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SUBCOMMITTEE ON CLINCH RIVER BREEDER REACTOR

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1 UNITED STATES NUCLEAR REGULATORY COMMISSION

2 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

3 SUBCOMMITTEE ON CLINCH RIVER BREEDER REACTOR

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5 Room 1046

6 1717 H Street, N.W.

7 Washington, D.C.

8 Tuesday, March 30, 1982

9 The Subcommittee on Clinch River Breeder

10 Reactor convened at 1:00 p.m.

11

12 PRESENT FOR THE ACRS:

13 SUBCOMMITTEE MEMBERS:

14 MAX CARBON, Subcommittee Chairman

15 J. CARSON MARK

16 MYRON BENDER

17 ACRS CONSULTANTS:

18 W. LIPINSKI

19 W. KASTENBERG

20 Z. ZUDANS

21 DESIGNATED FEDERAL EMPLOYEE:

22 PAUL BOEHNERT

23 ACRS FELLOW:

24 ALDEN BICE

25

1 P R O C E E D I N G S

2 MR. CARBON: The meeting will now come to
3 order. This is a meeting of the Advisory Committee on
4 Reactor Safeguards, Subcommittee on Clinch River Breeder
5 Reactor. My name is Max Carbon, Subcommittee Chairman.
6 The other ACRS members present today are Mr. Bender and
7 Mr. Mark. Mr. Okrent will attend tomorrow's session.
8 The purpose of this meeting is to review CRBR
9 principle design criteriia. The meeting is being
10 conducted in accordance with the provisions of the
11 Federal Advisory Committee Act and the Government in the
12 Sunshine Act. Mr. Paul Boehnert is the Designated
13 Federal Employee for the meeting.
14 The rules for participation in today's meeting
15 have been announced as part of the notice of this
16 meeting previously published in the Federal Register on
17 Monday, March 15, 1982. A transcript of the meeting is
18 being kept and will be made available as stated in the
19 Federal Register notice. We have the usual request to
20 speak loudly, and so on.
21 We have received no written statements from
22 members of the public. We have received no requests for
23 time to make statements by members of the public.
24 Let me lead off with a few comments here.
25 First, I think the idea of a roundtable discussion,

1 Paul, is a very good one. The purpose of this afternoon
2 and tomorrow morning is to review the principle design
3 criteria which the staff has developed to use in
4 evaluating licensing CRBR.
5 We do not have any intention at this time to
6 suggest that either the Subcommittee or the full
7 Committee make any written statement or anything.
8 Rather, the content of our review will come in the
9 discussions today and tomorrow, and of course they're
10 being recorded.
11 An agenda has been prepared which I think
12 everyone has, and I propose that we follow it in
13 general. But I'd like to suggest at the start here some
14 specific topics which we will have particular interest
15 in and encourage particularly the staff people to try
16 and bring out information relative to these points in
17 our discussion and in the presentation.
18 There are about six of these points and I
19 would like to go through them quickly. The first is,
20 what was the overall philosophy used in developing the
21 principle design criteria; why were some of the general
22 design criteria rejected and some modified and some new
23 principle design criteria instituted? I am using here
24 "principle design criteria" as related to CRBR, of
25 course, and general design criteria from Part 50 or

1 whatever. And I'm talking here in a broad sense, not on
2 an individual basis.
3 Question: How did you decide which or what
4 were acceptable, and again in a broad sense, aiming at
5 the basic approach? And then as a subset, I would ask
6 if you would bring out the relationship between the PDC
7 and any criteria that the project may have developed and
8 what effect the project developed did have on yours, and
9 also what influence the ANS -- ANSI, I guess, proposed
10 criteria have had on your principle design criteria.
11 The second question or comment relates to how
12 is the staff going to translate from a PDC to a plant
13 acceptance criteria. And in part I think this question
14 says, of what value are the principle design criteria,
15 how are you going to use them, how are you going to
16 apply them, and so on?
17 The subject is of particular importance to
18 Dave Okrent, as you are fully aware, and that's why it
19 is on -- well, it's not on the agenda. It's on the
20 agenda for today, but it will be deferred until tomorrow
21 morning. And we members of the Subcommittee and
22 consultants can in general hold off on this question of
23 how they will translate from the criteria, the design
24 criteria, to actual acceptance.
25 Number three in this list of points here:

1 There are several very important technical issues on
2 which the principle design criteria are either silent or
3 vague, and among these -- again, these are ones that I
4 personally consider very important issues on the safety
5 of the CRBR. One of these is the definition of design
6 basis accident and the second is the role of CDA's and
7 energetics. The third is the definition of the site
8 suitability source term. Fourth is the margin of safety
9 against seismic events. Fifth, the natural circulation
10 decay heat removal requirement. Sixth, containment
11 confinement considerations, including perhaps questions
12 about vented containment. And seven, sabotage.
13 Now, obviously some of those don't belong in
14 design criteria, but if you would do as much as you can
15 to relate the criteria to these issues and vice versa, I
16 think it would be helpful to our understanding. And of
17 course, some of those will be quite important in
18 tomorrow morning's discussion when we translate going
19 from design criteria to acceptance criteria.
20 Number four on this list is a personal
21 question that I want to raise: A basic concept for
22 safety in an LWR is simply the thought of keeping water
23 on the core and keeping containment intact, and we can
24 go to some pretty great and sophisticated ways to keep
25 water on the core, such as other people have mentioned:

1 fire hoses or hauling some water from a pond or river,
2 or some such thing.
3 You can't do things visually or theoretically
4 that simply with an LMFBR. It's a little bit more
5 sophisticated. And if you have any philosophical
6 thoughts on how you think in terms of relating safety of
7 an LWR to an LMFBR on some of these basic concepts, I
8 would welcome hearing about them.
9 Number five, there are some specific design
10 criteria on or about which we will have questions, and I
11 will mention a few of the perhaps more important ones of
12 these. One might be the requirement for containment
13 retention time. It is discussed in PDC No. 14, but it's
14 kind of vague.
15 Number two has to do with the use of the
16 single failure criterion in design criteria 19, 22, 35,
17 and elsewhere. The use of the criterion in those PDC's
18 raises several questions about the adequacy of the
19 single failure criterion, and I would welcome any
20 comments there.
21 Number three, are there questions about the
22 diversity required in the decay heat removal systems and
23 in the plant protection systems.
24 Number four on this list would include station
25 blackout, probably; and number five would be the

1 definition of "postulated accidents in PDC 41 and
2 elsewhere."
3 Then number six on this total list is --
4 obviously we are going to have several questions about
5 specific criteria, specific matter. An example might
6 be, why was GDC NO. 28, reactivity limits, deleted from
7 the PDC?
8 I don't think it's at all necessary that we
9 discuss every question that we will have. It's not very
10 likely, either. And some of the unanswered questions,
11 if we have them tabulated in written form -- and some of
12 them we do -- we'll forward them to you for such
13 consideration as you care to make of them.
14 That's all I wanted to say at the start here.
15 But let me call on the other Subcommittee members for
16 comments. Mike?
17 MR. BENDER: I think we've got a lot to chew
18 on without adding to the discussion. I'll pass for the
19 time being.
20 MR. CARBON: Carson?
21 MR. MARK: I have a number of questions, and
22 perhaps this is not the place to inject them. But I do
23 want to understand better than I do a number of things
24 which came to mind in the document that I have here,
25 research and technical assistance plan for CRBR

1 licensing. I'm not quite certain of the source of this
2 or the date of it. But it must have been some time late
3 last year.
4 And just to give an example -- there are
5 several -- major emphasis has been placed on the
6 avoidance of CDA's, the primary contributor to the risk
7 from an LMFBR. Now, is it or isn't it the primary
8 contributor to the risk? PRA's are probabilistic
9 assessments. Where they have not been done the source
10 term has not been defined.
11 Who is it that knows, and on what basis does
12 he know, that this is the primary contributor to risk?
13 I have the feeling that it's probably not, but I don't
14 know, and I would like to have, you know, something that
15 would make that clearer to me than it is now.
16 And there are questions of that sort that I
17 would like to introduce if and when -- and it may not
18 come out -- when the staff's plans reach such a point.
19 MR. CARBON: Walt?
20 MR. LIPINSKI: Preceding the exercise on CRBR
21 design criteria, FFTF had gone through a similar
22 exercise. And naturally, they took the position that
23 not every one of the criteria in Appendix A applied, and
24 some they adopted literally. But those they wrote that
25 were special amounted to 27 criteria over and above the

1 Appendix A criteria.

2 Now, some of these are specifically addressed

3 in that list of items you had run through where FFTF had

4 included these as criteria. But I wonder what the

5 staff's position is in terms of where they stood and the

6 reduced set that they have for Clinch River.

7 MR. CARBON: Bill?

8 MR. KASTENBERG: In addition to the questions

9 that you raised, I noticed in this statement of one or

10 several criteria that you propose you left off fuel

11 design limits where in the LWR criteria, I think with

12 respect to the residual heat removal system, they talk

13 about limits and you don't have that phrase in some of

14 yours, and I wonder if that's an editorial oversight or

15 is there some difference in philosophy.

16 MR. CARBON: Zenon?

17 MR. ZUDANS: All the questions I have fall

18 under the second part. So if we come to that one we

19 come to it --

20 MR. CARBON: Well, let's proceed with the

21 meeting, then. And Paul, I will call upon you to

22 begin.

23 I should say for the record, Mr. Check is

24 chief of the CRBR project office of the NRR staff.

25 Paul?

1 MR. CHECK: Thank you.

2 One of several unusual or special aspects of
3 this case is that it has had, or will have had, two CP
4 reviews. Neither one was or is complete, of course.
5 And we are trying to make the current review look like a
6 continuation of the earlier one, with some success. But
7 it isn't always easy.

8 The principal difficulty that we face is that
9 the former review was performed by a different team, and
10 all that they did isn't readily recoverable, especially
11 in basics. This is not meant as any criticism of them.
12 When the curtain came down on that earlier review, it
13 immediately fell to a very low priority in all of the
14 work in NRR, and so those people were off doing other
15 things pretty quickly.

16 At any rate, what this means is that in some
17 instances we, the present team, must now virtually do
18 again what was done before, and this certainly isn't
19 necessarily bad. When one is trying to maximize on
20 safety, then one increases the probability that things
21 that were overlooked will now be caught by doing things
22 twice.

23 We are here today to discuss the principle
24 design criteria for the Clinch River Breeder Reactor.
25 These -- and that is the present published version, as

1 the Committee knows -- were developed by the former
2 team. At, I believe it was, the January meeting at the
3 full Committee, Harold Denton suggested that the
4 principle design criteria might be considered by the
5 ACRS as the principle design criteria are being
6 revisited by the staff.
7 At our February meeting with the Subcommittee,
8 Bill Morris outlined the manner in which we are taking
9 up the principle design criteria. In that discussion,
10 Dr. Morris describe the relationship between the design
11 criteria and the events the plant is postulated to
12 encounter. He pointed out that the process of
13 developing and improving the principle design criteria
14 is in large measure a significant component of the
15 construction permit safety review.
16 Our safety review of the CP application is now
17 underway, and it's timely therefore that we meet with
18 you on design criteria so that we will have had the
19 benefit of your advice as our review matures and the
20 development of the principle design criteria
21 progresses.
22 MR. MARK: Paul, the request for a limited
23 work authorization has been rejected, I believe.
24 MR. CHECK: That's correct.
25 MR. MARK: Consequently, the CP -- or is it an

1 LWA -- will be the first thing that actually surfaces.
2 Which will be first, and when?
3 MR. CHECK: It is difficult to say precisely
4 what will happen first. Our schedules, the ones that we
5 advertise presently, show a limited work authorization,
6 one being granted in the neighborhood of the middle of
7 1983.
8 MR. MARK: That does not require a hearing?
9 MR. CHECK: Yes, it does. And the CP --
10 MR. MARK: It was the exception to the hearing
11 that was not accepted?
12 MR. CHECK: That's right. In the simplest
13 terms, I view the exemption request as only a request to
14 do something in parallel, to proceed with that first
15 stage of site preparation or construction, if you will,
16 at a time when the hearing for environmental purposes
17 was being conducted.
18 MR. MARK: So one anticipates there will be
19 time for a hearing and a decision on the LWA within a
20 year from now?
21 MR. CHECK: That's correct. In fact, the
22 hearing is scheduled to begin the end of August on NEPA
23 matters and this thing called site suitability.
24 MR. BENDER: Paul, while we're on the subject,
25 if I try to make a judgment about the design criteria

1 and in the terms of how the regulatory agency
2 establishes that they are the right criteria, there's an
3 approach that Dr. Carbon suggested, namely let's see
4 what the reasons are for the criteria and why some are
5 there and some are not.
6 One can deal with it that way, and it's a
7 logical thing to do. There's also a procedural question
8 that has to be addressed, and that is what is the
9 mechanism by which the Commissioners or the Commission
10 establishes that the rules say judge the plant by these
11 criteria. Now, you have rules that have to do with the
12 design criteria for LWR's, and I can find them in 10 CFR
13 50.
14 What is the comparable thing for the CRBR?
15 MR. CHECK: The short answer is that Appendix
16 A, to which I think you are referring, does not restrict
17 itself or is not restricted to light water reactors. It
18 is guidance as well for other kinds, and the preamble
19 says that. So it is a logical starting point for us, as
20 it would be for any LWR.
21 MR. BENDER: I think that's all right, and as
22 a matter of fact that's why from the beginning we've
23 encouraged the staff to say, well, look at the existing
24 criteria. But now if people raise questions as to why
25 there are some criteria that are omitted and others that

1 are added, then I have to say in a procedural sense we
2 are deciding that some of the criteria that the
3 regulations said apply do apply and others do not.
4 MR. CHECK: All will be addressed.
5 MR. BENDER: How?
6 MR. CHECK: In the application, for one, and
7 in our SER this whole process will be laid out.
8 MR. BENDER: I'm thinking in slightly
9 different terms. The SER and the application both must
10 deal with the Commission rules.
11 MR. CHECK: That's correct.
12 MR. BENDER: And we have agonized for months
13 and years sometimes over trivial changes in the rules as
14 they exist. And I'm saying now we're dealing with
15 something that represents somewhere near wholesale
16 changes in the way in which LWR's are dealt with. I
17 think they have to be done and I want to know what the
18 procedural aspects are that enable us to say that the
19 Commission's regulations permit you to invent new
20 criteria and put them into the rules through the SER
21 process.
22 MR. CHECK: Okay, we'll talk about that.
23 MR. BENDER: It's just a matter of
24 understanding.
25 MR. CHECK: When first Rich begins to talk a

1 little bit about how the current ones were developed and
2 as Billy develops how we're doing our current job, I
3 think this will come out. Let me complete my opening
4 remarks, if I may.
5 MR. BENDER: I'm done.
6 MR. CHECK: I think I remarked about the
7 timeliness of what we're doing here, and I left on the
8 thought that it's good that we do this because it is
9 going on with us now and it's useful to have the
10 impressions of how it should go on, what we're doing, so
11 that we can improve.
12 It would be unfair to the formulators of that
13 present version of PDC if I didn't say that I have a
14 strong inclination that we will probably, rather than
15 reinventing here in this current review, will probably
16 be converging on a sort of an embracement or a
17 reaffirmation of what was done before.
18 At any rate, this is the climate I want to
19 create here, that we've come to this meeting not to
20 defend principle design criteria -- we will be able to
21 in some instances -- but to discuss them with you and
22 with your consultants.
23 To help us with this, Rich Stark will lead us
24 through a history of the present principle design
25 criteria and then Bill Morris, over several sessions of

1 the agenda, will outline how we will get to the
2 principle design criteria that will be described in the
3 SER. And just before I turn it to Rich, I will comment
4 on that core dump.
5 We had at the outset a wonderful new idea that
6 we could put the agenda aside and start picking through
7 that and probably be here on Saturday. The good things
8 to talk about and that I think sets the right tone for
9 this meeting, what is it we're trying to do and why is
10 it okay. And then once we can agree or get a consensus
11 on what is the right approach, then we can go on and
12 start filling in some of the holes.
13 MR. ZUDANS: Mr. Bender's question raised
14 turmoil in my mind. What is it that CRBR will have to
15 satisfy? Is it Appendix A or the new set of criteria or
16 both, or Appendix A with some modifications to the
17 criteria that are discussed in the PDC of CRBR?
18 MR. CHECK: We'll try to show that it's a
19 derivative process and the rule is Appendix A, and then
20 we will derive from it something that fits here.
21 I would add one comment -- maybe it's a
22 request -- that I think the applicant participate in
23 some of this. I'm sure he's got good ideas. He's got a
24 safety philosophy that rivals our own, and I hope we are
25 converging on a common one. So as it seems appropriate,

1 I hope it would occur to us to invite him to comment.

2 I will just identify for you, Carson, the

3 source of the document that you're reading. It is the

4 product of a joint NRR-RES working group chaired by Bill

5 Morris, established by Dircks. So that the research

6 studies done on LMFBR's is the needed work to license

7 CRBR.

8 MR. MARK: I thought it was an excellent

9 document. I have some questions about statements in it,

10 but not an objection to the document.

11 MR. BENDER: I understand. I just wanted to

12 answer your question. You wanted to know about the

13 origin. And Bill will be prepared to talk about it.

14 So with that, unless there's something else to

15 be unloaded, I'm going to turn it over to Rich. Rich

16 Stark, the project manager.

17 MR. STARK: I have a few slides.

18 (Slide.)

19 MR. CHECK: While Rich is sorting things out,

20 let me give a fuller answer, I hope, to the question

21 about what is the starting point, what's the pole start

22 in all of this. There are sections in Part 50 itself

23 which address, for example, the findings which must be

24 made in 50.34, I believe is the section. And high among

25 the four findings that need to be made for a CP is the

1 establishment, the agreement on principle design
2 criteria for plants.
3 So it is the first fit task that the staff and
4 applicant must address as it proceeds. Appendix A came
5 later, of course, and represented the distillation of a
6 lot of experience and a lot of discussion, and
7 represents for us a good place to start.
8 MR. STARK: Good afternoon.
9 I guess before I get into my slides, I'm happy
10 to see that I think some of the same questions that
11 you're asking we were asking ourselves a few months
12 ago. For example, a continuation of your question, Mr.
13 Bender, how did Fort St. Vrain get through this
14 process? We talked to some of those people and I think
15 as we go through some of this you'll get the benefit of
16 some of the answers that we found.
17 The first item that I wanted to show here, I
18 wanted to take some time to put down how I think we got
19 to where we are today. 10 CFR Section 50.34 defines the
20 requirements for PSAR submittals and requires that the
21 applicant submit principle design criteria that indicate
22 the design criteria for the intended plant.
23 10 CFR Part 50, Appendix A in particular,
24 discusses general design criteria, and in the
25 introduction to Appendix A it indicates that Appendix A

1 also applies to non-water reactor plants, and some of
2 the words that are used in the introduction are as
3 follows. They feel Appendix A provides guidance in
4 establishing principal design criteria for nuclear
5 plants other than LWRs, and this basically was factored
6 in very early in the game and continues to be. And this
7 is what the people in Fort St. Vrain were trying to do,
8 to use the Appendix A criteria to the extent possible.
9 Back in 1963 there was an ANS Subcommittee 24
10 which was established. It later became ANS 54 and they
11 issued a draft criteria. The criteria had a title, it
12 was called "General Safety Design Criteria for an
13 LMFBR." That was issued, and back before 1974 the staff
14 considered both the general guidance that existed in
15 Appendix A as well as the draft criteria that the ANS
16 people had issued.
17 And in July 1974 the staff issued a document
18 called "Interim General Design Criteria for the Clinch
19 River Breeder Reactor Nuclear Power Plant." That
20 criteria was addressed by the applicant in their PSAR
21 whenever it was docketed, in April 1975.
22 Now, what the staff did at this particular
23 point was, they looked at the responses that the
24 applicant made to these interim design criteria and
25 reviewed them. And based on the staff criteria review

1 results, they revised the design criteria and more or
2 less published the design criteria. They sent a letter
3 to the applicant saying, these are the revised design
4 criteria, in January '76.
5 MR. MARK: Could I ask, ANS-54, issued in the
6 mid-sixties, is of course not a thing which you
7 automatically subscribe to?
8 MR. STARK: That's correct.
9 MR. MARK: But have you in effect subscribed
10 to it in 1974, or are you asking -- we'll take this
11 mostly, but we want the following changes?
12 MR. STARK: It's going to come up again on the
13 next slide, because ANS-54 played a more recent role.
14 But the way I view it right now is that ANS is trying to
15 write general design criteria for a loop-type LMFBR
16 plant. And what the staff was thinking about relative
17 to Clinch River was looking for general design criteria
18 for the demonstration plant, that are of course very
19 similar to the LMFBR plants that ANS is referring to.
20 They're almost the same, but not necessarily the same.
21 MR. CHECK: We have a more limited
22 responsibility, and that is to establish principle
23 design criteria for CRBR. Now, it may be that those are
24 a fairly good reflection of LMFBR general design
25 criteria that would be convenient. But to the extent

1 that they are not, it doesn't matter unless we've been
2 silly.
3 MR. MARK: However, you're specifically
4 restricting yourself to CRBR?
5 MR. CHECK: Correct.
6 MR. MARK: You're not looking at the CDS and
7 you're not considering pool-type plants, which would of
8 course in some respects be more favorable necessarily?
9 MR. CHECK: That's correct.
10 MR. ZUDANS: This is one thing that bothered
11 me when I read the criteria. It seems like what is
12 first, the chicken or the egg. You know, in principle
13 if you sat down today and with the best people you can
14 get and create a set of criteria and then said, here is
15 a set of criteria and you go and design it, things
16 wouldn't work out.
17 So what you're really doing is the criteria
18 that are being developed and the plants are being
19 designed at the same time. Now, you are in a better
20 position than your predecessors because they did not
21 have a design as far advanced as you have. Will it ever
22 become clear that the design came after the criteria and
23 not before? And if so, why is the criteria here? Why
24 not just do the safety review and satisfy Appendix A?
25 MR. CHECK: There are several questions in

1 there. With regard to the chicken and the egg, if we
2 look at how the general design criteria themselves were
3 established, then if reactors are chickens chickens came
4 before eggs. General design criteria were a
5 codification of good practice.
6 And you're right. You're right, you correctly
7 perceive the tighter loop that we're in. But I wouldn't
8 torture myself with making the case that this came first
9 and that came second. It's in the nature of things that
10 we sort of learn by doing. These are evolutionary
11 processes.
12 MR. ZUDANS: I understand the light water
13 reactor was an evolutionary aspect, and to use the best
14 experience from a couple of them. This is a specific
15 CRBR. It's not likely it will ever be repeated.
16 MR. CHECK: The more reason we should be more
17 restrictive, in our view.
18 MR. ZUDANS: But that's in the safety review.
19 It has nothing to do with criteria. Why not make a
20 larger set of criteria apply to CRBR, such as LMFBR
21 criteria?
22 MR. CHECK: I don't know how practical that
23 is. It would mean I would have to endorse the larger
24 set. I don't know how long that would take and what
25 would be involved. And the regulation is -- look in

1 Part 50.34 and it says one must agree upon principal
2 design criteria, and that's what we're working toward.
3 We realize there are more general statements
4 of criteria, but we're not sure -- in fact, we know they
5 are not required for establishing plant criteria.
6 MR. MARK: I presume that ANS-54 was not in
7 any way restricted to CRBR?
8 MR. CHECK: No, of course not. It's much
9 broader. And we are participating, too.
10 MR. BENDER: Paul, I don't want to appear to
11 be a proponent of Dr. Zundans' approach, although I can
12 see it as an effective way of doing things. But I do
13 think we need to be conscious of whether we are
14 tailoring the criteria to the existing design or
15 starting with a set of criteria by which the design will
16 be measured.
17 I don't want you to try to defend one approach
18 or the other, but I think the latter has to be the way
19 in which it's done. We have to say the criteria were
20 there and would have been there whether the plant was
21 designed or not, and then measure against it. You may
22 even have to take exceptions to the criteria for the
23 purpose of CRBR. I expect we will.
24 But I think for the purpose of public
25 credibility one has to think in terms of the regulatory

1 agency having a basis for judgment, and I think some
2 attention needs to be given to whether that basis is one
3 with which we are comfortable. I wouldn't be surprised
4 if you ought not to get the Commissioners concurrence in
5 whatever that set is, or at least the method of
6 evolution. That's all.
7 MR. CHECK: Bill Morris feels the need to add
8 to this.
9 MR. MORRIS: With regard to whether you can
10 have a set of criteria developed of the sort that we
11 deal with in the GDC and PDC before you know a good deal
12 about the design, I intend to try to show later on that
13 there is an intimate relationship between the kinds of
14 events that can be experienced and the kind of criteria
15 that you impose. And until you know something about the
16 design in some general form, you cannot adequately
17 demonstrate what those kinds of events might be.
18 And so it seems to me that it would be an
19 artificial thing to do to try to take the separation too
20 far. If you did it, what you would come up with, I
21 believe, would be a set of criteria that would look
22 different from our GDC's and the PDC's for Clinch
23 River. That is, they would be very generalized. And I
24 think this attitude that you take could be addressed to
25 the GDC's as well as to the PDC's.

1 The kinds of criteria that we actually have to
2 work with are intimately related to the events that can
3 occur for given kinds of designs, and you're talking, I
4 think, here about a completely different philosophy for
5 kinds of design criteria.
6 MR. CARBON: Let me take the Chairman's
7 prerogative here of moving us on. But let me also
8 emphasize that I think Mr. Bender expressed very well
9 what both colleagues had in mind, and I think it's a
10 very important concept and I would urge that you not
11 simply disregard it. Let's move on.
12 MR. STARK: Okay. I was going to make one
13 comment on ANS-54, that appears in a memo we have, I
14 think from Mr. Lipinski, and we'll be addressing it
15 perhaps tomorrow on the agenda. But in many cases, even
16 if you look at your letter, ANS deletes as many times as
17 it adds.
18 We're reviewing it very carefully because we
19 want to make sure we know what they're deleting before
20 they delete some things in the existing principle design
21 criteria. So I don't think it's a black and white
22 issue.
23 So with that I'll go back to the slide I have
24 up here.
25 MR. MARK: Could I ask, Mr. Chairman, at the

1 expense of prolonging this 30 seconds, if you contend --
2 I don't wish to push this on you -- that ANS-54 is the
3 perfect set of criteria or an acceptable set set, if you
4 imagine that there is such a thing, and that this
5 perhaps might be a sample -- it would, I imagine, be
6 capable of being used to license in this country, to
7 license Super-Phoenix, whereas what we're doing now will
8 not.
9 MR. CHECK: That's true. I don't know how we
10 came across before as sort of doctrinairely espousing a
11 particular approach as being clearly superior. We say
12 that one could approach it in several ways, or at least
13 two. But there is, I hope, seasoning our view a strong
14 element of practicality, and I think that --
15 MR. MARK: I think it's more immediacy than
16 practicality.
17 MR. CHECK: Correct. And we find that we
18 probably could not get ourselves to this embracement
19 position on ANS as fast as we can on plant principle
20 design criteria for CRBR.
21 MR. MARK: So there's still a follow-on job?
22 MR. CHECK: Oh, and Bill Morris belongs as a
23 participating member.
24
25

1 MR. STARK: The staff had discussed the
2 results of the Applicant's PSAR and had revised the GDC
3 and had called it the CRBR plant design criteria in
4 January of '76. I noticed that was also given to the
5 ACRS in January '76, and this version is the same design
6 criteria that we're looking at again today.
7 The Applicant -- I guess I have a typo in
8 there -- the Applicant adopted the proposed design
9 criteria, and these were the proposed design criteria in
10 January. They were adopted in June of '76 and have
11 since been incorporated into the PSAR. The same January
12 '76 documentary also appears as Appendix A to the site
13 suitability report that the staff had issued in March
14 1977.
15 And I have one more item.
16 (Slide.)
17 MR. CARBON: Before you leave that, did the
18 ACRS write anything in the January-June '76 period?
19 MR. STARK: I can't find it. I can't find
20 anything. I wish I could.
21 One other item is that ANS-54, which I guess
22 is now suspended in its activity, issued the general
23 safety design criteria for a loop-type LMFBR nuclear
24 power plant in November of 1980. And the situation we
25 have today is we have the principal design criteria on

1 the books, and we also have an opportunity to look at
2 ANS-54.1, which has recently been released, and try to
3 determine if the principal design criteria needs to be
4 amended or if it's satisfactory.
5 (Slide.)
6 I have one other slide that I think might be
7 helpful to put on. It's kind of a safety philosophy,
8 and I think it may help us somewhat. At least it will
9 give you some idea, I think, what the staff has been
10 trying to keep in the back of their mind.
11 The May 6th letter from NRC to the Applicant
12 -- Denise to Caffey -- stated that "CRBR should achieve
13 a level of safety comparable to current generation
14 light-water plants according to all current criteria for
15 evaluation, and that the design approaches to accomplish
16 the required level of safety should be similar or
17 analogous to LWR practice.
18 We've been trying to look at what do we do in
19 the light-water business. How do we use Appendix A and
20 how do we make Appendix A without -- if we could
21 maintain the maximum extent possible Appendix A, we
22 think our job would be consistent with the philosophy
23 that existed in 1976; so we are trying to utilize
24 Appendix A to the maximum extent possible.
25 MR. MARK: This historic letter from Denise to

1 Caffey six years ago, particularly said current
2 light-water plants, I presume while you want to follow
3 the text of that literally, you want to rephrase
4 "current" to mean 1982.
5 MR. STARK: I think you can read that into
6 it. I think we are.
7 MR. MARK: I was sure you were.
8 (Laughter.)
9 MR. BENDER: Excuse me. We keep coming back
10 to these water-cooled reactor design criteria. I guess
11 I am persuaded to say that there ought to be a document
12 which states why the criteria which are being proposed
13 and in fact will be used are an appropriate substitute
14 for what are now stated as the light-water criteria. I
15 think you just have to have something that explains why
16 they are okay, and I'm not sure that I have seen that
17 yet. I don't even know what form the explanation ought
18 to take.
19 It might be that they're okay because in the
20 time frame we've got that's all we can do, or it might
21 be they're okay because we've gone through them item by
22 item and compared what is needed for an LMFBR versus
23 what might be needed for an LWR and concluded that this
24 is an equivalent set. But somewhere along the way I
25 think you do have to lay it out and say we have a basis

1 for making this decision, and I don't think that was
2 done in 1976.
3 MR. CHECK: You're correct.
4 MR. BENDER: And so we're still in the
5 position of saying that we're using a letter written by
6 a member of the staff to a project manager is a
7 statement of NRC policy, when I think that individual
8 may not have been in a position to have had the
9 authority to say not only is that what we're going to do
10 but that's the policy of the NRC. And that's what I
11 think we need to be sure of.
12 MR. STARK: The introduction to Appendix A on
13 the GDC does foresee the use of design criteria for
14 other than light-water plants and indicates that these
15 shall provide guidance or will provide guidance,
16 whatever the appropriate words are.
17 So it's not a black and white answer to your
18 question, but the introduction attempted to address it
19 and felt these would still be a good starting point or a
20 point of comparison. They didn't indicate what the end
21 item would be.
22 MR. BENDER: I agree with what you're saying.
23 As a matter of fact, when we started reviewing the plant
24 back in the early '70s I think think that was the tack
25 the ACRS decided to take, was let's see how they match

1 up against light-water criteria. But I just haven't
2 seen the closure point yet. And when you're dealing
3 with things in a legal framework, I think the closures
4 need to be identifiable.
5 MR. CHECK: We must not have made our point
6 earlier.
7 First, we couldn't agree with you more. In
8 fact, some of our difficulty stems from the lack of that
9 document which describes the bases for the decisions
10 that were made.
11 MR. MARK: That's going to be essential.
12 MR. CHECK: It will certainly be, and it will
13 be a significant part of our SER. It will be in there.
14 I know we can't avoid that.
15 MR. BENDER: I think your timing is wrong. I
16 think you have to get that out before you put it in the
17 SER. That's what I think needs to be done. So that
18 when the SER is put out there is no opportunity to
19 challenge the question of whether you had a suitable set
20 of criteria to judge the plant by.
21 MR. CHECK: That's why we're doing certain
22 things in parallel. We're having a meeting such as this
23 so we can be as informed as we can. But there are
24 questions of practicality. I cannot string everything
25 out in series.

1 MR. BENDER: Practicality is an important
2 question, and immediacy is an important question; and I
3 think we are using the two terms interchangeably when
4 they don't mean the same thing. But in the absence of
5 the document you have I believe somewhere along the way
6 you need to get the Commissioners' concurrence that this
7 is a good approach. That's what I'm suggesting.
8 MR. CHECK: You've made that point several
9 times, and we'll consider it. I want to turn that
10 over. I'm suggestible, but even that takes time. And
11 if it isn't essential, then we may choose not to do it.
12 I want to be direct with you, but on the point
13 of the philosophy --
14 MR. BENDER: I'm thinking in terms of a letter
15 which you need to have from the committee. If you don't
16 need the letter from the committee, then most of the
17 questions that are being raised here are not even
18 relevant. But if you need a letter from the committee,
19 the committee has to have that guidance before it can
20 write a letter which has any usefulness in terms of your
21 SER.
22 MR. CHECK: Or alternatively, the SER will
23 describe to the committee's satisfaction the process.
24 MR. BENDER: I don't know that that's a
25 suitable alternative. Let me put it that way. It

1 wouldn't be a suitable one to me at this stage of the
2 game. Maybe later on I'll change my mind.
3 MR. CHECK: Can I go to the safety
4 philosophy? Was there anything particularly offensive
5 about that, leaving aside the first --
6 MR. BENDER: That's a good statement. What
7 you said is not offensive. That's a different term. I
8 said it's just not -- there's no basis for judging
9 unless you put the judgment criteria out before you
10 present your case.
11 Go ahead.
12 MR. STARK: That's the end of the history.
13 MR. CARBON: I would like to ask Paul a
14 question about this. It's a nice statement and
15 obviously a good thing to shoot for, but I guess I
16 personally don't really know how to do what it says
17 here. Do you worry much about that?
18 MR. CHECK: We think about it, and I'll let
19 Bill Morris tell you what he thinks.
20 MR. MORRIS: It has to do with the way in
21 which we would judge whether or not comparability to
22 light-water reactor safety has been achieved. I think
23 that there are a number of practices that are applied to
24 light-water reactor safety that are equally applicable
25 to this plant -- the protection system, seismic design,

1 any number of practices which can be judged to be
2 comparable on the face of it through the review
3 process. We believe we will achieve that.
4 There has been a PRA done by the Applicant
5 which is being revised. Throughout the review we will
6 have a team of reviewers evaluating that PRA work. We
7 will look at the feasibility of achieving that early in
8 the review; that is, we will look at the feasibility of
9 imposing design measures that will assure that we will
10 achieve that or can achieve that early in our review.
11 And we will have accident delineation studies, a number
12 of measures that we will be taking to assure
13 comparability. And ultimately it will be a matter of
14 judgment, and there will be no particular number that
15 will come out of a calculation that will allow you to
16 say aha, we've achieved it, but there will be a basis
17 for that judgment. I believe you will be presented with
18 that judgment as the review proceeds. And the PRA
19 ultimately is a tool for making that kind of a judgment,
20 but there are other ways that we intend to achieve that.
21 MR. CARBON: Well, I guess all I can say is
22 that outside of your final judgment I really don't think
23 there's any way you can do it. I don't think PRA will
24 do it for you because it's not that good, and we don't
25 have enough data on this plant. We speak of practices,

1 but practice doesn't compare sodium/water interaction to
2 high pressure primary lines and so on. And I think
3 maybe we'd better not dwell on this because I don't know
4 what to say about it, but it isn't clear to me that this
5 is any more than just a goal.
6 MR. CHECK: It's a philosophy, it's an
7 objective. It isn't a test.
8 MR. MORRIS: The decision will not be arrived
9 at by scientific methods necessarily, but rather a
10 number of ways to make an ultimate judgment will be used.
11 MR. CARBON: Fine.
12 MR. MARK: I'm sorry you seem to exclude the
13 general field of science --
14 (Laughter.)
15 -- From the business of trying to arrive at an
16 ultimate judgment.
17 MR. ZUDANS: If at any point one could
18 question the last line, in the couple of words in the
19 line before -- you cannot compare the safety of these
20 plants with a practice analogous to LWR acceptance
21 criteria.
22 MR. STARK: I think if you look -- what Bill
23 was trying to say, if you look at the PSAR --
24 MR. ZUDANS: I'm reading what's on the slide.
25 For example, it says light-water practice. There is no

1 such thing as high temperature in the light-water
2 practice; yet it is essential for you to demonstrate the
3 LMFBR can satisfy the industry requirements, the ASME
4 criteria. So right off the bat you already modify it by
5 an order of magnitude -- the acceptance levels. You
6 cannot do that. You don't have to do that in
7 light-water practice. So light-water practice alone is
8 not to be judged by this criteria.
9 MR. STARK: What you're saying is in fact
10 correct. We may look at current light-water practice
11 for QA procedures or for training operators, and it's
12 applicable. And what we've been doing is looking at the
13 test that's used for light-water and see if it's
14 applicable. And if we come to a high temperature case,
15 then it's not applicable and we have to look and see
16 what is the comparable or the acceptable; and in some
17 respects that's why we don't have a one-to-one
18 correspondence on the design criteria between Appendix A
19 and the PDC.
20 MR. ZUDANS: You don't have that similarity in
21 any of the structural responses. Seismic design is
22 entirely different than seismic design in LWRs, so you
23 do have problems.
24 MR. MORRIS: In that particular case, whereas
25 you have high temperatures in these designs, you don't

1 have high pressures. And even though there isn't a
2 one-to-one correspondence throughout the design, there
3 will be ways in which one can make judgments that we
4 have adequately achieved a comparable level of safety
5 for this plant to a light-water reactor plant.
6 MR. ZUDANS: Maybe I shouldn't say anything,
7 but I perceive it as an extremely difficult task to make
8 a comparison in levels of safety, because your
9 acceptance criteria have to be completely different, so
10 how are you going to make a judgment that this is safe
11 or vice-versa? It's very different.
12 I think this kind of design should have its
13 own set of safety goals, and there's no comparison to be
14 made, or what's similar to light-water reactors. I
15 don't think you'll be able to compare it.
16 MR. MORRIS: It would be interesting if it
17 were a goal that was not -- what would it be? It would
18 be -- the question I think would be --
19 MR. ZUDANS: Adequate levels of safety
20 consistent with whatever the design criteria says. You
21 shall not exceed certain limits.
22 MR. MORRIS: We're talking now about the
23 difference between a very general goal and a set of
24 specific acceptance criteria for limits.
25 MR. ZUDANS: This is a very general goal, this

1 statement here.

2 MR. MORRIS: That's right.

3 MR. STARK: That's the end of my discussion.

4 I'm just happy to hear the same questions that we've

5 been asking ourselves, and I see you're going down the

6 same path.

7 MR. STARK: Bill.

8 MR. MORRIS: Are you through?

9 MR. STARK: Yes, sir.

10 (Slide.)

11 MR. MORRIS: I believe in my discussion I

12 think I will be addressing some of the questions raised

13 in your questions at the beginning of the meeting, Mr.

14 Carbon. I guess I want to make several points here that

15 I think are important to understand; that is, we view

16 the principal and general design criteria -- we know

17 that they are related to events or circumstances which

18 could affect safety functions. And as our understanding

19 of those events changes, our perception of what the

20 important design basis events are changes.

21 We have a tendency to think there may be some

22 changes in the design criteria. And what we intend to

23 do is in the CP review make the selection, determine

24 that these criteria are justified, and evaluate the

25 feasibility for attaining and achieving the general

1 design criteria in the OL review which will come later.

2 In the OL review the objective is to

3 demonstrate that the principal design criteria have

4 indeed been implemented during the ultimate construction

5 of the plant. What we're talking about here is the

6 events and circumstances that could affect safety that

7 are within the design basis, and not events that are

8 without the design basis.

9 So the process of developing general and

10 principal design criteria is closely related to the

11 design basis spectrum events of the plant.

12 MR. CARBON: You say they relate only to

13 design basis events?

14 MR. MORRIS: I think that's a statement about

15 the general design criteria, and therefore, it's been a

16 guiding principle in the development of our principal

17 design criteria for Clinch River. If you look at the

18 general design criteria for light-water reactor plants,

19 you do not see there measures for taking account of

20 core-melt accidents or Class 9 accidents; and that is an

21 important way to understand why you see what you see in

22 the Clinch River PDC.

23 MR. CARBON: Well, then I raise the question

24 what good are they, because certainly you are going to

25 require some sorts of actions on things beyond the

1 design basis accident.

2 MR. MORRIS: There are indeed two parts of our

3 review. There's the part of our review that looks at

4 the principal design criteria and a part that looks at

5 events that are beyond design basis to see whether the

6 design can accommodate them.

7 In one case you design into the plant measures

8 to accommodate credible and postulated events. In the

9 other case you check the design to see whether it could

10 accommodate events more severe than the design basis.

11 And those are the two thrusts of the review.

12 And so if one were to wish to argue that the

13 general design criteria should contain some measures

14 related to Class 9 events, then one could make the

15 parallel argument about PDC for Clinch River. But it

16 doesn't seem to be something that we're compelled to do.

17 MR. KASTENBERG: Have you looked to see

18 whether or not you have to include events beyond the

19 design basis in order to satisfy your safety philosophy;

20 that is, in order to achieve a level of comparability

21 which you want did you look to see whether those events

22 beyond the design basis need to be included?

23 MR. MORRIS: We are considering events beyond

24 the design basis because we believe it's necessary to do

25 so to achieve the overall general safety goal.

1 MR. MARK: Do you have a cutoff point on the
2 kinds of things you feel necessary to consider? For
3 instance, I will think of the following scenario. I fly
4 over the plant and drop a one kiloton bomb through the
5 roof of the containment. Are you going to say you've
6 got to mitigate that?
7 MR. MORRIS: No.
8 MR. MARK: So you do have a cutoff somewhere.
9 MR. MORRIS: Yes.
10 MR. MARK: I think it will be necessary
11 sometime to say where that cutoff is, either at a level
12 of probability or something or other.
13 MR. MORRIS: I think what we would try to do
14 is say it would be comparable to the kind of cutoff we
15 would apply to a light-water reactor. If the Commission
16 began to consider that and took seriously an event such
17 as that for LWRs, I think we would feel a need to look
18 into the potential for such events at Clinch River.
19 MR. ZUDANS: A light-water reactor can -- you
20 are to show that the containment survives.
21 MR. MORRIS: That's right. The criterion is
22 the containment will survive what it sees, what it may
23 perceive as a threat, for a sufficiently long time to
24 have an acceptable impact from the loss of containment.
25 That doesn't mean that the pressure the containment can

1 take is the same for Clinch River as it is for a
2 light-water reactor because the threat is different.
3 MR. ZUDANS: Yes. Thank you.
4 MR. CARBON: I think the Commissioners take
5 degraded core accidents seriously because they worry
6 about hydrogen. Are you doing a comparable thing for
7 CRBR?
8 MR. MORRIS: I don't know -- let's say we are
9 looking at the products of core-melt accidents in
10 containment, and that would include hydrogen generated
11 through the reaction of sodium with concrete in the
12 reactor cavity, and aerosols generated from this
13 reaction, and sodium fires. So there is an analogy.
14 There is a combination of different species in
15 containment that is different from an LWR, but there is
16 a parallel and an analogy, and we are indeed looking at
17 that, and that is part of the review.
18 MR. CARBON: What CRBR accident do you
19 consider to be equivalent to the double-ended pipe break
20 design basis accident for an LWR?
21 MR. MORRIS: I don't believe there is a close
22 equivalent to that. We are looking at the potential for
23 pipe breaks in Clinch River, but the don't necessarily
24 have the same impact or the same consequences or look
25 like the pipe breaks that could occur in a light-water

1 reactor.

2 MR. CARBON: Quite so. I'm not trying to

3 relate a pipe break in one to a pipe break in the

4 other. I'm trying to say what is the accident in one

5 comparable to the other.

6 MR. MORRIS: I don't want to -- well, in terms

7 of an accident I think there are two things you have to

8 be concerned about in a major LOCA in a light-water

9 reactor; that is, what happens to the core and what

10 happens to containment. The containment must withstand

11 the pressures that are produced in that event, and the

12 core must remain coolable during that event.

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1 I don't know offhand a single event that
2 threatens both the core and the containment of CRBR in
3 the way that a major LOCA threatens the containment and
4 core of a light water reactor. There is probably an
5 analogy up to a point, but at some point, the processes
6 are different.
7 And I think, if you look at containment I
8 think a major sodium fire such as those described in the
9 PSAR in Section 50.6 would be the design basis event for
10 judging whether containment within the design basis is
11 adequate, and perhaps some other event might be the
12 limiting event for the core, and I don't offhand know
13 what that is. In fact, that is part of our review, to
14 determine what those events really are.
15 MR. CARBON: I think at least some people
16 within the technical community would maintain that an
17 energetic event -- core meltdown with an energetic
18 release coming from recriticality or some such thing
19 could maybe happen and certainly, that would be the
20 equivalent or more to a double-ended pipe break.
21 MR. MORRIS: The distinction is that for the
22 CRBR, an event for light water reactors involves that
23 large amount of core melting. That is a Class 9 event,
24 and the major LOCA is a design basis event for Clinch
25 River -- I mean for the light water reactor. So I

1 cannot compare directly a LOCA in a light water reactor
2 to a CDA in an LMFBR because one is within the design
3 basis and one is outside it. The CDA in an LMFBR is
4 essentially a Class 9 accident.
5 MR. ZUDANS: But a pipe break with a sodium
6 subsequent fire would be equivalent to a major LOCA in
7 principle.
8 MR. MORRIS: Well, it is an analog to it, yes.
9 MR. ZUDANS: Yes, so you could say the
10 double-ended pipe break sodium spilled in major quantity
11 and the fire following it -- it is the biggest challenge
12 before core melt. Is that designed for? Is the plant
13 designed for that?
14 MR. MORRIS: That is what we are trying to
15 judge. I think the applicant would say it is.
16 MR. ZUDANS: Would that be in your perception
17 equivalent to a double-ended pipe break in LWR's?
18 MR. MORRIS: I do not know what you mean by
19 equivalent.
20 MR. ZUDANS: Well, it is the highest event you
21 want to live with.
22 MR. MORRIS: The largest challenge to
23 containment?
24 MR. ZUDANS: Right, before the core melts.
25 MR. MORRIS: Offhand, I don't know what the

1 event that most challenges containment is. It may be
2 fire of another sort than the one that comes from a
3 pipebreak and maybe the applicant answer it. I do not
4 offhand know what the peak pressures in containment are
5 for various events. They are contained in the PSAR in
6 Chapter 15. And as a matter of fact, among the handouts
7 that may be available to you is a handout that is merely
8 a synopsis of the events in Chapter 15 of the PSAR. I
9 don't know whether you have that. It was provided for
10 your use in getting a quick look at what kinds of events
11 have been proposed in Chapter 15.
12 And let me say that it is not a judgment from
13 us. It is merely a tool for looking at the events, and
14 it just is a statement of what is in the PSAR. It does
15 not constitute any final view from us on that subject.
16 I looked through there, and if you will look at the
17 Section 15.6 in general, you will see a series of events
18 that would be sodium fires. However, there may be
19 another set of events that would have the most severe
20 impact on the core.
21 MR. CHECK: This discussion is interesting in
22 that it suggests the difficulty that there is in trying
23 to postulate for an LMFBR something like the
24 double-ended rupture of the largest pipe in a water
25 reactor where you do get, without a lot of argument, a

1 lot of people agreeing that this represents simultaneous
2 challenge to containment in core.
3 There just is not anything like that, as we
4 know it, within this design basis envelope that can be
5 postulated quite as readily as a single pipe break can
6 for a water reactor. And it may, in fact, be part of
7 the history which impelled earlier regulators and
8 researchers to push beyond the design basis envelope to
9 something bigger, something that would represent an
10 all-encompassing enveloping thread.
11 MR. CARBON: It seems to me that someone here
12 is simply adjusting the border or boundary of the
13 criteria, or adjusting whatever they want to term as the
14 design basis event. It may not be at all comparable in
15 seriousness in LMFBR's with the seriousness in an LWR.
16 It is not clear to me that the worst DBA in CRBR is
17 comparable in seriousness to the worst DBA in an LWR.
18 MR. CHECK: That may, in fact, be the case and
19 that knowledge, or at least that plateau of
20 understanding, may have impelled certain regulators to
21 look for a greater threat to design against. And having
22 done it, one could then relax in much the way the people
23 had five years ago with respect to the major accident
24 for the water reactor.
25 You look at the bounding event, you assure

1 yourself you have done everything for the bounding
2 event, and then you relax regarding those things which
3 are, at least initially, less severe. But we have tried
4 -- and I am sure others have as well -- to postulate
5 single events and explore consequences, and CDA's just
6 do not happen quite that easily as postulating a single,
7 double-ended rupture of a pipe.
8 MR. CARBON: It is possible there could be
9 considerable argument on that, but I guess do not really
10 know.
11 MR. MORRIS: Let me make sure that I clarified
12 the situation with regard to pipe breaks. Part of what
13 I have said is based upon our judgment made earlier in
14 the review -- that is, some years ago -- that it would
15 be possible to eliminate breaks in the cold leg as
16 design basis events by imposing leak detection measures
17 and pre-service inspection. And that those breaks are
18 not now considered as part of the design basis spectrum
19 because of those measures that we believe are
20 practicable.
21 And that if an instantaneous double-ended
22 guillotine break were postulated in the cold leg, it may
23 have implications for the core. I think it probably
24 could. But the breaks that are now part of our
25 consideration are hot leg breaks primarily. Those do

1 not seem on the surface to have that potential impact.
2 We are going to go through a review to evaluate what the
3 impact of those breaks will be, and I will show you
4 something about the schedule for that work in a few
5 minutes.
6 (Slide)
7 There has been some question I think about how
8 we go, within the NRC, about institutionalizing these
9 criteria, and I just wanted to remind you of some of the
10 factors that go into this process. As I said before,
11 the applicant has proposed principal design criteria.
12 They are in the PSAR in Chapter 3 and in each case he
13 has indicated how he thinks he has implemented and met
14 those criteria. And those are related to the proposed
15 design basis events.
16 The review of the various criteria will be
17 done within NRR in a manner similar to the reviews for
18 light water reactors. Various branches play lead and
19 secondary roles in evaluating these events and the
20 criteria that are related to them. In this, we have
21 pointed out one must take into account the 10 CFR
22 general design criteria and the ANS criteria in
23 evaluating the acceptability of these criteria.
24 We have consultants in research that have a
25 bearing on this review. The evaluation would be subject

1 to the input from the subcommittee which is what we are
2 doing today, and if we take an opinion that is different
3 from the applicant's proposal, we will provide him with
4 that evaluation in the review process and iterate this
5 until we get an acceptable set of design basis events
6 and close the principal design criteria.
7 At which time, assuming we have had this
8 ongoing dialogue with the subcommittee and we feel that
9 we understand your point of view here, we would then
10 propose to publish this in the SER rather than in some
11 preliminary document, as has been suggested here, and
12 that would then go to ACRS for their review.
13 We hope that this communication line here will
14 be the vehicle by which we will avoid having a
15 preliminary publication of the criteria that is separate
16 from the SER. The SER will contain the basis and
17 justification for the criteria.
18 MR. ZUDANS: The SER also will contain a
19 statement saying that Appendix A was satisfied?
20 MR. MORRIS: Appendix A in general, not the
21 specific criteria listed in Appendix A.
22 MR. CHECK: It will have been addressed. It
23 will have been acknowledged as a starting point.
24 MR. STARK: As guidance.
25 MR. CHECK: Let me try again. In Section

1 50.34, we are required by one of the four items in there
2 to make this finding on principal design criteria.
3 Then, sometime after that, as the result of a lot of
4 experience and design and a modest amount of operation,
5 general design criteria for light water reactors, we
6 will promulgate it actually in stages. As I am sure
7 some of you are aware, when I started in this process
8 there were 17; that was in the mid-sixties. And they
9 went up to 75 and they came down to whatever they are
10 today, but it happened in stages.
11 Now those, I take it, are, by the introduction
12 in Appendix A, a good starting point. They are most
13 suited for boiling water and pressurized water reactors,
14 but there is a general applicability to others as well.
15 MR. CARBON: Do you suppose if someone were
16 starting out today to draft general design criteria,
17 that as a consequence of TMI they would come up with
18 different ones today than the ones that are in Part 50?
19 And then I guess if you answer that yes, I would follow
20 up and say well, we are coming up with some new ones for
21 CRBR. Should TMI-2 have some influence on what we come
22 up with?
23 MR. CHECK: I will put off the answer to the
24 first one for the moment, but the second one is most
25 assuredly yes. The applicable lessons from TMI-2 I hope

1 are going to be reflected in our review.

2 MR. CARBON: Would you think if we were

3 starting from scratch with general design criteria that

4 you might go beyond the design basis event?

5 MR. CHECK: Well, I don't know the difficulty,

6 but there is no Moses for general design criteria, there

7 is no one person who leads us and speaks with

8 unquestioned authority about how things ought to go.

9 But I have had discussions with Guy Arlotto who has been

10 associated with these for a long, long time, and the

11 view he espouses is close to this thing I have said a

12 couple of times, that GDC are distillations. They are a

13 result of something, they do not precede. One cannot do

14 good standards and criteria writing -- general guidance

15 writing -- without some experience base.

16 MR. CARBON: That is certainly so, but once

17 you get the experience then you can come up with

18 something that serves as a guideline for future efforts.

19 MR. CHECK: I do not have experience here. I

20 have some limited experience with LMFBR's and will

21 factor into our work, but will try not to make a

22 pronouncement about how all LMFBR's ought to be designed.

23 MR. CARBON: I am not talking about that. I

24 am raising the question, would the experience with TMI

25 change the general design criteria for LWR's if you were

1 starting from scratch to write them.

2 MR. CHECK: Well, something is happening. I

3 happen to know -- you probably know too -- they

4 contemplate putting a general design criterion in about

5 human factors. But beyond that, I don't know, I don't

6 know of anything on a specific system-oriented or

7 structural related -- I don't know of anything like

8 that. But human factors are going to be, if standards

9 writers have their way, are going to find their way into

10 GDC over the next year.

11 MR. CARBON: Well, I would just comment that I

12 find the whole design criteria thing sort of

13 unsatisfactory or something. It does not really address

14 the safety issues; it addresses some artificial things

15 and stops there, and I find that very disappointing.

16 MR. CHECK: Maybe we are not making our

17 point. I think one of the things Bill tried to show was

18 that these things can be correlated with events.

19 MR. CARBON: You are correlating the criteria

20 for CRBR with the ones for LWR's, and I recognize that.

21 Maybe that is the thing you should do. I am not sure,

22 but I do find it disappointing in that each one really

23 covers the safety issue.

24 MR. CHECK: It is an interesting thing to

25 contemplate and I can only suggest that the committee

1 invite some of these big thinkers down here. I think an
2 afternoon with Arlotto would be useful. And maybe these
3 issues ought to be revisited on a larger scale.
4 MR. CARBON: Rather than Arlotto, have they
5 been raised with Denton?
6 MR. CHECK: I do not know.
7 MR. ZUDANS: I have one point for Paul. I
8 read in Paul's report that you have something called
9 standard format and content for a PSAR or FSAR for
10 LMFBR's which defines all these events. Isn't this the
11 first step in developing criteria to review that
12 particular document?
13 MR. CHECK: I am going to defer to somebody
14 who knows specifically the status of that document in a
15 moment. But let me say that one gets into the chicken
16 and egg business again. General design criteria,
17 standard review plan, standard format -- these things --
18 it is a circle.
19 MR. ZUDANS: I agree with you, it is a circle
20 and the criteria will affect the standard format and the
21 standard format is the basis of the events as you
22 described will affect criteria and design will affect
23 everything else. So why should the design criteria go
24 to such detail? I don't understand it.
25 MR. CHECK: Well again, we are trying to keep

1 this to some reasonable minimum.

2 MR. STARK: I think it reflects the evolution

3 of the licensing process and the light water process,

4 and we are looking at it in trying to apply 20 or 30

5 years of what the light water people went through in

6 this particular application. And we are asking the

7 question why did we do it that, and I understand your

8 comment.

9 MR. CARBON: Did you ask me a question? I am

10 not sure I caught all you said.

11 MR. STARK: No. We had the same discussion

12 ourselves because the GDC plus the standard review plan

13 plus the standard format and content and the NUREG's and

14 the Reg Guides -- they constitute the acceptance

15 criteria for this plant or any other plant, and that is

16 all in the SER. This is just a piece of it, as I see it.

17 MR. CARBON: Yes, that is certainly so. And I

18 guess what I was saying a moment ago in somewhat

19 different words is it is such a small piece or such an

20 arbitrarily-defined piece that one sometimes wonders why

21 make all the fuss about it.

22 MR. ZUDANS: Now, you are speaking my

23 language. In the general analysis, all those things

24 cover the safety issue precisely. This has created an

25 aritficial reference for future levels. It could not be

1 developed without design. I understand that point. It
2 is a circle and it is a very questionable thing whether
3 you ought to do what you have been doing.
4 MR. STARK: It does function as a good
5 checklist for us, though, because in referring to what
6 hurdles the light water people have to pass over, I
7 think we make sure that this applicant or this
8 application looks at the same bases or the same thinking
9 anyway, so it is still a key ingredient.
10 MR. CARBON: Yes, it is worthwhile. It
11 establishes some standards, some minimum requirements,
12 and many of those are very good. I do not question its
13 value.
14 (Slide)
15 MR. MORRIS: Just to give you an indication of
16 the degree of similarity between the CRBR principal
17 design criteria and the GDC criteria, I have indicated
18 in this table that 38 of the principal design criteria
19 are identical to the 10 CFR criteria. Ten are similar
20 with only a slight variation. That gives a total of
21 approximately 86% comparability between the light water
22 reactor criteria and the LMFBR criteria.
23 And I do not think, as Dr. Carbon suggested, I
24 don't think it is a trivial set of criteria because here
25 included are criteria to contend with seismic events,

1 floods, fires, and which address quality assurance, and
2 all of these features and all of these measures go into
3 assuring that the severe accidents or Class 9 accidents
4 are unlikely to occur.
5 There are eight of the principal design
6 criteria with no 10 CFR Appendix A counterpart, and
7 there are 9 10 CFR 50 criteria for which no comparable
8 CRBR principal design criteria exist. There is a
9 listing or tabulation of these various criteria in
10 various categories that you have been provided, and it
11 is there for your use in reviewing these similarities
12 and to help you see what the differences are.
13 (Slide)
14 What I am going to do is concentrate on some
15 of those differences now, and I am referring now to
16 first, Section 1 of this document that you have been
17 provided in which you have a listing of these criteria
18 in 10 CFR 50, for which no comparable principal design
19 criteria have been adopted yet. And I thought I would
20 spend a few minutes just discussing some of what I think
21 are the implications of these criteria being absent from
22 the PDC's in Clinch River.
23 First off, the one on reactivity limits -- if
24 you read criterion 28, and it is included here in
25 Section 1, if you read that you will see that it

1 specifically mentions rod drop and rod ejection
2 accidents. And rod drop accidents are associated with
3 BWR's and rod ejection accidents with PWR's. And it
4 relates to large rapid reactivity insertion accidents,
5 and addresses those events.
6 Now, that criterion does not appear in the
7 Clinch River criteria. We believe at this preliminary
8 stage of our review that it is because the applicant has
9 not identified an event that would insert reactivity
10 into the core in a way similar to light water reactor
11 rod drop or rod ejection accidents. And it is related
12 there again to the design of the control rods.
13 But we have under review the functional design
14 of those control rods to determine whether or not we
15 agree with that thinking and to see whether we believe
16 that this is an acceptable omissions. We also have
17 other activities that will allow us to determine the
18 potential for large reactivity insertions. Our Chapter
19 15 review and something we call an accident delineation
20 review, that attempts to identify whether there are
21 events currently proposed by the applicant in Chapter 15
22 that could lead to some kind of severe reactivity
23 insertion. So we are attempting to address this issue
24 in that way.
25 It does not mean that we just are essentially

1 -- have not determined what our viewpoint there is. But
2 what I am giving is my impression of the situation at
3 this time. The applicant may wish to say something
4 about this.
5 MR. ZUDANS: In Chapter 15, where you referred
6 to where these events are described, which essentially
7 described this, the control rod drop --
8 MR. MORRIS: A rod drop accident in a BWR
9 drops the rod out of the core. A rod drop accident in
10 this event drops the rod into the core and is another
11 kind of an event altogether.
12 MR. ZUDANS: Okay.
13 MR. CARBON: And what is it that happens in a
14 PWR?
15 MR. MORRIS: A PWR is the rod ejection event,
16 and that is again -- we don't know yet whether or not
17 some kind of mechanism could exist that could cause that
18 rod to be ejected.
19 MR. CARBON: I guess you are saying that you
20 do know a mechanism exists to eject one from a PWR.
21 MR. MORRIS: Yes, because it is here as
22 obvious evidence.
23 MR. CARBON: I have heard all the chicken and
24 egg sorts of arguments for the last hour, but those
25 criteria to me are still standards. And I guess I am

1 back with the chicken.

2 MR. MORRIS: I think we will have to get the

3 people who wrote the criteria to come down here and

4 discuss the basis for this in more detail than I can do.

5 MR. CARBON: Paul, do you --

6 MR. CHECK: Ask the question again.

7 MR. CARBON: Do you view the criteria as

8 standards to judge safety?

9 MR. CHECK: Well, the best I can do for you is

10 to read part of the introduction of what these things

11 are. The statement of minimum requirements for

12 structures, systems, components important to safety,

13 establish necessary design fabrication, construction

14 testing, performance requirements.

15 MR. CARBON: Well, if there are minimum

16 requirements I guess I would say that is the same as a

17 standard or something.

18 MR. CHECK: To put it another way, I do not

19 mean to be coy -- if somebody does not meet it, they are

20 in trouble.

21 MR. CARBON: If I look at it as a standard, if

22 you have got a standard there should not be a reactivity

23 insertion in an LWR. Why do you not have one in an

24 LMFBR?

25 MR. CHECK: I think we could have one; we just

1 do not see the need for it right now. It may work out
2 that way, we could probably put one on.
3 MR. ZUDANS: Well, it doesn't hurt.
4 MR. CHECK: Well, I take exception to adding
5 things that are not necessary.
6 MR. ZUDANS: You are removing things in this
7 case. It is already there. It is unfortuntely design
8 specific. The criteria are so general that they are
9 beautiful, like quality assurance, protection against
10 natural phenomenon, all of those things have general
11 criteria. They are really a general description of an
12 environment. That is what I would call really general
13 design criteria.
14 MR. CHECK: Which is not what we are doing.
15 MR. ZUDANS: Many things are very specific and
16 design-dependent.
17 MR. CHECK: I don't think we are leaving --
18 MR. ZUDANS: You have to review it, it is
19 already done. I think the same defect is with the
20 general design criteria. They are also not sacred in
21 that sense.
22 MR. LIPINSKI: They are looking at a design
23 that has certain features that are inherently safe on
24 reactivity, but if the designer were free to design --
25 and let us assume that he elects to put pneumatic drives

1 in all the control rods and he uses closed-loop position
2 control rather than screws, and I have an opportunity
3 for a common mode failure of all of my drives, to go
4 full velocity out, or if I have a single drive worth
5 more than a dollar then I could drive outwardly because
6 I have this peculiar mechanism, or if I design them all
7 to work the same way I can drive all the rods out.
8 But a common mode failure on an LMFBR, to have
9 a specification that limits the reactivity as well as
10 the rate of reactivity addition, I think it is real
11 important that the fact that these designers recognized
12 it and included it in their design does not mean there
13 should not be a GDC that covers that issue.
14 MR. CHECK: I agree with you.
15 MR. LIPINSKI: These designers recognize the
16 issue and LMFBR particularly is sensitive to reactivity
17 and it is inherent in this design. But the fact that
18 there is not a criterion to cover it is interesting.
19 MR. CHECK: I am not prepared to fight any
20 battle here. I do not think that that is that
21 significant an issue. Between us, we have the same
22 objective. We are looking at this right now and I
23 understand there are questions to the applicant on this
24 point. There is yet work to be done. I don't have any
25 major disagreement with what you are saying.

1 MR. CARBON: I don't think there really is a
2 technical problem here. We all recognize the importance
3 of it technically. I think the question is, what do
4 these criteria mean. And going back I guess to me,
5 probably Mike expressed it best way back there in
6 pointing out or making the statement that we have to be
7 sure that these are viewed as standards by which CRBR is
8 judged, rather than something that -- I think his words
9 were something along the lines of prepared to help
10 justify what we are doing.
11 It is a credibility problem in part. It
12 really is not a technical problem, I don't think. I
13 know the design has certain limits to it and so on.
14 MR. ZUDANS: It is interesting what Walt just
15 said. The designers recognize the need for this
16 protection, but that does not mean that should be a
17 reason to remove that criteria. Clearly, such a problem
18 would be postulated by improper design.
19 MR. BENDER: Well, let me offer an analogous
20 -- not specifically analogous but comparable kind of
21 circumstance in the LWR criteria. I don't think there
22 is anything that says pressure vessels cannot fail in
23 the LWR criteria. It is implicit, and there will be a
24 lot of implicit criteria here because you really don't
25 want criteria to state the obvious all the time.

1 I think I am concerned personally about being
2 sure we understand the logic. That is mostly why I am
3 pressing the matter. I would like to see something
4 written down. The criteria are kind of bald right now.
5 They just say, here are the criteria. But why they are
6 the criteria leaves a lot to the imagination, and while
7 I am very comfortable with what I understand about
8 LWR's, I do not think I have any reason to believe that
9 anybody here should have less discomfort than me with
10 the question of whether I understand why LMFBR's have
11 certain criteria.
12 I have looked at the criteria a long time, and
13 I do not know about a lot of them. Walt knows about
14 some and everybody in this room knows about some, but
15 none of us knows about the same ones.
16 MR. CHECK: I do not know why it occurred to
17 me only yesterday, but I started looking for a similar
18 document for these, for the Appendix A. There is not
19 one.
20 MR. BENDER: I am sure you are right, and
21 there is not one for Fort St. Vrain, and so the
22 precedent that we are searching for does not exist.
23 But there has been a change in the regulatory
24 process over the years. When water-cooled systems were
25 engineered there was no regulatory system. It was

1 created afterwards. When Fort S. Vrain was built, the
2 regulatory system was in transition. It was treated as
3 a special case, and this could be treated as a special
4 case if the law did not say that the CRBR must be put
5 through the licensing process. That is the thing that
6 I keep asking myself about and it is the thing you guys
7 need to think about in terms of what the regulatory
8 challenges will be when you want to stand up before the
9 hearing boards or the courts and say, we are ready to
10 license this because --. I don't put it forth as a
11 technical argument at all; I just say it is something
12 you need to be sure you do.
13 MR. CHECK: We do think about that a lot. We
14 are plotting strategy right now. Cecile Thomas is not
15 here because he is home working on that. We have
16 announced -- and I will say it again -- that unlike
17 perhaps any other case, we plan to postulate and defend
18 the same criteria for this plant.
19 Now, the timeliness of that defense is perhaps
20 troublesome to some, but it will be in our SER if not
21 sooner.
22 MR. MORRIS: Again, let me point out that we
23 do not consider the issue in criterion 28 closed. We
24 have a review in progress to determine whether we think
25 it should be added, and its potential omission is what

1 we are talking about now. We do not defend it; I was
2 just given an explanation of why it may not be there.
3 In that same vein, let me call to your
4 attention that there are other criteria requiring, in
5 the PDC for Clinch River, that there be protection
6 against anticipated operation occurrences and
7 accidents. And then if you want very general criteria,
8 that kind of criteria would suffice I believe for the
9 kinds of things Mr. Lipinski was talking about.
10 This particular one may be an artifact of a
11 particular licensing problem that arose with PWR's and
12 BWR's and for which this is there specifically to
13 address that issue rather than being a general
14 criterion. And it certainly does not appear to be
15 general when they specific identify certain events.
16 MR. CHECK: Wasn't it somebody's theory that
17 these were, in fact, dated? These are rather latecomers
18 to the GDC.
19 MR. MORRIS: A few years ago there was a lot
20 of debate about rod drop accident. A rod drop accident
21 for a BWR was considered new a few years ago and there
22 was a feature put on for BWR's that limits the kind of
23 reactivity insertions that you can get. In other words,
24 you limit the order in which you withdraw rods to meet
25 this criterion. And there was an issue at the time with

1 regard to whether the design of that system that limits
2 the rod withdrawal sequence had to be a safety system
3 and meet IEEE criteria or what kind of system it should
4 be. And I would suggest that this may be an artifact of
5 that whole process rather than something that came in
6 the way of a very generalized overall goal for reactor
7 design.

8 MR. BENDER: If you look at the history of
9 reactor accidents, you may get some insight into why
10 certain criteria exist. Some of them you could probaby
11 learn about by going down and talking to Mr. Hanauer who
12 is one of the main forces in getting some of them in
13 place. But I guess I would accept the idea that we do
14 not necessarily have to have a general set of criteria
15 for everything, for things that are obvious. But I
16 still think we ought to be able to say why the ones that
17 are here are here.

18 MR. MORRIS: Maybe an even more difficult one
19 is criterion 29. When I look at that I know that there
20 seem to be other criteria which address this same
21 issue. Principal design criteria 18, which is
22 equivalent to general design criterion 20; principal
23 design criterion 20 which is equivalent to GDC 21,
24 appear to impose the same types of requirements on the
25 design as this one. And one might think that if this

1 were adopted for Clinch River, that given we can
2 conclude that Clinch River meets those other criteria,
3 that this would impose no added requirements.
4 I just invite you to make those comparisons to
5 see what you think, but again, we are still considering
6 what should be the final set of criteria for Clinch
7 River, and this is under consideration.
8 Reactor coolant makeup is, to some extent,
9 equivalent to principal design criterion 27, which is
10 assurance of adequate reactor coolant inventory. So
11 even though they look somewhat different, there seems to
12 be a parallel or an analog criterion in the Clinch River
13 set.
14 If you look at the three criteria for
15 emergency core cooling, 35, 36 and 37, there again you
16 have a very specific event for which criteria have been
17 developed, even including criteria for inspection and
18 testing of the systems that achieve emergency core
19 cooling. So again, I would suggest that this is an ad
20 hoc kind of special set of criteria that were imposed on
21 the light water reactor designs.
22 Now again, we are re-evaluating or continuing
23 to evaluate the potential for pipe breaks in the Clinch
24 River design, and to determine whether we feel there
25 should be something like this added to the Clinch River

1 design criteria. But even though we may think that
2 there would be credible pipe breaks, it does not
3 necessarily mean that it would lead you to an ECCS
4 system such as suggested by these three criteria. But
5 perhaps the one that now exists, 27, is sufficient to
6 cover those events.
7 (Slide.)
8 Oh, on containment, in PSAR Section 15.6,
9 again, there are a set of events that include fires in
10 containment which could produce pressures and
11 temperatures in containment that might cause you to need
12 a cooling system. If you look at those events as
13 proposed by the applicant and evaluated by him, they do
14 not appear to produce a need for cooling systems.
15 We will re-evaluate those Chapter 15 events in
16 our review to determine whether we can agree with that.
17 If one can agree that it is unlikely, that you do not
18 need to cool containment to handle those kinds of design
19 basis events, then one may argue that you do not need
20 the specific criteria for testing and inspecting the
21 containment heat removal system.
22 Now I want to be clear here. There is a
23 containment heat removal system proposed for Clinch
24 River. It is there to cool containment during a core
25 disruptive accident followed by a core melt, and as I

1 understand it, under the applicant's proposal, would
2 only have to be used under those circumstances.
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1 MR. BENDER: Let me understand the containment
2 question a little bit more. We have something that says
3 there shall be containment of the following
4 capabilities. Do we have something that says there
5 shall be a containment which will be able to do certain
6 things?
7 MR. MORRIS: I'm not sure I know where that's
8 written.
9 MR. CHECK: It's in the Denise-to-Caffey, the
10 May 6th letter.
11 MR. MORRIS: I think you said if we specified
12 anything on containment that says --
13 MR. BENDER: And what kind of functions it
14 ought to have. That seems like a sort of fundamental
15 question, and it's not like the pressure vessel.
16 MR. MORRIS: I think some of these criteria
17 are in the principal design criteria now. A containment
18 isolation function must be there, and it's there; that
19 is, you should be able to isolate containment, and there
20 are certain accident conditions, and you should be able
21 to clean up the environment within the containment.
22 MR. BENDER: If I were trying to write
23 containment for this system -- go ahead, Bill.
24 MR. KASTENBERG: Criterion 41 is the
25 containment design basis. It's on page 10 of this

1 handout.

2 MR. LIPINSKI: Fourteen is containment design

3 on page 3, and the other one is design basis.

4 MR. KASTENBERG: Fourteen and 41, page 10,

5 criterion 41.

6 MR. ZUDANS: Fourteen is very general.

7 MR. BENDER: I'm not going to try to quibble

8 with this. I think that would be the wrong thing to

9 do. It's probably got everything in it, but if I were

10 trying to judge what is meant by exothermal chemical

11 reactions and release of fission products and potential

12 aerosols, and I were looking at how I would do this for

13 a water-cooled system, I would be looking at the LOCA

14 criteria, and so I have a basis for deciding whether the

15 containment is okay.

16 Now, what I'm trying to ask myself now is

17 given this particular set of information, where do I go

18 to tie these criteria to some kind of physical process

19 that might be going on that would be used to analyze the

20 adequacy of the containment. I know it's been done.

21 MR. MORRIS: That's in Chapter 15, your

22 proposed design basis events. In Chapter 6 you evaluate

23 the capability of the containment system to handle -- or

24 you identify which specific criteria the containment

25 must meet to achieve mitigation of those design basis

1 events. It's in the PSAR. That's where it's located.

2 MR. BENDER: Are they analytical processes in

3 the PSAR?

4 MR. MORRIS: Yes.

5 MR. BENDER: In a way which we know -- I don't

6 know whether to accept them or not. I'm trying to

7 figure out how I'm going to look at this and deal with

8 it.

9 MR. MORRIS: I would invite the Applicant to

10 say whether or not they do initiate cooling during any

11 design basis event and what the answer is.

12 MR. GOESER: Dick Goeser of the Applicant.

13 There is an initiation of a type of light-water reactor

14 during any of the design basis events. That's the

15 answer to one question.

16 The answer to the other question I think, Dr.

17 Bender, in Chapter 15 the results of the analyses for

18 the various events are indeed presented, and there is

19 reference made -- there is a description of the event,

20 and there are references made to codes that are being

21 used to analyze them.

22 Within the description of the event we treat

23 those things that are treated within the analysis and

24 try and describe the method of treatment, and the

25 details of the analysis are done by code.

1 MR. BENDER: So you think it's all wrapped up
2 in Chapter 6 and Chapter 15 in this criterion 41, and I
3 can find the things I need to look at to see whether
4 criterion 41 is met?
5 MR. GOESER: Yes. In addition there is Section
6 15-A of the PSAR which covers the containment analysis
7 with respect to the site suitability source term, which
8 also goes into what the containment has to be capable of.
9 MR. BENDER: I'm trying to develop a road map
10 for myself, and I'm just using this as an illustration.
11 I don't really know that I've even zeroed in on the
12 right set of questions. But if I were going to use the
13 criteria, I kind of think I've got to say well, here's
14 the criteria and here are the places where I go to see
15 whether they're satisfied.
16 MR. GOESER: The first thing in Chapter 3 of
17 the PSAR was a listing of these criteria with a
18 narrative description of how they are met with
19 references to other sections, which might be the first
20 place to go within the PSAR framework to get a better
21 road map written down by the Applicant in terms of how
22 we believe these criteria are being met.
23 MR. CARBON: I don't understand something that
24 both of you said back there. You referred to Chapter 15
25 and also Chapter 6, but both references were in terms of

1 design basis events.

2 Now, for design basis events you don't even

3 need containment.

4 MR. GOESER: Yes. Containment is needed; for

5 example, for the sodium fire that had been postulated.

6 But beyond that, the response to I guess an unstated

7 question is there is an additional part of the submittal

8 of the Applicant that covers the response of the primary

9 system in the containment to the various CDAs that have

10 been analyzed, and that's in another part of the

11 document --

12 MR. CARBON: In which part?

13 MR. GOESER: In CRBR-3, Volumes I and II, which

14 are reference documents to the PSAR.

15 MR. CARBON: So by reference those things are

16 incorporated in either Chapter 6 or Chapter 15, which

17 then leads on to the actual containment requirements, is

18 that correct, excluding the sodium fire, which I agree

19 with.

20 MR. GOESER: The requirements that have been

21 imposed in order to provide the margin within the plant

22 for the CDA combination are actually specified in

23 CRBR-3, Volumes I and II, depending on whether it's the

24 primary system or the containment. The design of the

25 containment itself is presented in Chapter 6.

1 MR. BENDER: Let me offer a thought without
2 setting it out here's something I think needs to be
3 done. We've become accustomed to the idea of having a
4 standard review plan for water-cooled plants, and it
5 didn't come about very easily, but we're used to it.
6 And there's a comparable circumstance here where we have
7 to say well, given that you have to review something,
8 there's a review process that's gone through, and I
9 suspect we need some kind of mini-plan, which I wouldn't
10 want you to lay on the table today, but probably
11 somewhere we have to say well, there was a review
12 process. It's not a helter-skelter kind of thing.
13 There was some order to it. And I suppose you've got in
14 mind to be able to display that plan in some way.
15 MR. CHECK: Well, of course again the ultimate
16 document will be the safety evaluation report that will
17 describe the review. I would observe that there is more
18 that is the same than is different for LMFBRs and LWRs.
19 There's a pretty good template for reviewers to use.
20 I haven't heard any complaints other than from
21 Bill about these exotic accidents, about reviewers not
22 knowing how to do the job, and I think they are
23 following well-trodden paths.
24 MR. ZUDANS: I guess they will get some
25 guidance from the PSAR itself with respect to acceptance

1 criteria, but those would be different from the current
2 standard review plan because of the high temperature
3 implications.
4 MR. CHECK: Yes. You point up another area
5 that's quite different -- mechanical design of the
6 primary system is getting special attention. I'm not
7 qualified to comment much on it, but I know this is not
8 a water reactor, and we need to do things here that go
9 well beyond traditional code guidance.
10 MR. CARBON: This would seem a good time to
11 take a break. Let's do so.
12 (Recess.)
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1 MR. CARBON: Let's go ahead and resume our
2 discussion.
3 (Slide.)
4 MR. MORRIS: Continuing in this handout I gave
5 you, we look now at those criteria that are in the
6 Clinch River principle design set, but which do not have
7 an analogue or do not appear in the light water
8 criteria. And I'm going to be fairly brief on these.
9 In some cases they speak for themselves, and if the
10 applicant wishes to expand on it I would hope he will do
11 so.
12 But in some cases these are indeed fairly
13 general, maybe more general than some of the light water
14 reactor criteria have been.
15 Criterion 4, protection against sodium
16 reactions, I think appears to be a prudent thing to
17 address, and it appears from reading the criterion that
18 they have addressed sodium reactions with water and
19 oxygen and with concrete, and those appear to be major
20 areas that must be taken into consideration in this
21 design.
22 And again, it was pointed out earlier, you
23 could look in Chapter 3 of the PSAR and have a very
24 brief discussion of how this is being implemented for
25 Clinch River. And if you wish to do that, that's where

1 you would find the discussion.

2 MR. MARK: Could I ask, with respect to No. 34

3 --

4 MR. MORRIS: Reactor and intermediate cooling

5 and cover gas purity control.

6 MR. MARK: I think we have, I understand, an

7 interest in trying to control the purity of the

8 coolant. It escapes me entirely what their interest in

9 the purity of the cover gas may be.

10 MR. MORRIS: I didn't understand your

11 question.

12 MR. MARK: Why is its purity control of

13 interest to them?

14 MR. MORRIS: I'll let the applicant answer

15 that question.

16 MR. DICKSON: This is Paul Dickson of

17 Westinghouse.

18 One of the major reasons you want purity is

19 for plugging up any of the passages, and another is to

20 minimize the radioactive isotope inventory in the cover

21 gas, and consideration of chemical attack and detection

22 of sodium-water reactions. All four of those are

23 specified in criterion 34.

24 MR. MARK: I can understand one's wish to

25 sample cover gas to detect things like delayed neutron

1 emitters. You can't keep it non-radioactive because
2 there will be some sodium vapor in there and that will
3 be radioactive. I don't suspect this cover gas of
4 plugging anything.
5 MR. DICKSON: You can't keep it totally
6 non-radioactive, but you mentioned yourself that you
7 must clean up the cover gas in order to be able to
8 detect subsequent failures.
9 MR. MARK: Okay, you circulate and take the
10 radioactivity out to keep it below some level, so that
11 you can see a new source in there?
12 MR. DICKSON: Yes, and I guess that's a point
13 I failed to make. In Clinch River we use a
14 recirculating cover gas. It's not a once-through
15 system.
16 MR. MARK: Thank you.
17 MR. MORRIS: Criterion 7, sodium heating
18 systems, is based on the potential for freezing of
19 sodium in the absence of an adequate heat source.
20 Criterion 26 is somewhat interesting in that you do not
21 find a criterion quite like this in the light water
22 reactor general design criteria, and what I think is the
23 important feature here is the statement that
24 consideration shall be given to provision of
25 independence and diversity to provide adequate

1 protection against common mode failures.

2 It's my understanding that in the development

3 of the criteria that the staf thought that some

4 attention should be given to this potential for common

5 mode failures for a system that we must admit had been

6 proposed to them, and they were evaluating. And they

7 felt that this was necessary.

8 Now, that may be a simplification of how this

9 was developed. That does not include some point of view

10 that the applicant may have had at that time. But

11 what's important, I think, is that there was a

12 recognition that there may be a need for diverse

13 capability in the shutdown system, that is, a separate

14 system different from the primary heat transport system

15 to remove decay heat. And there is no analogue for a

16 light water reactor that I know of.

17 If the staff eventually reviews the need for

18 additional shutdown heat removal capability in light

19 water reactors, you may some day see a proposed general

20 design criteria of this sort proposed for light water

21 reactors. I emphasize again that the mention of

22 diversity and independence is important.

23 MR. BENDER: Could I go back to criterion 7

24 for just a minute? When the staff is reviewing this

25 requirement, it has in it something that says, heating

1 systems and their controls shall be appropriately
2 designed with suitable redundancy to assure that
3 temperature distribution and rate of change of
4 temperature in the sodium system and components are
5 maintained within design limits, assuming a single
6 failure.
7 What kind of review process is that likely to
8 lead to? Does that mean that the electrical and control
9 group has to sit down and figure out, first, what single
10 failures have to be considered, and then whether there
11 is adequate redundancy, and do they have to consider the
12 electrical power sources in such an evaluation?
13 MR. MORRIS: I think the way it would proceed
14 would be that one would determine whether there is an
15 adequate set of specific criteria that support this
16 general criteria proposed. For instance, one might look
17 for a statement regarding the nature of the power source
18 that was used and look for an elaboration of the
19 criteria.
20 But the actual decision as to whether this
21 criterion has been met is something that's typically
22 deferred to the OL review, that is, the review to see
23 that in the design that has been proposed, that no
24 single failure will actually cause a deviation from this
25 criterion. That review is the OL review.

1 What we're doing in the CP review is trying to
2 determine the adequacy of the criteria that have been
3 proposed, including the principle design criteria and
4 those less general criteria that are the acceptance
5 criteria. We don't need to make the conclusion at this
6 time that the criterion has been met.
7 I think what you were talking about was deeply
8 involved in the consideration that the criterion had
9 been met.
10 MR. BENDER: Well, one of the places where a
11 lot of agonizing is going on now, -- because if the
12 criterion isn't met we'll have to wind up saying, what
13 backfits are needed. And there is a question of
14 timeliness in some of these reviews.
15 MR. MORRIS: What we do here is -- in
16 addition, let me add to that that what you do, I think
17 appropriately, in the CP review is you look at the
18 general criterion and then you look at those specific
19 statements about the proposed design that the applicant
20 has put in the PSAR regarding how he intends to use
21 power to produce this heating. And you look to the
22 question of feasibility. Does it seem reasonable that
23 it's feasible that he can meet that criterion in the
24 finalization of the design and the implementation of the
25 design?

1 And I believe if you come to the conclusion in
2 the CP review that he has stated adequately the criteria
3 he is to follow in his construction of the plant, you
4 can judge that it's feasible for him to meet those
5 criteria, then that is essentially to my mind a kind of
6 a contract that's been made with the applicant when this
7 review is complete for the CP. And we expect him to
8 come back and fulfil that contract, and we'll check it
9 in the OL process.
10 There is a possibility that we would not agree
11 with how he implemented it in the OL review, but we are
12 still looking at two-phase review here.
13 MR. BENDER: Well, I'm not sure that I agree
14 that that's the way in which the construction permits
15 are reviewed, but it is a way of doing things.
16 MR. MORRIS: I think it would be possible to
17 take the construction permit review beyond what I stated
18 as the minimum, and I think the basis for doing that is
19 the amount of time you've got and the degree to which
20 you feel you can make judgments about the feasibility of
21 his meeting the ultimate criteria individually. And I
22 think you will see that different reviewers will take
23 that farther than others.
24 MR. BENDER: I think it has to be related to
25 the status of the design, as well as the design intent.

1 If there has been no design done, then I certainly would
2 concede to the fact that all you can do is look at the
3 design intent, and that's often done in the CP stage.
4 In this particular case, where the design may
5 have been fully developed already, you're tacitly
6 accepting that design if you don't choose to comment on
7 it at that stage, and that's one of the things that's
8 bothering me somewhat.
9 MR. MORRIS: I think if we made the conclusion
10 you were discussing, I think we would be making an
11 operating license conclusion, and I think it would be
12 inappropriate and probably a misuse of resources to try
13 to do that now, rather than just confine ourselves to
14 the CP conclusions.
15 MR. STARK: The other thing is, the PSAR only
16 addresses the levels of drawing and documentation that
17 are required for preliminary review. The applicant has
18 yet to supply the final design documents for the FSAR.
19 So you would have to review it again anyway, because
20 they may not be the final design, and there should be a
21 lot more documentation in the FSAR.
22 MR. BENDER: Okay.
23 MR. MARK: What is meant by "assurance of
24 adequate reactor coolant inventory"? Is that a positive
25 measurement of the level of the surface of the sodium in

1 the primary system or a weighing of an overflow tank
2 that is supposed to have sodium in it?
3 MR. MORRIS: Are you in criterion 7 now?
4 MR. MARK: 27.
5 MR. MORRIS: 27. I would judge that it's an
6 adequate inventory to perform --
7 MR. CHECK: Are you asking how this design
8 meets that criterion?
9 MR. MARK: Well, it's a criterion. What is
10 the criterion hinged on? Looking at the sodium in the
11 reactor vessel or looking at the sodium in the supply
12 tank?
13 MR. LIPINSKI: This relates to the guard pipes
14 and the guard vessel, because assuming that you had a
15 pipe break, that the sodium runs somewhere, but it will
16 not continue to run in a closed -- this is stated
17 generally. But putting the guard vessels and the guard
18 pipes around the main components gives you a way to
19 guarantee that the sodium level doesn't drop below the
20 core.
21 MR. MARK: So it's not really a liquid level
22 recorder?
23 MR. LIPINSKI: No. You're losing inventory,
24 but you have other features that guarantee that you
25 don't just drain the main reactor vessel.

1 MR. MARK: Fine. Thank you, Walt.

2 MR. ZUDANS: This is the kind of criteria that

3 I like generally; you design to meet them.

4 MR. LIPINSKI: Maybe the project would like to

5 comment on that interpretation.

6 MR. GOESER: That would be ours also, Walt.

7 It's a limitation to make sure that you remain with your

8 outlet nozzles covered, so that you continue to cover

9 decay heat even in the face of a leak.

10 MR. MORRIS: I would just go on and point out,

11 for criterion 36 and 37, note that there is a principle

12 design criterion 35 on reactor residual heat extraction

13 system that is essentially the same as GDC-34. Now, the

14 general design criteria do not go ahead and include

15 provisions for inspection and testing of the reactor

16 residual heat extraction system.

17 So here the increment is that the principle

18 design criteria for Clinch River have addressed testing

19 and inspection, whereas the general design criteria for

20 light water reactors do not. I only note that for your

21 information, without making a judgment about what its

22 eventual implication is.

23 I think we discussed each of these.

24 MR. CHECK: I think in that case we could

25 observe that we're anticipating things that are going on

1 now in RHR systems.

2 MR. BENDER: Let me ask about criterion 29,

3 which, if I look at your table up there, is not listed.

4 MR. MORRIS: Which criterion?

5 MR. BENDER: Criterion 29, fracture prevention

6 of reactor coolant boundary. That's not listed as one

7 for which there is no comparable requirement under 10

8 CFR Appendix A. But I don't remember --

9 MR. STARK: It should be criterion 31, and I

10 can look at --

11 MR. MORRIS: If you've got the comparison

12 document that we provided, as you go through the PDC's

13 in each case, for instance criterion 39, you'll see it's

14 identical to Appendix A criterion 31.

15 MR. STARK: That is fracture prevention of

16 reactor coolant pressure boundary, and I guess they just

17 took the word "pressure" out.

18 MR. BENDER: Okay, I'm all right. I see what

19 you're saying. I guess maybe I was drawing a conclusion

20 here that was inappropriate.

21 But the intent in this particular criterion 31

22 was just to design for ductile kinds of properties. But

23 in criterion 29 I guess I had inferred the intent was to

24 make this the basis for showing that double-ended pipe

25 breaks could not occur. And maybe that was a wrong

1 inference.

2 What you were getting out of criterion 29,

3 that's not the same as in -- well, the only difference

4 between 29 and 31 is that one is the intermediate

5 boundary and the other is the primary boundary.

6 MR. STARK: I'm not sure I understood your

7 comment relative to a double-ended pipe break. I don't

8 think we've made the evaluation and the conclusion that

9 that is not a possible accident, in the staff's view,

10 anyway. Perhaps you can restate your question?

11 MR. BENDER: Okay. I was under the impression

12 that you were taking the position that double-ended pipe

13 breaks were not part of the design requirements. Are

14 they or not?

15 MR. MORRIS: The position as stated in the

16 previous review was we believed that one could eliminate

17 consideration of such breaks for the cold leg pending

18 our approval -- the applicant's proposal and our

19 approval of an adequate leak detection and pre- and

20 in-service inspection program. We left open the

21 question of the hot legs, and that is currently under

22 review.

23 This provision, appearing both in the light

24 water reactor and the CRBR criteria, does not mean that

25 you could eliminate breaks because it's there. It means

1 it's prudent to meet this criterion so you could
2 minimize the possibility of those breaks.
3 MR. BENDER: I apologize. I misinterpreted
4 what was in here, and probably I'm asking the wrong
5 question at this stage. Thank you.
6 (Slide.)
7 MR. MORRIS: Again, just a reminder of
8 something I've already said, that we think there are
9 these examples of cases where the design basis accidents
10 affect the way you view the criteria, and the rod
11 ejection and rod drop accidents appear to be somewhat
12 related -- the potential for those events in an LWR
13 appear to be related to the existence of GDC-28. And
14 for major loss of coolant accidents the GDC's that apply
15 there are 35, 36 and 37 on ECCS requirements, and on
16 containment cooling requirements are criteria related to
17 the potential for loss of coolant accidents and steam
18 line breaks inside containment of light water reactor
19 plants.
20 The way we're approaching this, as I said, is
21 to relook at these various kinds of analogues that could
22 occur for Clinch River to see whether we continue to
23 believe that these criteria could be -- legitimately
24 remain out of the set of criteria.
25 (Slide.)

1 The way we propose to do this is in part the
2 PSAR review of chapter 15, careful considerations of the
3 design basis events proposed there to see if we can
4 agree with the applicant that these events that he's
5 proposed represent the bounding events that are credible
6 within each number of categories, including reactivity
7 insertions, sodium leaks and spills, undercooling
8 events, et cetera.
9 We have a team of experienced LMFBR
10 technologists reviewing the possibility of accidents
11 that don't now appear here to see whether we think there
12 should be other categories or other events that haven't
13 yet been included here.
14 And another activity that could be useful here
15 is the front-end PRA review -- that is the part of the
16 PRA that establishes event sequences -- is being
17 conducted now by the applicant. We will have a team to
18 review that to determine whether they have done an
19 adequate job of investigating the need for additional
20 events for Clinch River.
21 There are other parts of the review, however,
22 that are related to this, and the the mechanical
23 engineering review that addresses piping integrity and
24 pipe break criteria is also progressing.
25 (Slide.)

1 Here is a schedule which shows you some of our
2 current goals for addressing these various potential
3 events. We're looking at potential loss of coolant, and
4 one of the things that has to be done as yet is to
5 complete the hot leg pipe break criteria review. Mr.
6 Holt is with us today from the Mechanical Engineering
7 Branch. He has a team of reviewers who are looking into
8 this situation to see what the pipe break criteria for
9 Clinch River should be, and that will have a bearing on
10 our eventual evaluation of these events.
11 We are continuing our review of the cold leg
12 leak detection and inspection evaluation. This may be
13 July. I'm not sure just exactly what the goal here is.
14 I believe we would expect to have a pretty good idea by
15 June of '82 what kinds of pipe break criteria we're
16 going to be dealing with.
17 We hope by October '82 to have completed our
18 evaluation for the proposal of cold leg leak detection
19 and inspection proposals from the applicant. We believe
20 that we will probably perform some analyses using the
21 Super Systems Code that was developed at Brookhaven on
22 some breaks in pipes to understand better what the
23 potential implications of them may be. And this should
24 hopefully be finished by October.
25 And finally, we'll have an integrated

1 evaluation some time in the time frame of December, in
2 which we will be coming to a final position on these
3 issues. In some sense this is an example of why we feel
4 that, given that we're trying to get the CP SER out
5 shortly some time in '83, that it's going to be very
6 difficult for us -- what is the date on that?
7 MR. STARK: March.
8 MR. MORRIS: March of '83. It's going to be
9 difficult for us to come to you with a clear picture any
10 time much before the SER is issued. But as these
11 reviews continue, it will be possible for us to discuss
12 with you further what our continued attitude towards
13 this kind of an event would be.
14 Again, we will consider in our review of PSAR
15 Section 15.6 of the SER the potential for overheating
16 containment. This is the section on sodium fires. We
17 hope we may be pretty far along in this review in July
18 of '82, and we would continue looking through our
19 accident delineation program at the possibility there
20 were some events that weren't considered here that could
21 lead to overpressurization or overheating of containment
22 within the design basis, and final evaluation of that
23 would hopefully be done by December.
24 I would remind you that this accident
25 delineation effort is one that is an attempt to assure

1 ourselves that the appropriate set of design basis
2 events has been proposed and is in the DBE spectrum for
3 Clinch River. It's also an attempt to determine whether
4 there are accident initiators that could lead to more
5 severe accidents, that is by the compounding of
6 failures, and to core disruptive accidents or Class 9
7 events.
8 So it has a twofold objective. There is our
9 review to determine whether there is a potential for
10 large rapid reactivity insertion. PSAR Chapter 15.2
11 characterizes the applicant's current proposal for these
12 events.
13 In addition to that, we're looking at the
14 functional design of the reactivity control system.
15 There are significant differences between the way the
16 control rods are designed and those for light water
17 reactors. This does not mean that we believe there is a
18 high probability we're going to find a mechanism for
19 reactivity insertion event such as could occur for a
20 light water reactor in this review, but it's part of the
21 process.
22 And again, the accident delineation effort
23 will have a potential impact on this final evaluation
24 which we hope to have out by December.
25

1 Perhaps the way to satisfy the need to have
2 the Subcommittee follow along with us on these issues
3 would be to perhaps have some interim reports on this at
4 a later time.
5 MR. MARK: You referred to the SSE super
6 systems code, I believe, in connection with the analyses
7 of pipe breaks. That is not capable of discussing the
8 likelihood --
9 MR. MORRIS: I believe it would be possible
10 --
11 MR. MARK: You mean you assume a pipe break
12 and ask where does the fluid flow go and how fast.
13 MR. MORRIS: Yes, that is right. The pipe
14 break criteria conversed in this schedule and once the
15 Mechanical Engineering Branch arrives at its criteria we
16 would probably have set up by that time the capability
17 to evaluate such an event and attempt to use the
18 postulated breaks in the Code and that would be input to
19 the Code.
20 MR. MARK: You have got guard vessels all over
21 the place. I assume you do not need to calculate the
22 sodium flow at all.
23 MR. MORRIS: What you would try to do -- and
24 again I do not want to overstate the capabilities of SSC
25 at this time -- it is our intention to attempt to

1 determine whether that break could cause a break of the
2 sort that could violate core design limits and it is the
3 process we hope to follow.
4 We, I believe, will be asking the applicant to
5 reconsider some of these breaks and tell us what he
6 thinks the analyses would show. It is my recollection
7 that the PSAR does not currently have analyses of pipe
8 breaks of any sort.
9 (Slide.)
10 I think that is about as far as I was planning
11 to go today. I think on the agenda for tomorrow is some
12 discussion of the process by which we will arrive at the
13 more specific acceptance criteria. However, there will
14 not be a very long discussion. That will give a long
15 time for questions.
16 MR. STARK: We will also have a response
17 tomorrow for the questions that we have received so
18 far.
19 MR. CHECK: Well, under the assumption, Max,
20 that you want to abide by this schedule, we are at the
21 end of Item IV, which is a continuation of Item III.
22 MR. CARBON: Do you have questions?
23 MR. BENDER: I do not think I really have a
24 question. I am not unhappy with the approach that is
25 being taken here. I think there are a lot of things in

1 it that represent good common sense.

2 If I were going to react at all to what I have

3 heard so far, it has more to do with how systematic the

4 approach is and it may be premature to ask how much

5 effort it will take to get to the end of this thing,

6 because I do not think anybody really knows right now,

7 but to get all this in the SER would be difficult, so we

8 are probably going to have to find some way to see it as

9 it develops, and I think we should think about how we

10 will do that.

11 MR. CHECK: We fully expect to be tested on

12 the question of our system and how closely we have

13 adhered to a logical plan developed and implemented.

14 I think that over the course of the several

15 meetings that we have projected, and I hope we will

16 discuss agenda, future agenda and schedules tomorrow

17 morning in the wrap-up. I think through the mechanism

18 of those meetings you will come to know more about our

19 system and we will have a chance to defend it -- display

20 it and defend it.

21 I am optimistic that ultimately we will

22 convert you.

23 MR. CARBON: Carson?

24 MR. MARK: I have a think which really was not

25 raised, I guess, by any of the discussions today and may

1 also seem out of place anywhere. I have a vague
2 recollection of the intention that in the past has gone
3 into speculation as to what would happen if you had an
4 instantaneous double-ended pipe break of the largest
5 pipe in the world, and that really occupied a lot of
6 peoples' time.
7 I believe the Germans have decided that is a
8 lot of nonsense and we will run our reactors without
9 discussing what would happen in such a case, because
10 while it is imaginable, it is not reasonable. I have a
11 little concern as to whether this CDA is not the LMFBR
12 analog of that.
13 You have not yet got through your
14 probabilistic risk assessment and yet it is stated in
15 one of the documents I referred to that this is the
16 biggest contributor risk. Is it or isn't it? If its
17 probability is something like one might wish it were but
18 may not be able to demonstrate, like 10^{-7} or so per
19 reactor year, then to put 85 percent of the research
20 effort on describing that thing that is totally
21 impossible to describe might be questioned.
22 What thought has been given to putting that in
23 its proper box? It is am imaginable, fascinating and
24 absolutely marvelous problem which can absorb millions
25 of dollars and hundreds of manyears and in the end you

1 will say well, of course, we have not necessarily
2 thought of everything. But you know that it could only
3 occur as a consequence of a loss of flow or a transient
4 over power or perhaps there are other things which could
5 bring it about. I am not sure.
6 Those are the ones that I believe are most
7 frequently pointed at and are we right in spending so
8 much time in discussing how it might proceed, or should
9 we not be putting all our eggs in the basket of saying
10 let's have a transient overpower which could lead to
11 it?
12 MR. MORRIS: I agree that one of the main
13 thrusts of our review must be to assure that CDA does
14 not occur or at least is very improbable, and those more
15 specific requirements or design measures that will be
16 built into Clinch River will be designed just for that
17 purpose.
18 A large part of our review is related to
19 avoiding CDAs.
20 MR. MARK: I am delighted. That is what I
21 think it should be.
22 MR. MORRIS: And I think what you may see in
23 that research document is a weighting on the side of
24 understanding CDAs. That is because that research is
25 one of the tools for doing that kind of thing, whereas

1 good engineering practice -- redundancy and a number of
2 other measures -- are the ways that you avoid CDAs.
3 MR. MARK: I can see that you need research if
4 you need to understand CDA because it is very hard to
5 think of how to understand it and it can soak up, as I
6 said, unlimited numbers of dollars and manyears. If you
7 could prove it was improbable, then you do not
8 necessarily have to understand this improbable event.
9 MR. MORRIS: Early on we decided that we must
10 provide in the review ample resources for those
11 activities related to preventing progression toward CDAs
12 and that was an intended design of the way we structured
13 the review and the technical assistance that we put into
14 this process and, to some extent, the research.
15 If you go back and look at that document I
16 think you will see there are a number of programs
17 related to understanding events before you progress
18 toward a CDA. It is an underlying principle.
19 MR. MARK: I realize that. It is just that
20 this is the toughest nut in the picture and so if it is
21 to be handled it takes an awful lot of handling.
22 MR. MORRIS: We hope to use some practical, I
23 would say, good judgment about what is an adequate
24 review of the CDA without carrying it to the ultimate
25 limit.

1 MR. MARK: It was with that in mind that I
2 asked you how prepared are you to put up with a kiloton
3 nuclear bomb dropped through the roof of the
4 containment. You say you are not considering it at
5 all. I am not sure the CDA is not almost in the same
6 basket, especially if you put the work on saying these
7 are the things which could cause it and we work on
8 those.
9 MR. CHECK: One is an external event.
10 MR. MARK: You have probability --
11 MR. CHECK: It is not pure probability.
12 MR. STARK: The Staff also has another
13 requirement. In the hearing there are at least three
14 contentions that deal with design basis accidents and
15 the inclusion of core disruptive accidents in the design
16 basis accidents.
17 There admitted contentions right now that we
18 have to address.
19 MR. MORRIS: The question of whether the core
20 disruptive accident is or is not a DBA, I think, a reply
21 to that is as follows: We believe that we should -- the
22 Applicant should take measures and we should impose
23 measures to assure that the CDA is to improbable that it
24 should not be necessary to consider it within the design
25 basis spectrum.

1 However, we believe that we should then check
2 the design to determine that if a CDA occurred that
3 there would not be an early containment failure and
4 hence an unacceptable consequence from that event that
5 is inevitable if the CDA is initiated. And we believe
6 that it is practical and feasible to impose design
7 features on the plant to assure that the CDA is an
8 improbable event sufficient to exclude it from the
9 design basis.
10 That is part of our effort.
11 MR. MARK: I did not mean to be challenging
12 your approach. I knew a little bit about it, I suppose,
13 and I think the way you have put it sounds very good to
14 me. You have got to consider it. I realize that.
15 MR. CHECK: I doubt there is anybody in this
16 room who would not grab at the mechanism for excluding
17 the CDA. I guess what we are doing is we are confessing
18 to you we do not know how to do that.
19 MR. MARK: And neither do I.
20 MR. CHECK: So we are going to do what we can
21 to minimize the probability and we are going to look at
22 ways to accommodate it even if it were to occur.
23 MR. MARK: For the latter, it is not
24 immediately clear to me that you need to describe the
25 way the fuel drips, slumps, glops, drizzles from here to

1 there. It might be possible to bound it by the ramp
2 rate and in fact even if we have such a ramp rate which
3 is higher than we can see, that it does not come off and
4 the sodium does not come out any faster than this.
5 MR. MORRIS: If we can find a way to
6 understand the event well enough to treat it by bounding
7 analyses we will follow that path. That may be
8 difficult to do.
9 MR. MARK: Look, I admit the difficulty and I
10 am just expressing a wish rather than a thought.
11 MR. CARBON: Walt, do you have questions?
12 MR. LIPINSKI: Your comment as to whether the
13 CDA could be a DBA, you did not emphasize the difference
14 in what you have to do with the analysis as to why the
15 CDA is not treated as a DBA, the conservative
16 assumptions that go into the DBA versus using realistic
17 assumptions in the CDA.
18 If you are going to include design features
19 you could call it a design basis, but then what do you
20 bring into play when it gets to be a DBA in terms of the
21 analysis you require? That is where the difference
22 lies.
23 MR. MORRIS: Is that a question?
24 MR. LIPINSKI: When you made your statement
25 that the Intervenors were asking it to be classified a

1 DBA, you did not proceed to explain why you wanted to
2 keep it a CDA as to what the difference was between DBA
3 analysis and CDA analysis.
4 MR. CHECK: Between Class 9 and Class 8 and
5 below that is a lot of money, a different design.
6 MR. MORRIS: The rules for analysis of a LOCA,
7 for instance, are codified. They are very formalized.
8 We do not believe it is imperative to do that for a CDA
9 or Class 9 events. Well, it may seem sort of
10 inappropriate to say there is an element of realism in
11 evaluating a CDA when some people think there is no
12 realism in assuming it could occur.
13 But we would imagine that our evaluations and
14 the analysis done by the applicant does not have to be
15 as conservative for a CDA as they do for a DBA and yet I
16 do not know how to gauge exactly what that level of
17 conservatism should be. It is usually a matter of
18 judgment after we understood the case better and better
19 understand where we are with regard to what
20 conservatisms need to be imposed on that analysis.
21 MR. BENDER: In this notorious Gamble-Caffey
22 letter numbers were kicked around like 1200 megawatt
23 seconds as being a number that had to be complied with.
24 I have never been comfortable that anybody had a
25 legitimate basis for deciding what the number ought to

1 be.

2 But can I judge from what you are saying that

3 you would leave open the question of what the number

4 ought to be?

5 MR. MORRIS: I believe that there is a new

6 core design which may involve different scenarios or

7 different details with regard to how the accident would

8 progress that we would not confine ourselves to 1200

9 mega-joules and that that was the number generated for a

10 specific core design.

11 We may have to arrive at a different number

12 and we may find that that number will be a sufficient

13 bounding number for the current core.

14 MR. BENDER: Well, it might be too much, as it

15 was before.

16 MR. MORRIS: If we can determine that there is

17 a basis for choosing another lower number we will do

18 that, and if we determine that there is a basis for

19 another higher number we would do that.

20 MR. BENDER: Well, I am going back to Dr.

21 Mark's comment a moment ago that the calculation was a

22 highly detailed computation involving very uncertain

23 processes and we know so little about whether the

24 processes will go in that direction in trying to

25 discriminate by factors of two or three in the numbers

1 that are computed seems to me like dealing with straws
2 in a high wind, which is working on parts of the problem
3 that do not make much difference.
4 MR. CARBON: Bill, do you have questions?
5 MR. KASTENBERG: Yes, two things.
6 I hate to beat a dead horse but I have been
7 sitting here and I have been troubled a little bit by
8 your reiteration of the CRBR safety philosophy based on
9 the May 6 Denise-to-Caffey letter, and my trouble comes
10 from the following.
11 In 1976 our current level of safety for that
12 generation or any generation of lightwater reactors --
13 our perception of that level of safety was different
14 than our perception of the level of safety of lightwater
15 reactors today. I think our perception was very narrow
16 in terms of what we felt the level of safety was.
17 I think today our perception has a big
18 spread. If you see estimates of core melt, they vary
19 from 10^{-3} to 10^{-6}, so I do not know -- at least I am
20 not comfortable with what is the level of safety that I
21 am trying to target and what the leads me to is to look
22 for a safety philosophy which is somewhat independent.
23 Granted it is still what we know about LWRs but it is
24 somewhat independent of how we arrive at a level of
25 safety for the lightwater reactor.

1 For example, if I were rewriting this, I might
2 write for the last sentence, we have the design
3 approaches to accomplish the required level of safety,
4 should be similar or analogous to LWR practice. I would
5 say "should be based on sound engineering practice."
6 And if that rests on something similar or analogous to
7 LWR practice, fine. If not, in accordance with
8 standards.
9 We do have ASME standards and we are
10 developing ANS standards. I would fall back on the LWR
11 practice when I had to, but I would look for something,
12 I think, which is somewhat stand-alone.
13 MR. STARK: I think we are doing what you are
14 saying because we have to look and see if the lightwater
15 standard fits, and if it does not why does it not fit
16 and what is the acceptance criteria for each section of
17 each chapter of the safety analysis report.
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1 We are looking for guidance or help wherever
2 we can find it. And if not, we have to see if we can
3 establish the acceptance criteria ourselves and come
4 down here later on and defend it before the whole ACRS.
5 MR. CHECK: You like your words obviously
6 better than those. Is it this statement or --
7 MR. KASTENBERG: I think this kind of
8 statement can get you into trouble later on.
9 MR. CHECK: Personally, I do not have any
10 problem with what you say. And like a lot of things
11 that we have inherited, we have been picking them up as
12 time permits and examining them, turning them around.
13 But you have got a point.
14 MR. MARK: The gist of what you were saying, I
15 think, is it ought to be put in terms that it would
16 stand alone, and if they compare with LWRs, fine, but
17 they should not be hung on them.
18 MR. KASTENBERG: That is basically the thrust,
19 yes.
20 MR. CHECK: I like that. You know, "safer
21 than coal mines."
22 [Laughter]
23 MR. CHECK: I understand. Why tie ourselves
24 to water reactors.
25 MR. MARK: Especially since we have seen in

1 the time since that was written that the light-water
2 thing is not what it used to be.
3 MR. KASTENBERG: It may be merely a matter of
4 perception. I do not know whether I would rest your
5 case on this statement that was written six years ago.
6 MR. MORRIS: Our case will be really resting
7 on the review and the findings in the SER. That, I
8 believe, was a kind of a statement of principle that was
9 early on in the process trying to set a common ground
10 for both the applicant and the NRC to proceed upon. And
11 maybe we should not read anything more in it than that.
12 I would tend to believe, though I find it
13 difficult to understand any problem in looking for a
14 safety goal for this plant, some kind of an overall
15 generalized goal that anything other than to make the
16 risk somewhat comparable to a current light-water
17 reactor plant --
18 MR. MARK: Would it satisfy, you, Bill, if
19 they should write our safety goal, as to have this be at
20 least as safe as we think light-water is?
21 [Laughter]
22 MR. CHECK: As they were in 1976.
23 MR. KASTENBERG: You get the point.
24 MR. MORRIS: We will not reach that conclusion
25 by clear scientific process. As almost everything in

1 the regulatory business is, there will be a large
2 measure of judgment involved.
3 MR. KASTENBERG: That is why I say I would
4 hinge it on what I consider sound engineering practice.
5 We do have standards and acceptance criteria, and that
6 is really how you are going to base this.
7 MR. MORRIS: Are you talking about LMFBR
8 standards now?
9 MR. KASTENBERG: We are developing LMFBR
10 standards, and we have other standards.
11 MR. MORRIS: It might be just as unwise to
12 base it on those standards, given that they do not have
13 quite the stature and long standing of the light-water
14 reactor standards. And if we had more experience in
15 LMFBRs and had the time to come to agreement about those
16 standards, I think that would be a fine way to proceed.
17 But I think at the time that was written,
18 those standards probably were not quite as mature as
19 they are now, and even now we have not had time to
20 review all the LMFBR standards and believe that they are
21 directly applicable.
22 MR. CARBON: Did you have a second question?
23 MR. KASTENBERG: Yes. You made the statement
24 earlier with regard to a question raised in containment
25 heat removal, and the statement was made that since

1 containment heat removal is not required for any design
2 basis event, it did not include a criteria for it. And
3 I wonder, is the converse true? If you found later on
4 that those events for which you need containment that
5 removal became design basis events, would you then have
6 a criteria for containment heat removal?
7 MR. MORRIS: The answer, I believe, is yes.
8 If we found there were events that we thought should be
9 in the design basis for which containment heat removal,
10 the answer is yes, we would require that that be added.
11 MR. KASTENBERG: I sense that the subcommittee
12 and the consultants were asking in different ways what
13 is your philosophy, why do you include things and why do
14 you not? Could I then extrapolate that example and say
15 that is part of your underlying philosophy, not the
16 whole philosophy, in trying to decide what your design
17 criteria that you want in which are --
18 MR. MORRIS: I believe the answer is yes, I
19 think the way I presented it is, as we investigate the
20 kinds of events that could occur, that may cause us to
21 change the criteria that we think should be imposed and
22 they seem to go together, to me.
23 MR. CARBON: Zenon.
24 MR. ZUDANS: Not much new. I still remain
25 with the feeling that the criteria should be more

1 general, that there are very specific things addressed
2 to very specific designs that should be addressed in a
3 second package, which should be called specific
4 criteria. So some of those criteria might be moved
5 around.
6 But I have a much better feeling now at the
7 end of the day because I think I understand a lot better
8 what the process is and I really see nothing wrong in
9 what you are doing.
10 MR. MORRIS: I think one of the things that I
11 noted when we talked about the circular process is that
12 it may not be necessarily logical but you can test it
13 for consistency and there must be consistency in the
14 process even though you can say something always follows
15 something else. But consistency would be the way I
16 would judge this process.
17 MR. ZUDANS: I would like to make one more
18 comment; namely, with respect to existing industry codes
19 and criteria. I agree that the LMFBR criteria float
20 around, they are not based on experience. So I do not
21 know how they could do much better than you could do.
22 The ASME high-temperature criteria were
23 developed to meet specific situations where you could
24 not get away with a Section 3 or Division 1 type of
25 criteria. So really this completes the whole circle.

1 MR. MORRIS: We have not brought it out, but
2 that particular area is one that we have devoted a lot
3 of effort to and will be devoting a lot of effort to in
4 our review, high-temperature and mechanical design is
5 getting a lot of attention. And the way we are
6 approaching it is that what we are trying to do is
7 determine whether the practices in code cases that have
8 been recently developed can be accepted by the NRC for
9 regulation.
10 MR. ZUDANS: That is a major job, because N-47
11 now stands -- this is Revision 21, and there is no end
12 to it. As new problems develop, the old criteria do not
13 satisfy them and something new is developed.
14 MR. CARBON: I guess perhaps that sort of
15 finishes everything this afternoon. I would call to
16 your attention in case you have not seen each of our
17 three consultants have turned in written comments with
18 regard to the criteria. And Paul should see that you
19 get those for your consideration.
20 MR. STARK: We have them and are prepared to
21 summarize on Roman numeral VI tomorrow morning our
22 comments on the consultants' comments.
23 MR. CARBON: Let me remind everyone that the
24 meeting tomorrow is in room 762 rather than here. And I
25 guess that is it until 8:30 tomorrow morning.

1 [Whereupon, at 4:25 p.m., the Subcommittee was
2 adjourned, to reconvene at 8:30 a.m. on Wednesday, March
3 31, 1982.]
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NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

In the matter of: ACRS/Subcommittee on Clinch River Breeder Reactor

Date of Proceeding: March 30, 1982

Docket Number:

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Ann Riley

Official Reporter (Typed)

Ann Riley

Official Reporter (Signature)

Synopsis of CRBR PSAR Chapter 15 Events
(March 8 , 1982; through Amendment 65)

Reactivity Insertion Design Events

15.2.1.1 Control Assembly Withdrawal at Startup

During startup, after criticality, a malfunction of an electronic control causes withdrawal of one primary control rod at maximum speed. Result: maximum cladding temperatures less than temperatures expected at full power operation.

15.2.1.2 Control Assembly Withdrawal at Power

Withdrawal of one control rod (assembly) at power can lead to transients to 115% of full power if begun at 100%, or to 120% if begun at <100%. Result: maximum cladding temperature not more than about 100 ^{o}F over temperatures at full power; no significant effect.

15.2.1.3 Seismic Reactivity Insertion - OBE

Operational basis earthquake causes loss of power to the pumps and a step reactivity insertion (by compaction of the core) 0.5 second thereafter, followed by control rod insertion slowed by seismically-induced friction. Result: maximum cladding temperatures less than 50 ^{o}F over temperatures at full power.

15.2.1.4 Small Reactivity Insertions

This analysis includes several possible small reactivity insertions (e.g., a small shift in position of a core component), either ramp or step reactivity insertions. Assuming worst case combinations of tolerances, maximum cladding temperatures of 1500 ^{o}F and 1560 ^{o}F (secondary scram) result; maximum temperatures at

normal full power operation are over 1400 ^{o}F.

15.2.1.5 Inadvertent Drop of a Single Control Rod at Full Power

An electrical or mechanical fault causes a control rod drive mechanism to release its rod. Result: due to action of the Plant Protection System, the transient is not more significant than that from a conventional plant trip shutdown.

15.2.2.1 Loss of Hydraulic Holddown

Loss of hydraulic holddown (by inconceivable" blockage of certain conduits) might cause upward motion of some fuel assemblies by as much as 2.5 inches; the possible reactivity insertions lead to transients resulting in maximum cladding temperatures of 1420 ^{o}F and 1430 ^{o}F, not significant to cladding lifetime.

15.2.2.2 Sudden Core Radial Movement

Stick-slip motion of fuel assemblies toward a more compact core arrangement causes a reactivity insertion. Maximum cladding temperature would not exceed 1470 ^{0}F for normal scram conditions, nor 1510 ^{o}F under SSE conditions; no significant degradation of cladding results.

15.2.2.3 Maloperation of Reactor Plant Controllers

Assuming failures of the sequencer and its limit circuits and the rod blocks permits seven control rods to move out. Reactivity ramp is terminated by the PPS within the operational incident limits.

15.2.3.1 Cold Sodium Insertion

Assuming the accidental startup of an inactive loop while on 2-loop operation and partial failure of the PPS, cold sodium could be pumped into the reactor vessel causing a reactivity transient. Results are cladding temperatures less than those at full power operation, because 2-loop operation would be at not more than 67% of full power.

15.2.3.2 Gas Bubble Passage Thru Fuel, Radial Blanket and Control Assemblies

Assuming that a 4-inch bubble could pass simultaneously into a row of 8 assemblies (> 3 cubic feet of gas), a reactivity insertion lasting less than 0.3 second occurs, resulting in a maximum cladding temperature increase of 68 ^{o}F.

15.2.3.3 Seismic Reactivity Insertion - SSE

Safe Shutdown Earthquake causes loss of power to the pump after 0.5 second, a step reactivity insertion (by compaction of the core,)followed by control rod insertion slowed by seismically-induced friction. Result: maximum cladding temperatures about 1505 ^{o}F, not more than 100 ^{o}F over normal full power maximum temperatures.

15.2.3.4 Control Assembly Withdrawal at Startup - Maximum Mechanical Speed

Assuming the worst failure of the electronic controller, such that the 9-inches per minute (ipm) limit is failed, when the CRDM reaches 73 ipm, its roller nut will disengage the lead screw resulting in a drop of the rod. The reactivity transient causes maximum cladding temperatures of about 800 ^{0}F, well below normal full power temperatures.

15.2.3.5 Control Assembly Withdrawal at Power-Maximum Mechanical Speed

Assuming the worst failure of the electronic controller such that the CRDM accelerates a rod withdrawal, at 73 ipm centrifugal force causes the rod to be dropped. The transient causes maximum cladding temperatures in the 1400 to 1470 ^{o} range, near the maximum normal full power temperature.

Undercooling Design Events

15.3.1.1 Loss of Off-Site Electrical Power

Simultaneous, multiple failures cause loss of all off-site power to the 13.8 KV buses and, due to loss of sodium pump power, reduction in core flow. Result: clad midwall hotspot temperature limited to 1410 ^{o}F by primary shutdown, to 1630 ^{o}F if shutdown is by the secondary system. Even the latter case is acceptable because the cladding damage does not exceed the limit.

15.3.1.2 Spurious Pump Trip

The trip of one primary sodium pump may be due to a pump failure or to an A.C. bus fault, which would also trip the corresponding intermediate loop sodium pump. The result is a maximum hot spot midwall clad temperature of 1390 ^{o}F.

15.3.1.3 Spurious Intermediate Pump Trip

The trip of one intermediate sodium pump may be due to a pump failure or to an A.C. bus fault. The PPS shuts the reactor down, with results similar to loss of off-site power with normal trip; clad hotspot temperature is not hotter, but is at high temperature longer.

15.3.1.4 Inadvertent Closure of One Evaporator or Superheater Module Isolation Valve

Inadvertent closure of an isolation valve results in rapid drying out of the affected unit followed by reactor trip due to steam/feed flow mismatch or, failing that, by trip on high IHTS cold leg temperature. Resulting maximum core temperature increases are about 30 ^{o}F.

15.3.1.5 Turbine Trip

Turbine trip for whatever reason (e.g., loss of lube oil pressure) leads to closing the turbine inlet valve, opening bypass valves, and reduction in reactor power at 3% per minute. No significant effects on core occur. The steam dump to the atmosphere releases whatever tritium has accumulated in the water system, a maximum of 0.25 μCi/g from which a 2-hour site boundary dose of about

99 mrem whole body dose was calculated(and about 1 mrem skin beta dose).

15.3.1.6 Loss of Normal Feedwater

Total loss of normal feedwater (e.g., from failure of a feedwater pump) leads to reactor trip by the steam-feedwater flow ratio and activation of the SGAHRS including opening of the safety relief valves. After pressure declines and relief valves close, the SGAHRS operates with the auxiliary feedwater and protected air-cooled condensers to remove heat. Core experiences normal plant trip temperatures.

15.3.1.7 Inadvertent Actuation of the Sodium-Water Reaction Pressure Relief System

A spurious activation of the water-steam side of the sodium-water reaction pressure relief system, without the sodium dump, isolates the evaporator-superheater module and dumps its water. The PPS trips the plant on steam-feedwater flow mismatch or on high IHTS cold leg temperature; core temperatures are less than those of normal full power operation. The IHX and associated equipment experiences some thermal stress.

15.3.2.1 Single Primary Pump Seizure

Seizure of a primary sodium coolant pump, considered with instantaneous stop of impeller, leads to plant trip by speed ratio of primary and intermediate pumps, or by flow ratio between primary and secondary loops; resulting maximum clad

midwall temperatures are 1400 and 1470 ^{o}F respectively. The trip times are so short that action of the check valve does not affect the maximum temperatures.

15.3.2.2 Single Intermediate Loop Pump Seizure

Seizure of an intermediate loop pump leads to plant trip by speed ratio of primary and intermediate pumps, or by flow ratio between primary and secondary loops. The core sees a normal plant trip, with temperatures not above its initial steady state.

15.3.2.3 Small Water-to-Sodium Leaks in Steam Generator Tubes

For small leaks, it is assumed that the water-to-sodium leak detection system will alert the operator who will then shut the plant down normally; this would be followed by controlled cooldown and depressurization of the affected steam generator and drainage of the IHTS loop. Plant trip drops the core temperatures so that the hot sodium from the affected loop does not cause temperatures to reach normal full power operating temperatures. Radiological impact of this event is insignificant.

15.3.2.4 Failure of the Steam Bypass System

The steam bypass system, thru 4 independent subsystems, maintains steam pressure while releasing steam to the main condenser following turbine trip. Failure of a bypass valve to open will cause

a pressure increase leading to relief valve action. Failure of all 4 bypass valves to open would lead to reactor trip by the three steam-feedwater flow ratio trips, or, later, by low steam drum level trip. Results are like the Loss of Normal Feedwater, i.e., the core experiences normal trip temperatures. (Venting from relief valve operation leads to small radiation doses, from tritium.*)

15.3.3.1 Steam or Feed Line Pipe Break

The events considered in this group include main steam line rupture, break in a steam line from a superheater to the main steam header, break in a saturated steam line between the steam drum and the superheater, feedline break, and recirculation line break. The reactor is tripped by a steam-to-feedwater flow ratio trip or by a secondary shutdown system (e.g., low steam drum level); the Na coolant flow is maintained by the pony motors at a lower velocity, so that the temperature transient is experienced by the core only after 150 seconds. The plant design must accommodate the temperature transients and the steam venting in cells in the steam generator building.

15.3.3.2 Loss of Normal Shutdown Cooling System

Loss of the main condenser is equivalent to loss of the normal shutdown cooling; consequences are similar to the case of failure of the steam bypass system. The reactor would be scrammed, and the pressure transient would lead to relief valve venting. Core temperatures would be like those following a normal trip.

* Editorial comment.

15.3.3.3 Large Sodium-Water Reaction

A large, sudden leak in a steam generator tube is the most likely source of a large sodium-water reaction. The leak could have a flow equivalent to half that possible from a double ended guilloting failure, and it is considered possible that the water/steam jet could propagate the leak to a maximum of 2 more tubes. The injection of water into sodium creates a high pressure pulse which is relieved by rupture disks in the IHTS, draining off the sodium and reaction products and automatically isolating and depressurizing the water side of the steam generators in that IHTS loop. The reactor is tripped on steam-feedwater flow mismatch or on the sodium-water reaction; in either case, the trip is long before the temperature transient is transported back to the core. Calculated radiological consequences, from tritium in vented steam and from small amounts of in-leakage primary sodium in fumes vented from the sodium-water reaction, are about 0.012 mrem whole body, 0.04 mrem lung and 0.02 mrem bone dose.

15.3.3.4 Primary Heat Transport System Pipe Leak

Analysis indicates that a leak of 75,000 gallons per minute would be required for core sodium temperatures to approach saturation; leak detection capability is estimated to be better than 3 gal/min. A 30 gal/min PHTS pipe leak would not result in a measurable core transient; a normal reactor shutdown could be made. Coastdown of the pumps (less than a minute) would reduce system pressure and the leak rate. Several days would then be available for further reduction of the leakage.

Consequences are plant downtime, sodium cleanup and pipe repair.

15.3.3.5 Intermediate Heat Transport System Pipe Leak

In most break locations, a large leak will trip the reactor on primary-to-intermediate flow ratio trip. Possibly a large leak between the flow meter and the IHX might not trip the reactor on either primary-to-intermediate flow ratio trip or primary-to-intermediate pump speed ratio trip; in such a case, primary cold leg temperature trip would shutdown the reactor. Reactor trip on flow or pump speed ratio would be early enough that the core would experience no temperatures above those of normal full power operation. A later trip, occurring one second after the primary cold leg temperature trip setting is reached, could lead to increases in core sodium temperature but it would be expected to remain below 1500 ^{o}F.

15.4.1.1 Stochastic Core Fuel Pin Failure

A stochastic failure is a random pin failure that is unpredictable; it could result from an undetected random cladding defect. "Stochastic failure of fuel pins is an anticipated occurrence for the CRBRP and such failures can be accommodated easily. Experiments in support of FFTF supplemented by supporting analyses have shown that for any postulated fission gas release mode, there are no serious thermal effects on adjacent pins or structures within the core. A transient jet of gas could not produce cladding overheating sufficient to cause failure of a neighboring pin, and even a steady jet could not cause failures because of the internal flow resistance in the failed pin. It has also been shown that

gas blanketing could not cause significant cladding overheating. Furthermore, a volumetrically large gas release from a pin could not stop local coolant flow long enough to cause significant cladding overheating. Based upon experience with stochastic failures in other sodium cooled plants, there should be no adverse mechanical effects from the expected mode of slow gas release through a small hole. For a postulated burst-release mode through a large rip in the gas plenum, no mechanical damage to the neighboring fuel pins or fuel assembly duct would occur. Finally, no adverse long-term effects of fuel pin failure are expected to occur even if some sodium logging of the fuel and some leaching of fission products would occur." Although fuel pin failure is expected to be such that only very small particles of fuel would be released, the potential for fuel release to cause flow blockage has been examined; from experience and analysis, no such blockage is expected. The plant is designed to accommodate operation with 1% failed fuel.

15.4.1.2 Overenriched Fuel Rod Failure

An overpower rod is one which contains pellets of an enrichment higher than intended for its position in the reactor core. Because of QA controls on pellet enrichment at the fuel fabricators, and because of mechanical keying of the fuel pins and fuel assemblies, it is highly unlikely that an overpower pin will be assembled and find its way into the wrong core location. Pins with maximum credible over-enrichment placed into a peak power location may not fail immediately under severe conditions but are marginal for long-term operation and are likely to fail during normal

expected transients. The CRBR statistical hot pin requires an overpower greater than 25% to reach the fuel incipient melting point; data for low burnup pins indicates even greater margins, by about 20%. Tests operating at overpower sufficient to melt fuel at the pellet middle have shown that operation with 20 to 30% of the pellet molten will not necessarily fail the cladding. If the cladding should fail with injection of molten fuel into the sodium, it does not appear that damage would be propagated further, to other fuel pins or to the duct.

15.4.1.3 Flow Blockage in a Core Assembly

This analysis considers first various potential sources of a blockage, and then the potential effects of blockage. Blockage initiators considered are wire wrap failure, excessive pin bowing or clad swelling, and foreign matter in the primary coolant. Experience and analysis indicate that wire wrap failure, excessive pin bowing or clad swelling are both improbable and not effective in forming blockages sufficient to be of concern. Inlet design precludes flat plate blockage as occurred in Fermi Unit I, and a combination of multiple orifices and strainers makes plugging by debris unlikely. For fuel damage to be experienced, an appreciable quantity of foreign matter (with diameters less than 0.25 inch) would have to be deposited in a non-random fashion in a fuel assembly. Mechanisms to prevent such occurrence are cleanliness during construction, requirements on cleanliness of sodium during filling and operation, pre-operational cleanup of the primary system with the core special filter assemblies, and use

of the cold trap during normal operation. Evaluation of potential effects of corrosion products, lubricants and other degradable materials, and failed fuel debris indicates little chance for blockage. Potential pin bowing or swelling may result in local hot spots on cladding but will not lead to significant blockage. Experimental data and analysis of blockages indicate extensive blockage, e.g., more than 50% of inlet flow area, must occur before there is reduction in flow significant with regard to temperature increases. Even a heat-generating blockage, comprised of a significant amount of fuel material from failed fuel, would need to be relatively non-porous to restrict flow enough to cause significant temperature effects on the fuel.

15.4.1.4 Postulated Module Inlet Blockage

The combination of multiple orifices with a variety of spacing and locations, some with intervening debris barriers, essentially precludes blockage in significant quantity. It would take a large mass of particles 0.5 inches diameter or smaller to plug enough of the strainer or spaces in the fuel assembly to result in significant temperature increases in the assembly.

15.4.1.5 Small Gas Bubble Passage Through Fuel, Radial Blanket and Control Rod Assemblies

Because bubbles passing through the core can cause a positive reactivity insertion and additional heating of the cladding, design to preclude bubbles entering the core includes venting from potential gas pockets at the core support cone, in the IHX

and elsewhere, and pump design and a low cover gas pressure to reduce gas entrainment. The lower plenum design also is intended to break up any large bubbles. If a small bubble, e.g,, 4 inches high and 4 assembly rows in radial extent, should pass through the core, the power burst of about 0.1 second at about 30% increase would cause a cladding temperature increase of less than 25 ^{o}F; repetition of this a million times would not be expected to cause significant cladding damage.

15.4.2.1 Stochastic Absorber Pin Failures

Quality assurance similar to that for fuel makes the likelihood of stochastic absorber pin failure small. Failure of an absorber pin causing a helium gas jet to impinge on an adjacent pin could raise its clad temperature by only 110 ^{o}F which for maximum initial clad midwall temperature and linear heat rate would be 1335 ^{o}F, well below the 1600 ^{o}F guideline limit for short transients. Gas blanketing duration would be only 0.15 seconds, much less than the 2.3 seconds required to reach 1600 ^{o}F. The maximum force impulse from a gas jet is also insufficient to damage an adjacent pin by mechanical action. Although clad failures are expected to be such small sizes that erosion of the boron carbide (B_4C) is unlikely, erosion tests of irradiated boron carbide have resulted in an average particle diameter of 0.0012 inch, with a small number of larger particles expected. From this, it is expected that the smaller particles will circulate with the coolant without causing significant blockage or erosion, and that larger particles will settle out before reaching the pump bearings.

15.4.2.2 Overpower Control Rod Assembly

Mechanisms to cause an overpower condition are a wrong core location, an overenriched pin in a lower enrichment assembly, overenriched pellets, and improper orificing. Mechanical discriminators act against mislocation of control assemblies or control pins; quality assurance reduces the likelihood of overenriched pellets and improper orificing. Primary control pins containing 55% boron-10 are used in outer assemblies; the central one uses natural, 20% boron-10; thus, an overenriched case must be in the central position. Overpower leads to higher temperatures, and increased helium release in the clad and thus to higher gas pressures. Because of the large margin below boron carbide melting temperature and no expectation of pellet-clad interaction, the expected pin failure mechanism is by gas overpressure. In the overpower case, at end of equilibrium cycle, the plenum pressure would be almost at the preliminary design guideline maximum of 4000 psi; therefore, clad failure may occur. Transients could raise the temperature and initiate failure earlier. It is expected that pinhole failure is more likely than gross rupture, but if rupture occurs, the erosion rate is expected to be small. Because the eroded particles are expected to be no bigger than 0.002 inch diameter and substantial portions dissolve, blockage due to their deposition is not likely.

15.4.2.3 Flow Blockage of a Control Assembly

Measures to avoid blockage of a control assembly are the same as those to avoid blockage of a fuel assembly. However, the

control assemblies are operated at lower linear power than fuel assemblies, and are overcooled. Thus control assemblies require an appreciably larger percent of their flow area to be blocked before they will start to boil sodium coolant.

15.4.3.1 Stochastic Radial Blanket Pin Failure

As with fuel and control rods, conservative design and quality assurance will keep the numbers of stochastic radial blanket pin failures small. Expected forms of failure are small leaks in the clad releasing gases, and entry of sodium into the fuel at cracks in the clad. Radial blanket pins have a large fission gas plenum and small fission gas pressure; these, in addition to high gas flow resistance due to small pellet-to-clad clearance, will make the outleakage of gas suboptimal for failure propagation. Under conservative maximum conditions, a gas jet will not cause adjacent clad to exceed 1541 ^{o}F nor to be blanketed longer than 0.05 second, which is adequately less than the 0.096 second calculated as needed for the clad temperature to reach 1600 ^{o}F. Similarly, the impact of the gas jet is not enough to cause mechanical damage. Most of the analysis is extrapolated from that for fuel, and usually favors the blanket due to larger diameter clad (0.253 inches vs. 0.107 inches) and smaller wire wrap spacing (2 inches vs. 6 inches). There is some experimental basis for concern about sodium-fuel reactions leading to swelling if sodium leaks into cracked pins; experience indicates this is not a serious danger.

15.4.3.2 Overpower Radial Blanket

Cooling requirements for radial blanket assemblies depend on reactor residence time and proximity to the reactor core center. The requirements are met by coolant flow control plus shuffling of the assemblies within the radial blanket zone. Coolant flow control is by fixed orifices in the inlet modules. A preprogrammed shuffling routine including numerous safeguards makes misplacement of an assembly highly unlikely. The event analyzed, judged to be the worst case, is repeated failure to shuffle elsewhere an assembly which remains in the highest flux position; this event requires only a human error in setting up the shuffling program. Within a year after failure to shuffle, some melting of the oxide fuel would be experienced. Cladding failure where cladding temperature is maximum would be expected; this is above the point of molten fuel ejection and below the top of the pellet stack so the release expected would be a less-than-maximum release of gas. The uncertainties are such that molten fuel ejection can not be excluded; a molten-fuel-coolant interaction is possible. Neither it nor a gas jet release are expected to lead to failure propagation from assembly to assembly. The molten fuel jet could create a small penetration in the duct wall and in the duct wall of the adjacent fuel assembly, but no further consequences.

15.4.3.3 Flow Blockage of a Radial Blanket Assembly

As in other parts of the core, flow blockages are considered extremely unlikely due to inlet design and precautions to be taken in cleanliness of the coolant system and sodium. The prevention and detection systems for blockages are the same as for fuel assemblies, except that Delayed Neutron Detectors will be relatively ineffective because blanket pellets contain fissile material fractions appreciably smaller than fuel pellets. As with fuel assemblies, a large blockage fraction is required for a significant reduction in coolant flow and increase in outlet temperature ; generally, to cause sodium boiling, a blockage occupying half or more of the flow area of the assembly is required. Smaller blockages might raise cladding temperatures locally but go undetected because of the minor increase in bulk sodium temperature; such hot spots would reduce the effective lifetime of the blanket pin cladding.

15.5.2.1 Fuel Assembly Dropped Within Reactor Vessel During Refueling

The In-Vessel Transfer Machine (IVTM) draws a fuel assembly out of the core into a tubular housing, then drops it into either an open lattice position or onto a location occupied by another assembly. Fission gases and fuel particles might be released through damaged cladding to the primary heat transport system. Severe mechanical damage to the IVTM and other hardware could occur, especially if the triple rotating plugs in the reactor head are operated. In the worst case, release of some cover gas might occur, with results as in Section 15.5.2.4.

15.5.2.2 Damage of Fuel Assembly Due to Attempt to Insert a Fuel Assembly Into an Occupied Position

A series of procedural errors and interlock failures would be necessary to lead to damage of a fuel assembly by attempting to insert it into an occupied location. Consequences are most likely within the umbrella of Section 15.5.2.4.

15.5.2.3 Single Fuel Assembly Cladding Failure and Subsequent Fission-Gas Release During Refueling

Spent fuel assemblies are in the core, the IVTM, the Ex-Vessel Transfer Machine, the Fuel Handling Cell, and the Ex-Vessel Storage Tank. It is judged that the EVTM is where dilution is smallest and the number of seals could lead to the largest release from a single failed fuel assembly. The analysis assumes release of 100% of the noble gases and halogens in the fuel assembly to the interior of the EVTM, with subsequent diffusion of the radioactivity through the EVTM seals into the combined space of the Reactor Containment Building and Reactor Services Building. (What happens then is not specified*.) Off-site doses are calculated for fuel with 36 hours decay time after reactor shutdown, and for fuel with 87 hours decay time; the maximum 2-hour dose at the site boundary is 1.89 rem to the thyroid.

15.5.2.4 Cover-Gas Release During Refueling

Cover gas may be released during refueling by mechanical damage, by improper sequencing of refueling motions, or by separation of the Auxiliary Handling Machine from an open floor valve during a seismic event. The earliest time this could occur is 30 hours

* Editorial comment.

after shutdown, during which period the RAPS has been in normal operation, cleaning up releases from 1% failed fuel. The cover gas is considered released instantaneously to the RCB and RSB and then to the environment, as if the railroad door in the RSB was open in violation of procedures. The maximum 2-hour site-boundary dose is estimated at 4.4 millirem whole body.

15.5.2.5 The Heaviest Crane Load Impacts the Reactor Closure Head

The reactor head structure is designed to accomodate the impact of the 100 ton Auxiliary Handling Machine at maximum lowering speed of 5 feet per minute. If, contrary to expectations, this should result in release of the cover gas, the results will not exceed those in Section 15.5.2.4, 4.4 mrem whole body at the site boundary.

15.5.3.1 Collision of EVTM with Control Rod Drive Mechanisms

The EVTM is a railway-mounted cask type fuel transfer machine which moves during refueling to within about one foot of the CRDMs. This only occurs when the reactor is shut down and all control rods are fully inserted and disconnected from the CRDMs. The rail structure includes anti-lift off and anti-over turning restraints. In addition to travel limit switches, there are fixed mechanical stops on the rails to limit the EVTM travel. If, in spite of procedural limits and all electrical controls and mechanical limits, the EVTM collides with the CRDMs, it could cause a release of cover gas through the CRDM seals; at the limit, the consequences would not be worse than those in Section 15.5.2.4, 4.4 mrem whole body at the site boundary.

15.6.1.1 Primary Sodium In-Containment Storage Tank Failure During Maintenance

Analysis of postulated in-containment sodium fires shows that the limiting event with respect to containment building temperature and pressure resistance capability is the postulated failure of a full primary sodium storage tank during maintenance, with the tank cell de-inerted and open to the upper containment volume. The tank is in normally inerted Cell 102A below the operating floor; the cell has a steel liner and its walls are 6-ft thick concrete. The event considered assumes that, with the tank cell de-inerted and interfacing with the atmosphere of the RCB, the maximum volume of 35,000 gallons of sodium is dumped from the tank onto the cell floor with immediate start of pool burning. It is assumed that the sodium coolant contains its maximum concentration of radionuclides. It is calculated that during the 550 hours of combustion, 179,000 lb of sodium is burned. Most of the fire takes place within the first 70 hours, during which the pressure peaks (0.8 psig at 20 hours) and the temperature peaks (wall temperature 122° at 70 hours). A total of 3.4 kg of aerosol containing 2.5 kg sodium is released during the 70 hours of containment overpressure. The estimated maximum 2-hour, site boundary dose to an individual is 0.288 rem to the bone.

15.6.1.2 Failure of the Ex-Vessel Storage Tank Sodium Cooling System During Operation

The EVST has two normal forced convection cooling circuits, used alternately, and one backup natural convection circuit. Each normal circuit is in a cell below grade in the Reactor Service

Building, as is the EVST. The event considered is the break of a cooling circuit at the low point of the pump suction line, maximizing the spill due to siphoning; about 7500 gallons of 475°F sodium would be spilled into the sealed, inerted cell of 14,950 cubic feet volume. A fire lasting about 9 hours ensues, peaking in 3.6 hours at 254°F and 3.8 psig in the cell (within the retention capability of the cell). The analysis assumes maximum radionuclide concentrations in the sodium, the same proportions of radionuclides in the combustion product aerosol, and that all the aerosol is released directly to the environment. The maximum individual dose calculated over 2 hours at the site boundary is 0.713 rem to the bone.

15.6.1.3 Failure of an Ex-Containment Primary Sodium Storage Tank

The two ex-containment primary sodium storage tanks are in a cell on the lowest level of the Steam Generator Building. These tanks will be used to store sodium only if maintenance requires the drainage of more than one PHTS loop or the EVST, etc. When a tank is to be used, the cell is sealed and inerted. The event considered is the spill of all the contents of one tank, 45,000 gallons of 450°F sodium onto the cell floor, where it is held by a 3/8 inch steel catch pan. The analysis assumes the ensuing fire consumes all the oxygen in the cell, that the proportion of radionuclides in the combustion product aerosol is the same as in the end-of-life sodium, and that all the aerosol is released to the environment without fallout or other depletion. The maximum individual dose calculated over 2 hours at the site boundary is 3.19 rem to the bone. The cell is designed to withstand the temperatures and pressures generated by this event.

15.6.1.4 Primary Heat Transport System Piping Leaks

The PHTS is entirely within the Reactor Containment Building, either in cells or in the reactor cavity. A Cell-Design-Basis Leak was chosen, with flow equivalent to the flow from a sharp edged circular orifice with an area equal to one half the pipe diameter times one half the pipe wall thickness. The maximum leak rates for the PHTS cell and the reactor cavity are 947 and 847 gallons per minute respectively. The design requirement is that there should be no damage to the cell that could impair the safety functions of other equipment. The leak in the PHTS cell is taken as 947 gpm for 4.5 minutes (until reactor trip on low sodium level) followed by pump coastdown of one minute and 500 minutes at pony-motor induced flow. The spill is taken as 35,000 gallons, at 1015°F. The leak in the Reactor Cavity is similar, beginning with a flow of 847 gpm and spilling 29,000 gallons of 750°sodium. The larger fire is estimated to burn about 423 pounds of sodium, and all the aerosol is assumed released to the containment. Automatic containment isolation follows detection of radioactivity in the RCB HVAC exhaust. Leakage for the containment is based on 0.5 psig pressure, applying a square-root relationship to the design leak rate of 0.1% of vol/day at 10 psig; bypass leakage is taken as 1% of the containment leakage. No credit was taken for aerosol depletion in the PHTS cell or outside the RCB; depletion inside containment was calculated with the HAA-3 code; a total of 3.6 grams of sodium is released to the atmosphere over a 30-day period. The maximum individual dose calculated over 2 hours at the site boundary is 0.112 mrem to the bone.

15.6.1.5 Intermediate Heat Transport System Pipe Leak

For the IHTS, the design basis leak was also taken to be equivalent to the flow from a sharp edged circular orifice with an area equal to one half the pipe diameter times one half the pipe wall thickness (e.g. for 24 inch pipe, 2.85 square inches). The limiting case is a leak in the 24 inch O.D. main loop hot leg pipe in Cell 226, at the low point in the pipe. It is assumed that the operator takes no action but that the system is tripped by primary-secondary flow mismatch, so that loop flow continues via the pony-motor, until the pump tank is emptied. The total spill is 300,000 pounds of sodium at an average bulk temperature of 800°F. For consequence evaluation, this Design Basis leak was combined with the maximum undetected IHX leak so that primary sodium leaks into the IHTS when its pressure drops; it was assumed that the leak continued at maximum rate for 2 hours and that operator action is taken within 24 hours to break the siphoning of primary sodium to the IHTS. Codes SPCA and SPRAY are used to calculate conditions in the cell, and HAA-3 to calculate the rate of discharge of aerosols to the environment. All aerosol formed is assumed released at ground level. An analysis was made to determine that the released aerosol would not impair the operation of the plant safety-related equipment. It is assumed in the analysis that when aerosols are detected in the Steam Generator Building HVAC exhausts, all SGB inlet and exhaust dampers are closed except for one 40 square foot opening which remains open 5 minutes. It is further assumed that Protected Air Cooled Condenser fan operation is interrupted for 5000 seconds, and that all other HVAC air intake openings throughout

the plant, with the exception of the Emergency Diesel Generator Rooms, are automatically closed within one minute. About 630 pounds of aerosol are released. The maximum 38-day dose to an individual at the boundary of the Low Population Zone is calculated to be 2.14 rem to the bone.

15.7.1.1 Loss of D.C. System

The plant design includes 3 independent battery supported Class 1E D.C. systems. The loss of one will not prevent the functioning of Class 1E D.C. loads because the redundant counterpart of the system that is lost will remain operational.

15.7.1.2 Loss of Instrumentation or Valve Air

All safety related, air-operated valves will be designed to move in a preferred direction (or in certain cases, fail as is) with the loss of air supply; e.g., containment isolation valves will fail closed. The systems supplying compressed air to safety-related valves or instruments will be designed such that a single credible failure will not cause interruption of the air supply. Loss of air supply will not have an adverse effect on safe operation of the plant.

15.7.1.3 IHX Leak

If an IHX leak occurs, in spite of design precautions, the leakage will be into the primary system from the intermediate system which is kept at higher pressure. Temperature-compensated sodium level sensing devices on each system will indicate to the operator any changes due to leakage.

15.7.1.4 Off-Normal Cover Gas Pressure in the Reactor Coolant Boundary

The subject cover gas pressure is maintained at 10±2 inches of water. There is a constant sweep flow into the cover gas spaces and through the shaft seals of the primary pumps. There is a gas pressure equalization line connecting the pumps, reactor vessel and overflow tank. Off-normal pressure could be due to failure of the regulators controlling flow to and from RAPS. Under pressure would be limited to about 1 psia and would have no significant effect. Because the cover gas volume within the primary coolant boundary is about 4500 cubic feet and the make-up rate is limited to about 50 SCFM, a over-pressure buildup will be gradual, allowing the operators to take effective action after the condition is annunciated. Relief valves set at 15 psig will limit the pressure, and CAPS action will prevent hazard fo the public.

15.7.1.15 Off-Normal Cover Gas Pressure in IHTS

The IHTS argon cover gas is taken from the steam generator building argon supply which is at 175 psig; the pressure in the pump and expansion tank cover gas is controlled to 100 psig. Failure of regulation on the supply could increase the pressure to 175 psig which would have no significant effect because the IHTS is designed to take 325 psig. Low cover gas pressure could result in sodium leakage from the primary system into the IHTS, with insignificant radiological effects since operator action would correct the pressure differential in a short time.

15.7.2.1 Inadvertent Release of Oil Through Pump Seal (PHTS)

The primary sodium pump has oil-lubricated bearings and seals above the sodium-containing pump tank. For oil leakage to enter the sodium, it must get past the system which collects leakage oil in an oversized reservoir, and also past a shaft oil slinger and second collection reservoir. The oil supply is a fixed capacity (about 6 gallons),smaller than either of the reservoirs, so that overflowing the collection reservoirs can be accomplished only by manual addition of excess oil. If oil should, however, enter the sodium, analysis indicates that the carbon particles in the sodium will be negligible, that the dissolved hydrogen will cause plugging temperatures low enough to have a good margin of safety, and that reactivity effects will be less than 1¢.

15.7.2.2 Inadvertent Release of Oil Through the Pump Seal into Sodium (IHTS)

The release of oil to the IHTS sodium from the pump oil bearing requires the failure of multiple barriers, as with the PHTS pump oil. An undetected loss of the entire seal oil supply to the IHTS sodium would have consequences no more severe than those of release of pump oil seal to the PHTS, or of intermediate loop pipe break or of intermediate loop pump seizure.

15.7.2.3 Generator Breaker Failure to Open at Turbine Trip

In the event of a turbine trip, the turbine trip logic signals a generator load break switch opening and, independently, a generator field breaker opening. Failure of the generator load break switch to open causes loss of the Preferred AC Power Supply, in which

case the Normal AC Distribution System and the Safety-Related AC Distribution System are transferred automatically to one of the Reserve Transformers and thus to the other offsite power supply. Thus consequences are negligible.

15.7.2.4 Rupture in RAPS Cold Box

The Radioactive Argon Processing System cold box contains the cryogenic still which separates krypton and xenon from the reactor argon cover gas stream. It is assumed that the reactor has been operating a long time with 1% failed fuel and that the cryostill has not been unloaded to the noble gas storage vessel for one year, and thus the cryostill contains a maximum inventory of radioactivity. Assuming a cryostill rupture, the cold box cell H&V radiation monitor will signal an alarm, close off the cold box influent and effluent lines and open the bypass line. The liquid nitrogen will be valved off on a cell high-pressure signal. It is assumed that the RCB refueling door is open and that the released gas vents out of the H&V exhausts, without retention in the cold box cell. The maximum individual dose at the site boundary in 2 hours is 3.0 rem whole body.

15.7.2.5 Liquid Radwaste System Failure (Leak or Rupture)

The liquid radwaste collection and processing system is located in a non-hardened building attached to the Reactor Service Building. The highest activity streams are processed below grade in concrete cells coated and sealed to prevent leakage and

facilitate decontamination; each has a sump and sump pump to transfer spills. Each cell is closed during operation. One of the two 20,000 gallon tanks when full would contain the largest inventory of radionuclides in the system. If failed, the contents would spread over the floor and drain into the sump. The operator, alerted by the low level indicator in the tank and the level indicator of the sump, would pump the sump contents to another tank. The most volatile radionuclide in the liquid waste is tritium; other radionuclides are primarily in salts formed from cleaning sodium-contaminated components with water. For conservatism, the spill is evaluated giving no credit to the floor sealing, drains or operator action, ie the spill is released to ground water, flows to the Clinch River and is diluted therein and available for intake at the potable water facility for the K-25 plant. Maximum dose to an individual exposed to the liquid (including drinking) at the site boundary is estimated to be 0.078 rem to the thyroid. Release of 10% of the tritium to the air would give an estimated whole body inhalation dose to an individual at the site boundary of 3.7 microrem in two hours.

15.7.2.6 Failure (Leak or Rupture) in the EVST NaK System

There are 3 NaK systems for cooling the EVST sodium, 2 active and one natural-draft standby. Each involves an Na/NaK heat exchanger outside the Ex-Vessel Storage Tank and a loop with a NaK/air heat exchanger using outside air for a heat sink. The NaK systems operate at about 100 psig and about 350°F and are all welded construction. Leaks of NaK in the inerted cells

containing EVS components will have smaller impacts than those involving EVS sodium because the NaK systems have smaller liquid volumes at lower temperatures and are non-radioactive. Leaks in either the air atmosphere cells or the heat exchangers will result in localized fires handled by catch pans, sealing of the volume and extinguishing by the fire protection system or by nitrogen flooding. Consequences to the public consist of the release to the air of minor amounts of NaK combustion products.

15.7.2.7 Leakage from Sodium Cold Traps

In the CRBRP auxiliary systems there are the primary sodium cold traps (cooled by liquid NaK in a jacket welded to the outside of the trap), the EVS sodium cold traps (cooled with nitrogen), and the intermediate sodium cold traps (air cooled). All cold traps collect oxide impurities, tritium and other radioactive materials as precipitates in their crystallizer component. A leak between the NaK and Na in a primary system cold trap will leak NaK into the primary sodium; loss of NaK inventory will signal the failure. A leak to cooling gas or to cell atmosphere will be signaled by leak detectors at the coolant outlet or in the cell; the trap will be isolated and replaced. A leak in any of the 3 types of cold traps will have negligible effect on either reactor safety or public safety. When a cold trap is to be removed, it is isolated from the processing system and the sodium allowed to freeze. The trap is cut out and the ends capped and seal-welded. Once removed, all traps remain frozen without supplementary cooling. Calculated consequences of a Design Basis Cold Trap Fire (within the RCB with RCB leakage at 0.032% per day) indicate a maximum individual dose for 2 hours at the site boundary of 1.02 mrem to the bone.

15.7.2.8 Rupture in RAPS Noble Gas Storage Vessel Cell

The RAPS noble gas storage vessel normally contains radioactive gas which is unloaded annually from the RAPS cryostill. The gas is mainly argon but also includes krypton and xenon; both stable and radioactive isotopes are present. The gas is bled slowly from the vessel into the Cell Atmosphere Processing System so that its pressure normally decreases over the year. The event analysis assumes that shortly after cryostill unloading, it is again unloaded so that the storage vessel contains two charges and is at maximum pressure when it ruptures. A radiation monitor initiates a signal closing off the cell vent line to CAPS. It is assumed that the cell leaks the gases quickly into the RCB, that the RCB refueling door is open, and that the released gases vent out through the H&V exhausts. The estimated maximum dose to an individual at the site boundary for 2 hours is 3.0 rem whole body.

15.7.2.9 Rupture in the CAPS Cold Box

The Cell Atmosphere Processing System cold box contains two charcoal delay beds in series, which absorb xenon and krypton from the process stream before it is discharged to H&V. The beds are cryogenically cooled by injecting liquid nitrogen into the influent stream. The event analysis assumes a rupture of the charcoal delay beds during refueling when the bed inventory of radionuclides is greatest. Normally the cell H&V radiation monitor would initiate an alarm and closing of the H&V vent, the tritium water removal drain, and process gas flow. High cell

pressure would trip off the liquid nitrogen flow. The event analysis assumes all the radioactivity is immediately released from the RSB, leading to an estimated maximum 2-hour site boundary individual dose of 0.14 rem whole body.

15.7.3.1 Leak in a Core Component Pot

A Core Component Pot is a long stainless steel thimble used to hold a fuel assembly or control assembly in transferring it from the reactor vessel to the EVST or reverse. A CCP contains about 22 gallons of sodium in addition to the core component; the plant may have 600 or more CCPs. A CCP could fail due to defective manufacture or to accidental damage or by corrosion. The limiting event analyzed is undetected loss of sodium from the CCP immediately before transporting a 20 KW-decay-heat fuel assembly in the CCP in the Ex-Vessel Transfer Machine. Normally, the time between a CCP emerging from the reactor sodium surface and submerging beneath the EVST sodium surface is 56 minutes. The event analysis assumes a fuel unloading procedure unperturbed by low-load signal from the grapple load cell, or by the monitor signalling high radiation levels in the EVTM. After 10 minutes, the fuel would heat to cladding temperatures of 1500°F and random clad failure could release some fission gas; the EVTM has a 20-KW capacity gas flow cooling system but the fuel would be insulated by the still gas between it and the CCP. After about 17 minutes, clad begins to melt; after 30 minutes it is melting on all rods. After about an hour, the fuel duct reaches the melting point locally. Even after 1.5 hours, the CCP (at 1900°F) would maintain its integrity

(if it had sufficient integrity initially).* Is is assumed that all the volatile fission products are released to the inside of the EVTM (if they are molten or vapor at 3500°F). The release is assumed to be by diffusion through the EVTM seals (calculated to reach 260°F) to the RSB/RCB and to involve only radionuclides mobile at 260°F. The results are said to be represented by those of Section 15.5.2.3, in which the maximum 2-hour site boundary dose to an individual is estimated at 1.89 rem to the thyroid.

15.7.3.2 Spent Fuel Shipping Cask Drop from Maximum Possible Height

The maximum possible height for a Spent Fuel Shipping Cask drop is 72 feet from the operating floor of the RSB to the bottom of the cask handling shaft. Due to design features (such as redundant crane rigging) and operational precautions, dropping a cask is considered extremely unlikely. Cask design is such that the drop is not expected to fail the seals. It is assumed that the cask is loaded with fuel cooled only 80 days (vs 100 minimum) such that it is at its capacity heat load, 26 KW. It is assumed that all fuel rods on 6 fuel assemblies fail, releasing their volatiles in the cask inner canister and that they leak out through the cask inner seals at the maximum rate of 6×10^{-5} Std. cm^3/sec, adjusted for the canister pressure; no credit is taken for the cask outer seals nor for holdup in the RSB/RCB. The maximum estimated dose to an offsite individual is 0.541 mrem to the thyroid, at the LPZ over 30 days.

* Editorial comment.

15.7.3.3 Maximum Possible Conventional Fires, Flood, Storms or Minimum River Level

Fire: The plant fire protection system is designed to provide adequate fire protection, detection and signalling in areas where a hazard may exist; the nearest tree line is kept 100 yards away to minimize forest fire impact.

Flood: Plant grade is at the 815 foot elevation while maximum flood is estimated at 809.2 feet including combined ½ Probable Maximum Flood,OBE with dam failures and wave runup.

Storms: Drainage facilities for safety-related structures are designed for a maximum 1-hour rain of 14 inches and an 8-hour depth of 29.5 inches; maximum recorded is 7.75 inches in 24 hours. Roof design is for 40 inches of snow, compared to a maximum monthly snow fall of 21 inches. Wind design for safety-related structures is for 90 mph at 30 feet above grade, compared to a peak recorded gust of 59 mph at Oak Ridge. Tornado design is for a maximum velocity of 360 mph.

Minimum River Level: The minimum river level is taken as 735 feet elevation, and the intake structure is 5.5 feet lower.

15.7.3.4 Failure of Plug Seals and Annuli

A transient causing excessively low cover gas pressure at the reactor head, followed by a return to normal, could displace the liquid in the plug annulus dip seals enough to allow cover gas under the seal blade and up into the inner buffer annulus between the dip seal and the inflatable seals. This could increase the rate of leakage of radioisotopes into the head access area; factors against this being significant are 1) there are

2 outer seals with a buffer space between them and these seals have more than adequate pressure resistance, 2) expected leak rates cause a small fraction of the allowable concentrations in the head access area, and 3) a radiation monitor in the head access area can signal any need for evacuation and isolation of containment.

15.7.3.5 Fuel Rod Leakage Combined with IHX and Steam Generator Leakage

This event assumes undetected IHX leakage in the same loop with a steam generator leak. Normally, pressure differentials cause leakage to flow from the steam generator into the IHTS and from the IHTS into the PHTS, thus preventing outward migration of radioactivity. Plant procedures are to maintain these pressure differentials or to drain loops; only unlikely events would reverse the pressure differentials and only for short time periods, so that risks of radiation exposure from such leakage are minimal.

15.7.3.6 Sodium Interaction with Chilled Water

For an interaction between sodium and water of the Chilled Water Systems to occur, two pipe failures and a third boundary must fail simultaneously. In recirculating gas fan cooler units and HVAC coolers serving areas containing sodium piping or equipment, water leakage will trigger redundant moisture or leakage detectors, isolating the unit flows and opening drain valves to remove the water. If automatic response fails, the design permits 2 hours for operator action before water enters areas containing sodium piping. In the RCB and RSB, floor drain system leak detectors will signal leakage, allowing isolation of the affected water line. Upon

confirmation of a sodium leak in area served by a recirculating gas cooling system, the water and gas lines to the cooler serving the area will be valved off.

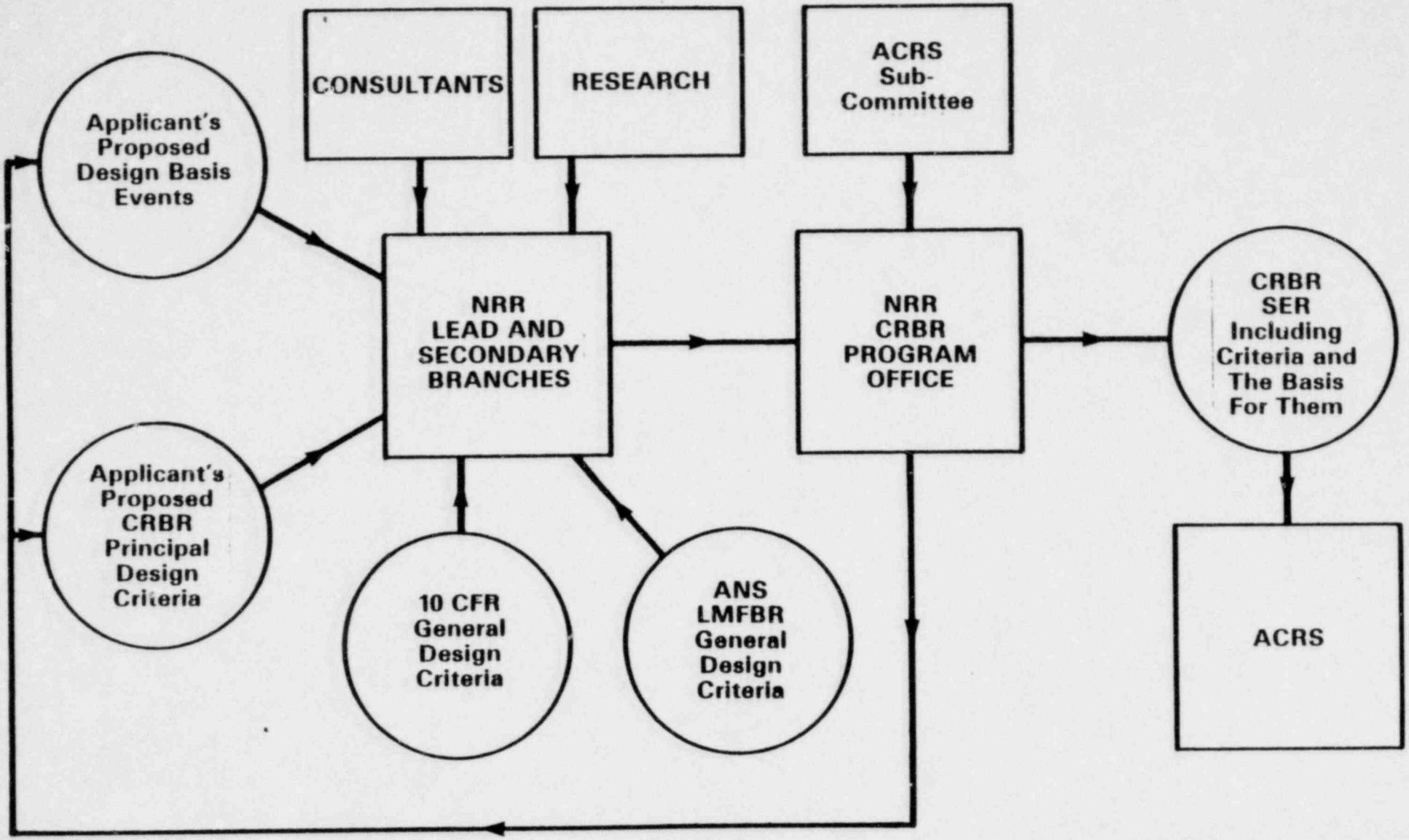
15.7.3.7 Sodium-Water Reaction in Large Component Cleaning Vessel

The Large Component Cleaning Vessel (LCCV) is located in the Large Component Cleaning Cell in the lower part of the Reactor Containment Building. The event considered is a runaway reaction of sodium with water in the LCCV. Under normal operations, residual sodium or NaK on a component is removed by a controlled reaction with water vapor at 150°F in a nitrogen purged atmosphere; the reaction is controlled by regulation of the water vapor concentration with the aim of maintaining hydrogen concentrations below 1 percent. A runaway reaction could occur by flooding the LCCV with water before the sodium was reacted with steam. The analysis considers cleaning the Intermediate Rotating Plug (a once-in-30-years task), and reacting water with 200 lb. of sodium on the bottom plate of the IRP; the resulting pressure in the LCCV is between 40 and 85 psig whereas it is designed for a static rupture pressure of 298 psig. The event assumes release of 100% of the radioactivity in the 200 lb. of sodium, decayed for 10 days. Such a release would isolate the RCB and the effluent would pass through the filter system enroute to the environment, with decontamination factors of 20 for iodine and 100 for particulates. The estimated maximum individual dose at the site boundary is 50 mrem to the thyroid.

PRINCIPAL AND GENERAL DESIGN CRITERIA

- Related to Events or Circumstances Which Could Affect Safety Functions
- Selected, Justified, and Evaluated for Feasibility in the CP Review
- Demonstrated to Have Been Implemented in the OL Review

MODEL FOR DEVELOPMENT OF CRBR PDC



CURRENT CRBR PRINCIPAL DESIGN CRITERIA

38	IDENTICAL TO 10 CFR 50 APP. A	(68%)
10	SIMILAR TO 10 CFR 50 APP. A WITH SLIGHT VARIATION	(18%)
8	WITH NO 10 CFR 50 APP. A COUNTERPART	(14%)
56*		

* THERE ARE 9 10 CFR 50 APP. A CRITERIA FOR WHICH NO COMPARABLE CRBR PDC EXIST

General Design Criteria of 10 CFR 50 - Appendix A not included as CRBR principal design criteria.

10 CFR Part 50, Appendix A

Criterion 28	Reactivity limits
Criterion 29	Protection against anticipated operational occurrences
Criterion 33	Reactor Coolant Makeup
Criterion 35	Emergency Core Cooling
Criterion 36	Inspection of emergency core cooling system
Criterion 37	Testing of emergency core cooling system
Criterion 38	Containment heat removal
Criterion 39	Inspection of containment heat removal system
Criterion 40	Testing of containment heat removal system

CRBR Design Criteria for which no comparable GDC in 10 CFR 50 - Appendix A exists

CRBR Design Criterion	
Criterion 4	Protection Against Sodium Reactions
Criterion 7	Sodium Heating Systems
Criterion 26	Heat Transport System Design
Criterion 27	Assurance of Adequate Reactor Coolant Inventory
Criterion 31	Intermediate Coolant System
Criterion 34	Reactor and Intermediate Coolant and Cover Gas Purity Control
Criterion 36	Inspection of Reactor Residual Heat Extraction System
Criterion 37	Testing of Reactor Residual Heat Extraction System

EXAMPLES OF RELATION BETWEEN GDCs AND LWR DESIGN BASIS ACCIDENTS*

(1) Rod Ejection and Rod Drop Accidents	GDC 28 (Reactivity Limits)
(2) Major Loss of Coolant Accidents	GDC 35, 36, 37, (Requirements on ECCS)
(3) Loss of Coolant Accidents and Steamline Breaks	GDC 38, 39, 40 (Requirements on Containment Cooling Systems)

* Note that each example corresponds to an accident not currently in the CRBR DBA spectrum and consequently for which no CRBR PDC has been proposed.

ACTIVITIES RELATED TO RESOLUTION OF CRBR DBA SPECTRUM

- PSAR Chapter 15 Review
- Accident Delineation
- Front-end PRA Review

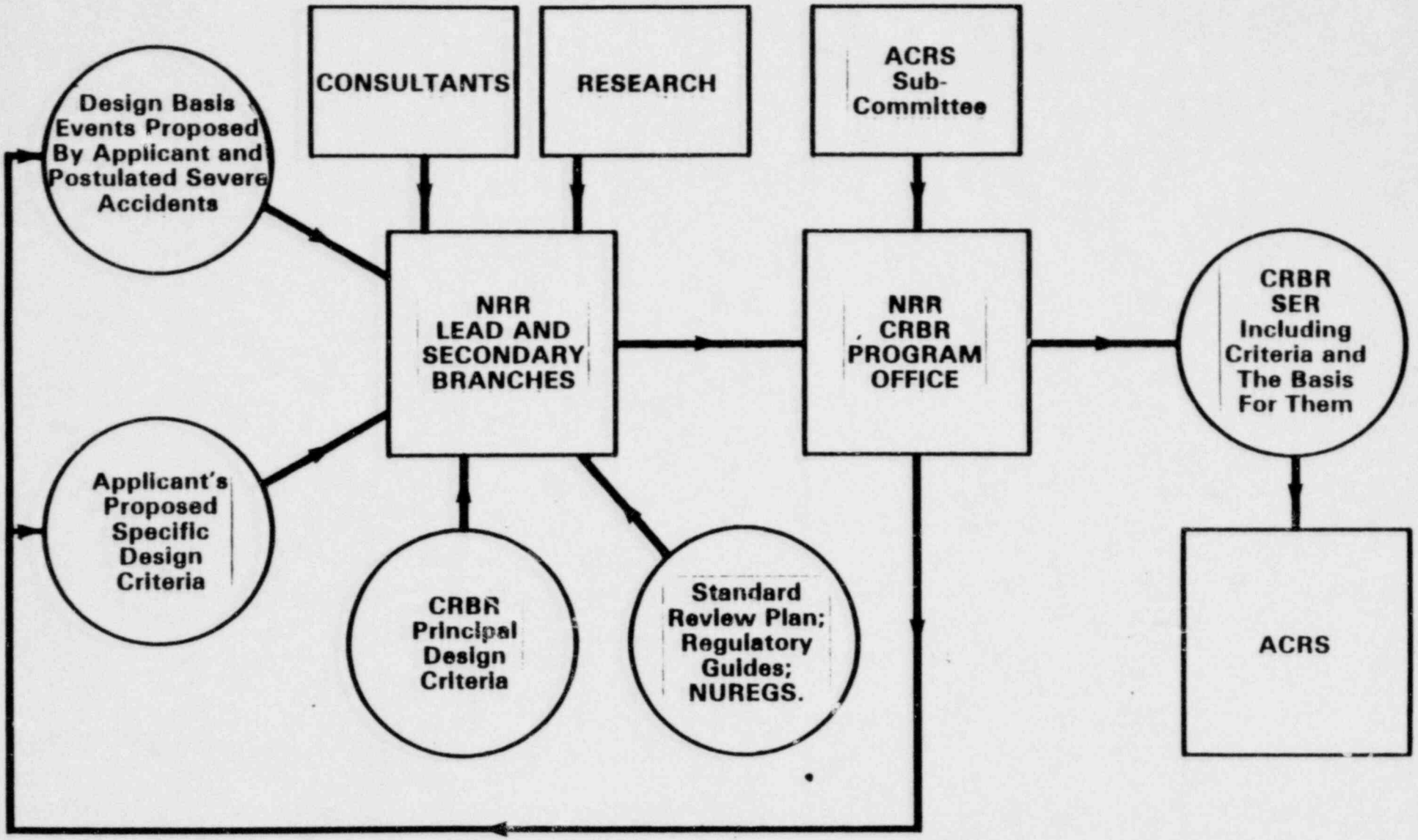
SCHEDULE FOR RESOLUTION OF SPECIFIC ISSUES

- o POTENTIAL FOR LOSS OF COOLANT
  - Hot Leg Pipe Break Criteria (June 82)
  - Cold Leg Leak Detection and Inspection Evaluation (Oct. 82)
  - Analyses of Pipe Breaks (Oct. 82)
  - Final Integrated Evaluation (Dec. 82)
- o POTENTIAL FOR OVER-HEATING CONTAINMENT
  - PSAR Chapter 15.6 Evaluation (July 82)
  - Accident Delineation Report (Oct. 82)
  - Final Evaluation (Dec. 82)
- o POTENTIAL FOR LARGE RAPID REACTIVITY INSERTIONS
  - PSAR Chapter 15.2 Evaluation (Oct. 82)
  - Functional Design of Reactivity Control System Review (Oct. 82)
  - Accident Delineation Report (Oct. 82)
  - Final Evaluation (Dec. 82)

CRBR DESIGN CRITERIA

- PROCESS FOR CRITERIA DEVELOPMENT
- PRINCIPAL DESIGN CRITERIA
- SPECIFIC ACCEPTANCE CRITERIA
- DESIGN BASIS EVENTS
- SEVERE ACCIDENTS BEYOND THE DESIGN BASIS

MODEL FOR CRBR REVIEW



EVOLUTION OF CRBR PRINCIPLE DESIGN CRITERIA

10 CFR SECTION 50.34 REQUIRES PRINCIPLE DESIGN CRITERIA TO BE INCLUDED IN PSAR

10 CFR 50 APPENDIX A PROVIDES GUIDANCE IN ESTABLISHING PRINCIPLE DESIGN CRITERIA FOR NUCLEAR POWER PLANTS OTHER THAN LWRS

ANS-24 ESTABLISHED IN 1963 (BECAME ANS-54) ISSUED DRAFT CRITERIA

NRC STAFF CONSIDERED 10 CFR 50 APPENDIX A AND ANS-54.1 DRAFT CRITERIA AND ISSUED "INTERIM GENERAL DESIGN CRITERIA FOR THE CLINCH RIVER BREEDER REACTOR NUCLEAR POWER PLANT" IN JULY 1974

APPLICANT ADDRESSED INTERIM CRITERIA IN PSAR DOCKETED IN APRIL 1975

STAFF ISSUED PSAR CRITERIA RESULTS AND PRESENTS CRBR DESIGN CRITERIA JAN. 1976

CRBR DESIGN CRITERIA GIVEN TO ACRS IN JAN. 1976

APPLICANT ADOPTS STAFF RPOPOSED CRBR DESIGN CRITERIA JUNE 1976

NRC ISSUED "CRBR PLANT DESIGN CRITERIA" AS A PART OF THE SITE SUITABILITY REPORT IN MARCH 1977

ANS-54.1 ISSUED "GENERAL SAFETY DESIGN CRITERIA FOR LOOP-TYPE LMFBR NUCLEAR POWER PLANT" IN NOVEMBER 1981

CRBR SAFETY PHILOSOPHY

MAY 6, 1976 DENISE TO CAFFEY LETTER STATED "CRBR SHOULD ACHIEVE A LEVEL OF SAFETY COMPARABLE TO CURRENT GENERATION LIGHT WATER PLANTS, ACCORDING TO ALL CURRENT CRITERIA FOR EVALUATION, AND THAT THE DESIGN APPROACHES TO ACCOMPLISH THE REQUIRED LEVEL OF SAFETY (SHOULD) BE SIMILAR OR ANALOGOUS TO LWR PRACTICE"

ATTACHMENT