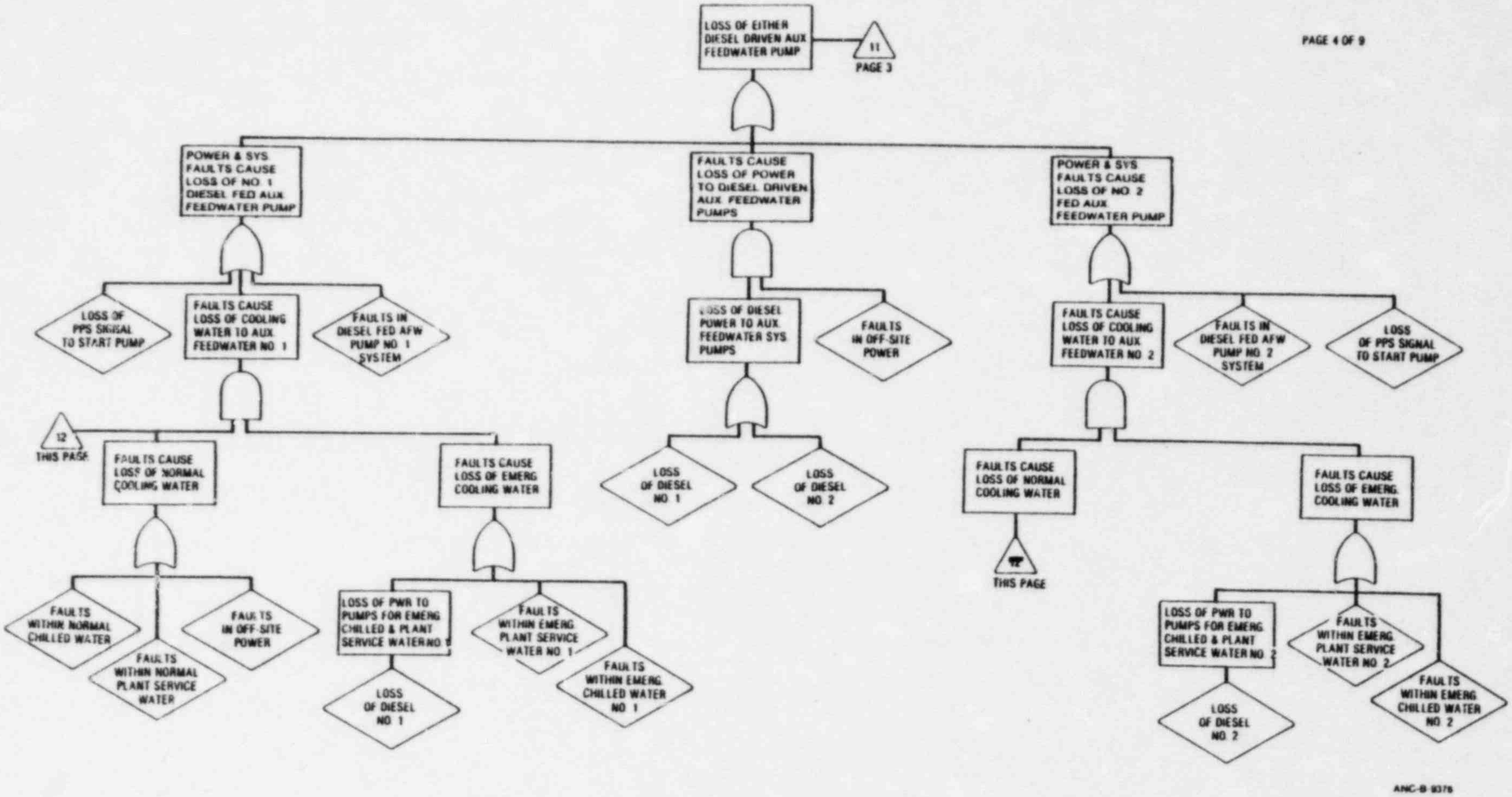


Figure II-1 (Con't): Generalized Fault Model Shutdown Heat Removal

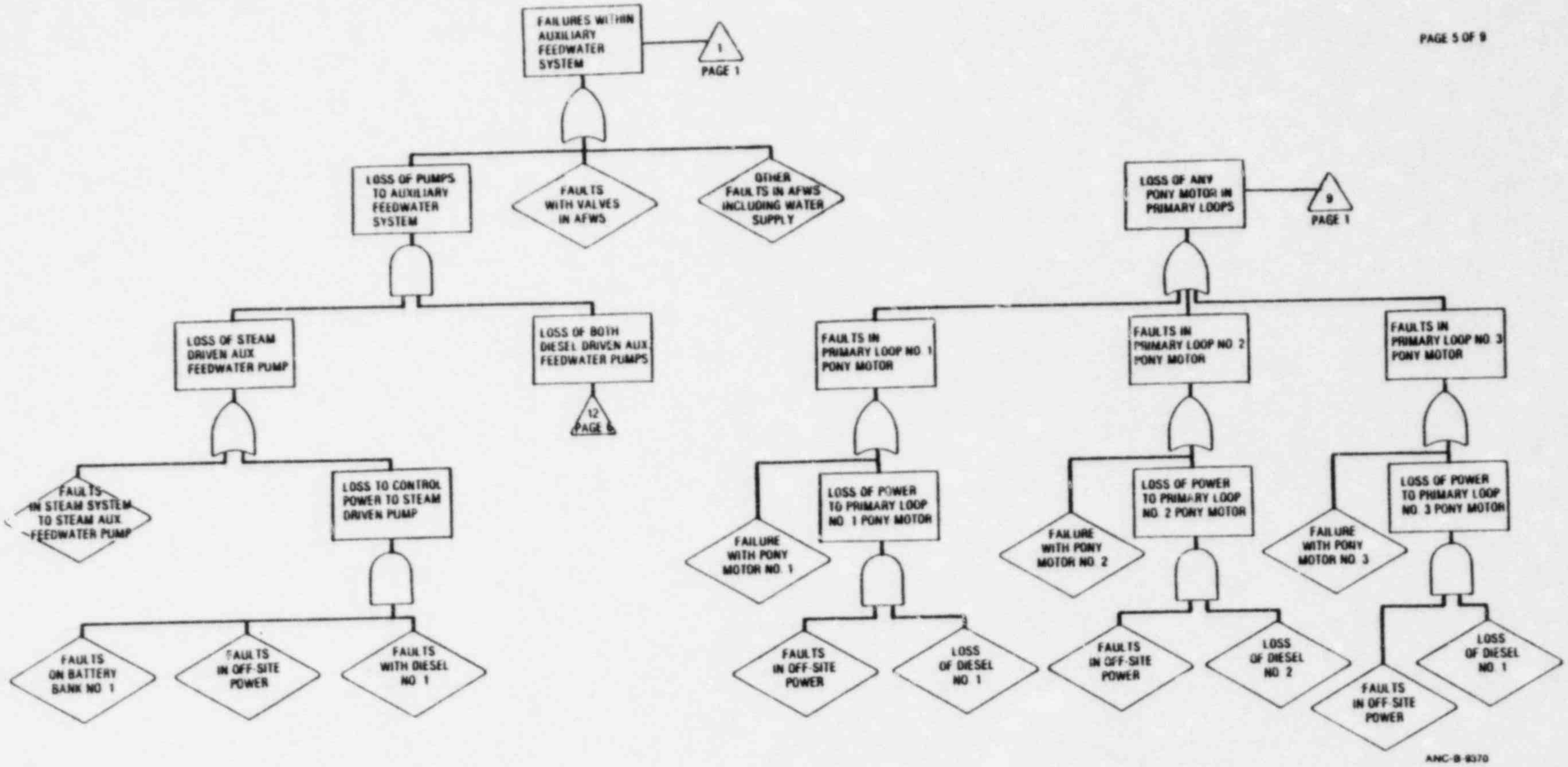


Figure II-1 (Con't): Generalized Fault Model Shutdown Heat Removal

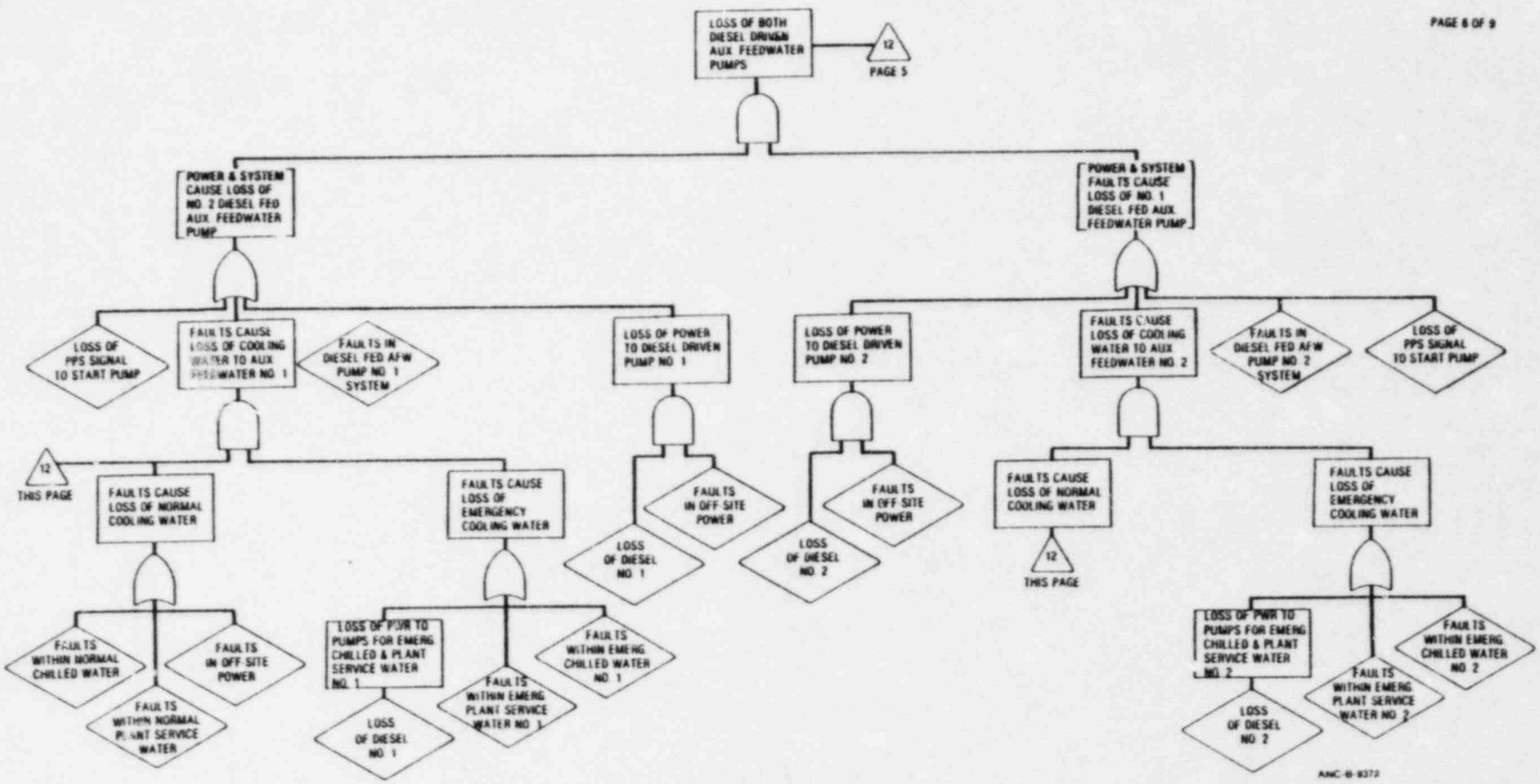


Figure II-1 (Con't): Generalized Fault Model Shutdown Heat Removal

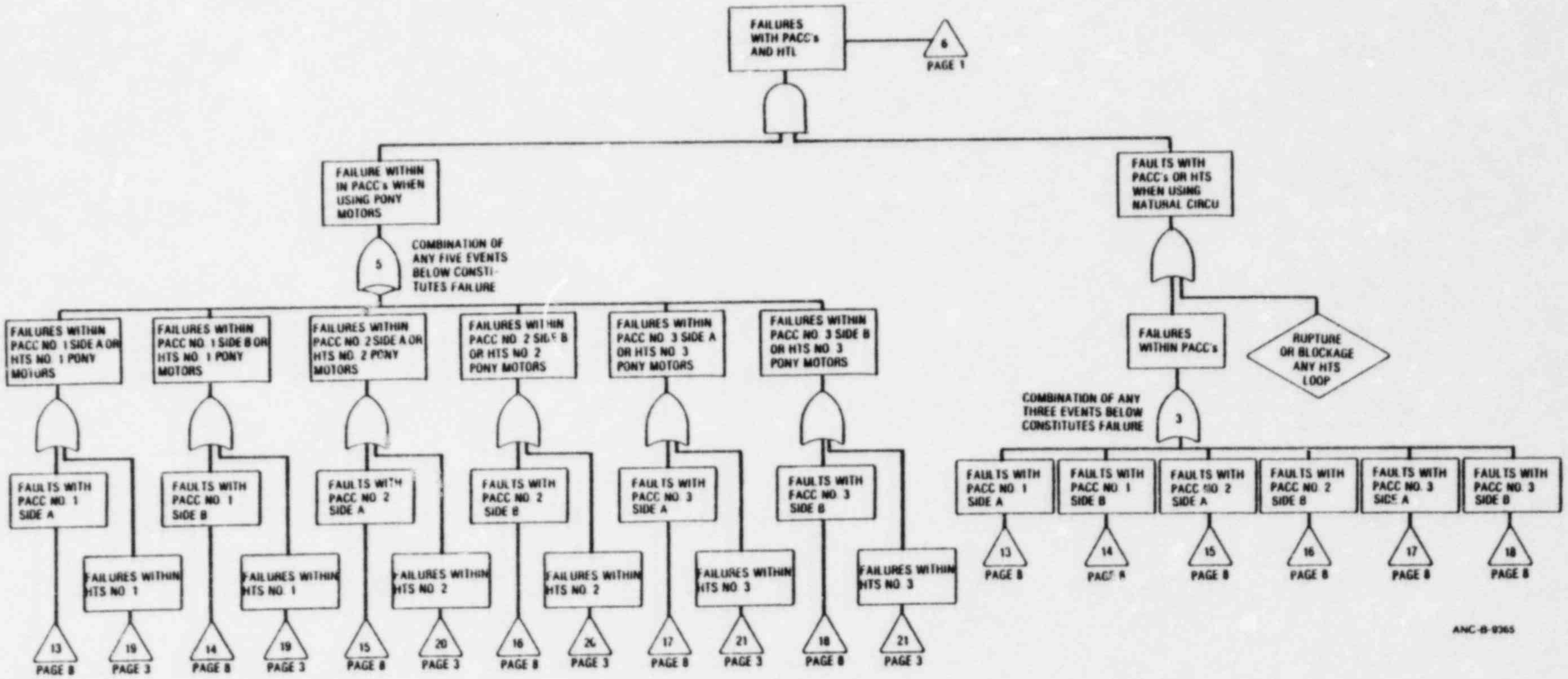


Figure II-1 (Con't): Generalized Fault Model Shutdown Heat Removal

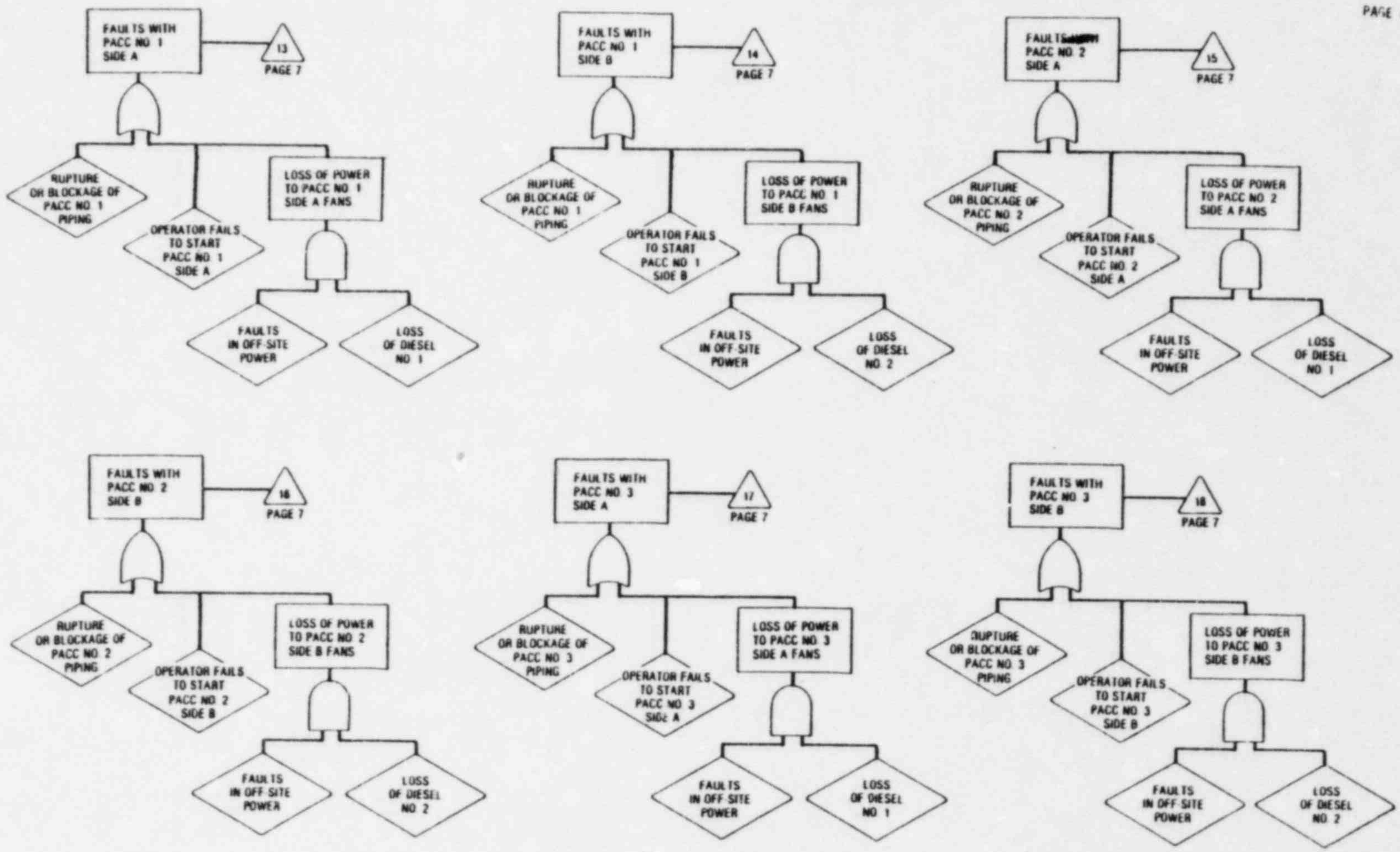


Figure II-1 (Cont'd) Generalized Fault Model Shutdown Heat Removal

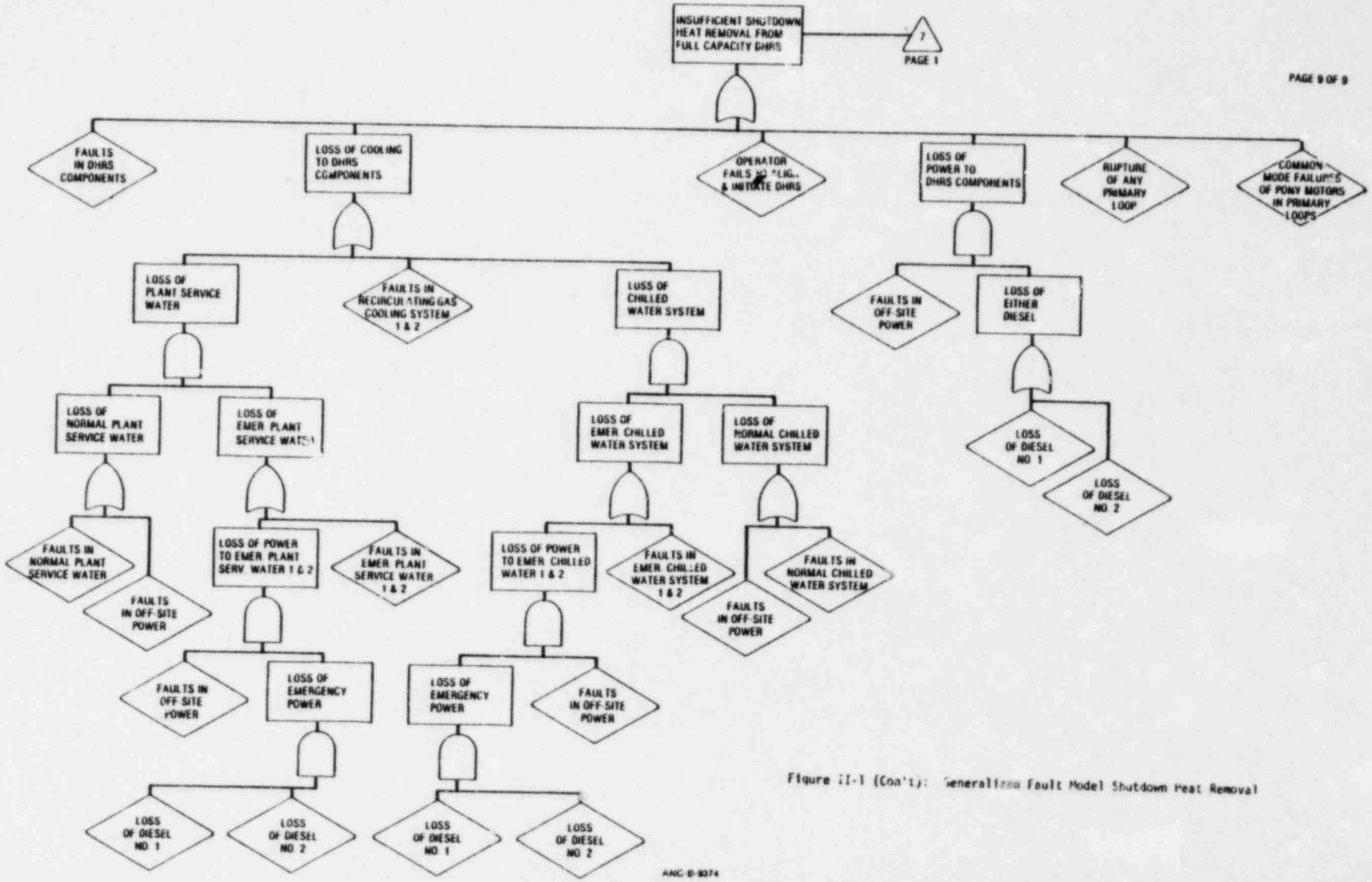


Figure II-1 (Con't): Generalized Fault Model Shutdown Heat Removal

Exhibit 2, Testimony of
COCHRAN, PART IV

6250

COMPTROLLER GENERAL OF THE UNITED STATES
WASHINGTON D.C. 20548

RESTRICTED — Not to be released outside the General Accounting Office except on the basis of specific approval by the Office of Congressional Relations.

B-164105

MAY 25 1982

The Honorable John D. Dingell
Chairman, Subcommittee on Oversight
and Investigations
Committee on Energy and Commerce
House of Representatives

Dear Mr. Chairman:

Subject: Revising the Clinch River Breeder Reactor Steam Generator Testing Program Can Reduce Risk (GAO/EMD-82-75)

Your September 2, 1981, letter asked that we review the technical outlook for several components of the Department of Energy's (DOE's) Clinch River Breeder Reactor (CRBR)--the Nation's first liquid metal fast breeder reactor demonstration plant. In February 1982, your office requested that we issue an interim report on DOE's program for testing CRBR's steam generators. This report responds to that request.

Steam generators for liquid metal fast breeder reactors have had a history of serious technical problems. Small breeder reactors in this country and demonstration breeder reactors in foreign countries have experienced steam generator failures. Steam generators for the CRBR have also experienced a number of problems during their development.

Despite that history, DOE does not plan to conduct complete and thorough tests of the steam generator design to be used in the CRBR. Instead, DOE plans to conduct (1) a series of limited tests on a steam generator which differs significantly from those designed for use in the CRBR, (2) a vibration test on a one-third scale model steam generator, and (3) some inplant testing on a CRBR steam generator after all CRBR steam generators have been fabricated. Without conducting more thorough tests of the CRBR steam generator design before building the CRBR units, DOE is assuming that the CRBR units will operate as predicted.

If DOE is correct, the CRBR will be able to proceed on its current schedule, and the cost will be lower than if more complete and thorough testing were done. If DOE is wrong, the costs and delays associated with redesigning and modifying or rebuilding the CRBR steam generators would be substantial.

(305178)

DOE's decision to forego more thorough tests is based on (1) a belief that the tests that will be done can be extrapolated to predict steam generator performance in the CRBR and (2) confidence that the steam generator design will be successful. Conversely, the history of problems with steam generators and with development of the CRBR steam generators argues for a more complete and thorough testing program.

The following sections present the objective, scope, and methodology of our review; a background on CRBR steam generators; our findings in more detail; and our conclusions and recommendations.

OBJECTIVE, SCOPE, AND METHODOLOGY

Our objective was to evaluate DOE's current program for testing the CRBR's steam generators. To accomplish that objective, we reviewed the history of the development of the steam generators, including the results of past tests and DOE's future plans for testing. We also compared the current CRBR steam generator design with the design of the steam generators tested in the past and currently being tested. Documents concerning the testing program were obtained from DOE headquarters in Washington, D.C.; the CRBR Project Office in Oak Ridge, Tennessee; the Energy Technology Engineering Center in Santa Susana, California; Westinghouse Advanced Reactors Division in Waltz Mill, Pennsylvania; and the Atomics International Division of Rockwell International Corporation at Canoga Park, California.

We also discussed DOE's testing program with the major contractors involved in the steam generator program and with DOE officials. Information concerning steam generator development in foreign countries was obtained from DOE subcontractors and technical publications. To assist us in the technical aspects of this assignment, we employed a consultant who has worked in the nuclear industry for over 30 years and who has an intimate knowledge of liquid metal fast breeder reactors and steam generators.

The information contained in this report represents the best information available at the time of our review. It should be recognized, however, that the testing program changed during our review and, even at the time we issued this report, DOE was considering other options.

We performed our work in accordance with GAO's "Standards for Audit of Governmental Organizations, Programs, Activities, and Functions."

BACKGROUND ON THE CRBR AND THE CRBR STEAM GENERATORS

In 1970, the Congress authorized the Atomic Energy Commission (AEC) 1/ to enter into cooperative arrangements with industry to build and operate the CRBR. During the early and mid-1970s, great urgency was attached to the CRBR program because predictions showed that current generation nuclear reactors would be running out of uranium fuel by the year 2000. The CRBR was initially scheduled to be completed by 1980 to permit a decision in the mid-1980s on commercial deployment of breeder reactors. We are currently completing work on a report which addresses the options available for the timing of the CRBR. That report includes information on a number of factors which have changed since the CRBR was originally authorized. Specifically:

--Current DOE data show sufficient natural uranium to fuel the light water nuclear industry well past the year 2020.

--Latest DOE data show breeders may not be economical until after the year 2025.

In commenting on a draft of that report, DOE argued that it is imperative to proceed with the CRBR schedule--current plans are to have the CRBR operating by 1990--and that any slowing of the program could lead to industrial disruption, constrained economic growth, and increased reliance on foreign energy supplies. While recognizing DOE's comments and concerns over possible delays in its current program, we concluded that the changes in the factors affecting the timing of when breeder reactors may be needed show that slowing the program has become a viable option.

Developing and demonstrating reliable steam generators have been and still are one of the most significant technical problems facing the CRBR project. Steam generators provide the transfer of heat from the reactor coolant to water, which is heated to steam to drive the plant's turbines. According to a Nuclear Regulatory Commission report, 33 of 45 operating nuclear plants with steam generators have experienced some form of steam generator problems. During the 1970s, these problems caused about 21 percent of forced outages at those plants. Many of these problems are operational problems and are not related to design deficiencies or inadequate testing. It is obvious, however, that steam generators are the source of considerable problems in existing nuclear plants. In

1/The Atomic Energy Commission and the Energy Research and Development Administration (ERDA) were predecessor agencies to DOE. AEC was abolished on Jan. 19, 1975, and many of its functions were transferred to ERDA. ERDA's functions were transferred to DOE on Oct. 1, 1977.

comparison to commercial reactors, the steam generators needed for the CRBR represent a more difficult challenge because sodium is used as the reactor coolant. Sodium steam generators impose severe mechanical stresses on the metal barrier between sodium and water within the steam generator. Even a small failure allowing contact between the two fluids raises the possibility of a fire or explosion resulting from a sodium-water interaction.

Breeder reactor steam generator history

According to Atomics International, the fabricator of the prototype steam generator for the CRBR, many designs have been used for breeder reactor steam generators around the world. Atomics International maintains that problems have been experienced in all cases where the steam generator design has not been thoroughly tested.

Smaller breeder reactors in the United States have experienced steam generator problems. For example, a steam generator in the Enrico Fermi reactor (near Detroit, Michigan) failed in 1962 when vibrations and other problems created holes in the metal tubing, allowing contact between the sodium and the water. Other countries have also experienced steam generator problems in breeder reactor plants. Structural integrity problems in a demonstration breeder plant in Russia caused leaks in four of six steam generators. Similar problems delayed full power operations at the British demonstration breeder plant when four of nine steam generators leaked. As recently as April 1982, the French demonstration breeder reactor was shutdown because two sodium leaks in a steam generator caused a fire.

CRBR steam generator program

In 1974, AEC chose a steam generator design for use in the CRBR that was quite different from any previous domestic steam generator, and it was also different from the steam generators used in foreign breeder reactors. During 1974 and 1975, Atomics International was selected to design and fabricate (1) two model steam generators, (2) a prototype steam generator, (3) nine steam generators for use in the CRBR, and (4) one backup unit. Until 1982, DOE's steam generator development program consisted of three major elements.

1. Testing the Model Steam Generators. The model steam generators, tested in 1978, were full-length steam generators but contained only 7 water-carrying tubes instead of the 757 tubes in a plant unit. The purpose of testing the model steam generators was to obtain data on full power steam generator performance and endurance.

2. Testing a Prototype Steam Generator. The prototype steam generator, to be tested in 1982 and 1983, was

originally to have been a full-size, 757 tube prototype of the CRBR steam generators. However, changes to the CRBR design resulting from the testing of the model steam generators and subsequent design reviews could not be fully incorporated in the prototype steam generator and, as a result, the prototype differs significantly from the CRBR steam generator design. The original purpose of building the prototype was to verify the steam generator manufacturing process and to test the structural integrity of the prototype under simulated operating conditions. Prototype steam generator testing is proceeding on schedule.

3. Fabricating and Installing the CRBR Steam Generators. The CRBR steam generators are the units which will ultimately be installed in the CRBR. As previously noted, the design of the CRBR steam generators has changed significantly over the past several years, and DOE does not plan to conduct complete and thorough testing of the current CRBR steam generator design prior to installation of the steam generators in the CRBR.

CRBR officials are currently adding another element to the CRBR steam generator testing program--fabrication of a one-third scale model of the CRBR steam generator--to test the design's ability to withstand flow-induced vibration.

DOE terminated the steam generator contract with Atomics International in 1981 and is currently resoliciting proposals to fabricate the nine redesigned CRBR steam generators and one backup unit. DOE expects to announce award of a contract in the near future.

DOE IS NOT MINIMIZING RISKS IN ITS STEAM GENERATOR TESTING PROGRAM

DOE's program for testing CRBR's steam generators is deficient in that

✓ --model steam generator testing and prototype fabrication were conducted concurrently, thus deficiencies found in the models were not corrected in the prototype;

✓ --prototype testing involves testing a design which is significantly different from the design for the CRBR steam generators;

✓ --prototype testing will not include simulating important operating conditions; and

✓ --the steam generator design to be used in the CRBR will not be completely and thoroughly tested prior to fabrication and installation of all CRBR steam generators.

Problems noted during model steam generator testing were not corrected on the prototype

Because of the perceived urgency of building the CRBR, program officials began fabrication of the prototype steam generator before completing testing of two model steam generators. Under normal conditions, the models should have been tested before fabrication of the prototype began. Initial tests on the model steam generators began in May 1978, but they were prematurely concluded in December 1978 because of deficient performance. Subsequent examination showed that the model steam generators could not withstand fluctuations in temperature because of fabrication errors and inadequate tube spacing and tube support.

The contract for the design and fabrication of the prototype was awarded in September 1975, thus fabrication of the prototype steam generator was well underway when the test results from the model steam generators became available in 1979. As a consequence, the design and fabrication problems noted in the model steam generators were not corrected in the prototype. Instead, major changes were made to the CRBR steam generator design. Therefore, the prototype steam generator scheduled for testing from May 1982 through March or April 1983 is not prototypic of the current CRBR design, and it contains many of the same deficiencies as the model steam generators. Thus, testing the prototype will not identify all the problems that could occur in the CRBR steam generators. In total, the cost of the prototype steam generator tests is about $8.2 million.

Prototype testing inadequate

DOE officials have concluded that the prototype might fail if tested to the limits originally specified to simulate anticipated CRBR operating conditions. As a result, the test program for the prototype was changed to delete or reduce the severity of the tests that were originally planned. The revised test plan approved in July 1981 does not include requirements to demonstrate the

--structural integrity of the steam generator, a major cause of failure in foreign breeder reactors, or

--ability of the steam generator to withstand large temperature changes occurring over a short period of time, the major cause of the model steam generator failure.

In addition, the prototype test never was planned to include the ability of the steam generator to withstand flow induced vibration, the major cause of the Fermi steam generator problems. These tests are critical to predicting performance because they involve the areas most likely to cause failure.

DOE will not fully test the CRBR steam generator design

As currently planned, DOE will not conduct complete and thorough tests of the steam generator design before they are installed in the CRBR. The nine CRBR steam generators and one backup unit are scheduled for delivery between January 1985 and May 1986. DOE plans to test a one-third scale model for flow-induced vibration and at a later date, install various performance-measuring instruments in two CRBR steam generator units and, after all units are installed, conduct pre-operational testing in the CRBR.

The one-third scale model tests will not provide all needed data on the structural integrity of the steam generator design or its ability to withstand large temperature changes over short periods of time. As mentioned previously, problems in these areas have plagued other breeder reactor steam generators. The inplant tests would provide some information related to these issues, but it would be conducted only after the CRBR steam generators have been completed, resulting in the same situation as the concurrent model steam generator tests and prototype fabrication. That is, by the time the inplant tests could occur, it would be too late to modify the CRBR steam generators to correct any major problems that may be discovered without incurring substantial costs and delays.

DOE previously considered complete and extensive testing of a full-scale CRBR steam generator at its Santa Susana, California test facility, in addition to the tests for flow induced vibrations. DOE currently, however, does not plan any additional tests of a full-size steam generator. DOE's Chief of the CRBR plant component branch said that the current steam generator test program is adequate to confirm the design, and that DOE does not wish to unnecessarily delay the CRBR project. According to DOE officials, testing a full-scale CRBR-design steam generator could delay the program by as much as 45 months if fabrication of the CRBR steam generators is halted. If fabrication of these units is not halted, eight CRBR steam generator units would be delivered by the time the test results are available in April 1986. The remaining CRBR steam generators and the backup unit would be substantially complete by that time and would be too far completed for major modifications without incurring large cost and schedule slippages.

Clinch River project officials contend that despite the problems that have been experienced with steam generators, more extensive CRBR steam generator tests are not required, and the tests being conducted are adequate and can be extrapolated to provide the information necessary to predict inplant performance. A Clinch River project official believes additional testing prior to fabrication of the remaining CRBR steam generators would unnecessarily delay the project. Our consultant recognizes the potential problems in the areas of structural integrity and ability of the CRBR steam generators to withstand temperature changes. He also acknowledges that the planned tests will not provide adequate data in these areas. However, he agrees with DOE that any steam generator tests that would result in a delay in the construction of the CRBR are not appropriate.

DOE's prime contractor for the CRBR--Westinghouse Electric--stated that the information gained from the prototype tests will be inadequate for resolving concerns about vibrations and recommended the one-third scale model tests. Westinghouse, however, also recognized that neither test would provide data concerning structural integrity or the CRBR steam generator's ability to withstand temperature changes.

In a February 26, 1982, letter to us, officials of Atomics International--the original designer and fabricator of the prototype steam generator--expressed disagreement with DOE's CRBR steam generator testing program. Atomics International officials recognized that it is highly desirable to minimize development cost, but that it is also highly desirable to minimize the risk of (1) forced outages from failure of untested features and (2) delays in licensing due to a lack of data from component testing under simulated reactor conditions. They noted that the CRBR steam generator design incorporates features which substantially differ from the prototype and are unsupported by tests. According to Atomics International officials, even after completing the prototype test, CRBR steam generator design and performance uncertainties will remain. Atomics International officials concluded that extensive testing of a full-scale CRBR steam generator and a scale model steam generator would eliminate the uncertainties.

In addition to delaying the program for up to 45 months, DOE officials estimate that installation and testing of a full-scale CRBR steam generator would cost about $7 million. This would however, eliminate the need for testing the prototype steam generator. Cancellation of the prototype test would save about $3.2 million, which would reduce the additional cost of testing a full-scale CRBR steam generator to less than $4 million. The resulting program delay and any accompanying inflationary increases would also, of course, impact on the overall CRBR cost and schedule.

We note that DOE's position on testing steam generators is inconsistent with its programs to develop other, perhaps less critical CRBR components. For example, DOE is testing the sodium pumps extensively. These tests have already proved worthwhile because a deficiency, which may result in a change in the plant unit design, has been discovered. It is exactly this type of situation which causes our concern over not testing the CRBR steam generators.

In lieu of tests to provide assurance that CRBR's steam generators will operate as required, DOE could obtain operability guarantees from the steam generator designer or fabricator. However, the contractor, which is selected to fabricate the CRBR steam generator, will have to guarantee only that the steam generators will be built in accordance with the design provided by Westinghouse. DOE officials stated that they will not request an operability guarantee for the fabricator because no company would provide such without first reviewing in detail the steam generator design. DOE officials stated that such a review would delay the program and increase program costs.

If the steam generators were to be built in accordance with the stated technical requirements, but failed because of design deficiencies, the Government would have to assume the additional costs of amending the design and reworking the steam generators because the design has not been guaranteed by Westinghouse--the lead reactor manufacturer. DOE officials explained that Westinghouse officials would not likely guarantee the steam generator design because it is developmental and a guarantee of that nature would be too risky.

CONCLUSIONS

In essence, DOE's steam generator testing program is based on the urgency of proceeding with the CRBR. This has been pointed out most recently in a DOE letter containing comments on a draft GAO report on options for the timing of the liquid metal fast breeder reactor program. (See p. 3.) While recognizing DOE's concerns and its desire to move forward as expeditiously as possible, our work shows that changes in the factors affecting the timing of when breeder reactors may be needed make slowing the breeder program and the CRBR a viable option.

The highly critical nature of the steam generator to overall CRBR success makes a strong argument for taking a cautious, conservative, and prudent approach to developing, fabricating and testing the CRBR steam generators. DOE--as well as our consultant--disagree and are confident that the steam generator, as currently designed, will operate as predicted. They base this position on their confidence in the technical design and testing program, and because they do not believe the CRBR program should be delayed by steam generator testing. This position, however, is not supported by (1) the history of steam generator development, (2) the test results to date, (3) DOE's program to test other CRBR components, and (4) the DOE contractor who designed and fabricated the prototype steam generator.

We recognize that all steam generator problems are not related to design deficiencies and that testing cannot eliminate all elements of risk. The ultimate test must come when the steam generators are operated in the CRBR. A good testing program can, however, minimize the risk involved. In this regard, DOE's current test program does not minimize the risk involved as it will not provide complete and thorough information in two critical areas where problems have been experienced in other breeder reactor steam generators, both in this country and abroad--the structural integrity of the steam generators and their ability to withstand large temperature changes over short periods of time. Without testing the CRBR steam generator design to obtain data in these two areas prior to fabricating the CRBR steam generators, DOE is assuming that the steam generators will work. If DOE is right, CRBR will be completed sooner at a lower overall cost. If wrong, it will prove a more costly and time-consuming risk to take.

In our view, DOE has several fundamental options to obtain the required data. More complete and thorough tests of the one-third scale model would provide much of the required data, but would be limited in that it would not provide full-scale data. Testing a full size CRBR steam generator could theoretically provide more complete data, but may not provide full vibration data. A third option would involve a combination of the scale model and full-scale tests and would provide data in all critical areas. Although conducting any additional testing would increase program costs and delay the program, we believe that minimizing the risks through a more complete and thorough testing program is far more attractive than the risk associated with purchasing steam generators which may not operate as required. Should the steam generators prove inadequate for optimal operation in CRBR, DOE would have to finance modification of the 10 completed steam generators or scrap the completed units and build 10 new steam generators.

We recognize that because of the complexity of the CRBR and because it is a research and development effort, some element of risk will always be involved. However, we believe a cautious, conservative, and prudent approach to developing, fabricating and testing this highly critical component should be taken to minimize that risk. For this reason, the information developed in our review is most supportive of the following courses of action.

--Stopping the CRBR prototype steam generator test program because of the limited value of testing a steam generator which differs significantly from the current CRBR design.

--Canceling the current solicitation for the fabrication of 10 CRBR steam generators.

--Developing a program for more complete and thorough testing of the CRBR steam generator design in as expeditious a timeframe as possible.

--Withholding a decision on procuring the CRBR steam generators until test results are received and evaluated and any necessary design modifications made.

RECOMMENDATION

We recommend that the Secretary of Energy evaluate the information presented in this report, as well as the risk assumed in not conducting more complete and thorough tests of the steam generator design, in deciding on how to proceed with the procurement of the CRBR steam generators.

- - - -

As arranged with your office, unless you release or publicly announce its contents earlier, we plan no further distribution of this report until 30 days from the date of the report. At that time, we will send copies of this report to the Director, Office of Management and Budget, the Secretary of Energy; and to other interested parties and make copies available to others upon request. At your request, in order to provide this report in time for use during the appropriation process, we did not solicit DOE's comments on this report. The information presented in this report was, however, discussed with responsible DOE officials to ensure accuracy.

Sincerely yours,

Charles A. Bowsher

Comptroller General
of the United States

Exhibit 2, TESTIMONY OF COCHRAN, PART IV

6261

17 November 1978

Mr. H.B. Piper
U.S. Department of Energy
Clinch River Breeder Reactor Plant
Project Office
Oak Ridge, Tennessee 37830

Dear Henry:

Attached are the results of SAI work on two FY-78 risk assessment tasks:

- Accident initiating event completeness and methodology review
- Resolution of project comments

As noted in my October 4, 1978 memo to you, the one outstanding project comment concerned pipe rupture probability. Accordingly, the enclosed report on our pipe rupture work completes the task on comment resolution.

Also enclosed are two of the references (based on earlier SAI work) which are referred to in the pipe rupture report. Please make these references available to LRM people as necessary. We will provide information on the references to EG&G as required.

Please feel free to contact me should you have any questions on the enclosed reports.

Sincerely,

Dave Leaver

David Leaver

DL/imp

cc: P.J. Wood, SAI/Pittsburgh
T.A. Zordan, W-LRM
R.J. Crump, EG&G

Enc/4

Science Applications, Inc. 5 Palo Alto Square, Suite 200, Palo Alto, CA 94304 (415) 493-4326

RELATIVE PIPE RUPTURE PROBABILITY FOR THE PRIMARY HEAT TRANSPORT SYSTEM OF CRBRP

By D. O. Harris
SCIENCE APPLICATIONS, INC.
Palo Alto, California

November 13, 1978

INTRODUCTION

This note is intended to summarize the results of work performed within the last year in estimating the probability of a pipe rupture within the primary heat transport system of the Clinch River Breeder Reactor Plant. An earlier note dated October 7, 1977, and included as Reference 1, discussed a possible means of tying the probability of pipe rupture in CRBR to values used for LWR's. LWR values have been suggested in the past, and are generally estimated with greater confidence than corresponding values for CRBR. A meeting between SAI, WARD and Westinghouse LWR personnel was held at WARD on December 15, 1977 in an attempt to obtain stress histories for LWR's that were calculated in the same manner as employed in the CRBR analysis. The use of stress histories for the two types of plants that were calculated by comparable means would allow the comparative rupture analysis to be performed with greater confidence. However, it was not possible to obtain such results for a LWR, and it was therefore necessary to fall back on stress analyses of LWR piping that were generated by vendors other than Westinghouse - using analytical techniques that may or may not be comparable to those employed for CRBR. This "fall-back" position had been employed earlier, with Reference 1 providing results obtained prior to October 1977.

Various questions regarding certain aspects of the analytical techniques for calculating pipe rupture probabilities were raised in discussions with Westinghouse personnel. These included the following items:

- Calculated results will depend strongly on the initial crack size distribution. What is the influence of using distributions other than the one originally employed?

- Why use weld volume to normalize the probability of having a defect? Wouldn't weld length or area provide a better basis for normalization.

- What criterion for a rupture is used? Is it merely a leak, or a guillotine failure?

The purpose of this note is to summarize results obtained using the results of stress analyses on LWR piping that were provided by vendors other than Westinghouse, and incorporating various initial crack size distributions and means of normalization of results. The end result to be included here is the ratio of overall time averaged failure rate of the primary piping of CRBR vs. various LWR's. The question of break size has not been addressed.

STRESS ANALYSIS AND CRACK GROWTH CALCULATIONS

As mentioned above, it was not possible to obtain the results of a LWR piping stress analysis that was performed in the same manner as used for CRBR. Therefore, it was necessary to employ results that are available to SAI from vendors other than Westinghouse. For instance, the cyclic peak stresses at various locations in the primary piping of a Babcock and Wilcox

PWR are summarized in Table 3, page 29 of Reference 2. A copy of this reference is enclosed. A fatigue crack growth analysis for various locations in the piping was performed. This analysis employed various conservative assumptions, as discussed in Reference 1, and the initial defect size in the hot-leg to pressure vessel joint that would just grow to the critical depth within the plant lifetime was used in comparison with CRBR results presented earlier. Such results from various reactors are presented below. These values are directly from Table 1 of Reference 1.

	CRBR hot-leg	CRBR cold-leg	PWR #1	PWR #2	PWR #3 from Ref. 2
joint considered	most highly stressed	most highly stressed	hot-leg -PV	hot-leg -PV	hot-leg -PV
a_{tol}, tolerable initial defect depth at end of life, in.	0.096	0.20	0.090	0.17	0.165
no. of weld joints in primary piping	57	96	37	36	33
joint thickness, in.	0.5	0.5	3.75	3.00	3.3125
pipe OD, in.	24	24	50	40	42.75

The cumulative probability of failure of the joint within the plant lifetime is simply the probability of having a defect in the joint of a size deeper than the tolerable depth given in the above table. This is a function of the as-fabricated crack depth distribution, the probability of having a defect to begin with, and the inspection procedure employed. Various initial defect distributions will be considered here, and a pre-service ultrasonic (UT) or radiographic (RT) inspection will be considered. Normalization of the probability of having a defect based on weld volume, weld area, and weld length will be employed - with the following notation used.

type of normalization	parameter for a given joint	prob. per unit of normalization of having a defect
volume	$2\pi Dh^2$	p_v^*
area	$2\pi Dh$	p_A^*
length	πD	p_ℓ^*

The weld volume and area include the heat affected zone. The parameters p_v^*, p_A^*, and p_ℓ^* are the least well known of the inputs to the analysis. Fortunately, these parameters cancel out in taking the ratio of CRBR to LWR ruptrue probabilities (assuming that they are about the same for the welds employed in the two types of plants).

AS FABRICATED CRACK DEPTH DISTRIBUTIONS

The as-fabricated crack depth distribution employed in References 1 and 2 was obtained from Wilson (Ref. 3), and was the following

$$P_{cond}\,(>a) = \frac{1}{2}\,\mathrm{erfc}\,\left(\frac{1}{\mu 2^{\frac{1}{2}}}\ln {}^{a}/_{\lambda}\right)$$

$$\mu = 1.53 \qquad \lambda = 1.36\mathrm{x}10^{-3}\ \mathrm{in.} \qquad \text{(Wilson)}$$

This corresponds to a log-normal distribution of crack depth.

Becher and Hansen[4] provide information on experimental measurements of crack size distributions in welds. A log-normal distribution provides a good fit to their data with

$\mu = 1 \quad \lambda = 0.04$ in. (Becher & Hansen)

The Marshall report[5] provides another estimate of crack depth distribution which is more applicable to nuclear pressure vessels. However, it will be assumed to also be applicable to piping. Reference 5 provides the following distribution

$$P_{cond}\ (>a) = e^{-a/a'} \qquad a' = 0.25 \text{ in.}$$

This distribution provides an appreciable probability of having a defect deeper than the pipe thickness--which is meaningless. To correct this deficiency, this exponential distribution will be truncated at a = h (h = thickness). This provides the following result

$$P_{cond}\ (>a) = \frac{e^{-a/a'} - e^{-h/a'}}{1 - e^{-h/a'}} \qquad \text{(truncated Marshall)}$$

(The term in the denominator is required so that $P_{cond}\ (>0) = 1$.)

DETECTION PROBABILITIES

Various pre-service inspections will be considered for the plants under consideration. PWR#3 will be taken to have a UT pre-service inspection, with the following probability of not detecting a defect of depth a being given by the following expression

$$P_{ND}(a) = \tfrac{1}{2}\,\mathrm{erfc}(\nu \ln a/a^*) \qquad \text{(UT).}$$
$$\nu = 1.33 \qquad a^* = \tfrac{1}{4} \text{ in.}$$

This relation is given in Reference 2, and was estimated from experimental data. PWR's# 1 and 2 will be considered to have had an RT pre-service inspection, in which case the following expression from Reference 2 is applicable

$$P_{ND}(a) = \tfrac{1}{2}\ \mathrm{erfc}\ (\nu\ \ln a/0.6h) \qquad \text{(RT)}$$

$$h = \text{thickness} \qquad \nu = 2.3$$

Which of these inspection procedures is employed for pre-service inspection does not have a large influence on the failure probabilities.

The non-detection probability for use in conjunction with the Marshall distribution was fitted to an exponential relation in order to simplify the analysis. The data summarized in Figure 15, page 62 of Reference 1 shows a great deal of scatter in P_{ND} -a for a RT inspection. Hence, it is not possible to tell if the data is better fit by a log normal or exponential distribution. The following relation was found for a radiographic inspection.

$$P_{ND} = \begin{cases} 1 \text{ for } \alpha = a/h < \alpha_0 = 0.76 \\ e^{-\beta(\alpha-\alpha_0)} \text{ for } \alpha > \alpha_0 \qquad (\beta = 9.5) \end{cases}$$

<u>POST-INSPECTION DISTRIBUTIONS AND FAILURE PROBABILITIES</u>

The crack depth distribution following pre-service inspection can be found from the as-fabricated distribution and non-detection probabilities as follows

$$P_{cond(post\text{-}insp)}\ (>a) = \int_a^{a_{upper}} p_o(x)\ P_{ND}(x)dx$$

where a_{upper} is some upper limit on crack depth, such as the wall thickness. The function $p_o(a)$ is obtainable from the above results for $P_{cond}(>a)$, because

$$p_o(a) = - \frac{d}{da} P_{cond}\ (a)$$

The conditional probability of failure of a given joint within the plant lifetime is then given by

$$P_{f(cond)} = P_{cond(post\text{-}insp)}\ (>a_{tol})$$

The average failure rate (per plant-year) for the given joint will then be

$$\overline{P}_{f(joint)} = P_{f(cond)}\ X(\text{prob. of having a defect})/(\text{lifetime})$$

The probability of having a defect in the joint depends on the basis of normalization (area, vol, etc.) as discussed above. For instance, using weld area as the basis of normalization, and assuming p_A^* is very small

$$\overline{P}_{f(joint)} = P_{f(cond)}\ p_A^*\ A/(\text{lifetime})$$

The plant liftime is taken as 30 years for CRBR and 40 years for the LWR's. The overall average failure rate for the plant will be $\overline{P}_{f(joint)}$ x(no. of joints). Taking ratios of CRBR to LWR values results in the factors such as p_A^* cancelling out (as was discussed above).

The ratios $\overline{P}_{f(CRBR)}/\overline{P}_{f(LWR)}$ for various bases of normalization and various crack size distribution are summarized in Table 1.

TABLE 1

$\overline{P}_{f\ (CRBR)} / \overline{P}_{f\ (LWR)}$ For Various Bases of Normalization

and Initial Crack Depth Distributions

(pre-service RT inspections, unless otherwise noted)

Basis of Normalization	Crack Dist.	PWR #1	PWR #2	PWR #3
weld volume	Wilson	.0186	.0618	.264*
	Becher & Hansen	.0174	.0493	.156*
	Marshall (no insp.)	.0309	.0423	.0697
	Marshall (RT)	.0276	.0378	.1298
weld area	Wilson	.139	.744	1.75*
	Becher & Hansen	.130	.592	1.04*
	Marshall (no insp.)	.232	.512	.463
	Marshall (RT)	.207	.457	.859
weld length	Wilson	1.04	4.45	11.62*
	Becher & Hansen	.974	3.54	6.89*
	Marshall(no insp.)	1.74	3.06	3.08
	Marshall (RT)	1.55	2.73	5.72

* UT pre-service inspection for PWR.

DISCUSSION

The results of Table 1 show a wide range of values, varying from 0.0186 to 11.62 (i.e., three orders of magnitude). However, for a given plant and basis of normalization, the numbers vary by much less--typically half an order of magnitude. Thus, it can be concluded that the crack size distribution and detection probabilities of the pre-seismic inspection do not have a large influence. In fact, the plant-to-plant variation from PWR to PWR are larger than the variations due to different initial crack depth distributions and inspections.

The variable having the largest influence on the results of Table 1 is the basis of normalization. This is because of the large differences in the pipe diameter and thickness of PWR piping as constrasted to the CRBR piping, as well as the large differences of the number of weld joints employed in the two types of plants. Normalization with respect to weld length does not seem to make as much sense as using volume or area, because the region affected by a weld includes the heat affected zone--which is generally about 2 wall thicknesses wide. Hence, it appears likely that 1 ft. long weld in a 4 in. thick plate would be much more likely to have a crack than a 1 ft. long weld in a ½ inch thick plate. However, whether the volume or surface area should be used is not clear. Fatigue cracks, such as are considered in this analysis, generally originate at surfaces, and their growth is accelerated by the environment at the surface. This would suggest that weld surface area is the controlling parameter. However, it would seem that constraint resulting from thicker weld section would result in a larger number of defects, so that volume may play a role. With the present state of knowledge, it is not possible to ascertain the controlling parameters. Discarding weld length as the basis, it would be conservative to assume that weld area is the controlling factor, in which case the ratio of average failure rates of CRBR to PWR's falls within the range of about 0.1 -1.

CONCLUSIONS

The above discussions lead to the conclusion that the failure rate of primary piping in CRBR is 0.1 -1 times the corresponding value for a PWR. The largest source of variation in this number is plant-to-plant variations in the three PWR's considered. The ratio of failure rates is not strongly influenced by pre-service inspections or the use of various candidate initial defect depth distributions.

REFERENCES

1. D. O. Harris, "A Note on the Pipe Rupture Probability Calculations for the Primary Heat Transport System of CRBRP," Science Applications, Inc., Palo Alto, California, October 7, 1977.

2. D. O. Harris, "An Analysis of the Probability of Pipe Rupture at Various Locations in the Primary Cooling Loop of a Babcock and Wilcox 177 Fuel Assembly Pressurized Water Reactor - Including the Effects of a Periodic Inspection," Science Applications, Inc., Report No. SAI-050-77-PA, Palo Alto, California, September 1977.

3. S. A. Wilson, Estimating the Relative Probability of Pipe Severance by Fault Cause, General Electric Company Report GEAP-20615, Boiling Water Reactor System Department, San Jose, California, September 1974.

4. P. E. Becher and B. Hansen, "Statistical Evaluation of Defects in Welds and Design Simplications," Danish Welding Institute, Danish Atomic Energy Commission, Research Establisment Riso.

5. "An Assessment of the Integrity of PWR Pressure Vessels," Report of a Study Group Chaired by W. Marshall, available from H. M. Stationary Office, London, Oct. 1976.

CLINCH RIVER BREEDER REACTOR

HEARING

BEFORE THE

SUBCOMMITTEE ON NUCLEAR REGULATION

OF THE

COMMITTEE ON ENVIRONMENT AND PUBLIC WORKS UNITED STATES SENATE

NINETY-FIFTH CONGRESS

FIRST SESSION

JULY 11, 1977

SERIAL NO. 95-H27

Printed for the use of the Committee on Environment and Public Works

Exhibit 3 to Cochran
Testimony, Part IV
Docket No. 50-537

6273

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CONTENTS

6274

In conclusion, the NRC staff found the proposed Clinch River site acceptable from an environmental and safety standpoint for the CRBR, assuming that the ERDA programmatic environmental statement was dispositive of the need for a demonstration-scale facility, including its timing and objectives. Naturally, if the findings in the programmatic statement in these critical areas were to change, we would be obligated to again review the environmental acceptability of the Clinch River site based on considerations relevant at that time.

Thank you, Mr. Chairman.

Senator HART. Thank you, Mr. Gossick. I would like to welcome Senator Bumpers, who is a member of the Energy Committee. He has indicated an interest in this subject and has been deeply involved in the general question of the project for the last number of weeks.

We are pleased to have him with us. Unless there is objection, I would like to go to the ERDA testimony and then have questions for both sets of panelists, if that is agreeable to everyone.

I would like to ask one procedural question, Mr. Gossick. That is, in your testimony you say that the NCR staff obtained from ERDA a copy of the Burns & Roe memorandum dated July 6, 1973. That is almost 4 years after the date of the memo. Can you explain to us, according to your own procedures for licensing, why a memorandum calling into serious question a project of serious counts as you described in your statement was not made available to the NRC in 4 years?

Mr. GOSSICK. Sir, as I understand the status of the document, it was an internal memorandum. It was not a part of the material filed by ERDA in the proceeding at that time.

Senator HART. Was it in ERDA's possession?

Mr. GOSSICK. I don't know, sir.

Mr. BECKJORD. No, sir. We received it about 2 weeks ago, after the document was released to the newspapers.

Senator HART. So, it was not in your possession during that period of time?

Mr. BECKJORD. No; it was not.

Senator MCCLURE. What is your customary procedure with respect to the internal memorandum of contracting agencies? Would you normally see this kind of a memorandum? What is the normal flow of that internal information with the contractor, with a regulatory agency or ERDA?

Mr. GOSSICK. Sir, I would point out in this case, of course, that ERDA constituted the applicant to NRC. I think the question is properly one that ERDA should address. We are not involved with contractors in a regulatory sense.

Senator MCCLURE. In the regulatory sense, then you would not ordinarily see the internal document of the applicant?

Mr. GOSSICK. No, sir, unless it would become a part of the——

Senator MCCLURE. Except those portions of their internal documents which they choose to present to the regulatory agency?

Mr. GOSSICK. Yes, sir.

Senator MCCLURE. In support of their application?

Mr. GOSSICK. Yes, sir.

Senator MCCLURE. If there are questions in the minds of what is now NRC would it [illegible]

Mr. GOSSICK. I know of no reason why we could not ask, depending upon the nature of our concern.

Senator MCCLURE. Have you done so?

Mr. GOSSICK. I am not aware of any incident in this particular case, Senator.

Senator MCCLURE. I wonder if the ERDA witness might also, Mr. Beckjord, perhaps you could indicate whether ERDA looks at applicants' internal memorandums.

Mr. BECKJORD. Senator, we see correspondence that is directed between the various participants in the project, as well as the correspondence that we receive directly. We participate in design review meetings. We ask questions to which there are responses. We feel that we see the important information through those methods of receiving it. As regards internal memorandums, if they are sent to us, we are aware of them. If they are not sent to us, we are not aware of them.

Senator BUMPERS. Mr. Chairman, I don't want to interrupt the proceedings, but I would like to ask a question. Due to the magnitude of the question raised in the internal memo about the suitability of the site which states that it is indeed the worst site ever selected for a nuclear powerplant, I am curious as to whether at that time or subsequent to that time, Burns & Roe called it to your attention that they thought it was the worst site ever selected.

Mr. BECKJORD. I am not aware of the use of these words, Senator. We certainly were aware that there were matters which had to be investigated with respect to the technical suitability of the site. I am going to cover that in my testimony.

Senator BUMPERS. But you don't have any direct correspondence indicating their concern about this site?

Mr. BECKJORD. There is considerable correspondence regarding the technical suitability of the site, Senator, work that was done at that time, particularly site borings, that type of information we were aware of. A complete evaluation of all that information was done before the final placement of the plant was decided upon, and considerable analysis was performed to support it.

Senator BUMPERS. I won't pursue this any further at this time.

Senator HART. I think it is obvious that the concern of the committee and many of us is whether the architect-engineer was saying one thing internally and another thing to the appropriate Government agencies. That is what we want to pursue. We would like to go forward with the ERDA testimony, Mr. Beckjord, accompanied by Mr. Lochlin Caffey, Director of the Clinch River project.

STATEMENT OF ERIC S. BECKJORD, ACCOMPANIED BY LOCHLIN CAFFEY

Mr. BECKJORD. Mr. Chairman and members of the committee, I appreciate this opportunity to discuss the environmental and safety matters related to the Clinch River Breeder Reactor Plant project which were raised in the July 6, 1973, internal Burns & Roe memorandum recently cited in the press.

With your permission, Mr. Chairman, I will submit my written testimony for the record and reduce the part that I give to you.

Senator HART. Without objection, that would be very agreeable to us.

Senator DOMENICI. Might I ask one clarifying question before he testifies? In the practical field of contracting, what does an internal memorandum mean? Who were they writing this to? What was the purpose of it? How does this occur in the day-to-day business of evaluating that kind of site?

Mr. BECKJORD. Senator, my understanding from the information available to us, which is the memorandum and the statement which Burns & Roe made to the press when this was released, is that this was an internal memorandum, the purpose of which was to advise the directors of Burns & Roe of the situation of the project with some recommendations regarding their subsequent business actions toward the project.

The was the purpose of the memo. As indicated, it was a private and internal memo. Evidently they did not intend to make that particular document available to the project.

Senator DOMENICI. It was their own assessment, directed at their people, as they proceeded to evaluate their job?

Mr. BECKJORD. At the time, there were evidently a number of important business decisions that the company intended to make. I think that is covered in that clarifying statement. The purpose of the memo was to address those decisions.

Senator DOMENICI. Thank you, Mr. Chairman.

Mr. BECKJORD. I reviewed the Burns & Roe memorandum in detail. My statements on it are based on information available to me.

The CRBRP project is a joint government-industry cooperative arrangement for demonstrating a liquid metal fast breeder reactor power plant as authorized by Congress on June 2, 1970—Public Law 91-273. The partners in this project are the Energy Research and Development Administration, Commonwealth Edison, Tennessee Valley Authority, and Project Management Corp.

The objectives of this project are to design, license, construct, test and operate an LMFBR demonstration plant. In May 1976, ERDA assumed full management control of the project with continued utility industry support and participation.

I have had the ERDA responsibility for this project since March 1976. During that time, project accomplishments have been good, with design now over 40 percent complete, all of the longlead equipment on order, and the final environmental statement and site suitability report issued by the Nuclear Regulatory Commission.

I have examined project records, reviewed the numerous reports and hearings concerning the project, and inquired extensively into project procedures and status, particularly in environmental, safety and related licensing matters. Generally, I can say that the project has also made good progress in these licensing areas during the past year, working toward its goal of a limited work authorization as required under the NEPA act of 1970, until the recent suspension of the environmental hearings in April. The environmental hearings suspension was requested by ERDA, pending a final decision on whether the project is to be terminated or continued.

I have reviewed the Burns & Roe memorandum in detail since it became available [illegible]

6276

based on the information available to me as a result of research done in the interim. Some of the issues raised were speculative and others were founded on incomplete or incorrect information. Of the remaining issues, I found either that they have already resolved or that work toward proper resolution is underway in conjunction with licensing activities as required by NRC.

Comments on the specific issues raised by the Burns & Roe memorandum are as follows: I refer now to numbers in the original memorandum, in the summary section, page 2, item 5, and also page 8, item 5. The issue here is the suitability of the site and the associated costs of site development.

The plant site was selected following consideration of several possible alternative sites. In late 1971, the AEC appointed a Senior Utility Steering Committee and Senior Utility Technical Advisory Panel to assist them in selecting a utility partner to design, build and operate the demonstration plant. Proposals were submitted to the Steering Committee and AEC by groups of utilities interested in participating in the demonstration plant program.

There were in fact three sites that were considered. The Steering Committee found that the proposal from Commonwealth Edison and the Tennessee Valley Authority offered increased siting flexibility over the other proposals. This was the proposal that finally was accepted by the Steering Committee, and following that, by the AEC. I will not go over the details of the site comparisons that were made.

The soundness of that original decision was supported by the comprehensive and detailed site investigation program conducted during 1973, subsequent to the Burns & Roe memorandum. In contrast to the Burns & Roe apprehension, the site was actually found to be similar to others utilized for nuclear powerplants in the region and was demonstrated to be fully acceptable from all standpoints.

The Nuclear Regulatory Commission also confirmed the acceptability of this site based on their independent review and assessment as documented in the final environmental statement for the CRBRP issued in February 1977, and the site suitability report issued in March 1977. In the site suitability report, NRC concluded that the foundation conditions were generally good and there were no subsurface conditions expected which would preclude the suitability of the site or the construction of the proposed plant.

As the nuclear powerplant siting criteria have undergone very substantial evolution over the past several years, the continued acceptability of this site further reinforces the soundness of its selection.

With regard to the cost of preparing this site, any additional costs incurred for preparation of this site compared to a hypothetical "optimum" site will be small when considered in the context of the many other factors influencing site selection. For example, the cost of highways that are necessary to transport equipment, can be a major variable in the cost of site preparation and this could vary considerably from site to site.

I refer now to the issue of compliance with licensing requirements. The statement in the Burns & Roe memorandum, page 8, item 5, page 9, paragraph 1 and page 9, paragraph 3, concerning compliance with 10 CFR 50 requirements appear to be in direct conflict with the requirements established by the AEC for this project in material submitted to the Congress prior to authorization.

In the original program justification data arrangement for this project submitted to the JCAE on August 11, 1972, it was clearly stated that "all applicable laws and regulations, including those pertaining to AEC licensing and regulations, will be complied with."

This same requirement, updated to reflect the establishment of the independent NRC, is in the Revised Program Justification Data Arrangement No. 77-106, which covers the project at this time.

As to my comment on it, the minutes of the Project Steering Committee have been reviewed and no record was found to support the statement made by Burns & Roe concerning compliance with 10 CFR 50 requirements.

Senator BUMPERS. Did you talk to the man who wrote the memo?

Mr. BECKJORD. I have not had detailed conversations with Mr. Young concerning the memo. I concluded that that was not proper in view of this hearing to be held.

Senator HART. But you have had some talks with him, or some contact with him?

Mr. BECKJORD. Oh, yes, I have had contacts with Mr. Young because he is responsible for the project for Burns & Roe. I mean with regard to this specific memo, I have not had detailed discuscions with him.

Senator HART. But you have had some discussions with him?

Mr. BECKJORD. I have had some discussions with him.

Senator HART. About the memo?

Mr. BECKJORD. The discussion concerned whether we wished to see the testimony which he planned to give. He indicated he would send a copy of the testimony.

Senator HART. But you didn't discuss the substance of the memo?

Mr. BECKJORD. Beyond a few comments, there was no detailed discussion of the substance of his testimony or the memo.

Senator HART. What was the nature of his comments?

Mr. BECKJORD. It concerned this passage regarding compliance with 10 CFR 50 requirements.

Senator HART. What was the nature of that discussion?

Mr. BECKJORD. I asked for clarification as to what was intended. He indicated that the clarification would be in his testimony.

Senator HART. He didn't go into it at that point?

Mr. BECKJORD. No.

Senator BUMPERS. Mr. Chairman, I don't want to interrupt his testimony further than necessary at this point, but I think this is very crucial. You say that—this is one of the most critical parts of the memo as far as I am concerned. You say you have talked to Messrs. Milton Shaw, Thomas Nemzek, Mr. Wagner of TVA, Mr. Wallace Behnke of Commonwealth Edison, Messrs. John Taylor and George Hardigg of Westinghouse, and each of them has assured you there was never either a policy or a practice of avoiding compliance with the AEC Division of Regulation licensing requirements. My question is did you ask him where he got that information and whether he based that information on the memo?

Mr. BECKJORD. Did I talk to Mr. Young?

6277

Senator BUMPERS. You were saying here you have talked to everybody who might have told Burns & Roe that they would not have to comply with some of these basic safety requirements and that all of them say, you say each of them assured you, there never was a policy or a practice of avoiding compliance with the AEC Division of Regulation licensing requirements.

If you talked to the writer of the memo, did you ask him what caused him to put that in the memo?

Mr. BECKJORD. I did not ask him that question, Senator.

Senator BUMPERS. Thank you, Mr. Chairman.

Senator HART. Proceed.

Mr. BECKJORD. I will read this in its entirety.

The minutes of the Project Steering Committee have been reviewed and no record was found to support the statement made by Burns and Roe concerning compliance with 10 CFR 50 requirements. In addition, I have personally called a number of men who were leaders in the early days of the project. These are Messrs. Milton Shaw and Thomas Nemzek, former Directors of ERDA's Reactor Development Division; Mr. Wagner of TVA, Mr. Wallace Behnke of Commonwealth Edison, Messrs. John Taylor and George Hardigg of Westinghouse.

Each of them assured me there was never either a policy or a practice of avoiding compliance with the AEC Division of Regulation licensing requirements. It was, in fact, the policy to go through the entire safety and licensing process as part of the project objectives.

It was understood by the project leaders that modifications to some of the 10 CFR 50 general design criteria would need to be developed, simply because of the technical differences between light water reactors, for which the general design criteria were originally written, and the Clinch River breeder reactor, for which general design criteria were not yet written in 1973.

These modifications were developed within the licensing process and are consistent with the evolution of the licensing process for LMFBR's. It should be noted that much work and discussion was required to resolve the differences of technical opinion prior to the final issuance of CRBRP general design criteria by NRC on January 9, 1976.

The fact that there were significant differences of technical opinion during this effort, however, does not lead to the conclusion that the project was trying to avoid compliance with safety requirements. The safety requirements were properly established when NRC issued, and the project accepted, these criteria.

The objective of the design criteria and the net effect of the CRBRP licensing process is to make the CRBRP at least as safe as a light water reactor located at the same site. To suggest, as the Burns and Roe memorandum does, that there was an intent not to comply with licensing requirements or that the AEC desired to avoid including needed safety features because of cost considerations is simply not supported by the facts.

I can further testify that during my association with the project the policy has been, is now, and will continue to be, to comply with the Nuclear Regulatory Commission's licensing requirements.

The three level defense-in-depth safety philosophy currently being used for design of LWR's was also adopted for CRBRP. This r quires design measures to prevent accidents, to provide protection against either anticipated or unlikely faults that might occur, and beyond this, to provide appropriate engineered safety features in the design to safely accommodate extremely unlikely faults, if they somehow should occur, in order to protect the health and safety of the public.

Furthermore, ERDA and NRC have agreed that, for the CRBRP, it is prudent to include additional measures in design to further limit potential consequences to the health and safety of the public. Accordingly, the project has included margins, beyond the necessary design basis, in order to reduce the postulated consequences of hypothetical accidents involving core meltdown and energetic disassembly.

At the time of the Burns and Roe memorandum, there were ongoing discussions between RRD and DRL concerning whether hypothetical core disruptive accidents HCDA should be included in the design basis for LMFBR's. The resolution with DRL was that, to avoid schedule delay, two CRBRP designs would be submitted for concurrent review, one without and one with HCDA's in the design basis, the reference design and a parallel design.

In a May 1976 letter, the NRC agreed that HCDA's can and should be excluded from the design basis. Subsequently, the project withdrew the parallel design from further consideration by NRC, but it was mutually agreed that margins would be provided in the plant design in order to reduce the postulated consequences of such hypothetical accidents so that the CRBRP would be comparable to current LWR's.

Senator HART. Let me run through that in English so I understand what that means. It seems to me what happened here was in the discussions within ERDA and with the contractors, that it was decided, for purposes of determining the safety of the project, that there would be two hypotheticals, or there would be two critical paths followed, one which included the so-called hypothetical core disruptive accidents which, I assume, are core meltdowns and things of that sort, and one which did not.

Because in an effort to avoid what are called schedule delays—you took the two path method to avoid the delays. Later on in an agreement by May 1976 letters, the NRC agreed, I don't know with whom, that the path including the hypothetical core disruptive accidents, which is an interesting phrase in itself, would be excluded from what is called the design basis; presumably the basis upon which the decision to go forward would be made.

So the project withdrew the so-called parallel design, including the hypothetical core disruptive accident presumptions. Then you proceed, it was mutually agreed that margins would be provided in the design to reduce the postulated consequence of such hypothetical accidents.

What does that mean?

Mr. BECKJORD. It means this, Mr. Chairman, and I will try to explain it in English. A design basis accident is an accident which is assumed to happen and the course of the accident is evaluated. By the definition of design basis accident, what is meant is the particular part of the system or the plant in its entirety has to accommodate the consequences of that accident and control them with no adverse consequences within design limits.

What that means, design limit for example, I will attempt to explain in a simplified manner. In the case of a metal bar, let us say, if we were to accommodate an accident within the design limit for a metal bar, after the accident had occurred, there would be no deformation of the bar because design limit requires the design stresses not be exceeded. So the bar might deform, but when the event was over, the bar would be elastic and it would return to its initial condition.

I can consider accidents which go beyond the design basis. In that case, I will evaluate that accident. If such an accident occurred that did go beyond the design basis, the bar might be deformed so that it would not return to its initial condition. That doesn't mean that anything has happened. It does not mean that there are adverse consequences. It simply means that the system is capable of accommodating one time occurrences far beyond, in many cases, the limit of the design.

So the question here is whether this HCDA should be a design basis accident and the entire system should accommodate it within design margins or whether going beyond design limits should be permitted with the accident contained in other ways. That is a one time event.

I believe that correctly summarizes the differences between a design basis accident and an accident which goes beyond a design basis accident.

Senator HART. A hypothetical core disruptive accident, that means a core meltdown?

Mr. BECKJORD. It means a core meltdown or it could mean a core disassembly through some means of sudden energy release which would cause it to disperse.

Senator HART. Is it safe to say that that phrase includes the worst possible things that could happen?

Mr. BECKJORD. I believe it does.

Senator HART. So in this case, in order to save time, it seems to me two decisions were made, that you would go on two paths in your planning, one which included these most serious accidents for design purposes and one which did not. That went on for awhile, it is a little unclear for how long.

Mr. BECKJORD. I think it was midsummer of 1974 until the May 6 letter, of 1976.

Senator HART. So almost 2 years, you went on a two path basis, one with the accidents, one without.

Mr. BECKJORD. Yes.

Senator HART. Then NRC agreed, with whom and for what reason it is unclear, that the planning path that included the most serious accidents can and should be excluded. Then the project withdrew the path with the most critical accidents included from further consideration by the NRC, but it was mutually agreed that margins would be provided.

What does that mean, margins would be provided?

Mr. BECKJORD. As when I was trying to come up with a simple example, margins would be provided to accommodate the consequences of a hypothetical core disruptive accident, but not as a matter of a design basis—that is, not to say with the bar returning to its initial original position.

6278

Specifically, these margins have to do with the structure of the plant and the containment, the heat capacity in the base of the floor under the reactor, and in the ability for heat removal from the containment so that if this accident still occurred, it would reach an equilibrium point.

It is a very low probability accident, but nonetheless, these design margins would make it possible to control the accident.

Senator HART. There are two questions that come to mind. First of all, who defines the margins, specifically and, second, why not, for purposes of public safety, accept the design course that included the most serious accidents? Why go on the two-path method in the first place? Why not take the worst case basis for a design study?

Mr. CASE. Senator Hart, may I respond to that?

First, one should understand there are two aspects to this hypothetical core disruptive accident. First, you do everything you reasonably can do to prevent the accident. There has been no change with the NRC requirements with regard to that. In other words, we still require all the features necessary to prevent the occurrence of such an accident.

The other side of the coin is to assume, nevertheless, having all the features, the accident occurs anyway for some hypothetical reason. On that score, we took a course of action in between the so-called reference design and parallel design requiring the plant be designed to accommodate some of the effects of this accident, but not all of them. In other words, we continue to require that the containment system maintain its integrity for at least 24 hours following the occurrence of this same hypothetical core-disruptive accident.

The reason we don't require all of the other features of plant beyond that time is simply that we don't think it is necessary from a safety standpoint in view of the very low probability of the occurrence in the first place due to the features required to prevent the accident occurrence.

Senator HART. On the breeder reactor program, are the same standards used for the light water reactors in this regard, in both regards?

Mr. CASE. The standards here are more severe than those for the light water reactors. For the light water reactors, we require all the features to prevent such accidents. We do not require it to accommodate the accident, in the event it should occur. In this plant, we do.

Senator MCCLURE. Might I just ask this question? Some reference has been made to, in order to save time or to avoid schedule delay, that refers, if I understand it correctly, that refers only to the parallel design feature for a period of time and not to the ultimate decision. Am I correct?

Mr. CASE. That is not correct. That factor did not enter into our decision. Our decision was strictly based on our judgment that the risks of this reactor should be comparable to light water reactors.

Senator MCCLURE. As a matter of fact, you required a design beyond that required of the light water reactor?

Mr. CASE. Yes, but I want to make the record clear that light water reactors have some inherent features that this plant does not. In requiring features on this plant, our objective was to make the risks comparable.

Senator MCCLURE. So that there was no element of saving time or meeting a schedule that was motivating in your decision not to require the most severe accident containment?

Mr. CASE. That is correct, Senator.

Senator BUMPERS. May I ask one question? What is an energetic disassembly? Is that an explosion?

Mr. CASE. In layman's terms, it would be called an explosion. Yes, sir.

Senator MCCLURE. It sounded like some OSHA language.

Senator HART. Mr. Beckjord, proceed, please.

Mr. BECKJORD. All of the relevant CRBRP safety issues, including those raised by the Burns & Roe memorandum, are being properly and thoroughly analyzed during the course of the licensing process. Most of the issues have been resolved in a manner mutually acceptable to ERDA and NRC. Work is continuing on the remainder of these issues at this time. No unusually difficult problems in design have been identified.

To date, the project has made design changes estimated to ultimately cost $60 million in order to meet additional licensing requirements which have evolved during the interactions with NRC, and it is possible that other changes may yet be required. You may be assured, however, that we have always been, and are at present, dedicated to meeting all necessary licensing requirements.

Referring again to the Burns & Roe memorandum, on page 14, item 5, and on page 17, item C, there is an issue raised regarding project requests for special licensing variances. The CRBRP project has asked for no special licensing variances.

Consistent with one of the major CRBRP project objectives of demonstrating the licensability of the LMFBR concept, the CRBRP is being subjected to the identical licensing process by the NRC as would any commercial nuclear powerplant.

At the time of the Burns & Roe memorandum, the project was expecting to request an exemption to conduct certain site preparation activities prior to receipt of a construction permit, as was permitted by the AEC regulations under 10 CFR 50.12(b).

However, that procedure was changed, I believe, in anticipation of the establishment of NRC. That attempt was dropped and instead we began to pursue the limited work authorization, which is the end of the process required under the NEPA Act of 1970. The limited work authorization would permit us to begin site preparation activities.

When the CRBRP environmental hearing activity was suspended in April of this year, we were in the process of pursuing the request for a limited work authorization. My point is that, with the exception I have just expressed, there have been no requests for special variances.

Regarding other NRC requirements, the project will meet all of the applicable requirements. However, as already stated, some of the NRC requirements were formulated for LWR's and have either no applicability or only partial applicability to the CRBRP. In these cases, the project will meet the intent of the LWR requirements by developing modified or new requirements in cooperation with NRC; that is, 27 of the 56 general design criteria were modified, plutonium dose guidelines were developed, and new containment criteria were developed.

Referring again to the Burns & Roe memorandum, on page 17, item D, on page 17, item 7, on page 18, items A through D, and on page 18, the first full paragraph, the technical suitability of the plantsite is discussed. I won't read through all of those points.

These apprehensions of Burns & Roe about the site were based on 24 core borings at the proposed site, of which only 4 were in the immediate vicinity of the plant location. After a comprehensive and detailed site investigation program, the final plant location at the Clinch River site was proven to be sound.

This site investigation program included over 100 additional core borings, a test grouting program to confirm the homogeneity of the foundation stratum, detailed geophysical studies, and other extensive analyses and tests. All the points raised by the Burns & Roe memorandum were fully and thoroughly reviewed with NRC prior to their issuance of the final environmental statement and the site suitability report for the CRBRP. The NRC staff concluded that the foundation conditions are good and that the site is suitable for construction of the plant.

Referring to the Burns & Roe memorandum on page 22, item F, the issues presented are safety approaches and plant licensability. This comment on the licensing process was made at an early point in the plant design.

As has already been explained, one of the key objectives of this project has been to license this plant in the same manner as a commercial LWR plant. Many of the specific approaches and features which were ultimately incorporated into the design required extensive study, analysis, and development.

The problems identified in the Burns & Roe memorandum have each been addressed in the licensing process as the design has evolved. Either they have been resolved or appropriate work is underway to resolve them.

In conclusion, I wish to emphasize the following points: the goal of the CRBRP design has been to provide a plant which is at least as safe as an LWR located at the same site. Since the commencement of the project, it has been the policy to go through the entire licensing process and to comply with licensing requirements established by the AEC Division of Regulation and its heir, the Nuclear Regulatory Commission. All NRC licensing requirements are being fulfilled in the project implementation.

The internal Burns & Roe memorandum is over 4 years old. Some of the issues raised in it were speculative, and we have not found a basis for them. The remaining issues have each been properly addressed in our detailed design and site investigations and with the NRC in the licensing process. Each issue has been fully and completely resolved or appropriate work toward resolution is currently proceeding.

NRC has agreed that the comprehensive site investigation program has established that the site meets NRC requirements. Good progress was made by the project in the licensing area during the past year until the suspension of licensing hearings in April.

That concludes my statement, Mr. Chairman. I would be glad to answer any other questions that you have.

Senator HART. Thank you, Mr. Beckjord.

I think we will direct questions to both Mr. Gossick and Mr. Case as well as the earlier witnesses. One thing that concerns me, Mr. Gossick, about your statement is the tone and passive character of some of the sentences where you talk about the site and the project as being not inconsistent with NRC objectives and standards.

At very few points in your statement do you go out of your way to give extraordinary assurances to us for the American public about this project. For example, you say the staff review has been aimed at assuring that these concerns were resolved in a manner consistent with a safe facility design and operation.

That is a very carefully worded statement. You use words like "aimed at" and "matters consistent with." In matters of this sort, what the American people want, at least what I want, is something a little more than that, how safe these facilities are and that the way that the Clinch River project has been going is not inconsistent with other projects and things of that sort.

There is lacking, I think, a kind of positive note in your testimony that I think we would like. Is that a problem for you?

Mr. GOSSICK. Sir, I think I must address the point with regard to the site as we have concluded in my statement. We are convinced that the site is a satisfactory site. We have not finished the safety review, the other part of the review of the CRBR, Mr. Chairman. It is still undergoing staff review. It is a process not yet complete.

Therefore, we must not speculate about the outcome of that until the hearing is finished on the safety aspects and the staff action is completed. All I can say is it is going along as any other application, recognizing it is a first of its kind.

So there is no intent to indicate either pessimism or, for that matter, no particular grounds that I can cite at this point for saying that we are convinced that it will be a safe design. We have just not completed that process.

Senator HART. The tentative nature of your statement is attributable more to the fact that you are still in the process and not that you have lingering hesitancy itself.

Mr. GOSSICK. That is correct.

Senator HART. Specifically, in your testimony in connection with the Burns & Roe statement about it being one of the worst sites ever selected, you have the following language, "Reduction in accident risks achievable with remote location"—talking about the staff balance—"against the resulting costs and inability of the demonstration plant to accomplish its goals on a time frame compatible with the present timing goals of the LMFBR program."

What that says to me is there is a balancing of risk against cost and time. You resolve it slightly, at least in the direction of cost and time. If I am wrong, correct me. I want to quote in that connection the context from which that assurance came or that statement came, the final environmental statement dated February 1977 has the following sentence in it, or paragraph, that I will extract:

> Another measure of the relative differences among the sites was obtained by [illegible] in terms of overall population exposure and

to 50 miles. The radiological dosage at the alternative sites would be roughly a factor of 10 less than the Clinch River site by this measure.

I think the question is in this balancing: How much does the risk go up in order to keep cost and time down?

Mr. GOSSICK. Sir, I would like to ask Mr. Case to address the details of this. I would say at the outset, however, as I have already indicated, that the objectives of the CRBR program within the context of the overall LMFBR program and the ability to meet those objectives on a certain time scale have been stated by ERDA as required, and have been discussed with the Congress and the administration, and were taken into account in that balancing process.

Specifically how that was treated, I would like to ask Mr. Case to address.

Mr. CASE. First, risk is a product of probability times consequences. Your question really was what is——

Senator HART. Say that again.

Mr. CASE. Risk is a product of probability times consequences—the probability of an accident times consequences of an accident. The factor of 10 which you mentioned, which comes from our environmental statement, deals only with the consequence side.

It is indeed true, taking into account the population distribution at the alternative sites considered, that the consequences, should a serious accident occur, would be 10 times higher at the Clinch River site as compared to these alternative sites.

Senator HART. Because of population density?

Mr. CASE. Primarily because of population density.

Senator HART. I think there is a quarter of a million people living within 50 miles of the Clinch River site.

Mr. CASE. Yes; the element we must also consider is the probability of the accident. In both cases, due to the design requirements, the probability will be very low.

However, the consequences would be 10 times higher, although this is a 10 times change in a small risk. That is the point which had to be balanced against, in accordance with the Commission's decision, the effect of moving from the Clinch River site to these alternative sites, the effect on the timing goals set forth by the ERDA Administrator in his programmatic statement, since there would be some delay involved going from one site to the other.

Taking that timing into account, it was our view that you could not meet the programmatic goals as set forth by the ERDA Administrator.

Senator HART. What about the degrees of probability among the various sites?

Mr. CASE. Essentially, no difference at all.

Senator HART. Probability remains constant, consequences increase by virtue of staying at Clinch River?

Mr. CASE. Yes.

Senator HART. You mentioned, Mr. Gossick, the need for finding where "solution cavities exist" at the site. Can you assure the committee that this will take place or has taken place, talking about the site questions?

Mr. GOSSICK. With regard to the possible cavities?

Senator HART. Yes.

Mr. GOSSICK. Yes, sir. Certainly, that will be under continuing review and scrutiny by the NRC. It would continue if the project continues and, certainly, I assure the committee that will be looked at very carefully.

Senator BUMPERS. Would the chairman yield at this point?

What is pressure grouting?

Mr. GOSSICK. Sir, it is an injection, as I understand it, of cement or concrete into the subsurface, into the areas where it is suspected or known that there are cavities that have been formed by erosion.

Senator BUMPERS. I believe it was in your testimony that you said that would be a possible suitable solution to solving the cavities problems?

Mr. GOSSICK. Yes, sir. It is a common technique. As I understand it, many of the dams in the Tennessee area, one in particular I am familiar with, have used that technique.

Senator BUMPERS. The one I am familiar with is the Teton Dam. They used that technique there.

Mr. GOSSICK. I am not familiar with that, but that is putting it into the rocks. I think that was dealing with an earth dam. We are talking here about rock.

Senator BUMPERS. Are you not aware of the fact that that is precisely what caused the Teton Dam to——

Senator MCCLURE. I would say to the Senator that is not what caused the Teton Dam failing. The pressure grouting worked. They didn't do some other things that should have been done.

Senator HART. Rather than debate the Teton Dam, Mr. Case, I think you referred to the atmosphere in connection with consequences. Do I understand that among the alternative sites that the atmospheric conditions at Clinch River are such that any escaping radioactivity would remain in the area longer than the alternative sites?

Mr. CASE. The diffusion conditions are worse at the Clinch River site as compared to the alternative sites, so, the answer is yes.

Senator HART. On the question of containment and the consequences of core meltdown, since that has come up, we will quantify that, if we may. If you could, describe very briefly how such an incident would occur, or accident. It is my understanding what happens is the core eats its way down through the containment, possibly, and would potentially release large amounts of radioactive materials.

Second, in view of the seriousness of those consequences, what is the justification for excluding the so-called CDA from the required design criteria?

Mr. CASE. Yes, to your first question, a possible way of violating containment integrity following an extensive core meltdown would be for the core to melt down through the concrete and then violate integrity by moving into the ground.

An important consideration before that sequence of events is another possible method of losing containment integrity. That would be to literally blow the containment up due to overpressurization during a much shorter period of time. That is our principal concern with regard to the Clinch River reactor.

Our requirements are to avoid loss of containment integrity during the first 24 hours due to overpressurization, admitting the possibility,

6281

as is true in light water reactors, that you might lose containment integrity after that time through this meltdown process which you have described. The advantage of maintaining the integrity through the 24-hour period is to reduce the potential consequences of accidents due to radioactive decay during the 24-hour period.

The basis for accepting the small risk of the loss of containment integrity due to the meltdown phenomenon is the low probability that we believe of such an accident due to other design provisions.

Senator HART. Does the Clinch River design include a so-called core catcher?

Mr. CASE. The specific method by which they would assure this requirement of 24-hour containment integrity, I don't believe the project has figured it out yet, nor submitted it for our review.

Senator HART. It hasn't included or excluded it?

Mr. CASE. Right.

Senator HART. The French and British do include that feature?

Mr. CASE. Yes.

Senator DOMENICI. Did you say it had to be a core catcher?

Mr. CASE. The method used to satisfy this requirement has not been proposed by the applicant.

Senator DOMENICI. Thank you.

Senator HART. Senator McClure?

Senator MCCLURE. Thank you, Mr. Chairman.

Can you gentlemen tell us how long we have had liquid metal fast breeder reactors in operation?

Mr. BECKJORD. Since 1951.

Senator MCCLURE. EBR-1 went operational in 1951 and EBR-2 in 1963. There are others in the world besides those two experimental breeder reactors in the United States, is that correct?

Mr. BECKJORD. There are, I believe, eight that have been placed into operation, Senator, in the world.

Senator MCCLURE. Some of the design criteria in Clinch River are not necessarily just dreamed up out of engineers' dreams? They are based upon some experience with a breeder reactor of this kind?

Mr. CASE. Yes, sir.

Senator MCCLURE. The difference between this and those experimental breeder reactors is that of scale and the problems on scaling up to a demonstration plant and applying new techniques learned during the experimental breeder reactor operation. Is that correct?

Mr. CASE. Yes, sir.

Senator MCCLURE. Mr. Gossick, in your statement, you say, "Informational deficiencies were identified by the staff in a letter of November 1, 1974."

Have you compared those informational deficiencies with the allegations of the Burns & Roe memorandum?

Mr. GOSSICK. Let me check with the staff.

Senator MCCLURE. I see a number of heads shaking behind you.

Mr. GOSSICK. I am advised that some of the questions were involved and relate to the matters we have discussed here this morning that are in the memorandum.

I will ask Mr. Case to elaborate, but this is a normal part of our licensing process where the application is received and there is needed

6282

information missing or information that needs to be clarified for the purposes of our staff review.

Mr. CASE. This is the usual case for us to find information deficiencies in a tendered application and to require that the deficiencies be remedied in the application to be docketed for review. There is nothing unusual in this case.

Senator MCCLURE. What I am interested in is whether or not the information which was in the Burns & Roe memorandum was by one means or another made known to or made a concern of the NRC.

Mr. CASE. The concerns with regard to grouting, solution cavities, were made known to the NRC, and were followed up in our review. The concerns relating to the physical characteristics of the site were made known to us, yes.

Senator MCCLURE. Even though the memorandum was not furnished to you and you didn't know of it until 2 weeks ago?

Mr. CASE. That is correct.

Senator MCCLURE. Nevertheless, the design criteria or the site selection problems that were outlined in that memorandum were either known to you or discussed by you over the period of the last 4 years?

Mr. CASE. Yes, sir.

Senator MCCLURE. I guess the bottom line would be, is there anything in the Burns & Roe memorandum which would change the NRC position on the site?

Mr. CASE. No, sir.

Senator MCCLURE. Mr. Beckjord, you were asked the question if you had discussed with Mr. Young the background of his assertions.

You said you had not discussed that with Mr. Young. In spite of the fact that you have not discussed it with him directly since the memorandum was called to your attention, do you have any knowledge of the background for his assertion?

Mr. BECKJORD. No, sir, I do not.

Senator MCCLURE. I suppose one thing that would concern me is the complexity of management of a plant of this kind, particularly with the way in which it was originally conceived.

As I understand it, and correct me if I am wrong, Consolidated Edison and TVA were copartners with ERDA in the development of this plant originally.

Mr. BECKJORD. Commonwealth Edison.

Senator MCCLURE. Excuse me, Commonwealth Edison and TVA. They were the essential prime participants in the Project Management Corp., that, as the cost overruns began to mount and the cost of the project and the delay of the project increased, in May of 1976, ERDA took over the management of the project, is that correct?

Mr. BECKJORD. That is correct, Senator.

Senator MCCLURE. ERDA has primary responsibility now, although Commonwealth Edison and TVA are still involved in supervision of the project?

Mr. BECKJORD. Yes, sir.

Senator MCCLURE. Do you see any difference in the difficulty of overseeing the project from ERDA's standpoint? Was there greater difficulty prior to May of 1976 than there is at the present time?

Mr. BECKJORD. There was greater difficulty prior to May 1976.

Senator McClure. Simply because there were more cooks stirring the broth?

Mr. Beckjord. Yes, sir, at this point ERDA is solely responsible for the project and ERDA can act. There were possible situations before the change in May of last year where the activity could have become deadlocked because of disagreement.

If a disagreement had occurred among the principals, activity could have been brought to a stop. But that can't happen now.

Senator McClure. Mr. Gossick, could you comment on the same question, from the NRC standpoint?

Mr. Gossick. Senator McClure, from the information that we have, our staff does not consider that the Project Management constitutes a safety issue as far as the difficulty in managing the program is concerned. We consider that purely ERDA's concern.

Senator McClure. Again, the bottom line, I assume, from the standpoint of the hearing today is that there is nothing in the Burns & Roe memorandum of 1973 which you have not dealt with or are not dealing with currently?

Mr. Gossick. That is correct, sir.

Mr. Case. Restricting it to those things that affect safety. There are a number of aspects that don't affect safety that we didn't even follow.

Senator McClure. They would not be your responsibility?

Mr. Case. That is correct.

Senator McClure. Might I address the same question to ERDA. There are some aspects that are not simply from the standpoint of safety, that NRC would not be involved with, that ERDA might be concerned with.

ERDA has dealt with or is dealing with all of the items that are listed in the Burns & Roe memorandum of 1973?

Mr. Beckjord. I would ask Mr. Caffey to comment on that.

Mr. Caffey. I would say, Senator McClure, that all aspects and apprehensions and concerns listed in the Burns & Roe memorandum which affect the project, this is aside from business matters of Burns & Roe, have been adequately dealt with except for those individual items of safety issues which we are still interfacing with NRC about.

All of the management aspects have been adequately dealt with.

Senator Hart. Senator Domenici?

Senator Domenici. Thank you, Mr. Chairman. Just a few questions.

Mr. Case, with reference to your statement defining risk as probability times consequences. Could you enlighten me with some specifics?

What kind of probability are you talking about in the two areas that have been discussed here today?

Mr. Case. The probability that we are talking about, in our judgment, for a core disruptive accident is about 1 in 1 million or less per reactor year. In other words, the probability of such an accident, we believe, is less than one in a million per reactor per year.

Senator Domenici. You would be multiplying that times consequences of various alternative sites to arrive at your risk?

Mr. Case. Yes, sir.

Senator Domenici. You made the conclusion then that because the probability is so small, when it is multiplied times a higher consequence, the risk is not increased that much in terms of other considerations. Is that correct?

Mr. Case. That is correct. The risk is acceptable in either location. There is less risk at these alternative sites. But taking that smaller risk must be balanced against meeting the program objectives.

Senator Domenici. Just one last summary question for myself. I have been through the Clinch River project in the Energy Committee as a new member for a couple of months. In the process, I find we have been on this project for years with all kinds of differing scientific positions.

There have been scientists on both sides of this issue from its inception. There have been energy people on each side of this issue.

Is there anything about the internal memorandum which you now have in your possession which in any way changes your decisions to this point in time about its value?

Mr. Gossick. There is not, sir.

Senator Domenici. How about ERDA?

Mr. Beckjord. None, sir.

Senator Domenici. If you had known about the memorandum 6 months after it was written, can you tell us that nothing would have changed with reference to the way you have proceeded with this project?

Mr. Beckjord. There might have been a lot of activity when we discovered it as there has been over the past 2 weeks, Senator. I think that——

Senator Domenici. Would we be where we are today with this project, with the same requirements imposed at this point and the same licensing procedure?

Mr. Beckjord. That is a fair statement, Senator. I believe so.

Senator Domenici. How about the NRC?

Mr. Gossick. I would concur in that. The matters in the memorandum that deal with the site have been brought out. So, there is nothing that would change matters as far as I can see.

Senator Domenici. Has there been a recent comparison of the three sites from the point of view of the allegations in the internal memorandum? Do we have that kind of evaluation somewhere in the record of the Federal Government?

Mr. Beckjord. I guess I would refer to the report, the final environmental statement in which the alternate sites were evaluated. The general considerations were looked at at the alternative sites as well as Clinch River, the difference is that I don't believe extensive new borings were taken at those alternative sites.

If serious consideration were to be given at a future time to a different site, then that is the kind of work that you would do to establish that it is in fact suitable.

General considerations were looked at, at alternative sites, but not the specific structural mechanics of the sites.

Senator Domenici. Is it true that when you did do the specifics on this site, it proved out satisfactory with reference to meeting the necessary safety requirements?

Mr. Beckjord. To the best of my knowledge, that site is wholly acceptable.

Senator Domenici. Thank you, Mr. Chairman.

6283

Senator HART. Senator Bumpers?

Senator BUMPERS. Mr. Chairman, I just have one item I want to pursue at the expense of going over territory we have already covered.

I would like to ask Mr. Beckjord this: the thing that has caused me more concern, I think, over the Burns & Roe memo than anything else is the statement here, for example, where Mr. Young says:

> The overall approach to reactor safety matters has to date been based upon the Fast Flux Test Facility approaches, the policies established by Mr. Shaw in RRD, which are in many ways contrary to those of the AEC Nuclear Commission.

For example, Westinghouse and Burns & Roe have been told orally by RRD and PMC that we should not comply with the requirements of 10 CFR 50. They cite the DRL safety considerations and would not necessarily provide a simple reliable plan. Then he goes ahead to say this is part of the power struggle between the AEC and so on.

In your testimony, Mr. Beckjord, you say you started developing parallel systems; then you say, to cover hypothetical core disruptions, and then you drop that.

> In a May 1976 letter, the NRC agreed that these hypothetical core disruptions can and should be excluded from the design basis. Subsequently, the project withdrew the parallel design from further consideration by NRC, but it was mutually agreed that margins would be provided in the plant in order to reduce the postulated consequences of such hypothetical accidents.

It really seems to me, and I admit that I may be in error and I may be inferring something here that is in error, but it occurs to me that what Mr. Young has been told orally is precisely what happened, that we have cut corners on the safety specifications.

Mr. BECKJORD. Senator, I don't believe that is the case. Let me take your second question first, relating to the HCDA. The question that relates to the HCDA is whether the HCDA is to be accommodated within design basis.

That comes back to a discussion which I was trying to clarify earlier this morning, how a design is accomplished; as to whether the accident is fully contained and controlled within the design limits.

In the case of the HCDA, what has been decided is that the HCDA is not accommodated within design limits; it is accommodated in another way with margins built into the plant design so as to mitigate the consequences of that accident.

Mr. Case was explaining what the rationale for this is, namely, the probability of an HCDA is very low. My figures are somewhat lower than his. I would say that the probability of an HCDA is reckoned to be of the order of 10 to the minus 8 per year or less. So it has a very low probability of occurrence. The question is, what do you do about it.

Senator BUMPERS. You are not suggesting that you are entirely accurate on the probability, are you?

Mr. BECKJORD. No, sir, 10^{-8} or less.

Senator BUMPERS. OK.

Mr. BECKJORD. This accident has been studied extensively. For it to occur—let me just say a little more about it. I know of no mechanistic way that it can happen. It is called hypothetical because for the purposes of analysis and discussion, we assume it can happen, but nobody has come up with a mechanism by which it could logically occur.

6284

As an example, the core of the Clinch River reactor consists of fuel material and it is encased in cladding material and structural material. In order for the worst HCDA to occur, they would have to develop some way in which the cladding material and the structural material would fall away. It might melt, but the fuel would stay in place. I don't know of a way that this can happen, so it is considerations like this.

I am trying to describe it in a very simple fashion which has been studied extensively. It is by reasoning such as this that the probability that it could happen is reduced; and 10^{-8} is a very small number.

What do you do about it? Do you conceive of a design which will accommodate this very unlikely event within design limits or variables or do you find some other way to handle it? The path that has been chosen is to build other margins into the plant.

Senator BUMPERS. Mr. Beckjord, what are the probabilities by ERDA's estimates of an explosion occurring in a breeder reactor plant?

Mr. BECKJORD. That would be the same order, 10^{-8} per reactor year. I might add that one of the margins that is to be included in this plant design is the capability to withstand a very sharp explosion.

The words "energetic disassembly" came up earlier. Maybe that is overly technical, but we have been in discussions with the Nuclear Regulatory Commission on the amount of energy, the amount of explosive force that must be accommodated within the structure. That matter is not settled yet.

Senator BUMPERS. Incidentally, the one that Senator McClure referred to that was put in operation in 1951 did explode, didn't it?

Mr. BECKJORD. No; it did not. That was a meltdown.

Senator HART. I think that was the original question. You say you are using figures of 1 out of 10^{-8}, when in fact six breeders have been developed where two of them have had meltdowns which I understand to be contained in the definition of a core disruptive accident.

When you use the term hypothetical because you can't conceive of it ever happening, it has happened twice, at the Idaho Falls plant and the plant in Detroit. Am I missing something here?

Mr. BECKJORD. Yes. The hypothetical accident we are talking about here is a lot more severe.

Senator HART. Let's talk about one that is not so severe because I understand the definition of hypothetical core disruptive accidents includes core meltdown and it has happened two times out of six.

Mr. BECKJORD. We are talking about a total core meltdown.

Senator HART. Well, let's talk about a little core meltdown.

Mr. BECKJORD. One occurred at the plant in Detroit. Part of the subassembly did melt.

Senator HART. Does HCDA include a little core meltdown?

Mr. BECKJORD. No; that is a big one.

Senator HART. What do you call a little one?

Mr. BECKJORD. A little one is a core melt.

Senator HART. A hypothetical core disruptive accident——

Mr. BECKJORD. That is the big accident.

Senator HART. What is the dividing line between big and little?

Mr. Beckjord. A little one, I would define that as the accident that occurred at the Fermi plant. Part of the assembly melted. The reactor was shut down. It was safely shut down without activity released to the environment or injuries to the public.

Senator Bumpers. It is still shut down, isn't it.

Mr. Beckjord. After that accident, the vessel was opened, the cause of the accident was determined, the deficiencies were corrected, and that plant was placed back into operation. It operated, I don't know, for 2 or 3 years. It was finally shut down based on economic considerations; but the plant did operate again after that accident.

Senator Hart. It seems to me there is little circular reasoning here, it is little if nothing bad happens. If something bad happens, it is big; but a big one can't happen.

Mr. Beckjord. That is certainly not the impression I am trying to convey, Mr. Chairman.

Senator Hart. The Fermi meltdown, little because nothing got away from it, the operator?

Mr. Beckjord. Yes.

Senator McClure. I thought he said the Fermi could be characterized as big.

Mr. Beckjord. No.

Senator Hart. It can't be big because a big one can't happen.

Mr. Beckjord. The Fermi accident occurred. There was a flaw in design. It happened one day that the coolant flow channel was blocked. That is what happened at the Fermi reactor. With no flow permissible in that channel, there was melting. When the assembly was melting, the plant was shut down right away. It was detected.

Senator Hart. What we are trying to get at is what a hypothetical core disruptive accident is.

Mr. Beckjord. A hypothetical core disruptive accident is the worst accident that can be conceived of for this reactor.

Senator Hart. But it can't happen, but it can be conceived of?

Mr. Beckjord. No; it can be conceived of; but what I am saying is that I can't give you a mechanism by which it could happen. In other words, we assume that something like that could happen and we look at the consequences; but I am telling you I don't know how it could happen. I can't come up and give you a sequence of events that will lead to that accident.

It is typical in the accident analysis of nuclear reactors that we don't always go into the mechanism. We assume that the worst possible thing can happen. We try to figure out a way in which it might happen. If we can figure out a way, then we do something about it.

Senator Hart. The key point here is you structure your design studies and analyses by a standard called a hypothetical core disruptive accident, but by your own definition, that is a set of circumstances which cannot occur or which you cannot conceive of occurring?

Mr. Beckjord. No. sir. I don't know of a way it could happen. The studies have shown that the probabilities of it happening are very small. That is what we are saying. However, nonetheless, even though they are very small, there are margins in the design to accommodate such an event and to mitigate its consequences. Those have been required by the Nuclear Regulatory Commission.

6285

Senator Hart. I apologize for interrupting, Senator Bumpers.

Senator Bumpers. I am about finished anyway. The term meltdown could not have occurred if we had used the so-called core catcher technology—I am sorry, the pool technology which the French and British are using?

Mr. Beckjord. Yes, sir. Could you repeat that?

Senator Bumpers. Could the Fermi meltdown have occurred if we were using the so-called liquid sodium pool technology?

Mr. Beckjord. The pool or the loop would make no difference. That it would not have an effect on meltdown—it could happen. If there was the same design defect in the pool system, it could have happened there.

Senator Bumpers. Do you personally feel as far as you know, anybody in the agency feel that the loop method which we are going to use is preferable to the pool techniques?

Mr. Beckjord. Let me give you a short answer on that, Senator. I believe that a safe system can be built using either approach. Each one has advantages and disadvantages. I think that from a safety point of view, they can and will be equivalent.

What we don't really know, what nobody knows is which one is going to be more economical in the end. The French cite important advantages for their system. There are important advantages for ours. One which we think is important and which the Germans also think is important is the ability to inspect the entire system during periods of shutdown. That is not totally possible with the pool system. That is an advantage for the loop type of system.

Senator Bumpers. Have you seen this memo dated June 20, 1976, submitted to ERDA and the Electric Power Research Institute? It has Burns & Roe and Rockwell International at the bottom of that. Have you seen that? It is NRB 76-1. I assume that this is something that came to ERDA from Rockwell and Burns & Roe. Their conclusion is that the pool concept is favored over both the hybrid and the loop designs and they set out numerous reasons why.

Mr. Beckjord. Yes; I am aware. I recall now that report. I think that I will stand on my statement. I think that most of the people in the business in this country will agree that either system can be made, that the two systems can be made equally safe, Senator; but as I say, there is this controversy over which one will ultimately be more economical.

Senator Bumpers. They go ahead to say that the total probability of the core disruptive accident occurring by the pool concept is calculated to be approximately one-fifth and two-fifths that of the loop and hybrid concepts, respectively. That is contrary to what you said a minute ago. These are the people that are building it.

Mr. Beckjord. Can I provide an answer for the record on that point, Senator? I will stand on my statement.

Senator Bumpers. Yes. Of course, we are going to debate this thing this afternoon. If you don't have it to me before 2 o'clock, I will take dramatic liberties with this memo and debate on the floor.

Mr. Beckjord. All right, sir, 2 o'clock.

Senator McClure. Mr. Chairman, I think it might be helpful if we would put in the record a listing of the liquid metal fast breede reactor plants that have been either operated or under design and include a prototype which has been talked about for operation in 1988.

It starts with Clementine, an experimental reactor in 1946 which has been decommissioned; EBR-1 which was operational in 1951 and has been decommissioned. Incidentally, EBR-1 is the plant that first produced commercial electricity. It lighted a small city in Idaho near the test station.

The Fermi plant was decommissioned, but became operational in 1963; EBR-2 in 1963; SEFOR, which was located in Arkansas, started in 1969, has been decommissioned; and the FFTF, Clinch River breeder reactor; if that list might be made part of the record.

Senator HART. Without objection.

[The list follows:]

U.S. LMFBR PLANTS

Name	Type	Location	Power level	Initial Operation	Current status
Clementine	Experimental	Los Alamos	0.25 MWt	1946	Decommissioned.
Experimental Breeder Reactor I (EBR I).	Experimental and demonstration.	National Reactor Testing Station (NRTS), Idaho.	1.2 MWt	1951	Do.
Fermi Atomic Powerplant	Power	Michigan	66 MWe	1963	Do.
Experimental Breeder Reactor II (EBR II).	Testing	NRTS, Idaho	62 MWt (20 MWe)	1963	Operational.
SEFOR	do	Arkansas	20 MWt	1969	Decommissioned.
Fast Flux Test Facility (FFTF)	do	Hanford, Wash	400 MWt	1979	Under construction.
Clinch River Breeder Reactor (CRBR).	Power (demonstration).	Tennessee	350 MWe	1983	Under design.
Prototype Large Breeder Reactor (PLBR).	Power (near commercial).	Undetermined	1000 MWe	1988	Under conceptual design.

Senator HART. Gentlemen, thank you, very much.

Senator DOMENICI. If you will supply that answer that you were going to provide for Senator Bumpers.

Mr. BECKJORD. Yes, sir, before 2 o'clock.

[The information requested by Senator Bumpers and Mr. Beckjord's prepared statement follow:]

HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS FOR LMFBR'S POOL VERSUS LOOP

The risk associated with a postulated HCDA is the product of the consequences (magnitude) and probability of occurrence. The magnitude of such a postulated event is dependent on the core composition and geometry, and therefore consequences are not affected by whether a pool or loop design is assumed. A report dated June 25, 1976, FBR-76-1, by a single contractor team (AI/BRI) concludes that the probability of occurrence of an HCDA may be a factor of five less for a pool type LMFBR than for a comparable loop plant similar to the CRBRP.

Both the loop and pool concepts are safe and either would meet all appropriate safety requirements. The comparisons made by AI/BRI were to consider overall design advantages and not to compare absolute safety.

The AI/BRI conclusion that the pool design has a factor of five lower probability for occurrence of an HCDA than a loop plant is based solely on the larger sodium inventory immediately surrounding the core. For postulated events such as loss of offsite power or large earthquakes coupled with a simultaneous loss of core cooling, they compute that about 14 hours are available for corrective action for the pool plant as opposed to four hours for the loop plant. Either time is sufficient to take corrective action but this difference does affect the probabilities somewhat. However, since they are comparing very small numbers like 1 chance in a million to 1 chance in a billion, uncertainties in the input data are too great to claim factor of five difference between pool and loop designs.

6286

STATEMENT OF ERIC S. BECKJORD, DIRECTOR, DIVISION OF REACTOR DEVELOPMENT AND DEMONSTRATION

Mr. Chairman and Members of the Committee, I appreciate this opportunity to discuss the environmental and safety matters related to the Clinch River Breeder Reactor Plant (CRBRP) Project which were raised in the July 6, 1973, internal Burns and Roe memorandum recently cited in the press.

The CRBRP Project is a joint government-industry cooperative arrangement for demonstrating a Liquid Metal Fast Breeder Reactor power plant as authorized by Congress on June 2, 1970 (Public Law 91-273). The partners in this project are the Energy Research and Development Administration (ERDA), Commonwealth Edison (CE), Tennessee Valley Authority (TVA) and Project Management Corporation (PMC). The objectives of this Project are to design, license, construct, test and operate an LMFBR demonstration plant. In May 1976, ERDA assumed full management control of the Project with continued utility industry support and participation.

I have had the ERDA responsibility for this Project since March 1976. During that time, Project accomplishments have been good, with design now over 40 percent complete, all of the long lead equipment on order, and the Final Environmental Statement and Site Suitability Report issued by the Nuclear Regulatory Commission (NRC). I have examined Project records, reviewed the numerous reports and hearings concerning the Project, and inquired extensively into Project procedures and status, particularly in environmental, safety and related licensing matters. Generally, I can say that the Project has also made good progress in these licensing areas during the past year, working toward its goal of a Limited Work Authorization (LWA) as required under the NEPA Act of 1970, until the recent suspension of the environmental hearings in April. The environmental hearings suspension was requested by ERDA pending a final decision on whether the Project is to be terminated or continued.

I have reviewed the Burns and Roe memorandum in detail since it became available to me about two weeks ago. My statements on it are based on the information available to me as a result of research done in the interim. Some of the issues raised were speculative and others were founded on incomplete or incorrect information. Of the remaining issues, I found either that they have already been resolved or that work toward proper resolution is underway in conjunction with licensing activities as required by NRC.

Comments on the specific issues raised by the Burns and Roe memorandum are as follows:

In the "Summary" section, pages 2 and 3, Burns and Roe stated:

"The site selected is likely to be very costly to prepare and could even be unsuitable.* * *"

The cost of preparing the Clinch River site will have been proven to be substantially more than estimated. The site costs and problems could be such as to indicate a change of site."

The plant site was selected following consideration of several possible alternative sites. In late 1971, the AEC appointed a Senior Utility Steering Committee and Senior Utility Technical Advisory Panel to assist them in selecting a utility partner to design, build and operate the demonstration plant. Proposals were submitted to the Steering Committee and AEC by groups of utilities interested in participating in the demonstration plant program. Each of the principal sites advanced in the proposals received appeared to meet the general requirement that the proposed site should require no unusual design features or special consideration in licensing. The Steering Committee found, however, that the proposal from CE/TVA offered increased siting flexibility over the other proposals. This CE/TVA proposal for building and operating the LMFBR demonstration plant was ultimately accepted by the Steering Committee and the AEC.

Three candidate sites within the TVA area were considered: Widow's Creek John Sevier and Clinch River. Analysis of the relevant siting, environmental and direct cost factors for the three sites disclosed no clear-cut or overriding advantages for any single site. Such differences as existed were considered amenable to treatment in the design within the limits of existing technology. As a practical matter, the three candidate sites were found to be equivalent from site characteristic and environmental standpoints.

Although comparisons of direct site cost slightly favored the Widow's Creek and John Sevier sites because of the availability of some site services, the differences were within the range of uncertainty inherent in such cost estimates

An overall analysis of the three sites, including considerations of meeting project and program objectives, showed Clinch River to have a decisive advantage, because the new site services to be provided at Clinch River would be more compatible with the nuclear steam supply system.

The soundness of that original decision was supported by the comprehensive and detailed site investigation program conducted during 1973, subsequent to the Burns and Roe memorandum. In contrast to the Burns and Roe apprehension, the site was actually found to be similar to others utilized for nuclear power plants in the region and was demonstrated to be fully acceptable from all standpoints. The Nuclear Regulatory Commission also confirmed the acceptability of this site based on their independent review and assessment as documented in the Final Environmental Statement for the CRBRP issued in February 1977 and the Site Suitability Report issued in March 1977. In the Site Suitability Report, NRC concluded that the foundation conditions were generally good and there was no subsurface conditions expected which would preclude the suitability of the site or the construction of the proposed plant. As the nuclear power plant siting criteria have undergone very substantial evolution over the past several years, the continued acceptability of this site further reinforces the soundness of its selection.

With regard to the cost of preparing this site, any additional costs incurred for preparation of this site compared to a hypothetical "optimum" site will be small when considered in the context of the many other factors influencing site selection.

In the "Background" section, pages 8 and 9, the Burns and Roe memorandum states:

"The overall approach to LMFBR reactor safety matters has to date been based on FFTF [Fast Flux Test Facility] approaches and policies established by Mr. Shaw and RRD [Division of Reactor Research and Development] which are in many ways contrary to those of the AEC Division of Regulation (DRL). For example, Westinghouse and Burns and Roe have been told orally by RRD and PMC that we should not comply with the requirements of 10CFR50 Appendix A (General Design Requirements) for LMFBR where such requirements arise from theoretical DRL safety considerations and would not necessarily provide a simple, reliable plant. * * * "This approach is being fostered in full knowledge that it may not result in meeting DRL's licensing requirements and that many issues would have to be taken to the AEC Commissioners for resolution. It is part of a power struggle between parts of the AEC. The LMFBR Demonstration Plant is viewed as a test case in which RRD and PMC can knock out many theoretical safety-oriented design features which complicate commercial plants and make them more expensive, and in which a new approach to safety and licensing can be established. In addition, the Demonstration Plant is viewed as having to be consistent with FFTF in order to justify the approaches on that project. Unfortunately, some safety approaches on FFTF were apparently decided on because of the severe cost bind that project is in. * * *

"A number of existing approaches based on FFTF practices are already known as potential problem areas. These include the lack of specific safety criteria for the project; present emergency core cooling provisions and natural circulation assumptions; the current assumption that a double-ended pipe break is not a credible accident; the assumptions as to the extent of the Hypothetical Core Disruptive Accident (HCDA) and features needed to contain it; the effects of sodium spills and fires; radioactivity release above the operating floor; plutonium leakage and levels at the site boundaries; and the ability to design an effective system to contain a core and reactor vessel meltdown. * * *"

This statement concerning compliance with 10CFR50 requirements appears to be in direct conflict with the requirements established by the AEC for this Project in material submitted to the Congress prior to authorization. In the original Program Justification Data Arrangement for this Project submitted to the JCAE on August 11, 1972, it was clearly stated that:

"All applicable laws and regulations, including those pertaining to AEC licensing and regulations, will be complied with."

This same requirement, updated to reflect the establishment of the independent NRC, is in the Revised Program Justification Data Arrangement No. 77-106 which covers the Project at this time.

The minutes of the Project Steering Committee have been reviewed and no record was found to support the statement made by Burns and Roe concerning compliance with 10CFR50 requirements. In addition, I have personally called a number of men who were leaders in the early days of the Project. These are

Messrs. Milton Shaw and Thomas Nemzek, former directors of ERDA's reactor development division, Mr. Wagner of TVA, Mr. Wallace Behnke of Commonwealth Edison, Messrs. John Taylor and George Hardigg of Westinghouse. Each of them has assured me there was never either a policy or a practice of avoiding compliance with the AEC Division of Regulation licensing requirements. It was in fact the policy to go through the entire safety and licensing process as part of the project objectives. It was understood by the project leaders that modifications to some of the 10CFR50 General Design Criteria would need to be developed, simply because of the technical differences between Light Water Reactors (LWRs), for which the General Design Criteria were originally written, and the Clinch River Breeder Reactor, for which general design criteria were not yet written in 1973. These modifications were developed within the licensing process and are consistent with the evolution of the licensing process for LMFBRs. It should be noted that much work and discussion was required to resolve the differences of technical opinion prior to the final issuance of CRBRP general design criteria by NRC on January 9, 1976. The fact that there were significant differences of technical opinion during this effort, however, does not lead to the conclusion that the Project was trying to avoid compliance with safety requirements. The safety requirements were properly established when NRC issued, and the Project accepted, these criteria.

The objective of the design criteria and the net effect of the CRBRP licensing process is to make the CRBRP at least as safe as a light water reactor located at the same site. To suggest, as the Burns and Roe memorandum does, that there was an intent not to comply with licensing requirements or that the AEC desired to avoid including needed safety features because of cost considerations, is simply not supported by the facts.

I can further testify that during my association with the Project, the policy has been, is now, and will continue to be, to comply with the Nuclear Regulatory Commission's licensing requirements.

The three level defense-in-depth safety philosophy currently being used for design of LWRs was also adopted for CRBRP. This requires design measures to prevent accidents, to provide protection against either anticipated or unlikely faults that might occur, and beyond this to provide appropriate engineered safety features in the design to safely accommodate extremely unlikely faults, if they somehow should occur, in order to protect the health and safety of the public. Furthermore, ERDA and NRC have agreed that, for the CRBRP, it is prudent to include additional measures in design to further limit potential consequences to the health and safety of the public. Accordingly, the Project has included margins beyond the necessary design basis in order to reduce the postulated consequences of hypothetical accidents involving core meltdown and energetic disassembly. At the time of the Burns and Roe memorandum, there were on-going discussions between RRD and DRL concerning whether hypothetical core disruptive accidents (HCDAs) should be included in the design basis (level three) for LMFBRs. The resolution with DRL was that, to avoid schedule delay, two CRBRP designs would be submitted for concurrent review, one without and one with HCDAs in the design basis (the reference design and a parallel design).

In a May 1976 letter, the NRC agreed that HCDAs can and should be excluded from the design basis. Subsequently, the Project withdrew the parallel design from further consideration by NRC, but it was mutually agreed that margins would be provided in the plant design in order to reduce the postulated consequences of such hypothetical accidents so that the CRBRP would be comparable to current LWRs.

All of the relevant CRBRP safety issues, including those raised by the Burns and Roe memorandum, are being properly and thoroughly analyzed during the course of the licensing process. Most of the issues have been resolved in a manner mutually acceptable to ERDA and NRC. Work is continuing on the remainder of these issues at this time. No unusually difficult problems in design have been identified. To date, the Project has made design changes estimated to ultimately cost $60 million in order to meet additional licensing requirements which have evolved during the interactions with NRC, and it is possible that other changes may yet be required. You may be assured, however, that we have always been and are at present, dedicated to meeting all necessary licensing requirements.

In the "Background" section, pages 11 and 17, the Burns and Roe memorandum states:

"The licensing approach involves numerous variance requests and submittals

6287

"It appears likely that the Regulatory group of the AEC will be made independent of the development part of AEC soon. This would mean far less chance of early and unique licensing approvals. * * *"

The CRBRP Project has asked for no special licensing variances. Consistent with one of the major CRBRP Project objectives of demonstrating the licensability of the LMFBR concept, the CRBPR is being subjected to the identical licensing process by the NRC as would any commercial nuclear power plant. At the time of the Burn and Roe memorandum, the Project was expecting to request an exemption to conduct certain site preparation activities prior to receipt of a Construction Permit, as was permitted by the AEC Regulations under 10 CFR 50.12(b). At that time, the AEC was granting exemptions for commercial nuclear power plants under this regulation since this was prior to institution of the use of LWAs. When the regulations were changed to incorporate the LWA procedure, the Project abandoned consideration of an exemption request and oriented licensing activities toward obtaining an LWA.

Regarding other NRC requirements, the Project will meet all of the applicable requirements. However, as already stated, some of the NRC requirements were formulated for LWRs and have either no applicability or only partial applicability to the CRBRP. In these cases, the Project will meet the intent of the LWR requirements by developing modified or new requirements in cooperation with NRC (e.g., 27 of the 56 General Design Criteria were modified, plutonium dose guidelines were developed, and new containment criteria were developed).

In the "Background" section, pages 17 and 18, of the Burns and Roe memorandum, additional statements concerning the site appear:

"The site conditions described below may delay establishment of the suitability of the site.* * *

"The Clinch River site selected for the LMFBR Demonstration Plant is one of the worst sites ever selected for a nuclear power plant based on its topography and rock conditions. The suitability of the site will not be confirmed until after an extensive soil boring program. There is a possibility that the site may not be acceptable. As a minimum, site development costs will be high. The reasons for the above conclusions are as follows:

"(a) The site has varying rock conditions. The rock on which we are attempting to place the plant is known to be somewhat nonhomogeneous and to be subject to possible solution activity problems and perhaps voids and cavities. These conditions may require some rock treatment such as grouting, and verification of the results by an added soil boring program. Previous sites with similar problems have been difficult to license and have been difficult and costly to prepare.

"(b) The areas surrounding the present estimated plant location are known to have an as yet undetermined degree of voids and cavities. Because of this condition and the large amount of excavation required by the design depth of containment at the present time, an extensive rock treatment (grouting) effort appears to be required, followed by a detailed soil boring program to verify that the results are satisfactory. This effort is anticipated to be required to avoid possible severe subsidence problems, which could be the equivalent of a seismic event. The AEC has insisted on such actions for previous sites with less extents of voids and cavities; considerable costs and delays have been involved.

"(c) Slope stability will be a problem during construction due to the nature of the site material.

"(d) Extensive excavation, including much into bedrock, and backfill is presently estimated to be required because of the hilly terrain and subsurface conditions at the site.

"The results of the above could mean a minimum of more than six months' delay and millions of dollars in cost increases. In addition, final location and orientation of the plant will be delayed pending results of the soil boring program. * * *"

These apprehensions of Burns and Roe about the site were based on twenty-four core borings at the proposed site, of which only four were in the immediate vicinity of the plant location. After a comprehensive and detailed site investigation program, the final plant location at the Clinch River site was proven to be sound. This site investigation program included over one hundred additional core borings, a test grouting program to confirm the homogeneity of the foundation stratum, detailed geophysical studies, and other extensive analyses and

tests. All these points raised by the Burns and Roe memorandum were fully and thoroughly reviewed with NRC prior to their issuance of the Final Environmental Statement and the Site Suitability Report for the CRBRP. The NRC staff concluded that the foundation conditions are good and that the site was suitable for construction of the plant.

In the "Background" section, page 22, the Burns and Roe memorandum states:

"Many safety approaches incorporated in FFTF and planned for the LMFBR Demonstration Plant may not be commercially licensable. These plant features could be addressed and resolved during the Demonstration Plant licensing process."

This comment was made at an early point in the plant design. As has already been explained, one of the key objectives of this Project has been to license this plant in the same manner as a commercial LWR plant. Many of the specific approaches and features which were ultimately incorporated into the design required extensive study, analysis and development. The problems identified in the Burns and Roe memorandum have each been addressed in the licensing process as the design has evolved. Either they have been resolved or appropriate work is underway to resolve them.

In conclusion, I wish to emphasize the following points:

The goal of the CRBRP design has been to provide a plant which is at least as safe as an LWR located at the same site.

Since the commencement of the project, it has been the policy to go through the entire licensing process and to comply with licensing requirements established by the AEC Division of Regulation and its heir, the Nuclear Regulatory Commission. All NRC licensing requirements are being fulfilled in the project implementation.

The internal Burns and Roe memorandum is over four years old. Some of the issues raised in it were speculative and we have not found a basis for them. The remaining issues have each been properly addressed in our detailed design and site investigations and with the NRC in the licensing procedures. Each issue has been fully and completely resolved or appropriate work toward resolution is currently proceeding.

NRC has agreed that the comprehensive site investigation program has established that the site meets NRC requirements.

Good progress was made by the Project in the licensing area during the past year until the suspension of licensing hearings in April.

That concludes my statement, Mr. Chairman. I will be glad to answer any additional questions the Committee may have on this subject.

Senator HART. The next witness will be Mr. William Young, vice president of the Breeder Reactor Division of Burns & Roe.

Would you identify for the record, those who are accompanying you?

STATEMENT OF WILLIAM H. YOUNG, VICE PRESIDENT, BREEDER REACTOR DIVISION, BURNS & ROE, INC., ACCOMPANIED BY DR. SEYMOUR BARON, SENIOR CORPORATE VICE PRESIDENT FOR ENGINEERING AND TECHNOLOGY

Mr. YOUNG. Yes, sir. I am William H. Young. This is Dr. Seymour Baron, who is senior corporate vice president for engineering and technology.

I would like to read through my prepared statement which has been submitted along with a number of detailed attachments which I will not read.

On the point that Senator Bumpers just brought up, after my prepared statement, I certainly would be willing to answer questions on that document that he held up. I think it might be quite important.

Senator HART. At an appropriate time. I would encourage you to condense where possible your prepared statement.

6288

MR. SWANSON: The Staff would now like to offer into evidence Exhibit 20. That was Reference 1 to Dr. Cochran's own attachment.

JUDGE MILLER: Oh, yes.

Is there any objection?

MR. EDGAR: No objection.

MS. FINAMORE: No objectior.

JUDGE MILLER: It will be received.

(Staff's Exhibit No. 20 was received in evidence.)

MR. EDGAR: The Applicants would offer Exhibits 53 through 58.

JUDGE MILLER: Any objection?

MR. SWANSON: None.

MS. FINAMORE: No.

JUDGE MILLER: 58? I have --

MR. EDGAR: I'm sorry. 57.

JUDGE MILLER: Through 57 will be admitted.

(Applicants' Exhibits Nos. 53 through 57 were received in evidence.)

JUDGE MILLER: What's next now?

MR. EDGAR: The Applicants call to the witness stand our panel on Contentions 7(a) and (b): Mr. John Longenecker, Narinder Kaushal and Dr. Carl Anderson.

JUDGE MILLER: Will the witnesses come forward, please.

Have any of them been sworn?

MR. EDGAR: None of these witnesses have been sworn, Your Honor.

JUDGE MILLER: Two wish to take the oath; is that correct, or all three?

MR. EDGAR: Dr. Anderson would like the affirmation.

JUDGE MILLER: We'll hold Dr. Anderson for a moment. Will the other two witnesses raise their right hands for the oath, please.

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Whereupon,

JOHN R. LONGENECKER

was called as a witness and, having been duly sworn, was examined and testified as follows:

CARL A. ANDERSON, JR.

was called as a witness and, having duly affirmed his testimony to be the truth, was examined and testified as follows:

NARINDER N. KAUSHAL

was called as a witness and, having been duly sworn, was examined and testified as follows:

MR. EDGAR: I have handed out two things.

First, for the convenience of the Board and the parties, a glossary of terms and acronisms that we have reason to believe might appear during the course the discussion on this piece of testimony.

JUDGE MILLER: All right.

We have that glossary on Contentions 7(a) and 7(b) inserted in the record at this point, prior to the commencement of the testimony.

(Glossary of terms relating to Contentions 7(a) and 7(b) follows on next page.)

GLOSSARY
(Contentions 7a and 7b)

BOP	-	Balance of Plant
EBR-II	-	Experimental Breeder Reactor - II
E Specs	-	Equipment Specifications
ESF	-	Engineered Safety Feature
FEIS	-	Final Environmental Impact Statement
FFTF	-	Fast Flux Test Facility
HCDA	-	Hypothetical Core Disruptive Accident
HTS	-	Heat Transport System
ICD	-	Interface Control Drawing
IHTS	-	Intermediate Heat Transport System
IHX	-	Intermediate Heat Exchanger
LDP	-	Large Developmental Plant
LMFBR	-	Liquid Metal Fast Breeder Reactor
LOF	-	Loss of Flow
LWR	-	Light Water Reactor
MPR	-	Management Policies and Requirements
MWe	-	Megawatt electric
MWt	-	Megawatt thermal
OPDD	-	Overall Plant Design Description
PDP	-	Project Definition Phase
PO Cog-Engineer	-	CRBRP (DOE) Project Office Cognizant Engineer
RAPS	-	Radioactive Argon Processing System
RDT	-	Reactor Development and Technology
RM/A-E	-	Reactor Manufactuer/Architect - engineer
SDD	-	System Design Description
SEFOR	-	Southwest Experimental Fast Oxide Reactor
SGS	-	Steam Generator System
TOP	-	Transient Overpower
ZPPR	-	Zero Power Plutonium Reactor

JUDGE MILLER: When I say inserted in the record, it will also be inserted in the transcript.

MR. EDGAR: Mr. Chairman, I've also handed out a copy to all parties with pen and ink changes reflecting errata of Applicants direct testimony concerning NRDC's Contention 7(a) and 7(b) dated November 1, 1982.

This is the pre-filed written direct testimony marked with errata.

I would request that that be marked for identification as Applicant Exhibit No. 58.

JUDGE MILLER: It may be marked.

(Applicants Exhibit No. 58 was marked for identification.)

DIRECT EXAMINATION

BY MR. EDGAR:

Q. Would each of you please state your name and business address for the record?

BY WITNESS ANDERSON:

A. I'm Carl Anderson. Post Office Box 158, Madison, Pennsylvania, Westinghouse Electric Company.

BY WITNESS LONGENECKER:

A. My name is John Longenecker, U.S. Department of Energy in Washington, D. C.

BY WITNESS KAUSHAL;

A. My name is Narinder Kaushal, Post Office Box U, Clinch River Breeder Reactor Plant Office, Oak Ridge, Tennessee.

BY MR. EDGAR:

Q. Now, would each of you respond to the next series of questions in turn.

Are the opinions and statements in Applicants Exhibit 58 your own?

BY WITNESS ANDERSON:

A. Yes.

BY WITNESS LONGENECKER:

A. Yes.

BY WITNESS KAUSHAL:

A. Yes.

Q. And are the opinions and statements expressed in Applicants Exhibit 58 true and correct to the best of your information and belief?

BY WITNESS ANDERSON:

A. Yes.

BY WITNESS LONGENECKER:

A. Yes.

BY WITNESS KAUSHAL:

A. Yes.

Q And do you adopt Exhibit 58 as your testimony in this proceeding?

BY WITNESS ANDERSON:

A Yes.

BY WITNESS LONGENECKER:

A Yes.

BY WITNESS KAUSHAL:

A Yes.

MR. EDGAR: I would like to make a proffer of the expertise of this panel .

Mr. Longenecker's qualifications appear starting at Page 48 of Applicants Exhibit 58.

Dr. Kaushal's qualifications appear at Page 50 of Exhibit 58 and Dr. Anderson's qualifications appear at Page 52 of Applicants Exhibit 58.

This panel is, through reason of training and experience, representing or provides expertise in regard to LMFBR technology and CRBRP design.

With that, the panel is ready for cross-examination.

JUDGE MILLER: You may cross-examine.

MR.MIZUNO: Does the Chairman wish the Staff to go and cross-examine first?

JUDGE MILLER: Yes.

CROSS-EXAMINATION

BY MR. MIZUNO:

Q. Mr. Longenecker, are you the lead witness for the panel?

BY WITNESS LONGENECKER:

A. That's correct.

Q. I'll direct questions to you and if you can't answer them, then direct me to the proper witness.

BY WITNESS LONGENECKER:

A. All right.

Q. Turn to Page 16 of the testimony.

You discussed the design and testing of the Clinch River steam generators.

Were failure experiences at foreign liquid metal fast breeder reactors steam generators evaluated and taken into account in the Applicants technical performance assessment of the Clinch River steam generators?

BY WITNESS LONGENECKER:

A. Yes, they were.

Q. Can you describe that, please?

BY WITNESS LONGENECKER:

A. Very summarily, we did consider in developing the design and the test plans for the Clinch River steam generators, the other applicable world experience which would, to my knowledge, include that from the British program, the French, the Germans, Japanese, Soviet program to the degree that there is information in the open literature, as well as some information that's available from the Dutch program.

They've done some testing for both the German and French programs.

Q. You looked at the information for specific steam generators, foreign steam generators?

BY WITNESS LONGENECKER:

A. Yes, we did. Again, for sodium cooled steam generators. The ones that I mentioned have a combination of test and or operating experience for liquid sodium cooled generators.

Q. And can you describe what you have learned as part of your research into this area?

BY WITNESS LONGENECKER:

A. Again, very generally. The information -- I guess I could put it in three basic categories, the information that we obtained.

We obtained quite a bit of information in the area of design of the units and the design methods.

Second, in the area of materials quality, materials performance and fabrication techniques.

And third, from an overall performance point of view, performance and maintenance and operations. Availability-type data.

Generally in those three areas we obtained information on each unit, to the degree that it was available.

Incorporated it into our considerations and weighed our results to the extent possible, against foreign information.

Q. I wonder if you could go into just a little bit more detail as to how you incorporated that information learned into your technical performance assessments for Clinch River steam generators?

BY WITNESS LONGENECKER:

A. Could you explain a little bit more what you mean by technical performance assessments?

Do you mean in the design, in the test program or --

Q. Well, in both.

How did you -- I believe it was your testimony that you gained a certain amount of information in three different areas.

Did you use that information in design of the Clinch River steam generators and, second of all, did you also use that information in developing a testing

7-8

program for the steam generators?

BY WITNESS LONGENECKER:

A. Let me try it this way. If I could.

Let me try to explain in each area how we used the information.

In the first area, as far as design, we did consider the applicable design information and operating and test information in design of our unit.

There are things that we learned from the foreign experience as far as the configuration of the unit, flow rates, heat rates, materials compatibility and the like, which we incorporated in the design.

Furthermore, as information was available in the fabrication of the units, we used that foreign experience.

One of the best examples is, to assure that we have good quality on the pressure boundary, particularly in the tubes, we purchased and did further development on some Dutch x-ray equipment. The rod anode x-ray equipment which we used in examining the full penetration welds, which we made in the tube sheet .

We learned quite a bit and incorporated the experience in the type of materials we used, led to a degree to our using the very pure vacuum arc remelt forgings and tubing material, which we did in the units.

We looked at their testing experience and some of their operational experience and by gauging to the degree that we could, the types of operational problems that they had encountered, we planned our test program to assure that we tested out those conditions.

Q. How did you incorporate or did you incorporate information concerning maintainability of foreign steam generators in your own, I guess, steam generator development?

And if you did do that, how did you do that?

BY WITNESS LONGENECKER:

A. We did. That's one of the considerations, and as far as maintainability, what we by and large -- we want these units to be -- have as high an availability and as low a maintainability as possible.

We took the approach again, of -- in looking at the world experience, you find that three things make them fail.

It's pretty common. One,you have bad materials. Two, you have bad fabrication techniques or dirty fabrication techniques, bad welding and, three, either by design or lack of testing, you have some operational phenomenon such as flow loose vibration -- one of the best known -- that you have not properly accounted for.

So we have, from the beginning, planned a unit that we believe will be high reliability and low maintainability, at the same time, within that, we have made the provision to be able to do internal inspection of the unit in the routine maintenance, so that we can get in there and assess whether there is a problem that is yet very small that could be detected and fixed during that routine maintenance period.

We obviously think that we have a very high -- a unit of very high integrity and which will be reliable through the life of the plant.

Q Okay.

JUDGE LINENBERGER: Excuse me, Mr. Mizuno; but two or three times you have used the phrase "low maintainability", which to me means you're going to have one devil of a time doing anything to something that is not performing right or has developed a problem.

Is that really what you mean when you say "low maintainability"?

WITNESS LONGENECKER: No, sir.

What I mean is high availability and a unit that, due to its high structural integrity, will require very little maintenance while it's in service.

JUDGE LINENBERGER: Thank you.

JUDGE HAND: If I could pursue that on second.

Is there something special about this steam generator that makes it easier to maintain? Is it different than an ordinary generator?

WITNESS LONGENECKER: As far as maintainability is concerned, it's a smaller unit than an LWR which, tc a degree has 757 tubes, where you've got probably four times that in an LWR unit.

It, as we believe on of the other panels discussed, does not have the problem with having the feedwater on the shell side and the inherent problems with tube denting and corrosion buildup that can come into. So, it's a pretty clean unit, having sodium on the one side.

It is also, using modular units with large manways and the other features that we have, we believe has some rather superior features for maintainability.

By and large, maintainability for a unit depends on building it in so that you can get in and have access.

We think we've designed it into this unit. You can really do that on any unit.

JUDGE HAND: Thank you.

BY MR. MIZUNO:

Q Continuing on Page 16, you refer to a full size prototype that is currently being tested to assure

7-12

that the plant units will meet the design parameters for steam conditions.

Is that prototype an exact prototype of the steam generator that is currently planned to go into the Clinch River?

BY WITNESS LONGENECKER:

A. To avoid the problem with semantics. As far as exact prototype, it is not exactly the same in all design details with the plant unit design.

It is -- the design of the steam generator unit is one of evolution, by the nature of the process. We have been developing the design since 1969.

The prototype was built and delivered in August of last year. We have learned some things from model testing that we have incorporated.

We have learned some things -- we talked about maintenance. We have learned a few things about seals and access to the unit, which we would incorporate in the plant unit.

What I could say, to answer you question, in a summary way; the differencs that exist between the prototype and the plant unit are small and they are part of the natural design evolution.

In my opinion and to the most important point, I don't believe that they in any way invalidate the

7-13

applicability of the results from testing that unit to the ultimate plant unit.

Q Do you intend to do additional tests on the prototype steam generator?

BY WITNESS LONGENECKER:

A. Yes, we do.

Q Could you describe those tests?

BY WITNESS LONGENECKER:

A. Well, the prototype tests which just began this year in sodium, at 70 megawatts,will run through 1983. There will be a series tests of increasingly severe conditions over that period.

In addition, there will be -- the test program really runs up through 1989. As I told you -- we believe it's a very high integrity unit, in that by the time we put the plant in operation, finish the last test, we will have accumulated 20 years R and D test experience on the unit.

To me, that's pretty impressive and I believe that it will be the most extensively tested unit in the world at that time.

Specifically, what we plan to do is, we will finish the sodium test on the prototype . There will be in 1983 time frame a one-third scale water test of the unit for flow induced vibration.

7-14

The first plant unit which will constitute the spare, will be produced in 1985. That unit will be tested in 1985-86 in water up to 125 percent full flow.

In 1988, according to the reference schedule, we will install those units in the plant and perform pre-operational tests in sodium up to 110 percent flow at temperature in the 1988-89 time frame.

So that the test program runs, really, through 1989 for the units.

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Q. Is there any slack in your construction schedule that would allow you to do fabrication of an exact prototype of a steam generator, test it, and then put it in the plant without any delay in the startup date for Clinch River?

BY WITNESS LONGENECKER:

A. I'm sorry. Could you define what you mean by "exact prototype"?

Q. Yes. The prototype which would incorporate those slight changes, I guess.

BY WITNESS LONGENECKER:

A. I guess what you really mean is the first plant unit?

Q. Yes.

BY WITNESS LONGENECKER:

A. Okay. If you take the dates that I just gave you, the first that you could -- The answer to your question is no.

The first that you could produce a plant unit is 1985 and 1986. That's physically as soon as the first one is going to roll off the line.

If you then take that to E-Tech where we are testing this unit, provided that there is time in the facility -- but let's assume we had priority -- it takes about six months to a year to install it, pipe it up and

8-2

prepare for the tests. You begin preparing tests in '87, and going with the logic that there was something to be learned from that test to incorporate in the other plant units, you wouldn't fabricate those.

If we had test results by '88, you couldn't deliver the units until probably four to five years later.

There is not that much slack. You are obviously talking about a several year, I would estimate three to five-year delay in the project to do that.

Furthermore, if I could add, with a 20-year test program, I just don't think it makes sense to go do one more test.

The amount of information you would get from that, I just don't think would be really significant, nor would it give us much higher confidence that that's going to work.

Q Thank you. Turning to Pages 18 through 21, Question and Answer 10, have the Applicants developed the data collection system for collecting and evaluating data on the maintainability of Clinch River, once it goes into operation?

BY WITNESS LONGENECKER:

A. Your question is a system. We do plan, obviously, since the demonstration of maintainability is one of our principal objectives, to during the demonstration

8-3

period to document carefully the maintainability experience that we have as to be used in further plants in the LMFBR program.

So yes, we do plan to document that carefully.

Q Is that data -- Can you describe the level of data that you would be collecting? Is it at the system level or how far down does it go?

Does it go to individual components or major components?

BY WITNESS LONGENECKER:

A For maintainability, the information that we collect and the information in our planning estimates will go down to the individual components of system level, because we do have allocations in our maintenance estimate, both for manpower and in our availability studies that go down to that level.

Q Will this data that you collect on maintainability of Clinch River be useful in the design of future liquid metal fast breeder reactors?

BY WITNESS LONGENECKER:

A Obviously, yes.

Q Okay. Turning to Page 24 of your testimony, and this is Question and Answer 12, could you describe how Clinch River will generate data that will be useful in designing future LMFBR's from an environmental acceptability standpoint?

BY WITNESS LONGENECKER:

A. Principally, information that we generate will demonstrate that LMFBR's are environmentally acceptable, that they can meet, as we say here, federal and state environmental regulations, and in particular any environmental advantages to LMFBR's, I think, due to this process of having them approved and measuring their physical and environmental impact will be demonstrated during the five-year demonstration period.

Q. Would it be fair to state that by showing that Clinch River can be licensed in accordance with appropriate federal, state and local laws that that goes a long way toward showing the environmental acceptability of future LMFBR's?

MS. FINAMORE: Objection. Leading the witness.

JUDGE MILLER: Yes, it is leading.

MR. MIZUNO: Withdraw the question.

JUDGE MILLER: You may have the witness describe the attainability of such objectives.

MR. MIZUNO: All right.

WITNESS LONGENECKER: If I could refer you to the answer, the first sentence on Page 24, Answer 12, we do state that: "We will meet the objective of environmental

8-5

acceptability by conducting construction and operation in conformance with the applicable federal and state environmental regulations.

So meeting that objective does require that we demonstrate that we can meet those, and yes, that is --

JUDGE MILLER: What are the applicable regulations, if there's going to be a distinction between applicable and non-applicable federal and state environmental requirements?

WITNESS LONGENECKER: Your Honor, that just meant to refer to the environmental regulations, any that are applicable.

We wanted to differentiate from other types of regulations which would not pertain to the environment.

JUDGE MILLER: The differentiation is between other non-environmental regulations and not between regulations, federal or state, where it may be arguably non-applicable; is that right?

WITNESS LONGENECKER: Yes, sir.

JUDGE MILLER: Okay.

BY MR. MIZUNO:

Q Okay. On Pages 24 and 25, you talk about the objective of demonstrating economic feasibility.

Do you have appropriate procedures, criteria or guidelines for separating costs into one-time developmental costs and non-prototype recurrent costs?

BY WITNESS LONGENECKER:

A. Yes, we do.

Q. Can you describe that system or those procedures, please?

BY WITNESS LONGENECKER:

A. In general, the plan that we have for accumulating cost data is one that is used -- has been used on the project since the 1974-75 initial detailed cost estimate.

It involves preparing a cost estimate and documenting actual cost experience at the total plant, total system, subsystem, component and actual materials quantity level.

For instance, those types of things go down to for various types of piping, the actual number of feet of different size piping and the cost per foot, and the cost of installing that.

We do plan to collect data at that level of detail.

Furthermore, for the plant we do have a system by which we can for each of the major cost categories distinguish and document. First-of-a-kind costs, what we would assume to be non-recurring costs, research and development costs, and then actual base capital costs for each of the systems.

8-7

Q. Will this cost accounting be useful in demonstrating the economic feasibility of future liquid metal fast breeder reactors; and if so, how?

BY WITNESS LONGENECKER:

A. Well, if I might, again I'd refer to Page 24, the first sentence to Answer 13, I believe, says it best.

That is that, "The economic feasibility objective will be achieved by developing this comprehensive cost materials quantity and performance information for the plant."

That does provide a data base which will allow us to extrapolate those costs to commercial size LMFBR's.

We also state in here we are using those currently in the rest of the program, namely for the large developmental plant.

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Q Okay. Turning to Page 42 of your testimony, Question and Answer 27, you conclude that, "The loss of flow HCDA will not be influenced by a choice of lower operating temperatures for the Clinch River Breeder Reactor System.

Are there any hypothetical accidents beyond the HCDA which might be favorably influenced if lower operating temperatures were adopted; and by "favorably influenced," I mean that you lower the probability of that accident occurring or you reduce the consequences of that accident?

BY WITNESS LONGENECKER:

A. Dr. Anderson.

BY WITNESS ANDERSON:

A. I know of no such accidents that would be more favorably affected by lower operating temperature.

Q Turn to Page 44, Question and Answer 29. You refer to the core catcher system. I think it's also known as a core retention system.

Under (b), you state that, "any active features provided in the core catcher have to perform in an extremely hostile environment...and are inaccessible at a time when they are required to function."

I wonder whether you could go into that a little further. What was the intended substance of that Part (b), Phrase (b)?

8-9

BY WITNESS ANDERSON:

A. Okay. The assumption in this case is that the core will have melted through the reactor vessel and the guard vessel. The environment, then, within which the core catcher has to work is the environment of high temperature, molten dispersed fuel in sodium.

The core catcher materials must withstand that. Its construction must withstand that and must not allow it to pass through in order for it to perform any function greater than that performed by the feature presently enclosed in the Clinch River design.

All of this, of course, is impossible at that time because we have no means of getting through the high temperature, sodium and fuel debris that would be within the core catcher in order to get at it and monitor its functions.

Q. Let us hypothesize that you did have an HCDA and there was a core melt and you had an active system, but when that HCDA occurred, the active core catcher did not work.

Would you be able to go and fix it subsequently and would that have an effect?

BY WITNESS ANDERSON:

A. I cannot think of any design in which you would be able to go and fix it subsequently; therefore, one would

8-10

have to have the features presently incorporated in the Clinch River design as a backup in order to insure that the health and safety of the public were protected, even in the event that we had a core catcher in the design.

Q. Okay. Turning to Pages 45 and 46, Question and Answer 30, you talk about elimination of venting of the containment during normal operation, making containment access during normal operation difficult.

Could you specifically identify or give some examples of some operational and maintenance functions which would be difficult if you did not have continuous venting.

BY WITNESS LONGENECKER:

A. Dr. Kaushal.

BY WITNESS KAUSHAL:

A. The idea here is that if the containment is completely closed and unvented, then the atmosphere within the containment will not be hospitable to the workers who might want to approach within that containment to do some maintenance work.

Making it more difficult would mean that under the way we would envision it, the operators will have to put on breathing equipment like the air bags in order to go in, and that would make access into relatively confined areas difficult and would also make their maneuverability

8-11

in case of an emergency considerably less.

I might add to that, also, that a completely enclosed containment, in order for someone to enter it, you have to go through a relatively elaborate procedure to open up a maintenance hatch into the containment to get in there.

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MR. MIZUNO: Thank you. The Staff has no further questions at this time.

JUDGE MILLER: Intervenors?

MS. FINAMORE: Yes.

CROSS-EXAMINATION

BY MS. FINAMORE:

Q I'd like to turn to Page 4 of the testimony. Mr. Longenecker, you state in the bottom of Answer 5 that the programmatic timing of CRBRP has been established by the DOE FEIS, and the record of decision is "as soon as possible."

Wasn't there an earlier timing goal of the project, which was expressed as a date certain?

MR. EDGAR: Objection. Relevance.

MS. FINAMORE: I believe this is very relevant. We're trying to establish the ability of the plant to meet its timing objective.

MR. EDGAR: Well, the Commission --

JUDGE MILLER: Yes, that's true. But the Commission's order did establish certain things to be taken as given, which my recollection is includes the timing.

Although the timing may have been changed in its statement, nonetheless, the Board feels bound by the Commission's directive.

So you may inquire into anything that you wish as far as the timing and the objectives in connection with it, but I don't see that you're going to gain anything by going back to a different period of time when there were different stated objectives because we -- the Board doesn't have the power to go into -- or make decisions regarding timing. We're to take that as a given, I believe.

MS. FINAMORE: But the Board must determine whether or not that objective has been met or has to decide --

JUDGE MILLER: Well, it's likely -- reasonably likely to be met, I believe. Isn't that the statement of the objectives in terms of the contention?

MS. FINAMORE: Well, the reason for this question was not to challenge this timing objective. The reason was to determine the meaning of this timing objective, and as such, by going into the background of it, we can establish what the meaning is of this fairly ambiguous statement.

That's the reason why I'm asking these questions, not to challenge it.

JUDGE MILLER: Well, isn't it a little sophistry to say that you're going to go into the change in order to see the present meaning? Why don't you just

9-3

go into the present meaning?

BY MS. FINAMORE:

Q. Can you explain the meaning of the term "as soon as possible"?

BY WITNESS LONGENECKER:

A. Yes. The term "as soon as possible" means that it is incumbent upon the Department of Energy, and we, as the Applicants, to do everything within our power, within the constraints, of meeting other laws and the other project objectives to complete the project as expeditiously as possible.

Q. How do you determine whether or not you've met this timing objective?

BY WITNESS LONGENECKER:

A. It is a measure of judgment in weighing various alternative means for pursuing the project. Again, the -- within the constraint of meeting the other project objectives after having weighed those alternatives, the one which completes the project at the earliest date is by definition as expeditiously as possible.

Q. How much weight do you give to that so-called timing objective in relation to other programmatic objectives?

BY WITNESS LONGENECKER:

A. It is one of the objectives. It must be weighed along with all of the others.

Q. But you stated that first you analyze the other programmatic objectives, and after you analyze them, you go ahead with one that most likely meets the timing objective; is that correct?

BY WITNESS LONGENECKER:

A. No, that's not what I stated.

Q. Can you explain what you meant by you weigh the timing objective after you have considered other program objectives?

MR. EDGAR: Objection. That's not what the witness said. The question presumes that's what the witness said.

JUDGE MILLER: I think that's correct. I don't think that's an accurate characterization of this testimony.

I think you are putting in -- or at least emphasizing the term "after" in a manner in which the witness did not.

BY MS. FINAMORE:

Q. Am I correct that you stated that you look at the timing objective after you've considered other program alternatives?

BY WITNESS LONGENECKER:

A. No.

9-5

Q At what point in your analysis do you consider the program alternatives and the timing objective?

BY WITNESS LONGENECKER:

A As -- Well, excuse me, but I think you just asked a different question.

Are you asking program alternatives or timing?

Q The relative timing of your analysis.

JUDGE MILLER: That is a different question. What are you asking now? You had better rephrase it. You've asked two different questions. I'm not sure which you want, really.

BY MS. FINAMORE:

Q At what point in your analysis do you consider the timing objective?

BY WITNESS LONGENECKER:

A The timing for Clinch River in its completion is one of the overall objectives. We must meet all of the program project objectives, and it is weighed concurrently with the other objectives.

Q So, therefore, unless you meet all of the objectives, you cannot -- Let me -- Am I correct in understanding you that you feel the program must meet every objective?

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BY WITNESS LONGENECKER:

A. I believe you're correct in assuming that we wouldn't have those objectives unless we intended to meet them. That's -- To me, yes, that's the definition of an objective. We do plan to meet all of the program project objectives.

Q. If it's possible that an alternative, such as an alternative site, is found to be substantially better, would the results in delay in moving to that site prevent completion of the CRBR as soon as possible?

MR. EDGAR: Objection. Scope of the contention. We've already had a full day of hearing on alternative sites.

I don't believe this panel is up here to address alternative sites.

MS. FINAMORE: No. I'm just addressing the timing objectives.

MR. EDGAR: You said alternative sites.

MS. FINAMORE: Yes. And --

JUDGE MILLER: We are getting into an area that has both already been covered and is contained in other issues under the pleadings, namely, alternative sites. We don't want to go back into alternative sites.

MS. FINAMORE: Well, to the extent that it relates to the timing objective, it is covered by this contention which says the plant has to meet its objectives.

JUDGE MILLER: Well, it may be that there's overlap. Where there's overlap, we've already indicated for the purpose of completion of this phase of the hearing in a timely fashion, that we wish to avoid such overlaps or redundancies --

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MS. FINAMORE: If I may point out --

JUDGE MILLER: The objection will, therefore, be sustained.

MS. FINAMORE: -- when the earlier panel on alternative sites ...

I'd like to make an offer of proof, if I may.

JUDGE MILLER: How do you make an offer of proof on cross-examination? An offer of proof is affirmative evidence. You're not entitled to make an offer of proof on affirmative evidence at this time, although you may do so appropriately at a later time, such as rebuttal or something of that kind.

You don't make offers of proof on cross-examination.

MS. FINAMORE: I just wanted to point out that the earlier panel --

JUDGE MILLER: You're arguing. We've ruled. I've told you several times that once we have ruled, right or wrong, we wish you to proceed expeditiously and in a timely fashion without further delay.

Proceed.

BY MS. FINAMORE:

Q Mr. Longenecker, if an alternative design is found to be substantially better, would the resulting delay prevent completion of CRBR as soon as possible?

BY WITNESS LONGENECKER:

A. I don't understand your question. I'm sorry.

I'm sorry. I just don't understand your question. Would you repeat it?

Q. I'll repeat it. If an alternative design was found by you to be substantially better than the one proposed, would the resulting delay prevent completion of CRBR as soon as possible?

BY WITNESS LONGENECKER:

A. I'm sorry. I'm just having trouble with that answer.

JUDGE MILLER: I'm having the same trouble. I think it's a confusing question, frankly.

BY MS. FINAMORE:

Q. Assume hypothetically that you discovered that an alternative steam generator design was substantially better than the present one, would the resulting delay in moving to that new steam generator design prevent completion of the CRBR in -- as soon as possible?

BY WITNESS LONGENECKER:

A. Again -- Let me try to answer that by saying -- I'm having trouble with the hypothetical.

We believe the current steam generator design to be adequate, as a tremendous weight of demonstration, as said before, a total of 20 years of experience before we put it into operation.

If we were to find during the subsequent test program that there was some feature of that design that would not meet the overall performance objective for the generator and would, in our belief, jeopardize our ability to meet our objectives, we -- the other program objectives of demonstration of technical performance, we would obviously take the time to repair that.

We don't think that's the case. If your hypothetical -- If I can assume that your hypothetical means that were there a feature of a steam generator design that, taken in the abstract, might be deemed to be somewhat better, that feature incorporated in the design would not, in my opinion, make the plant as a whole better, or the steam generator system --

JUDGE MILLER: Remember, the term is "substantially superior," as the Commission has instructed this Board, and hence you, to consider it. And that was incorporated in the question, too, I believe.

So consider it from that point of view because otherwise we're beyond our powers anyway.

WITNESS LONGENECKER: If -- Well, I think I've given you the best answer I can.

BY MS. FINAMORE:

Q. In the hypothetical case that I've just given you where an alternative steam generator was found by you to be substantially better, am I correct that it would involve some amount of time for you to change to the new steam generator design?

BY WITNESS LONGENECKER:

A. I just can't answer that hypothetical. I'd have to know how much -- what "substantially better" meant, the terms of the technical superiority as -- to know what that timing means and whether it be incorporated and what that would mean to the overall timing objectives.

I'm sorry. I just can't --

Q. I'm asking you to assume hypothetically that you have discovered an alternative steam generator design which is substantially better.

BY WITNESS LONGENECKER:

A. How substantially better?

Q. As defined by the Commission.

BY WITNESS LONGENECKER:

A. I have never seen a definition by the Commission of a substantially better steam generator.

Q. Substantially better, in your judgment. Assuming -- Isn't it possible that you could have a design that is substantially better, or a component that

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is substantially better?

BY WITNESS LONGENECKER:

A. I'll grant you that anything is possible. I think it's highly unlikely that you could develop a steam generator that's substantially better, or even measurably better than the one that we have.

Q. But isn't it true that your present ongoing testing program could discover previously unknown faults, as you've mentioned earlier, for operating difficulties?

BY WITNESS LONGENECKER:

A. It certainly is possible that the testing program might uncover things that we had heretofore not expected.

I believe that's extremely unlikely, given the base of experience that we have. But anything is possible.

Q. Thank you.

Now, assuming, hypothetically, that you have discovered an operating difficulty heretofore unnoticed that would cause a move to an alternative steam generator that you consider substantially better -- assuming that hypothetical, am I correct that it would require some amount of time for you to move to the new steam generator design?

MR. EDGAR: Objection. First, that question was asked and answered. The witness said he could not answer that question without knowing what specific type of generator change we're talking about so he could translate that into a time factor.

And, furthermore, he expressed a lack of knowledge of the surrounding circumstances.

The second point is, as the record now stands, the predicate for the hypothetical is that it is theoretically possible in the sense that anything is possible, but it is extremely unlikely that the hypothetical will occur.

It seems to me that by definition now, not only are we having redundant cross-examination, but we are getting into cross-examination which is speculative to the extreme.

MS. FINAMORE: If I may respond.

When I originally asked the question, the witness expressed some uncertainty regarding the term "substantially better." That has to go -- That goes to the hypothetical situation.

I asked and he then agreed that this hypothetical situation was possible, based on the presence of their present and ongoing testing program.

Now, once that situation is possible, I wish to use it as a hypothetical situation and ask him the

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simple question, if, in such a hypothetical, some amount of time would be required.

I'm not asking him for the extent of time required, which is what he asked about earlier. I'm asking if any amount of time would be required. It's a very simple question.

JUDGE HAND: Ms. Finamore, if at some point in time before CRBR goes into operation, some piece of equipment -- whether it's the steam generator or anything else -- was suddenly found to be so bad that it was environmentally unacceptable, I think one has to assume that the program would stand still until that problem was solved.

And eventually, as soon as possible, it would be solved; and it would go on. It might be a day. It might be a thousand days or a thousand years. I can't imagine.

It seems to me that you're asking a question that has a terribly obvious answer. And you wouldn't have learned anything when you get the answer.

MS. FINAMORE: I think the problem, Dr. Hand, is that -- as we'll explain in our closing argument -- I think you're absolutely right; and that is because the way the timing objective is stated now, there is no way that the present design could not meet it because it has no meaning. And that's what we're trying to establish on the record here.

MR. EDGAR: Now, I object to the entire line of questioning. She just said the magic word. She said, "The timing objective has no meaning."

Now we're going to the merits of the timing objective. That's exactly what she said, and she should be held to that representation.

JUDGE MILLER: All right. The --

MS. FINAMORE: What I meant to say in that situation --

JUDGE MILLER: The Board has heard enough on this.

We believe that the effort to go into the question of timing has to be within the constraints of the ruling of the Commission, which was leaving open the consideration of alternative designs to meet the objectives taken as given.

Now, to be a substantially better design would certainly have to be something that was susceptible -- both being described and understood by any technical witness to give any testimony that had the timing aspect. To go into that, obviously the question doesn't do it.

If your effort is, as you latterly put it, to take the position that it was impossible to achieve

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anything as soon as possible, and, hence, that the timing is simply illusory, that is properly a matter to be left to argument, rather than to argue at this time with either the witness or the Board.

So, therefore, we sustain the objection.

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BY MS. FINAMORE:

Q Mr. Longenecker, in your judgment, are there any alternative designs to the present one that could be met that could meet the programmatic objectives if applied in a timely fashion?

BY WITNESS LONGENECKER:

A I apologize again. I just don't understand your question.

Q Are you aware of alternative design features that have been proposed to the CRBR present design?

BY WITNESS LONGENECKER:

A Yes.

Q Can you explain what those features are to me, please, in a general sense?

BY WITNESS LONGENECKER:

A In a general sense, I suspect that during the ten years the project has been under way, there has probably been an alternative evaluated for each aspect of the design, from the fuel itself to the turbine generator.

That has been part -- as we describe in the testimony, that is part and parcel of how we got to the reference design, was evaluating alternatives to designs and picking one that considering all the design features in the aggregate best met the over-all project objectives.

So you take the totality of all the alternatives

10-2

for each feature in the plant, and at one time or another alternatives to each of those has been proposed and evaluated by us.

Q Are you aware at the present time of alternatives to the CRBR design that are still being proposed?

BY WITNESS LONGENECKER:

A I'm aware that you have proposed several in your contentions.

Q And which ones of those are you aware?

BY WITNESS LONGENECKER:

A That's contained in Answer 24 in our testimony, Page 37 through 39. I can read that if you'd like.

Q Well --

JUDGE MILLER: No. If it's already in your testimony, you won't have to say it twice. You've already said it once, you see, because this written document is your testimony given here under oath or affirmation.

BY MS. FINAMORE:

Q Referring to the alternative designs in your Answer 24, namely the pool-type system, flywheels on the sodium pumps, lower system operating temperatures, third shutdown system, core catcher, and no-vent containment, in your judgment could the use of any of these features enable the plant to meet its timing objective?

10-3

BY WITNESS LONGENECKER:

A. I'm sorry. We just went from alternative designs to timing objective again. You just confused me with your question.

JUDGE MILLER: I think that this whole thing boils down to some fallacious underlying and unarticulated reasoning.

If there's a substantially better design to meet the stated objectives to be taken as given, then whatever time is required to meet and to adopt that substantially better design is as soon as possible.

Now, you keep using "as soon as possible" on the present when you are trying to transpose it to something else to imply that there is some intellectual distinction, which I don't think that the witness is recognizing and I don't think that the Board is recognizing.

As soon as possible, I think, is in terms of what you're talking about, and a substantially better design which should be adopted to meet their own objectives, the witness has said would be, and whatever physical time required would be as soon as possible to meet the objectives with the obviously or substantially better design.

Now, we are getting those two concepts confused and that's why he's looking at it from one point of view and you from another, I believe.

So if you'll define what you mean when you go into -- or back to "as soon as possible," then we'll get direct answers and then we'll go on with it.

If you don't want to do it that way, you are always going to be having the witness saying he doesn't understand, because he really doesn't.

BY MS. FINAMORE:

Q When I say "as soon as possible." I am referring to your timing objective, which is to complete the plant in a manner as soon as possible.

BY WITNESS LONGENECKER:

A Let me try to answer. I'll do the best I can. There are several objectives to the project, one of which is timing.

We must meet all the project objectives in their entirety. You have mentioned design alternatives.

We have analyzed those ourselves. We have discussed in the testimony in the Answers 25 through 31 some alternatives.

The first test is the threshold decision of are any of these substantially better than the features that we have on the plant, or do they in the aggregate provide us with a substantially enhanced ability to meet the project technical objectives.

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As we say in the testimony, in evaluating those, each of those failed to make that showing, that threshold that they are substantially better.

So we, therefore, since they did not give us a substantial enhancement in our ability to meet the project technical objectives, we have not incorporated those, and proceeding with this plant, which we believe to be fully technically adequate and to be able to meet our technical performance objectives does meet the timing objective as expeditiously as possible.

On the other hand, if we were to incorporate one of these design alternatives for some reason which does not substantially enhance our ability to meet the project objectives and take a delay in the project -- for some reason, if we were to decide, even though it weren't to make the plant better, we were going to put in one of these features, that delay would be in violation of the project objective, as expeditiously as possible.

Q So am I correct to assume from your statement that if a particular alternative to the CRBR was found to be substantially better, the resultant delay in changing to that particular alternative would not violate your timing objective, since it would be a necessary change?

BY WITNESS LONGENECKER:

A. Again, if I could refer to our test, a design

10-6

feature in and of itself being superior does not mean that its incorporation into the plant would make the total plant design superior, nor would it significantly necessarily enhance our ability to meet the project technical objectives.

There are none of these that we believe would cause us to fail to meet the project technical objectives.

If we were to pick an advanced material, of which there are many, and put it in the plant, and although that might be judged to be some type of technological advancement, it would not significantly enhance our ability to meet the project technical objectives.

The design we have is adequate. The materials we have are more than adequate.

If we were to take a delay to do that, that would not be consistent with our objective, as expeditiously as possible.

Q Well, given that answer, let me ask you this: If the use of an alternative design for a particular system in the plant would enable the entire CRBR Project to meet its program objectives in a substantially better manner, is it true that the delay that would be necessitated in moving to that alternative design would still be consistent with your timing objective?

10-7

BY WITNESS LONGENECKER:

A. I'm sorry, I just can't answer the question. Again, you've got so many hypotheticals in there that I would have to know what the feature was and how it would enhance our ability to meet -- I think you've taken me back to program rather than project objectives, but I just can't answer the hypothetical. I'm sorry.

Q. Okay. Isn't it possible that there could be an alternative design, such as one of the ones mentioned in Answer 24, that would enable the project to meet its programmatic objectives in a substantially better manner? Is it possible?

BY WITNESS LONGENECKER:

A. Yes. In the theoretical definition of possibility, anything is possible. I think it is extremely remote and very highly unlikely, in my professional judgment.

We have looked since 1968 at these design features. We discuss in the testimony a very rigorous procedure for doing that, and I honestly don't think that any substantially better design alternatives exist.

That's why I'm having trouble with your hypothetical. I just can't imagine what it would be that we would put into this plant that would make it substantially better.

Q. Well, isn't it possible that the Staff, when it completes its detailed design review, might in fact find that use of one of these alternative design features or another design feature might enable the project to better, or to substantially better meet its program objectives? Isn't it possible?

BY WITNESS LONGENECKER:

A. Your question now is program, rather than project technical objectives?

Q. No. It's the objectives that you've stated in your testimony that's the subject --

BY WITNESS LONGENECKER:

A. I wasn't aware that the Staff was reviewing our program objectives as part of the licensing review. That's why I'm having trouble with your question.

JUDGE MILLER: Let's take a ten-minute recess.

(Recess taken.)

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JUDGE MILLER: You may proceed, Ms. Finamore.

BY MS. FINAMORE:

Q Mr. Longenecker, I'd like you to assume as a hypothetical that a particular alternative design to the CRBR would enable the project to meet its programatic objectives in a substantially better manner.

Isn't it true, in such a case, that despite the delay necessary to incorporate such a substantially better design alternative, the timing objective would still be met?

BY WITNESS LONGENECKER:

A I'm having the same problem with your question. I'm sorry.

Q You said it was possible that an alternative design feature would enable the project to meet its programatic objectives in a substantially better manner; did you not?

JUDGE MILLER: Pardon me.

I think that part of the problem here is the "substantially better" feature.

I think what the witness said is, a feature or features could or could not -- might or might not result in a substantially better overall design.

I think that's the substance of his testimony.

Now, if you put your question in the same context, I think he can answer it, but if you do the other, he's going to be continued --

MS. FINAMORE: Yes.

BY MS. FINAMORE:

Q. Assuming, Mr. Longenecker, that the use of a particular alternative design feature would change the overall design of the plant in such a manner as to meet the programatic objectives in a substantially better manner, isn't it true that the time necessary to incorporate this alternative design feature into the design would still meet the timing objective?

BY WITNESS LONGENECKER:

A. I'm still having trouble with the question because it's so long and involved.

Let me try. It's got so many hypotheticals in it and so many unknowns.

If we were required to meet the project objectives, to incorporate a design feature that, in effect, took some period of time to incorporate, such that we would delay the project completion by that period of time and that were required to meet the other project objectives, that technical change, that we would meet as expeditiously as possible.

Q. And when you say required to meet the programatic objectives, doesn't that mean that such an alternative is substantially better?

BY WITNESS LONGENECKER:

A. Required to me means that we would weigh all of the design features, we would weigh the total plant design against the overall project and program objectives and we would deem that there was something in the plant that -- or in the plan as a whole that would prevent us, unless changed, from meeting the program objectives.

Q. Now, that's a different answer than my question, then.

BY WITNESS LONGENECKER:

A. I told you I didn't understand your question. I'm sorry.

JUDGE MILLER: I don't think it's a different answer.

I think if you follow what I've tried to explain to you were the parameters of the information you are conveying by the hypothetical question, I think that the answer is perfectly consistent with the partial answers given before and by the tangential nature of your question.

BY MS. FINAMORE:

Q. Then, are you saying that an alternative design would not be substantially better unless it were -- unless failure to use such a feature would prevent you from meeting the objectives?

BY WITNESS LONGENECKER:

A. I think the answer is no, that is not what I'm saying.

Q. So, in other words, even though your present design might meet the program objectives, it's possible that an alternative design would be substantially better in meeting those objectives?

BY WITNESS LONGENECKER:

A. It is possible but I think very highly unlikely. It's within the realm of theoretical possibility, where anything is possible.

Q. And in such a case, where you could meet the programatic objectives with your present design, but an alternative design would enable you to meet the programatic objectives in a substantially better manner , is it not also true that changing the design and the time involved, would still enable you to meet the timing objectives?

BY WITNESS LONGENECKER:

A. I don't understand the question. I'm sorry.

Q. You stated previously that it's possible that you could meet the present design -- the programatic objectives with your present design but that an alternative design might enable you to meet the objectives in a substantially better manner; is that correct?

BY WITNESS LONGENECKER:

A. No, I don't believe I stated that.

Q. Is it possible, then, that you can meet your present -- that the present CRBR design can meet its programatic objectives?

BY WITNESS LONGENECKER:

A. Yes, it does meet the programatic objectives. The entire project, the design, timing, all the rest, does meet the overall LMFBR program objectives.

Q. And is it possible that an alternative feature or an alternative design might enable you to meet the programatic objectives in a manner that's substantially better?

BY WITNESS LONGENECKER:

A. I just can't answer the question.

Unless you could define the specific feature and we could weigh it as far as timing, I just can't answer your question. I'm sorry.

Q. Isn't it possible, however --

BY WITNESS LONGENECKER:

A. Anything is possible, I'll grant you. Theoretically it's possible but I think it is highly unlikely.

Q Thank you.

Now, I'm asking you to assume hypothetically that such is the case and in such a case, isn't it true that the time required for you to change to that substantially better feature, would still meet the timing objectives?

BY WITNESS LONGENECKER:

A I'm sorry. I just don't understand the question.

JUDGE MILLER: That's probably about as far as we can go. We've taken up quite a bit of time on this.

WITNESS LONGENECKER: If you can give me a specific --

JUDGE MILLER: Well, no, now. Let's not make suggestions.

WITNESS LONGENECKER: I'm sorry.

JUDGE MILLER: Just leave things where they are. We think we've exhausted the area.

I think the Board understands what some of the problems are but then we can't get you together, so we suggest that you move on to some other area.

BY MS. FINAMORE:

Q I'd like to turn to Page 28 of your testimony.

11-7

And you state in the final sentence that the CRBRP systems design provides a basis for all the LDP systems design, referring to a table on Page 29 to 32.

When you say that the systems design provides a basis for the LDP, are you saying that there will be no changes in the design, other than a scale-up in size?

BY WITNESS LONGENECKER:

A. No.

Q. What changes are you anticipating in the LDP, from the present CRBRP design?

BY WITNESS LONGENECKER:

A. I can't list them all for you. I can say that they fall in two general categories.

Design changes that are required due to the increased size of the plant and some design features are obviously size-dependent.

And other features that would be incorporated due to the technological advancements that may be made in the time intervening between CRBR and large developmental plant.

Q. What do you mean "time for CRBR"?

BY WITNESS LONGENECKER:

A. On its reference schedule.

Q. Are you saying the difference between the

11-8

design, once it's operating or from the present time?

Which changes are you referring to?

BY WITNESS LONGENECKER:

A. Clinch River is obviously -- the design is almost complete and when it's constructed and operated we will learn some things from that which we will incorporate into the technological base.

We, as described in the program statement, also have an ongoing base technology program that, for plants which will be built in the time frame of the LDP, which is anticipated to be on the order of five to ten years after Clinch River, if some of those features prove to be promising, they may be incorporated in the LDP.

I can't tell you what they are because they don't exist at this time but that is the reason for our base technology program.

Q. But setting aside the base program and also setting aside the necessary changes due to the scale-up in size, am I correct that the LDP is presently anticipated to involve the same design as the CRBR, except for changes that you might make as you learn more about the plant?

BY WITNESS LONGENECKER:

A. In a very general sense, that's true.

As I told you, there will be in a relative sense, a substantial number of changes between the two.

There will be more cable in LDP because it's a larger plant.

There will be more concrete. There will be more feet of pipe but in the conceptual sense, the LDP concept, as currently developed, is a loop-type plant with gneral same systems and layout as the CRBR.

So, conceptually, they are generally equivalent.

Q. Now, again, setting aside the changes that are necessary due to the size of the plant, are you expecting there to be changes in the design due to the information that you are learning or you expect to learn from the CRBR?

BY WITNESS LONGENECKER:

A. I would say that -- this is LDP or CRBR?

Q. You said that you --

BY WITNESS LONGENECKER:

A. I'm sorry.

Changes in the design to LDP?

Q. Yes.

BY WITNESS LONGENECKER:

A. I'm not expecting that there will be changes to the design.

What I am expecting is that the construction and operation of CRBR will confirm the validity of the

11-10

design of the LDP.

Q. Well, given that, then, am I correct that you are assuming that other than changes due to size, the LDP will be pretty much the same design as the CRBR?

BY WITNESS LONGENECKER:

A. That's not what I said.

What I said was, that conceptually they are the same type of plant. They are both loop plants. One is obviously larger.

They are both LMFBR's. They both have the same general system layout as is shown in the Page 29, 30 31 and 32.

There will be changes, obviously in the plants due to size. There will be changes due to any technilogical advancements that occur between now and the time that the LDP final design and construction is undertaken.

So there will be both of those. It's not just designs.

Q. Yes, but then, at the present time, other than changes due to size and other than changes in the base program, is the LDP design the same as the CRBR design?

BY WITNESS LONGENECKER:

A. It is in concept very similar. It is not identical to the CRBR.

Q Now, again, other than changes due to size --

JUDGE MILLER: This is getting redundant. We're growing short on time. You're going to regret it very shortly, so I suggest that you get onto something else.

Or else --

MS. FINAMORE: I have a final question.

JUDGE MILLER: All right. Go ahead.

BY MS FINAMORE:

Q What, if any, features are different in the LDP from the CRBR, at the present time, other than those due to size?

BY WITNESS LONGENECKER:

A. It's again in a very absolute sense,there are quite a few changes in the plant.

Let me explain what I mean by due to the size. I think you're talking about the LDP as a 1000 megawatt plant.

When you go to a larger plant, one of the notable changes between the two is the refueling system. When you get to a core of the size of the large developmental plant, you need a different fuel handling scheme than we've used on FFTF and CRBR to be able to reguel the plant in the nominal outage that you allocate; something like 16 days each year.

11-12

Therefore, you do have a different fuel handling system, an A-frame fuel handling system on LDP in the present conceptual design, where you have the EVTM, exvessel transfer machine on Clinch River, which is the same as FFTF.

That is a size difference.

You have a different decay heat removal system on LDP than you do on Clinch River. The reason for that is that you have more decay heat in a 1000 megawatt or 25,500 megawatt thermal core in LDP, compared to the 975 megawatt thermal core -- I could go through the plant and there are, you know, differences like that.

It is a four-loop plant because, again, of the size factor.

I could continue to enumerate changes for you but by and large, they are related to the difference in the size factor and the things that we must do to make the plant operate and meet its objectives in the large size.

Q. So there are no differences that are due to factors other than size; is that correct?

BY WITNESS LONGENECKER:

A. No, that isn't what I said.

What I said is, there will be differences due to size and due to advancements in the technological advancements which come about from the LMFBR base program.

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Q. And are they incorporated in the LDP at the present time?

BY WITNESS LONGENECKER:

A. As I said, I'm talking about future advancements, so obviously they are not incorporated at this time.

Q. On Page 33 of your testimony you state in the middle of the page:

"That is the same manner that a large portion of the information obtained from CRBRP is directly relevant to LDP. The information from the design, construction and operation of CRBRP can also be reasonably expected to provide significant information of relevance to commercial LMFBR's in the future."

On what grounds do you believe that any future LMFBR's will be of a design similar to CRBRP?

Briefly.

BY WITNESS LONGENECKER:

A. I'm having trouble relating that question to the sentence in the testimony you just read.

What that sentence means to imply, is not that future plants will be identical. It obviously doesn't say that.

It says that CRBR can also be reasonably expected to provide significant information of relevance to commercial LMFBR's of the future.

The maintenance information that we obtain from Clinch River, that I previously spoke of, information on component performance, on core design and performance, on the compatability of various materials with the coolants, instrumentation and controls, operability, plant availability, all of that information will be relevant to commercial LMFBR's.

Q Well, then, do you have any basis for believing that the future LMFBR's would use a design that is similar to the CRBR?

BY WITNESS LONGENECKER:

A. Yes.

Q What basis do you have for that?

BY WITNESS LONGENECKER:

A. My professional judgment that the concept that we have is sound and is reasonably viable.

Obviously I don't think the -- we have stated in the testimony that it is obvious that this CRBRP is not a commercial size LMFBR but I believe that the concept which it demonstrates has a very reasonable probability of being adopted in future central station commercial size LMFBR's.

Q Have any utilities expressed a concrete interest in ordering an LMFBR with a design similar to CRBR?

MR. EDGAR: Objection. Relevance.

JUDGE MILLER: Sustained. Irrelevant.

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12-1

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Q. On Page 34 of your testimony, you state that, "As a result of tests at the ZPPR on heterogeneous core, the core design on the LDP and larger LMFBR's can proceed with a higher degree of confidence."

Is this enhancement of confidence considered important, in your judgment?

BY WITNESS LONGENECKER:

A. I'm sorry. Could you point me to the sentence that you are reading from? Where on Page 34?

Q. It actually begins in the last word on Page 34 and carries over to Page 35.

BY WITNESS LONGENECKER:

A. Okay, and your question was, please?

Q. Is this enhancement of confidence considered important, in your judgment?

BY WITNESS LONGENECKER:

A. I think it's important that we did the tests at ZPPR, and that gave us confidence to be able to predict the core performance. So I believe the tests were important, yes.

Q. Is there room for improvement in the confidence of the design of the CRBR core?

BY WITNESS LONGENECKER:

A. I can't answer the question. I don't know that -- I can't grapple with it. Improve the confidence?

12-2

Q. Yes. You stated --

BY WITNESS LONGENECKER:

A. I don't know how I could improve my confidence in the CRBR core.

Q. You said the results of the ZPPR tests improved your confidence in the core design; isn't that correct?

BY WITNESS LONGENECKER:

A. I think test data in any manner will improve to some degree your confidence that a system or component will function, and this is meant to imply that having performed that test, we have a higher confidence in the core performance than if we had not performed the tests in ZPPR.

Q. I'm relating it to your degree of confidence at the moment. Is there any room for further improvement in confidence as the result of further tests?

MR. EDGAR: For what core, LDP or CRBR?

MS. FINAMORE: CRBR. Oh, excuse me. This sentence refers to LDP and so does my question.

MR. EDGAR: Well, you asked the same question while ago and it was confused on that point.

JUDGE MILLER: Let's proceed.

BY MS. FINAMORE:

Q. Are you performing tests on the core design

12-3

of the LDP?

BY WITNESS LONGENECKER:

A. Are we? We have, to my knowledge, not yet done any criticals in ZPPR. They are scheduled for later, of course.

The CRBR -- or the LDP, excuse me, does have a heterogeneous core design. As the project progresses, we would plan to do critical experiments for LDP, most likely in ZPPR, just like we did for Clinch River.

This statement, again, says that the criticals done for Clinch River and the experience that we attain from the core will give us high confidence in the LDP core.

Q. On Page 37 of your testimony, Answer 23, the final paragraph in that answer, you state that:

"The major design features of CRBR were the product of a systematic review and were responsive to the needs identified by the ultimate user - the utility industry."

Am I correct that those needs of the utility industry were identified during the period from 1968 through '72, based on your earlier testimony?

BY WITNESS LONGENECKER:

A. I don't recall giving any testimony on when the utility needs were defined. Are you speaking of --

Q. Written testimony.

12-4

BY WITNESS LONGENECKER:

A. -- a statement from my testimony?

Q. Yes.

BY WITNESS LONGENECKER:

A. The project definition phase of the project was conducted in that time frame. If you'll look at the Answer 23 on Page 36, I think that's pretty well described there, when we talk about during the project definition phase, having a cooperative activity with the utility industry to define the objectives for the project.

They are set forward in our quadripartite contract between DOE and the utilities.

Q. And are those similar or the same as the needs identified by the utility industry that you mentioned in the final sentence on Answer 23?

BY WITNESS LONGENECKER:

A. Yes, I would say so. I wouldn't limit it -- I just can't limit it to that time frame, though.

The utilities, obviously, are a partner in this project. They have over a hundred utility representatives in the project with us.

The utility input and the assurance that we meet the utility and user needs is done on a continuing basis. So the time frame we are talking about was the establishment of the original objectives of the utilities.

12-5

Since that date, on a continuing basis, through both the Project Management Corporation, representatives from Tennessee Valley Authority, Commonwealth Edison, the Breeder Reactor Corporation, all those entities have periodically and continuously reviewed the design to assure that it is responsive to their requirements.

Q. Have the objectives or needs of the utilities changed since they were identified in the project definition phase, to your knowledge?

BY WITNESS LONGENECKER:

A. With regard to LMFBR's or central station power plants or Clinch River or -- I'm having trouble with your question. Have their needs changed?

Q. Well, according to your testimony, the project definition phase refers to the LMFBR program and its demonstration facility; is that correct?

BY WITNESS LONGENECKER:

A. The testimony refers to an interaction to define what would be the appropriate design features of CRBRP, as identified by the both cooperative effort between the government, industry and the utilities.

So has their needs changed as regards CRBR? No.

Q. Or as regards to the major features of the CRBR?

BY WITNESS LONGENECKER:

A. No.

Q. But there have been changes to the major features of the CRBR since that time, have there not?

BY WITNESS LONGENECKER:

A. No, not to my knowledge.

Q. Such as core change?

BY WITNESS LONGENECKER:

A. You said "major changes to the CRBR." There has been a change to the core design.

There have been other evolutionary changes since 1968. I don't think any of those, in the terms of major changes to the concept, I don't believe so. In the detail, yes.

Q. On Page 38 of your testimony you state that, "Recent evaluations performed in the U.S. have indicated no clear superiority of one system over the other," referring to pool type or loop type.

Can you tell me -- second sentence from the bottom.

Can you tell me who performed these recent evaluations?

BY WITNESS LONGENECKER:

A. There have been a number of them. The one with which I am most familiar is the one that was done in evaluating the concept that should be pursued for the Large Developmental Plant.

That evaluation was performed by Rockwell International, General Electric Advanced Reactor System Division, Westinghouse Advanced Reactor Division, Babcock & Wilcox Company, Combustion Engineering, Bechtel National Corporation, Burns & Rowe, and Stone & Webster.

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Q. On Page 39 of your testimony, Answer 25, you state that, "There is a lack of large pool-type reactor construction experience in this country."

Can you tell me what the construction skills are that are peculiar to pool-type reactor construction?

BY WITNESS LONGENECKER:

A. I wouldn't say that they are peculiar. I would say that they are unique.

A pool-type reactor has a much larger vessel in diameter and typically larger components than does a loop type.

As a result of this, at various sites -- well, at any site, it does depend largely as to whether there is railroad or barge access to the site. There is a larger amount of field labor required to construct a pool-type reactor.

The best example is at Super Phenix where they built the vessel and many of the major components on site. You typically don't need to do that for a loop plant.

It is generally thought that the amountof field labor that must be executed, the amount of operations, particularly machining and welding of large components that must be done on site in some way typically is not done with the same speed and the same quality as is done in the shops.

So there is a substantial difference in the amount of field labor requirements, notably in the area of the nuclear steam supply system and its components.

Q. On Page 40 of your testimony, you state that, "With flywheels on the sodium pumps, the time for initiation of boiling would increase slightly."

Can you tell me how much that time would increase?

BY WITNESS LONGENECKER:

A. Dr. Anderson.

BY WITNESS ANDERSON:

A. It depends on how big a flywheel you put on.

Q. Can you give me an estimate?

BY WITNESS ANDERSON:

A. No. Can you give me an estimate of the flywheel size? I don't know, the bigger the flywheel, the longer the time.

Q. What I'm trying to find out -- well, what range of flywheels are in existence that you might consider using in the CRBR?

BY WITNESS ANDERSON:

A. You don't have to have a flywheel. You can build inertia into the motor, as we did on Clinch River.

Q. Yes. I'm asking you to explain for the record what types of flywheels are in existence today --

12-10

BY WITNESS ANDFRSON:

A. A flywheel could be anything from an ounce to a million pounds. It's arbitrary.

Q. Yes. If one were to use a flywheel in the Clinch River Breeder Reactor Plant, what range of size would one reasonably expect to include?

BY WITNESS ANDERSON:

A. The range of size one should include is that one which balances the need for coastdown with the need for limiting the transients imposed on the components in the design.

When we designed Clinch River, we considered putting a flywheel on and we found that we could build in the requisite inertia within the motor and didn't need a flywheel.

So anyplace in that range is satisfactory. The optimum is the one we chose in Clinch River.

Q. And in your testimony when you said, "The time for initiation of boilding would increase slightly," given the range that you've just mentioned, what would be the range of time that you would envision?

BY WITNESS ANDERSON:

A. I don't know the numbers.

Q. So you had no idea what you meant by "slightly"?

12-11

BY WITNESS ANDERSON:

A. No. I do have an idea. I just don't have the numbers for me.

Q. Well, can you explain it qualitatively?

BY WITNESS ANDERSON:

A. Surely. If you put on a huge flywheel, it will extend the time at which boiling occurs greatly.

If you put on a small flywheel, you extend the time at which boiling occur very little.

Q. Are you talking about a matter of hours?

BY WITNESS ANDERSON:

A. If you put on a very small one, it will be a small time; if you put on a big one, it will be a big time.

You can put on a flywheel which will give you hours, if you choose. Whether it's practical or not, I don't know, and I know it will make the transients worse.

BY WITNESS LONGENECKER:

A. Could I add to that? I believe there's a reference made previously to flywheels added to Super Phenix.

They did that as a means of balancing the inertia. As Dr. Anderson said, we do that by actually determining the proper size of the rotor on the pump.

In the case of Super Phenix that, as I recall, was designed to add seconds to the total coastdown of the

12-12

motor.

So I can't tell you what the time is to induce boiling, but the total added coastdown time that they are adding there is, to my best recollection, on the order of 60 to 90 seconds.

So we are not talking about minutes or hours.

Q Dr. Anderson, assuming that one used a flywheel that increased the total coastdown time by 60 to 90 seconds, can you tell me what increase in time to boiling that would correspond to?

BY WITNESS ANDERSON:

A. No.

Q Can anyone else?

BY WITNESS LONGENECKER:

A. No.

BY WITNESS KAUSHAL:

A. No.

Q On the bottom of Page 40 through Page 41, you state that:

"Increasing the pump inertia by means of a flywheel beyond that required to provide adequate coolant flow increases the rate of temperature change associated with system thermal transients."

Wouldn't it actually decrease the rate of

12-13

temperature increase, due to a loss of flow?

BY WITNESS ANDERSON:

A. Yes.

Q. On Answer 27 on Page 41, you refer to "power systems operating temperatures."

Dr. Anderson, does increasing the time to initiate boiling enhance the likelihood that the flow will be restored before boiling begins in a loss of flow accident?

BY WITNESS ANDERSON:

A. Would you repeat that, please, slowly?

Q. Does increasing the time to initiate boiling enhance the likelihood that flow will be restored before boiling begins --

BY WITNESS ANDERSON:

A. No.

Q. -- in a loss of flow HCDA?

BY WITNESS ANDERSON:

A. No.

Q. Would it increase that likelihood in any degree?

BY WITNESS ANDERSON:

A. No.

Q. Page 42 -- Page 44, you discuss the core-catcher alternative.

Isn't it true that the Super Phenix does contain a core catcher in its design?

BY WITNESS ANDERSON:

A. The kind of core catcher you are talking about is not included in the Super Phenix design.

Q. What kind of core catcher is included in the design of the Super Phenix?

BY WITNESS ANDERSON:

A. What do you mean by "core catcher"?

Q. Any kind of core retention device.

BY WITNESS ANDERSON:

A. There are places within the reactor vessel that have the capability of holding a portion of the core. I don't define that as a core catcher.

Q. Page 44 you state that:

"It should be noted that (a) the core catcher does not in any way reduce the likelihood of an HCDA and that (b) any active features provided in the core catcher have to perform in an extremely hostile environment subsequent to an HCDA and are inacceissible at a ti: when they are required to function."

Is your Point (b) not true of any other safety system in the breeder?

BY WITNESS LONGENECKER:

A. I don't know, can't say categorically. The point we're trying to make there is that we put our emphasis on preventing core melts, and we think that things like core catchers that can act only after to accommodate it in this type of hostile environment are not effective features of the plant.

That's why we have not included them.

Q. Wouldn't the core catcher rely less on active features than most other safety systems in the CRBR?

BY WITNESS LONGENECKER:

A. The systems that we have in CRBR are there to prevent accidents from occurring.

This is merely a catch pan that's there after the accident has occurred, and as I say, we don't believe that that's an effective approach to plant design.

Q. Would the core catcher rely less on active features, since it is just a catch pan?

BY WITNESS LONGENECKER:

A. I don't know.

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WITNESS LONGENECKER: It would depend on the specific core catcher design.

BY MS. FINAMORE:

Q. But isn't the core catcher primarily a passive system?

BY WITNESS LONGENECKER:

A. I don't know. If you could give me a specific design Catch pans are not inherently passive. Some are cooled. Some have different active devices in them.

A core catcher is not an inherently passive device.

Q. Isn't it more passive in any type of design than the other systems you have for mitigating accident conditions of the CDA?

BY WITNESS LONGENECKER:

A. What other systems?

Q. TMBDB and SMBDB.

BY WITNESS LONGENECKER:

A. Those aren't systems. Those are acronyms for accident sequences.

Q. The systems designed to prevent those accident sequences from having high consequences, such as event-purge system?

BY WITNESS LONGENECKER:

A. No.

Q. Do you believe then that the core catcher in any design is more of an active system than the containment mitigation systems you've got in the design?

BY WITNESS LONGENECKER:

A. No. If you could specify for me a specific core catcher design, I could do that comparison. I can't in the abstract.

Q. Isn't it possible --

JUDGE MILLER: I suggest that we're spending a lot of time quibbling, and that we're going to have to cut short your time.

I would think you could make a more profitable use of it.

BY MS. FINAMORE:

Q. Wouldn't the inclusion of a core catcher reduce the consequences of a CDA once it occurred?

BY WITNESS LONGENECKER:

A. I don't know.

Q. Isn't it possible that such would be the case?

BY WITNESS LONGENECKER:

A. Anything --

JUDGE MILLER: That's a meaningless question. We're not into some philosophical possibilities. Let's get beyond that.

Let's get -- Let's conclude on a substantive basis what you're want to go into, because you're almost through.

BY MS. FINAMORE:

Q. Assuming that one has a HCDA Class 4, as described by Applicants, in which there is cracking of the concrete base mat, in such a situation wouldn't the inclusion of a core catcher reduce the consequences of a CDA, if it functions as designed?

BY WITNESS LONGENECKER:

A. No. Under your hypothesis that it has already reached the base mat, it would, by definition, have already penetrated any device that you would deem to be a core catcher.

So, in that sequence, no.

Q. Wouldn't the core catcher aid you in avoiding the reaching of that portion of the accident?

BY WITNESS LONGENECKER:

A. I don't know without looking at the specific design again. If one can postulate that you've gotten that far, you have melted the whole core, it has melted the vessel, it has melted the guard vessel, it has melted the cell liner -- I don't know what would lead me to believe hypothetically that it would not also melt through this core catcher, which you have hypothesized.

So I can't say that that would give me -- that adding that one additional hypothetical barrier would give me any substantial mitigation.

Q. Have you analyzed the impact of a core catcher on NCDA, Classes 1, 2 or 3 for the Applicants?

BY WITNESS LONGENECKER:

A. I don't know. I have not personally.

Q. Have the Applicants?

BY WITNESS LONGENECKER:

A. I don't know.

Q. Have you included any such analysis in your testimony on Contentions 7(a) or (b)?

BY WITNESS LONGENECKER:

A. No.

There is no discussion in my testimony, obviously, of core melt accidents or analyses in the details of core catchers for Classes 1, 2 or 3.

Q. Isn't it true that the Applicants at one time did analyze a core catcher --

BY WITNESS LONGENECKER:

A. I don't know.

Q. -- as part of the design?

BY WITNESS LONGENECKER:

A. I don't know.

Q. Are you familiar with the parallel design of the Applicants?

BY WITNESS LONGENECKER:

A. Yes. I'm familiar with the parallel design of the Applicants.

Q. Doesn't that include a core catcher?

BY WITNESS LONGENECKER:

A. I don't know. Again --

JUDGE MILLER: I thought we sustained an objection to that two hours ago.

WITNESS LONGENECKER: My reason for saying I don't know --

JUDGE MILLER: Stop, stop.

WITNESS LONGENECKER: I'm sorry.

BY MS. FINAMORE:

Q. At Page 44 of your testimony regarding the no-vent containment, you state on Page 45 that in the event that any significant radioactivity levels are detected in the containment effluent, the containment atmosphere is isolated through the use of containment isolation valves.

Are these valves active or passive features?

BY WITNESS LONGENECKER:

A. Dr. Kaushal.

BY WITNESS KAUSHAL:

A. These are active features.

Q. Do they close automatically or must they be activated by operator action?

BY WITNESS KAUSHAL:

A. They close automatically.

Q. You state on the bottom of Page 45 that elimination of venting during normal operation makes containment access difficult. Have you weighed the advantages of -- that no-venting or the venting situation against the difficulties you've mentioned earlier regarding maintainability during a no-vent situation?

BY WITNESS KAUSHAL:

A. Would you repeat that question? I couldn't quite understand it.

Q. Okay.

In analyzing the costs and benefits of a no-vent containment, have you weighed the advantages, in terms of dose consequences, of a no-vent containment against the difficulties in repair of such a containment that you mentioned earlier?

BY WITNESS KAUSHAL:

A. There are no particular advantages to a no-vent containment in this case, as our testimony states.

Q. There are no differences in dose consequences resulting from an assumption of no venting?

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BY WITNESS KAUSHAL:

A. The dose guidelines are met for the containment as designed.

Q. But wouldn't the doses change, depending on whether or not you include venting of the containment, as indicated in your testimony?

BY WITNESS KAUSHAL:

A. I don't believe so.

Q. Are you sure?

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BY WITNESS KAUSHAL:

A. I stated that I don't believe so.

Q. Does anyone else know?

(No response.)

Q. Mr. Longenecker, are you aware whether -- Isn't it true that in the Applicants' own analysis of dose consequences, the doses from a vented containment are much higher than the doses from a non-vented containment?

BY WITNESS LONGENECKER:

A. I'm not familiar with those analyses.

Q. Are you, Dr. Anderson?

BY WITNESS ANDERSON:

A. No.

Q. At Page 46 of your testimony, you state that "Design measures could be taken to increase the probability that no vent would be required. One cannot in practice, however, foresee all contingencies. Therefore, it is prudent and advantageous to include a filtered control vent capable to assure the containment integrity cannot be challenged."

Are you saying in this statement that the control vent capability would prevent challenge to the containment?

BY WITNESS LONGENECKER:

A. Dr. Kaushal.

BY WITNESS KAUSHAL:

A. That's correct.

Q. Are you saying that there would be no way that containment would fail because of the -- if controlled vent capability is included?

BY WITNESS KAUSHAL:

A. Yes, assuming that appropriate operator action would be taken, the containment will not fail.

Q. So if appropriate operator action is not taken, your statement would not be the same; is that correct?

BY WITNESS KAUSHAL:

A. You'll have to restate your premise. I don't understand it.

Q. Isn't it true that if the operator action was not appropriate, the containment integrity could be challenged?

BY WITNESS KAUSHAL:

A. If that's a theoretical possibility, yes, that could happen.

MS. FINAMORE: I have no further questions.

JUDGE MILLER: Thank you. Is there anything further?

MR. MIZUNO: Could we have a few moments, please?

JUDGE MILLER: Well, we're trying to utilize

13-10

all of our moments very carefully.

Dr. Hand, do you have any questions?

JUDGE HAND: No, thank you.

MR. MIZUNO: No more questions.

JUDGE MILLER: Judge Linenberger.

BOARD EXAMINATION

BY JUDGE LINENBERGER:

Q At one point, the panel made a distinction in its response to some question, between -- I think the panel -- I heard the panel make a distinction between project objectives and program objectives.

Could you explain very briefly what you were referring to in that distinction?

BY WITNESS LONGENECKER:

A. Yes, sir. In the context of the program there are objectives which are set forth in the initial portion of our testimony. What that describes is the objectives that the CRBR is expected to meet in its contribution to the total LMFBR program, which, as I've described before, does include demonstration plants, base program and attendant fuel cycle activities.

The project objectives to which I was referring are those which we established at the beginning on the project cooperatively with the utilities. The objectives, technically, economically and otherwise, that

Clinch River would meet in and of itself. Those are set forward in our key project documents.

We think we discussed the flow of those. Those flow down to all of our requirements to assure that we meet the project objectives.

Q. Do any of the program objectives fall within your responsibility?

BY WITNESS LONGENECKER:

A. Yes, sir, they do.

Q. I guess I'm a little -- more than a little bit interested in one aspect of things here. It is frequently the case that the better one tries to make a product, the less useful it becomes.

It seems to me that a major overriding consideration with respect to Clinch River is, first, whether it will or will not achieve a breeding gain of some worthwhile amount -- and I won't define "worthwhile" right now.

I have seen or heard nothing in what you gentlemen have discussed, either in your testimony or orally, that indicates whenever you try to optimize system design or component feature designs or whatever, that you or somebody has immediately jumped in and said, "Well, how much worse does this make the breeding gain?

Now who does this? Are you gentlemen responsible for that sort of thing? If not, who is? Can you talk about this for a little bit, please?

BY WITNESS LONGENECKER:

A. Yes, sir. We are -- In the aggregate, from different perspectives, we have each participated in the review and evaluation of the technical features and the overall ability of CRBR to meet its project objectives.

I would begin by saying that in the initial guidelines that were established for the project, which are contained in the initial contracts, there was -- It is included. Reading gain is as one of the overall project design guidelines. So it is included there.

It has been evaluated continually over the past. It has certainly been considered in the design of the cor and in the design and performance of the fuel.

We, in our testimony and in some of the other discussions where we are describing how we will meet the objectives, specifically that of demonstrating its viability as a central station power producer, have emphasized the other features: heat transfer, electricity generation, reliability and availability, those aspects, rather than the breeding gain, merely because we feel that the other work that we have done at the various test facilities that we have give us high confidence that

13-13

an adequate breeding ratio meeting the design guidelines can be met.

We believe that from a utility perspective the particular aspect of the project which we are trying to demonstrate is the reliability of this machine operating in a utility environment as large as a central station power producer.

In our testimony that's why we have emphasized it. It does not, in any way, diminish that importance as a project objective or guideline.

BY WITNESS ANDERSON:

A. May I add to that?

Q. Please

BY WITNESS ANDERSON:

A. In the original design of the Clinch River Breeder Reactor -- the original core design, we had planned to use light water reactor discharged plutonium, which would have given us a very good breeding ratio.

Later, it became apparent that that was not going to be available on the right time scale; and we had to use higher Pu 239 isotopes -- plutonium.

That led to a lower breeding ratio than we had set as a guideline on the project, and we sought many ways of improving that.

One of the most important attributes of the

13-14

heterogenous core -- not the only one -- was improvement of the breeding ratio in the reactor.

By putting in the heterogenous core, we did indeed get our breeding ratio. So we did look continually at the breeding ratio.

Another aspect of that that affects the design of the plant is that, in addition to breeding gain, we have to look at the doubling time of the plant. That involves the power density of the reactor, and that, in turn, affects the flow rate through the reactor which affects the pump design.

The pump design is a relatively high-head pump, in order to insure that it will meet the doubling time of the plant.

So throughout the design -- not only the core, but also the plant components, we have to look at features that would insure that we meet the breeding gain and doubling time desired for the plant.

Q. What is it that has given you confidence that the design parameters that you have adopted, such as the ones you've just been discussing, will indeed enable the system to achieve a desirable breeding gain and doubling time?

BY WITNESS ANDERSON:

A. The physics test that we performed in the

13-15

zero power plutonium reactor mocked up exactly the isotopic densities of all of the materials in the reactor, its size and verified the breeding gain that can be expected on a physics basis; that is, the number of atoms produced per atom destroyed.

In addition to that, we've done a lot of testing in EBR-2 of the fuels to be used in the plant. Knowing that we can get the power density from the fuel and that we can get the right physics from the fuel itself, we have very high confidence that we can meet the breeding gain, power density and doubling time of the plant.

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14-1

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Q Your answer to Question 15, which starts at the bottom of Page 25 and is completed the upper half of Page 26 your prefiled testimony, addresses the -- in a sense, the conservation of non-renewable resources, and yet it is not at all clear from the words used in the answer that breeding is an objective at all, and it's not at all clear from the words used in the answer what good the Uranium-238 is doing, unless one assumes something which is not said, namely that the whole point of the things is that it breeds.

Isn't the conservation of non-renewable resources rather intimately tied to the question of breeding?

BY WITNESS LONGENECKER:

A. Yes, sir, it is. We in the answer are perhaps assuming -- again, the answer assumes that breeding will occur in accordance with the design guidelines.

What we are making here is rather the straightforward statement that the use of an otherwise unused isotope of uranium, Uranium-238, in the reactor to be used to breed plutonium to be used as a future fuel thus extends the U.S. resource base, and to the degree that it is used would prevent us from needing to burn other exhaustible resources such as coal or oil.

14-2

It certainly -- Breeding in this instance, though, is certainly an important element of proving this objective.

Q While we are on the subject, let me ask -- again, I care not who answers -- what breeding gain and doubling time is currently considered to be achievable through the present Clinch River Breeder design approach?

BY WITNESS ANDERSON:

A. The breeding ratio we believe we will achieve is 1.2, and I'm weak on the doubling time. I think it's 30 years. Is that right; do you know?

BY WITNESS KAUSHAL:

A. That's approximately correct. I don't know the exact number, and the breeding ratio is substantially greater than 1.2.

Q Greater than 1.2?

BY WITNESS KAUSHAL:

A. Yes, sir.

Q Well, then, it --

BY WITNESS LONGENECKER:

A. I believe the distinction is you had asked the question as to what is achievable. We do have a plan, as we go from first core to future cores, have the option to go with higher performance in some circumstances.

I believe -- I'll let Dr. Anderson speak. I

14-3

believe he was addressing first core, and what Dr. Kaushal was speaking of was ultimately what you could achieve through some other combination of events of fuel features.

BY WITNESS ANDERSON:

A. I might also mention, if I may, that the same core design features in a larger plant lead to better performance in this area, because inherently there's less leakage of neutrons from a larger plant.

So with the same general design features you would get considerably better than a breeding ratio of 1.2 at a larger plant.

Q. What you are saying is considerations of scaling to a larger size are favoring an improvement in that performance?

BY WITNESS ANDERSON:

A. Right. Yes.

Q. Again, and I'm not sure which page it occurs on, but somewhere in here is a discussion of CRBR meeting an economic objective.

Perhaps you gentlemen can tell me faster than I can find it --

BY WITNESS LONGENECKER:

A. Yes, sir, that's Question 13 on Page 24.

Q. I again see a discussion in answer to that question that would -- could conceivably be satisfied by another kind of plant design. By "another kind," I mean something other than a lightwater reactor, that wasn't intended to breed at all.

Again, it seems to me that economic feasibility is to my way of thinking -- correct me if I'm wrong -- not divorceable from a consideration of whether the CRBR demonstrates that it can breed successfully, and yet none of this discussion touches on that point.

Are you saying that even if the breeding gain is 1.00, it's still going to achieve its economic objectives, or demonstration of economic feasibility?

BY WITNESS LONGENECKER:

A. Judge Linenberger, if I might try to put that in the context of the total program again, one of the elements, again, of the program is a fuel cycle development, going along with demonstration plants, which is Clinch River and the Base Tech Program.

We are assuming and have high confidence that we will meet our design guideline for breeding ratio.

The economics on the fuel cycle, of course, are highly dependent on other circumstances which are outside the control of Clinch River.

That has to do with the costs of building reprocessing plants, transportation, the operating modes therein.

The other element of the program, the fuel cycle development, will determine to a great degree the economics in that area.

We obviously do our part of that with the fuel that we deliver to them and the amount of plutonium that we breed; but since we are only one element of that, that demonstration in and of itself will come from another portion of the total program, that total demonstration.

Q It seems to me, however, that the ease with which one -- or the margin with which one might meet economic objectives is considerably widened -- or to put it another way, the task is made easier if the plant contributes some plutonium somewhere along the line as a result of the operation.

BY WITNESS LONGENECKER:

A. Yes, sir, and again I'll say we do have high confidence that we can meet the breeding ratio objective as stated in the design guidelines of 1.2, as Dr. Anderson said.

Similarly, the core and LMFBR cores do have the inherent physics capability of achieving higher breeding ratios, if necessary in the scheme of the over-all LMFBR deployment scheme.

We have high confidence in that based on the

14-6

work that we've done. We think this operation FFTF/EBR-II continued base program will demonstrate that.

My point was that as far as the portion of that which we control, we believe that we can do our adequate share to breed enough plutonium.

The questions that must be demonstrated are those other elements in the fuel cycle of the economics of physically separating it, putting it back into a usable product.

But those are really outside of the control of the project, per se.

Q. Not an immediately relevant point, but where is ZPPR located?

BY WITNESS LONGENECKER:

A. That's in Idaho, sir, the Idaho National Engineering Laboratory. It's in the vicinity of EBR-II.

Q. Where are the plant models and mockups that are discussed and pictured in your testimony located?

BY WITNESS LONGENECKER:

A. I believe you are referring to the ones we discussed -- there are several models and mockups of the plant.

I believe the ones we referred to in the testimony, Judge Linenberger, particularly on Page 22, the ones we discussed regarding maintainability, the main model which is shown in Figure 2 is located at Burns & Rowe in Oradale, New Jersey.

That is a whole plant model, as it's discussed there.

Q What about the mockup shown on Figure 3?

BY WITNESS LONGENECKER:

A. Dr. Kaushal.

BY WITNESS KAUSHAL:

A. The mockup was made at Walsh Mill, the Westinghouse Advanced Reactor Division, Walsh Mill.

BY WITNESS ANDERSON:

A. If I may add something, the illustration Figure 2 is the best we could get our picture, but the model is immense, but the model fills a whole room and is well worth seeing if you ever have a chance to go look at it.

Q Very good.

On Page 27 in the middle of the page, contained in answer to Question 17 is a statement about size of extrapolation factors that are considered prudent -- a prudent compromise between technology advancement and risk of scale up.

Are the numbers 2.5 to 3.5 the result of somebody's seat-of-the-pants judgment, or are they real hard analyses of scaling problems that went into that?

14-8

BY WITNESS LONGENECKER:

A. Judge Linenberger, I believe that there is a reasonable amount of analysis that goes into determining those factors, in particular the ones that we have evaluated, as we have looked at scale ups in foreign breeding programs, we have looked at scale ups which were taken in lightwater reactors.

We have looked at other technology, such as in gas turbines, jet engines, and things like that, and what we find is very roughly speaking in the range of two to thre[illegible] a half is a generally accepted scaling factor, and [illegible] more high technology applications is believed to be, without some other extenuating circumstance, in the reasonable range of engineering judgment to be a prudent scale-up step.

Q. Well, when you say "high technology applications," are you saying that same range of scale-up factors will attain for airplanes, jet planes, for example?

BY WITNESS LONGENECKER:

A. To the jet engines, I believe you will find some definite similarities. Can't make that with the weight and wing span and all the others that go with it, but by and large, for the jet engines, that is true.

Q. Page 35, in answer to Question 22, that answers a popularly asked question, but I don't think really says

14-9

why it is true.

A popularly asked question is: Why does the United States need to do something new and different in breeder technology, at least so far as newspaper and many periodicals are concerned in their coverage of breeder programs? The efforts of the French have been given a lot of attention, and I think it is reasonable to wonder why we should spend -- the U.S. should spend several billions of dollars to do something that maybe the money the French have already spent will do in an acceptable way, also.

Implicit in what I read into the answer of this question is that is not a good enough approach of the U.S. Can you shed light on that?

BY WITNESS LONGENECKER:

A. Well, Judge Linenberger, in answer to the specific Question 22, what we are trying to imply there is that the design of Clinch River and in fact the way the United States approaches designing any FRLMFBR plants is a very detailed integral process and each portion of that demonstrates the highest degree of confidence in the technology that we believe we can manifest in the particular part of the plant.

As such, when a design as involved as this one has almost 90 percent design completion, we can look at

14-10

other features in other plants which in and of themselves may have on the appearance to have some advantage, some slight advantage over the features we have.

When you look at incorporating a new feature into a total design, what we generally design is for that advantage one derives a number of attendant disadvantages for the system as a whole, and the description that we've laid out in here as to how we designed the plant, starting from the over-all requirements to the OPDD to the system level to the subsystems to the components to the materials is a well-integrated one.

By and large, we believe on balance that we have what is the optimum set of design features for Clinch River.

In looking at the French approach, they have taken a different approach to plant design. Theirs has been different in terms of the scale of the technology, the components that they incorporation; but by and large, when you look at the basic technology, it is by and large the same.

It is more like the -- the differences are more like the differences that one would notice between a Ford and a Chevrolet, and the debates as to how you go about designing each of those right down to the hood ornament, than it is some basic differences in the technology.

14-11

I personally believe that our plants have advantages over the French designs, and we find that in their support for our going forward with Clinch River, and their interest, their keen interest in some of our features, particularly heterogeneous core and a very intense interest on our development on steam generators.

So I would say it is recognition by both countries that we desire to have the capability, the domestic indigenous capability and somewhat different scientific approach to how we achieve obtaining an LMFRB capability.

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15-1

Q Should I infer that somewhere in this country there has been an effort of some sort that has looked at the French approach and decided that the LMFBR approach is more desirable?

From a technical point of view, now.

BY WITNESS LONGENECKER:

A. Yes.

Q Let's face it now. It's a way to spend money, sure, but that's -- let's not bless it on that basis.

Has there been an engineering evaluation that you know of that gives one reason to want to spend this money to do something in a different way, rather than adopt somebody else's approach?

BY WITNESS LONGENECKER:

A. There are a number of technical reasons. There are a number of reasons in the national interest for having our own indigenous energy supply, obviously.

But technically, we think design of our breeders and technology we develop is equivalent and in some cases, is superior to the French technology, which is the reason we have gone the way we have.

At the same time, say, in the context of the total program, not speaking just for Clinch River, but it's obviously, in my opinion, desirable to have our own indigenous supply, our own supply capability.

15-2

I would personally find it unacceptable to have to import LMFBR's as we've imported oil and other things through the past.

I believe we have a technological base in the industry that can perform as well or better, producing these, as can the French.

Q. Okay. So be it. I don't want to debate with you, but I would certainly note that the Japanese economy has done very well based on a starting point of importing technology.

I just hope we're not too proud to use somebody else's technology that can save us some money. Certainly, I'm not proposing that we import plutonium from France but I hope we're not too proud to make use of things they've done right.

BY WITNESS LONGENECKER:

A. No, sir, and I apologize.

I didn't mean to debate the point with you. I thought we were coming at it from the same angle.

I would point out that we do have with the French, the Japanese, the British, the Germans and others extensive technical interchange programs and we've learned some things from them.

They have test faciities like we do. Similarly, they have learned some things from us but I'd

15-3

say in the aggregate, our opinion is that CRBR is an adequate product and, in fact, our technology as a whole I believe is -- has some obvious advantages to theirs.

By and large, they have taken a different approach than we have. We have a very extensive, thorough research and development program. They have a much more plant oriented program.

They construct plants and they do their -- get their experience from operation but I wouldn't say that the differences of approach in any way denigrates the effectiveness of our technology.

Q On Page 40, there is a discussion concerning transient overpower and loss of flow events and in other testimony and discussions we've had in the recent past, with respect to such events, the term is usually -- has been preceded by the word "unprotected", transient overpower. "Unprotected" loss of flow.

I'm just curious. Are you talking about something different here than as represented by the terminology "uprotected TOP" or "UTOP" and"ULOF"?

BY WITNESS LONGENECKER:

A. Could I have Dr. Kashual address that, please?

BY WITNESS KASHUAL:

A. I believe these are the same things, YOur Honor.

15-4

If you notice on about the fifth or sixth line:

"For the postulated transient overpower events that assume failure of both reactor shutdown systems --"

By protection, that's what it means. That neither of the reactor shutdown systems operate in conjunction with the transient overpower.

Q. All right, sir.

I'm close to winding up here but, getting back to the core catcher concept for just a moment, on Page 44 and your statement (b) there, at the top of Page 44, I heard your words in answer to the question by Intervenors' Counsel.

They didn't tell me, though, how or whether anyone has looked at the possibility of putting a refractory material of some kind, fabricated into a cup or bucket or dish or whatever underneath this whole thing to further reduce the possibility that if something drains out the bottom of the concrete foundation path and gets underground water or does something bad; now, has anybody really done some kind of analysis with respect to utilizing just a passive refractory dish to --

BY WITNESS ANDERSON:

A. Yeah.

Q. And I haven't heard why that's so objectionable. I don't -- the statement following (b) doesn't explain that.

BY WITNESS ANDERSON:

A. We have looked at that and other nations have looked at it as well.

The rub is that given the probability of the event, the addition of that device doesn't change the probability of consequences very much because, to the same degree of conservatism you assume the event occurs, you cannot prove that it won't eat through that passive core catcher.

So the next thing you do to increase the reliability of not eating the core catcher then, is you add cooling and then you've got the problem of relying on the cooling.

But I won't get into that stage.

Staying with it as a passive unit, we haven't been able to convince ourselves that the debris and sodium which gets into the core catcher wouldn't also penetrate it.

For example, if you make it in blocks, it can go in the cracks between the blocks.

If you cement it up, it can eat through the cement.

5-6

If you make it a powder, it can float the powder.

We have looked at these. The Germans have looked at it, as someone has said in the proceeding some place, I'm sure. The SNR-300 plant in Germany is going to put in a core catcher but not because the designers feel it adds to the health and safety of public but because they are being ordered to put it in.

Now, that's still up in the air. I don't know how that's going to come down.

But, basically, what we tried to do is put in features that would add to the protection of the health and safety of the public and we didn't see that the core catcher option added to that.

Q Okay.

Finally, the very last sentence of your pre-filed testimony on Page 47, the answer to Question 31.

It's not clear to me that the answer stated there implicitly embraces the consideration of program objectives.

In other words, I can read that as saying, "We'll look at individual CRBR features --", and that's the language that's used there, "features", "and we've satisifed ourself that each feature -- we can't replace any given feature with something potentially or

15-7

substantially better".

Now, was that statement meant to include that putting these features together, that we can't substantially make better, will give us a good shot at the program objectives? Or were you just looking strictly at engineering considerations of individual features?

BY WITNESS LONGENECKER:

A. Judge Linenberger, that one was intended primarily to be a summary statement on the evaluation of alternative design features.

I would say -- I believe it's contained in a prior portion of our testiony, in the first half of the testimony. We treat the ability to attain the program objectives. I do believe that could make -- clearly the statement that having evaluated all the alternatives, the ones mentioned here and others and we do mention in the testimony -- you've seen our design review evaluation sheet.

I think one of the entries in there is Have You Considered Design Alternatives? And we do that periodically.

We have over the life of the project.

I can say that no features that have been evaluated would be deemed by us to be a substantial

15-8

improvement in our ability to meet the program objectives.

Q One final question.

I believe one of you gentlemen testified that or stated that assuming some breeding gain accomplishment in the Clinch River design, whatever it is, will probably look better when you go to the LDP system, because of the advantages of larger size.

Now, some things in life are unfortunately non-linear. I can see perhaps if you had a breeding gain of 1.2 and you went to the LDP, it might be 1.3.

I can see also, however, that if you had breeding gain of 1.0 and you went to the LDP, it might still be 1.0. I don't know.

I'm asking you. Can one automatically depend on breeding improvement as you make the transition to LDP, even if Clinch River itself fails to have a breeding gain?

BY WITNESS ANDERSON:

A Yeah. The inherent layout of the reactor assemblies, the breeder blankets, the core fuel assemblies and control assemblies, and the materials are essentially replicated, as we went from the Clinch River design to the LDP design.

In other words, we had the same ratios of materials and essentially the same arrangement. So, the

15-9

only thing that really changes when we go to a larger plant, is that there's less surface area per unit volume.

And, thereby, fewer of the neutrons leak out of core to be absorbed in reflector or structural materials within the vessel, and, therefore, those neutrons there in the larger plant are available to enter into the chain reaction and increase the breeding ratio of the plant.

So, it's a fundamental fact of physics, that going to a larger breeder, given all other things constant, gives a better breeding ratio.

And even if you had a 1.0 in a small plant, go to the same layout of assemblies of a large plant would give you more than 1.0.

JUDGE LINENBERGER: Thank you.

I believe that's all the questions I have.

JUDGE MILLER: Thank you, gentlemen.

You may be excused.

(Witnesses excused.)

MR. EDGAR: I would like to offer Applicants Exhibit 58 at this time.

JUDGE MILLER: Any objections?

MR.FINAMORE: No objection.

JUDGE MILLER: Staff?

15-10

MR. SWANSON: No.

JUDGE MILLER: Very well.

EXhibit 58 of the Applicants will be admitted into evidence.

(Applicant Exhibit No. 58 was entered into evidence and made a part of the record and follows.)

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of UNITED STATES DEPARTMENT OF ENERGY PROJECT MANAGEMENT CORPORATION TENNESSEE VALLEY AUTHORITY (Clinch River Breeder Reactor Plant)	Docket No. 50-537

APPLICANTS' DIRECT TESTIMONY
CONCERNING NRDC CONTENTIONS 7a) and 7b)

Dated: November 1, 1982

Q.1. Please state your names and affiliations.

A.1. John R. Longenecker, Acting Director, Office of the Clinch River Breeder Reactor Plant (CRBRP) Project, Office of Breeder Reactor Programs, U.S. Department of Energy.

Carl A. Anderson, Jr., Project Manager, Large Plant Projects, Westinghouse Advanced Reactors Division.

Narinder N. Kaushal, Deputy Assistant Director for Engineering, Clinch River Breeder Reactor Plant Project.

Q.2. Have you prepared statements of your professional qualifications?

A.2. Yes. Copies are attached to this testimony.

Q.3. What subject matter does your testimony address?

A.3. This testimony addresses NRDC Contentions 7a) and b) which allege that adequate analyses of alternatives to the CRBRP have not been performed.[1] Specifically, NRDC contends that

> 7. Neither Applicants nor Staff have adequately analyzed the alternatives to the CRBRP for the following reasons:
>
> a) Neither Applicants nor Staff have adequately demonstrated that the CRBRP 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

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[1] Contention 7 c) (Site Selection) is the subject of separate testimony.

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.

Q.4. What will this testimony show in relation to NRDC Contentions 7a) and b)?

A.4. This testimony shows that:

1) the CRBRP will meet the objectives established for it in the LMFBR Program Final Environmental Impact Statement (FEIS) (Supplement to ERDA-1535) in a timely manner. (See Q/A 5-16)

2) completion of design, construction and operation of the CRBRP will provide a demonstration which will be relevant to the development of commercial LMFBR central station electric plants. (See Q/A 17-21)

3) the informational requirements of the LMFBR Program or of a demonstration-scale facility will not be substantially better satisfied by alternative design features such as are embodied in certain foreign breeder reactors. (See Q/A 22-31)

Q.5. What are the LMFBR Program objectives and timing for CRBRP?

A.5. The LMFBR Program objectives for CRBRP are set forth in the DOE Final Environmental Impact Statement (FEIS) on page 57 as follows:

- to demonstrate the technical performance, reliability, maintainability, safety, environmental acceptability, and economic feasibility of an LMFBR central station electric power plant in a utility environment;
- to confirm the value of this concept for conserving important nonrenewable natural resources.

In addition, the programmatic timing of the CRBRP has been established by the DOE FEIS and the record of decision as "as soon as possible."[2]

Q.6. How were the CRBRP objectives implemented in the design?

A.6. Rather than completing the plant design to the end and then subjecting it to a check to see if the Project objectives would be met, the Applicants have followed a systematic approach in which the basic objectives of the Project were made an integral part of the design at the outset. The design was then developed around these objectives and has been continually checked at each stage of the process to assure that they would be met.

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[2] See 47 Fed. Reg. 33771 (August 4, 1982).

The Project objectives quoted above have been translated into four tiers of requirements for the design. The first tier consists of a set of design guidelines which define the characteristics and criteria for the Project which, if implemented, were judged to be necessary and sufficient to assure meeting the Project objectives.

Examples of design characteristics include three loops, power level (approximately 1000 MWt) and design lifetime (30 years). Examples of design criteria include high availability, low containment leakage (less than 0.1% per day) and low refueling time (less than 20 days per year).

The design guidelines then flow down to the second tier requirements in the Overall Plant Design Description (OPDD). The OPDD defines the characteristics and criteria for the plant as a whole. The OPDD includes, for example, general design criteria, codes and standards (e.g., ASME, IEEE), overall availability requirements, and maintenance requirements. The OPDD also identifies the 56 plant systems and defines the scope of each.

The OPDD requirements flow down to the third tier requirements for each of the 56 system design descriptions (SDD). For each plant system, the SDD defines the system performance requirements, including the interface requirements between systems. These interface requirements include, for a given system, all interface requirements imposed on it by other systems and the specific interface requirements imposed on other systems by the given system. The SDD also provides a description of each system and the components within that system. Finally, the SDD describes operation, maintenance, and test outlines for the particular system.

The requirements of the SDD flow down to the fourth tier to form requirements for Equipment Specifications (E Specs). The E Specs establish detailed component design requirements necessary to meet the system performance requirements set forth in the SDD.

These five descending tiers of requirements--from the Project objectives, to design guidelines, to overall plant design descriptions, to system design descriptions, and finally, to equipment specifications--assure that the Project objectives are an integral part of each level of the design.

Q.7. How have the Applicants assured that the objectives implemented for the CRBRP will be met?

A.7. The Applicants have established systems for in-process measurement and control over all elements of the Project to assure that the objectives and requirements implementing those objectives will be met. The Applicants have established a formal system of management policies and requirements, including three major elements which are pertinent here. These are design reviews, configuration management, and quality assurance.

Formal design reviews must be conducted for all systems and major subsystems of the plant. The design reviews are conducted by teams of independent reviewers which, for any given system or subsystem, include all disciplines necessary for review of the technical subject at hand. The design review teams evaluate a given system or subsystem against the requirements of the SDD and OPDD and, if any deficiencies are noted, recommend actions to assure that these requirements are met. Figure 1 is a typical checklist employed by these design teams, which illustrates the scope of a design review.

Figure 1

RECOMMENDED
DESIGN REVIEW CHECK-LIST

DATE____________________

CONTRACTOR____________________
DESIGN ENGINEER____________________ SYSTEM/COMPONENT____________________
TYPE OF REVIEW ____________________ PO COG ENGINEER____________________

	YES	NO
1. Has the design been baselined? Date of baselined or expected__________________		
2. If not baselined, is design in conformity with: (a) CRBRP Plant Reference Design Report? (b) Demonstration Plant Guidelines? (c) System Design Description?		
3. Has the requirement of PSAR 1.1.3 concerning NRC Regulatory Guides been satisfied?		
4. Have all applicable RDT Standards been applied in accordance with Appendix B of the MPR?		
5. Have any new or supplemented standards been developed in accordance with RDT Standard F2-2?		
6. Does the design satisfy RDT Standard F2-2 for: (a) Design criteria? (b) Codes, standards and practices? (c) Engineering studies? (d) Parts, materials and processes? (e) Design descriptions? (f) Specifications, drawings and instructions? (g) Identifications? (h) Acceptance criteria? (i) Interface control?		
7. Does the design satisfy OPDD-10 requirements for: (a) Performance? (b) Safety? (c) Interfaces? (d) Limits of the individual system? (e) Overall plant layout?		
8. Has the contractor submitted Work Agreements and/or 189's as appropriate?		
9. Were they properly approved?		
10. Has the contractor prepared a SDD?		

Figure 1 (cont.)

	YES	NO
11. Does the SDD cover the following system requirements:		
(a) Overall?		
(b) Interface?		
(c) Safety?		
(d) Reliability?		
(e) Maintainability?		
(f) Operations?		
(g) Arrangement?		
(h) Design parameters?		
(i) Flow paths?		
12. Does the SDD provide definitive system related requirements for the generation of component specifications?		
13. Have specifications been developed for:		
(a) Configuration?		
(b) Arrangement?		
(c) Performance parameters?		
(d) Materials?		
(e) Processes?		
(f) Selection of parts?		
14. Does the SDD confirm that the design will meet all requirements?		
15. Are necessary inspection devices provided for in the SDD?		
16. Are the needs for maintenance and inspections adequately covered?		
17. In regards to safety, is the design for		
(a) Level 1?		
(b) Level 2?		
(c) Level 3?		
18. Does the design emphasize and enhance safety?		
19. Is the design adequately supported by studies, evaluations, analysis and/or calculations?		
20. Are the above documents available and have they been properly identified in the Engineers Data Book?		
21. Will the design provide reliability of operations as required by Section 10, MPR?		

Figure 1 (cont.)

	YES	NO
22. Will the design system/component be easily installed into the Plant of System?		
23. Do Design Layout Drawings indicate that E-Spec requirements will be met?		
24. Are all critical dimension identified?		
25. Is the approval procedure for any critical dimension properly defined?		
26. Have all interfaces been considered?		
27. Does the system/component require special sources of support, I.E. power, air or environment?		
28. Have these special sources of support been adequately defined?		
29. Has sufficient access been provided for necessary inservice inspection as required by the ASME Code Section XI, 3?		
30. Has the contractor developed procedures for forwarding design review reports as required by Section 3.1.7.1.2. of the MPR?		
31. Have trade-off studies or analyses been conducted for alternate designs?		
32. If so, were these studies justified and properly approved?		
33. Have adequate studies and analyses been conducted to justify fabricability? .		
34. Is the cost properly justified? If not, amplify.		
35. Is the schedule realistic? If not, amplify.		
36. Are adequate technical data available to support any changes proposed by the contractor?		
37. Can the same performance be achieved by utilizing an existing or off-the-shelf design?		
38. Can a cheaper material be used without degrading plant safety?		

Figure 1 (cont.)

	YES	NO
39. Can the system/component be fabricated cheaper:		
(a) At the plant?		
(b) At the site?		
(c) Within the building housing system/component?		
40. Have plans been developed for any special transportation required?		
41. Has a plan been developed for spares?		
42. Has the RM/A-E, as appropriate, established Design Requirements Baselines for contractors through E-Specs and ICD's?		
43. Have any ICD's been released based on "limited" vice "precise" data?		
44. Is the design engineer conversant with RDT Standard F1-2 (PREPARATION OF SYSTEM DESIGN DESCRIPTIONS)?		
45. Is the design engineer conversant with RDT Standard F2-2 (QUALITY ASSURANCE PROGRAM REQUIREMENTS)?		
46. Are all principal design data under formal control?		
47. Does the contractor have an adequate procedure for working level control of the design?		

At a minimum, design reviews are conducted for each system or major subsystem at each of three principal stages of design: (a) conceptual, (b) preliminary, and (c) final. At the conceptual design stage (at approximately 30% design completion), a review is conducted at the system or subsystem level. At the preliminary design stage (at roughly 60% design completion), a design review is conducted at the system, subsystem and component level. At the final design stage (when the design is essentially complete), the design is reviewed at the system, subsystem and component level to assure that all SDD and OPDD requirements are met.

In addition to the three stages of design review discussed above, special design reviews are conducted on an as-needed basis for key systems and subsystems, key components, and for purposes of systems integration. In terms of key systems and subsystems, examples include the core restraint system review and the heterogeneous core review. In terms of key components, examples of reviews include those conducted for the reactor vessel and the main sodium pumps. In terms of system integration reviews, examples include the availability review of the entire nuclear steam supply system, and the maintainability review conducted for the head access area.

The second major element used for in-process measurement and control to assure that the Project objectives will be met is configuration management. As results are obtained from a design review, the Project proceeds to "baseline" the design of that specific system. The Project has implemented a formal configuration management plan whereby specific elements of given system design are baselined, i.e., formally approved and established as the reference plant design.

At the conceptual design stage, the system requirements in the SDDs are baselined. At the preliminary ~~preminiary~~ design stage, the system descriptions in the SDDs are baselined. At the final design stage, the outlines for operations, maintenance and tests are baselined.

After a given system is baselined, any changes to that design require formal review and approval of an engineering change proposal. All engineering change proposals are reviewed to insure that a particular change satisfies the higher tier requirements established for that system.

The third major element, the Quality Assurance Program, assures that throughout the process of design, construction and operation, procedures are adhered to and that documentation is both traceable and complete. In addition, the Quality Assurance Program includes inspection of equipment to assure that the requirements of Equipment Specifications and SDDs are met.

In summary, through design reviews, configuration management, and quality assurance, the tiers of requirements which implement the CRBRP objectives are subjected to in-process management and control to insure that the objectives implemented for the CRBRP will be met.

Q.8. How will the CRBRP achieve its objective of demonstrating technical performance?

A.8. The actual testing and operation of the plant will demonstrate the achievement of the technical performance objective. The technical performance objective encompasses the technical parameters established for the Project. CRBRP will be designed, constructed, and operated to achieve the plant's technical parameters. The major parameters of interest here are: plant thermal power production, steam conditions, and electrical power production. The specific design characteristics of the plant provide a high degree of assurance that each of these parameters will be met.

Thermal power production is a function of core heat generation, core flow, and heat transport from the core in the heat transport system. The thermal power production parameter for the CRBRP is 975 MWt.

There is reasonable assurance that the CRBRP core will perform as expected. A series of experiments have been conducted at the Zero Power Plutonium Reactor (ZPPR) in Idaho using a CRBRP core configuration mock-up. In these experiments various core-physics related parameters were measured and were compared against calculated values to test the ability to predict nuclear power production in the core. The agreement between the calculated and measured values provides confidence that the calculational techniques are ~~vaild~~ valid and that the CRBRP core power distribution will be properly calculated.

It is reasonable to expect that the heat transport from the core will meet the thermal power production design parameters for CRBRP. The basic flow characteristics through the core have been determined by scale-model hydraulic tests. The analytical tools for calculating basic heat transfer from the core are well established through experience with the Experimental Breeder Reactor-II (EBR II), the Fast Flux Test Facility (FFTF), and light water reactors (LWRs).

Similarly, the overall heat transport system can be expected to meet the design parameters for plant thermal power production based upon experience from EBR II and FFTF. The major HTS components are sodium pumps and Intermediate Heat Exchangers. A prototype of the main sodium pump is currently being tested and has been found to perform satisfactorily to date. The Intermediate Heat Exchanger (IHX) is similar to the one sucessfully used in FFTF and can be reasonably expected to perform acceptably in CRBRP. Thus, the core heat generation, coolant flow through the core, and heat transport from the core is reasonably likely to meet the design parameter for thermal power production.

The design parameters of importance to the CRBRP steam conditions are pressure, temperature, and flow. The

CRBRP design parameters for steam conditions are: (1) pressure 1450 psi; (2) temperature 900° F; and (3) flow 3.3 millions pounds per hour. The steam, feedwater and condensate systems for CRBRP are similar to those currently in use in LWR's and fossil power plants, and the CRBRP conditions of pressure, temperature and flow fall within the range of parameters experienced for LWRs and fossil-fueled plants. The design of the CRBRP steam generator module has been verified by model and feature tests in both water and sodium. A full size prototype is currently being tested to assure that the plant units will meet the design parameters for steam conditions. Based upon the results of testing to date and the experience available for LWRs and fossil fueled plants, it is likely that the CRBRP will meet the design parameter for steam conditions.

The CRBRP design parameter for electrical power production is 350 MW electric. Electrical power production for CRBRP will be achieved through the use of a turbine generator which is similar to those currently in use in LWRs or fossil fired plants. The turbine will operate at conditions of temperature, pressure, and flow which fall within the range of parameters experienced for LWRs and fossil plants and, thus, is reasonably likely to meet the design parameter for electrical power production.

In summary, it is likely that CRBRP will achieve its design parameters for thermal power production, steam conditions, and electrical power production, and thus, meet the overall objective of technical performance.

Q.9. How will the CRBRP meet its reliability objective?

A.9. The CRBRP is being designed to function as a baseload unit, for which it is generally accepted to mean that it will be available from 60-90% of the time. The CRBRP has been designed to reach the baseload reliability of about 75% within the 5-year demonstration period. Rather than designing the plant and evaluating availability after the fact, the Applicants have made reliability analyses an integral part of the design process from the outset. The plant has been specifically engineered using these reliability analysis techniques to assure that the availability goal will be met. These analyses made use of an existing data base for the availability performance of similar components and systems. The CRBRP systems were subdivided into subsystems, and the subsystems were in turn subdivided into components. The availability of each CRBRP component or subsystem was then assessed using the existing data base. The CRBRP reliability assessment showed that the plant would meet its reliability goal. In addition, however, the assessment identified specific elements of the design which could be improved so that the Project's ability to meet the goal would be enhanced. Two specific examples illustrate this point. First, as a result of this review, it was determined that providing redundant heaters on the equalization lines would increase the plant availability by 1.6%.

Secondly, inclusion of certain piping changes would allow maintenance operations for one Radioactive Argon Processing System (RAPS) compressor while the other is operating, thereby increasing the plant availability by 0.6%. Action has been taken to implement both of these design modifications for the CRBRP.

Thus, specific analyses conducted for CRBRP reliability, along with the actions taken to enhance reliability, provide reasonable assurance that the CRBRP is likely to meet it reliability objectives.

Q.10. How will the CRBRP achieve its objective of maintainability?

A.10. Maintainability encompasses the ability of the plant operator to perform preventive and corrective maintenance on the plant with minimal adverse impact on the amount of time the plant is available for generation of electricity. The actual goals to be achieved in maintainability are constrained by requiring maintenance in a time frame that supports the plant reliability goal. The Clinch River Breeder Reactor Plant design includes specific features and requirements to enhance maintainability. Maintainability reviews are required parts of the design and design review process. The OPDD, SDD and E Specs establish specific maintainability requirements for CRBRP systems and components in successively greater detail. These requirements include:

- o All in-sodium components must be designed to drain freely of sodium so that, upon removal, liquid sodium does not freeze inside the components and thus complicate maintenance operations.
- o Major components must be either removable or repairable in place.
- o Ample space must be provided around all major equipment to assure ease of access for maintenance. In order to assure that this requirement would be met, the Applicants developed a detailed scale-model of the Clinch River Breeder Reactor Plant (one-half inch to one foot). This scale-model has been applied as an engineering tool in review of all equipment arrangements to assure that no unforeseen interferences would occur which could impact maintainability. (See Figure 2.)

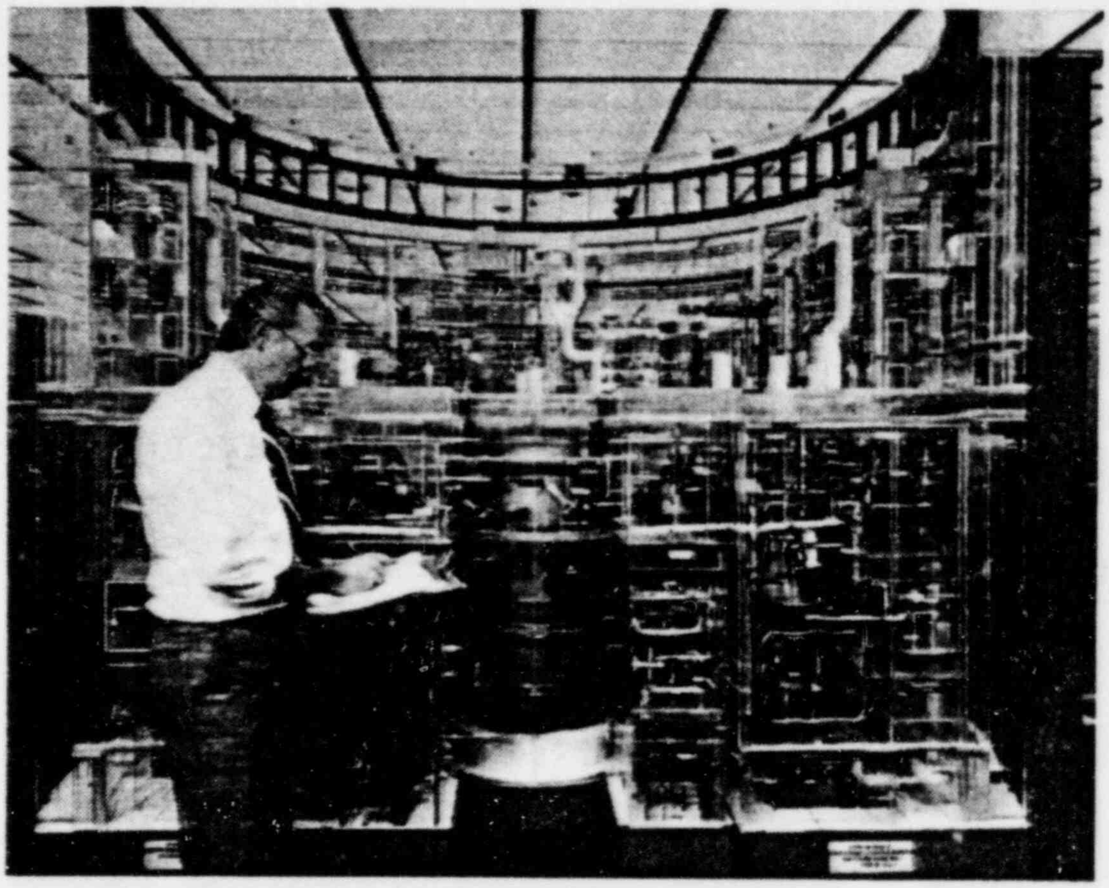

Figure 2

In specific areas of the design where maintenance operations are expected to be critical to meeting the availability objectives, detailed models were built to verify that maintenance operations could be performed satisfactorily. For example, the reactor head access area is a portion of the plant in which there is a relatively high density of equipment. In addition, during refueling operations, there are equipment movements (e.g., rotating plugs on the reactor closure head) in this area. These conditions required careful review to assure that maintenance operations can be satisfactorily accomplished. In order to ensure that this could be done, a full-scale mock-up of the reactor head access area was constructed and used by the reactor component and systems designers to ensure that necessary operations and maintenance activities could be accomplished in the reactor head access area. (See Figure 3.)

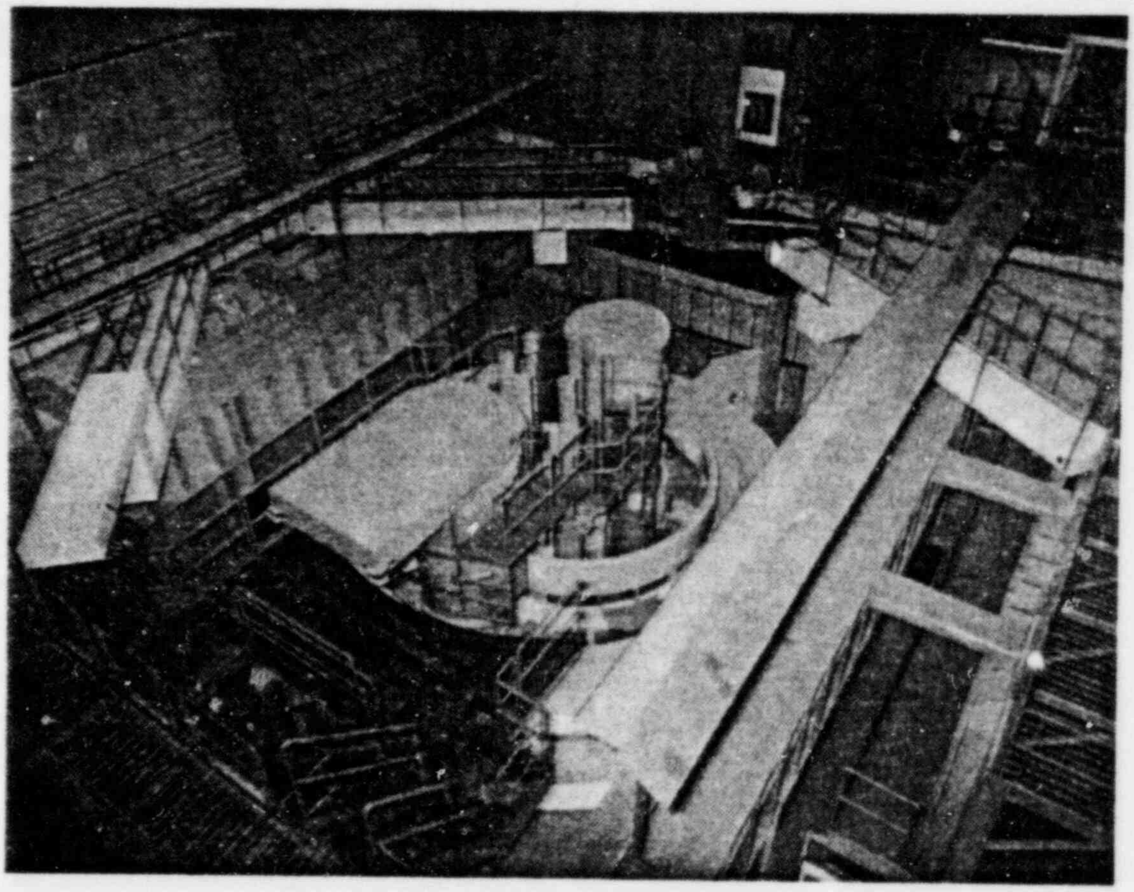

Figure 3

As a second example, the high density of equipment in the area surrounding the reactor head made it necessary to construct a full-scale mock-up of the secondary control rod drive mechanism so that the designers could simulate and fully characterize the actual maintenance operations anticipated for those components.

In summary, the systematic application of maintenance requirements, maintenance reviews, and specific scale modeling make it likely that the CRBRP will meet its maintainability objectives.

Q.11. How will the CRBRP design achieve its safety objective?

A.11. The demonstration of the safety objective of the Clinch River Breeder Reactor Plant will be achieved when the plant is licensed and operated within the limitations imposed by applicable regulations, guides, and instructions, while achieving the other objectives established for the Project. While the ultimate demonstration concerning this objective must await the Construction Permit and Operating License proceedings and completion of the demonstration period, the present record suggests that it is reasonably likely that this objective will be met. The NRC Staff's June 1982 Site Suitability Report concluded that "...the proposed CRBRP site is suitable for a facility of the general size and type proposed from the standpoint of radiological health and safety considerations." In addition, the NRC Staff's February 1977 Final Environmental Statement concluded that

"...it is within the state-of-the-art to design, construct and operate the CRBRP in such a manner that the consequences of accidents will not be significantly different from those already assessed for LWRs." The NRC Staff's July 1982 Draft Supplement to the Final Environmental Statement does not alter this conclusion. The Applicants' testimony concerning NRDC Contentions 1, 2, and 3 confirms this FES conclusion.

Q.12. How will the CRBRP achieve its objective of demonstrating environmental acceptability?

A.12. The CRBRP will achieve its objective of environmental acceptability by conducting construction and operation in conformance with applicable Federal and State environmental regulations. The CRBRP will satisfy all applicable Federal and State regulatory requirements (see Chapter 12.0 of Applicants Environmental Report). The NRC Staff's Final Environmental Statement concluded that the environmental impacts of construction and operation were acceptable. Thus, it is likely that the CRBRP will meet the objective of environmental acceptability.

Q.13. How will the CRBRP achieve its objective of demonstrating economic feasibility?

A.13. The economic feasibility objective will be achieved by developing comprehensive cost, material quantities, and performance information for the CRBRP, thus providing a data base from which one could extrapolate to commercial-size central station power plants. The cost data base will include planned and actual costs for all elements of the project including design, construction, and plant operation. The project has established a system for compiling this comprehensive cost information in a form which permits cost analysis and evaluation for all the plant elements at a very detailed level. Examples of materials quantities data include length of piping, length of electrical cabling, and volumes of concrete in the various plant structures. This CRBRP data is currently being used in development of the LDP cost estimate. In the future, the cost and performance data established for the CRBRP can be used to project the cost and economics of other future LMFBR plants. Thus, the CRBRP is reasonably likely to meet the objective of demonstrating economic feasibility.

Q.14. How will the CRBRP achieve its objective of operating the plant in a utility environment?

A.14. The objective of operating the Clinch River Breeder Reactor Plant in a utility environment will be met by operation on the Tennessee Valley Authority (TVA) system, supplying power to that grid, while being operated by personnel of TVA.

Q.15. How will the CRBRP achieve its objective of confirming the value of the LMFBR in conserving non-renewable resources?

A.15. The objective of confirming the value of the LMFBR in conserving important non-renewable natural resources will be demonstrated by the plant's ability to generate electricity utilizing available uranium resources, including the otherwise unused U-238. The only currently available resources for central station generation of electricity are hydroelectric, natural gas, oil, coal and uranium-235. A large share of U.S. central station electrical generating capability now uses non-renewable natural resources--oil, natural gas, coal, uranium U-235--and hydroelectric power is limited. The CRBRP will demonstrate the ability to generate electricity utilizing an otherwise unusuable natural resource--uranium-238. Thus, operation of the CRBRP will meet the objective of confirming the value of the LMFBR concept for conserving important non-renewable resources.

Q.16. How will CRBRP achieve its objectives in a timely manner?

A.16. The programmatic timing of CRBRP contemplates completion of CRBRP as soon as possible. Consistent with satisfaction of all other Project objectives, and the exercise of all lawful means to that end, the Applicants are committed to take all actions necessary to complete CRBRP as soon as possible. Project research and development is approximately 97% complete, and the design is approximately 87% complete. Seventy percent of the hardware is on order or delivered. Site preparation activities have commenced. The NRC Staff has issued its Site Suitability Report and Final Environmental Statement for the Project. On the basis of these factors, it is likely that CRBRP will meet its objectives in a timely manner.

Q.17. How were the type and size of the CRBRP configuration established?

A.17. The type is by definition an LMFBR. The overall configuration of the plant, after intensive reviews, was selected as a loop-type plant. The discussion in Answer 25 below provides a summary comparison of the loop-type plant versus a pool-type plant.

The size, or the gross power rating (975 MWt, 325 MWt per loop), of the CRBRP was selected as a reasonable midpoint between FFTF (400 MWt or 133 MWt per loop) and commercial size reactors (2400-3800 MWt, 600-1270 MWt per loop). Extrapolations of size by a factor of 2.5 to 3.5 are considered to be a prudent compromise between the need for advancement in technology and keeping the scale up risks acceptably low. Development of LWR technology followed approximately the same path. Foreign LMFBR programs have utilized similar extrapolation factors. The information obtained from a plant of the size of CRBRP is relevant to a commercial size reactor in that a similar extrapolation of the technological base from the CRBRP would lead to a commercial size LMFBR.

Q.18. How is the CRBRP design relevant to the development of commercial size LMFBRs?

A.18. The next plant under development by DOE and U.S. electric utilities and private industry is the Large Developmental

Plant (LDP). This is a 1000 MWe or 2550 MWt plant. Note that the LDP size extrapolation from CRBRP is similar to the extrapolation from CRBRP to FFTF. This extrapolation factor for LDP was established after an intensive interaction and analysis by the industry and DOE based on balancing considerations of advancements in technology and attaining a low risk basic design. The overall configuration of the plant is again "loop-type," established after studies and analyses conducted independent of CRBRP. Furthermore, based on the concept already developed for LDP, an assessment was made by DOE and the industry on the bases available for the design of LDP systems. Table 1 shows the results of this assessment. As can be seen from this table, CRBRP systems design provides a basis for all the LDP systems designs.

TABLE 1

EXTRAPOLATION BASE FOR LDP

LMFBR SYSTEMS

System No.	Title	Experience Base CRBRP	FFTF	LWR
11	Power Transmission	X	X	X
12	Building Electrical	X	X	X
13	Grounding & Cathodic Protection	X	X	X
15	Communication	X	X	X
16	Lighting	X	X	X
19	Site Improvements	X	X	X
20	Balance of Plant (BOP) Building	X		X
21	Reactor Support Building	X	X	
22	Compressed Gas	X	X	
23	Auxiliary Coolant	X	X	
24	Radioactive Waste	X	X	X
25	Heating, Ventilation, and Air Conditioning (HVAC)	X	X	X
26	Plant Fire Protection	X	X	*
27	Reactor Containment	X	X	
28	Recirculation Gas Cooling	X	X	
31	Reactor System	X	X	
32	Reactor Enclosure	X	X	
41	Reactor Refueling	X	X	

TABLE 1 (Continued)

EXTRAPOLATION BASE FOR LDP

LMFBR SYSTEMS

System No.	Title	Experience Base CRBRP	FFTF	LWR
44	Maintenance (Nuclear Island)	X	X	
45	Maintenance (BOP)	X		X
51	Reactor Heat Transport	X	X	
52	Steam Gen. Aux. Heat Removal	X		
53	Steam Generator	X		
54	Recirculating Gas Instrumentation	X	X	
55	Reactor Containment Instrumentation	X	X	
56	Reactor Heat Transport Instrumentation	X	*	
57	Auxiliary Coolant Fluid Instrumentation	X	X	
58	Radioactive Waste Instrumentation	X	X	X
59	HVAC Instrumention	X	X	X
60	Plant Fire Protection Instrumentation	X	X	*
61	Inert Gas Receiving & Processing Instrumentation System	X	X	

TABLE 1 (Continued)

EXTRAPOLATION BASE FOR LDP

LMFBR SYSTEMS

System No.	Title	Experience Base CRBRP	FFTF	LWR
62	Impurity Monitoring & Analysis Instrumentation System	X	X	
63	Aux. Liquid Metal Instrumentation System	X	X	
64	Reactor Refueling Instrumentation System	X	X	
66	Leak Detection Instrumentation System	X	X	
67	Plant Annunciator System	X	X	X
68	Piping & Equipment Electrical Heating and Control System	X	X	
69	BOP Instrumentation & Control	X		X
71	Feedwater & Condensate System	X		X
72	Main and Auxiliary Steam System	X		X
73	Heat Rejection System	X		X
74	River Water Service System	X		X
75	Treated Water System	X		X
76	Waste Water Treatment	X		X

TABLE 1 (Continued)

EXTRAPOLATION BASE FOR LDP

LMFBR SYSTEMS

System No.	Title	Experience Base CRBRP	FFTF	LWR
81	Auxiliary Liquid Metal	X	X	
82	Inert Gas Receiving and Processing	X	X	
85	Impurity Monitoring and Analysis	X	X	
90	Plant Control System	X	X	X
91	Data Handling and Display	X	X	X
92	Reactor and Vessel Instrumentation	X	X	
94	Fuel Failure Monitoring	X	X	
95	Flux Monitoring	X	X	X
96	Radiation Monitoring	X	X	X
97	Site Investigation	X	X	X
98	Construction Facilities Equipment & Services	X	X	X
99	Plant Protection	X	X	X
TOTAL NUMBER OF SYSTEMS		56	44	29

X Direct data base
* Partial data base

In addition to similarities at the system level, there are strong similarities between CRBRP and LDP at the subsystem and component level. For example, in the reactor system the LDP reactor core is of heterogeneous design, as is the CRBRP core. The fuel material, structural material, fuel assemblies, blanket assemblies, shield assemblies, control assemblies, control rod drive mechanisms, upper internals structure, core restraint, instrumentation, reactor head and shielding are essentially identical. Thus, in the case of the independent effort to develop the design for the LDP, which is essentially of commercial size, CRBRP provided much relevant information.

In the same manner that a large portion of the information obtained from CRBRP is directly relevant to LDP, the information from the design, construction, and operation of CRBRP can also be reasonably expected to provide significant information of relevance to commercial LMFBRs of the future.

Q.19. What is the relevance of information generated by CRBRP which is independent of the specific design?

A.19. A significant contribution of the CRBRP to the overall LMFBR program is development of a strong base of technological information. This technological base encompasses the sum-total of the experience with the design, construction, and operation of the CRBRP. Examples of this technological base would be information concerning materials properties, analytical methods (e.g., thermal hydraulic analysis codes) and the associated data bases. This technological base then forms the foundation for the next step in the development of technology. The process continues, ultimately leading to a final product, in this case a commercial size breeder plant with the desired characteristics. In this context, even experience which leads to rejection of certain design concepts is relevant inasmuch as the rejection of a design concept is based on prior knowledge and experience. CRBRP will provide substantial information which is independent of the specific design concerning materials, properties, analytical methods, and design studies which will be of substantial value to future LMFBRs.

Q.20. How will the CRBRP core design be relevant to core design in commercial size LMFBRs?

A.20. The heterogeneous core configuration as used in CRBRP is expected to be adopted in future LMFBRs. The design of the core assemblies, blanket assemblies, shield assemblies, and control assemblies is not expected to change radically. The core restraint is expected to be similar. Most importantly, the methodology developed for heterogeneous core analysis will be directly applicable to design of larger LMFBRs.

As previously noted, extensive tests of the CRBRP heterogeneous core configuration were conducted at the Zero Power Plutonium Reactor (ZPPR). These tests provided valuable feedback on the validity of analytical tools. As a result of this experience, the core design of ~~on~~ the LDP and larger LMFBRs can proceed with a higher degree of confidence.

Q.21. How will CRBRP engineered safety features be relevant to commercial size LMFBRs?

A.21. The major engineered safety features (ESFs) in CRBRP, such as reactor containment, the liners in the cells containing sodium piping, features to mitigate the effects of sodium spills and fires are all relevant to larger or commercial LMFBRs. The types of events against which these ESF's must be designed are characteristic of the LMFBR, regardless of size. Design, construction, testing, and operation of these engineeering safety features will, as indicated earlier, demonstrate the acceptability of these features and provide relevant information for future LMFBRs.

Q.22. Has consideration been given to whether the information objectives of the CRBRP might be substantially better satisfied by design features found in other reactors?

A.22. Yes. There are no design features which have been identified in either the U.S. LMFBR Program or in the designs utilized in foreign programs which are substantially better alternatives for satisfying Project objectives than the design features that have been incorporated in the CRBRP.

Q.23. How were the basic design characteristics for CRBRP initially chosen?

A.23. The design characteristics of CRBRP were the product of intensive review and assessment and analysis of the LMFBR program needs. This process established the design objectives and guidelines, as well as the more detailed top-level design requirements. The basic plant features, concepts and parameters had their genesis in conceptual design studies performed by Atomics International, General Electric, and Westinghouse, each teamed with a utility and an architect-engineer, during the Project Definition Phase (PDP) of the LMFBR program in 1968. The PDP was initiated to determine the major features of an LMFBR demonstration facility. Due to the competitive nature of the PDP and the availability of information from foreign breeder programs through international exchange agreements, the PDP designs (completed in 1971) encompassed all of the information then available to each of the participating reactor manaufacturers.

The designs proposed by the PDP studies, which also included consideration of on-going studies on the optimum features of commercial LMFBR's, were evaluated in detail in 1972 by two utility/industry advisory committees. This evaluation resulted in the formulation of the initial design characteristics.

Competitive designs and proposals responsive to the above-mentioned design characteristics were solicited by the Atomic Energy Commission (AEC) for the demonstration plant project. The proposals received were evaluated and features from each were factored into the basic design concepts of the CRBRP.

After selection of the Commonweath Edison/TVA proposal, trade-off studies were performed, including but not limited to plant size, primary loop configuration (pool vs. loop), primary pump location, refueling concept, number of heat transport loops, steam cycle selection, steam generator concept, and decay heat removal concept. The results from the studies were factored into the decisions made relative to specific Clinch River Breeder Reactor Plant features.

Thus, the major design ~~feature~~ features of CRBRP were the product of a systematic review and were responsive to the needs identified by the ultimate user--the utility industry.

Q.24. What design features, different from those contained in the CRBRP design, have been identified by NRDC as potentially advantageous?

A.24. These alternative design features are: (1) the pool-type primary system configuration, (2) use of flywheels on sodium pumps, (3) lower system operating temperatures, (4) third shutdown system, (5) core catcher, and (6) no-vent containment.

Q.25. Is the pool-type system configuration a substantially better alternative?

A.25. No. In a loop-type configuration, such as CRBRP, the major primary heat transport system components are interconnected with the reactor vessel by means of coolant-carrying piping. In a "pool-type" configuration, the primary system components are in a "pool" of sodium contained within a vessel which also houses the reactor core.

It should be noted, however, that many features, for example, intermediate heat transport system (IHTS), steam generator system (SGS), the turbine generator, and auxiliary ~~auxilary~~ systems, are common to both concepts. Therefore, much of the information obtained from a loop plant such as CRBRP, including contributions to the overall technology base, is relevant to either concept. Pool-type systems have been considered since the very early period in LMFBR development. Early U.S. fast reactors were built both in pool (EBR-II) and loop (SEFOR and Fermi-I) configurations. In foreign reactors, early test LMFBRs were built only in loop configurations (DFR, Rapsodie and BR-5). The current generation of larger plants includes both loop (SNR-300, BN-350, Joyo and Monju) and pool (Phenix, PFR, Superphenix, BN-600). Recent evaluations performed in the U.S. have indicated no clear superiority of one system over the other. In these evaluations, attention was given to safety, maintainability, cost and duration of fabrication and construction, and economy of operation. On a purely functional basis, both pool and loop-type LMFBRs are feasible and neither has a significant overall advantage over the other. Considering fabrication/construction differences between pool and loop-type reactors, the cost and schedule estimate differences are generally recognized to be within the range of uncertainty of the estimating accuracy. However, there is a lack of large pool-type reactor construction experience in this country, and there is a schedule risk associated with the greater estimated field labor requirements for a pool-type reactor. Therefore, there is no substantial advantage of the pool concept over the loop concept, and CRBRP in a loop plant configuration has a higher likelihood of meeting its objectives and timing.

Q.26. Is the use of flywheels on sodium pumps a substantially better alternative?

A.26. No. The CRBRP primary flow coastdown characteristics (the flow vs. time after power is removed from pumps) have been selected by balancing two competing requirements:

1. The need to provide adequate coolant flow to the core and radial blanket for all design basis events including postulated loss of power to all three primary pumps, and
2. The need to minimize the thermal transients associated with reactor and plant trips.

Too little flow might result in inadequate core cooling, while too much flow may result in overcooling and thermally stressing plant components during transients. The required flow coastdown characteristics for the CRBRP sodium pumps are being provided by building directly into the pump drive rotor (as opposed to the addition of a separate flywheel) sufficient inertia so that the required momentum of the pump-drive motor assembly will be available. This inertia satisfies both of the above requirements.

Transient overpower (TOP) and loss-of-flow (LOF) events which are beyond the design base have also been considered. The addition of a heavy flywheel would be ineffective in significantly reducing the likelihood or consequences of such events. For the postulated transient overpower (TOP) events that assume failure of both reactor shutdown systems, there would be no advantage for a heavy flywheel because the pumps continue to run in that event. For the postulated loss of flow (LOF) events that assume failure of both reactor shutdown systems, the addition of heavy flywheels would not change the overall conclusions. The time for initiation of boiling would increase slightly, but once boiling is initiated, the sequence of events is controlled by the phenomena related to boiling, which are not affected by a flywheel. Increased pump inertia produced by the flywheel would not change the likelihood of sodium boiling and the resultant consequence of a non-energetic core meltdown.

On the other hand, increasing the pump inertia by means of a flywheel beyond that required to provide adequate coolant flow increases the rate of temperature change associated with system thermal transients, thereby adding to the fatigue damage associated with transients. Thus, adding a pump flywheel would not be a substantially better design alternative than the CRBRP design.

Q.27. Is system operation at lower temperatures a substantially better alternative?

A.27. No. The system operating temperatures of the CRBRP were selected based upon plant performance analyses that considered equipment constraints, steam conditions, desired fuel performance, thermal transient and creep effects and cycle efficiency. Lower system temperatures have been considered for CRBRP as well as for future plants. For normal operations and accidents within the design basis, a balancing of the advantages and disadvantages of lower system operating temperatures shows that this is not a substantially better design alternative. Lowering the operating temperatures without lowering the design temperatures would have the effect of increasing equipment sizes and costs and decreasing efficiency, while providing more margin to system limiting conditions and slightly improved fuel performance. However, at any given design temperature, the prudent designer would provide the same structural design margins between operation and design temperatures, and there is no net benefit to be derived from lower operating temperatures.

Events beyond the design base have also been considered in relation to selection of system operating temperatures. The effect of choosing a lower plant operating temperature would not significantly change the transient overpower hypothetical core disruptive accident (HCDA) consequences because the current transient overpower scenario results in molten fuel release from the pin before coolant boiling occurs. Thus the overall conclusions regarding the transient overpower HCDA would not be influenced by a choice of lower operating temperature.

The effect of lower operating temperatures on the likelihood and consequence of a loss-of-flow HCDA is similar to that described for pump inertia selection. The time to initiate boiling would be slightly increased, but the likelihood or consequences of sodium boiling would not change. Thus, the overall conclusions regarding the loss-of-flow HCDA would not be influenced by a choice of lower operating temperature.

In summary, when all of the above factors are considered, lower CRBRP operating temperatures would not be a substantially better alternative for meeting project objectives.

Q.28. Is a third shutdown system a substantially better alternative?

A.28. No. As discussed in Applicants' testimony concerning NRDC Contentions 1, 2, and 3, there are two control rod systems in CRBRP. The systems are diverse--that is, they have different operating principles and use different components, and they are redundant--that is, each system is designed to shut the reactor down without action by the other system. Each system is also internally redundant--that is, each system by itself is designed to shut down the reactor even if any one control rod in that system does not function.

A third shutdown system is unnecessary, because, as shown in Applicants testimony concerning NRDC Contentions 1, 2, and 3, all credible failure modes are addressed by the primary and secondary shutdown systems. A third shutdown system would not address any other known failure modes, and would not provide a significant reduction in risk to the public health and safety. Therefore, the addition of a third shutdown system would not be a substantially better alternative.

Q.29. Is inclusion of a core catcher in the design a substantially better alternative?

A.29. No. Core catcher is the name associated with the features in a plant design that would provide for the ability to retain some or all of the core subsequent to an over-power or undercooling accident that results in melting of the core and subsequent meltthrough of the reactor vessel and guard vessel. The Applicants have analyzed the benefits which would be obtained from inclusion of a core catcher in the design and have compared the protection afforded to the public health and safety with and without the inclusion of a core catcher in the design. A core catcher is generally assumed to include means for keeping this core debris from penetrating further into the bottom of the reactor cavity. It should be noted that (a) the core catcher does not in any way reduce the likelihood of an HCDA and that (b) any active features provided in the core catcher have to perform in an extremely hostile environment subsequent to an HCDA and are inaccessible at a time when they are required to function.

As shown in Applicants' testimony concerning NRDC Contentions 1, 2, and 3, the overall approach to CRBRP design has been to include in the design such features that make the likelihood of a core melt so unlikely that one need not include a core melt in the spectrum of Design Basis Accidents. However, as further shown by the Applicants' testimony concerning NRDC's Contentions 1, 2, and 3, the Applicants have provided margins and design features in CRBRP to mitigate the consequences of HCDAs and to assure that the residual risks from HCDAs can be made acceptably low. There is no substantial further advantage to inclusion of a core catcher in the design.

Q.30. Is a no-vent containment a substantially better alternative?

A.30. No. CRBRP containment design includes provisions for venting the containment in a controlled manner. In the normal mode of operation, there is negligible radioactivity in the containment atmosphere and the containment is continuously vented. This provides for access to the containment during operation, thus improving operability and maintainability of the plant.

The atmosphere in the containment is continously monitored. In the event that any significant radioactivity levels are detected in the containment effluent, the containment atmosphere is isolated through the use of containment isolation valves. Under such circumstances, the containment is essentially unvented and for all design bases may be kept unvented for as long as it is desired.

The CRBRP design has provision for filtering and cleanup of the vent discharge from the containment. Thus, when a decision to vent the containment is made, protection to public health and safety is provided by assuring, through cleanup of the discharge, that the radiological releases are acceptably low.

Elimination of the capability to vent the containment in normal operations is not advantageous to demonstrating the CRBRP objectives. The analysis has shown that, even with venting, the radiation dose guidelines are not exceeded. Thus, the health and safety of the public is assured even with a vented containment. On the other hand, elimination of venting during normal operation makes the containment access during normal operation (operability, maintainability) difficult. This is contrary to basic objectives of the CRBRP. Thus, under normal operation and design basic ~~condition,~~ conditions, a no-vent containment is not a significantly better alternative.

In the event of a beyond-the-design-base Hypothetical Core Disruptive Accident (HCDA), the containment is isolated through the containment isolation system on detection of high levels of radioactivity in the atmosphere. CRBRP analysis shows that, subsequent to an HCDA, the containment may have to be vented in order to maintain the containment pressure within the containment vessel capability.

Even under the HCDA condition, there is still no particular advantage to elimination of the containment capability to vent through a cleanup system and to maintain a no-vent condition indefinitely. CRBRP analysis shows that even under the HCDA conditions, through the use of controlled filtered vent, the radiological releases for such accidents are acceptably low.

Design measures could be taken to increase the probability that no vent would be required. One cannot in practice, however, foresee all contingencies. Therefore, it is prudent and advantageous to include a filtered controlled vent capability to assure that containment integrity cannot be challenged.

On balance, a no-vent containment is not a substantially better alternative.

Q.31. What conclusions do you draw?

A.31. Although alternatives to specific CRBRP features have been identified by NRDC as potentially advantageous, none are substantially better than those presently incorporated in the design.

STATEMENT OF QUALIFICATIONS

John R. Longenecker
Acting Director, Office of the Clinch River
Breeder Reactor Plant Project
Office of Breeder Reactor Programs
U.S. Department of Energy

John R. Longenecker is Acting Director of the Office of the Clinch River Breeder Reactor Plant (CRBRP) Project in the Department of Energy (DOE). Included within his responsbilitiy is the licensing and program management of the CRBRP, and direction of the conceptual design of the Liquid Metal Fast Breeder Reactor (LMFBR) Large Developmental Plant.

Prior to this assignment, Mr. Longenecker served in the Department of Energy as Director, Division of Plant Development; Chief, Conceptual Design Study, Division of Reactor Research and Technology (RRT); Technical Assistant to the Program Director, Nuclear Energy Programs; and in various capacities in the Energy Research and Development Administration's (ERDA) Division of Reactor Research and Development, including Special Assistant to the Director, Acting Assistant Project Director for Procurement for the CRBRP, Acting Chief of CRBRP Mechanical Components Branch, and Reactor Engineer for various LMFBR projects. He joined the Atomic Energy Commission in 1973 and served there in the Division of Reactor Development and Technology prior to the formation of ERDA in 1975, and DOE in 1978.

Prior to entering Government service, Mr. Longenecker was employed by the Ford Motor Company as a research engineer and by the firm of John Robinson and Associates as a structural engineer.

Mr. Longenecker received both Bachelor of Science and Master of Science degrees in solid state mechanics from the Pennsylvania State University.

STATEMENT OF QUALIFICATIONS

Narinder N. Kaushal

Engineering Division

Clinch River Breeder Reactor Plant Project Office

P.O. Box U

Oak Ridge, TN 37830

Dr. Kaushal is the Deputy Assistant Director for Engineering at the Clinch River Breeder Reactor Plant Project. In this capacity he serves as the principal technical, administrative and operating official of the Engineering Division, coordinating and executing approved programs, policies and decisions of the Assistant Director for Engineering.

From February 1978 until August 1982, Dr. Kaushal served as the Chief, Reactor and Plant Systems Branch. In that capacity, he directed the day-to-day activities of CRBRP participants involved in the design, development, fabrication, test, evaluation, installation, checkout, startup test, safety, operation, and plant security of the major systems and components of the reactor and balance-of-plant.

From February 1975 to 1978, Dr. Kaushal served as the Chief, Instrumentation, Control and Electrical Branch, with a full range of CRBRP management responsibilities for the reactor and plant controls, instrumentation and electrical systems.

He holds bachelor's degrees in mathematics and physics, a masters degree in both physics and electronics, and a doctorate degree from Rensselaer Polytechnic Institute in nuclear physics and solid-state physics. From 1967, when he received his doctorate degree, until he joined the CRBRP Project in 1974, he was a Research Associate in the Nuclear Engineering Department of Rensselaer Polytechnic Institute (RPI). In this capacity he conducted research at the RPI Linear Accelerator Laboratory, and had supervisory responsibility for the Fast Neutron Spectrum Program.

STATEMENT OF QUALIFICATIONS

Carl A. Anderson, Jr.

Project Manager, Large Plant Projects

Westinghouse Advanced Reactors Division

Madison, Pennsylvania 15663

Since January 1979, I have been Project Manager, Large Plant Projects, with responsibility for reactor design and development of the LMFBR plant to follow Clinch River; technical interactions with LMFBR programs in the United Kingdom, France, Germany and Japan; fusion development; and other advanced reactor programs.

I received the degree of Mechanical Engineer from Stevens Institute of Technology in 1956, and the degrees of Master of Science in Mechanical Engineering in 1957, and Doctor of Philosophy in Nuclear Engineering in 1961 from Massachusetts Institute of Technology. In 1971, I completed the Program for Management Development at the Harvard Business School.

From 1961 to 1964, I was employed by Sandia Corporation at Sandia National Laboratory, Albuquerque, New Mexico, first as a Section Supervisor and later as a Division Supervisor. I directed design, construction and operation of the Sandia Engineering Reactor and the Sandia Nuclear Assembly for Reactor Experiments. I supervised a nuclear dosimetry laboratory, a gamma irradiation facility, cryostats, hot laboratories, remotely operated machinery, and pulsed neutron experiments.

From 1964 to 1967, as a Staff Member of the Los Alamos Scientific Laboratory in Los Alamos, New Mexico, I participated in fast reactor research and development. The work centered on the Fast Reactor Core Test Facility and the LMFBR cores planned for operation therein, with associated work on correlation function analysis, fission product yields, heat transfer, mechanical design and plant transient analysis.

I joined the Westinghouse Electric Corporation Advanced Reactors Division in 1967.

From 1967 to 1968, I was Project Manager responsible for the 1000 MWe LMFBR Design Study and the Large Sodium Pump Study.

From 1968 to 1971, I was Manager, LMFBR Reactor Engineering, responsible for conceptual design and analysis of the reactor for the Westinghouse LMFBR Demonstration Plant. This included the nuclear, thermal-hydraulic and mechanical design of components within the reactor vessel.

From 1971 to 1974, I was Manager, FFTF Reactor Engineering. I directed the design, procurement and fabrication of the FFTF reactor, including reactor vessel, head, instrument trees, in-vessel handling machines, shielding, core support structure, core basket, core restraint system, flux monitor system and control rod systems.

From 1974 to 1975, I was Manager of Technology, responsible for the Division's fuels and materials research and development, sodium loop testing, friction and wear testing, fast breeder fuel fabrication and testing, and stress analysis methods development.

From 1975 to 1976, I was Reactor Plant Project Manager in the CRBRP Project. I supervised the analysis, design, procurement and fabrication of those portions of the plant for which Westinghouse was technically responsible, including: fuel, radial blanket, shield, control system, reactor vessel and head, core support structure, reactor internals, primary piping, intermediate heat exchanger, check valves, instrumentation and controls.

From 1976 to 1979, I was Project Manager, Prototype Large Breeder Reactor. In this position I was responsible for all aspects of the effort to design the Prototype Large Breeder Reactor, a near-commercial LMFBR.

I am a member of the American Society of Mechanical Engineers, the American Nuclear Society, the American Association for the Advancement of Science, and the Society of the Sigma Xi.

15-11

JUDGE MILLER: I think there is one more panel by the Staff; is that correct?

Call your witnesses.

We will take a five-minute recess.

(Short recess.)

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16-1
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JUDGE MILLER: Is the Staff ready to proceed?

MR. MIZUNO: The Staff is ready to proceed.

JUDGE MILLER: Have all of the witnesses been previously sworn?

MR. MIZUNO: I'm not that sure. Mr. Becker, have you --

JUDGE MILLER: Has anyone not been previously sworn?

Whereupon,

PAUL H. LEECH,

RICHARD A. BECKER

-and-

JOHN K. LONG

were recalled as witnesses by and on behalf of the Staff and, having been previously duly sworn, were examined and testified as follows:

DIRECT EXAMINATION

BY MR. MIZUNO:

Q. Gentlemen, do you have the document entitled "Testimony of Paul H. Leech, Richard A. Becker and John K. Long Relative to NRDC Contention 7(a) and 7(b)" before you?

BY WITNESS LEECH:

A. Yes.

/

BY WITNESS BECKER:

A. Yes

BY WITNESS LONG:

A. Yes.

MR. MIZUNO: Mr. Chairman, I'd like to have this document identified and marked as Staff Exhibit No. 21.

JUDGE MILLER: It may be marked as Staff's Exhibit 21.

(Staff's Exhibit No. 21 was marked for identification.)

BY MR. MIZUNO:

Q. Gentlemen, do you have any changes or corrections to make at this time?

Please go ahead.

BY WITNESS LONG:

A. Yes, I have a couple. On Page 20 in the Answer A.41, in the eighth line it says "The Prototype Fast Reactor (PFR) operated continuously from 1977 to the present."

I'd like to modify that to say "intermittently."

JUDGE MILLER: Instead of "continuously"?

WITNESS LONG: Instead of "continuously."

JUDGE MILLER: Very well.

WITNESS LONG: I think that's more accurate.

On the following page, 21, there is a typo that may not be understandable.

In the eighth line I think the word "containment" was meant to be "contamination."

JUDGE MILLER: "Contamination control"?

WITNESS LONG: Yes.

JUDGE MILLER: Okay.

WITNESS LONG: On Page 26 in the Answer A47, the second paragraph, the first sentence which reads -- well, I won't read it -- should be qualified by the insertion of the words, "in the United States," after the word "development."

JUDGE MILLER: Okay.

WITNESS LONG: That's all I have.

BY MR. MIZUNO:

Q. Mr. Leech, do you have any corrections to make?

BY WITNESS LEECH:

A. No, I don't.

Q. Mr. Becker, do you have any corrections?

BY WITNESS BECKER:

A. No, I have none.

Q. Gentlemen, as corrected, does this testimony represent your testimony in this proceeding?

/

BY WITNESS LONG:

A. Yes.

BY WITNESS LEECH:

A. Yes.

BY WITNESS BECKER:

A. Yes.

Q. Is it true and correct to the best of your belief?

BY WITNESS LONG:

A. Yes.

BY WITNESS LEECH:

A. Yes.

BY WITNESS BECKER:

A. Yes.

MR. MIZUNO: I tender the panel for cross-examination.

JUDGE MILLER: Very well. Mr. Edgar, cross-examination?

MR. EDGAR: We have no questions -- Oh, no, I do have one.

CROSS-EXAMINATION

BY MR. EDGAR:

Q. There is discussion in the testimony -- the question of steam generator considerations, and I wonder which of the witnesses is the correct one to

16-5

address the question to.

MR. MIZUNO: Mr. Becker would be the appropriate person.

MR. EDGAR: Okay. He's the man who prepared --

MR. MIZUNO: Perhaps I should mention the areas of expertise for the different witnesses. Would this help?

MR. EDGARD: Yes.

MR. MIZUNO: Okay. I believe Mr. Leech will be speaking to the timing objectives -- timing programmatic objective and the ability to meet that objection.

Mr. Becker will be speaking about the steam generator program. He has expertise in that area.

Mr. Long will be speaking about the alternative design concepts.

MR. EDGAR: All right. Well, my question involves the steam generator experience and the discussion which, I assume, would be on Page 7, running over through Page 11.

BY MR. EDGAR:

Q The question is, Mr. Becker: In analyzing the issue of steam generators and the discussion in that testimony, what is your judgment concerning the safety significance of that class of steam generator issues discussed in the testimony?

BY WITNESS BECKER:

A. My judgment with respect to that class of issues regarding the steam generator -- at this point I guess I would have to say is incomplete since we're in the process of doing the safety evaluation. That has not been published yet.

I think some judgments can be drawn of a general nature. The steam generator being located in a secondary loop removed from the reactor has a tendency to have small impact, as far as failure rates are concerned.

There is considerable redundancy as far as a component in the decay heat removal chain, and, therefore, the loss of a single steam generator disables one of the main loops and its ability to remove heat from -- for decay heat removal.

But there are two additional loops, in addition, a decay heat removal service that's available.

The other aspect I guess that should be addressed would be the sodium/water reaction. I think there has been testimony given that the sodium/water reactions have been anticipated, and there are a number of systems associated with the steam generator that will mitigate the consequences of a sodium/water reaction.

MR. EDGAR: I have no further questions.

JUDGE MILLER: Intervenors?

MS. FINAMORE: Yes.

CROSS-EXAMINATION

BY MS. FINAMORE:

Q Dr. Long, on Page 26 of your testimony you just added the words, "in the United States," to that second paragraph of Answer 47.

Am I correct to infer from that that self-actuated shutdown systems have yet reached such a stage of development in other countries that would permit their use in the CRBR?

BY WITNESS LONG:

A Yes.

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Q Can you explain which systems and which countries you're referring to?

BY WITNESS LONG:

A. Well, I have recently learned that they have been used in Phenix in France.

Q Is that the only one you're referring to?

BY WITNESS LONG:

A. Yes.

Q And when you say "self-actuated shutdown systems," can you describe that briefly?

BY WITNESS LONG:

A. Conceptually, I can describe the one that was used in Phenix. It is a -- what's called a curie-point sensitive magnetic latch, so that a control rod is triggered to be inserted into the core by the direct temperature of the coolant, rather than going through an electronic sensing and triggering device.

Q Would you characterize that as a totally passive device?

BY WITNESS LONG:

A. I don't regard it as totally passive since it has a latch, and it has motion -- it has to move. It has fewer required working parts than the systems that have the electrical and electronic sensing and control.

Q So would you characterize the shutdown systems in the present CRBR design as more active than those you've just described for the Phenix?

BY WITNESS LONG:

A. Well, may I say that I define the degree of activity in terms of the number of working parts required. And I think my answer to the previous question is that there are more working parts on the systems in CRBR.

Q. Mr. Leech, did the Staff conduct a review to determine whether the Clinch River Breeder Reactor will be constructed as expeditiously as possible?

BY WITNESS LEECH:

A. No, we did not.

Q. Is it your view that the Staff did not need to determine whether the Clinch River Reactor will be built as expeditiously as possible in order to obtain an LWA or recommendation from the Staff for an LWA?

BY WITNESS LEECH:

A. That's correct.

Q. Dr. Long, if the Clinch River Breeder Reactor had a steam generator explosion during the five-year demonstration, would the plant still meet its technical performance and reliability objectives?

BY WITNESS LONG:

A. I think -- I hate to use that word "explosion." I know we've debated quite a bit about that in

16-10

these proceedings.

The Staff prefers to use more precise language.

Q With that amendment and whatever precise language you wish to use for the phenomenon that I'm referring to as an explosion, if that event occurred during the five-year demonstration, would the plant still meet its technical performance and reliability objectives?

BY WITNESS LONG:

A Now, our reading of Contention 7 is that the objectives under discussion are the objectives of CRBR providing useful information for the LMFBR program; that is, Contention 7, as we understand it, questions whether CRBR is a useful information generator.

With that proviso, much can be learned from the operation of CRBR even if there are problems with the steam generator. We think in that sense that it will meet its objectives.

Q The technical performance and reliability objectives?

BY WITNESS LONG:

A The objectives of providing technical performance and reliability information for the LMFBR program, which is the subject, as I understand it, of the contention.

Q So your answer is yes?

BY WITNESS LONG:

A. Yes.

Q. Is your answer also yes, if you had a core disruptive accident exceeding the 661-megajewel primary containment design basis?

BY WITNESS LONG:

A. Of course, we don't think that's likely. We think that the CRBR will, through its safety review, be adequately protected against such accidents.

However, if accidents occur -- and we think that any accident would be of a much more modest nature -- it certainly would be logically to provide information relevant to the program.

It might provide information causing us to re-examine the nature of the program. But it would certainly provide information for the program.

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17-1

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BY MS. FINAMORE:

Q Would the plant still meet its technical performance and reliability objectives with such an accident?

BY WITNESS LONG:

A With regard to the program, yes.

Q If such an accident happened in the first five --

BY WITNESS LONG:

A By "such an accident," do you mean the accident is not even described -- it's somewhat modified from your original 661-megajewel?

Q No, I'd like the answer for the core disruptive accident exceeding the 661-megajewel primary containment design basis.

BY WITNESS LONG:

A Well, that certainly would provide some information, too.

(Laughter.)

Q And in such a case, would you consider that the technical performance and reliability objectives have been met?

BY WITNESS LONG:

A As stated in Contention 7, yes.

Q And if that accident that I just described happened in the first five years of CRBR's operation, would the reactor meet its technical performance and reliability objectives?

BY WITNESS LONG:

A. Yes.

MR. MIZUNO: Could I have a clarification from counsel for Intervenors? Is she referring to the overall programmatic objectives, which are the subject of this contention, or specific project objectives?

I believe that distinction was made earlier by Applicants' panel.

MS. FINAMORE: Well, I --

JUDGE MILLER: The witnesses seem to be consistently describing and responding in terms of programmatic objectives, as I understand their testimony.

BY MS. FINAMORE:

Q. I'd like to turn to Page 7, Answer 13. Mr. Becker, you state that "EBR-II has operated a steam generator for 19 years without having a water-to-sodium leak."

Is the CRBR sodium generator design similar to that of EBR-II?

BY WITNESS BECKER:

A. No.

Q. Also, in Answer 13 you briefly

17-3

catalogue the steam generator experience at other breeders. Is the CRBR steam generator design similar to the steam generator design at any of those other breeders?

BY WITNESS BECKER:

A. In what respect to you define "similarity"?

Q. In the normal sense of the word. Is it similar in any respect that you can identify?

BY WITNESS BECKER:

A. I have some difficulty with the question because physical similarity is lacking in almost all of these steam generators. They have different configurations. They do use sodium on one side, water on the other.

Q. You state on Page 14 -- Page 8, Answer 14, that "Several physical configurations have produced leak-free designs," and that "It is ... important that the configuration selected be capable of incorporating proper design features and the lessons learned from available steam generator experience."

What proper design features are you referring to?

BY WITNESS BECKER:

A. The design features referred to here are the ability to inspect and provide proper metallurgical treatment for welds, spacing in some cases for -- to avoid flow-induced vibration problems, things of that type.

Q On Page 18 of your testimony, Anwer 35 -- Anyone may answer.

You state that "The economic projects for an LMFBR utility plant will be guided by a detailed cost accounting of capital and operating expenses for the CRBRP."

Have you analyzed the effectiveness or the extent of this cost accounting method in any way?

BY WITNESS LONG:

A I'm responsible for the answer, and I was basing it on the descriptions of the cost accounting program that were furnished to us in the -- I believe it's in the Applicants' Draft Environmental Statement, but it was summarized in Mr. Longenecker's testimony this morning.

Q Now, regarding the objective of economic feasibility, do you define that objective solely to mean whether or not there is an effective detailed cost accounting program, such as you've described?

BY WITNESS LONG:

A Essentially, yes. As far as the generation of information useful to the LMFBR program is concerned, it seems to me that CRBR's contribution will be in its ability to separate the first-of-a-kind costs from the relevant costs for repetitive construction programs.

Q Now, setting aside the first-of-a-kind costs,

17-5

in analyzing whether CRBR demonstrates that an LMFBR breeder is economically feasible, do you consider at all the extent of the costs, in other words, how high the costs are? Do you consider that a factor in your analysis?

BY WITNESS LONG:

A. No, not as I am looking at the question here. I'm looking at the question of whether CRBR will provide information for the LMFBR program. Now, they may provide information which says this program is too expensive or higher than we used to think it was, or something like that.

That would be information.

Q. Now, suppose that the information you've described does show the plant is too expensive, as you've put it, would you consider that the Clinch River plant has demonstrated its economic feasibility objectives --

MR. MIZUNO: Objection. That's a hypothetical, and I think he has already indicated this this information can only be gotten together and evaluated once you have all that information in.

The plant hasn't been completely designed and constructed and operated yet.

JUDGE MILLER: Yes. But the question assumes that in the future there will be such operation and so

17-6

forth and some results of an economic or other nature, which as susceptible of evaluation.

The question then inquires of the relationship between those economic or cost factors and the programmatic objectives of the project.

You may answer.

WITNESS LONG: I -- Could you repeat then the question, just so I make sure?

JUDGE MILLER: Yes. Go ahead, Ms. Finamore.

BY MS. FINAMORE:

Q If I may rephrase it.

Does the actual cost of the plant -- or a judgment by you that the plant is too expensive, in your words, affect in any way whether the plant meets its economic feasibility objective?

JUDGE MILLER: Just a moment. Now he didn't say he was going to judge whether or not it was too expensive.

He was saying the data would be obtained from which programmatic judgments could be made, in terms of the programmatic objectives. But he did not say that the NRC was going to make that judgment, any more than the NRC makes a judgment whether XYZ private utility should put in a plant.

You did change -- and you changed all together,

7-7

I think, the nature of the inquiry.

BY MS. FINAMORE:

Q Well, has the Staff made a judgment at this time that the plant has met its economic feasibility objective?

BY WITNESS LONG:

A With respect to providing information for the LMFBR program, yes.

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17-8

Q Now is it possible that review of the plant information that you've just described might indicate that the plant is too expensive?

BY WITNESS LONG:

A Yes. Too expensive to be attractive to utilities is what you mean.

Q Yes.

BY WITNESS LONG:

A Yes. That's possible.

Q Well, assuming such is the case hypothetically, would you consider -- or would the Staff consider the cost of the plant in analyzing whether it has, in fact, met the economic feasibility objectives?

MR. MIZUNO: I have to object again. I don't see how the -- Mr. Long can answer that question. He has already stated that the information is going to be collected and evaluated.

He can't -- It will meet the objectives for obtaining information. But as far as having a post facto decision, I don't see what the relevance of that is.

MS. FINAMORE: If I may respond. I believe that the plant is going to go through the five-year demonstration phase, and even before that phase it will be generating cost information at later stages in this proceeding, and judgments such as the plant's meeting of programmatic objectives, are under the Board's views still open for re-analysis on the basis of new information.

And --

JUDGE MILLER: Wait a minute. What's the basis for that? I'm not following you there.

MS. FINAMORE: Well, for example, at the next stage of these proceedings --

JUDGE MILLER: You mean the LWA-2?

MS. FINAMORE: The CP.

JUDGE MILLER: Oh, the CP. All right.

You're projecting now that we're getting to the construction and permit hearing stage --

MS. FINAMORE: Well, just as a hypothetical.

JUDGE MILLER: It may be not so hypothetical, but go ahead.

(Laughter.)

MR. EDGAR: We're going to have continuing jurisdiction and monitoring --

JUDGE MILLER: You may be working for free.

However, let's assume, yes, that we move into the construction and permit stage. Go ahead.

MS. FINAMORE: Now it's my understanding from earlier discussions we've had with the Board that although

17-10

the impact statement is completed and the Board is going to make a decision upon its adequacy --

JUDGE MILLER: Is this for environmental purposes -- NEPA --

MS. FINAMORE: Oh, yes.

JUDGE MILLER: Okay. I follw you then.

Yes, that's right. We said you wouldn't be precluded at a later stage, even though we had gone into it somewhat at the LWA stage.

That doesn't answer the question of whether or not you should go into it at the LWA stage.

MS. FINAMORE: On the basis of new information.

I'm trying --

JUDGE MILLER: Well, new and relevant information that qualifies under late-filed contentions, if that be it, and so forth.

The Board hesitates to rule in advance and in a vacuum, because we could be quoted -- you could pull this out of the transcript a year from now and say, "You said."

I, therefore, must qualify it.

MS. FINAMORE: I'm not asking whether or not it will meet the objectives. I'm just asking the Staff how it defines this objective and what factors it would

17-11

consider in this objective.

It seems to me that one reasonable factor to consider in determining economic feasibility is how much the plant costs.

And I'm trying to determine if the Staff shares that description of the economic feasibility documents -- objective.

MR. EDGAR: That wasn't the question --

JUDGE MILLER: No, that wasn't the question.

However, what I'm considering is: To what extent is that material or relevant here at this stage of the inquiry?

MS. FINAMORE: Well --

JUDGE MILLER: -- if the plans as they progress -- whatever degree: CP stage or what not, you're going to have generated cost information, presumably. But, query: Are you entitled at this point to go beyond the issue of whether or not the cost information that's generated has some reasonable relationship to the programmatic objective, not whether the Staff study makes a decision on whether it's worthwhile to go ahead or not.

And you keep, I think, veering over into that area.

MS. FINAMORE: Okay. Just referring to the present --

17-12

JUDGE MILLER: And to the Staff's role.

MS. FINAMORE: The Staff's role, that's what I'm trying to ascertain, how they define their role at the present time.

JUDGE MILLER: Why don't you ask them what they consider their role to be with this whole question of costs and economic -- We'll recess for lunch in five minutes.

But go ahead, you have time.

BY MS. FINAMORE:

Q. Do you consider a -- What do you consider the Staff's role to be at this stage in the proceeding in determining whether the CRBR is likely to demonstrate the economic feasibility of a relevant commercial LMFBR central station electric plant?

BY WITNESS LONG:

A. Well, we believe that the Staff's role in that area is to determine whether the proper accounting systems are being kept so that information relative to the future LMFBR program can be derived from the CRBR accounts and utilized.

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Q Now, do you believe the Staff's role should also include evaluating the cost of the plant at this present time, or cost estimates at this present time?

BY WITNESS LONG:

A. Well, are you still within your previous question? That is, in determining whether it meets its goals relative to the LMFBR Program?

Q In determining whether it meets its economic feasibility objective?

BY WITNESS LONG:

A. I don't believe that the cost of CRBR is something we should judge at this time. I don't believe it's within our role to decide at this time whether CRBR cost is acceptable or not.

JUDGE MILLER: Finish your answer and then we will recess an hour for lunch, and we'll return at 2:00 o'clock.

Finish your answer, if you hadn't finished, sir. I didn't mean to interrupt you.

WITNESS LONG: I had nothing to add, except to refer to my previous answer, that we are looking at the methods of generating cost accounting information to see that they will be applicable.

If it turns out that CRBR costs more than we expect, so be it.

18-2

We, of course, will factor -- that will be useful information in deciding whether you want to go on with the program.

BY MS. FINAMORE:

Q. But that could in no way prevent the CRBR from meeting its economic feasibility objective in your mind?

BY WITNESS LONG:

A. Yes.

JUDGE MILLER: Okay, 2:00 o'clock.

(Whereupon, at 1:00 p.m., the hearing was recessed, to reconvene at 2:00 p.m., the same day.)

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AFTERNOON SESSION

2:00 p.m.

JUDGE MILLER: All right. Let's resume the hearing.

Ms. Finamore, I believe you were examining.

MS. FINAMORE: Yes.

BY MS. FINAMORE:

Q I'd like to turn to Page 26, Answer 48. You state that:

"The installation of a core retention device in CRBR would not likely generate any useful operating data for future reactors, since the probability of its being called into use is extremely low."

First of all, are you making any distinction between a core retention device and a core catcher?

BY WITNESS LONG:

A No.

Q Dr. Long, are you saying in this statement that useful information from a core catcher (I'll refer to it in that manner) could only be obtained through examining how it is used or how it functions in the case of an accident?

BY WITNESS LONG:

A No.

19-2

Q Isn't it possible that one could also gain useful information from the construction of a core catcher?

BY WITNESS LONG:

A I presume so.

Q And isn't it true that one could also gain the useful information in the design and testing of a core catcher?

BY WITNESS LONG:

A Yes.

Q On Page 27, Answer 49, you discuss the sodium pump flywheels and note that Applicants claim that the inertia in the flowing sodium is sufficient for coastdown purposes, and that you have the matter under review.

That's at the end of the first paragraph of Answer 49.

Is it a correct characterization of this statement that you have not yet been able to conclude that you agree with Applicants' claim?

BY WITNESS LONG:

A We have not yet reached our conclusion on the loss of flow accident to which this is a part, and so I hesitate to indicate complete agreement on it.

However, I do agree in principle with the idea that the decay in power should govern the required coastdown rate of the coolant, and that this can be

19-3

achieved through inertia in the coolant, and I perhaps should have added in the pump, although I felt that that was implied when I said in the inertia of the flowing sodium; I mean the sodium and the pump itself.

Q. In the beginning of the next paragraph you state:

"It is not known whether or not the coastdown would need to be augmented in some way for the larger LMFBR's."

But isn't the import of your previous paragraph, which we just discussed, that you also don't know whether augmentation is necessary for smaller LMFBR's?

BY WITNESS LONG:

A. Yes.

Q. In the last paragraph on Page 27, you characterized flywheels on the sodium pumps as fairly standard devices; is that correct?

BY WITNESS LONG:

A. Well, flywheels themselves are fairly standard devices.

Q. The fact that they are fairly standard, doesn't that indicate the general perception of the need for such components?

BY WITNESS LONG:

A. Oh, no. I mean they are fairly standard ways

19-4

of extending the inertia of a rotating system. They have been used in steam engines, you know.

They are used in your automobile, for example.

Q Haven't they been used in other nuclear reactors as well?

BY WITNESS LONG:

A I think the French have used some additional rotating inertia in Phenix.

Q I believe in your written testimony you stated that the hydraulic performance is crucial to avoiding cracks in the pipes; is that correct?

BY WITNESS LONG:

A Can you point out where that is?

Q I'm having a little difficulty. I thought you might recall.

BY WITNESS LONG:

A I don't recall saying exactly that. That's why I wanted to make sure that we were in agreement on what we're talking about.

Q Mr. Becker, am I correct that you discuss hydraulic performance? I'm trying to find it.

BY WITNESS BECKER:

A No, that's not correct.

Q Dr. Long, am I correct that the parallel design of the Applicants included a core catcher?

19-5

BY WITNESS LONG:

A. I think so. Now, it's been quite a number of years since we discontinued our review of that, and I'm not really sure of details, but I think it did have -- it may have had an option or it may have actually had a core retention device. I'm not sure.

MS. FINAMORE: I have no further questions.

JUDGE MILLER: Thank you. Who hasn't yet cross-examined? I guess everybody has.

Is there any redirect?

MR. MIZUNO: A minute to confer with the witnesses.

(Discussion off the record.)

REDIRECT EXAMINATION

BY MR. MIZUNO:

Q. Dr. Long, have you evaluated whether the use of a self-actuated shutdown system at Clinch River will be substantially superior to the present Clinch River design for meeting its programmatic objectives?

BY WITNESS LONG:

A. We have concluded from the information that we have several aspects of that. It seems to us rather important in deciding whether it will meet its informational objectives to know whether self-actuating shutdown systems are indeed likely to be used in the LMFBR Program, and that judgment, I guess, you'd have to say remains open, although their use is not contemplated in the LDP.

In view of the fact it isn't really known whether they will be used and isn't really expected that they will be used, it's hard to conclude that you would gain useful information for the program by placing them in CRBRP.

It's also true that you can gain all the information that you need concerning them in other facilities.

We have the FFTF for the purpose of testing fuel and developing devices of that sort.

So it does not seem that the informational requirements of CRBR would require that such systems be inserted and tested in CRBR itself.

Q. Okay. Will the construction of a core catcher demonstrate whether the core catcher can operate or not?

BY WITNESS LONG:

A. No.

Q. Can you explain why? Why not?

BY WITNESS LONG:

A. Well, the construction will demonstrate certain technical problems in putting these things together and designing them, but to determine whether they will operate requires tests with very high temperature materials, and the sort of tests that are being done on a relatively small scale in National Laboratories at Sandia, for example, and developments are taking place in those areas which have some bearing on the way high temperature materials behave.

But it's unlikely that a particular reactor would have its core catcher called into play, so that as for providing operating information for the future program, that would be unlikely.

Q What is your conclusion regarding the use of a core catcher design as an alternative for Clinch River Breeder Reactor? Is it a substantially better alternative --

MS. FINAMORE: Objection. I don't think this is properly within the scope of redirect examination.

MR. MIZUNO: I believe that it is. We had a series of questions involving the core catcher and its use in Clinch River, whether they have been demonstrated, and its testing.

I think this is a --

MS. FINAMORE: The question of whether it's substantially better could have been put in the testimony originally.

It doesn't relate to the questions that were asked, which had specifically to do with whether or not you could get useful information from construction and testing.

19-8

MR. MIZUNO: I think that Counsel has stated it precisely.

She asked a question regarding whether Clinch River would bring about more information.

All I'm asking the witness is given his answer to that, does it represent a substantially better alternative.

MS. FINAMORE: That was not the question.

JUDGE MILLER: In cross-examination Counsel did go into the question of whether you could judge the operational capabilities of a core catcher until the core catcher catches core, and now, I think, this is just a refinement of that.

He's asking now it's relationship, if any, to the substantially better matter. So you may answer the question.

WITNESS LONG: Well, you have to evaluate it in several -- There are several different ways in which you could evaluate it.

One is would it provide -- If it were added in addition to the features now present in the CRBR, would it add substantial information beyond that; and there I think the answer is no, pending the outcome of our safety evaluation, of course.

We will review the present containment purge vent system and TMBDB features which are represented to us as being sufficient to cope with core disruptive accidents.

The further addition of some other device would add little to that.

Then I think you have to also consider suppose you had the core retention device and not the other feature, would it be reliable enough that you could depend on it solely, and that's in doubt.

So there is a possibility that you would still need the containment purge and vent features.

In sum, we don't see that the core catcher would contribute much additional information to the LMFBR Program beyond what we can get from the design as we look at it as present.

MR. MIZUNO: No more redirect.

JUDGE MILLER: Recross?

MS. FINAMORE: No.

JUDGE MILLER: Okay. Judge Hand?

JUDGE HAND: No, thank you.

JUDGE MILLER: Judge Linenberger?

BOARD EXAMINATION

BY JUDGE LINENBERGER:

Q Mr. Becker, with apologies to you, we do have your professional qualifications available from an earlier session. I don't have that with me and I don't

19-10

remember them.

I would just like you to briefly summarize what you consider to be your qualifications by reason of training or experience to discuss the steam generator considerations that you discuss in this testimony.

BY WITNESS BECKER:

A. Well, I think the most immediate qualifications is that I am the reviewer with the principal responsibility on the hydraulic systems, as far as the safety evaluation is concerned.

My biography beyond that, I was the plant manager of the SEFOR reactor, which was a liquid metal cooled fast system that did not have a steam generator.

The heat was rejected, as in FFTF, directly to the atmosphere.

I have worked on steam-cooled, super-heated steam-cooled reactors, boiling water reactors.

I have done background -- or I have done calculational work in gas-cooled reactors.

I have been in the nuclear business for 25 years.

I think that's a thumbnail sketch of my biography.

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Q What is the field of your professional training?

BY WITNESS BECKER:

A My professional training was physics, engineering physics.

Q Your testimony seems to indicate that whereas prior experience with the use of sodium and heat exchanger relates to systems, that are related to systems that don't look very much like those proposed for the Clinch River Plant, that the basic experience is meaningful and applicable and that one should not be too concerned about the lack of this look-alike aspect.

From that I would infer that a significant amount of analysis has gone into prior experience to assure that the conceptual approach and the detailed design approach of Clinch River is a reasonable one.

Now is this the kind of atypical effort that the Staff has performed or are you only monitoring -- is the Staff only monitoring what others are doing in this field?

BY WITNESS BECKER:

A Predominantly, this is a monitoring of what is done in the field.

The development program of the steam generator for the Department of Energy. Other steam generator performance and experience with steam generators. Sodium steam generators.

Q The answer to Question 51, appearing on Page 28 of this testimony, interests me.

I will intentionally put that information in a light that illustrates my question, rather than illustrates the truth of the matter.

But what that answer says to me is that adopting the heterogeneous core design is desirable because it will give more meaningful information in terms of core physics kinds of problems than a easier to calculate homogeneous design and I can certainly see it easier to model analytically, a homogeneous core than a heterogeneous core, but the objective is not just the learning more about the core physics per se, so in the context of the overall objectives of the Clinch River Plant, why does a hetereogeneous core make sense compared to a homogeneous?

BY WITNESS LONG:

A I guess I was responsible for that answer and your point is well taken, that it's -- if it were only an exercise in physics calculations, that would not justify a unusual geometry, but it turns out that by taking advantage of really an additional variable, that you're able to incorporate into the core design, by

20-3

making it in various sections, you can improve the sodium void co-efficient and perhaps other parameters of the core operation.

So, it does provide the designer with options that can lead to improvements and in my own view, whatever improvements and refinements can be built into future LMFBR's, will probably be exploited and this will -- this project, CRBR, will provide useful information leading to the exploitation of these geometries in the future.

Q I hear you. I haven't heard you say anything that would indicate you think a heterogeneous core will yield a significantly higher breeding gain.

For example, do you think it will?

BY WITNESS LONG:

A. Yes, I do.

Q Is this opinion on your part?

Is this an opinion on your part or is it the result of any analysis you or the Staff have done or is it based on information given to you by Applicant?

BY WITNESS LONG:

A. I think it was first called to our attention that heterogeneous designs could provide better breeding ratios before the option was adopted in CRBR.

Analysisist have calculated this for some time however, we on the Staff have not made an independent study of that, as far as I know, but in the case of CRBR we feel that the Applicants calculations are consistent with what we had been led to expect by previous calculations and I should also add a word of caution, that when you increase the breeding ratio by this device, you do increase the core inventory and that slows down the doubling time.

Overall, the Applicants figures indicate that there is a gain and I believe that such gains will be exploited in the larger reactors.

Q. Have there been experiments performed in critical experimental facilities, such as ZPPR or others that have compared homogeneous with heterogeneous core configurations that lend credence to the conclusion that the breeding gain may be improved in a heterogeneous core design?

BY WITNESS LONG:

A. Yes, there have.

Q. Are those reported somewhere in a -- are they reasonable available documents such as I, for one, just with my own curiousity could go review these, take it from a library somewhere, perhaps?

BY WITNESS LONG:

A. I don't have that information with me but I believe that the reports could be made available.

Q Do you know whether ZPPR itself has done any experiments of this type?

BY WITNESS LONG:

A. Yes.

The critical assemblies can do experiments which provide a lot of information on the breeding ratio.

But operating at the low power that they do, they have a certain amount of uncertainty in the breeding ratio that they calculate and there's also the limitation that they use fuel that's not burned up, so, in a sense, what you get is the breeding ratio at beginning of cycle.

For that reason, CRBR itself will augment the information we get out of critical assemblies.

But there certainly has been a test to determine the breeding ratios for critical assemblies.

Q While we're on the subject of the heterogeneous core, let's go to the bottom of Page 14, the last paragraph there, which speaks with respect to the heterogeneity of the design of the blanket region, the availability of a new latitude in adjusting the sodium-void coefficient, Doppler coefficient and breeding characteristics?

Let's take those one at a time.

Phenomenologically, explain to me how this parameter heterogeneity of the blanket region, I presume is what that refers to, permits latitude in adjustment of sodium-void coefficient.

BY WITNESS LONG:

A. Is it clear that I was talking about placing the internal blanket within the core?

Q. I understand that, yes.

At least I took your words literally, where it said "in the core" meaning not outside.

BY WITNESS LONG:

A. Yes. That was my intention.

It's hard to explain the sodium-void coefficient without just saying that it's a complicated calculation and you use the multi-group cross sections that's been calculated.

But there is one aspect of it that's physically graspable and that is that you have, in a sense, removed that portion of the core which normally has the most positive sodium-void coefficient.

That is the center of the core, that normally has the most positive sodium-void coefficient and you place blanket material in that region so that changes in worth of the sodium are not so important there.

Q. Are you saying, then, that by placing this blanket region within the core, that you are going from a more positive sodium-void coefficient to a less positive sodium-void coefficient?

Or do you really change the sign of it in the case of Clinch River?

BY WITNESS LONG:

A. Overall, it's still positive, but less than it was before. Substantially less.

Q. Why do you believe that?

BY WITNESS LONG:

A. Why do I have the confidence in the calculated values?

Q. Yes.

BY WITNESS LONG:

A. They are substantiated by critical experiments in the ZPPR where you can take sodium out and measure the changes in reactivity.

Q. So-called danger coefficient --

BY WITNESS LONG:

A. Yes.

Q. -- kinds of measurements?

BY WITNESS LONG:

A. Yes.

Q. Now, again, let me ask the same question except to focus on Doppler coefficient instead of sodium-

20-8

void coefficient.

And don't be afraid to get technical. That is quite all right.

BY WITNESS LONG:

A. It's a long time since I've gotten very technical.

It is associated with changes in the spectrum, the neutron spectrum, and --

Q. Well, let's focus in on that.

Changes in the neutron spectrum. Is the spectrum hardened or softened by introducing internal blanket?

BY WITN$SS LONG:

A. Well, it changes locally throughout the core in different places but an overall softening of the spectrum would be expected to improve the Doppler coefficient and I presume that's what happens.

Q. Do you happen to know approximately what range of neutron energies are effective with respect to the captured residence of interest to Doppler broadening?

BY WITNESS LONG:

A. Well, it's especially effective in the range below about three kilovolts.

Q. Okay, now, then, the same question with respect to the third element there, breeding characteristics?

BY WITNESS LONG:

A. There, I think it's just a question of bringing fertile material in close proximity to the neutrons at the point of their birth, so that you get capture in the U-238, which will augment the breeding.

Q. Okay, but somewhat simplisitically I can say, that if I take fuel out of the center of the core to make room for fertile material, that rather than having the flux peak near the axis of the core, I might expect a flux depression near the axis of the core, so if I put a blanket in there, the fertile material is going to seek a lower neutron density and so looking at it that way, that view of it, I don't see it being consistent with your answer.

But I must admit I don't understand these things.

BY WITNESS LONG:

A. I see what you mean and my understanding of it is, that it takes careful design and you have to make the blanket region small enough so that it is, in effect, relatively transparent to neutron.

If it were larger than the effect that you say would predominate, it would not be an effective breeder.

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But, in a fast reactor, regions up to an inch or two in dimensions are relatively transparent to neutrons, so the breeding is effective in these internal blanket regions.

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BY JUDGE LINENBERGER:

Q The second sentence -- the second word in the next sentence is "efficiency" as one of the three things that could be better optimized by going to a heterogenous blanket region within the core.

What's the -- In what context is that word "efficiency" used here? What does it mean?

BY WITNESS LONG:

A Just a moment.

I can't think specifically why I put that word in there. It -- I may have had in mind that it would improve the -- all of these features mentioned above, the sodium void coefficient, the Doppler coefficient and the breeding ratio, which work together, provide some improvement in the operation and economics and many of the general features of the performance of the plant.

Q In that sense then, it sounds as though you're using the word "efficiency" in the context that it might have appeal to a utility executive rather than in the context of something of interest to a core physics analytical type. Do you think you meant it --

BY WITNESS LONG:

A I don't see now a connection between that and core physics, per se.

Q So it's not related to neutron economy in any sense, but in overall performance of --

BY WITNESS LONG:

A. Well, I have mentioned that it would improve the breeding ratio. But I think I probably meant here the combination of the various things that would improve -- leading to an overall improved performance.

Q. Presumably, one of the overall -- Well, more than presumably. One of the overall objectives of the Clinch River Project is to attempt to demonstrate that people have somehow built something -- a better mousetrap, if you will, that's going to appeal to the utility industry.

And I presume that part of at least the so-called five-year demonstration program will be looking, in part at least, at things that seem to make this design concept attractive to the utility industry. Is that indeed a correct picture of the situation?

BY WITNESS LONG:

A. Yes, that has been our -- We have given some consideration to this five-year demonstration program.

Q. Well, in terms of what can you extract from it that will make utility objectives come -- executive come knocking on somebody's door and saying, "Where can I sign up for one?"

BY WITNESS LONG:

A. Well, I guess we -- We're not sales people.

Q. Okay. But I'm just trying to get a feeling for what the Staff is focusing on and --

BY WITNESS LONG:

A. Well, we feel that the utility people should have an honest appraisal of what the LMFBR can do -- good or bad.

Q. Okay. Now that's -- Those are certainly two alternatives that face us all the time: Good or bad.

(Laughter.)

JUDGE MILLER: Warts and all!

BY JUDGE LINENBERGER:

Q. In one respect I can say that if the -- for some reason, whatever, Clinch River is built and operated and demonstrated to have a breeding gain of .87, a lot of people are going to be disappointed -- I would think so.

On the other hand, that result, it seems to me, in itself has some value because it says, "Look" -- I would think it would say, "Look, don't go building LPPs or commercial breeder reactors cast in the same mold as Clinch River. You had better look at Clinch River as a way not to build commercial breeders."

Now, there have been a lot of words talked about in reports and in testimony about these programmatic objectives.

Is, indeed, finding out from Clinch River that this is not the right way to approach commercial breeders a meaningful programmatic objective in itself, do you think -- does the Staff thing?

I -- Excuse me. When I say "you" to you gentlemen, unless I say otherwise, I'm talking about the Staff position.

BY WITNESS LEECH:

A. I'll take that.

I believe that -- Obviously whatever information comes out of the program should be of considerable interest to the utility industry, to manufacturers, everybody interested in the subject.

If it comes out .87, there will be a lot of disappointed people; and I suppose it's so unexpected that it's hard to imagine that that will be the result, judging by our review among us. We don't expect that to happen.

Q. Well, what you're saying is that you reject my hypothesis, and you don't have to face up to answering my question.

(Laughter.)

But that's not quite what I was looking for.

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BY WITNESS LEECH:

A. Put your question again.

Q. What I'm saying is: Suppose Clinch River demonstrates that it's the wrong way to go about building a breeder?

Does the Staff consider that that is an important thing to learn and is a worthwhile program objective in itself?

BY WITNESS LEECH:

A. Well, it certainly is an important thing to learn, absolutely. It wouldn't -- If you could anticipate that that would be the result, then it would seem a poor idea to proceed with building it.

Q. I agree.

I thought I heard something from Applicants' witnesses on this morning -- if you gentlemen were here for that testimony -- it was indicated that there's not necessarily one-to-one correspondence between breeding gain and doubling time as one changes core design -- core designs.

Is that generally true? Is the transition from a given breeding gain to -- "transition" is a bad word.

Is the derivation of the doubling time based on a given breeding gain expected to be different in

Clinch River than in a large breeder project, for example?

In other words -- Let me put it a little different way.

Suppose you have a breeding gain of 1.20 in Clinch River. If you look at the formula for compound interest, that says that looking at it strictly that way, you have a 35-year -- I mean 1.02, you have like a 35-year doubling time. If it was 1.2, it would be shorter obviously.

Would you expect for the same breeding gain an LDP that the doubling time would be longer, shorter or the same?

BY WITNESS LONG:

A. All I can recall at the moment is that the equation for converting doubling time to breeding gain, or vice versa, includes -- It's basically a compound interest formula.

It includes the total inventory of material that has to be doubled. It includes the fuel cycle, as well as the reactor. And it includes delay times in cooling off the fuel to be ready for the fuel cycle.

In other words, if a core has a one-year life in the reactor, and there's a one-year cooling time, you have to have an additional core in the pipeline to run

21-7

your reactor during that cooling time and then additional material to run it during the reprocessing time.

So insofar as those things enter into the equation, if they differ, then there will be a different relation between breeding gain and doubling time.

In general, the more dealsy you have in the processing, the slower the doubling time because it increases your effective inventory.

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BY WITNESS LONG:

A. (Continuing) The simple way to answer your question is that everything else being equal, the breeding gain and doubling time have a relationship. But if other things creep into the equation as a result of your going to the larger scale, then there could be a difference.

Q. Are you saying that as far as you view the matters right now, if you came out with a breeding gain of 1.0 for Clinch River, you would not necessarily expect then a higher breeding gain in the LDP?

BY WITNESS LONG:

A. Oh, no. There is a further -- The breeding ratio might be better in the LDP, just because of the spectral changes and core configuration and elimination of leakage -- neutrons.

So you would expect that increase. And then if the inventory -- If the relation between core inventory and total cycle inventory were the same, then the doubling time would be the same, I believe.

Q. Let me ask you the same question I asked the Staff panel this morning. I'll let you decide whether I get the same answer.

The Clinch River Project -- the LDP, the development of commercial breeders is kind of a pleasant way to spend money. And if things work out well, it even contributes to the nation's pride of accomplishment, and that is certainly worthwhile.

But if one looks from a slightly different viewpoint at this whole undertaking, why isn't attempting to say to yourself, if you can demonstrate this to be so -- "The direction the French are going is not so bad, and they've got a lot of the R&D under their belt and paid for it themselves. Why can't we take advantage of this, take off from where they are and save the taxpayers a heck of a lot of money? Why do we have to have a new and different type of system?" Is it pride of authorship or something, or is there really hard, fast engineering design reasons why the Staff things Clinch River is a better approach than any of the foreign breeder development programs?

BY WITNESS LONG:

A. Well, there are a number of other reasons, in addition to the ones you've mentioned of national pride and accomplishment for one country having its own program. And one is the broadness of the technological base that's achieved by having your own engineers engage in the development and implementation of the project. That's an expertise that's hard to purchase.

And my views are formed largely by a GAO report that I believe we referenced in the original Environmental Statement where they indicated a number of areas where international exchanges might have shortcomings.

You don't, certainly, in the extreme case want to buy just a black box from the French and have it operate. You want to have your own understanding of it.

Presumably, that could be worked out. There's also -- There are also political implications in buying something -- It's unlike a sale of an article between U. S. companies -- or between a U. S. company and a U.S. utility, because there are political overtones to it.

And the GAO report concluded that there are pluses and minuses to that approach.

That's about as far as I can say.

BY WITNESS LEECH:

A. Let me add that it seems to me that from the beginning one objective of this whole program has been the involvement of the utility industry. Obviously, for the purpose of not only letting the utility engineers and others participate in having hands-on experience -- let's put it -- but also for -- I suppose selling themselves on the future of this kind of technology.

I don't have a feeling that they could very readily do that, if you simply buy something from the

French.

You might ship utility people to France, I suppose, if you could make that arrangement. But it doesn't seem practical to do.

I don't know whether Dr. Long wants to talk about the difference in the reactor type. This Phenix reactor is a different kind of reactor. The one we have here is an opportunity, I suppose, to carry forward in the loop type a development project which offers a variation in some respects from what the French are doing.

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22-1

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BY JUDGE LINENBERGER:

Q Okay. Well, maybe it's because of the way I phrased my question, and if so, I'll give you one more chance here.

But I don't here either one of you say, "Well, by golly, the technology of this concept is potentially so much superior that we don't care how anybody else does it. It's worth really riding that horse."

Is it not that straightforward a situation in comparison with other programs in other countries?

BY WITNESS LONG:

A You mean that our technology is so much better than theirs that we --

Q That this breeder reactor concept is potentially so much more superior technologically than --

BY WITNESS LONG:

A Than the LWR concept.

Q No, the breeder approaches taken by Germany or France or some of the other countries, that it doesn't make sense to import their technology. We have such a better idea, we should go it ourselves with Clinch River.

BY WITNESS LONG:

A I haven't been able to come to that conclusion.

Q Okay, I was just interested.

There's been considerable discussion of one

22-2

aspect of this program, namely that things be done as expeditiously as possible, and various inquiry into whether as expeditiously as possible as one of several objectives in itself goes down the drain if you run into a roadblock somewhere and have to make a change in something.

Now, from what I've read in the several testimonies presented, I get the impression that as expeditiously as possible is in and of itself kind of a moving target that says we will take whatever time it needs within reason and expedition so long as we come up with something that's safe and -- well, to use your word, Dr. Long, efficient and attractive to the utility industry.

Is that taking things out of context, or indeed, is the time scale of achievement perhaps of somewhat subordinate importance to doing it safely, doing it in a way which will really demonstrate that it breeds effectively, and that it really represents an attractive system to the electric utility industry?

Can you comment on that?

BY WITNESS LEECH:

A. Well, I will take the last of your comment or your question first.

It seems to me that the project has to achieve its -- it has to provide information that will be useful to the over-all LMFBR Program objectives.

If it's necessary because of some safety consideration, for example, to resolve that kind of a problem before you can proceed, then the expeditiously as possible has to adjust in time to meet that requirement.

Certainly, the project has to have a high likelihood of being able to achieve its objectives in terms of information provided by the project.

As expeditiously as possible, to me, means, first of all, and we have said this in my Answer No. 4, that we believe the Clinch River Project is capable of making substantial contributions in all of these objectives areas within the five-year demonstration period.

That's in terms of meeting the informational needs of the program.

Now, that's a judgment made today based on the experience of people like Dr. Long and what we know about the project.

Now, obviously, if something comes up in the safety review that shows something has to be resolved, I'm sure the Applicants are carrying the burden of resolving that as quickly as they can.

If by some remote imagination here, someone were purposely to decide, let's say in the Applicants' organization, not to do something, of course, that would obviously negate the purpose.

Q I guess in essence, then, you are saying that you consider the time scale of accomplishment somewhat subordinate in objectives to --

BY WITNESS LEECH:

A. To achieving the informational needs, yes.

JUDGE LINENBERGER: Thank you. That's all I have, Mr. Chairman.

JUDGE MILLER: Thank you.

You gentlemen may be excused, and thank you.

(The witnesses were excused.)

MR. MIZUNO: The Staff would like to offer, or move that Staff Exhibit No. 21 be admitted into evidence.

JUDGE MILLER: Staff Exhibit 21?

MR. MIZUNO: Yes, the testimony of Paul Leech --

JUDGE MILLER: All right. Any objections?

MR. EDGAR: No objections.

MS. FINAMORE: No.

JUDGE MILLER: It may be admitted.

MR. MIZUNO: Thank you.

(Staff's Exhibit No. 21 was received in evidence, and follows:

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STAFF EXHIBIT # 24

6522

UNITES STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of UNITED STATES DEPARTMENT OF ENERGY PROJECT MANAGEMENT CORPORATION TENNESSEE VALLEY AUTHORITY (Clinch River Breeder Reactor plant)	)	Docket No. 50-537

TESTIMONY OF PAUL H. LEECH, RICHARD A. BECKER AND JOHN K. LONG RELATIVE TO NRDC CONTENTIONS 7(a) AND 7(b)

CONTENTION 7(a)(1)

Q1. Mr. Leech, what is your name, by whom are you employed, and what is your position?

A1. My name is Paul H. Leech and I am employed by the U. S. Nuclear Regulatory Commission as a Senior Project Manager in the Clinch River Breeder Reactor Program Office of the Office of Nuclear Reactor Regulation. A statement of my professional qualifications is attached to this testimony.

Q2. What is the subject matter of your testimony?

A2. This testimony addresses Natural Resources Defense Council (NRDC) Contention 7(a)(1), which states:

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.

Q3. What is the timing of the LMFBR demonstration plant (CRBRP) under the LMFBR Program?

A3. As stated on page 57 of the DOE 1982 Supplement to ERDA-1535 (the Final Environmental Impact Statement on the LMFBR Program), the current timing objective for the CRBRP is to complete its construction "as expeditiously as possible." The Applicants now plan to complete the construction in 1989 and operation of the plant is scheduled to begin in February 1990. Then, a demonstration period of approximately five years is planned to achieve the major objectives of the CRBRP project, which are:

- o to demonstrate the technical performance, reliability, maintainability, safety, environmental acceptability, and economic feasibility of an LMFBR cent al station steam electric power plant in a utility environment; and
- o to confirm the value of this concept for conserving important non-renewable natural resources.

Q4. Did the Staff conclude that the CRBR Project is likely to meet the above objectives in a timely manner?

A4. Yes. This conclusion is based, in part, on the Staff's review of the information provided by the Applicants in Section 1.3 of their Environmental Report and in the LMFBR Program Environmental Statement (ERDA-1535 and DOE Supplement) and, in part, upon the Staff's independent knowledge and experience. The Staff believes that the CRBRP is capable of making substantial contributions, within the 5-year demonstration period, to meeting the informational needs of the LMFBR Program in all of the areas set forth in the above objectives. Further discussions relative to this subject are found in Section 8.3 of the NRC's Final Environmental Statement (FES) and its Supplement 1 and in the testimony (below) of John K. Long.

Q5. Are there alternatives which might lead to more timely achievement of the programmatic objectives for the CRBR than the approach presently being pursued?

A5. The Staff has not identified any such alternatives. However, the Intervenors have suggested two possibilities:

(1) "The GAO has expressed serious concern about Applicant's decision to install untested steam generators in CRBR. (U. S. General Accounting Office, Revising the Clinch River Breeder Reactor Steam Generator Testing Program Can Reduce Risk, GAO/EMD-82-75, May 25, 1982)." "If the untested steam generators prove to be defective after installation at CRBR, the likelihood is very high that achievement of the informational objectives for CRBR will be delayed for a very substantial period of time - perhaps years - or that the objectives will never be achieved." "A more prudent approach of testing the questionable steam generators prior to making the decision to install them might well lead to more timely achievement of the programmatic objectives for CRBR than the approach presently being pursued."[1]

[1] Affidavit of Thomas B. Cochran, in the matter of Clinch River Breeder Reactor Plant, October 19, 1982, paragraphs 17, 18, 20.

Richard A. Becker addresses the above concern later in this testimony.

(2) "The choice of a more appropriate site than the CRBR site for the demonstration LMFBR plant could result in more timely achievement of the programmatic objectives for the demonstration plant. The Licensing Board might find the CRBR site unsuitable."[2]

The Staff agrees that an alternative site might better meet the timing objective _if_ the proposed CRBR site were found unsuitable by the Atomic Safety and Licensing Board (ASLB). However, the Staff's position is that the proposed site is acceptable for the CRBRP, as indicated in the Staff's Site Suitability Report (NUREG-0786) and the Staff's Final Environmental Statement (NUREG-0139) and its Supplement Number 1. If the ASLB agrees with this Staff position, a decision to choose a different site would result in an unnecessary delay of approximately three to four years in plant construction and operation (see the testimony of Paul Leech et. al. relative to NRDC Contentions 5(a) and 7(c)). Consequently, that choice would not be compatible with the timing objective of completing the plant as expediously as possible.

[2] _Id._ at paragraph 21.

STEAM GENERATOR DEVELOPMENT PROGRAM

Q6. Mr. Becker, what is your name, by whom are you employed, and what is your position?

A6. My name is Richard A. Becker and I am employed by the U. S. Nuclear Regulatory Commission as a Reactor Engineer in the Clinch River Breeder Reactor Program Office of the Office of Nuclear Reactor Regulation. A statement of my professional qualifications has previously been received in evidence at Tr. 2480.

Q7. What is the subject matter of your testimony?

A7. This testimony addresses that portion of NRDC's Contention 7(a)(1), which raises issues about the timeliness of accomplishing the CRBR informational goals with respect to the steam generators and the GAO report cited above in A5(1).

Q8. Has the Staff evaluated the likelihood of timely achievement of the informational objectives with regard to the testing of the steam generators?

A8. Yes, the likelihood of the timely achievement of the informational objectives proposed for CRBR has been evaluated relative to alternatives involving extended steam generator testing. It would certainly be imprudent to install an untested steam generator design, since this choice has high technical risks and could lead to delays in the long run. The factors supporting the steam generator design, therefore, have been reviewed to assure that a sound basis exists for the proposed plans.

Q9. Are you familiar with the GAO report on the CRBR steam generator test program?

A9. Yes, it was considered as part of the ongoing review of the CRBR plant.

Q10. What are the conclusions of the GAO report?

A10. The GAO report concluded that the CRBR steam generator design development program did not minimize technical risk and that a more exhaustive test program was indicated. The GAO report cited did not conclude that the CRBR steam generator design proposed was untested.

GAO acknowledged that "some element of risk will always be involved." Furthermore, the GAO technical consultant did not agree with the conclusion[3] of the report and apparently had concluded, as stated in the report, that he was "confident that the steam generator, as currently designed, will operate as predicted."

Q11. Can testing eliminate technical risk?

A11. No, a thorough and well conceived component development program including proper phenomenon, special features and total system testing, can minimize, but cannot eliminate residual technical risk.

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[3] U. S. General Accounting Office, Revising the Clinch River Breeder Reactor Steam Generator Testing Program Can Reduce Risk, (GAO/EMD -82-75, May 25, 1982) pg. 9.

Q12. What is the view of the Staff with respect to the experience with steam generator designs on nuclear power plants?

A12. Steam generators are used on pressurized water reactors (PWR) and LMFBRs. Although there are similarities and the PWR problems should not be ignored, the PWR steam generator should be viewed as different and distinct from LMFBR steam generators because liquid sodium is used as one of the coolants in LMFBR, steam generators.

Q13. What has been the experience with respect to steam generators used on LMFBRs?

A13. Experience is usually categorized in terms of leaks or the absence of leaks between the high pressure water and the sodium. The root cause of the leak, such as design defect, Q/A lapse or operational error, will dictate the consequences or severity of the leak. LMFBR steam generator experience has been mixed. LMFBR steam generators have operated without water-to-sodium leaks, while other plants have had persistent water-to-sodium leaks. EBR-II has operated a steam generator for 19 years without having a water-to-sodium leak. The French demonstration reactor Phenix operated 10 years before experiencing its first water-to-sodium leak. The British PFR and the Soviet BN-350 experienced extensive and persistent water-to-sodium leaks in their steam generators. The U.S. FERMI reactor experienced water-to-sodium leaks during its operating history.

Q14. What conclusions can be drawn from the operating LMFBR steam generator experience?

A14. Several physical configurations have produced leak-free designs. These would lead to the conclusion that careful engineering design, materials selection and control, quality fabrication and full inspection are more important than configuration. Material selection, weld design, accessibility and inspection seem to be common to most of the steam generator problems. Meticulous quality assurance is an absolute necessity in all phases of design, fabrication and operation. It is not alarming that CRBR steam generator does not resemble, for example, the successful EBR-II steam generator. It is more important that the configuration selected be capable of incorporating proper design features and the lessons learned from available steam generator experience.

Q15. Has the CRBR steam generator design incorporated the lessons learned from operating LMFBR steam generators?

A15. Yes, the Staff's ongoing review of the development program and design indicates that the lessons learned from problems with other sodium-to-water steam generators have been understood and assimilated into the CRBR steam generator design. Also, pertinent conclusions from the PWR steam generator experience have been factored into the design.

Q16. Has this been confirmed by component development tests?

A16. Yes, the basic configuration, design approach to welds, inspection, quality assurance, materials, phenomena and stability have all been confirmed in individual effects tests and model tests. From these tests, mechanical corrections for tolerances and materials compatibility were incorporated in the component or system integration test currently in progress. This component or systems integration test is called the CRBR prototype steam generator.

Q17. Were all the design changes included in the prototype steam generator?

A17. No, at the same time that design changes were being made to include corrections indicated by test results, several design improvements were adopted which could not be included in the prototype and still maintain schedule.

Q18. Are these design improvements major changes?

A18. No, these design improvements are minor in nature and are not involved with any of the fundamental aspects of the steam generator concept or structure.

Q19. Will these design improvements be completely untested in the installed CRBR steam generator units?

A19. No, although these improvements could not be included in the prototype steam generator on test, hydraulic testing of a 0.42-size scale model is planned in the future to confirm analytical predictions that there will be no flow-induced vibration problems with these design improvements. Finally, as a confirmation test of the 0.42-scale model tests, the plant spare steam generator will be hydraulically tested. Ten steam generator units are being built; nine of these units will be installed and one unit will be a spare.

Q20. Based on the Staff's evaluation of the development program, will the informational objectives of CRBR more likely be met in a timely way by the more exhaustive testing recommended by GAO or the steam generator development program as currently constituted?

A20. GAO acknowledged in their conclusions that all steam generator problems are not related to design deficiencies, testing cannot eliminate all elements of risk, and the ultimate test must come when the steam generators are operated in CRBR.[4] The Staff and the GAO's technical consultant agree with the applicant that the assurances gained from testing a precise prototype prior to manufacturing the plant units cannot technically justify the delay. Based on the status of our review of the CRBR steam generator design to date, it appears that the technical risk of a major design defect, requiring redesign and lengthy delay after installation, which had gone undetected by testing is very small. The hydraulic testing on the 0.42-scale model and the plant spare unit

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[4] Id., pg. 9.

should disclose any flow-induced vibration problem which has not been theoretically disclosed. Therefore, steam generator problems which might occur would be corrected in place, probably by plant operations personnel and designers working together. Component maintainability in a utility environment is one facet of the informational objective of CRBR so that lengthy delays in contributing to the informational goals seem unlikely from the planned course of action.

The alternative course advocated by GAO would require a precise prototype to be fabricated and tested before contracting for production of the plant units. This would cause a certain delay of at least two years and possibly longer. This is a direct and certain forestalling of accomplishing of any facet of the CRBR informational objective and, therefore, less acceptable.

CONTENTION 7a) (2), (3)

Q21. Please state your name and affiliation.

A21. My name is John K. Long. I am employed as a Reactor Engineer, Reactor Systems Branch, Division of Systems Integration in the Office of Nuclear Reactor Regulation. My involvement in the CRBR review has been to review those aspects of core disruptive accidents involving thermal margins beyond the design basis. A copy of my professional qualifications was received into evidence and appears at Tr. 2533-37.

Q22. What subject matter does this testimony address?

A22. This testimony addresses the issue of alternatives to CRBR with respect to the ability of CRBR to meet its objectives as set forth in NRDC, et. al., Contention 7a) (2) as follows:

> 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.
>
> (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.

Q23. What is the programmatic objective for a demonstration CRBR pertinent to the above contention?

A23. The proposed Final Environmental Statement for the LMFBR Program (WASH-1535), endorsed by ERDA-1535, at p. 3.5-2, states the following relevant CRBR objective:

> "to demonstrate the technical performance, reliability, maintainability, safety, environmental acceptability, and economic feasibility of an LMFBR central station electric power plant in a utility environment," and

In the Final Supplement to the programmatic statement (Supplement to ERDA-1535, dated May 1982, pp. 38-39) DOE states that:

> "Technical feasibility of the LMFBR has been clearly demonstrated and the remaining work is to conduct engineering scale demonstration of the technology at a size leading up to that of commercial plants."

Q24. What specific design features of the 375MWe CRBR are likely to demonstrate relevant information with respect to an economic, reliable and licensable LMFBR (presumed to produce about 1000 MWe)?

A24. There are many LMFBR characteristics that are relevant to economics, reliability and licensability that are not directly dependent on reactor size. CRBR is designed to be an integrated demonstration of many of these characteristics. Among them are the particular combinations of thermal-hydraulic and high temperature properties. There are a number of specific features designed to demonstrate relevant data in this regard. An extensive list has been compiled and presented in the Applicants' programmatic Environmental Impact Statement, Supplement to ERDA-1535, DOE/EIS-0085-FS, p. 61. Among the most important of these are the fuel elements and assemblies themselves; the reactor closure rotating plug seals, bearings, insulation and cooling; in-vessel refueling equipment; and instrumentation and control equipment systems.

Q25. Are any of those features likely to provide relevant information for the LMFBR program, independent of their size?

A25. Yes. A demonstration on the scale of CRBR would be directly pertinent and relevant to a LMFBR of a practical commercial size (for purposes of this evaluation, the practical commercial size is considered to be in the 1000 MWe range).

Q26. What differences are expected between the fuel elements and related hardware of CRBR and those of a practical commercial size LMFBR?

A26. The fuel and assembly hardware of the CRBR could be identical externally to those of a large future LMFBR, but the fuel enrichment would be different. Other features being equal the fuel enrichment is generally less for larger reactors, but heat ratings (kw/foot) are generally the same.

Q27. In view of this difference, what relevance to larger reactors is there in the CRBR design?

A27. Thermal hydraulic effects during normal operation are expected to be quite similar.

The change in enrichment is accompanied by various changes in core physics, including sodium-void coefficient, Doppler coefficient and breeding ratio. These variations are well known, have been calculated for many years for reactors that differ principally in scale, and are important for safety analyses. Each new reactor that comes on line provides data that serves as a check point to verify or adjust and correct the previous calculations. CRBR will provide such a check point and in this fashion will have relevance to the design of larger reactors.

The use of heterogeneous blanket regions in the core of CRBR introduces a variable in the design parameters that has not previously been utilized. The availability of this parameter permits designers a new latitude in the adjustment of sodium-void coefficient, Doppler coefficient and breeding characteristics. Safety, efficiency, and breeding performance can be better optimized by taking advantage of this parameter. Thus CRBR will be relevant to the optimization of the design of subsequent reactors.

Q28. Are the rotating seals, bearings, insulation, cooling, in-vessel refueling equipment and instrumentation/control equipment relevant to future reactors?

A28. Yes. They could be essentially copied with appropriate scaling for later designs. The CRBR experience with these items will therefore be transferable to commercial LMFBRs. Their demonstration on CRBR is therefore extremely relevant for the reactor program.

Q29. What does Contention 7a) (3) address?

A29. Contention 7a) (3) states:

> 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.
>
> (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.

Q30. In what fashion is the CRBR reasonably likely to demonstrate the reliability of a commercial LMFBR central station electric plant?

A30. In a sequence of facilities with scaled-up dimensions, culminating in a large commercial plant, each member of the sequence contributes some data on reliability to be factored into the design of the subsequent members. The direct applicability of the reliability data is increased as the similarity of the general features, design details, and designs of individual components is greater. Although the reliability of any single detail, such as fuel performance, might be fairly well established in test reactors or sodium test facilities, the demonstration of the reliable interaction of components requires a facility in which whole systems can interact with each other. The CRBR is such a facility, tying together the nuclear, steam and electric systems to demonstrate their reliability in the commercial environment of a power grid. This matter is further discussed in Section 8.3 of the Staff's FES and the FES Supplement (FESS).

Q31. Will subsequent plants in the sequence mentioned above be closely similar to the CRBR?

A31. If the LMFBR program as currently formulated is followed, successive plants will be similar to CRBR. The general concept of a loop-type, sodium-cooled, breeder facility is a viable choice for a commercial plant. If this became the choice for the commercial plant, many components would be very similar in materials, principles of operation and temperatures. The commercial plant scale would be different to accommodate larger power outputs in some components. CRBR would provide a data base for future reliability data.

Q32. What would be the relevance of CRBR if the commercial plant were of a different design, such as a pool plant?

A32. Many of the general principles would be the same between the CRBR and the commercial plant, even if the latter were a different design. The maintenance of purity of the sodium, the operation in a radioactive sodium environment, the production of superheated steam of high quality, and the isolation between steam and nuclear systems by an intermediate sodium loop, all have important consequences for reliability, and all are features of both loop and pool systems.

Details of piping, seals and pumping would be different, so that reliability data would not be as effectively generated in these respects, but overall, the demonstration would be extremely valuable.

Q33. Is the CRBR reasonably likely to demonstrate the maintainability of a relevant commercial LMFBR central station electric plant?

A33. The maintainability aspects of the CRBR will have to be divided into those which are related to first-of-a-kind test facilities and those which are related to more routine operations in order to provide useful projections for commercial plants. When this division is made the Staff believes that the maintainability records of CRBR would indeed be valuable input for the decision on commercialization, again provided the loop concept is followed. Maintenance of equipment within the primary and intermediate systems of pool type reactors requires different techniques, and the CRBR experience would be of less benefit if that direction of comerciali-zation is taken. Equipment beyond the intermediate heat exchanger is not fundamentally different in the two systems.

Q34. Are there maintainability data which could not be obtained in a test facility, and which would require experience with a complete working plant?

A34. Yes. Some information to be obtained that would have programmatic relevance includes the economic cost of maintenance, the enforced reduction in plant operating factor, and the personnel hazards involved. Definitive measures of these problems can only be obtained through an actual demonstration under realistic operating conditions.

Q35. To what extent is the CRBR likely to demonstrate the economic feasibility of a relevant commercial LMFBR central station electric plant?

A35. The economic projections for an LMFBR utility plant will be guided by a detailed cost accounting of capital and operating expenses for the CRBRP, after proper corrections for non-repetitive, non-prototypic costs associated with the first-of-a-kind nature of the plant. The project is undertaking a very comprehensive cost-reporting system to provide the information for such an evaluation. The costs reported for the CRBRP will also be adjusted for possible improvements as the scale of plant is increased, in order to provide information relevant to commercial LMFBRs. Such adjustments are determined subjectively and are partly based on other experiences with small scale plants that have later been extrapolated to larger sizes.

Q36. Can this process lead to useful estimates of the cost of future plants?

A36. Yes. Although this process of cost extrapolation is not precise, the cost data from the CRBRP would provide a better basis than currently exists for such estimates. Without CRBR, the degree of extrapolation would be considerably larger.

Q37. How can CRBR make a significant contribution to the demonstration of technical performance of a relevant commercial LMFBR?

A37. If the full rated technical performance of CRBR is achieved, it will serve as an effective confirmatory demonstration in the sequence leading to full commercial sizes.

However, the achievement of full rated technical performance in CRBR is not essential to provide technical performance information relevant to the LMFBR program. Negative information from CRBR could be factored in as corrections to the LMFBR program, or partial technical performance in CRBR could provide a data base for further improvements in the program.

Q38. Based on past experience, is it expected that the energy conversion systems of LMFBRs are more likely to present problems in demonstrating technical performance than the nuclear or control systems?

A38. Yes. For example, steam generators have presented problems in some foreign reactors.

Q39. If there are problems with the energy conversion system which prevent the attainment of full technical performance, would the value of CRBR as a demonstration of a central station power plant in a utility environment be seriously compromised?

A39. It is expected that deficiencies in the energy conversion system, if any should develop, would be correctable within a reasonable time, as they have been on other LMFBRs, without modification to the nuclear system. This is due to the almost completely non-radioactive nature of the secondary sodium and steam systems. Thus, deficiencies in the conversion system would be likely to delay, but not prevent, the demonstration of full technical performance in a utility environment.

Q40. What evidence is there that a high level of technical performance can be achieved?

A40. The evidence is in the successful record of construction and operation of LMFBRs in the U. S. and in foreign countries.

Q41. To what extent has world-wide fast reactor experience demonstrated the technical performance of a relevant commercial LMFBR central station electric plant?

A41. The experience from demonstration reactors in foreign countries is considered in evaluating the potential of the CRBR as a contributor to the ability to predict the technical performance of commercial LMFBRs. The record of performance of the major breeder reactors is considerable and it is impressive. Except for major shutdowns in 1977 for intermediate heat exchanger repair, Phenix has operated intermittently ~~continuously~~ from 1975 until the present. The Prototype Fast Reactor (PFR) operated intermittently ~~continuously~~ from 1977 to the present, except for one major shutdown of about 8 months for steam generator repairs. BN-350 has operated extensively since 1973. BN-600 commenced operation in 1980. Japan has placed its JOYO in operation and has broken ground for its successor, MONJU.

Q42. To what extent is the CRBR reasonably likely to demonstrate the environmental acceptability of a relevant commercial LMFBR central station electric plant?

A42. The ability of the CRBRP to demonstrate environmental acceptability of LMFBRs will depend in large measure on the scalability of impacts resulting from its construction and operation. The various LMFBR concepts are not expected to have substantially different radioactive effluent generation from one another; the Staff therefore believes that the demonstration results provided by the CRBRP will be scalable with minor modifications, to any of the future LMFBRs now proposed. All LMFBRs would have an inert cover-gas system in conjunction with the sodium coolant, and all concepts would include systems to clean up the radioactive contamination in this cover-gas. Moreover, the conditions encountered by these systems in ~~containment~~ contamination control or release are not substantially different among the various designs.

All LMFBRs will have to restrict and control the release of tritium. As has been demonstrated, much of the tritium is retained in the system cold traps. The quantities of tritium produced are somewhat design dependent, but they are not so different among the various designs that the demonstration provided by CRBRP would be inapplicable if another design concept were adopted.

The other considerations of environmental impacts of the CRBRP, which are discussed in Chapters 4 and 5 of the FES and the FESS have been reviewed by the Staff and no items have been found which could not be scaled to larger LMFBRs, or modified slightly to accommodate different LMFBR concepts. The Staff therefore finds that the CRBRP would provide a useful demonstration of the environmental impact of liquid metal fast breeder reactor technology.

Fuel cycle and waste disposal aspects of LMFBR technology are the subject of separate studies which will include the environmental impact of the balance of the cycle. The entire impact of the LMFBR program will be estimated by DOE using all available sources of information. The CRBRP is capable of making a significant contribution to this study.

Q43. To what extent is the CRBR reasonably likely to demonstrate the safety of a relevant commercial LMFBR central station electric plant?

A43. There are several safety areas where the CRBR would make a significant contribution. The first of these is a demonstration of safe operation of an integrated system. Although all of the components of the CRBR would be of a quality that can be regarded as safe individually, the demonstration of their performance in the total system would provide additional confidence that they would all work together in a satisfactory manner, and, consequently, that similar large scale systems can also be made to work.

Secondly, initial operation of CRBR can confirm natural circulation predictions that have been developed from tests on smaller systems like FFTF. This will provide a further bridge for extension of the natural circulation calculations to still larger systems.

However, the entire objective of demonstrating the safety of LMFBRs will not be achieved merely by safe operation of CRBR. Although a satisfactory record of performance based on (1) reliable operation of systems and components important to normal safe operation, and (2) the effectiveness of measures to control off-normal events should they occur would be encouraging, it would not provide a direct indication of the total safety of larger LMFBRs. Much of the safety program relevant to the larger reactors is being carried out in separate studies. These are being done in reactor test facilities and in out-of-pile-tests.

The CRBR core design has been modified to include internal breeding blankets. This introduces a degree of heterogeneity that complicates the analysis of bowing, Doppler, and local reactivity effects. The CRBR in its current heterogeneous design will be a valuable demonstration of the ability to calculate such complex fast reactor systems.

Still another area in which the CRBR will provide a large scale demonstration of a safety feature is in the core clamping and support design. There has been no way of demonstrating on an engineering mockup the full combination of thermal and hydraulic effects that influence the expansion and bowing behavior of the fuel elements and assemblies in a reactor the size of CRBR. FFTF testing and data in this area may not be directly applicable to CRBR due to the size difference and the fact that FFTF is a homogeneous core. Elaborate calculations of this type of behavior have been done to supplement an engineering test program, but the actual behavior of the reactor is required for final validation of the engineering predictions. The additional effects of irradiation on fuel assembly behavior, through irradiation swelling and constrained creep, will also be demonstrated. These effects are essential to calculations of power coefficient and transient behavior, and are thus safety related. Although these will also be demonstrated by FFTF, experience with the CRBRP will permit a demonstration of these phenomena on a scale that can be extrapolated to commercial plants. Thus, the CRBRP can make a significant contribution to knowledge of the safety of LMFBRs by narrowing the uncertainties in component and system behavior that now exist.

Q44. What is your conclusion regarding the NRDC, et. al., Contentions 7a) (2) and (3)?

A44. It has been shown that the CRBR is reasonably likely to provide information relevant to an economic, reliable and licensable LMFBR of a practical commercial size (assumed to mean about 1000 MWe) with respect to technical performance, reliability, maintainability, safety, environmental acceptability and economic feasibility. I therefore conclude that the Staff analysis has been adequate to demonstrate that the CRBR as now planned will achieve its objectives of generating information relevant to practical commercial LMFBRs.

<u>CONTENTION 7b)</u>

Q45. Dr. Long, what does NRDC, et. al., Contention 7b) address?

A45. Contention 7b) states as follows:

> 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.

Q46. How might the informational requirements of the LMFBR program be served by the incorporation of alternative features in the CRBR?

A46. The informational needs of the LMFBR program will best be served by incorporating in CRBR, as far as practical, systems that are similar to those most likely to be chosen for use in the LMFBR program.

In general, the CRBR has been designed to incorporate those features which in the present judgement of its sponsors, designers, consultants and DOE participants will be most likely to appear in the LMFBR program. The consideration of alternatives for additional informational purposes therefore is related to a second, less likely, level of informational need.

The required information relative to candidate alternative features to meet this less likely level of need may be obtained in various ways. The features may be studied in research and development programs out-of-pile, in reactors other than CRBR, or in CRBR itself. Since the primary objectives of CRBR as a generator of information for the LMFBR programs are served by the present design, it would be detrimental to the program to require the incorporation into CRBR of alternatives to meet this secondary need unless they were:

1) clearly necessary in the LMFBR programs, and
2) fully developed to the extent that their incorporation in CRBR would not delay or jeopardize the primary informational mission of CRBR.

The Staff considers that both of the above requirements would have to be met in order to justify the incorporation of an alternative feature into a design that already meets its primary informational goals.

Q47. Has the use of self-actuated shutdown systems been examined against the above informational goals?

A47. Self-actuated shutdown systems are not considered to be an essential need for the LMFBR program. The Staff knows of no reason why more conventional shutdown systems could not satisfy the safety requirements of the program. The development of these systems can be continued on an out-of-pile mode in the event that they are later determined to be needed.

Self-actuated shutdown systems have not yet reached the stage of development in the United States that would permit their use in the CRBR. The second level information that would be obtained through their use in CRBR does not justify a requirement for their use at this time.

Q48. Has the Staff examined the possible value of a core retention device in generating information for the LMFBR program?

A48. The installation of a core retention device in CRBR would not likely generate any useful operating data for future reactors. since the probability of its being called into use is extremely low. Even if the probability of its being called into use were as great as 10^{-3} (and the Staff believes this to be too large a figure) we might have to wait hundreds of reactor operating years to see how it functions. Obviously any informational requirements in connection with core retention devices will have to come from out-of-pile studies, not from CRBR.

Q49. Has the Staff considered the informational value of installing a large flywheel on the primary sodium pumps.

A49. Each reactor is designed so that the coastdown characteristics of the flowing sodium provide a sufficient match to the shutdown cooling requirements to prevent fuel overheating in a sudden shutdown. In the absence of an added flywheel, there is still a large inertia in the flowing sodium itself. The Applicants claim that this is sufficient, in the case of CRBR, to provide the necessary coolant coastdown period, and the Staff has this matter under review.

It is not known whether or not the coastdown would need to be augmented in some way for the larger LMFBRs, but it is not anticipated that augmentation would be a serious problem if the need arises. Flywheels can be installed on the motor-generator sets outside the containment, as is planned for the Super-Phenix reactor, or the stored energy to furnish additional coastdown flow can be provided in other ways. The Staff does not consider that the installation of such fairly standard devices on CRBR would augment the information generated for the LMFBR program in any significant way. The decision whether or not to use them should therefore be based on the coolant flow-coastdown requirements of the CRBR itself and not on a need for information for other reactors.

Q50. Has the Staff considered whether the use of a fully contained system would significantly augment the informational input to the LMFBR program as compared to a filtered-vented containment system?

A50. There are many fully contained systems in existence and relatively few filtered vented systems. Provided that the CRBR filtered vented system can be designed to satisfy the safety and environmental requirements, its design, construction and test operation will provide new information with greater potential for value in the LMFBR program than would the construction of another conventional containment.

Q51. Has the Staff considered the informational value to the program of building the CRBR with a geometry that would provide an improved reactivity coefficient associated with thermal expansion?

A51. The thermal expansion of the sodium in the heterogeneous core is calculated to be associated with less positive components of reactivity than in the homogeneous core. It is anticipated that the detailed verification of this effect in CRBR will provide information of considerable value to the LMFBR program. On the other hand, the construction of a homogeneous core as originally proposed would only provide a verification of coefficients that are considered to be more straightforward to calculate, and which have been verified in other reactors. Clearly, the information value for the LMFBR program is greater with the heterogeneous core than with the alternative homogeneous core originally proposed.

Q52. Has the Staff considered the informational value to the program of building the CRBR with a pool rather than a loop configruation?

A52. The informational value to the LMFBR program is greatest if the CRBR configuration is similar to the configuration of the larger reactors that follow. These larger reactors are presently expected to be of the loop type, so that the loop choice for CRBR maximizes its expected information value to the program. If the larger reactors should finally choose the pool rather than the loop design, the CRBR demonstration would provide less relevant information to the program but would nevertheless provide sufficient information to be of considerable value, as indicated under Q32 and in the FES.

Q53. What is your conclusion in regard to NRDC, et. al. Contention 7b)?

A53. The Staff has examined the information value to the LMFBR program or to a large demonstration-scale facility of various alternative design features such as are embodied in certain foreign breeder reactors and has found that they do not provide substantially better satisfaction of the informational requirements for the larger reactors than does the present CRBR design.

PAUL H. LEECH

PROFESSIONAL QUALIFICATIONS

I am presently employed by the U. S. Nuclear Regulatory Commission as a Project Manager in the Clinch River Breeder Reactor Program Office of the Office of Nuclear Reactor Regulation. My specific responsibility is to manage the NRC's environmental review of the application to the Commission for a permit to construct the Clinch River Breeder Reactor Plant near Oak Ridge, Tennessee. I had that same responsibility during 1975-1977.

My formal education was obtained at: San Jose (California) State College (pre-engineering, 1939-40); University of Colorado, Boulder, Colorado (B. S. degree in Electrical Engineering, 1943); and Columbia University, New York City (courses in psychology, world trade, literature). Short courses sponsored by various employers included the following subjects: electrical design; management, underground power transmission; ecosystems; nuclear power and environmental assessment; environmental quality and natural resources; PWR Technology.

After graduation from the University of Colorado, my initial experience was predominantly in the application and sale of electrical apparatus, analyzing and reporting technical developments and experience in the electric utility industry, and analysis of the environmental effects of all types of power plants and power transmission and distribution systems.

Beginning in 1945, I was employed for 13 years by the General Electric Company in various assignments related to the design of electrical products and their applications in industry.

Beginning in 1959, I was employed for eleven years as the Western Editor of Electrical World, a technical trade magazine published by McGraw-Hill for the electric utility industry. In this capacity I specialized in the fields of electric power transmission and distribution, system engineering and power generation.

During 1971, I was employed for eight months in the Bechtel Corporation Power and Industrial Division as a senior engineer concerned primarily with environmental effects of nuclear power plants. In September of that year I left Bechtel to accept a position with the Atomic Energy Commission's Office of Regulation (now the Nuclear Regulatory Commission).

I have served the Commission primarily as an environmental project manager for preparation of environmental statements on various applications for construction permits and operating licenses for nuclear power plants, including: Fort Calhoun Station near Omaha, Nebraska; Millstone Power Station at Waterford, Connecticut; Surry Power Station and North Anna Power Station in Virginia; Skagit Nuclear Power Station in Washington; and the Sundesert Nuclear Plant near Blythe, California. I was also the environmental project manager for preparation of the Programmatic Environmental Impact Statement related to decontamination and disposal of radioactive wastes resulting from the March 28, 1979 accident at Three Mile Island Nuclear Station Unit 2. In addition, I served briefly as the licensing project manager for review of the Pebble Springs Nuclear Plant in the State of Oregon.

22-5

JUDGE MILLER: All right. I guess that probably concludes, does it not, the testimony for this phase of the LWA-1 evidentiary proceeding.

We have asked now that the next matter to come before the Board be arguments of Counsel in the nature of closing arguments upon the issues, and that you go into them in sufficient depth to contain the equivalent, at any rate, of proposed findings of fact.

We have asked Counsel to consider, to discuss among themselves which contentions they wish to address in what order, considering that they may be clustered by number in order to get to the appropriate subjects which have some logical connection.

Since the Applicants bear the over-all burden of proof, I suppose they should be given the opportunity to go first in that regard.

MR. SWANSON: Could we take a short break before we get started on this?

JUDGE MILLER: All right. We'll take a short recess.

(Recess taken.)

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JUDGE MILLER: Mr. Edgar, which contentions do you wish to address first?

MR. EDGAR: Let me give the Board a sequence. We have discussed it with the parties and it seems to be agreeable.

We may find ourselves, because we haven't reviewed each other's arguments, that we have a little disconnect, but we think we understand.

The first set of things would be the Contention 1, 2 and 3 set of things.

JUDGE MILLER: One, two and three?

MR. EDGAR: Yes, sir, and now I'll explain. It's really a sub -- it is a major part of, but not all of, 1, 2 and 3 together, and it's your issues of whether the CDA should be a DBA and whether the containment would maintain doses below the dose guidelines, and the issues that revolve around those two central points.

I can introduce some of this in argument.

Let's call that one broad category at the outset.

Then the next category would be what has been labeled the "Environmental Effects of Accidents." It's basically --

JUDGE MILLER: What are the contention numbers, first?

23-2

MR. EDGAR: This is what I am going to explain. It is a set of portions of Contentions 2 and 3.

JUDGE MILLER: Okay, and those portions dealing with what issues now?

MR. EDGAR: Well, that really gets involved with the risks of beyond design basis accidents or hypothetical core disruptive accidents, and the Staff's Appendix J analysis.

MS. FINAMORE: If I may interject just one minor point. The Intervenors have included a portion of Contention 1 in that section, also. The Applicants and the Staff have not.

MR. EDGAR: The third subject would be Contention 5(b), which is the diffusion plant in Y-12, the effects on the diffusion plant in Y-12.

The fourth subject would be Contention 2(e), which incorporates Contention 11(d)(1) and (d)(2), and has been or relates to the subject of the 10 CFR 100.11(a), dose guidelines, the dose guideline values for site suitability analysis.

The next subject would be 11(b) and (c), genetic and somatic effects.

The sixth subject would be Contention 4 and 6(b)(4), relating to Safeguards.

23-3

The next would be Contentions 6(b)(1) and (b)(3), fuel cycle.

The next and eighth subject is Contention 5(a) and 7(c), alternative sites.

The ninth and final subject would be 7(a) and (b) which is the testimony just considered relating to design alternatives.

JUDGE MILLER: Now, for convenience of handling, is it your thought that the parties would take up sequentially each of the Roman numerals over all or cluster them or what?

You've already clustered in part, of course, by issues.

MR. EDGAR: Yes, sir. Now what we've got here is an understanding that we would go issue by issue. We have identified nine sets here.

There is generally a relationship between the issues in terms of how they are located to the next nearest neighbor.

The two that involve the health effects area tend to go together. The accidents items tend to go together.

Safeguards and fuel cycle tend to go together, and alternatives tend to go together.

I'm not sure there's much more that can be done

23-4

in terms of lumping, and to do it in a way that gets the issues sifted out.

JUDGE MILLER: That's all right.

Anything else?

MS. FINAMORE: I have one additional matter that I'd like to go into before we begin.

The Board may recall, the first day of this proceeding we ran into a discussion on additional discovery concerning the gas sparging, and just a few minutes ago we were provided some documents by the Applicants in response to our request.

We would like to introduce two of them into evidence, if we may at this time.

Applicants --

JUDGE MILLER: I don't know what they are or whether or not there's any objections. Outside of that, I'm fully prepared to rule.

MR. SWANSON: Staff doesn't even know what they are. We haven't seen them.

MS. FINAMORE: Yes, I was just going to say I can show them to Staff at this time. We did just get them ourselves.

If I could just read them to you. One of them is entitled, "Clinch River Breeder Reactor Project: Postulated Accidents, Offsite Dose Estimates," and it

23-5

contains a list of the data behind the Applicants' estimates of the doses in its testimony for HCDA Class 2.

The second document we were just handed I believe goes along with that document. It's entitled, "Worst Sector X/Q's," and it gives X/Q values used by the Applicants in its analysis of HCDA Class 2 doses for 5 percent and 50 percent, 16 hours, 30-day, 26-day and annual average.

If I could, I'd like to introduce those as -- have them marked for identification and introduced as Intervenors' Exhibits -- I believe 23 and 24.

(Intervenors' Exhibits Nos. 23 and 24 were marked for identification.)

JUDGE MILLER: Is there any objection?

MR. EDGAR: Your Honor, we made those documents available as an accommodation. We received a request for information at the lunch hour on the 14th.

We think we are entitled to present our own case in the way we see fit.

That is underlying information as to our calculations. We see no necessity for introduction into the record.

If this were the normal procedure, why, I suppose that anybody could just simply say that, "Here's an underlying document from Dr. Cochran's testimony. We'll throw it all in," without prop. · foundation, without any discussion on the record.

I see -- we'd oppose admission of the documents.

MS. FINAMORE: The request was for the information. That's the underlying documentation for a particular answer in Applicants' testimony, and that discussion is on the record.

This document was provided to us in specific response to that request. I think it's more than relevant, since it constitutes the underlying documentation for these calculations.

JUDGE MILLER: The question isn't one of relevance. The question is that you asked for information.

There was a discussion as to whether you could have or should have obtained it before as a matter of discovery in a period prior to the commencement of the hearing.

I think as a courtesy to you we said that you could probably obtain the information by making a request which you should have made earlier on the filing of the direct written testimony, November 1, which was the evidentiary aspect which brought the matter to the Board's attention.

You have been supplied the information. We

23-7

haven't made any rulings that went beyond that.

MS. FINAMORE: We were more than willing to cross-examine the witnesses on that information. However, it was not provided to us until this moment.

We are perfectly willing to cross-examine them on it right now if they are here.

JUDGE MILLER: Well, the point is, we are closing the evidence. I'm about ready to close the record on the evidence.

We are about to take up closing arguments and you tell me what you would have done. So we could have some ham and eggs if we had some eggs if we had some ham.

MS. FINAMORE: Yes. Unless we are able --

JUDGE MILLER: You are trying to tell us to go back to November 1, things that should have been done then in order to get you to wherever you wanted to be in the course of the trial.

We are not going to go backwards and start taking more evidence.

MS. FINAMORE: But there was no function in providing us the information, then. It doesn't relieve our prejudice in any way, unless we are able to use it now.

JUDGE MILLER: Your prejudice was self-induced. Your prejudice was self-induced, which was what was our ruling, that you should have requested it, and if

23-8

you weren't able to get it, to file a motion with the Board to obtain discovery, as our rules provide and as our practice is, as of November 1, not as December 13, 14, 15, 16.

MS. FINAMORE: We feel this information is extremely important to this case now and we are very prejudiced if we are not able to introduce it in the record now.

JUDGE MILLER: Well, I think that the importance of it is directly related to your dilatoriness in seeking to obtain it.

MS. FINAMORE: Again, I state it was after the close of the discovery period. We were relying on the Board's ruling.

JUDGE MILLER: Now, look, let's get this straight. You know our rules on discovery. The rules themselves provide that discovery may be reopened for a good cause.

That's just routine. You know, also, that any lawyer can file a motion, and under the Comanche Peak procedure, which we made known to you for months and months and months, that you are requested to confer with your colleagues and then to file motions indicating results of the conference.

Now there's nothing new and you are trying to

23-9

make something sound like it's unusual when it's routine.

MS. FINAMORE: For that very same reason is why I'm requesting that the Board allow us to introduce this evidence at this time, even though it is outside the receipt of other testimony, because we feel there is good cause because we only received it --

JUDGE MILLER: Good cause refers to pretrial discovery. So don't talk about good cause for pretrial discovery as though it gives you some right at the end of the evidentiary hearing and the commencement of closing argument to go backwards in time.

MS. FINAMORE: Well --

JUDGE MILLER: Now you are a lawyer. You understand these things. I can't believe that you are trying to distort the effect of our rules for some unknown reason.

MS. FINAMORE: No. We just feel in this situation --

JUDGE MILLER: All right. Your feelings are overruled. Your motion, if it be a motion is denied. You have the information and I trust will make appropriate use of it.

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23-10

JUDGE MILLER: The evidentiary record now is closed.

We are now going into the closing arguments and you know what the schedule is.

The 24th you are to file in writing your proposed findings of fact and conclusions of law.

We propose, however, and I've told you in advance we want you today to go in depth into the matters, the underlying facts and the references to transcript pages, to exhibit numbers and the like, so we will know what the controversies are on this record between and among the parties, the bases for their own affirmative contentions, the way in which they respond for the consideration of the Board as decision maker to the arguments and proposed findings of their adversaries.

Now, we are on the verge of that.

Is there anything further?

MR. SWANSON: Could I just get a clarification?

JUDGE MILLER: Yes.

MR. SWANSON: The Chairman just indicated the record was closed, and just so we are absolutely clear, since the Board approved or gave authorization for a witness in this proceeding, Dr. Cochran, to make argument, that in fact any statements he makes from here on are that, argument, and they may not be cited as evidence in this

23-11

proceeding.

JUDGE MILLER: I think that's very abundantly clear.

MR. SWANSON: Thank you.

JUDGE MILLER: We have granted the courtesy to Dr. Cochran and out of consideration for Ms. Finamore's representation previously, these are technical matters and that she would feel better able to present her client's position if Dr. Cochran could participate.

We previously followed our normal rules that we don't want persons to be both advocates, lawyers and witnesses.

We recognize that there are some grounds here. We are exercising our discretion. We have asked Ms. Finamore to be sure that there is no duplication of the arguments. They have divided them up fairly.

It is also further understood of record that the arguments presented now by Dr. Cochran are simply that, arguments. They are in no way evidentiary, and no more evidentiary than what you say, Mr. Swanson or Mr. Edgar or Ms. Finamore.

All right. Any other ground rules now that we need to consider before we go into our closing arguments?

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I. CONTENTIONS 1, 2 3
SITE SUITABILITY

JUDGE MILLER: Mr. Edgar, you may proceed.

MR. EDGAR: The first issue relates to Contentions 1, 2 and 3 on Site Suitability and on this issue I'd like to present some summary discussion first to attempt to draw some balance around the issues and what we regard as some critical points of affirmative evidence.

I will not repeat that procedure on most other issues but I think if I do it the first time on this one, it will save time on some subsequent discussion.

Then I propose to go on and attempt to define where we see the real contest, but if the Board will bear with me, it may take a while to get through it but I think on this particular contention, that serves a useful purpose.

JUDGE MILLER: Very well.

MR. EDGAR: We see three basic questions bound up within the Site Suitability issues on Contentions 1,2 and 3.

We also believe that the key elements of an affirmative testimony are captured within three documents.

The first and most fundamental of these documents, is the Site Suitability Report which is the Staff's Exhibit 1.

The second document that we regard as important is the Applicants Exhibit 1, which appears at Tr. 1989.

JUDGE MILLER: Tr. what?

MR. EDGAR: 1989.

JUDGE MILLER: Thank you.

MR. EDGAR: And the third package, which really consists of two pieces, is Staff's Exhibit 2 and 3 which appear at Tr.2446 and 2483, respectively.

Now, from examination of this basic material there are three basic questions that emerge.

The first is whether the hypothetical core disruptive accident should be a design basis accident. That issue is bound up in NRDC Contentions 1(a), 2(a) and (b) and 3(b) and (d).

The second fundamental question is whether the Site Suitability source term has been selected so as to envelope the consequences of credible accidents.

This basic inquiry is bound up in NRDC Contentions 2(a) and (b) and I put a parenthetical note on that, partial, 2(c) and 2(d).

The third basic inquiry is whether the containment will limit off-site dose consequences to levels within the 10 CFR, Part 100 Dose Guideline Values.

That inquiry is bound up within NRDC

24-3

Contention 2(d).

I should note two related issues which would be, in my judgment, deferred and we will address separately because we believe they are separable.

One is Contention 2(e), which is the issue of the validity of the dose guideline values recommended by the Staff in the Site Suitability Report for application to Clinch River.

And the second related issue is the issue of the environmental effects of accidents and particularly, the environmental risks associated with core disruptive accidents beyond the plant design base.

That issue is bound up,in our judgment, within NRDC Contentions 2(d),(f),(g) and (h), 3(c) and 3(d).

So, with that basic introduction, I'd like to turn to the three inquiries; that is, whether an HCDA should be a DBA; whether the source term envelopes the consequences of credible accidents and; whether the containment will reduce off-site doses to levels within the dose guidelines.

Turning to the first question, whether an HCDA should be a DBA.

There are two fundamentals that should preface this conversation.

24-4

The first is that one can and the project, in fact, did review lists of sequences from lightwater reactors, Staff standard review plans, FFTF, to determine what kind of sequences are important for examination of the conditions that could lead to a CDA, but getting lost in specific sequences may not be always the most meaningful approach.

So, we would commend in this regard to the Board's attention, the analysis that is summarized in Applicants Exhibit 1 in Sections 2 and 3, which start with a much more fundamental physical perspective.

The Staff's Exhibit 2 and the S Safety or Site Suitability Report, Staff's Exhibit 1, also approach the analysis on a similar level.

If one divorces oneself from accident sequences, one can reduce the problem to the following propostion:

All accident sequences of important to initiation of an HCDA must involve one of two basic conditions. Either reduced heat removal or excessive heat generation.

It doesn't matter how you get there but you're going to end up there at HCDA conditions and that's the important element of the analysis.

If you examine these conditions in light of

24-5

the basic design characteristics of the reactor, one can yield four systems within the design concept, which are necessary to maintain conditions of heat generation and heat removal, so one would not progress to HCDA conditions.

And there is not in the record, to our judgment, a real dispute as to what the four key systems are.

The four key systems are, first, the reactor shutdown system; secondly, the shutdown heat removal system; thirdly, features to prevent the primary heat transport system from incurring a leak at leak rates in excess of the design basis and; fourth, features and measures to prevent a local imbalance between heat generation and heat removal from extending into some wider scale involvement in the core.

Essentially, features that will limit fuel failures to very local conditions.

Once these four systems are lined up and the relevant descriptions of these said systems can be found in Applicants Exhibit 1, starting at Page 26 or Tr.2015; they can be found in the Staff's Exhibit 1, the Site Suitability Report and they can be found in the Staff's Exhibit 2.

There are several salient design characteristics of each of those systems which we feel are crucial in determining the issue of the HCDA and the DBA and each of those systems can be considered in turn in that light.

The first, the reactor shutdown system. We feel the most important element of that issue is that in Clinch River here is a recognition of the importance of the reactor shutdown system for terminating excessive heat generation.

The Clinch River design contemplates two redundant,diverse and independent shutdown systems in order to achieve high reliability against excessive heat generation.

Both systems use a different operating physical principle and employ well-established technology.

One is electromechanical and the other is electrohydraulic.

Both operate on fail-safe principles and loss of motive force or power will cause scram.

The relevant elements of the reactor shutdown system are found at Tr. 2016 through 2024.

The next system that we feel is important in its salient characteristics -- I will go on to describe -- is the shutdown heat removal system.

In the parlance of the lightwater reactor, it's the decay heat removal system.

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MR. EDGAR: Once one shuts a reactor down or terminates the fission reaction, it's important to deal with the stored and decay heat in the system. The salient general design characteristics of the shutdown heat removal system, in our judgment, are: Its redundancy, diversity and independence, that this reactor can remove heat through four independent paths, three through the primary loops and one through a separate system known as the direct heat removal system.

This plant can also remove decay heat without offsite power. It can do so in natural circulation, in turbine-driven or auxiliary feedwater pumps.

The shutdown heat removal system description can be summarized at TR 2024 through 2029.

The third element of importance, in terms of design features, are those measures necessary to prevent a large leak in a primary heat transport system pipe. These are described at TR 2029 through -32.

There are several levels of approach in the design to assuring that design basis leaks will not be exceeded. The first and most basic is to select materials which show properties of toughness, ductility and which can accommodate the high temperature service experienced in the reactor.

The second is to provide redundant, diverse leak detection, which can detect small leaks at levels of leak rate orders of magnitude below the design basis.

In addition, there are specific programs that have been undertaken on the primary system piping to assure that (a) flaws in piping which are inherent in the material are restricted in extent; that (b) the flaws will not grow under expected stress conditions to become a crack; (c) that even if a crack develops, the technology is present to detect that crack through leak detection and take appropriate action; and, last, that if a crack develops, the material properties are such, and the stress levels are such, that the extent of the crack will be confined so that leak rates will not exceed design basis conditions.

The fourth and final set of design characteristics of importance to this issue are those which are necessary to maintain individual subassembly, heat generation and heat removal bounds; in other words, measures which prevent a local fuel failure which can be accommodated within the design from becoming a widespread involvement which might then progress to core disruptive accident conditions.

These features are described and summarized at TR 2032 through -35.

There are basically two subsets of

considerations within this category of local fuel failure features.

The first are features which would preclude a rapid flow reduction to the fuel assemblies. At the bottom of the fuel assemblies -- and there are some good illustrations of this at TR 2033 through -34 -- there are special inlet ports, both primary and auxiliary, that are designed so that even if foreign material were introduced at the bottom of the reactor, flow would not be starved up in the fuel assemblies.

In short, these features are going to preclude a Fermi-type incident where a plate came off within the reactor, blocked the inlet to the fuel assembly; and there was partial melting in two fuel assemblies in that reactor.

Some discussion of that subject and the relationship of these design features to that type of event can be found at TR 1828 through 1831.

Now, that deals with the flow issue and how one assures that there will be sufficient flow to a fuel bundle to prevent local fuel failure propagation.

But there is another set of features that provide additional assurance that, assuming you have a feature locally or an imbalance in heat generation and in heat removal locally, that it will not propagate to

25-4

wider-scale involvement.

These features are found at TR 2034 through -35.

The three basic sets of information on this subject consists of packaging the fuel assemblies within individual steel hexagonal subassembly ducts. They enclose each fuel bundle so that there is an inherent limitation of propagation of effects from one bundle to the next and across the core.

The other element is experimental information and analysis of fuel failure and how fuel failure mechanisms are limited so that one fuel pin doesn't interact with its next nearest neighbor and cause a propagation.

The final element of this consideration involves detection devices that, assuming that one has a fuel failure, it is prudent to know that and to be in a position to take action prior to the time that any more widespread involvement could occur.

So there are two classes of instrumentation that are important here. There are fission gas detectors which will determine -- are sensitive enough to pick up a single rod failure.

They will detect fission gas release from the rod, and at least give an indication of single rod failures.

In addition, there are delayed neutron detectors which will detect the event of fuel contact with sodium, so that there is a detection mechanism available at levels below those resulting in local blockage; that is, when the cladding loses integrity in sufficient amount, that you have fuel in contact with sodium, the delayed neutron detectors would tell you that.

Now, that's a long way of saying that we believe that the fundamental physical characteristics of this reactor are well understood, that examination of the problem at the basic level of heat removal and heat generation tells one that there are four basic design features of fundamental importance here.

One can also conclude that the shutdown system -- the shutdown heat removal system, the primary heat transport system integrity and the measures necessary to prevent fuel failure propagation are in place, well understood and that it is clearly feasible to design the Clinch River reactor so that the progression of heat removal and heat generation to CDA conditions is highly unlikely and that CDA conditions should not be part of the design basis.

Now, once one reaches that point, in our judgment, it is a straightforward matter then to address the two remaining issues under site suitability.

The first one is selection of a site suitability source term. The site suitability source term recommended by the NRC Staff is the same as that for light water reactors, except that consideration has been given to the plutonium content in the core and the Staff specified an additional factor of one percent plutonium for the site suitability source term.

The Applicants' analysis, Applicants' Exhibit 1, at TR 2037, shows that the consequences of the site suitability source term envelopes the consequences of those events within the design basis.

Next, the Applicants' analysis at Applicants' Exhibit 1, TR 2040, the Staff's analysis in the site suitability report, Staff Exhibit 1, at Page III-11, and the Staff's Exhibit 3, which appears at 2498, all show that given the source term recommended by the Staff, given meteorology in accordance with Regulatory Guide 1.145, and a design basis leak rate of one percent -- volume percent per day that the containment will limit doses to within Part 100 dose guideline values.

Thus, we believe that affirmatively the record clearly shows that the hypothetical core disruptive accident should not be a design basis accident, that the consequences of the credible accidents are

25-7

enveloped by the site suitability source term, and that the plant will meet the dose guidelines.

Now, I appreciate your patience, and I will now -- I really think that information is important on an affirmative basis.

But let me now address what I perceive to be the issues that have been raised in opposition and our response to each.

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On the first question of whether a hypothetical core disruptive accident should be a design basis accident, NRDC raises four issues.

First, experience with domestic reactors indicates that the HCDA should be a DBA.

Second, experience with foreign reactors indicates that an HCDA should be a DBA.

Third, the HCDA was a DBA in the parallel design.

Fourth, the Applicants and Staff have not assigned a proper role to reliability in the decision as to whether the HCDA should be a DBA, and when I say reliability here, I mean quantitative reliability considerations.

As to the first of the issues raised by NRDC this involves experience with domestic reactors as the basis for the HCDA being a DBA. This is an argument drawn from history.

It can be found in Intervenors Exhibit 3, Tr.2822.

What you find there are references to a series of prior projects and the argument made that because a prior project treated CDA's in one way or the other, they should be treated that way here.

Well, we don't believe that the argument

26-2

follows but, nevertheless, we believe that you can go through each case and I'll try to confine myself to the major ones, and explain why they don't prove the proposition sought.

The first is EBR-I, Experimental Breeder Reactor-I. The relevant citations here are Tr.2628 through -30. The Final Environmental Statement Supplement, which is Staff Exhibit 8 in Appendix J.

In EBR-I, in the experimental breeder reactor, the reactor was being subjected to a test to impose upon it an intentional power excursion.

The shutdown system was disconnected at the time, in order to allow the reactor to reach an overpower condition. Given that the shutdown system was disconnected, and the operator did not respond, there was an incident in which fuel damage occurred, but it did not result in significant releases.

We think that provides no proof for the proposition that because there was an incident at EBR-I that it follows that a CDA must be a DBA at Clinch River.

We think that that situation is distinguishable and, indeed, there is no relevance to the issue at hand.

The next concern, the Fermi Reactor. The Fermi Reactor was an LMFBR in which the design basis specified for that reactor was the meltdown of one

26-3

assembly.

The reactor experienced a situation in which a plate came loose within the primary system, lodge up against the bottom of the core, flow was restricted and melting occurred in two fuel assemblies.

However, no significant releases occurred. Furthermore, the total amount of melting was one assembly.

We think, particularly given the design features provided within Clinch River, which incorporate the lessons learned at Fermi, that that again is not relevant information or substantial information, the evidence suggesting the need for a CDA being a DBA.

The citations of importance here would be Tr. 2636 through -37. Tr. 1828 through -30 and Staff Exhibit 8, in Appendix J, discussing the Fermi incident.

There are two additional domestic reactors which are advanced by NRDC as proving the point that an HCDA should be a DBA.

They are SEFOR and FFTF.

As to SEFOR, if the Board will consider Tr. 2396 through -97, Tr. 2638 through -39, it will be seen that the SEFOR reactor was designed for an event involving core disruption.

However, recognize that SEFOR had a particular mission. It was a reactor specifically designed to

26-4

undergo rapid power excursions to develop information on the Doppler coefficient.

It is simply not a reasonable analogy here. It was not a power reactor and it was specifically developed to undergo transient testing.

The final argument presented on domestic reactors involves FFTF. The argument is made by reference to a series of documents that FFTF had a core disruptive accident as a design basis accident.

The fact is, that is not true.

You will find at Tr.2395 through -96 and Tr. 1825 through -26 the fact that the CDA was simply not a DBA.

The safety approach for SEFOR, while articulated in somewhat different language, was in all substantive respects the same as CRBR.

That is the basic set of issues in terms of whether domestic reactor experience,in and of itself, can support the proposition that the CDA should be a DBA.

We don't think that that is substantial evidence at all on which the Board should rely for a decision here; plus, we think that careful examination of that information will show that it does not prove the point asserted.

The next level -- the next issue defined involves experience with foreign reactors.

If the Board will consider the following sources of information.

First of all, Applicants Exhibit 29, you will see that contrary to NRDC's assertions, the approach adopted by the French in their reactor programs is essentially identical to that employed by CRBRP.

There is an interesting sidelight to this conversation and the Board should consider here Tr.2727 through -37, Tr. 2644 through 2649 and then two exhibits. Intervenors Exhibit 5 and Intervenors Exhibit 7.

NRDC made the argument that a so-called dome or cupola within the French breeder reactor was the equivalent of the so-called CRBRP parallel design, in that it had a sealed head access area and a core catcher.

If you will examine Intervenors Exhibit 5, you'll see that what that dome really is, is a container and, indeed, that's the primary containment for the reactor.

Then the facility has a concrete confinement building and running from this dome, out through the concrete confinement building is a vent system with filtration.

Indeed, it is a direct analog of Clinch River.

The difference between this and Clinch River is that the steel confinement building is larger and the annulus space is smaller.

We, thus, do not believe that there is any analogy to be drawn and no argument which will hold water, that one can examine the French program and determine that a CDA should be a DBA for Clinch River.

Similar conclusions follow with respect to other foreign reactors. The Board should consider there Tr. 2707, Applicants Exhibit 31.

Our bottom line at that point is, that the foreign reactor experience is, indeed, consistent with domestic experience, but that, again, is not the basis upon which we would urge the Board to make its decision.

We would urge the Board to make its decision upon consideration of the technical issues and consideration of the four basic systems which are important to the CDA versus DBA issue in Clinch River.

The next argument advanced by NRDC is, that the hypothetical core disruptive accident was a DBA in the so-called parallel design and that the NRC Staff had determined that a core disruptive accident had to be a design basis accident for Clinch River and then they went through a dramatic reversal of their position.

If one steps back from those facts and looks at the sequence of events on that issue, that is simply incorrect.

If you examine Dr. Cochran's testimony in this respect, he cites a series of Staff documents and statements which indicates that the Staff had taken the position that it was incumbent on the Applicants to prove that a CDA should not be a DBA, and express skepticism about the Applicants ability to make that proof.

Well,it's more important that you consider the dates of all those Staff statements.

If you look at Tr. 2650, you will see an admission by Dr. Cochran that each and every reference to Staff positions which suggest that an HCDA should be a DBA was dated before the Clinch River application was filed.

The Staff had not even reviewed the design at that point.

See Tr. 1837, for the date on which the application was filed and you will see that that element of the story cannot hold together.

The point is simple here.

The Staff never decided that an HCDA shouldn't be a DBA. They hadn't reviewed the matter and they required the Applicants to convince them.

The more substantial evidence here is Staff

26-8

Exhibit 5. That is the May 6, 1976 letter from Richard Denise of NRC to Lockland Caffey of the Clinch River Breeder Reactor Plant Project Office, and that establishes and importantly notes that after the Staff had a chance to review the design, they were able to establish some basic ground rules for their view of the safety issues.

There is another important predicate fact in this line of argument; that in going from the two track system of the parallel and reference design and in the Staff's making the decision that they would consider the reference design where the hypothetical core disruptive accident could and should be precluded as a design basis accident, there was a basic change in the containment design.

In the original reference design, Clinch River had a single steel shell containment, when before the May 6 letter, and if you look at Tr. 1837, you will see that prior to the May 6 letter, the Applicants filed an amendment which modified the containment design to use the dual containment confinement approach, rather than a single steel shell.

It had a steel shell and an outer concrete confinement building.

What we see in the pattern of events here is not a dramatic reversal. What we see is simply the Staff's

26-9

decision after having had a chance to review the application.

In addition, we submit, that this has no relevance for the Board's consideration today. This is not the kind of evidence which the Board should rely upon in rendering its decision.

The Board should focus on the substantive technical information and past history, however revisionist one's approach is to interpreting it, has no bearing on a decision of this importance for the Board.

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27-1 bm

MR. EDGAR: There is a fourth and final category of arguments in this respect. I will first define, as I perceive the issue, that there is a disagreement about the role of quantitative reliability and a decision as to whether the HCDA should be the DBA.

Now there's a threshold point here, and that is, that Intervenors have persisted in their view that the Board's April 22nd ruling is incorrect, that one must demonstrate with fault tree/event tree analyses and quantitative reliability assessments by detailed reviews that the CDA is not a DBA; and if you don't do that, it just can't be done.

We do not wish to rehash that. It has been burdened at length, but that is the major thrust of the argument.

Beyond that, there are some specifics. The first deals with reactor shutdown system reliability.

In that respect, the NRDC offers, first, the proposed ATWS rules for light water reactors. That came up the other day, and the Board's attention was brought to the Commission's rulemaking decision on ATWS.

The ATWS rule indicates that certain LWRs may have failure rates above or approaching 10^{-3} and that action should be taken to remedy the situation.

We don't think that that is convincing here in that CRBR has two shutdown systems and redundancy, diversity and independence -- and if the Clinch River design is properly executed, there is no reason why one should simply blindly apply the ATWS rule for the proposition that shutdown reliability cannot be achieved.

There is nothing -- and I repeat, nothing -- in the record to indicate a fundamental physics difference in control characteristics or kinetics between LWRs and LMFBRs such that it is beyond the state of technology to design a reliable, effective, redundant, diverse, nad independent, fast-acting shutdown system for Clinch River.

Applicants' Exhibit 46, which is the testimony introduced the other day, contains discussion starting at the beginning of reactor kinetics and control characteristics and demonstrates this point.

There is an additional consideration in connection with NRDC's argument about quantitative reliability programs.

We submit that NRDC has misconceived the nature of the Clinch River reliability program and the role perceived by the Staff and the Applicants for a reliability program.

The reliability program is not set up to disprove that an HCDA should be a CDA or a DBA. As Mr. Morris indicated yesterday at TR 5647, the reliability

28-2

program role is to confirm the ability to realize the potential for the reliability inherent in the redundant, diverse and independent systems.

This is something that's done after the fact in real time with hardware. It is not simply an analytical program designed to prove a negative.

We submit, Your Honor, that NRDC's four fundamental issues: domestic reactors, foreign reactors, their consideration of the parallel design history and last, but not least, their consideration of quantitative reliability cannot hold water.

We submit on the contrary that this Board should turn itself to the affirmative evidence and consider the merits of the four features that I discussed at the outset.

There is a final level of issues bound up within the site suitability contentions. These involve the question of how one does the source term contention and the basis for the selection of the source term itself.

The first item on that agenda is that if one can establish -- and we submit that that is a conclusion that's well supported in the record -- that the source term should not be predicated upon a hypothetical core disruptive accident being a design basis accident, then there is no great reason to consider those issues which might involve the mechanistic derivation of the source term; that is, comparing the source term to specific event sequences or mechanistic HCDA consequences.

The issues here have really focused down on how the source term consequences were calculated and whether the dose guidelines have been met.

These arguments can be outlined in short form by reference to Dr. Morgan's testimony, which appears at TR 3119.

In our judgment, there are two primary arguments advanced by Dr. Morgan, and then several minor points. The two primary arguments are whether the dose calculations properly considered the entire passage of the cloud from release at the source term.

The Staff's initial calculations truncated the calculation at the end of 30 days at the low population zone, so the argument goes, "You should have considered the longer passage of the cloud."

The next issue is whether the dose calculations properly considered plutonium isotopic concentrations with higher plutonium 238 and 241. You will obviously have some familiarity with that issue since it came up in the fuel cycle contention as well.

There is a slightly different cast on it here in site suitability, but the answers are very nearly the same.

28-4

The two arguments -- the entire passage of the cloud and the isotopics -- appear at TR 3126 through -28 and TR 3128 through -32, respectively.

Turning, first, to the entire passage of the cloud argument, this argument came up because the Staff was asked a question in discovery about -- what about effects after 30 days?

The Staff, for the purpose of analysis, and to test the extreme bounds of sensitivity did a calculation which is known as a puff release. TR 2939 describes that.

But what it is, if you'll examine that portion of the record, is that you run out the site suitability source term calculation for 30 days, assuming release over that period, and then at the end of the 30 days, you assume instantaneous release in zero time of all of the remaining contents of the containment.

This results in an increase in doses so that you're at, roughly, half of the site suitability dose guideline values. I should not say half. I'll retract that.

I should say a substantial portion of the dose guideline values. I don't have the numbers in my head.

The real question, though, is whether this analysis is meaningful. Applicants' Witness Strawbridge at

TR 1830 through 1832, pointed out that he had examined his analysis of the source term and determined that if you examine the dose as a function of time, 90 percent of the dose is incurred on the first day and 98 percent of the dose is incurred in the first week.

Now, also, in relation to the same portion of the transcript, Mr. Strawbridge pointed out that during this 30-day period, if one release is off-site and one still has material within that containment, that logically that material in the form of aerosols is going to deplete and fall out.

So there's a countervailing consideration of depletion of the source term as a function of time.

When the Staff reanalyzed the problem to make a more reasonable assumption on the puff release, and they considered depletion and fallout, the values that they calculated indicated the contribution after 30 days was essentially insignificant.

That can be found at TR 2400 through 2404.

In our judgment, the evidence is clear that the puff release is not a valid argument, that the effects beyond 30 days are simply not significant.

Now that brings up the issue of isotopics. We will return to address that in connection with the fuel cycle.

28-6

This argument appears in Dr. Morgan's testimony, Intervenors' Exhibit 9, at 10 through 15, at TR 3128 through 3133.

And what this argument boils down to is this: The Staff made certain assumptions about plutonium isotopics. In particular, they assumed conservative values; that is, higher values for plutonium 238 and 241 content relative to that fuel which the Applicant intends to use and upon which the Applicant has based the application.

Morgan then computes a hazard index which is not really a subject of dispute. Plutonium 238 and 241 are more toxic than other isotopes of plutonium.

But the problem then breaks down because there's a giant leak at this point. If you look at Transcript 3133 (sic) through -32, the argument is made --

JUDGE MILLER: What was that again?

MR. EDGAR: I'm sorry. Let me repeat it.

It is Transcript 3131 through -32. I thought I might have stuttered.

JUDGE MILLER: I think you transposed -- interpolated.

MR. EDGAR: I think I did.

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gsd

MR. EDGAR: Here's where the link comes in. The argument is made that because plutonium recycle through repeated recycle in lightwater reactors would result in the buildup relatively of the isotopes 238 and 241.

Then we make a big assumption that recycle in Clinch River would result in the buildup of 238 and 241.

JUDGE MILLER: We will go about ten minutes more and then have a recess.

MR. EDGAR: And because you've got a higher burden of 238 and 241, higher doses would result.

This is a significant error. We will cover it in more detail in fuel cycle, but at this point it's significant to note three basic points briefly.

First of all, Clinch River will be licensed with a certain fuel, and if the Clinch River reactor uses other fuel which is outside the bounds of the isotopics, then an amendment would have to be sought.

That's standard NRC licensing practice. See Tr. 1833 for that proposition.

Secondly, and I think much more importantly, if you recycle fuel in Clinch River in a breeder reactor with a fast neutron spectrum rather than a thermal neutron spectrum as a lightwater reactor, 238 and 241 are going to get burned on subsequent recycle.

Their relative concentrations go down, rather than up, as they do in a lightwater reactor.

The proof of that is found in Applicants -- or Intervenors -- wait a minute.

The proof of that would be in Applicants' Exhibit 36, which is Volume 3 of the ER, Amendment 14.4(a) to Chapter 5.7 of the ER.

Furthermore, Dr. Cochran does not dispute this analysis. See Tr. 4539.

We think there are two points that are -- or one remaining point of lesser interest and not of as major importance as the puff release and the isotopics.

I'll just mention it in passing. The argument is made that dose models incorporating ICRP-30 methodology rather than ICRP-2 methodology would make a big difference.

I believe the burden of the testimony is that at most this is a factor in the order of two, and that in addition the dose calculated for the source term are well below the guideline values.

The sources for this information would be Applicants' Exhibit 25 at Page 11; Dr. Thompson, Tr. 1903; Mr. Strawbridge, 5282 through 3; Mr. Hibbitts, 5218 through 19; Mr. Bell, 2344 of the Staff; and Mr. Bell, 2351.

We submit, Your Honor, that if you go down the list of the major questions: Should the CDA be a DBA? Does the source term envelope consequences of credible

29-3

accidents? And will the containment reduce offsite doses to levels within the dose guideline values?

If you examine the substantive technical information, the answer is clear. We submit that the Board should find affirmatively on all three questions.

JUDGE MILLER: All right. We'll take a recess.

(Recess taken.)

JUDGE MILLER: All right. Who wishes to go next? Take your choice. The Board does not care.

MR. SWANSON: Counsel talked briefly and I think we will alternate. Staff will go first this time.

JUDGE MILLER: Okay.

MR. SWANSON: In following the Applicants, I will try not to repeat, and in doing so, I think that will greatly shorten the presentation I would have otherwise made, because as the Board has recognized, the positions are quite similar.

In responding to the issues, Mr. Edgar cited many Staff sources of testimony, as well as Applicant.

I would like to briefly characterize, however, the Staff position in our own words.

As Mr. Edgar correctly pointed out, the Staff's main sources of testimony on site suitability matters can be found in the Staff's first three exhibits.

Mr. Edgar identified them. They are the

29-4

Site Suitability Report, Staff Exhibit 1; the testimony filed by Dr. Morris and others, which is Staff Exhibit 2; and the testimony filed by Mr. Bell and others, Staff Exhibit 3.

In approaching the issues that were discussed and litigated in the August site suitability hearing, the Staff indicated that the purpose of assigning or defining design basis accidents in response to Contention 1(a) was to establish an analytical test of the safety systems and features of the Clinch River reactor. Following the practice which is customary in licensing lightwater reactors, accidents involving very improbable multiple failures of safety systems or failure of conservatively designed safety features need not be included in the design basis accident spectrum.

Staff Exhibit 2, the testimony of Dr. Morris, et al., which starts at Transcript Page 2445, contains the Staff discussion and analysis of the site suitability source term accident and its basis for not including accidents beyond the design basis, such as core disruptive accidents in the design basis by establishing the site suitability accident as described in that document.

In the Site Suitability Report the Staff selected an accident which bounded all design basis accidents and in fact bounded many core disruptive accidents,

29-5

as well.

A discussion of that can be found in Answer 7, which appears on Pages 8 and 9 of Staff's Exhibit 2.

The Staff identified feasible design in operational measures, including those normally applied to lightwater reactors and those special measures needed for an LMFBR, such as the CRBR, which would be implemented at CRBR to insure that the conditions which could lead to CDA's are very improbable.

At this point I think it's very instructive for the Board to refer to a fairly lengthy answer in Staff Exhibit 2; that is, Answer 13, which begins on Page 14 of that document and runs until Page 23.

In that answer the Staff identifies five safety functions which must be achieved in order to prevent CDA's from being included within the design basis.

I will just mention them. They are, first, that you must shut down the reactor nuclear chain reaction upon initiation of transients.

Second, you must be able to maintain sufficient coolant inventory.

Third, you must be able to maintain sufficient coolant flow.

Fourth, you must be able to remove sufficient heat from the fuel.

29-6

Fifth, you must be able to avoid propagation of local fuel faults beyond an assembly.

This is a somewhat different way of stating the same factors I think Mr. Edgar indicated at the beginning of his argument.

In the pages that follow, that I just referenced in Staff Exhibit 2, the Staff sets down in greater detail a discussion and actually a roadmap for the reader to follow to ascertain the Staff's bases for its basic conclusions.

The roadmap I mentioned because it refers to the sections in the Staff's Site Suitability Report, Staff Exhibit 1, where in greater length the Staff describes its bases for analysis and conclusions.

I would like to reference just a few points, however, that the Staff considers quite important and which serve as the primary basis for the Staff's conclusions.

Regarding the ability to shut down nuclear chain reactions, the Staff points out that in its review of Clinch River, and as described in the Site Suitability Report, the Staff will require dual redundant shutdown systems which are independent of one another and which employ diversity of design by designing each of these dual systems to itself meet the single failure criterion; and this is a requirement which is normally applied only

29-7

to the total shutdown system of an LWR.

It becomes necessary for four simultaneous component failures to occur before defeating a reactivity shutdown function.

This is again discussed at Page 15 of the Staff's Exhibit 2.

Independence and diversity are designed into the dual systems to minimize the possibility that such a simultaneous occurrence could in fact exist at Clinch River.

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The Staff describes the standards that are applied in its review to assure that the single failure criterion is met, and also to indicate the basis for the Staff's assurance that independence is achieved.

The diversity is achieved by employing different types of components as sensors, logic, reactivity insertion mechanisms, and sometimes by requiring that design and maintenance functions be performed by different groups.

You can refer to Section II.C.1. of the Site Suitability Report for a more lengthy discussion of this matter.

The Staff similarly in that same answer in its Exhibit 2 describes in greater detail the basis for its conclusion that you can maintain sufficient coolant inventory.

The Staff indicates, for example, that principal measures to achieve this goal are to perform pre-service and in-service inspection of primary coolant boundary to verify continuing piping integrity, and to install a detection system which can detect small leaks should they occur before they grow to an unacceptable size.

The LMFBR primary coolant systems operate at low pressure, below the boiling temperature of sodium, so

29-9

that an emergency core cooling system as you might counter with an LWR is not necessary.

Instead, the Staff has concluded that it is sufficient to provide guard vessels to catch hypothetical coolant leakage from portions of the system, and piping, to have piping elevated above the top of the core for other portions of the coolant system to preclude siphoning.

Again, the features and components of the Staff analysis in this area are described at Pages 16 and 17 of Staff Exhibit 2.

The ability to maintain sufficient coolant flow is described at Pages 17 and 18 of Staff Exhibit 2, and in greater detail in Section II.C.3. of the Site Suitability Report, which is Staff Exhibit 1.

The Staff indicated that by assuring a clear path for coolant flow to fuel assemblies, a sudden flow blockage and damage to subassemblies can be precluded, such as occurred at Fermi.

Measures were taken in the general design of Clinch River to preclude this blockage from occurring, include items such as multiple coolant inlet ports at different planes and strainers in the flow path.

Even though high quality of fabrication can be expected, non-mechanistic deposits of debris or other

29-10

loose parts can be postulated, and flow blockage from these parts can be avoided by employment of core outlet thermal couples or loose parts monitoring systems to aid operators in diagnosing and correcting such systems.

Again, these are the type of measures that the Staff relies on in reaching assurance that sufficient coolant flow will be maintained at the Clinch River Breeder Reactor.

The other components that I mentioned, the ability to remove sufficient heat from the fuel, is described on Pages 18 and 19 of Staff's Exhibit 2, and in greater detail at II.C.4. of the Site Suitability Report.

The ability to avoid propagation of local fuel faults is described in Pages 19 and 20 of Staff Exhibit 2, and in greater detail in II.C.3. of the Site Suitability Report.

The Staff position is that -- in conclusion, is that given the current state of the art of reliability analysis methodology for reactor systems, it is more appropriate to continue to rely on established deterministic criteria and engineering judgment than on reliability analysis and goals in establishing which accidents are included in the design basis accident spectrum.

The Staff has factored into the criteria to be applied to CRBR lessons that have been learned from

29-11

previous reactor accidents.

The discussion of prior accidents, information that has been learned, I think was alluded to and references were given by Mr. Edgar previously. I won't repeat them, but I mentioned one already, the Fermi-1 accident.

Another example, the TMI-2 accident falls into a separate category; that is, the human error aspect and its ability to cause CDA's.

In that respect, I would refer the Board to Answer 15 of Staff Exhibit 2, which begins on Page 23 and runs through Page 25.

In that answer the Staff describes the bases for its conclusion that human error is very unlikely to cause a core disruptive accident at CRBR.

As the Staff indicates in that answer, special emphasis on this matter has been the case since the TMI-2 accident.

The Staff has relied on its experience and knowledge of characteristics of LMFBR's in previously operating LMFBR reactors, and it is believed that there are no special LMFBR characteristics which require extra-ordinary capability on the part of the operator, as compared with LWR's.

Rapid operator action in responding to accidents

29-12

will not be necessary, because the NRC criteria normally applied to LWR's and also to be applied to CRBR require that fast-acting safety systems be installed to mitigate rapidly developing accidents.

Because of the low primary coolant pressures of the LMFBR's, the operators at CRBR would not be faced with the challenges of performing any actions relating to depressurization during small pipe breaks or loss of offsite power, as might be the case for pressurized water reactors.

Because of the large heat capacity margin of an LMFBR reactor coolant system, there is ample time for operator action in transferring to the backup decay heat removal system in responding to loss of all primary heat transport capability.

I would also refer to two instances in the transcript where Staff witnesses indicated, first, that human error at Clinch River would not differ significantly from the type of error to be expected at LWR's. I would refer you to Transcript Page 2244.

Secondly, Staff testified that the likelihood of unrecognized interdependence which might cause an accident is very low. The transcript reference there is 2256.

The Staff position was that at this stage of

29-13

review for the site suitability LWA analysis, it is unnecessary to complete the kind of detailed review of the acceptability of the human factors considerations that will eventually be carried out during the CP and, ultimately, the OL reviews.

However, for the reasons discussed in the Staff testimony that I described, primarily Answer 15 in Exhibit 2, the Staff is confident that by following the NRC review policy, core disruptive accidents at Clinch River resulting from human error will be very unlikely.

The Staff believes that by taking into account the above factors that I've referenced thus far, it is reasonable to exclude core disruptive accidents from the Clinch River design basis accident envelope.

Grouping together other parts of Intervenor contentions which deal with core disruptive accidents, Contentions 2(b) and 3(b) and (c), the Staff concluded on Pages 29 through 31 of Staff Exhibit 2 that the Intervenors had not made the case on those contentions, and the Staff summarized its reasons therefor.

Briefly, the Staff position was that although -- that for the reasons set forth above (and I will not repeat them), core disruptive accidents need not be included in the design basis, nor need they be considered to be credible accidents from the viewpoint of 10 CFR Part 100.

29-14

Although core disruptive accidents may not be regarded as credible accidents, in keeping with the guidance of Part 100 in the practice for lightwater reactor licensing, a non-mechanistic release of fission products supplemented with plutonium had been assumed as the site suitability source term.

Consistent with this lightwater reactor practice, and again, consistent with Part 100, the magnitude of the release assumed for the site suitability source term bounds a range of core melt accidents and many core -- as I indicated, bounds all credible accidents and even many core disruptive, Class 9 type accidents.

Again, I gave a prior reference to Answer 7 of Staff Exhibit 2 for that position.

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MR. SWANSON: Mr. Edgar indicated several arguments that Intervenors had raised; and he responded to them citing frequently Staff documents. I will not repeat those.

There are a couple of arguments that we see recurring, however, in the Intervenors' testimony which we'll briefly address.

One recurring argument is that the Staff cannot demonstrate feasibility nor reasonably demonstrate the initiators of core destructive accidents because of a lack of experience with LMFBRs.

I would just like to mention three points that the Staff has established in these proceedings. The first is that Staff -- speaking for the Staff at this point -- the Staff does have considerable experience dealing with LMFBRs.

Just limiting ourselves to the panel that testified last summer, there was over 50 years of experience dealing with reviews and operation of LMFBRs. The transcript reference for that is 2394.

That includes Staff and consultants on the panel.

The Staff has utilized operating experience with Fermi, EBR-I and II, SEFOR and FFTF and incorporated that operating review into its analysis, which formed the

30-2

basis for much of the judgment that the Staff exercised.

The reference for that statement is Staff Exhibit 2 at Page 13.

And, in addition, one can just look at the statements of qualifications of the Staff panel members who testified during the summer. As was indicated, there is considerable experience either operating or reviewing several of the facilities I just mentioned.

Finally, the Staff established the point on Page 13 of Staff Exhibit 2 that many of the systems for LMFBRs are applicable from LWRs. Therefore, the experience gained from reviews and operation of those systems are transferable to the Clinch River Breeder Reactor.

In addition, the Staff concluded -- I would indicate that the Staff concluded that the safety functions I previously alluded to were found by the Staff to be feasible to achieve.

The reference for that is Answer 13 of Staff Exhibit 2, the lengthy answer that I referenced earlier.

I won't repeat some of the other arguments that Mr. Edgar mentioned. But I would indicate one final argument that Intervenors have repeated in connection with their contentions dealings with the extent of the magnitude of the accident that should be considered to be included in the design basis; and that is the argument by Intervenors that the Staff has inadequately considered sodium/concrete reactions. That is, the heat and pressure buildup inside containment when you do have a sodium/ concrete reaction.

I would just refer the Board briefly to the site suitability report, Pages II-18 and 19, where the Staff discusses the accommodation of sodium/concrete reactions in its analysis in consideration of a site suitability source term.

And then later on when we get to the environmental phase, Appendix J, particularly Page J-6, the Staff again considers the accommodation of sodium/concrete reactions.

The Staff did so in its environmental analysis by assuming core melt in at least four different categories of accidents in the sodium/concrete reaction, which is encompassed in the TMBDB design features -- that is, the annulus cooling and vent purge systems, which are used to mitigate the effects of a sodium/concrete reaction, as well as other facets of hypothetical core disruptive accidents.

In some of the classes considered by the Staff in Appendix J, the Staff even assumes that these features fail. And even with the failure, the Staff finds the risk to be acceptably low from the resultant doses that are received from such an accident.

The Staff concluded that even in such a situation, the risks are comparable to those that the Staff has found for light water reactors for similar type accidents.

Turning in greater detail to the derivation of the site suitability source term, I will again try not to repeat anything that Mr. Edgar has said. But I would like to briefly outline the Staff's position on derivation of the source term.

The Staff analysis which arrived at the site suitability source term, in compliance with 10 CFR Section 100.11A and Footnote 1 to that section, derived the source term by computing the source term in a non-mechanistic method provided for in that regulation, and also as provided for in the document TID 14844, which was attached to Staff Exhibit 3 and which is referenced by the regulation I cited.

These documents assumed releases beyond those which would be produced by any accident considered credible, and that, in fact, served as the basis for the Staff's derivation of the site suitability source term.

Discussion of that can be found at Pages 5 and 6 of Staff Exhibit 3. Again, Staff Exhibit 3 is the

30-5

testimony of Messrs. Bell and others of the Staff, which was introduced last summer.

By choosing the conservative source term, as represented in that testimony and the site suitability report, Staff Exhibit 1, the Staff concluded that not only are all credible accidents bounded, but that only some core disruptive accidents might not be bounded.

The appropriateness of Staff assumptions, I think can be demonstrated by the following discussion. The Staff argument begins with a requirement that you assume a design basis accident plus substantial core meltdown, as provided for by the regulation cited. That's also discussed on Page 8 of Staff Exhibit 3.

The light water reactor methodology that the Staff used is applicable to Clinch River because of similarities in function and because releases are based on percentage of core content, which account for changes in the isotopic contents of the core.

This is discussed at Page 9 of Staff Exhibit 3.

As a method of conservatism, the Staff took no credit for sodium absorption of iodine in considering the halogens, although sodium may completely absorb iodine in such an accident, preventing such a release.

Reference to that is Page 10 of Staff Exhibit

30-6

3.

Further, although the accidents which are likely to release iodine would only likely release ten percent of the fission product inventory, the Staff conservatively assumed a release of 50 percent of the iodine in the inventory.

That is mentioned on Page 11 of the same document.

Further reductions in iodine release would occur as a result of the attenuation along the path to the containment. That's referenced on Page 12 of that Staff testimony, Exhibit 3.

The Staff, however, because of attenuation, condensation and oxygen limitations assumed 180,000 -- excuse me. Let me start over.

The Staff assumed 180,000 pounds of sodium would be available for dumping into the containment. This is conservative because of attenuation, condensation and oxygen limitations.

I would refer you to Page 17 of the testimony, Staff Exhibit 3, for that point.

Again, it was determined that the sodium contribution to dose was negligible and need not be considered. In fact, that was a conservatism.

The basis for assuming that it was conservative not to assume -- or include sodium, as I mentioned, was that sodium could well cause substantial plateout of iodine if it were considered.

In terms of assessing toxicity by ignoring sodium, the Staff testified that based on computations and testimony, the toxicity of plutonium in the site suitability source term is over 1200 times more toxic than the entire dispersable sodium inventory.

That's discussed at Page 18 of the testimony.

Therefore, plutonium was the dominant material to be considered in that source term compared with sodium.

The doses that the Staff calculated, based on that source term, were derived from using the guidelines on 10 CFR Part 100. That is discussed starting at Page 26 of Staff Exhibit 3.

That Part 100 provides guidelines for doses to whole body and thyroid from such doses.

As a method of conservatism, these values were modified downward; that is, the dose guidelines, for the CP stage. And doses to several additional organs, in addition to those required by Part 100, were assessed for purposes of analysis.

This is discussed at Pages 7 and 8 and also Page 26 of Staff Exhibit 3.

The Staff analysis concluded that the dose

30-8

guidelines computed for Clinch River were appropriate. That conclusion appears at Page 34 of Staff Exhibit 3.

The Staff used the dose to the thyroid as the reference point, rather than to the whole body. This resulted in guidelines three times more limiting than if the whole body dose limits were used as a reference point.

That's discussed at Page 28 of the Staff testimony.

The weighting factors used were consistent with values from the major radiation protection agencies world-wide. That's discussed at Pages 28 to 29 of Staff Exhibit 3.

I want to remind the Board that the guidelines used are not intended to determine acceptable doses for the public, but for the purpose of site suitability were used for comparing sites and determining the acceptability of the Clinch River site.

We discuss that at Page 29.

Very briefly, I would just reference that at Pages 31 and 32 of the Staff testimony, the Staff presented its basis for concluding that the hot particle theory argued by Dr. Morgan had been discredited.

Further, at Pages 33 and 34, the Staff pointed out its basis for concluding that the collective judgment of the scientific community supports the Staff use of the

30-9

dose response model -- the linear dose response model in developing the guidelines that were used in this analysis.

The dose guidelines for the additional body organs used and analyzed were based on mortality risk weighting factors recommended -- again -- by one of the major radiation protection organizations; that is, ICRP-26.

That's discussed at Page 34 of the Staff testimony.

Mr. Edgar previously discussed the arguments of Dr. Morgan and the reasons why those arguments should be discounted by this Board. I won't repeat them. I think the arguments that were cited effectively did refute the arguments cited by Dr. Morgan.

Again, I would mention the same thing as Mr. Edgar, that when we get to the fuel cycle analysis, we see in that testimony even greater detail why the isotopic concentration argument raised by Dr. Morgan simply doesn't -- does not support the conclusions that he asserted.

I would add one further reference, though, in response to that.

One of the points mentioned by Mr. Edgar is that if Applicants in the future chose to use some different isotopic concentration of fuel, that they would have to go through the normal licensing reviews.

20-10

Mr. Edgar gave one reference. I would also refer you to a statement by Mr. Hulman of the Staff at Transcript Page 2348 in support of the same conclusion.

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31-1

Contention 2(d), regarding the containment capability, was addressed by Staff witness Eltawila in his discussion which begins on Page 23 of Staff Exhibit 3.

His conclusions were that the containment volume for the proposed Clinch River Reactor is similar to that of larger LWR plants which have been built to hold significantly higher pressures than is the CRBR design pressure.

Also on the same page, Dr. Eltawila testified that the size and strength of the proposed Clinch River Containment is clearly within the feasibility of current practice and that he relies in this conclusion upon experience building containments for other sodium-cooled reactors, and, hence, those designed to withstand sodium fire accidents.

He made that statement on Page 23 and 24 of that testimony.

Current lightwater reactor containments as well as the Fast Flux Text Facility are designed, constructed and tested to leak rates similar to that of a design leak rate assumed for Clinch River. That is, not more than 1/10th of 1 percent, by volume, per day. The reference for that is Page 24 of his testimony.

Dr. Eltawila testified that it's feasible to design Clinch River to achieve a by-pass leakage of

0.001 percent by volume per day, as we have with current lightwater reactor containments.

Page 24. Again, he relies on experience designing and the operation of lightwater reactor containments and experience with sodium cooled plants.

An additional conservatism cited by Dr. Eltawila is the annulus filtration system, outside of the steel shell containment, where 95 percent of leakage from containment would enter, and where it would be filtered and recirculated.

This is referencedon Page 25 and at that same point, Dr. Eltawila indicated that the components of the system are common in the nuclear industry and, thus, are feasible to implement.

Regarding Contentions 2(f),(g) and (h), dealing with the adequacy of computer modeling and their inputs, the Staff presented testimony regarding the computer models and codes it used in its analogy. The TACT code, the PAVAN Code and HAA-3 codes were used for the site suitability review and they are discussed, starting on Page 35 of Staff Exhibit 3.

The CRAC Code was described and was used, as was used in the Staff's environmental review and that is discussed in Staff Exhibit 17. That is the accident testimony filed this week.

That testimony begins on Page 5748 of the

31-3

transcript and the discussion of the codes and their inputs begins on Page 31, that's Answer 38, of that testimony.

I think that the issue is basically uncontested on this point, is the Intervenors did not file testimony on Contention 2(f),(g) and (h), which discredits the Staff testimony that codes were properly verified or that they are valid and that there was nothing offered to suggest that the documentation of the codes is necessary in any form, other than that in which the codes exist.

I believe there was no cross-examination by Intervenors on this issue of Staff witnesses and, as I indicated, the computer codes were described and the basis for their use was explained by the Staff in the testimony I mentioned.

The codes were documented and input for the codes was also validated and explained.

I think the conclusion the Board has to draw on this issue is, the Staff properly defended its use of computer codes and models, as used in the Site Suitability and NEPA analyses.

I won't repeat all the conclusions that one has to make to respond to each of the Contentions 1, 2,3. They are contained at the end of each of the Staff pieces

31-4

of testimony filed and, at this point, I think we are confining ourselves to Site Suitability analysis.

That would be Staff Exhibits 2 and 3, but I think in light of the information presented on this record, the Board has a basis for finding that the Intervenors have failed to assert or prove the Contentions asserted in those Contentions, insofar as they relate to Site Suitability matters.

That concludes my argument on this part.

It's my understanding that we are deferring the environmental phase of argument for a second round following Site Suitability argument.

JUDGE MILLER: I don't know.

Is that corect?

MR. EDGAR: Well, when we started we had the nine subject categories and that is consistent with the nine that we set out in the beginning.

JUDGE MILLER: Ms. Finamore or Dr. Cochran?

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DR. COCHRAN: Judge Miller, I would like to begin with a summary statement setting forth how I propose to go through my argumert here and much in the same manner that Mr. Edgar did.

I think Mr. Edgar has reasonably represented

31-5

the three major issues that are addressed by Contentions 1, 2 and 3.

The first issue being whether the CDA should be a design basis accident and the second issue, whether the site suitability source term selected envelopes the design basis accidents, all credible accidents and, third, whether the containment will limit the releases to 10 CFR 100 and I would say there are really two parts to that; one is the analysis called for under the requirements of 10CFR100 and the other one is -- the other aspect of that is, just the analysis of HCDA's using more realistic assumptions done in Appendix J.

Now, I would like to first address the question of whether the CDA should be a design basis accident.

First is, why is that important?

I believe that it's conceded by the Applicants and the Staff that if the Board reaches the conclusion that at this time the CDA cannot be demonstrated to be outside of the design basis accident spectrum, then they have to revisit the site suitability source term.

So, in effect, you must conclude that the CDA is within the design basis accident spectrum, in order to give a limited work authorization at the end of these hearings.

I will demonstrate the importance of that when we go through the site suitability source term calculations that are required under 10CFR100.

With regard to this issue, whether the CDA should be a DBA, I believe there are two sort of different cuts at this problem.

First is the cut that was made in our first hearings of the first week and, if I could characterize what I think are the differences that Intervenors see between the Intervenors and the other two parties.

On the one hand, I think everyone agrees that the burden of proof is on the Applicants or Applicants and Staff and we are saying that they haven't established that proof.

That they, in effect, have said they can do it but haven't demonstrated that they can do it.

Now, this is -- there is a difference, I believe, between the feasibility of designing equipment and demonstrating that that equipment, when designed, will meet its performance objectives.

Now, in this respect, I'm reminded of the statements that are attributed to Abraham Lincoln, where he raised the question with another party, if a cow's tail -- if you call a cow's tail a leg, how many legs does a cow have? And the other party answers, five.

31-7

And Lincoln responds, "No, calling a cow's tail a leg, doesn't make it a leg."

And I think that the differences here are that Applicants and Staff are calling a CDA outside of the design basis accident spectrum and we are saying that the proof has not been provided.

Now, in the early phase of the hearings, we argued on the -- basically with the Intervenors saying, "You really need to look at some probablistic analysis to make this demonstration and that the analysis that you provided really, on, you know, when you look at it scientifically, it really doesn't demonstrate that you've met the criteria."

Subsequent to that phase of the hearings, we now have Appendix J and of Staff Exhibit 8 and I believe you can go through Appendix J and demonstrate that the CDA at this phase of the process -- that the Staff and Applicants' best estimates would demonstrate that the CDA should be a design basis accident, on the basis of their own analysis.

Particularly the analysis provided by the Staff in Appendix J.

So, I want to first take you through that argument and then revisit the issues raised in the early phase of the hearing, the first week.

31-8

Subsequent to that, I will get into the site suitability source term analysis.

Now, first requirement is a criterion by which one should judge whether the CDA is a design basis accident. In order to determine whether CDA should be within the CRBR design basis, the Staff currently uses the safety objective that there be no greater than one chance in a million per reactor year of a CRBR radioactive release with the potential consequences greater than the 10CFR, Part 100 Dose Guidelines.

That's found at Tr.2277 to -79, Dr. Morris, Staff Exhibit 5.

The Applicants have also proposed this approach, Tr.1483,Clare, in Intervenors Exhibit 1, Pages 7 through 8.

JUDGE LINENBERGER: Excuse me. What was that Tr. reference?

DR. COCHRAN: Tr. 1483.

JUDGE LINENBERGER: Thank you.

DR. COCHRAN: Clare, and Intervenors Exhibit 1, Pages 7 to 8.

Now, I want to take you to, I believe it's Staff's Exhibit 8. In any case, it's the supplement to the Final Environmental Statement, Appendix J.

And we will begin at Page J-8, where Staff has estimated that the upper bound on the CDA probability is 10^{-4}. In other words, on the basis of the Staff's best estimate today, they cannot establish that the -- they have established that the CDA probability could be as high as 10^{-4}.

Now, in Mr. Soffer's, I believe, testimony with regard to Contention 5(b) and the National Security and Energy Security Issues, he argued that in looking at a particular direction, particular wind direction, such as towards the gaseous diffusion plant or the Y-12 plant, the probability would be a factor of 10 less than the -- you should account for a factor of 10 in probability to account for the fact that there is one chance in ten it will be in that particular direction in the wind rows.

So, we will be discussing a CDA frequency of less that 10^{-5} in what I will call the worst case condition, worst case direction, which is also the direction one should analyze in a site suitability source term analysis.

At 10^{-5}, which is the upper bound that the Staff has placed on the probability of the CDA in that particular direction, is less than the Staff's safety objective of 10^{-6}. So, it's appropriate, I believe, to analyze the consequences of such an accident with the design proposed by the Staff.

Excuse me. I mean proposed by the Applicant.

Now, you would believe -- you might believe that given that safety objective, you could find somewhere in the Environmental Impact Statement or the testimony of the Staff, an analysis testing whether an HCDA exceeds those -- of that probability would exceed the dose consequences of 10CFR 100 and, in fact, you do not find an analysis of the HCDA using realistic assumptions.

I will not use the site suitability source term, so-called conservative assumptions, in that particular direction by the Staff.

The Applicant, however, has analyzed that particular accident and I will -- in Applicants Exhibit 46, at Page 34, using what Applicants claim to be new data, replacing the calculation in Applicants Exhibit 1.

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DR. COCHRAN: Before -- I will revisit the Applicants' analysis, but, first, I believe that the Staff and the Applicants together have provided sufficient data in their own testimony and in Appendix J in order to make the calculation for the dosages in the worst direction for a CDA of -- in the Class 1 category, the 10^{-4} CDA, 10^{-5} in the worst case direction.

In Staff's Exhibit 5B -- excuse me, Staff's testimony on Contention 5(b), they do an analysis -- an HCDA analysis for the Oak Ridge Gaseous Diffusion Plant and for the Y-12 plant.

But the Gaseous Diffusion Plant is roughly 2 1/2 miles just outside of the low-population zone. However, it's not in the wind direction of the worst case conditions.

So we will have to take their doses that they derived in the direction and at the point of the Gaseous Diffusion Plant and sweep them to the worst case direction and at the low population zone boundary and see what the doses are there.

Now, that's a proper procedure, and, in fact, it was the procedure that Mr. Thadani used to calculate the dosages at Y-12 from the dosages at ORGDP.

It derives from the fact that the dosages -- you assume a source term and you multiply that by the

32-2

fraction of material released from the containment and multiply that by the X/Q, and then multiply that by the dose conversion factor to get the resulting dose.

And so the resulting dosages are simply proportional to the X/Q assumptions.

If you look at the Staff's testimony on Exhibit 18 -- Staff testimony on 5(b), you will find the K-25 doses for the thyroid and the whole body given at Page 6 in Answer 11 and on Page 7 in Answer 16.

I will just -- The site suitability source term thyroid dose -- and I will be looking at the thyroid in this particular calculation -- is -- in the K-25 direction is given as 320 millirem.

In the -- Wait just a minute.

In the direction of the worst case direction, the dose -- again using the site suitability analysis -- the dose at the LPZ boundary is -- to the thyroid using the site suitability source term calculation -- is seven rem or 7000 millirem.

That's found in the Staff's Exhibit 1, which is the site suitability report.

So they've done a site suitability calculation in both the direction of the ORGDP and in the worst case direction -- in the site suitability report; and those two doses -- those two dosages, 320 millirem and 7000

32-3

millirem -- differ by a factor of 20. That represents the difference in the X/Q's between the ORGDP direction and the worst case direction.

Now you can do a similar calculation for the whole body. The whole body dose is 19 millirem at K-25. In the site suitability report it's .3 rem or 300 millirem.

That's a difference of a factor of 15.

Those factors should be approximately the same. You may round off -- round off errors or some hidden assumptions.

But the ratio of the X/Q's is roughly a factor of 15 to 20.

Now, one might say, "Well, that's inappropriate because you're ratioing the 95 percent X/Q values, and we ought to" -- For this calculation, I wish to do it for the more realistic releases. So you should do the comparison for the 50 percent X/Q, and you can do that using the Staff's data.

That's -- Excuse me. Using the Applicants' data.

Applicants also calculate the dose at the K-25 plant and also in the direction of maximum dose. The Applicants' thyroid dose for HCDA's is seven rems at the ORGDP. That's in their testimony on 5(b) at Page 13.

32-4

And their thyroid dose for the HCDA conditions in the worst case direction is 85 rems. That's found in Applicants' Exhibit -- I believe it's 46 -- at Page 34.

The ratio between those two, 85 divided by 7 is 12, which again says that the -- using the 50 percent X/Q's and the Applicant's meteorology, the effect of shifting directions from the ORGDP site to the LPZ worst case direction is -- would be to increase the dosages by a factor of 12.

Now when you look at the Staff's HCDA analysis for the ORGDP, you will find that the thyroid dose at K-25, which is at the LPZ distance, is 100 rems.

So when you shift around to the worst case conditions, you should multiply, to account for the higher X/Q values, that dose by roughly an order of magnitude.

And in so doing, you end up with a thyroid dose of approximately a thousand rem, which far exceeds the 10 CFR 100 guideline values.

So in this calculation I have demonstrated that you take -- with the Staff's own best estimate to date of the probabilities of the CDA in the worst direction, which would be 10^{-5}, and demonstrated that the thyroid dose assumes the 10 CFR 100 guideline values, and I would conclude from that that the safety objective is not met. Therefore, that accident should be within the

32-5

design basis, and some corrections should be made.

Now, one can also -- I now want to revisit the Applicant's analysis, since the Applicant did the calculation directly in the worst case direction.

You'll recall that the Applicant first did the calculation, as I noted earlier, in their Exhibit 1, and then recalculated it in Exhibit 36 at Page 34.

The Applicants give several cases. I will look at Case 1, since that's the most benign of their cases. You see that they calculate in their Exhibit 46 a thyroid dose of about 85 rems, which is below the site suitability -- I mean, below the 10 CFR 100 guideline values.

But it also happens to be well below the calculation I just gave, based on the Staff's data.

How do you account for the difference between the two?

That you can find -- Well, first of all, you can calculate directly what that difference is because both the Staff and the Applicant calculate the thyroid dose at the ORGDP.

Staff's value was 100 rems, and the Applicants' value was roughly seven rems, there being -- in other words, the Applicants' thyroid dose is -- the Staff's thyroid dose under this HCDA realistic calculation is 14

32-6

times higher than that of the Applicants.

The Applicants' seven-rem figure is found in their Exhibit 18 at Page 13.

Now, the Staff testified -- Mr. Thadani testified in his testimony on Contention 5(b) that these differences were due to two factors: One, they used different -- the Staff and Applicant used different meteorology; and, two, they used different filter efficiencies.

You will find in his testimony that the differences are in opposite directions. The Staff has been more conservative with regard to filter efficiencies, and the Applicant has been more conservative with regard to meteorology.

I believe you will also find that in the cross-examination that the principal meteorological difference is due to the assumptions regarding the height of release.

Therefore, the factor of 14 difference between the Staff and the Applicant with regard to the thyroid dose would be actually higher if the Applicants and Staff had both used the Staff meteorology -- or if the Staff -- Let me restate that.

The effect of using the Staff's filter efficiencies, rather than the Applicants' -- all other

32-7

factors considered equal, would be larger than a factor of 14 because part of that difference has been offset by Staff's less conservative meteorology.

Now, you will find nothing in the record indicating that the Applicant disagreed with the Staff's filter efficiencies and their dose calculations. You will find nothing in the record indicating -- Well, I say "nothing in the record," nothing in the record other than the Applicants' own analysis, but they did not attempt to question the accuracy of the Staff's calculations or assumptions regarding those doses.

Similarly, you will find that in the cross-examination of Applicants' witnesses on Exhibit 46, that they had prepared no independent estimates of probabilities; and, therefore, the Staff's probability estimates in Appendix J are the only ones that are available.

The Applicants did point out several areas of conservatism in the Staff's analysis in Appendix J, but upon cross-examination they admitted that there were other errors that would lead to non-conservative -- in the non-conservative -- not errors, but there were other differences between --

JUDGE MILLER: Pardon me, Dr. Cochran.

Conclude, if you can, this portion. We're going to have to vacate the building. I promised we'd leave at five till six.

Finish your thought, and then you may resume in the morning at 8:00.

DR. COCHRAN: How much time --

JUDGE MILLER: Is this a convenient breaking place?

DR. COCHRAN: I'm almost there, if I could have about five minutes.

JUDGE MILLER: Well, I promised them we'd leave at five till six, which it is now.

People have to get some equipment here. I promised them an hour ago that we'd be out by 6:00, because I knew that we would carry over until in the morning. So if you could make some convenient note for yourself.

DR. COCHRAN: I can pick up tomorrow.

JUDGE MILLER: Okay.

DR. COCHRAN: -- with no trouble.

JUDGE MILLER: Until 8:00 in the morning.

(Whereupon, at 5:55 p.m. the hearing was recessed, to reconvene at 8:00 a.m., Friday, December 17, 1982, in the same place.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

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in the matter of: TENNESSEE VALLEY AUTHORITY (CLINCH RIVER BREEDER REACTOR)

Date of Proceeding: December 16, 1982

Docket Number: 50-537

Place of Proceeding: Oak Ridge, Tennessee

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Mary L. Bagby

Official Reporter (Typed)

Mary L. Bagby

Official Reporter (Signature)