UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF TESTIMONY REGARDING
ANALYSES OF CORE DISRUPTIVE ACCIDENTS

Q1. Please state your names and affiliations.

A1. My name is Cardis L. Allen. I am employed by the U.S. Nuclear Regulatory Commission as a Senior Reactor Engineer in the Clinch River Breeder Reactor Program Office, in the Office of Nuclear Reactor Regulation. My involvement in the Clinch River Breeder Reactor (CRBR) review is the coordination of the review of the energetics associated with core disruptive accidents (CDAs).

My name is Larry W. Bell. I am employed by the U.S. Nuclear Regulatory Commission as a Nuclear Engineer in the Accident Evaluation Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. I performed the NRC Staff's review of the CRBR control room habitability and Design Basis Accident analyses as set forth in the Applicants' Preliminary Safety Analysis Report (PSAR), Chapters 6 and 15, respectively. I reviewed the Applicants' responses to questions, conducted independent analyses

of the radiological consequences of DBAs and prepared input for the Staff's evaluation of the control room habitability systems and DBAs as set forth in the CRBR SER. In addition, I performed a CDA analysis, the results of which are set forth in Supplement No. 2 to the CRBR SER, at pp. A.5-1.

My name is Howard B. Holz. I am employed as a Reactor Engineer, Technical Review Branch, CRBR Program Office, in the Office of Nuclear Reactor Regulation. My involvement in the CRBR review has been to review those aspects of core disruptive accidents (CDAs) associated with the response of structures and components to potentially energetic CDAs. For this type of CDA, the structural margins beyond the design basis (SMBDB) have been reviewed.

My name is Lewis G. Hulman. I am Chief of the Accident Evaluation Branch in the Division of Systems Integration, Office of Nuclear Reactor Regulation at the U.S. Nuclear Regulatory Commission.

My name is Dr. John K. Long. I am employed as a Reactor Engineer, Technical Review Branch, CRBR Program Office, in the Office of Nuclear Reactor Regulation. My involvement in the CRBR review has been to review those aspects of core disruptive accidents (CDAs) which occur subsequent to the loss of core geometry.

My name is Dr. Bill Morris. I am employed by the Nuclear Regulatory Commission as Branch Chief, Electrical Engineering Branch in the

Office of Reactor Research. During the construction permit review for the CRBR until March 1983, I supervised the review as Section Leader, Technical Review Section, Clinch River Breeder Reactor Program Office, Office of Nuclear Reactor Regulation, and I participated extensively in the issuance of the Safety Evaluation Report for CRBR (NUREG-0968, March 1983) (SER).

My name is Dr. Jerry J. Swift. I am employed by the U.S. Nuclear Regulatory Commission as a Reactor Engineer, CRBR Program Office in the Office of Nuclear Reactor Regulation. My involvement with the CRBR review has been to coordinate the review of the radioactive source term analyses and the review of the Probabilistic Risk Assessment of CRBR.

My name is Dr. Charles R. Bell. I am employed as the Associate Group Leader of the Safety Analysis Group at the Los Alamos National Laboratory. I am also presently providing consultant services to the Nuclear Regulatory Commission. My involvement with the CRBR review has been as a member of the management group assigned the task of developing the basis for an independent licensing position on the energetics associated with core disruptive accidents.

My name is Thomas A. Butler. I am employed by the Los Alamos National Laboratory as a staff member in the Advanced Engineering Technology Group. I have managed the structural mechanics analysis for the Group's NRC CRBR technical assistance efforts, and am

responsible for providing technical evaluation assessments supporting sections of the SER dealing with the review of the structural design criteria for features provided to mitigate CDAs. I have also had direct supervisory responsibility for the review of the feasibility of the annulus cooling system and the containment vent/cleanup system.

My name is Dr. Edmund T. Rumble, III. I am employed as a Corporate Vice President of Science Applications, Inc. (SAI). Presently, I am providing consultant services to the Nuclear Regulatory Commission. My involvement with the CRBR review has been as a member of an SAI team providing technical assistance to the Office of Nuclear Reactor Regulation on safety matters related to the proposed CRBR.

My name is Dr. David Swanson. I am an independent consultant specializing in the high temperature reactions between metals and other materials and in related chemical engineering problems. I have reviewed the reactions of sodium and core debris with concrete as presented in the Applicants' CRBR analyses and compared them with a wide range of relevant experimental data.

My name is Dr. Theofanis G. Theofanous. I am employed as a Professor of Nuclear Engineering at Purdue University. In addition, I have been providing consultant services to the Nuclear Regulatory Commission. My involvement with the CRBR review has been as Chairman of the Management Group which was formed to develop an

independent NRC position on the level of energetics associated with core disruptive accidents in the CRBR.

Q2. Have you prepared statements of professional qualifications?

A2. (Panel) Yes. Copies are attached to this testimony.

I. INTRODUCTION AND OVERVIEW

Q3. What subject matter does this testimony address?

A3. (Allen, Long) This testimony addresses the adequacy of the Staff's analysis of core disruptive accidents (CDAs) for the CRBR.

Q4. Please summarize the conclusions reached in this testimony with respect to the CDA analyses performed by the Staff.

A4. (Allen, Long) The Staff's testimony will demonstrate that a considerable level of attention has been devoted to the evaluation of CDAs for the CRBR. These efforts have focused on two major areas. The first area is the evaluation of the potential for energetic consequences sufficient to fail the reactor vessel head; the reactor vessel head is the principal barrier preventing early containment challenges from such events. The second area is the evaluation of the longer term capability of the CRBR to accommodate the consequences of less energetic CDAs, i.e., where failure of the reactor vessel head has been ruled out. This area includes the evaluation of the capability to accommodate heat and radioactivity released to various parts of the plant from such events.

It is the Staff's conclusion that energetic CDAs sufficient to fail the reactor vessel head are physically unreasonable and that the consequences of such behavior are not a significant safety concern for the CRBR. Further, it is the Staff's conclusion that the CRBR will be capable of accommodating the effects of a CDA for a period of time sufficient that radiological doses to individuals at the LPZ boundary can be expected to be within 10 C.F.R. Part 100 guidelines.

This testimony summarizes the principal analyses and conclusions reached in the Staff's evaluation of CDAs. More detailed information concerning the Staff's evaluation are reported in Appendix A of the SER, in SER Supplement No. 2 (May 1983), and in supporting documents identified therein.

Q5. How is a CDA defined for the CRBR?

A5. (Allen, Long) For discussion purposes in this testimony we define a CDA to be a core melt accident in which sufficient fuel and clad can be relocated to substantially affect the neutronic (and hence power) behavior of the reactor. Such behavior generally disassembles the core either by energetic or by non-energetic phenomena or by both types of phenomena.

Q6. Why are such events of interest?

A6. (Allen, Morris) Although CDAs are very unlikely, severe radiological consequences could result from severe core damage. To assure that the risk from CDAs is acceptably low the Staff has

evaluated the capability of CRBR to accommodate the energetic loads, thermal loads, and radiological releases from the core. In addition, because of the lack of extensive operating experience with this type of reactor, the Staff believes it prudent to understand the phenomena and potential consequences associated with such events should they occur.

Q7. Are CDAs considered in the design of CRBR?

A7. (Allen, Morris) Yes. However, as explained in the Staff's testimony concerning the design basis accident spectrum, they are not used as design basis accidents.

Q8. Since the CRBR design basis accident spectrum does not include CDAs, please explain in more detail why they are analyzed at all for CRBR?

A8. (Allen, Morris) LMFBR designers have historically included in their design documentation the results of analyses of CDAs. These CDAs were not considered DBAs but were nevertheless analyzed for the purpose of determining the hypothetical consequences of a severe accident at the facility. Physically, the reason CDAs have been analyzed for LMFBRs is that, in an LMFBR, the possible positive coolant void reactivity effect and the higher enrichment of the fuel result in the potential for an accident which might lead to an energetic event. Such behavior is not possible in LWRs due to their low enrichment fuel and the need for neutronic moderation. CDAs could also lead to a non-energetic reactor vessel melt through event; such events for CRBR are also analyzed.

For the CRBR, it is the Staff's position that the likelihood of core disruptive accidents can and must be reduced to a sufficiently low level to justify their exclusion from the design basis accident spectrum. As discussed in the Staff's testimony on the design basis accident spectrum, this is accomplished in the CRBR by application of deterministic criteria to the design. These criteria require sufficient redundancy, diversity and independence of safety systems to make failure of these systems very unlikely. However, because of the difference in the amount of experience between LMFBRs and LWRs, the Staff has required that additional measures be taken to limit consequences and reduce residual risks from potential core disruptive accidents.

Q9. What is the significance of considering CDAs in the design of the CRBR but not classifying them as DBAs?

A9. (Allen, Morris) Bearing in mind that consideration of CDAs is not specifically required by the regulations, such a procedure allows flexibility in the application of criteria and use of conservatisms in the analysis of such unlikely events. It is the Staff's judgement that this flexibility is appropriate considering the low likelihood of CDAs. Further, while CDAs are not expected, the facility is not likely to be restarted if a CDA has occurred. Thus, the acceptance criteria applied to CDA evaluations need not include the capability for restart as is the case for DBAs. For example, deformations in the primary system that would be unacceptable under existing guidelines for DBAs can be tolerated for CDAs.

Q10. Is there a general pattern of progression that all CDAs may be expected to follow?

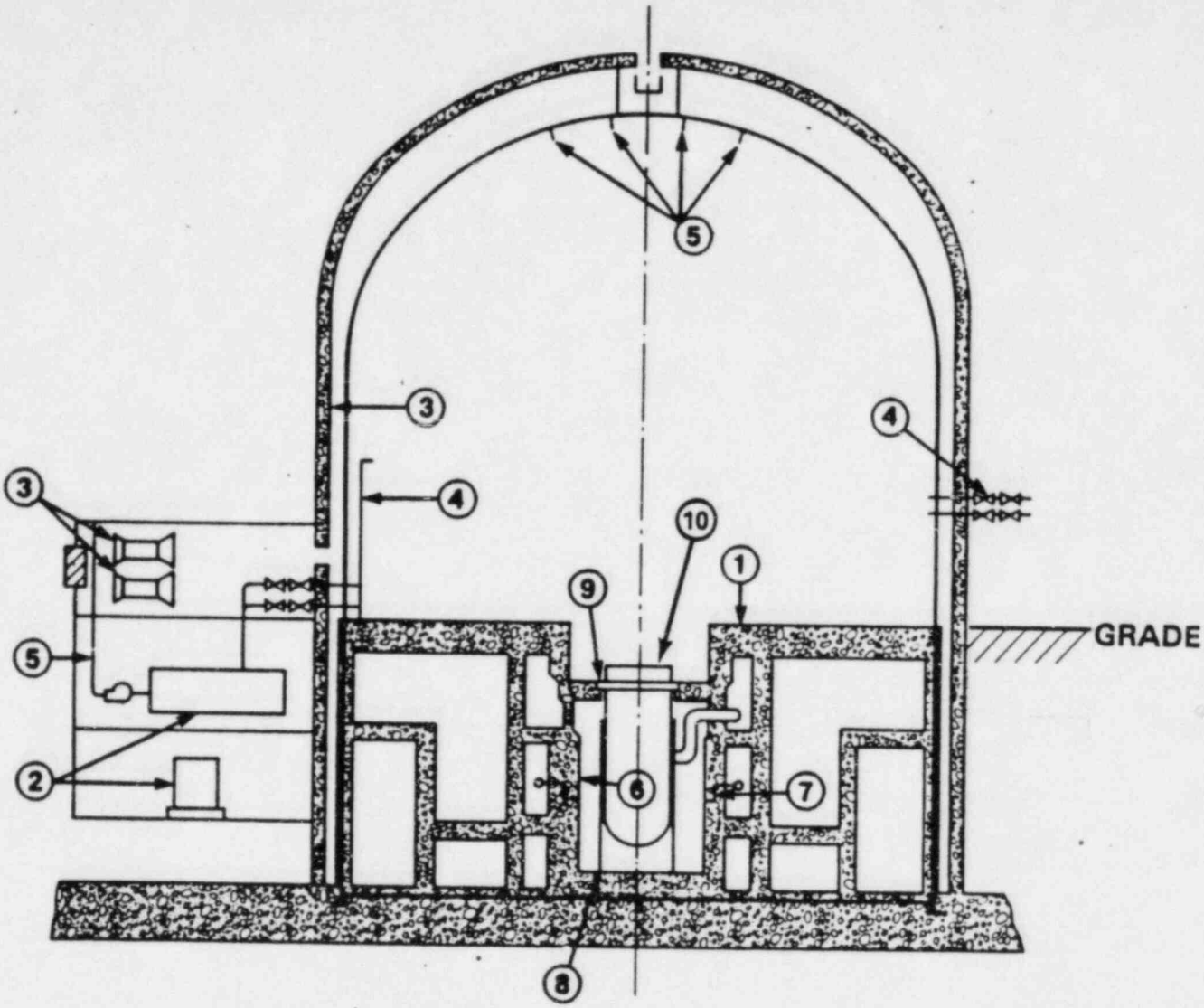
A10. (Allen, Long) Yes. In general, all CDAs progress through various stages of core disruption, from an initating event through clad and fuel melting and relocation (during which power bursts may occur with some related energy release) to a permanently subcritical end state. That end state can result either from an energetic dispersal of the core material (the energetic dispersal can take place within the primary system or, if energetic enough, the material could be dispersed outside the primary system), or from a non-energetic dispersal into the structures surrounding the core. Both dispersal modes could ultimately lead to penetration of the primary system via thermal attack.

Q11. Does this testimony address the "energetics" aspect of CDA behavior as well as that associated with penetration of the reactor vessel by thermal attack?

A11. (Allen, Long) Yes. The Staff's testimony includes information on the entire CDA sequence.

Q12. How were potential CDA initiating mechanisms evaluated?

A12. (Allen, Theofanous, C. Bell, Rumble) Several potential CDA initiators were considered but only those which lead to a gross undercooling or gross overpower condition can produce sufficiently severe temperatures to result in fuel and clad melting. In general, a gross overpower condition can occur from an unprotected (i.e., no



1. OPERATING FLOOR
2. CONTAINMENT CLEANUP SYSTEM
3. ANNULUS COOLING SYSTEM
4. CONTAINMENT VENT AND PURGE SYSTEM
5. INSTRUMENTATION
6. REACTOR CAVITY AND PIPEWAY
7. LINER VENT SYSTEM
8. GUARD VESSEL SUPPORT
9. REACTOR CAVITY TO HEAD ACCESS AREA SEALS
10. REACTOR HEAD

Figure 1 Schematic Showing Major Components Associated with the Reactor Containment Building

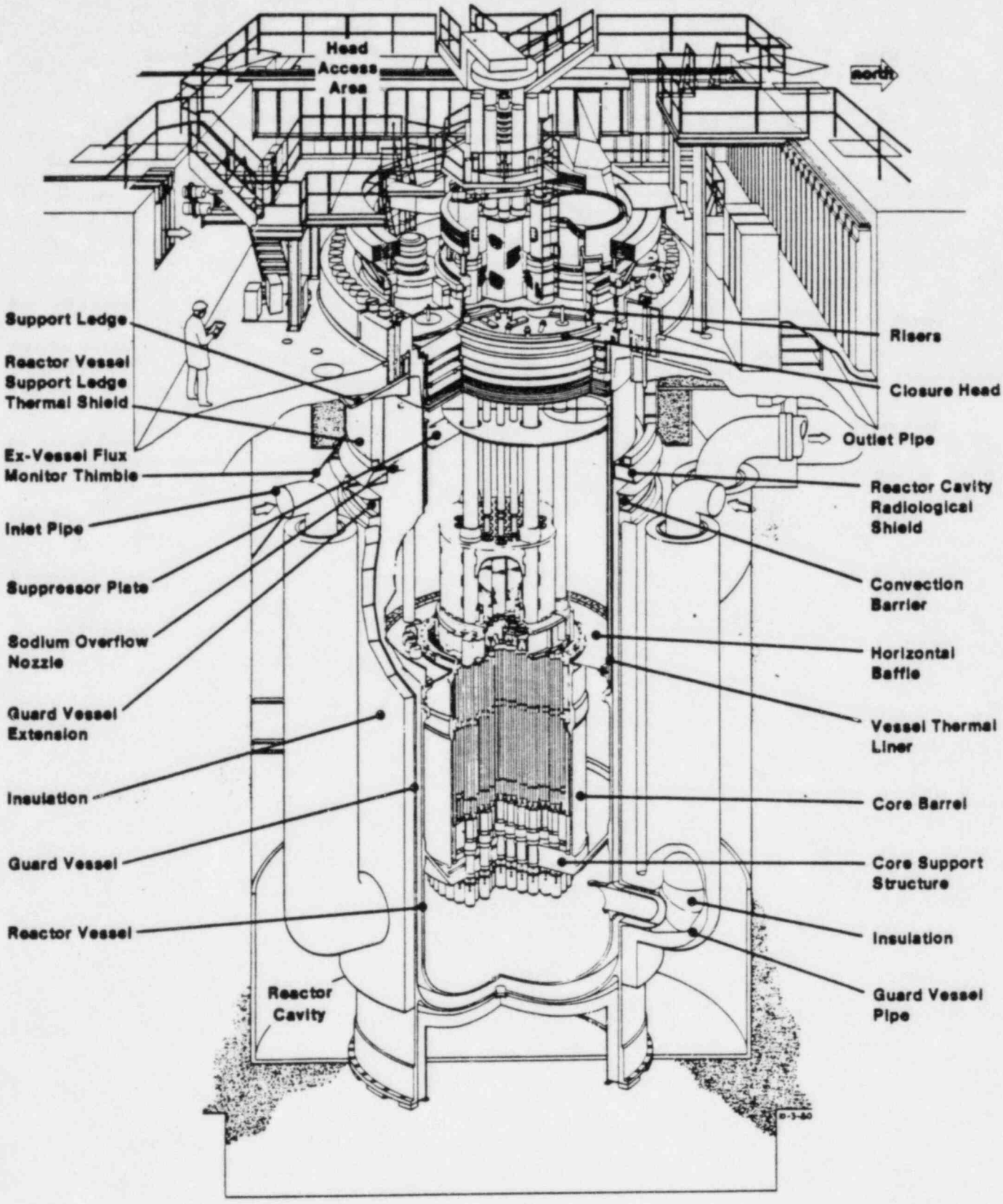


Figure 2 Reactor Enclosure System and Parts of Interfacing Systems.

scram) reactivity insertion event. A gross undercooling condition can occur as a result of an unprotected loss of flow event, or from a loss of heat sink event.

Specific initiating mechanisms were identified to provide limiting cases for these events based on reasonable judgements regarding the likelihood of occurrence of these mechanisms. Bounding mechanisms were selected to represent limiting cases involving the potentially different ranges of phenomenology incurred in the various generic classes, i.e., reactivity insertion, loss-of-flow and loss-of-heat-sink (LOHS) events.

For the reactivity insertion mechanism coupled with an assumed failure to scram, a limit on the reactivity insertion rate from an uncontrolled control rod withdrawal was selected. The basis for selection of this reactivity insertion rate was that significantly higher rates have a diminishingly small likelihood because of the additional independent failures that must be postulated.

For the loss-of-flow mechanism a total loss of power to the coolant pumps was assumed coupled with an assumed failure to scram.

For the LOHS we selected the protected event because it involves a different range of phenomena than is associated with the unprotected loss-of-flow and unprotected TOP events, and is much more likely than the unprotected LOHS event. Our understanding of the phenomena

involved, and our detailed analysis of the protected LOHS and unprotected LOFA, leads us to the conclusion that no unique (larger) energetic circumstances are to be expected from the unprotected LOHS than the other events we have considered in detail.

Q13. Please describe the consequences of the energetic aspects of CDAs?

A13. (Allen) As noted in response to Question 5 above, the core may be dispersed by energetic phenomena. If energetic enough, such events could threaten the integrity of the upper reactor vessel (RV) closure head. This head provides a barrier between the reactor vessel internals (reactor core) and the reactor containment building (RCB) environment. Figure 1 illustrates this point. Figure 2 provides some detailed perspective of the reactor, vessel, head and cavity regions. The operating floor (which is illustrated in Figure 1) together with the head isolates the regions containing primary sodium from the containment environment. If the head should fail, radioactive materials could be released directly from the disrupted core to the RCB environment. These materials would then be available to leak to the atmosphere early in the CDA sequence. In addition, such a failure could challenge the integrity of the containment by sodium fires or missiles. If the head remains intact the disrupted core will be retained within the reactor vessel or the debris will eventually be discharged to the reactor cavity where it will (at least initially) still be isolated from the containment environment.

Q14. What are the major aspects of the evaluation of the energetics associated with CDAs?

A14. (Allen, Theofanous, C. Bell) There are three major aspects. The initiation of CDAs, the production of energetic behavior during CDAs and the consequences of such behavior. We will briefly address each of these in turn here. They are discussed more fully in subsequent parts of this testimony.

Energetic behavior can occur during CDAs because the cores of commercial power-producing LMFBRS are not arranged in their most reactive configurations. Relocations of the core materials can, therefore, produce states of higher reactivity and hence higher power. These relocation processes can only occur following a postulated gross mismatch between power production and cooling capability sufficient to produce coolant voiding and/or clad and fuel melting. The way in which such mismatches are brought about can have an effect on the subsequent events. Thus one major aspect involves the consideration of a proper choice of CDA initiators.

Not all relocations of core materials produce an increase in reactivity nor do such relocations take place at arbitrary rates. Rather they are limited by certain well known physical phenomena. In essence gravity and pressure forces produce motions while the resulting reactivity feedback affects power and thus vapor pressures. A tight coupling between reactivity increasing motions and power level is an inherent characteristic of relocation events.

The second major aspect of our review and evaluation of CDA energetics in CRBR has been to determine physically meaningful upper bounds on the reactivity states and associated power and energy releases against which the structural capability of the primary system may be assessed.

For the reasons explained in answer to Question 13 above, the focus of the assessment of the primary system integrity was the reactor vessel head. The evaluation of the response of the primary system, especially of the reactor vessel head to loads produced by energetic events is the third major aspect of CDA energetics evaluations.

Q15. Was the response of the reactor vessel head to loads generated by a CDA the only dynamic load capability requirement considered for CDAs in the CRBR?

A15. (Allen, Holz, Butler) No. All of the major components connected with the primary coolant system are being considered by the Applicants with regard to their ability to accommodate the dynamic loads produced by CDAs in the CRBR. These components are described in Section 5.2 of CRBRP-3, Volume 1, incorporated by reference in the PSAR, Section 1.6. The Applicants have supplied appropriate criteria and have considered the response of these components to loads developed during a CDA as is described in Sections 5.3 and 5.4 of CRBRP-3, Volume 1. The Staff and its consultants have reviewed the criteria, load requirements, and analytical approaches used by the Applicants and have concluded that, since the load requirements

are conservatively high and the analytical approaches are acceptable, the components can resist loads generated during a CDA. Distortion of components may occur but gross failure is not expected.

Q16. Is an evaluation of the effect of CDA energetics on the primary system (especially the RV closure head) the only necessary consideration in an assessment of CDAs?

A16. (Long) No. Even if the CDA produced no energetic consequences one must consider the ultimate disposition of the core debris. The Applicants have assumed that it is not possible, at this time, to demonstrate that the debris can be cooled and retained within the reactor vessel. The Staff agrees with this judgement. Thus the CDA analysis must consider the consequences of failure of the bottom of the reactor vessel due to thermal attack by the hot core debris.

Q17. Please provide an outline of the course of events that is expected to occur subsequent to the initial in-vessel phase of the CDA.

A17. (Long) The Staff has evaluated in detail CDA sequences in which the primary reactor coolant system loses its integrity as a result of penetration of the bottom head of the reactor vessel by the hot core debris. Because of the total transfer of primary coolant and core debris to the reactor cavity, assumed for this failure location, the consequences of this failure path bound the consequences of other failure locations within the primary system. The basic sequence of events following the loss of primary system integrity involves the draining of the primary coolant system sodium inventory together

with the disrupted core debris into the reactor cavity. The steel liner on the floor of the reactor cavity is assumed to fail, allowing the interaction of sodium with the concrete structure of the cavity. Sodium in the reactor cavity eventually boils away and the debris penetrates into the concrete structure below the reactor vessel. Gas products of these reactions are vented to the containment and, when required, the containment atmosphere can be vented through a wet scrubber cleanup system to the environment. Overheating of the containment structures is prevented by an annulus cooling system. Hydrogen is generated during these processes but is prevented from accumulating to high concentrations by being burned when oxygen is present or by being diluted by containment venting and purging with outside air.

Over the long term, the radioactive solid debris generated from the CDA is expected to be retained within the containment system. Some of this will accumulate within the cleanup system during the venting and purging of the containment atmosphere. Downward penetration of the core debris into the concrete basemat is predicted to stop just short of the lower surface of the basemat. Evaluations were also made for the situation in which complete penetration is assumed to occur; these evaluations are summarized in Section A.4.2 of the CRBR SER.

Q18. What features have been included in the design of the CRBR specifically to mitigate potential consequences of CDAs?

A18. (Long) These features are (1) the annulus cooling systems, (2) the containment cleanup (filtration) system, (3) the reactor cavity vent system, (4) the containment vent and purge system and (5) certain containment instrumentation systems. All of these systems are associated with that part of the CDA sequence which follows penetration of the reactor vessel, and are applicable to mitigation of the consequences involving sodium and core debris interactions with concrete. These features are described in Section 2.1 of CRBRP-3, Volume 2, incorporated by reference in PSAR Section 1.6. The Staff's evaluation of these systems is given in Section A.4 of the SER and is summarized later in this testimony.

Q19. How were the potential consequences of CDAs evaluated?

A19. (Allen, Long) The Applicants provided several documents presenting their analysis of CDAs (see SER Appendix A). The Staff and its consultants have reviewed these analyses and have held numerous meetings with the Applicants on the material presented in those documents. In addition, the Staff has obtained independent assessments in several specific areas from highly qualified experts.

Q20. In what areas did the Staff obtain independent assessments?

A20. (Allen, Long) The Staff obtained independent assesssments on the subjects of: (1) the level of energetics associated with CDAs in the CRBR, (2) the potential consequences of that level of energetics, (3) the phenomena associated with the interaction of sodium and concrete, and (4) the response of the containment

structures (including the role of the annulus cooling and containment cleanup systems) to the consequences associated with CDAs in the CRBR. In addition, the Staff obtained independent audits and analysis of the response of the containment environment (temperature, pressure, aerosol concentrations, and hydrogen concentration) to CDAs, and the radiological consequences associated with CDAs.

Q21. How is the remainder of the Staff's testimony on its evaluation of CDAs presented?

A21. (Allen, Long) The remainder of the Staff's testimony on this subject is presented in three parts. In Part II we address the Staff's evaluation of the potential for energetic behavior during CDAs that could challenge the structural capability of the primary system (especially the RV closure head). In that testimony, we will demonstrate that failure of the reactor vessel closure head due to energetic behavior associated with CDAs is sufficiently unlikely that it need not be considered in the assessment of risk for CRBR. In Part III we describe the Staff's evaluation of the phenomena following failure of the reactor vessel and guard vessel via thermal attack. We will demonstrate that the risks from this scenario are no greater than those associated with LWRs for similar accidents. Finally, in Part IV, we demonstrate that in the event of either an energetic or a non-energetic CDA, not accompanied by containment failure, the radiological consequences will not exceed the dose guidelines of 10 C.F.R. Part 100.

Q22. What do you conclude from the Staff's evaluations of CDAs for CRBR?

A22. (Allen, Long) The Staff's evaluation of the potential consequences of CDAs in the CRBR has been adequate in scope and depth, as is demonstrated in the remainder of the testimony on this subject. The Staff's testimony will show that sufficient attention has been given to core disruptive accidents. Further, the Staff's testimony supports the judgement that the risk from such events in the CRBR is acceptably low.

II. EVALUATION OF THE ENERGETICS ASSOCIATED WITH A CDA

A. Introduction

Q23. What subject matter does Part II of the Staff's testimony address?

A23. (Allen) Part II of this testimony addresses the subject of the Staff's review and independent assessment of the potential for energetic behavior and the consequences of such behavior during CDAs in the CRBR.

Q24. Please summarize the conclusions reached in Part II of this testimony?

A24. (Allen) In Part II of this testimony, we demonstrate that the Staff has given considerable attention to the "energetics" aspect of postulated core disruptive accidents (CDAs) in the CRBR. Further, we show from the results of the Staff's independent assessment that energetic behavior during CDAs is not a significant factor affecting the safety of the CRBR.

Q25. What has been the objective of the Staff's independent assessment of the energetic behavior associated with CDAs in the CRBR?

A25. (Allen) The objective of the assessment is to define the magnitude of the energy releases against which the capability for maintaining the integrity of the primary system, and of the reactor vessel head in particular, should be assessed.

Q26. What are the concerns associated with the magnitude of energetics associated with CDAs?

A26. (Allen) As explained in Part I of this testimony, the level of energetics associated with CDAs is important because, if large enough, it might lead to failure of the reactor vessel closure head. This "head" provides the barrier between the reactor core and the containment building. Failure of this barrier would allow relatively direct communication between the disrupted core and the containment building environment. Sodium fires or missiles associated with head failure might also challenge the integrity of the containment building.

Q27. How will Part II of this testimony be presented?

A27. (Allen, Theofanous, C. Bell) Part II of this testimony will be presented in nine additional subsections, as follows:

B. <u>Evaluation of Potential CDA Initiating Mechanisms</u>

In this subsection we describe the general approach used in selecting the potential CDA initiating mechanisms to be evaluated in detail.

C. Approach Used in the Evaluation

In this subsection we describe the overall approach used in our review and independent analysis.

D. Evaluation of the Relationship Between Ramp Rate and System Loads

Here we describe the relationship between the severity of neutronic events (ramp rates) during CDAs and the resulting loads on the primary system. Based on this relationship and the capability of the system to withstand these loads, we estimate the most severe neutronic events that can be accommodated.

E. Evaluation of the Capability of the Reactor Vessel Head to Accommodate a 75 MJ Impact

In this subsection we describe the Staff's evaluation of the capability of the CRBR to accommodate the head design requirements.

F. Evaluation of Ramp Rates Associated with the LOFA

Having established the range of ramp rates required to reach energy release levels that might challenge the integrity of the primary system, we describe here the analysis of the ramp rates developed during a LOFA.

G. Evaluation of TOP CDAs

We describe here the analysis of the energetic potential developed during TOP CDAs.

H. Evaluation of Protected LOHS CDAs

We describe here the characteristics and energetics potential of the protected LOHS.

I. Conclusions

Here we summarize the conclusions we have drawn from our studies on the energetics associated with CDAs in the CRBR.

J. Answers to Board Questions 11 and 17

B. Evaluation Of Potential CDA Initiating Mechanisms

Q28. How might CDAs occur in the CRBR?

A28. (Allen, Theofanous, C. Bell) In general, a CDA can be initiated only by failure to shutdown the reactor when required or by failure to remove heat when required.

Q29. Could CDAs occur under a variety of conditions?

A29. (Allen, Theofanous, C. Bell) Yes. However, our experience with CDA evaluations has shown that these conditions can be represented by a few generic cases.

Q30. Please explain how these conditions can be represented by a few generic cases?

A30. (Allen, Theofanous, C. Bell) Depending upon whether reactor shutdown has been achieved, core disruption may be initiated at powers ranging from near normal to decay levels. The corresponding heating rates vary by two orders of magnitude and define the first

major classification of CDAs into "unprotected" and "protected" respectively. The unprotected CDA assumes the reactor to be at full power (the reactor scram function is assumed to fail) and initial core disruption may occur due to either an unterminated undercooling or an unterminated overpower condition. Mechanistically, the undercooling event is assumed to be the result of loss-of-coolant flow or the loss-of-heat sink, known as the Loss of Flow Accident (LOFA) and LOHS respectively. The unprotected overpower event is assumed to be due to an uncontrolled reactivity insertion. This is commonly referred to as the Transient Overpower Accident (TOP). In this study a protected CDA is studied by assuming there is a sustained failure to remove decay heat. In general terms, these three accidents exemplify the generic behavior over the whole CDA spectra. Hence, they can be used to adequately characterize the potential for energetics during CDAs.

Q31. Have other potential CDA intiating events and conditions been considered?

A31. (Allen, Theofanous, C. Bell, Rumble) Yes, several potential initiating events and conditions have been considered aside from the ones just mentioned. For example, another class of CDA initiators, that of Fuel Failure Propagation (FFP), has also been identified and extensively studied in the past. All evidence to date indicates that the attainment of whole core disruption through such a mechanism does not occur. This subject is discussed at some length in the Staff's testimony concerning the design basis accident

spectrum. Other failures or combinations of failure events can be postulated such as an unprotected TOP in conjunction with an unprotected loss-of-flow (TOP/LOF), core support failures due to earthquakes beyond the SSE, or loss of piping integrity. Detailed analyses of these events have not been carried out. It is the Staff's judgment that detailed analyses of these events are not warranted since their likelihoods are small compared to those associated with LOFA, LOHS, and TOP events and are diminishingly small in an absolute sense.

C. Approach Used In The Evaluation

Q32. Please describe the approach utilized in your review and independent assessment of CDA energetics?

A32. (Allen) The effort began with a detailed review and evaluation of the Applicants' positions and their technical bases. Following completion of the initial review, a management group (Drs. T.G. Theofanous and C.R. Bell) was formed to direct the efforts of a team (selected from several national laboratories, universities, and consulting firms) to develop an independent assessment of the energetics associated with CDAs in the CRBR. The independent assessment effort (documented in NUREG/CR-3224) was completed in a period of about seven months and contained original elements on one or more of the following aspects: (a) accidents, phenomena, or effects taken into account, (b) analysis methods utilized, and (c) experimental evidence brought to bear.

As a result of its initial review, the Staff and its consultants identified the "eight areas of concern" listed in Section I, Table 2 of NUREG/CR-3224. These concerns were officially transmitted to the Applicants as questions in June 1982. The Applicants' responses were received in August 1982 and were reviewed by the CDA energetics review team which was already working on the comprehensive independent assessment reported in NUREG/CR-3224. The evaluation of these responses by the team is documented in Section 7 of the Compendium to NUREG/CR-3224. Resolution of these eight areas was a part of the team's independent assessment effort. Each has been resolved to the Staff's satisfaction, and the technical bases for resolution are found in various sections of NUREG/CR-3224.

Q33. Is the status of the "eight areas of concern" described in this testimony?

A33. (Allen) Yes. A discussion of this subject is provided in Subsection J of Part II of this testimony.

Q34. Aside from considering the Applicants' evaluations, what methods have you used in your own independent analyses?

A34. (Allen, Theofanous, C. Bell) Our approach consists of realistically following each one of the three generic CDA initiators (identified in subsection B above) through the core disruption phases until termination of neutronic activity which, for purposes of energetics analysis, is taken as the point at which sufficient fuel is removed to assure permanent subcriticality (see Answer 56 below). These CDA

analyses provide an overall framework against which the potential for energetic phenomena is assessed with due regard for the controlling physical processes.

Q35. Did you calculate a specific unique value that defines the magnitude of the energetics generated by a given initiating event?

A35. (Allen, Theofanous, C. Bell) It would be in error to expect that such mechanistic analyses can, at this time, predict uniquely the complete evolution of a postulated CDA from initiation to termination. There is considerable complexity in the underlying physical processes that has not yet been appropriately modeled. We believe that such limitations may alter the overall timing of some events, and may even affect the actual character and sequence of the intermediate states. However, we also believe that these uncertainties can be adequately handled within a properly oriented overall effort. With this in mind we did not attempt to associate a single outcome to a given initiator. Rather, we attempted to establish a "range of phenomenology" consistent with experience and known physical principles. Within this range we searched for energetically-prone circumstances, identified the important mechanisms, and quantified the energy release in a reasonably conservative manner (i.e., avoiding excessive and clearly non-physical conservatisms). Similarly, we identified the termination-favoring phenomena, the important mechanisms involved and the various paths to termination. Based on these results we completed the assessment by considering various combinations of

sequences and their associated likelihoods. For example the various paths that a CDA can take are illustrated for a LOFA in Figure 3. The terms used in this figure to describe the various paths are described later (see the response to Question 56 below). Our assessment calls for judgements in assigning likelihoods to these paths. These judgements are based on our experience and on the insights and knowledge developed in the process of carrying out the evaluations.

Q36. Did you perform a quantitative analysis of CDA phenomena?

A36. (C. Bell) Yes. These analyses were carried out by means of the system codes SAS3D (and to a limited extent the most recent version SAS4A) and SIMMER-II.

Q37. What do the SAS3D and SAS4A computer codes calculate?

A37. (C. Bell) These codes calculate the coupled neutronic, thermal and hydraulic behavior of the reactor during the initial stages of a CDA while the core geometry remains largely intact. Limited fuel and clad relocation models are included in these codes.

Q38. What does the SIMMER-II computer code calculate?

A38. (C. Bell) This code calculates the coupled neutronic and fluid dynamic behavior of the core as it loses its original geometry and substantial material relocation is taking place.

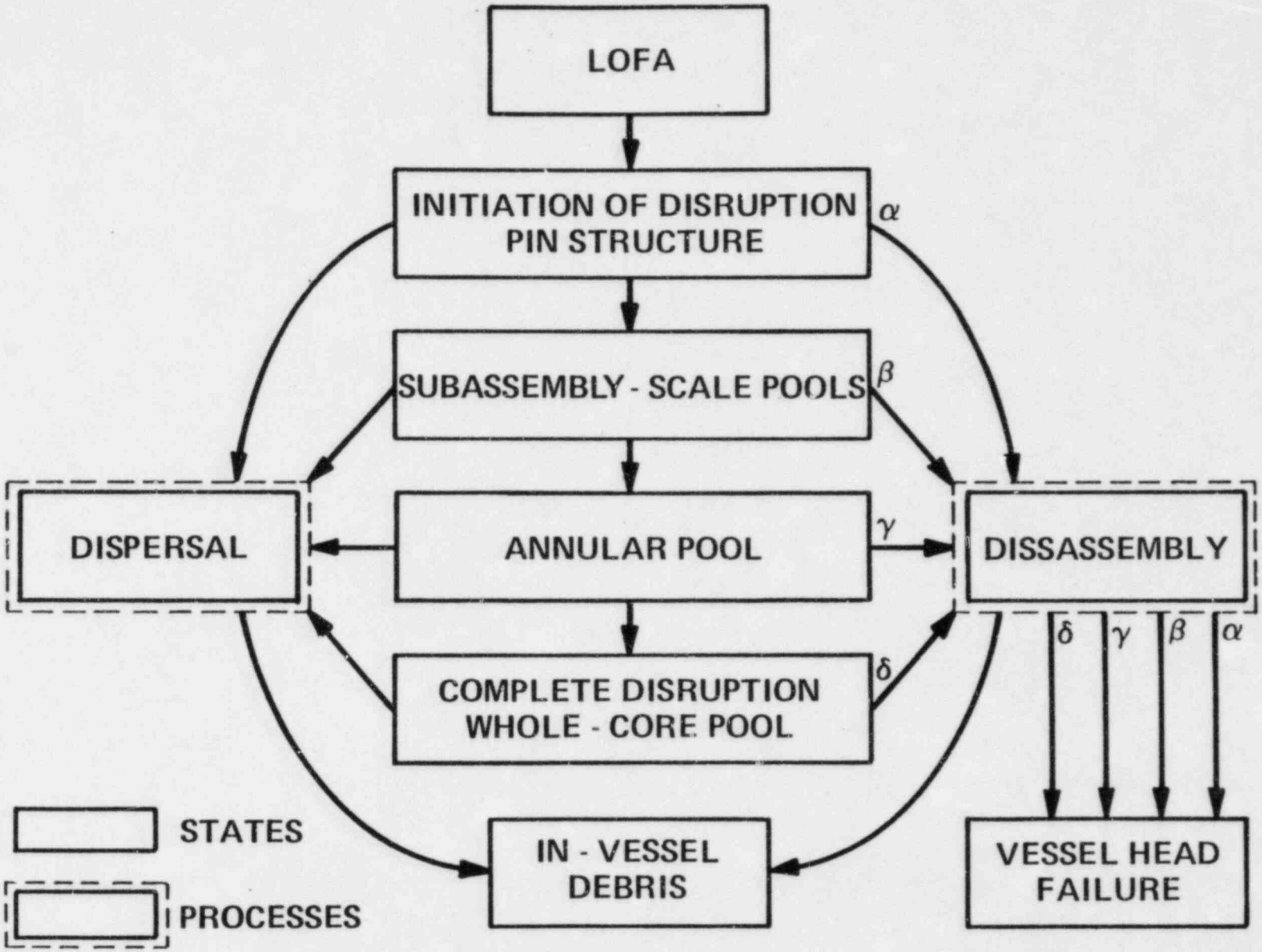


Figure 3. FRAMEWORK FOR CDA ASSESSMENT

Q39. Are the codes used in the Staff's evaluation sufficient by themselves to provide the answers required?

A39. (Allen, Theofanous, C. Bell) No. The codes are not used as "black boxes" that require only geometry and initial condition data, punching a "button" and taking the resulting output as the unique verbatim answer. These codes are used as "integrators" of the technical base and their results are interpreted, guided, scrutinized and/or augmented by employing special purpose analytical techniques, in-pile experimental data and out-of-pile simulant experiments as appropriate. As in all safety studies, the synthesis of experimental data and analysis techniques to produce a quantified basis for the conclusions requires approximations, involves uncertainties and must be appropriately focused. Engineering judgement is utilized to provide overall guidance in this regard. Further details regarding the use of these codes are provided in NUREG/CR-3224 (for example, see p. I-6 of NUREG/CR-3224).

Q40. Were all potential CDA sequences studied in the same detail?

A40. (Allen, Theofanous, C. Bell) As an initial step in our independent assessment effort we made the judgement that among all core disruptive accidents the LOFA should be chosen as the subject of our most detailed considerations. This was based on the following: (a) the LOFA phenomenology spans the range of energetically significant CDA behavior; (b) within the LOFA sequences our specific and significant areas of concern were identified early in our review; and (c) exploratory examination of all other CDAs indicated an

energetically benign behavior as compared to that projected for the LOFA. Furthermore, this emphasis was to reflect the relative complexity of the LOFA sequence as compared to that of the TOP and LOHS accidents rather than any disregard of the unique aspects of these other (TOP and LOHS) CDA initiators. Indeed, these unique aspects were also studied in detail and, with all assessments complete, the choice of this distribution of effort was found appropriate.

D. Evaluation Of The Relationship Between Ramp Rate And System Loads

Q41. How can the significance of resulting levels of energetics appropriately be evaluated?

A41. (Allen, Theofanous, C. Bell) The structural capability of the primary system provides an appropriate perspective against which the damage potential of a given energetic event must be viewed. In particular, the capability of the reactor vessel (RV) closure head to accommodate the energetics associated with CDAs provides a basis yardstick against which these energetics can be measured. As noted earlier (see Answer 13 of this testimony), if the integrity of this head is maintained there will be no significant direct release of radioactive materials and sodium to the containment or missiles directed at the containment early in the CDA. Thus, containment integrity would not be challenged early in the accident and control would be maintained over potential releases from containment.

Although in-vessel cooling of the debris is then possible as we shall see later (see the answer to Question 83) credit is not currently given for this possibility and the CDA is conservatively assumed to lead to failure of the reactor vessel and guard vessel by thermal attack. The significant point is that failure of the reactor vessel due to energetic behavior at locations other than the closure head would also result in discharging the primary coolant and debris to the reactor cavity just as in the case of failure by thermal attack. The subsequent events in this case (that of failure by thermal attack) have been analyzed and the Staff's review is presented in Part III of this testimony.

Thus, the capability of the RV closure head to accommodate loads from energetic behavior is the appropriate measure of the significance of such phenomena.

Q42. What are the energetic loads of interest in this aspect of the analysis?

A42. (Allen, Theofanous, C. Bell) In terms of evaluating the capability of the head to accommodate such events, the item of major interest is the magnitude of kinetic energy imparted to the sodium above the core by fuel vapor expansion that is developed during an energetic CDA progression. This expansion process accelerates the sodium pool through the cover gas space that exists above the sodium surface within the reactor vessel. The expansion thus imparts a certain kinetic energy to the sodium pool. The kinetic energy of the sodium

slug is specified at the time of the sodium slug impact with the underside of the vessel closure head.

Q43. Is the CRBR head to be designed to accommodate such an impact?

A43. (Allen, Theofanous, C. Bell) Yes. The Applicants have stated that the head shall accommodate such an event without failing. The kinetic energy of the sodium slug, for which the CRBR primary system is designed, is generated nonmechanistically by the Applicants. It is determined by the pressure-volume (P-V) curve representing a hypothesized isentropic fuel vapor expansion starting from an initial, highly disrupted core state, and proceeds to an end state defined by a final pressure of one atmosphere. Based on their own estimate of the distribution of the energy associated with such an expansion process (661 MJ) the Applicants have committed to a head design that must accommodate an associated slug impact kinetic energy of 75 megajoules (75 MJ). These terms are discussed below.

Q44. Did the Staff independently estimate the loads on the head?

A44. (Allen, Theofanous, C. Bell) Yes. The Staff independently calculated the fuel vapor expansion process in a conservative but physically realistic manner.

Q45. Please describe the basic concepts involved in the Staff's evaluation of the expansion process.

A45. (Allen, Theofanous, C. Bell) The term "isentropic expansion yield to one atmosphere" is utilized to "characterize" the work <u>potential</u>

from a given high pressure, high temperature fluid. It is used because it is unambiguous in that it represents the maximum possible mechanical energy release in an unconstrained expansion to a final pressure of one atmosphere. In practice this yield, which we have termed Ultimate Work Potential (UWP) to emphasize the idealized conditions under which it is obtained, can be estimated by straightforward thermodynamic (isentropic) expansion and an integration of the resulting P-V curve. The UWP is commonly cited in analyses because it conveys the severity of a calculated energetic event irrespective of the volume actually available for expansion (i.e., irrespective of reactor design).

A closely related concept is that of Impact Work Potential (IWP). This is used to define an ideal (lossless) expansion similar to the expansion above, carried out, however, only to the volume available inside the reactor vessel (i.e., the cover gas volume). This is a more meaningful number than the UWP since it represents an upper bound on the mechanical energy released to the vessel (primarily to the head area).

However, in any real expansion, losses will be present such that the theoretical IWP is never achieved. The actual reductions will depend on the particular geometry and strength of any constraints interfering with the expansion as well as mixing and heat transfer phenomena occurring during the process (see Section II.2 of NUREG/CR-3224).

Q46. What are the major features of the reactor system that are involved in the Staff's evaluation of energy levels required to fail the head?

A46. (Allen, Theofanous, C. Bell) The levels of energetics required to produce significant structural damage in the CRBR were evaluated taking into account an "inner containment" formed by the Core Barrel (CB)/Upper Internal Structure (UIS)/Core Support Structure (CSS) envelope. This configuration is illustrated schematically in Figure 4. In addition, the pressure transmission characteristics of the two phase expanding core medium and other materials found within were also taken into account. These characteristics have important implications on the resulting short term loading of the local structures (Core Barrel and Core Support Structure). This mitigating behavior is the result of a compliant core state (distributed voids) and it must be taken into account particularly since such compliance is one of the crucial prerequisites for highly energetic behavior to start with.

Q47. How was the Staff's evaluation of the energy levels required to fail the head conducted?

A47. (Allen, Theofanous, C. Bell) The analysis of the energy level required to fail the head was conducted in two steps. The first step involved evaluation of the response of the "inner containment" (i.e., the "cage" formed by the CB-UIS-CSS envelope) to the fuel vapor expansion process. If the "cage" boundary fails, the fuel vapor can then expand against the sodium pool above the upper cage

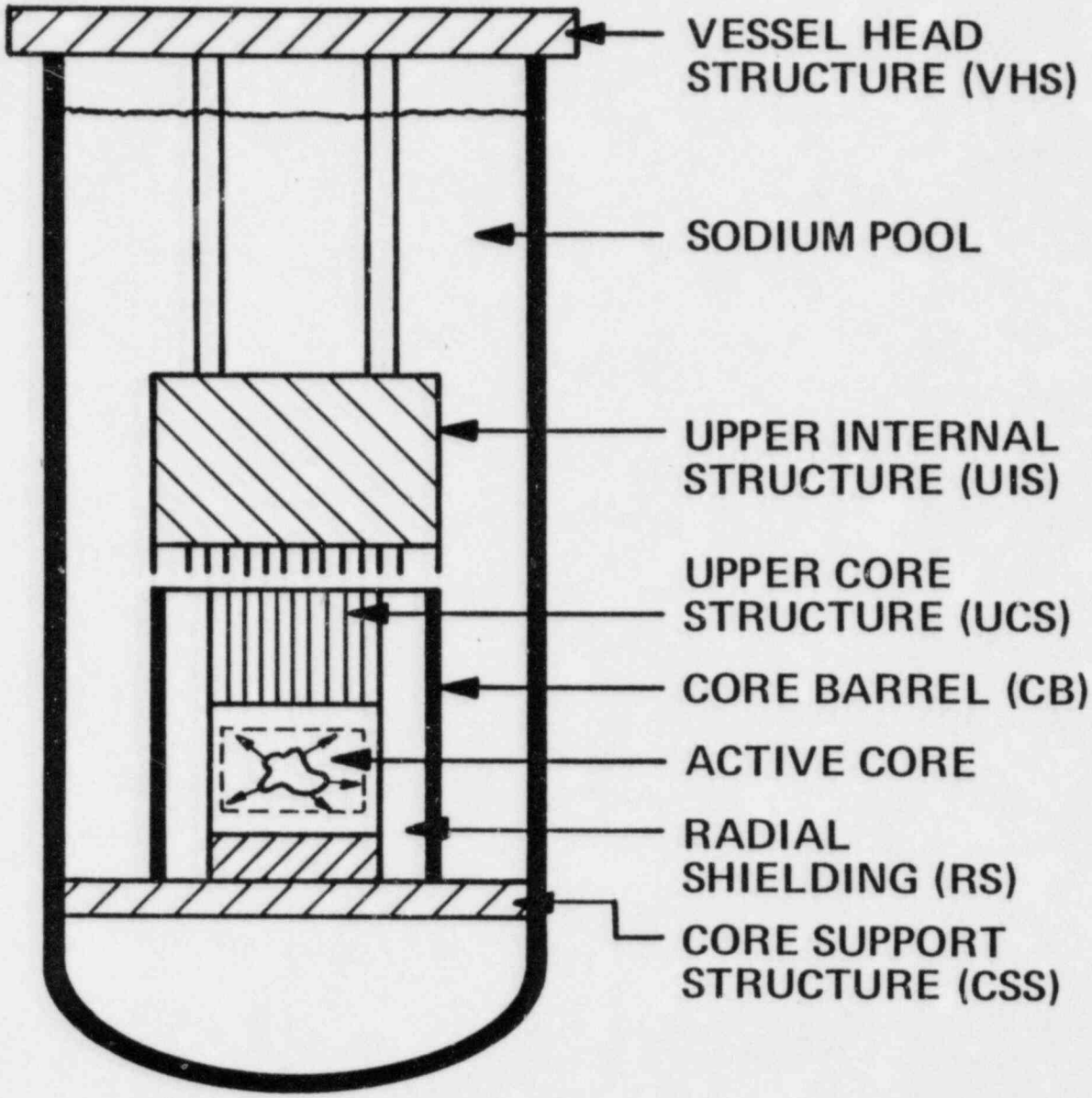


Figure 4. SCHEMATIC OF REACTOR VESSEL AND MAJOR INTERNAL COMPONENTS

boundary (i.e., the UIS). The second step in the evaluation involves the analysis of the expansion into the sodium pool. To assure conservatism in our analysis, all losses expected in a real expansion were not included. The analyses of both steps are described in detail in Section II.2 of NUREG/CR-3224.

Q48. What are the results of your analyses?

A48. (Allen, Theofanous, C. Bell) Our structural analyses indicate a level of energetics on the order of 1130 MJ (isentropic expansion yield to one atmosphere) would be required to breach the inner containment. That is, minimal energetic release against the boundary of the primary system can be expected for any energetics below this level. At still higher levels an upward displacement of the UIS and a longer-term expansion against the sodium pool would take place. Evaluations of the long-term expansion phenomena indicate that an energetic event of nearly twice the above magnitude, approximately 2550 MJ, would be required to produce a slug impact kinetic energy close to the vessel head design capability of 75 MJ committed to by the Applicants.

Q49. How can the energetics levels of 1130 MJ and 2550 MJ, referred to above, be related to reactivity insertion rates generated during a CDA?

A49. (Allen, Theofanous, C. Bell) The 1130 and 2550 MJ energetic levels referred to above correspond to 100 and 200 $/s ramp rate disassemblies respectively occurring in the two-phase regime. The

significance of these ramp rates is discussed below in Part F. As demonstrated there, the expected values for ramp rates during CDAs are significantly lower than these values.

E. Evaluation Of The Capability Of The Reactor Vessel Head To Accommodate A 75 MJ Impact

Q50. Has the Staff evaluated the capacity of the reactor vessel head to withstand a sodium slug impact of 75 MJ?

A50. (Allen, Holz) Yes. The Staff has reviewed the Applicants' analysis and has carried out its own independent assessment of the energy accommodation capability of the present head design.

Q51. What is the result of the Staff's review and independent assessment?

A51. (Allen, Holz, Butler) Our evaluation of the energy partitioning during sodium slug impact and the energy absorption capability of the head indicates that the present head design may not be able to accommodate a 75 MJ sodium slug impact.

Q52. Please describe the mechanism by which the present head design may fail?

A52. (Allen, Holz, Butler) Evidence from the hydrostatic test of the scale model of the vessel head indicates that failure occurs when the intermediate rotating plug disengages from the large rotating plug. The rotating plugs are the primary load carrying components of the vessel head (see Figure 2 in this testimony). The disengagement has been determined to occur principally because of a

kinematic condition that exists at the interface of the rotating plugs. As the head deflects upward, gaps between the plugs at the base of the head close. This forms a hinge at the base of the plugs that forces the margin shear ring and plug lip to separate as the head continues to deflect upward.

Q53. Can the head be modified to provide an acceptable design?

A53. (Allen, Holz, Butler) Analysis of the failure mode suggests that an acceptable design modification to increase head capacity may be to machine away non-load bearing portions of the head at the hinge point. This modification would increase the amount of deflection required before failure would occur and would, therefore, increase the energy absorption capability.

Q54. How has the Staff evaluated whether such a modification would be effective?

A54. (Allen, Holz, Butler) The Applicants have analyzed the effect of this modification by using layout drawings of the head along with deflection information taken during the scale model hydrostatic test. We have reviewed this analysis and concur with their conclusion that appropriate machining of the existing head is likely to give the head an energy absorbing capability beyond that needed to resist a sodium slug with 75 MJ of kinetic energy.

Q55. Please summarize the Staff's position with regard to the capability of the head to withstand a sodium slug impact kinetic energy of 75 MJ.

A55. (Allen, Holz, Butler) Appropriate criteria have been established with respect to the capability of the reactor vessel head to accommodate a sodium slug impact. While the present head may not meet these criteria, the Applicants have committed to develop a modified head design to correct this deficiency, as set forth in a letter from John R. Longenecker (DOE) to Dr. J. Nelson Grace (NRC) dated February 14, 1983. The Applicants have also proposed an additional testing program to verify the capability of the final head design. The Staff and its consultants have reviewed the proposed design modifications and test program and believe that modification of the head as required is feasible.

F. Evaluation Of Ramp Rates Associated With The LOFA

Q56. Describe the phenomena involved in a CDA initiated by a LOFA?

A56. (Allen, Theofanous, C. Bell) From the initiation of core disruption (initial clad melting) the LOFA will evolve through a continuum of gradually escalating core disruption states until complete disruption (melting of all fuel and inner blanket materials found within the original core confines, also known as a whole-core pool) occurs.

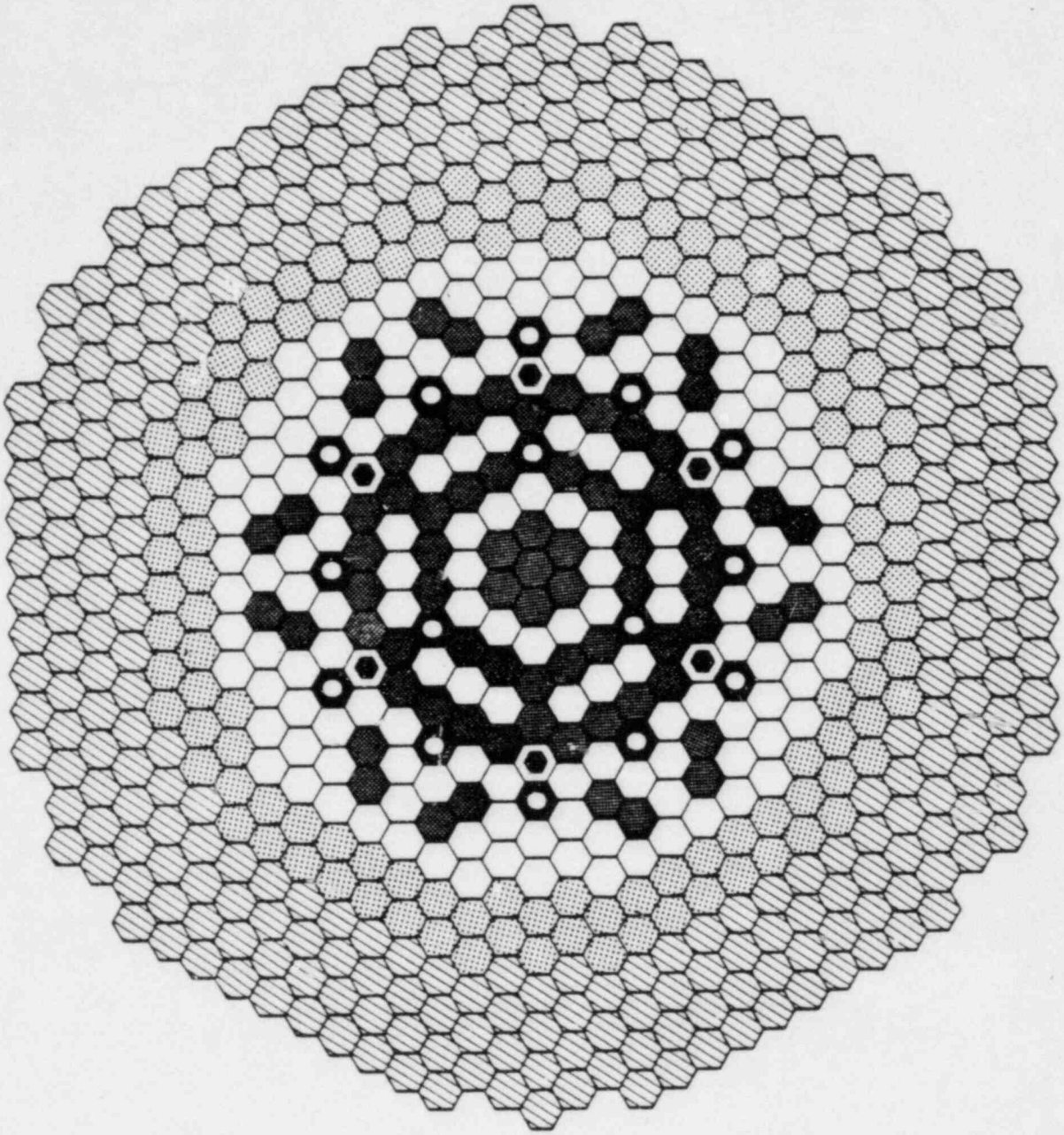
A cross section of the core is shown in Figure 5. This illustrates the arrangement of the various hexagonal subassembly units

(described in the legend below Figure 5). The open hexagons represent fuel subassemblies. The geometric arrangement of the fuel in three concentric rings is obvious from the figure (the outer ring has several rows of subassemblies whereas the inner rings consist of single rows of subassemblies). This configuration is the basis for the terms used in Figure 3, i.e., "subassembly scale pools", "annular pool" and "whole core pool".

Energetically, the progression through the various stages of disruption is important while a sufficient fraction (approximately 60% for the CRBR) of the initially present fuel remains within the active core region. Neutronically active states are then possible through a variety of rearrangements of driver, blanket, structural, control, and coolant materials. Permanent subcriticality, or "termination" (i.e., termination of energetic concerns) may occur from any point along the continuum of core disruption states. When the relocation of the appropriate quantity of driver fuel occurs in a forceful manner, we speak of "energetic termination" or hydrodynamic "disassembly." When this relocation is benign, we speak of "mild termination" or simple "dispersal."

Q57. What was the objective of your LOFA evaluation?

A57. (Allen, Theofanous, C. Bell) Our overall objective was to determine the relative likelihood of the two termination paths (dispersal versus disassembly) as a function of the degree of core disruption and to quantify the damage potential of the energetic ones.



Fuel Assemblies (156)

Inner Blanket Assemblies (76)

Radial Blanket Assemblies (126)

Alternate Fuel/Blanket Assemblies (6)

Control Assemblies (15)

Removable Radial Shield Assemblies (312)

Figure 5 Schematic View of Reactor Core Cross Section.

Q58. How was the evaluation carried out?

A58. (Allen, Theofanous, C. Bell) The evaluation was performed in two major stages. The first involved the CDA progression up to significant clad and fuel melting. This is called the "initiation phase". The second involves the CDA progression from that point on. It is called the "disruption phase". Energetic recriticalities are possible only during the disruption phase, since significant amounts of fuel and clad will then have melted and are available for relocation. The relocation can result in either dispersal or compaction of the fuel.

Q59. Are ramp rate values of 200 $/s (which was shown in section D above to be needed to challenge the head) typical of the results to be expected during a LOFA?

A59. (Allen, Theofanous, C. Bell) Not at all. During the initiation phase of the LOFA we found much lower ramp rates with essentially no energetics. During the disruption phase the ramp rates may be larger (conservatively bounded by 100 $/s) than in the initiating phase but still result in a minimal energetics release against the primary system boundary because of the mitigating effect of the cage (see the answer to Question 46 above).

Q60. How did you reach these conclusions with regard to initiating phase energetics?

A60. (Allen, Theofanous, C. Bell) A number of SAS3D analyses covering a broad range of the important parameters were carried out to charac-

terize the range of initiating phase LOFA behavior. With one exception, we found no significant energetics during this phase of the CDA (see the answer to Question 61 below). Details of these analyses can be found in Section II.3 of NUREG/CR-3224.

Q61. Please describe the exception referred to in Question 60?

A61. (Allen, Theofanous, C. Bell) We have identified plenum fission gas induced fuel compaction as a mechanism for initiating-phase energetics. In the presence of plenum pressure, accompanied by fuel column disruption, the fuel pin is subjected to unbalanced forces resulting in rapid downward motion of the blanket and undisrupted driver fuel pellets. We have been able to bound the reactivity insertion rates from the fuel compaction process per se at approximately 50 $/s. This ramp rate would, by itself, result in a minimal level of energetics on the primary boundary. However, at the time of this energetic event, only one-half of the core has been voided and the resulting high overpower could induce a LOF-d-TOP event in the unvoided half. Because of the potential augmentation of the reactivity from fuel motion within the pins of the unvoided regions, such a combination of events is highly undesirable. We have recommended, therefore, that steps be taken to limit the action of the plenum fission gas pressures during the initiating phase of the LOFA.

Q62. Does the Staff anticipate that its concerns with respect to plenum fission gas compactions will be resolved satisfactorily?

A62. (Allen, Theofanous, C. Bell) Yes. The Applicants have agreed to review this matter further. If further analysis confirms this concern, they have committed to make a design change to prevent the plenum fission gas from rapidly acting on the fuel in a compacting manner.

Q63. Is such a design modification feasible?

A63. (Allen, Theofanous, C. Bell) Yes. The Staff has considered a proposed design modification involving a flow restriction in the fission gas plenum region. The Staff has made a preliminary determination that such a device could be made effective. The Staff's review of this matter will be continued early in the OL stage.

Q64. How did you reach your conclusions with regard to the potential for energetics during the post-initiation (disruption) phase of a CDA?

A64. (Allen, Theofanous, C. Bell) The general behavior of the post-initiation period was examined both in terms of a SIMMER-II integral calculation as well as in terms of separate effects evaluations of fuel dispersal and recriticality phenomena.

Q65. Please describe your general findings from the disruption phase analyses.

A65. (Allen, Theofanous, C. Bell) Based upon the disruption phase analyses, we have concluded that the most likely path for a CDA to follow during the post-initiation phase is one in which mild

low-energetic bursts occur while the core is being simultaneously homogenized and dispersed, with dispersal being adequate to assure permanent neutronic termination prior to formation of an homogeneous whole-core pool.

Q66. How does the core become dispersed?

A66. (Allen, Theofanous, C. Bell) After fuel becomes molten and vapor pressures develop, the molten fuel is forced axially upward and downward through coolant channels and gaps between internal blanket subassemblies, and radially outward through the gaps between radial blanket and shielding subassemblies. The pressures required to do this are small and can easily be generated by mild bursts associated with gravity slumping during disruption.

Q67. Have you considered the extreme case in which a homogeneous whole-core pool is formed before sufficient fuel is dispersed to assure permanent subcriticality?

A67. (Allen, Theofanous, C. Bell) Yes. For the whole-core, homogeneous pool under perfectly symmetric conditions (geometry and power distribution), a sloshing action is observed which, under certain conditions of material configuration, may produce high reactivity insertion rates. In those cases, single phase expansions dominate neutronic shutdown and negligible energetics result.

Q68. Please explain what is meant by your conclusion that single-phase expansions dominate.

A68. (Allen, Theofanous, C. Bell) High pressures which will disperse the fuel and terminate the excursion are generated promptly with little addition of energy when there are few voids in the material undergoing the excursion. In that case the material is said to be in a single-phase (liquid) state, whereas when gas or vapor is present it is said to be in the two-phase state. In our studies we found generally that to attain high ramp rate recriticalities, the recriticalities must involve recompactions to the single-phase state. Because of the dominance of the high single-phase pressures produced in this case, the process tends to be self limiting with regard to the production of energy.

Q69. How effective is the single phase dispersal in limiting the energetics?

A69. (Allen, Theofanous, C. Bell) As an example of the effectiveness of this factor, in one whole-core pool case we considered an in-slosh with 300 $/s ramp at prompt critical. This yielded quick, single-phase thermal expansion shutdown and produced negligible energy release. This analysis demonstrated the effectiveness of single-phase dispersal in limiting the energetics.

Q70. Please quantify your conclusions concerning the energetics during the post-initiation phase of a CDA?

A70. (Allen, Theofanous, C. Bell) For the subassembly pool and annular pool phases, we estimate ramp rates of approximately 40 $/s and an upper bound limit of 80 to 100 $/s. These ramp rates (including the

upper bound limit) result in minimal loads on the primary system boundary.

G. Evaluation Of TOP CDAs

Q71. What are the differences between a CDA resulting from a LOFA and one initiated by a TOP?

A71. (Allen, Theofanous, C. Bell) The TOP-unique behavior develops during the very early stages of the initiating phase. As a result of the reactivity insertion the power rises quickly and produces fuel melting and pin failure well before coolant and cladding overheating. For a postulated mid-plane failure location, fuel motion within the fuel pin can have a significant reactivity augmentation effect and, unless it is moderated by an equally rapid dispersal of the fuel escaping into the coolant channels, an autocatalytic behavior could potentially develop.

Q72. How was the effect of potential mid-plane failures evaluated?

A72. (Allen, Theofanous, C. Bell, Rumble) Our assessment focused on defining the margins for autocatalytic behavior for assumed mid-plane failures. This behavior is controlled by the competition between pin-internal fuel motion and pin-external dispersal, usually referred to as sweepout. The relevant time scale is determined by the core-wide coherence of such pin failures which, in turn, is affected by the core configuration and the imposed reactivity ramp rate (coherence increases with ramp rate). For the CRBR the end-of-cycle-three (EOC-3) core with the replacement of the six high

power driver fuel assemblies with blanket assemblies has the highest potential for coherent (concurrent) pin failures when compared to other core states. The reactivity ramp rate used was selected on the basis of failure modes and effects analyses of the reactor control system (see NUREG/CR-3224, pgs. III-3 and III-4). We selected the 10-12 ¢/s TOP as a conservative upper limit for this investigation. Higher reactivity insertion rates are at least three orders of magnitude less likely.

Q73. How was your analysis for these conditions performed?

A73. (Allen, Theofanous, C. Bell) The EOC-3 core CRBR TOP accident was simulated with the PLUTO2/SAS4A computer code. A failure incoherence (time between failures) of more than 300 milliseconds (ms) for the first six groups of subassemblies was deduced. The PLUTO2 sweepout calculation was adjusted to reflect available experimental data from the L8 TREAT test. (The L8 test involved three full length irradiated pins in a flowing sodium loop which was subjected to a power transient). The results indicate that the PLUTO2 calculations are sufficiently accurate for use in conservative modelling of fuel motion reactivity effects in TOP accidents.

Q74. What are the results obtained from this analysis?

A74. (Allen, Theofanous, C. Bell) The calculated sweepout was seen to successfully cancel pin internal fuel motion reactivity (and a small amount of sodium voiding reactivity) and to produce shutdown with

the first 100 ms. Thus, even under the most coherent core conditions(i.e., flattest power distribution) and the most conservative pin failure location (midplane), no energetic behavior could be found for TOP events of up to 10-12 ¢/s.

H. Evaluation Of Protected LOHS CDAs

Q75. What are the differences between a CDA initiated by a LOFA and one initiated by a protected LOHS?

A75. (Allen, Theofanous, C. Bell) The protected LOHS-unique circumstances originate from the fact that, in this case, core disruption occurs at very low power and in the absence of sodium coolant. Under low power conditions the absence of coolant is required to initiate the CDA since boiling can remove heat at decay power levels. The core may become uncovered because of coolant boiloff or failure of the primary coolant system boundary at the high temperature LOHS environment. The actual mechanism is not important because it will affect only the disruption stage power level which, in any case, is very small. Characteristically, however, disruption would not occur until many hours into the accident, indicating significant margins for recovery.

Q76. Describe the phenomena involved in a CDA initiated by a protected LOHS.

A76. (Allen, Theofanous, C. Bell) At the characteristically low heating conditions all steel within the core will melt, relocate downward and form a plug at the lower axial blanket region. The system will

remain subcritical, and hence will continue to heat slowly, until fuel settling occurs either due to softening of the pellets (as the melting point is approached) or simply due to toppling and compaction to a lower porosity. The initial porosity is approximately 65%, while a porosity of approximately 50% would be required to approach criticality. This eventual approach to criticality would accelerate the melting rate thus producing, at most, a moderate scale recriticality estimated at approximately 60 $/s. Such an event would be sufficient to disperse the core into the vessel and provide permanent neutronic termination. A smaller recriticality, however, i.e., approximately 10-20 $/s, would be considered more likely under these circumstances and it would be insufficient to provide termination by fuel removal. A whole core pool, with homogenization of all internal, axial and radial blankets would result in this case. The resulting dilution would then be adequate to render the system permanently subcritical even in the absence of the steel and control rod materials which will eventually separate out.

Q77. Would energetic events during a LOHS accident produce the same consequences as in a LOFA?

A77. (Allen, Theofanous, C. Bell) In the absence of the sodium pool typical of the protected LOHS, even the most severe recriticalities could provide no slug impact challenge to the reactor vessel head. The only potential challenge could occur from impact of the UIS on the head. To explore this potential, we considered the consequences

of a postulated LOHS with a 200 $/s ramp rate. The expansion forces on the UIS, conservatively assuming absence of significant resistance by the UIS support columns, were evaluated using the SIMMER-II code. An upper bound UIS kinetic energy (in the upward direction) of approximately 5 MJ was thus estimated. Such a missile is of little mechanical consequence to the reactor vessel head.

I. Conclusions

Q78. Please summarize the conclusions you have reached as a result of your independent assessment of CDA energetics.

A78. (Allen, Theofanous, C. Bell, Butler, Rumble) We have systematically evaluated the possible progression of all three classes of CDAs as exemplified by the LOF, TOP, and protected LOHS accidents. Non-negligible energetic circumstances were identified only as a consequence of recriticalities within the LOFA sequences (assuming that the plenum fission gas fuel compaction mechanism is eliminated).

The magnitude of recriticality events in the S/A-scale (i.e., prior to S/A wall failure) and annular pool (i.e., prior to melting of the inner blankets, which leads to the whole-core pool) phases are limited to the order of 50 $/s or less, because of incoherence. Neutronic activity, throughout both of these stages of core disruption is substantial and contributes to pressurization and fuel dispersal away from the core region. Thus, benign termination prior to entering the whole-core, homogeneous pool phase, is projected

even under restrictive assumptions for fuel removal path availablity and fuel removal mechanics.

Whole core pool recriticalities exhibit a narrow range of significant energetic behavior. This energetic regime is associated with idealized perfectly symmetric geometry and completely homogeneous pools. Even so, the resulting level of energetics does not exceed the structural capability of the primary system boundary.

The levels of energetics required to produce significant structural damage in the CRBR were evaluated, taking into account, for the first time, the structural enclosure formed by the Core Barrel/Core Support Structure/Upper Internal Structure, and the pressure transmission characteristic of the expanding core medium and other materials within that enclosure. We conclude that an accident with an energetic yield of about 1130 MJ (expressed as ultimate work potential for expansion to one atmosphere) would be required to fail this inner containing structure, and about 2550 MJ would be required to challenge the reactor vessel head structure, i.e., to produce a slug impact kinetic energy close to the CRBR vessel head design value of 75 MJ. These levels of energetics roughly correspond to two-phase whole-core disassemblies with 100 $/s and 200 $/s driving reactivity ramp rates.

The capability of the CRBR vessel head to absorb the kinetic energy transferred to it from impact by a sodium slug (with kinetic energy

of 75 MJ at time of impact) has been evaluated. The Staff has determined that the Applicants' commitment to a head design with this capability can and will be met.

Based on these results we conclude that a CDA-induced energetic vessel head failure is physically unreasonable.

Q79. Has the Staff's assessment been reviewed by other knowledgeable bodies?

A79. (Allen, Theofanous, C. Bell) A review of the independent assessment reported in NUREG/CR-3224 was made by a number of individuals and institutions. The review process is described in NUREG/CR-3224 (pages I-4, 5). The results are given in the compendium to NUREG/CR-3224. The results of those reviews support the approach used and the general conclusions expressed above. The results of the independent assessment and its review were also presented to the ACRS at its meeting of April 14, 1983. The ACRS has concurred in the Staff's position with regard to energetics and provided the following comments:

> An historical liquid-metal fast-breeder-reactor safety concern has been the potential for large reactivity excursions caused by, for example, a combination of failure to scram and either a loss of coolant flow or an insertion of reactivity. It is sometimes postulated that such an excursion could lead to vaporization of coolant and fuel and to rupture of the primary containment (i.e. reactor vessel, etc.) and possibly secondary containment (i.e., the steel containment shell) due to the pressures resulting from the vaporization. This event is termed an energetic core disruptive accident (CDA). Both the Applicants and the NRC Saff have independently reviewed this potential and have concluded

> that the probability of such an accident is quite low. Further, both conclude that, even if such a combination of events did occur, the magnitude of the resulting mechanical forces in the CRBRP design would be well below the capability of the primary containment system to withstand such forces without rupture. We concur in the NRC Staff position.

(The ACRS letter containing these comments is reproduced in full in Appendix I of CRBR SER Supplement No. 1.)

J. Answers To Board Questions 11 And 17

Q80. Please respond to Board Question 11, which states as follows:

> In discussing the energetics of accidents beyond design basis, the Staff offers the statement that there will be an "isentropic expansion yield to one atmosphere" (NUREG-0968, Vol. 2, p. A. 2-5). The Staff is requested to discuss briefly what is the physical significance of this statement and the extent to which it contributes to any conservatism in the analyses of energy releases. Phenomenologically, how has the Staff satisfied itself that "approximately 2550 MJ would be required to produce a slug impact kinetic energy close to the head design capability of 75 MJ" (Ibid).

A80. (Allen, Theofanous, C. Bell) The concept of an "isentropic expansion yield to one atmosphere" (the ultimate work potential (UWP) described in answer to Question 45 above) is used only as a reference point to indicate the relative potential severity resulting from disrupted core conditions. It has been widely used because it is an unambiguous and easily defined quantity. It has no physical application in that such yields cannot be realized in real systems. Since it is only a reference value, it is not actually used in analyzing the capability of the system to accommodate CDA loads. Thus, it makes no real contribution to the conservatism in the analysis of energy releases.

The Staff has satisfied itself that "approximately 2550 MJ would be required to produce a slug impact kinetic energy close to the head design capability of 75 MJ" through a detailed analysis of a realistic expansion process. The analysis is summarized briefly in the answers to Questions 46, 47 and 48 above. A more detailed explanation is provided in Section II.2 of NUREG/CR-3224.

Q81. Please respond to Board Question 17, which states as follows:

> What is the status of the Staff's review of, and what is the Staff's position with respect to, "The Eight Areas of Concern" listed in Section I, Table II of NUREG/CR-3224?

A81. (Allen, Theofanous, C. Bell) As noted in the answer to Question 32 above, these areas were developed from an initial review of the Applicants' analysis of CDA energetics. The Applicants' responses to questions relating to these eight areas was factored into our independent assessment efforts (also described in the answer to Question 32 above). Thus, the resolution of these areas has been included as part of the tasks associated with the independent assessment. As such, the Staff's conclusion regarding each of these areas is reported in various sections of NUREG/CR-3224. As discussed in answer to Question 32 above, each of the "eight areas of concern" has been resolved to the Staff's satisfaction. The status of the Staff's review and the Staff's position with regard to each of the "eight areas of concern" is provided below, together with citations to the appropriate sections of NUREG/CR-3224.

AREA 1

Can TOP become prompt-critical in such a way that internal fuel motion in lower power channels is the key factor in the energetics determination? Is such an event possible only for mid-plane failures with low sweepout? How is the degree of sweepout determined? What is the effect of intrasubassembly incoherence on sweepout?

RESOLUTION 1

TOP behavior is important in quantifying the energetics potential because of an associated autocatalysis potential (self-amplifying power transient) and because of the greater damage potential associated with in-core sodium (higher and more sustained pressures following core disassembly). This potentially energetic regime is avoided if the reactor does not approach the prompt-critical neutronic condition. The potential for producing this condition depends on the axial location of fuel-pin failures, the number of pins failing concurrently (coherence), and the efficiency at which failed fuel is swept from the core by the flowing sodium, fission gas, and locally generated sodium vapor.

The concern was resolved by assessing the worse-case situation in which midplane pin failure was assumed, experimentally supported fuel sweepout characteristics were utilized, the core state (burnup) that exhibits maximum pin-failure coherence was utilized, and the maximum reactivity insertion rate with a significant probability of occurrence was imposed. Even for the worst-case situation, a margin

against the autocatalysis regime was found (see Section III of NUREG/CR-3224).

AREA 2

A LOF-d-TOP might still occur if the sodium void worth is 50-60 percent higher and internal fuel motion in TOP type channels can occur. What are the reactivity uncertainties of sodium void, Doppler, axial expansion and lead channel fuel motion? How do you interpret the significance of these uncertainties?

RESOLUTION 2

The LOF-d-TOP concern is similar to that for the TOP (Area 1 above), i.e., autocatalysis and in-core sodium. The potentially energetic situation is avoided if the combined reactivity feedbacks from fuel expansion, Doppler, sodium voiding, cladding relocation, and initial fuel disruption are small (less than +1$) until voiding of the entire core has occurred.

The resolution was achieved by establishing reasonable values and associated uncertainties for the neutronic parameters, developing reasonable ranges for assumptions in material relocation models through comparison with experiments, and performing a conservative integrated analysis including sensitivity studies to define the boundaries of the LOF-d-TOP regime. The Applicants' best-estimate (selected to be most representative of reality) analysis showed no LOF-d-TOP tendencies even for the upward revised sodium-void

reactivity. Our independent assessment (see Section II.3 of NUREG/CR-3224), which combined parameters and assumptions in a pessimistic manner for conservatism, indicated that this LOF-d-TOP regime was not attainable unless unrealistic assumptions were made or the plenum fission gas compaction mechanism is involved.

AREA 3

What is the potential for autocatalysis due to plenum fission gas acting on the fuel column to force axial compaction as disruption occurs in the initiating phase of the LOF?

RESOLUTION 3

The pressurized gas plena above the core pose the potential for rapid compaction of the fuel from above. The result of some local compaction is that the power increases, more disruption occurs, more extensive compaction is initiated, and the power escalates rapidly; thus, the potential for autocatalysis. This is also a mechanism for promoting early positive reactivity feedback from initial fuel disruption (early fuel disruption is generally dispersive with associated negative reactivity effects after a minimal fuel burnup) in the context of the concern in Area 2.

The resolution of this concern was based on analyses by both the Applicants and ourselves. The Applicants' best-estimate results indicated little compactive potential because of rapid discharge of the gas from the plena prior to fuel column disruption. Our

analysis (see Section II.4 of NUREG/CR-3224), which incorporated more allowance for uncertainties, indicated the lack of a direct autocatalysis tendency from pressure-driven compaction but showed a general tendency for development of the LOF-d-TOP situation (Area 2), particularly near the end of the burnup cycle. Resolution of this concern was obtained with an agreement by the Applicants to eliminate this strong compactive tendency.

AREA 4

To what extent can steel blockages form throughout the core to prevent fuel removal through normal axial blanket flow channels during the early phase of the LOF? What is the location and character of the steel blockages in these channels?

RESOLUTION 4

The concern in this area is associated with the potential for closing the major path for early fuel removal and thereby assuring the progression of the disruption phase to the high-inventory, whole-core pool state with its potential for neutronic amplification by sloshing. For this concern, the opposite end of the uncertainty and assumption spectrum from that considered in Areas 2 and 3 must be addressed. This is necessary because an energetically benign initiating phase of the LOFA provides the greatest opportunity for extensive steel relocation from the core and plugging of the coolant channels in the colder axial blankets.

This benign initiating-phase regime was analyzed by both the Applicants and the Staff. The Applicants' analysis as well as our own do not indicate complete, core-wide steel plugging prior to substantial fuel disruption in the hottest subassemblies. Our assessment is that the mobile fuel in these hottest subassemblies, with its large reactivity effect, controls the subsequent power response of the core and promotes power oscillations because of its natural tendency to compact by gravity. The result is the co-melting of fuel and cladding in a large part of the core, instead of the separate melting and relocation of the cladding, thereby assuring a significant number of fuel removal paths through the axial blankets (see Section II.3 of NUREG/CR-3224).

AREA 5

What is the basis for maintaining continuous subcriticality in the high heat loss environment of early meltout phase? What are the fuel losses (quantified) taking into account uncertainties in removal path geometrics, driving pressures and freezing mechanisms?

RESOLUTION 5

The concern associated with the assumption of a continuous subcritical condition involves the time frame to reach a completely disrupted core state (whole-core pool). If subcriticality is maintained (core boilup by steel vapor formation at decay power), the heatup rate of the core would be on the order of 10 K/s requiring many tens of seconds for complete disruption. Therefore, the

core disruption process could be analyzed using simple quasistatic approaches uncoupled from the neutronic behavior. However, if subcriticality is not maintained, the disruption process becomes highly transient, nonlinear, much more rapid, and much more difficult to assess, particularly for assurance of neutronic termination by dispersal (gradual fuel removal).

This concern was resolved through our detailed reference analysis of the overall accident sequence using state-of-the-art analysis tools (see Section II.5 of NUREG/CR-3224), separate effects analyses (see Section II.6 of NUREG/CR-3224) and fuel freezing and plugging data. Our findings were that subcriticality would not be maintained continuously, that the sustained neutronic activity would drive fuel from the core as disruption progressed, that the number of fuel removal paths would increase as disruption progressed, and that the fuel removal process would dominate the accident sequence more and more as disruption progressed. The complexity is greater but the accident behavior is well bounded.

AREA 6

What degree of subcriticality is required to prevent pool recriticality from thermal and fluid dynamics upset conditions? What is your position on the potential for small recriticalities to amplify? What is the justification for your position?

RESOLUTION 6

The concern in Area 6 was that whole-core pool transients could be self-escalating if they contained sufficient fuel to be critical. A small recriticality, initiated by a small thermal or fluid dynamic upset condition, could promote coherent outward movements of core materials (nonenergetic disassembly) followed by coherent, gravity-driven reassembly. The momentum associated with the reassembly could cause a secondary recriticality larger than the first. The process could repeat itself in an undamped manner until termination by energetic disassembly. The magnitude of this energetic disassembly was an open question and the focus of our concern.

This concern was resolved partially through static criticality calculations by both the Applicants and the Staff. There was agreement that permanent removal of about 40% of the original fuel inventory was required to eliminate the recriticality potential completely. Through our own analyses, and special experiments (see Section II.5 and II.7 of NUREG/CR-3224), we determined that sloshing amplification was likely if the molten pool inventory was greater than 60%. However, the dominance of fuel removal reduced the time interval for sloshing amplification to a degree that effectively eliminated the concern. Also, the energetic yield of the high ramp-rate sloshes was found to be highly mitigated (see answers to Questions 46, 67 and 68 above).

AREA 7

In assessing benign termination from the boiled-up pool (upward removal), justify the fuel removal mechanisms and rates. In particular assess the potential for upper pool sodium entry via rapid condensation of steel vapor pressure.

RESOLUTION 7

The concern here relates to the ability for fuel to discharge upward from a boiling pool at decay power. At this low power, vapor velocities are insufficient to fluidize the liquid to a sufficient extent to provide massive discharge. As associated concern is sodium re-entry as the molten core pool discharges through openings in the upper axial blanket region and into the cold upper core structure where rapid condensation could occur with associated subambient local pressure. If this low pressure produced sodium re-entry prior to sufficient fuel removal to prevent recriticality, a pool transient such as temporary collapse of a boilup state could result, leading to a recriticality or perhaps initiating the sloshing amplification process (see Area 6 above).

Our independent analysis of the disruption process indicates that this upward discharge from local openings in a sealed, boiled-up core is irrelevant.

AREA 8

What is your estimate of the force required to produce a mechanically induced relief path via upper internals structure displacement?

RESOLUTION 8

The buckling of the support columns of the upper internal structure is a threshold event that results in easy upward displacement of the upper internal structure and upper core structure. This displacement removes a major impedance from the core expansion process thereby permitting the high core pressure, generated from an energetic disassembly, to act directly against the sodium pool. This core pressure can produce larger kinetic energy in the pool and, subsequently, larger loads against the vessel head. Energetic events that are incapable of buckling these columns produce negligible loads on the head. In addition, the opportunity for strong thermal interaction between the core materials and the sodium pool is effectively denied.

This area was resolved using the Applicants' finite-element analysis of the column buckling threshold, our own transient analysis models (see Section II.2, Appendix B of NUREG/CR-3224), and comparisons of model predictions with SRI column buckling experiments (using scale models of the CRBR upper internal structure and columns). As indicated in response to Question 49 above, the column failure threshold in terms of accident severity, or ramp rate, was found to be approximately 100 $/s.

III. EVALUATION OF EVENTS FOLLOWING LOSS OF CORE GEOMETRY

A. Introduction

Q82. What subject matter does Part III of this testimony address?

A82. (Long, Swift) Part III of this testimony addresses the adequacy of the Staff's analysis of core disruptive accidents in the CRBR, with respect to those aspects of the CDAs which occur subsequent to the loss of core geometry.

Q83. Describe the initial steps following loss of core geometry in the subsequent long-term behavior of the disrupted core material?

A83. (Long) There is a possibility that the disrupted core would remain coolable and subcritical within the primary vessel. The uncertainty in this course of events is great enough that we have assumed that it could not be satisfactorily quantified. Therefore, we have studied the more severe alternate course, namely that the core debris was not coolable.

By the time the core debris has descended to the bottom of the reactor vessel, it is expected to be in the form of small particles (as predicted by experimental data) as a result of melting and quenching and will have incorporated a significant amount of blanket material and structures with it. This comes about whether the initial dispersal of the core was rapidly energetic or whether it followed a slow series of melt, slump and remelt sequences. If rapid, the initial dispersal would be at least partially radial, so that a small fraction of the relatively undiluted core material

would fall initially to the bottom head, followed by remelted core material incorporating radial blanket material. If the initial dispersal is slow, radial dispersal would not occur, but the core would incorporate substantial lower axial blanket material in its downward melting and remelting progress.

On the bottom head, the finely fragmented core debris would form a bed whose thickness and particle size would prevent it from being cooled by the overlying sodium. Before the core debris bed would reach its melting temperature, the lower head would heat and fail by creep-rupture under the temperature and weight loading. The guard vessel would fail soon thereafter in a similar manner.

The hot core debris in particulate form, with debris from structures and the vessel head, would fall to the reactor cavity floor about 1000 seconds into the accident. The following rush of sodium at about 1000°F would disperse the fragmented core relatively uniformly throughout the floor of the reactor cavity. A reactor cavity vent diaphragm would rupture at this time. The reactor cavity is filled with an inert atmosphere (less than 2% O_2) so that little sodium oxidation would take place.

Q84. Can the core debris become critical again at this stage?

A84. (Long) There is little geometrical constraint to preclude criticality, and the mass of fissile material is sufficient to allow criticality. Nevertheless, criticality is unlikely in this

situation. Considerable blanket material would be incorporated with the core by this time. If initial disruption had occurred rapidly, this additional blanket material would be largely provided from the radial and inner blankets when radially dispersed core material melted and incorporated additional 238-U in the debris. If initial disruption had followed the slow melt-slump progression, much of the 238-U from the lower axial blanket would be incorporated. The rapid flow of sodium into the cavity during and after the descent of the core debris would lead toward uniform dispersal of the debris throughout the cavity floor, in a thin slab. The volume of fuel and cladding alone, if spread throughout the cavity floor, would make a slab less than 0.5 inches thick, which is clearly subscritical.

If criticality should occur in some intermediate configuration of this progression, the ramp rate, gravity driven, would be low enough (based on the in-vessel energetics calculations) that no large energetics comparable to that in the primary system would occur. The pulses of energy accompanying recriticalities, when averaged over time, would comprise only a small addition to the decay heat. Low level energetics accompanying a recriticality would aid in the dispersal of the debris toward a uniform slab. No sustained criticality can be achieved by the debris bed because of the lack of geometric controls. The net energy contribution of recriticalities would therefore be neglibible compared to the sustained decay heat of the core debris. Eventually, dilution by blanket material and

concrete reaction products would prevent further criticalities from occurring.

Q85. What would be the effect of the hot core-debris and sodium in the steel-lined reactor cavity?

A85. (Long) The Staff expects that the reactor cavity floor liner would be rapidly penetrated by the initial contact with the high temperature core debris. The few inches of insulating concrete below the floor liner is not highly resistant to sodium and would also be rapidly penetrated. Thus the surface of the concrete basemat is assumed to be exposed to the sodium and core debris almost immediately after vessel failure. From this point onward, there remains some uncertainty about the rates at which the detailed course of events will take place and about the challenges to containment that will result. There is, however, a sufficient body of experimental data to define the most likely courses that the accident is likely to take and to bound the other uncertainties and their consequences.

Q86. After penetration of the reactor cavity floor liner, what are the next principal events in the course of the accident sequence?

A86. (Long) The mixture of hot sodium and core debris will react with the concrete. Water will be driven out of the concrete as its temperature is increased. This water will react with sodium to generate hydrogen and sodium hydroxide. Exothermic chemical reactions between sodium and concrete are possible, and the reaction

zone will advance downward into the concrete basemat comprising the reactor cavity floor. The chemical reaction heat is significant but not as great as the decay heat. Thermal energy is added to the sodium pool bringing it up to its boiling temperature. A mixture of sodium vapors and hydrogen is evolved through the cavity vent into the containment building where it is ignited, forming copious amounts of sodium oxidic aerosols and water (which will quickly react to form sodium hydroxide). The oxygen content of the containment atmosphere is reduced, until at about 6% O_2, hydrogen will no longer be ignited but sodium vapors will continue to burn. The hydrogen content of the containment atmosphere will again begin to increase.

The burning of the sodium and hydrogen transfers energy to the containment atmosphere. Some of this energy will be dissipated through the containment walls to the annulus cooling system. Eventually either the increase in pressure or hydrogen content in the containment building or the decrease in the oxygen content will require that the atmosphere be vented.

Venting will take place simply by opening relief valves that permit the atmosphere to blow down through a wet cleanup and filtration system. The reduction in pressure at the time of venting will induce considerable additional vaporization of sodium and the creation of more aerosols.

When the pressure in the containment is reduced to about one atmosphere, exhaust fans in the vent lines are turned on and purge valves are opened to admit fresh air. The fresh supply of air assures that all sodium vapors in the containment will be oxidized before they enter the wet cleanup system. The continued dilution of the atmosphere will prevent the hydrogen content from exceeding about 6%, so that it will not generate much pressure if it should burn.

Sodium will be boiled away in the reactor cavity after about five days, or less. The remaining core debris will continue to melt its way into the concrete, without chemical reaction, until decay heat declines and the quantity of involved molten concrete increases such that heat fluxes are low. Hydrogen and aerosol production will be greatly reduced. After six months to a year, it is estimated that heat fluxes will be so low that further penetration is at a negligible rate when the core debris has penetrated 20 ± 5 ft. into the 26 ft. basemat.

Q87. What methods of review and independent analysis did the NRC Staff use to analyze the melt-through sequence?

A87. (Long) The various steps in the review undertaken by the Staff are illustrated in the attached Figure 6. Each of the steps in the sequence was examined by independent calculations, duplicate calculations, sensitivity studies, comparisons with experiments and with experience, and qualitative judgements as appropriate. The

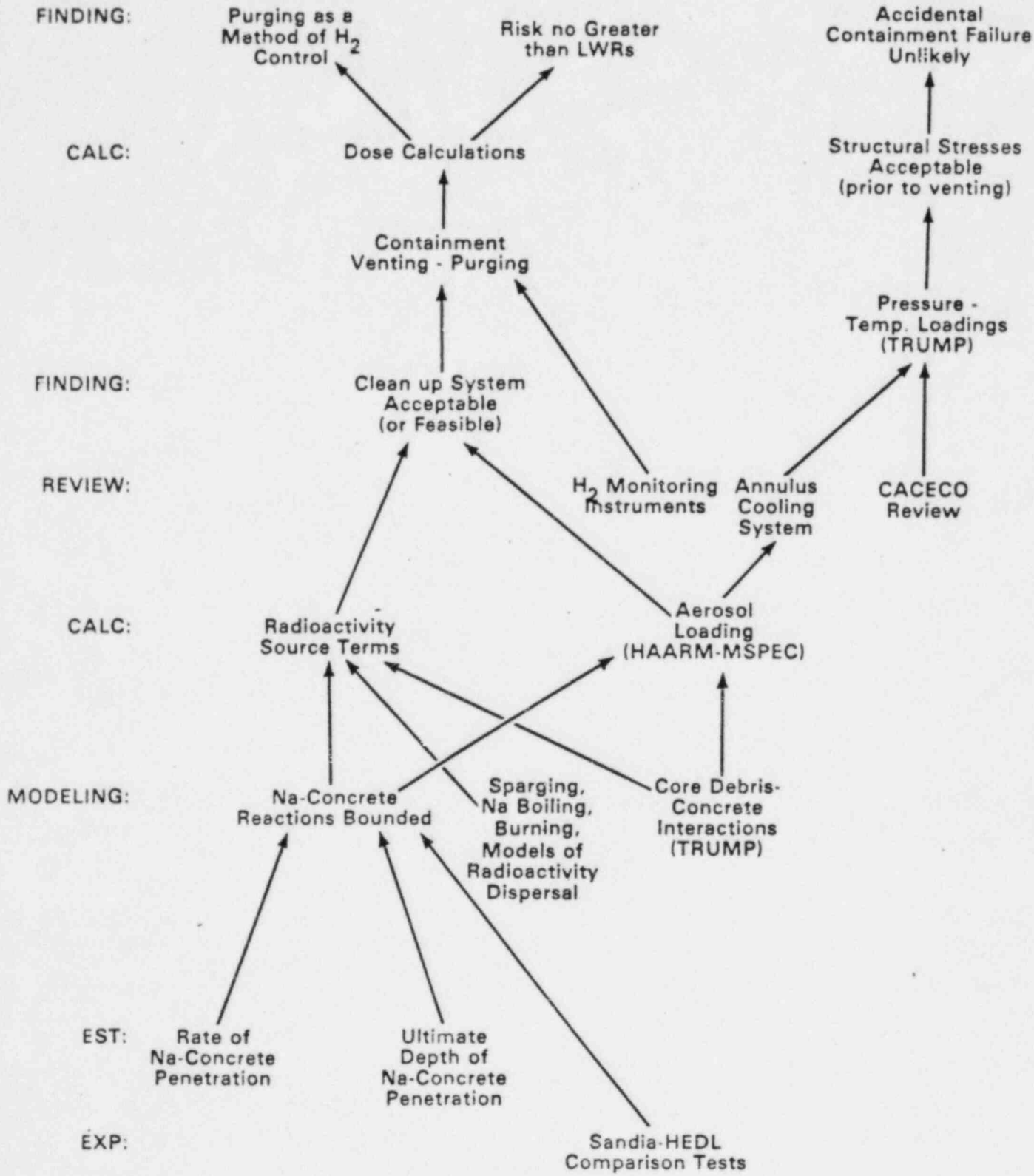


Figure 6 CDA PART III REVIEW PLAN

Staff reviewed variations from the Applicants' proposed sequences at almost every step.

The Staff engaged the services of expert consultants to provide additional review capability in the areas of high-temperature reactions among sodium, core-material, and concrete; aerosol behavior; and the response and survivability of key structures.

Q88. Has a base case scenario been chosen with respect to which other variations can be comparatively studied?

A88. (Long) Both Applicants and Staff have adopted as a base case the scenario with penetration of sodium into concrete at the rate of one-half inch per hour for four hours. This scenario is characterized by a rather extended (5 day) period for the boiloff of the sodium, exposing the structures of the containment to the maximum heat immersion. Containment venting is required at 36 hours, due to the build-up of hydrogen in this scenario.

The principal events and characteristics of the Staff's base case melt-through accident are illustrated in the attached Figure 7. The consequences of this scenario, in terms of doses at the LPZ boundary, are given later in this testimony, in Part IV.

Q89. Were alternatives to this base case also reviewed?

A89. (Long) Some degree of study was devoted to each of the junctures in Figure 6. In some instances it was possible by a qualitative review

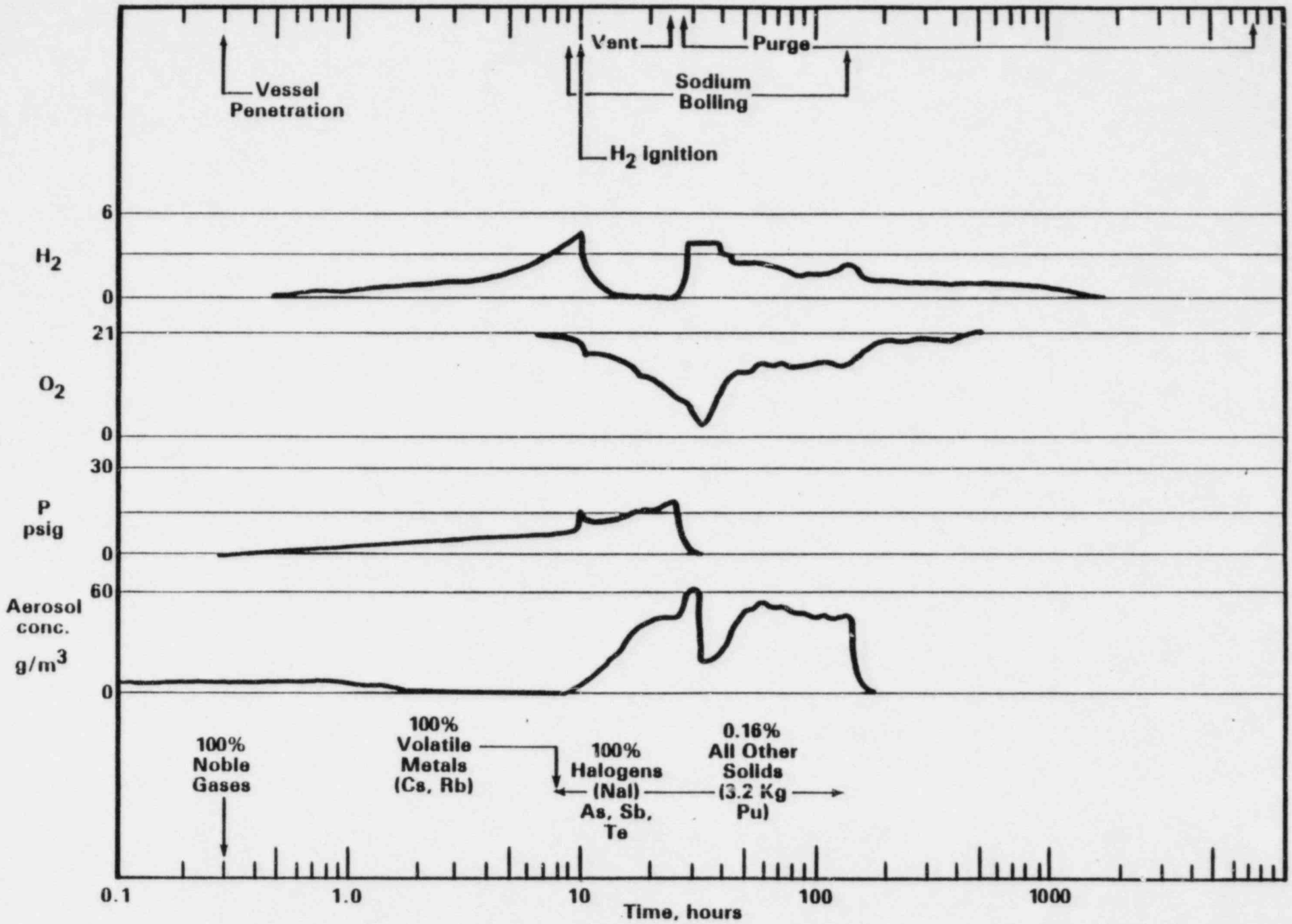


Figure 7 THERMAL MARGIN SCENARIO

BASE CASE - STAFF VERSION

to determine how the events in Figure 7 would be displaced or altered. In other cases independent analyses were necessary.

The principal independent Staff analyses on variations of the base case scenario were in the following areas:

a. Sodium-concrete reaction rates CACECO calculations:

1/2 inch per hour for 4 hours (base case).

7 inches per hour for 1/3 hour followed by 1 inch per hour until sodium boils dry.

7 inches per hour for 1 hour followed by 1 inch per hour until sodium boils dry.

These three variations were analyzed with and without wall plateout. The latter two cases envelope all observed sodium concrete interaction experiments by a wide margin, as shown in Figure 8.

It was observed that the decay heat predominated over the sodium-concrete reaction heat in all the above cases. Energy input to containment is increased but consequences are not severely affected by the rate of concrete reaction. Boil-dry time is foreshortened to 70 hours in the more rapid cases. The time of venting is reduced from 36 to about 24 hours.

b. Aerosol Behavior

Quantities in suspension were independently checked using the HAARM-3 code at various aerosol source rates. Applicants' HAA-3 calculations were determined to over-predict experimen-

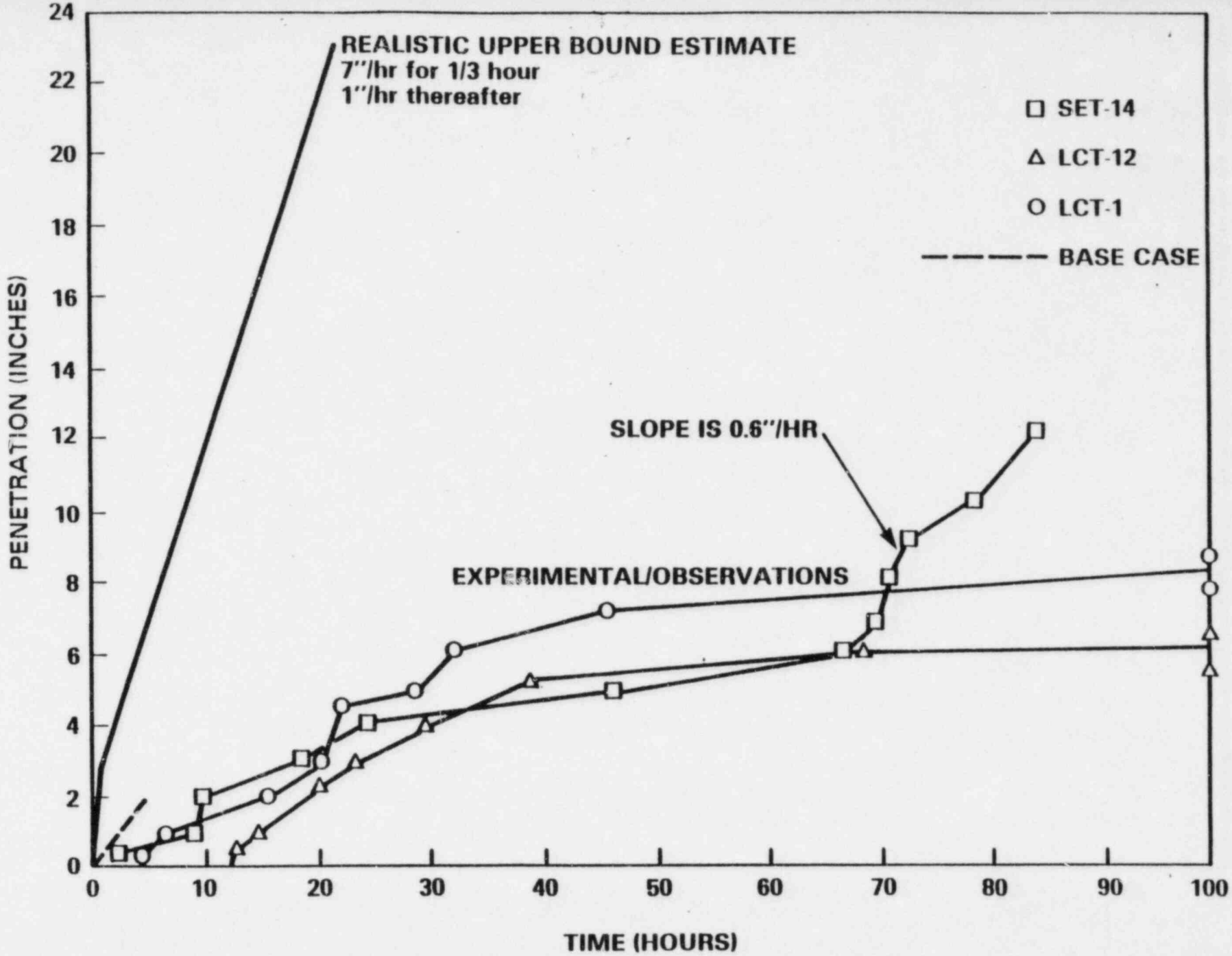


Figure 8 VARIOUS PENETRATION RATES OF SODIUM INTO CONCRETE

tally determined suspended concentrations. This is an important conservatism, since the concentration of suspended aerosol governs the release of radioactivity to the cleanup system.

c. Radiological Source Term Variations

Release of 137-Cs without fallout (base case includes a fallout fraction).

Release of 10% of 131-I without fallout (base case includes a fallout fraction for all iodine).

These were sensitivity studies based on hand calculations. The increased cesium source resulted in an increase of 5 rem to bone, and 3 rem to liver and whole body doses. The increased iodine source produced a net dose increase of 80 rem to thyroid at the outer boundary of the low population zone.

d. Wall Deposition of Aerosols

Supplementary calculations of the wall deposition of sodium oxidic aerosols were performed using the MSPEC aerosol code, in order to determine the extent of possible interference with the annulus cooling system. This topic is discussed in response to Questions 109 and 110 below.

e. Plutonium Source Term

The Staff has taken a more conservative view than the Applicants of the amount of plutonium that might be released from the sodium pool to form aerosols. This has taken the form of an increase by a factor of ten in the amount of plutonium released during the boil-off period, and a consequent increase in the bone dose of approximately the same amount. The release

of plutonium is discussed in response to Questions 148 and 149 below.

In addition, the Staff has reviewed, but not recalculated a number of variations on the base case that have been submitted by the Applicants. These include:

a. Reduced Decay-Heat Studies

In some of these alternatives to the scenario, substantial hydrogen was produced before the sodium boiled sufficiently to ensure ignition. It was determined that the levels of hydrogen concentration could be controlled by purging.

b. Concrete Thermophysical Data

Sensitivity studies were performed by the Applicants on water content and thermal conductivity.

c. Reactor Vessel and Floor Liner Penetration Time

Over a wide range (100 - 10,000 seconds) of sensitivity studies, variations of these parameters had negligible effects on the consequences.

d. Core Debris Bed Leveling Characteristics

Sensitivity studies with respect to core debris bed coolability were submitted. The base case assumption that the debris bed is not coolable appears conservative with respect to a wide range of likely configurations.

The Staff's conclusions from these reviews were that sufficient variations had been calculated to cover the uncertainties in the

details of the sequence and to provide confidence that the overall results of the accident are understood.

Q90. What other activities contributed to the Staff's review of melt-through scenarios?

A90. (Long) The Applicants' general requirements for those systems and features provided in the design to mitigate a CDA (SER p. A.4-19) were reviewed with respect to the Staff's criteria (SER p. A.4-1). Design specifications for the key structures and equipment were reviewed in comparison with ASME and other applicable engineering standards. Feasibility of the Applicants' proposed designs was verified. Two items resulted from this review where confirmatory effort is required. These are in regard to the effect of a wall plate-out of aerosols on the performance of the annulus cooling system, and the confirmation of the failure criteria for cell liners. Both of these items are scheduled for review and confirmation by the Staff before the pertinent equipment is installed.

Q91. How is the remainder of Part III of this testimony organized?

A91. (Long) The remainder of Part III of this testimony addresses in sequence the following: (1) sodium-concrete interactions; (2) aerosols; (3) hydrogen; (4) structural response; (5) annulus cooling; and (6) vent cleanup system.

B. Sodium-Concrete Interactions

Q92. What are some of the problems caused by sodium-concrete interactions?

A92. (Swanson) Sodium can attack concrete and penetrate some distance into it. As sodium attacks concrete, hydrogen will be generated. At the same time, sodium oxide aerosols can be formed in the atmosphere. Aerosols can plug vent paths that are a key factor in the release of radioactivity. Hydrogen, generated from the reaction between sodium and water released from the concrete, can present a combustion threat and is important in determining the time for containment venting.

Q93. What factors influence the extent of sodium-concrete interactions?

A93. (Swanson) A number of factors influence the extent of sodium-concrete interactions: factors affecting reaction chemistry include the type of concrete, whether the concrete is dehydrated, and the sodium pool temperature. Other considerations include the the formation of reaction products, cracking, spallation, stress conditions, the presence of rebars, pool depth, scale and orientation of the surface being attacked.

Q94. What is a realistic upper bound to the penetration rate of sodium attacking concrete?

A94. (Swanson) A sodium concrete penetration rate of 7 in/hr for the first 20 minutes, followed by a rate of 1 in/hr thereafter, will envelope all existing experimental data.

Very high initial penetration rates (approximately 7 in/hr) have been observed in a few experiments but these rates have persisted for only a short time. Long term penetration rates are significantly less (less than 0.2 in/hr). Thus, the above penetration rate is a realistic upper bound, adequate to include all available data.

Q95. What is the maximum penetration depth of sodium into concrete?

A95. (Swanson) Two factors tend to limit the penetration of sodium into concrete. These are the formation of large quantities of reaction products and the exhaustion of water from the concrete. Based on the quantity of water to be present in the concrete forming the CRBR basemat, Sandia National Laboratories has calculated a maximum penetration of 30 in. This proposed maximum penetration exceeds the maximum penetration observed experimentally (14 inches -- see Figure 8) by a considerable margin, even in tests with a duration of 100 hours.

Q96. What are the effects of the products of chemical reactions on the chemical attack of sodium on concrete?

A96. (Swanson) Large volumes of reaction products are generated during the course of sodium-concrete reactions. These products mix with sodium hydroxide formed during the reaction of sodium with the water released from the concrete.

At the conclusion of a sodium-concrete test, the reaction layer looks like sandstone and gives the appearance of having been liquid. At room temperature the layer appears to have substantial mechanical strength. The layer rapidly becomes quite thick as the sodium attack on concrete proceeds. Eventually, the layer is thick enough to prevent contact between sodium and concrete. The available evidence from moderately large scale experiments strongly suggests that the presence of reaction products limits both the rate and depth of penetration of sodium into concrete.

Q97. What is the effect of orientation of the concrete surface?

A97. (Swanson) The orientation of the concrete surface (i.e., horizontal or vertical) determines whether the reaction products can form a protective layer. If the concrete surface is horizontal, then reaction products will reduce the extent of sodium attack. On the other hand, if the concrete surface is vertical, greater penetration should be observed under otherwise identical conditions. The available experimental data confirms this view and provides strong support for the role of reaction products in limiting sodium penetration into concrete.

Q98. Will the extent of sodium-concrete reactions vary significantly with the type of concrete?

A98. (Swanson) The choice of concrete aggregate (e.g., basalt, magnetite, and limestone) affects the extent of sodium-concrete interactions. Originally, the Applicants proposed the use of

calcitic limestone aggregate concrete; more recently, the use of dolomitic limestone aggregate (higher magnesium content) concrete has been suggested. Recent experiments at HEDL and Sandia suggest that the use of dolomitic concrete will not present any greater containment challenge than the use of calcitic concrete. Although the dolomitic concrete database is limited, the material appears to behave comparably to calcitic concrete with regard to sodium. The effect of selecting dolomitic aggregate rather than calcitic aggregate on aerosol behavior is discussed in response to Question 114 below.

Q99. What is the effect of core debris on concrete?

A99. (Swanson) Fully oxidized core debris will not react chemically with concrete, but will thermally penetrate by melting the concrete. The interface betwen the core debris and the molten concrete will not greatly exceed the temperature of the melting concrete, about 2200°F, because of the high thermal conductivity of molten concrete.

Core debris by itself can penetrate 15-25 feet into the 26-foot thick basemat over a period of time on the order of months, according to TRUMP heat transfer calculations. The average penetration rate is about 0.85 inches per hour for 70 hours, and falls off substantially thereafter.

Q100. What are the combined effects of core debris and sodium on concrete?

A100. (Swanson) There are two cases:

1. Core debris coolable - In this case the core debris transfers its heat to the sodium which reacts with the concrete as previously described. The presence of the heat source (debris) immediately adjacent to the reaction front may cause some slight increase in penetration rate. We have allowed a factor of five in penetration rate to take care of this. i.e., we have used one inch per hour whereas the experiments with heated sodium would be bounded by 0.2 inch per hour.
2. Core debris not coolable - Sodium vapor prevents liquid sodium from contacting the particles of core debris. Liquid sodium is also excluded from the reaction zone by vapor. The penetration of the debris into the concrete is a thermal effect. As described above in response to Question 99 above, the rate would average less than one inch per hour.

These considerations indicate that the proposed realistic upper bound penetration rate incorporates a margin sufficient to encompass the combined attack of sodium and core debris on concrete.

Q101. Is there any problem in applying the results of the small scale sodium-concrete tests that have been conducted to the size of the CRBR reactor cavity?

A101. (Swanson) Sodium-concrete tests have been conducted on a small scale. Scale effects have not been observed in tests in a size range from one to three feet in diameter. However, this is smaller

than the reactor scale by an order of magnitude. Sodium penetration into concrete appears to be limited by the formation of a protective layer of reaction products. It has been surmised that this layer could be swept away by turbulent convection currents in an accident occurring on a large scale. However, experiments have been performed where the concrete surface was oriented vertically so that a protective layer cannot adhere. These experiments establish a maximum long term penetration rate of 0.2 inch per hour, which is still bounded by the proposed penetration rate. Therefore, the results of these tests can be applied with confidence to the CRBR reactor cavity.

Q102. Will concrete cracking and spallation affect the concrete penetration rate?

A102. (Swanson) The extent of cracking and spallation may be scale-dependent. Also, the effect of rebars on cracking and spallation may be scale-dependent. Therefore, the small scale tests were conducted in such a way as to reproduce the temperature and stress fields of a large event. The results suggest that the proposed penetration rates will envelope large scale events. Experiments at HEDL and at Sandia indicate that spallation processes will generate only small chunks of concrete.

It is likely that substantial amounts of concrete will be removed from the initial surface by cracking, based on observations in tests at HEDL. This is apparently related to the initial high temperature

gradient, and does not continue after temperatures become more uniformly distributed.

Sodium does not penetrate small cracks in concrete, and hence such cracks do not provide a short-cut path for increased interactions.

Rebars, observed in one test at HEDL, did not increase cracking. Their presence should limit the extent of spallation.

The effects of cracking and spallation have been taken into consideration by the Staff, to the extent appropriate, in the high rate of initial concrete penetration modeled by the Staff.

C. Aerosol Analysis

Q103. Are there experimental bases and well developed theoretical interpretations to aid in the understanding of aerosol behavior?

A103. (Long) Yes. The experimental basis for sodium oxide aerosols is very broad and there are also aerosol experiments dealing with fuels and mixed materials. The theoretical treatment of agglomeration is based on kinetic theory of collisions as formulated by Smoluchowski and many others. The Staff has relied principally on the references cited in the testimony in the LWA-1 hearings (Tr. 2523) and on the user's manuals for the HAA-3, HAARM-3 and MSPEC aerosol codes.

Q104. How are aerosols of sodium compounds formed during the accident scenarios under discussion?

A104. (Long) When sodium vapors come in contact with O_2, CO_2, or H_2O vapor, small particles of the reaction products, sodium oxides, sodium carbonate or sodium hydroxide are formed. In the following discussion these will be referred to collectively as sodium oxidic aerosols. These particles are initially in a size range of less than one micron in diameter. Their subsequent behavior in regard to agglomeration, fallout and plateout follows the characteristics of aerosol behavior.

Q105. What principal roles do aerosols of sodium compounds play in the accident?

A105. (Long) There are three principal roles: (1) they serve as carriers for particles of fuel and fission products; (2) accumulations of these fine particles can create interference problems for mechanical or electrical devices or by blocking flow channels; and (3) they may serve as heat transfer barriers when deposited on surfaces.

Q106. During which phase of the accident is the ability of aerosols to carry fuel and fission products important?

A106. (Long) This is particularly important during and after the period when the sodium boils away in the reactor cavity. Fission products other than noble gases that are released prior to or during the boiloff period have been found to coagglomerate with the sodium oxidic aerosols in the containment building and their subsequent behavior will be governed by the movements of the combined aerosols.

A certain fraction of these will fall-out or plate-out on surfaces, from which their availability for subsequent release will be greatly reduced. Fission products that will be likely to boiloff before or during the sodium boiloff period include the cesium - rubidium group, the halogens, the arsenic-antimony - tellurium group, and a small fraction of the refractory materials, including plutonium oxide. It is estimated that 80 to 99% of the amount of these materials released to containment could become unavailable for future release through the mechanism of coagglomeration and fallout with sodium compounds. As discussed in Answer 108 below, 70% of these materials has been conservatively assumed to be unavailable for release in the Staff's analysis.

Q107. Does the radioactivity carried by the aerosols affect their agglomeration and settling behavior?

A107. (Long) The energy of the radioactivity is insufficient to have a significant effect on the temperature of the aerosols. Recoil effects are also too insignificant to alter the agglomeration process. Aerosol particles carry small electrical charges which may be augmented by the radioactivity. However, the net effect of electrical charges is not as important as Brownian motion in determining the collision frequency of the particles and their agglomeration.

Q108. What is the potential for these aerosols to plug vital passages and mechanical equipment?

A108. (Long) The reactor cavity and the passages from this cavity to the containment building are filled with an inert atmosphere (less than 2% oxygen) and consequently the amounts of oxide aerosols that can form are very limited. The capacity of the cavity vent tubes is ample to provide for passage of these limited amounts of oxide aerosols without plugging.

When sodium vapor issues from the reactor cavity to the containment building, it is completely oxidized by the air atmosphere. Not all of the sodium will reach the containment, since some reacts with the concrete. It is estimated, however, that up to a million pounds of aerosols could be generated in containment from the primary sodium. It is conservatively calculated that at least 70% of this would be deposited as fall-out and plate-out on walls and floors. This leaves 300,000 pounds of airborne particulate as the maximum amount that would have to be transferred to the cleanup system. The Applicants have committed to demonstrate that the cleanup system can handle this amount (CRBRP-3, Vol. 2, p. 2-8). Vent pathways from the containment building to the cleanup system are two straight pipes of extremely large (36 inch) diameter. These pipes are unlikely to plug because of their size. In view of the large size chosen for these passages and the Applicants' further commitment to demonstrate the acceptability of the capacity of the system, the plugging of these pathways need not be considered a probable mode of failure.

The operation of mechanical equipment within the containment building is not required during or after the stage when large quantities of aerosols are produced in the core-melt accident.

Sampling tubes for determining the composition of the atmosphere can be designed with some prefiltering and can be back-flushed if necessary to prevent plugging.

Q109. What heat transfer effects are expected as a result of the deposition of sodium oxide aerosols on the walls?

A109. (Long) In the worst case projections, aerosol particles that deposit on and adhere to the containment building walls are assumed to behave like loosely consolidated insulating materials. The deposition is expected to be controlled by thermophoresis, which is the term given to a diffusional process driven by a temperature gradient. The deposits would increase the thermal resistance of the containment inner shell which in turn affects the heat removal capability of the annulus cooling system. Since some of the deposition will consist of sodium hydroxide which melts at 650°F, and the containment atmosphere may exceed this, partial melting and consolidation of wall deposits may occur. This would reduce the fluffiness and insulation value of the deposits. Since the extent of consolidation is unknown, this is a favorable but not a quantifiable effect and has not been included in heat transfer calculations.

Q110. How has the insulating effect been calculated, and what is the calculated containment temperature?

A110. (Long) Thermophoresis parameters in the MSPEC aerosol code were adjusted to agree with observed wall plating data from HEDL. This permits a calculation of the mass deposited on the walls as a function of time. In order to estimate the insulation value of the deposits, a density must be attributed to them (the density of wall deposits has not been measured). It was therefore assumed that the wall deposits would include the maximum amount of porosity consistent with their holding together on a vertical surface. The containment atmosphere temperature was calculated throughout the duration of the accident while a material of 90% porosity and low conductivity was gradually added to the surface as calculated by MSPEC up to a thickness of 1.2 inches. The maximum containment temperature was estimated to be 1550°F in the base case scenario with these deposits.

The value of 1550°F represents a higher temperature than that for which the severe accident equipment was originally scheduled to be qualified, but is not a level that would be beyond qualification feasibility. Further work is being performed on the analyses of variations of the base case scenario and their effects on containment temperature. This further analysis will be reviewed by the Staff at the OL stage. The Applicants have committed to qualify the containment and other necessary equipment to whatever temperatures are finally determined to represent the accident.

The time when venting is required is only advanced by 2 or 3 hours, to 33 hours at the earliest, due to the deposits.

Q111. If a region of this wall deposit were to become locally dislodged, would the excess temperature to which that region of the wall is exposed cause the containment shell to fail?

A111. (Long) No. Calculations have shown that under the conditions existing in containment, the stress due to a localized overheating would be relieved by yielding before buckling or failure occurs.

Q112. What is the likelihood of resuspension of sodium oxidic aerosols after they have been deposited?

A112. (Long) In response to questions that the Staff raised with regard to the asymmetric turbulent flow field set up by the burning sodium vapor, the Applicants have performed analyses of flow velocities throughout containment. These analyses have indicated that velocities at floor level were on the order of 6 ft./sec. (4 mph). During venting, air velocities at floor level are expected to be, at most, 50% higher. The Staff has accepted the Applicants' analyses as providing a reasonable evaluation of this phenomenon.

Since deposited and resuspended aerosols are generally coarser particles than the original suspension, it is unlikely that these low velocities would cause much material to be elevated to the vents, located 50 feet above the operating floor.

Q113. What conclusions has the Staff reached as to the analyses of the role of aerosols in core disruptive accidents?

A113. (Long) The Staff has concluded that the well-developed collision theory of aerosol behavior has been appropriately applied to the analysis of CDAs for CRBR, in analyzing the fallout of radioactive materials, the plugging of important passageways, and the insulation effects of aerosol deposits.

Q114. Please respond to Board Question 16, which states as follows:

> The SER discusses the impact of aerosol behavior on containment shell cooling. The Staff is requested to comment on whether changing concrete aggregate from calcitic to dolomitic limestone could significantly alter the behavior of the aerosols, and explain the basis for the answer.

A114. (Long, Swanson) In order for a material to behave like an aerosol, it must be in the form of very fine particles, of the order of a few microns in diameter or less. In these sizes it can remain suspended in the atmosphere for an appreciable length of time and can participate in the Brownian motion and agglomeration that are typical of aerosols.

In the containment volume, aerosols are formed from the combustion of sodium vapors and their subsequent condensation, the principal mechanism by which such fine particles are produced. The vapor is predominantly sodium combustion products (NaOH, Na_2O), and contains only minute traces of concrete materials because of their lower volatility. The layer of wall deposits thus consists primarily of sodium oxidic compounds with traces of concrete materials.

After the sodium boil-off period, the rate of formation of concrete aerosols as a result of core debris - concrete interactions is very much less than the earlier rate of formation of sodium oxidic aerosols, because of the low volatility of the concrete. The Staff's calculations of the effect of aerosols on containment shell cooling have therefore only addressed the period during which sodium is boiling in the reactor cavity.

The insulation value of a wall-deposit of non-metallic materials depends on its porosity more than on its chemical composition. The Staff accepts as most reasonable the calculations in which a high value of porosity (90%) was assumed, as discussed above in response to Question 110. In view of the high porosity and low concrete fractions, the insulation value of the deposits is independent of whether concrete with calcitic or dolomitic aggregate is used.

The Staff also investigated whether the use of calcitic or dolomitic aggregate would make a difference in the rate of attack of sodium on concrete (see response to Question 98 above). No differences of this nature were found. From this information it was inferred that the rate of buildup of insulating wall deposits would also be independent of whether concrete with calcitic or dolomitic aggregate is used.

Having reached these conclusions, the Staff has determined that changing concrete aggregate from calcitic to dolomitic limestone would not significantly alter the behavior of the aerosols.

D. Hydrogen

Q115. What is the significance of hydrogen in connection with core melt-through accidents?

A115. (Long) Free hydrogen can be generated from the reaction of water with sodium. The concrete contains water which can be released when the concrete is heated. The hydrogen generated by the reaction of water released from concrete with sodium is a source of pressure in the containment atmosphere by burning at low concentrations, and could potentially rupture the containment if allowed to build up and ignite at concentrations greater than approximately 9%. The burning of hydrogen must also be regarded as a potential source of damage to the vent cleanup system which might result from burn pressure. Accordingly, the containment design includes a vent purge system to control the hydrogen concentration inside containment. A 6% hydrogen concentration has been selected as an upper limit allowed in containment to avoid reaching potentially damaging concentrations. The 6% hydrogen concentration value is to be used as a criterion for deciding when to vent containment.

Detonation of hydrogen is not considered a problem since this can only occur at concentrations well above 9%. To reach those hydrogen

concentrations would require a failure to properly operate the containment vent/purge system.

Q116. How has the Staff calculated the concentration of hydrogen in the containment atmosphere?

A116. (Long) The CACECO code has been used. This code calculates the temperature profile in the concrete when its surface is exposed to a heat source. Each change in temperature at each position in the concrete is related to a water content of the concrete at that point. Reductions in these water contents as the concrete temperature increases are interpreted as releases of water, which is then made available for reaction with sodium.

The Staff and its consultants have worked with the CACECO code and have reviewed the supporting documentation and experimental data. We have concluded that the code provides reasonable guidance in determining the rates of hydrogen release that are likely to be encountered.

Q117. In the base case thermal margin beyond design basis (TMBDB) scenario, what are the peak concentrations of hydrogen that would be reached in the containment atmosphere?

A117. (Long) Applicants have submitted base case calculations with the CACECO code that indicate a peak hydrogen concentration in the containment of 4.4%, allowing for some holdup dissolved in the

primary sodium. Without this holdup, the concentration could reach 5.2%.

Based on its review of the CACECO code, the Staff believes that these figures represent acceptable concentrations for the purpose of accident calculations. In the base case scenario, hydrogen is burned at the above concentration as the sodium vapor boil-off becomes sufficient to ensure ignition. Hydrogen burning at these lean concentrations would not generate pressures that would jeopardize the containment structure. Subsequent purging assures that hydrogen will not accummulate beyond its flammable limit.

Q118. Have variations on this base case scenario been examined to determine if a worse hydrogen situation could develop?

A118. (Long) Many features of the scenario have been varied without altering this picture of the resultant hydrogen behavior other than slightly foreshortening or extending it. Variations have included the sodium-concrete reaction rates, the thermophysical property data for concrete, the reactor vessel and cavity floor liner penetration times, and the fractional failed area of the cavity floor liner.

Studies of the effect of reduced decay heat, as might occur early in cycle, or in a scenario where core material was partially retained in the primary system, illustrated a somewhat different but still acceptable hydrogen behavior. Reduction to 60% decay heat extended the ignition time to 24 hours, at which time hydrogen concentration

had increased to 6.1%. Further reduction in decay heat extended the ignition to even later times, but in these cases, purging can be initiated to control the hydrogen concentrations after 24 hours. These lean concentrations do not develop large burn pressures, and it is considered feasible to design a vent cleanup system capable of accommodating these pressures.

E. Structural Response

Q119. Has the Staff evaluated CRBR structures to determine their response to loads generated during a core-melt accident?

A119. (Butler, Holz) Yes. The Staff reviewed the analysis performed by the Applicants as set forth in CRBRP-3, Vol. 2, incorporated by reference in PSAR Section 1.6. In addition, the Staff performed an independent analysis of the containment building pressure boundary and the thermal loading on the containment building. The Staff's review is summarized in Appendix A to the CRBR SER, Section A.4.6 (p. A.4-10) and in Attachment 2 to SER Appendix A.4, "Final Technical Evaluation Report of Thermal Margin Beyond Design Basis Features".

Q120. Which CRBR structures were analyzed to determine their response to loads generated during a core-melt accident?

A120. (Butler, Holz) Plant structures that must be considered during this accident are the containment building pressure boundary, the confinement building, containment internal concrete structures, and the reactor cavity wall liners and pipeway cell liners. These have

been analyzed. Of these structures, only the containment building pressure boundary and confinement building must remain intact indefinitely following a core melt accident. In this accident it is not important that the concrete internal structures or cell liners survive after sodium boildry because their function is not required after boildry.

Q121. What are the specific loads that act on the steel containment boundary?

A121. (Butler, Holz) The critical portion of the containment building, from a structural viewpoint, is the free-standing cylindrical steel shell above the operating floor that has an ellipsoidal/spherical head. The boundary is not loaded until a sodium/hydrogen flame begins burning above the operating floor. Starting at that time, the temperature of the steel shell and the pressure acting on it increase until the containment vent/purge systems are actuated. After that, the pressure loads are essentially zero and the temperature of the structure increases little, if any. The Applicants have calculated that the maximum pressure during the base case TMBDB scenario is less than 24 psig; the Staff's review of that determination has found it to be acceptable.

Q122. Is it feasible to design a containment which is capable of withstanding the TMBDB pressure load?

A122. (Butler, Long, Holz) Yes. In fact, the currently proposed containment design has been evaluated by Applicants and Staff and has been

determined to be capable of withstanding a TMBDB pressure load of 40 psig. The Staff's review at the OL stage will ensure that the final containment design is capable of withstanding TMBDB loads.

It should be noted that the design of the containment is based upon design basis accidents and is specified to be 10 psig. The containment is designed to withstand the design pressure in combination with other loads such as those experienced during a Safe Shutdown Earthquake (SSE). However, the thickness of the containment shell is more strongly influenced by the SSE component of the loads than by the pressure component. Therefore, when the only primary loads present, as in a core melt accident, are pressure and dead weight, the containment shell has a pressure capability that is approximately four times higher than the design pressure.

Q123. How were temperature distributions in the containment and confinement structures determined?

A123. (Butler, Long, Holz) The TRUMP heat transfer computer code was used by the Applicants to calculate temperatures in the containment and confinement structures during the core-melt accident. The thermal forcing function for the TRUMP model was based on heat loads predicted by the CACECO computer code, increased by 10%. The model included effects of the annulus cooling system. The Staff reviewed the Applicants' model with an independent analysis using the ABAQUS computer code. The Staff has determined that the Applicants'

predictions give conservatively high temperatures for the sodium aerosol deposition thickness assumed.

Q124. What kind of analyses have been performed to ensure that the confinement building will not fail?

A124. (Butler, Long, Holz) The confinement building was modeled by the Applicants with the finite element computer code ANSYS. The Staff has determined that the design techniques are conservative as are the failure criteria that, for the temperatures experienced in the confinement structure, are based on the American Concrete Institute (ACI) building code requirements. The primary loads in the confinement building are from dead weight and are low. The loads from the thermal gradients are secondary in nature (that is, they are self limiting because as the structure begins to fail, it becomes more flexible, thus reducing the internal stresses). The Staff has determined that it is feasible to design a confinement structure capable of withstanding TMBDB conditions.

Q125. Do the internal structures have to remain intact in a TMBDB event?

A125. (Butler, Holz, Long) The reactor cavity walls and reactor vessel support ledge must remain intact for at least 50 hours, to ensure that the heat capacity of the pipeway cell walls is employed in extracting heat from the gases generated in the reactor cavity. All the structures except the reactor cavity floor and floor liners have to remain intact until containment venting begins. Other structures, except for those that support the containment pressure

boundary, can fail after sodium boildry. The structures that support the containment shell must survive indefinitely. The Staff's review indicates that the structural design criteria selected by the Applicants, as described above, are satisfactory. Further, the Staff has concluded that it is feasible to design these internal structures consistent with these criteria.

Q126. What would be the consequences if the reactor cavity wall liners and pipeway cell liners failed before the times specified in the criteria for the core-melt scenario?

A126. (Butler) As noted in response to Question 125 above, these liners have to remain intact until containment venting begins (assumed by the Applicants to be no earlier than 30 hours). In evaluating the core melt consequences, the Applicants have estimated that different portions of these liners fail starting no sooner than 30 hours. Earlier liner failures would have some impact on the rate of sodium/concrete reactions and could therefore have some impact on the required vent time and the survival times of certain internal structures.

Q127. What assurance is there that the reactor cavity wall liners and pipeway cell liners will remain functional for the periods of time assumed in the core-melt evaluation?

A127. (Butler) The Applicants are performing, at the Staff's request, a combined analysis and test program to verify analytical models and to substantiate the proposed cell liner failure criteria (i.e.,

design limits). If in-depth analysis of the present liner design shows that it cannot meet the appropriate failure criteria, several fallback positions have been identified by the Applicants that are expected to lead to a cell liner design that is acceptable to the Staff.

F. Annulus Cooling System

Q128. What is the function of the annulus cooling system?

A128. (Butler) The annulus cooling system controls the temperature of the containment and confinement structures. Basically, this system breaks the direct thermal path in the annulus between those two structures by introducing a significant heat removal path. The annulus cooling system is to be designed to remove approximately 5×10^7 BTU/hr (15 MW) from the containment. This heat removal rate is sufficient to maintain the containment at a safe temperature.

Q129. What are the components of the annulus cooling system?

A129. (Butler) Redundant fans, located in the Reactor Service Building, supply air to the bottom of the annulus space. Partitions in the annulus force the air through a spiral flow path around the containment shell. The resulting flow paths ensure sufficient air velocities to achieve the needed heat removal rates. Leak-tight motorized dampers are provided at the system boundaries. All power requirements of the system are supplied from Class IE redundant power systems. All active components are provided with backup capabilities so that failure of any one active component will not

preclude 100% operation of the system (see CRBRP-3, Vol. 2, Sec. 2.2.10).

Q130. Has the Staff evaluated the annulus cooling system and, if so, what are the conclusions reached by the Staff's evaluation?

A130. (Butler, Long) The Staff has evaluated the currently proposed annulus cooling system to determine that a system with the required capacity is feasible to design. To estimate the annulus cooling system capacity, the Staff performed calculations assuming steady state conditions and average system parameters and boundary conditions. Results show that the cooling system proposed by Applicants can remove approximately 5×10^7 BTU/hr (15 MW) from the containment at 24 hours into the base case core-melt scenario. This heat removal capability compares favorably with the decay heat load during the core melt, which is 6 MW at 24 hours. Accordingly, the Staff has determined that the design criteria for this system are acceptable, and that it is feasible to design an acceptable annulus cooling system.

G. Vent Cleanup System

Q131. What are the primary components of the containment vent cleanup system?

A131. (Butler) The cleanup system consists of an air washer (quench tank), a venturi jet scrubber, a high-efficiency wetted-fiber-bed scrubber, and a redundant set of blowers. Gases and aerosols from

the containment building pass through these components sequentially before the cleaned gas is released to the atmosphere.

Q132. What are the functions of these primary components?

A132. (Butler) The air washer ensures that all sodium oxide is converted to sodium hydroxide and that the gas temperature is reduced from 1100°F to 160°F. It will also remove some of the larger particles from the gas. The venturi scrubber removes remaining large particles; the wet-fiber-bed scrubber removes smaller particles and provides sufficient contact to remove condensible vapor-phase species (CRBRP-3, Vol. 2, Sec. 2.2).

Q133. Are these or similar components commonly used by other industries?

A133. (Butler) All of the major components are widely used and are commercially available. The quench tank venturi-scrubber combination is used in many process applications, the most common being combustion related (such as incineration and coal burning). Other off-gas cleaning uses include blast furnaces, coke operations, and gray iron foundry operations.

Fibrous scrubbers are most commonly used for vapor removal. However, they are excellent devices for collection of submicron particles. These scrubbers are often used for removal of mist from acid plant gas streams.

Q134. Has the Staff evaluated the containment vent cleanup system?

A134. (Butler, Long) Yes. The Staff has evaluated the system design criteria to determine the feasibility of achieving an acceptable design with the required cleanup efficiency.

Because the system is not completely designed, an in-depth analysis of the complete system is not possible. To determine system feasibility, we reviewed results of tests AC1-AC6 that were performed at Hanford Engineering Development Laboratory (HEDL-TME-81-1). The systems tested in the AC test series included the three major components proposed for the CRBR design, and utilized an aerosol from a sodium spray fire. We also reviewed empirical equations developed from these test results to determine whether large differences in system efficiency might result when the components are scaled up to the size and put into the specific configuration required for CRBR.

Q135. What were the Staff's conclusions concerning the system design criteria and feasibility of designing an acceptable containment vent cleanup system?

A135. (Butler, Long) We have concluded that the design criteria are acceptable and that an acceptable cleanup system can be designed. Components similar to those proposed for the CRBR cleanup system are readily available and are widely used. They give satisfactory results when appropriately sized for the specific filtration demand. Thorough similtude analyses need to be performed to apply available data to the final design, and to allow differences in component size

and operating characteristics and aerosol parameters to be appropriately taken into consideration.

Q136. What are the Staff's conclusions regarding core melt-through studies?

A136. (Long) Seven principal conclusions have been reached by the Staff. These are as follows:

1. Accidents associated with core melt-through following loss of core geometry have been adequately analyzed.
2. The general course of events in such accidents has been determined.
3. Possible variations at each key juncture of events have been studied to determine the conservative bounds of the accidents.
4. Detailed studies of each phase of the accidents and its variations have been accomplished, including vessel penetration, concrete interactions with sodium and core debris, aerosol release and behavior, hydrogen production and behavior, challenges to structures and equipment, key mitigating features, and radiological releases.
5. The understanding of these events is adequate to conclude that sufficient attention has been given to these accidents in the Staff's CDA analyses.
6. As is discussed in Part IV below, the overall results of the analyses show that radiological doses to individuals at the LPZ boundary in the event of a Class 1 Category II CDA can be expected to be within 10 C.F.R. Part 100 guidelines.

7. The design criteria developed by the Applicants for the features to specifically mitigate CDAs are acceptable, and it is feasible within the design concepts or fallback possibilities to meet these criteria.

IV. RADIOLOGICAL CONSIDERATIONS ASSOCIATED WITH CORE DISRUPTIVE ACCIDENTS

Q137. How is Part IV of this testimony organized?

A137. (Swift, L. Bell, Hulman) Part IV of this testimony addresses, in sequence, (1) the radiological source term developed by the Staff for evaluation of CDAs and (2) the radiological consequences of these CDAs.

A. Radiological Source Term

Q138. Has the Staff developed a radiological source term that is representative of a CDA at CRBR?

A138. (Swift) Yes. The radiological source term developed by the Staff is representative of both energetic and non-energetic CDAs. The postulated CDAs do not include either an initial head release or an early failure of the containment or vent/purge system. This source term is consistent with the conclusions described above concerning the energetics expected in a CDA at CRBR. The source term corresponds to a Class 1 CDA involving Category II primary system failure, i.e., the "primary system [is] initially intact, but later fails due to ineffective long-term decay heat removal (of the order of hours or more)," as defined in Appendix J of the FES Supplement.

Q139. What is the sequence of releases of radioactive materials from the reactor cavity that is expected to occur in a core disruptive accident?

A139. (Swift) The first significant release consists of some hot gases and sodium oxide aerosols, from the initial reaction of the primary sodium with the reactor cavity atmosphere, plus fission product noble gases. This is followed by a flow of cavity-atmosphere nitrogen plus hydrogen from the sodium-concrete reaction; this flow contains a small fraction of aerosols and vapors of sodium, cesium, rubidium and other volatiles. After the sodium in the reactor cavity begins to boil, the fraction of sodium vapor increases significantly; at 6 g/m^3 it causes ignition of the hydrogen in the containment atmosphere. As the rate of boiling increases, so does the amount of entrained aerosols, including sodium iodide and solids such as plutonium oxide. This transport of materials in the flow of hydrogen and sodium vapor to the containment atmosphere continues until all the sodium is boiled away, a point in time called "boildry". The period of sodium boiling may extend from 70 to 130 hours, depending primarily upon the rate of reaction of the sodium with concrete. As boildry approaches, the materials dissolved and suspended in the remaining sodium become more concentrated and the boiling temperature increases. As the last of the sodium boils away, other volatile materials (tellurium, selenium, antimony and arsenic), which are less volatile than sodium, are also driven off.

When the sodium has boiled away, the nature of the transport changes. The sodium is no longer present to react with the water and carbon dioxide from the concrete; what remains is the core debris with its radioactive decay heat generation, plus many tons of sodium hydroxide, sodium carbonate, sodium silicate, calcium oxide, magnesium oxide, iron oxide or more complicated combinations of these, plus perhaps a half ton of fission products. As the concrete is heated and decomposed, water vapor and carbon dioxide are released, providing a flow of gases to transport radioactive materials to containment. However, the sparging of solids by this gas flow carries insignificant amounts of materials to the containment. This type of release to containment begins after boildry (70 to 130 hours after accident initiation) and may continue for several months, until the decay heat generation rate has diminished to the point where it is insufficient to cause further melting or dissolution of concrete.

The sodium boiling stage of this sequence is perturbed by the containment venting. The pressure buildup in the containment, to 2 or 3 times its design pressure, is relieved at about 24 hours by allowing it to vent down to normal atmospheric pressure through the containment cleanup system. Relieving the pressure results in a short period of more rapid sodium boiling and venting of the reactor cavity and pipeway cell atmospheres to containment. The short-term overcooling due to the ventdown leads to a period of reduced sodium vapor flow; this lasts for only a few hours, and then a strong vapor

flow resumes and continues at a relatively steady rate until boildry.

Q140. How did the Staff model the release of noble gases to containment?

A140. (Swift) The Staff's model postulates that 100% of the noble gases are released to the containment atmosphere at the initiation of the accident.

Q141. Does that model represent what the Staff expects would occur in a CDA?

A141. (Swift) In a CDA, almost all the noble gases would be released from the reactor core at the time of core disruption. Some however, might move into the primary heat transport system when the reactor guard vessel is penetrated and the primary sodium is drained into the reactor cavity. In the reactor cavity, the sodium level would initially be well above the level of the hole in the guard vessel where the sodium drained out, and thus some of the noble gases might be trapped in the reactor vessel. Such effects might delay the release to containment of a portion of the noble gases, allowing some time for their radioactive decay in a location which affords more shielding.

Also, rather than at t=0.0, the noble gases would not be released to containment from the reactor cavity until the vent to containment opens, modeled as 1000 seconds after initiation of the CDA. Further, even with the vents open, the noble gases would not be

swept out of the reactor cavity and pipeway cell atmospheres all at once, but only over a period of hours.

Because the noble gases contribute only a fraction of the doses, these conservatisms in the model result in only a minor conservatism in the calculated doses.

Q142. How did the Staff model the release of halogens to the containment?

A142. (Swift) The Staff's model postulates that 100% of the halogens are released into the primary sodium at the time of core disruption, and are subsequently 100% transported to the containment atmosphere at a rate directly proportional to the rate of sodium vapor release to the containment. The radioactive iodine is the halogen of concern; the radioisotopes of bromine contribute negligibly to the doses.

Q143. Does that model represent what the Staff expects would occur in a CDA?

A143. (Swift, Long) When the iodine is released from the core materials, it will interact with the sodium and form sodium iodide, which is somewhat less volatile than sodium. Initially, this is a dilute solution, there being about 500,000 kg of primary sodium and only about 7 kg of iodine. Therefore, the reaction will tend to go 100% to sodium iodide. Because sodium iodide is less volatile than sodium, it is boiled off at a lesser rate than is the sodium. Research results indicate that by the time 50% of the sodium has boiled off, only 10% of the iodine has boiled off. The Staff's

model, then, releases the iodine to the containment atmosphere earlier than is realistically expected. Thus, the source term for iodine has had less time than is realistically expected to be diminished by radioactive decay before it is available to be leaked or vented to the environment (through a cleanup system) in the Staff's model. This is a conservatism, i.e., it tends to cause the calculated doses to be higher than are realistically expected.

A further conservatism is employed in postulating that 100% of the iodine is released to the containment atmosphere. It is possible that some iodine would be retained on surfaces within the primary heat transport system or within the reactor cavity and pipeway cells before reaching the vent to the containment atmosphere. Also, at the bottom of the reactor cavity, where the sodium and core debris have been reacting with the concrete, considerable quantities (tens of tons) of sodium hydroxide and other compounds are formed; a fraction of the iodine may also be retained within this mass of reaction products.

Uncertainties associated with this model are:

(1) The iodine (and sodium, etc.) released to the containment atmosphere before venting has a greater probability of being depleted from the containment atmosphere by plateout and fallout than does the iodine released to the containment atmosphere during venting or during the operation of the vent/purge system. The Staff's model assumes that about 20% of the sodium (and thus

20% of the iodine) is boiled up to the containment before initiation of venting, and that most of this 20% would have been depleted from the containment atmosphere before venting is initiated. However, even if all the iodine is assumed to be released to containment during venting and vent/purge operation, the quantity released to the environment would not be increased by more than 25% over that which is assumed in the Staff's model, resulting in approximately a 50 rem increase in dose to the thyroid.

(2) During the period of boiling of sodium, the rapid rate of injection of large quantities of sodium aerosols into the containment atmosphere (2 to 5 tons per hour) forces such a rapid depletion of the aerosol by fallout and plateout that the relative fraction of aerosol vented to the cleanup system is not more than 30%. If some of the iodine, as sodium iodide, remains in the pool until boiling and is driven off only as the temperature of the residual materials increases, it may arrive in the containment atmosphere at a time when depletion by fallout has slowed significantly, thus allowing a larger fraction to be vented to the cleanup system (assuming a uniform vent/purge flow). If 10% of the iodine is vented to the cleanup system without depletion in the containment, the calculated thyroid dose would show a net increase of approximately 80 rem.

Having considered these uncertainties and the conservatisms contained in this model, the Staff has determined that the releases to the environment calculated by this model represent that which is expected to occur in a CDA (corresponding to the Class 1, Category II CDA discussed in the FES Supplement, Appendix J, Table J.2). In addition, the Staff has determined that sufficient improvements in the containment cleanup system filtration efficiency are easily achievable, should the uncertainties in the Staff's model remain at the OL stage of review.

Q144. How did the Staff model the release of the volatiles cesium and rubidium to the containment?

A144. (Swift) The Staff's model postulates that 100% of the cesium and rubidium is released to the containment atmosphere at one time, i.e., at 10 hours after core disruption.

Q145. Does that model represent what the Staff expects would occur in a CDA?

A145. (Swift) Research results indicate that, because cesium and rubidium are more volatile than sodium, as the sodium boils off, dissolved cesium and rubidium boil off at a faster rate than the sodium. These results indicate that almost all the cesium and rubidium boil off by the time 10% of the sodium has boiled off. It is expected that almost all the cesium and rubidium would arrive in the containment around the time of ignition of the vent plume, i.e., around the time of the initial hydrogen ignition. This time is

sufficiently early that the containment atmosphere would have been almost completely depleted of the cesium and rubidium aerosols by fallout and plateout before the time venting is initiated. This is reflected in the Staff's model.

It has been suggested that the release of some of the cesium might be delayed. We have performed a calculation of a release of cesium-137 without depletion by fallout in the containment; if 100% of the cesium were released to the cleanup system, the resulting 30-day LPZ doses would be 5 rem to the liver and 3 rem to the whole body and other organs, in addition to those of our base case. The Staff considers that its model represents that which is expected to occur in a CDA. However, this sensitivity analysis indicates that even if no fallout of cesium-137 occurs, the resulting doses do not cause the dose consequences of the CDA to exceed 10 C.F.R. Part 100 guidelines.

Q146. How did the Staff model the release of other volatile fission products to the containment?

A146. (Swift) Other volatile fission products are selenium, tellurium, antimony and arsenic. The Staff modeled these as 100% initially released into the primary sodium and later 100% transported with sodium vapor to the containment atmosphere at a rate directly proportional to the rate of the release of sodium vapor to the containment.

Q147. Does that model represent what the Staff expects would occur in a CDA?

A147. (Swift) These fission products are volatile, but less volatile than sodium; their volatility is like that of sodium iodide. Because of low fission yields, short half-lives, and small dose conversion factors, all radionuclides of antimony, arsenic and selenium and most radionuclides of tellurium can contribute little to doses. Some conservatism is added in the model by releasing these fission products earlier than is expected. If their release to containment is delayed until after boildry, only certain radioisotopes of tellurium would contribute to the doses, but without a significant effect on the total offsite dose.

Q148. How did the Staff model the release of the other radionuclides to the containment?

A148. (Swift) The Staff's model postulates that 0.16% of all other radionuclides (i.e., the Ba-Sr, Ru, and La groups, including Pu) are transported to the containment atmosphere at a rate directly proportional to the rate of sodium vapor release to the containment.

Q149. Does that model represent what the Staff expects would occur in a CDA?

A149. (Swift) The model simplifies the situation, but in general it is conservative. Most of the remaining radionuclides have volatilities like those of the fuel materials uranium and plutonium. (Barium and strontium are usually considered to be slightly more volatile.) The

uranium and plutonium dioxides are chemically quite resistant to the conditions in CRBR during and after core disruption. However, their nature is such that when molten or vaporized and quenched in sodium, they fragment into particles generally ranging in size downward from about one millimeter in diameter. An appreciable fraction, estimated to be as much as 15%, forms particles so small that they may remain in suspension in the sodium. Some will interact with the sodium and form sodium plutonate and sodium uranate, dissolved in the sodium. A small fraction of these particles, uranates, and plutonates will be entrained in bubbles passing up through the sodium and will then be carried up to the containment atmosphere with the flow of hydrogen and sodium vapor. Experimental data for sodium boiling rates comparable to those which might be expected in the reactor cavity indicates that the concentration of fuel materials transported away with the vapors is generally less than one part in one thousand of the concentration of the fuel materials in the boiling liquid sodium. Other data show that for higher boiling rates, the partitioning factor might be as much as ten times higher, i.e., perhaps one part in one hundred. The Staff assumed these values, 15% and one in one hundred partitioning, for the Staff's model; these values lead to a release to the containment of 0.15% of the core inventory.

Another mechanism for release of radionuclides to the containment could come into effect under certain conditions. In this mechanism, gases passing through a hot bed of core debris (perhaps as bubbles

through molten core materials) would entrain small particles of the core debris materials, principally by evaporating or subliming the materials. This mechanism is sometimes called gas "sparging"; sparging might occur after boildry. The Staff considers that the amounts transported by this mechanism would be insignificant, but has nonetheless included them in the model by adding 0.01% of the core to the 0.15% modeled as boiled up with the sodium, so that the fraction modeled as transported to the containment atmosphere with the sodium vapor totals 0.16%.

Q150. What is your conclusion as to the radiological source term developed by the Staff for a CDA at CRBR?

A150. (Swift) The Staff has assumed a simplified but adequate model of the releases of radionuclides to containment following a CDA. Though primarily realistic, the model incorporates a number of conservatisms which tend to increase the estimates of CDA doses. As discussed above, there are certain uncertainties associated with this model. None of these uncertainties other than those involving iodine could cause the 10 C.F.R. Part 100 guidelines to be exceeded; with respect to iodine, if uncertainties remain at the OL review stage, additional filtration capability can be provided in order to keep calculated doses within the 10 C.F.R. Part 100 dose guidelines.

B. Radiological Consequences Of CDAs

Q151. Has the Staff evaluated the radiological consequences of a CDA?

A151. (L. Bell, Hulman) Yes. The Staff reviewed the Applicants' analysis of the radiological consequences of a CDA. In addition, the Staff performed an independent analysis of a Class 1, Category II CDA, as defined in the FES Supplement, Appendix J. The parameters and results of the Staff's analysis are provided in Appendix A of the CRBR SER Supplement No. 2 (NUREG-0968, May 1983).

Q152. Do you consider that the CDA scenario the Staff has selected is appropriately conservative and bounding for Class 1 CDAs?

A152. (Swift, L. Bell, Hulman) Yes, because of the conservatisms inherent in the assumptions utilized in the Staff's analysis.

Q153. Please describe some of the important assumptions you used in modeling the CDA and contrast them with those of the Applicants?

A153. (Swift, L. Bell, Hulman) The scenarios selected by the Staff and Applicants, respectively, assumed the following:

1. The Staff assumed that venting of the containment starts 24 hours into the accident, whereas venting at 36 hours was assumed by the Applicants.
2. The Staff assumed that the containment is vented at a constant rate of 20,000 cfm over a three hour period, whereas the Applicants assumed that venting commenced at a flow rate of 24,000 cfm and decreased to about 8,000 cfm over the three-hour vent period.
3. The Staff assumed that any fission product debris suspended over the reactor cavity would fall back into the pool for

re-evolution during the pool boil-off, whereas the Applicants' model assumed that none of the fallout is re-evolved.

4. In the Staff's model, any activity released to the secondary containment for the first 24 hours of the accident and present in the secondary containment for the duration of the accident after 24 hours was released untreated to the environment at a flow rate of 100% of the secondary containment volume per day, starting 24 hours into the accident. It is not clear to us as to whether the Applicants made any assumptions with regard to this potential source of radioactive release.

5. The primary containment was assumed by the Staff to leak at the technical specification leak rate limit for the first 24 hours of the accident, and at one-half the technical specification leak rate after 24 hours (in addition to the vent/purge leakage). All containment leakage in the first 24 hours is leaked to the environment via the secondary containment and is treated by the annulus filtration system. All containment leakage (excluding vent/purge leakage) after 24 hours into the accident was assumed to be untreated during the course of the accident. It is unclear to us as to whether the Applicants included the technical specification leak rate as a radiological pathway in their CDA analysis.

6. The plutonium source term was assumed by the Staff to be 0.16% of the inventory, compared to the Applicants' source term of about 0.04% of the inventory.

7. The fission products released from the pool boil-off were assumed by the Staff to be released to the containment at a constant linear rate over a 120 hour period starting with the initiation of pool boiling at 10 hours into the accident. The Applicants used a variable release rate over 120 hours.
8. In the Staff's model, the fall-out rate was held constant during the course of the accident and was considered to be an average rate, but fallout was not initiated until 13 hours into the accident. The Applicants assumed a variable rate for fallout, and initiated fallout at 10 hours into the accident.
9. All of the noble gases (100%) were assumed by the Staff to be released at the start of the accident, and all of the cesium and rubidium were assumed to be released at the initiation of pool boiling at 10 hours into the accident, whereas all other volatile isotopes such as tellurium, iodine, etc., were assumed to be 100% released over the 120 hour pool boil-off period starting at 10 hours into the accident. The Applicants' assumptions in this area were very similar to the Staff's assumptions.

Q154. Which computer codes were used by the Staff in the calculation of the doses for the CDA?

A154. (L. Bell, Hulman) The Staff used the TACT V computer code to calculate both the DBA and CDA doses. The TACT V code has been validated through hand calculations and in comparing similar model

runs betwen the TACT III code and the TACT V code. The TACT V code has been documented in draft form.

Q155. Please describe the resulting doses calculated by the Staff, using the above assumptions?

A155. (L. Bell, Hulman) The results of the Staff's CDA dose calculations are provided in SER Supplement No. 2, at p. A.5.1. All of the resulting organ doses calculated are within the 10 C.F.R. Part 100 dose guidelines. The thyroid was the governing critical organ with an estimated dose of 192 rem. In addition, the dose guidelines for CRBR constitute a risk equivalent to the dose guidelines used for the siting of LWRs. It was determined that the doses computed for the above-described CRBR CDA constitute a risk no greater than that allowed for LWRs in the dose calculations performed to show LWR compliance with 10 C.F.R. Part 100.

Q156. Please respond to Board Question No. 1, which states as follows:

> In its Safety Goal Development Program announcement (48 Fed. Reg. 10772, March 14, 1983) the Commission stated that during the 90-day period (ending June 8, 1983) for public comment on the proposed evaluation plan "it is expected that preliminary information on new radiological source terms will become available..." (Id., at 10778). The Staff is requested to advise whether that information will be evaluated for any impact on this proceeding, and the reason for its answer.

Q156. (L. Bell, Hulman) The Accident Source Term Program Office plans to address the severe accident source terms for LWRs, and not for LMFBRs such as CRBR which involve different coolant, fuel and

design. A recent memorandum providing a status report on the LWR accident source term reassessment from William J. Dircks, Executive Director for Operations, to the Commissioners, dated June 6, 1983, is attached hereto ("Attachment 1 to CDA Testimony").

Two source terms were used to evaluate the CRBR design from a safety perspective: the source term used for Site Suitability and the controlled CDA (CDA Class 1) source term. In addition, source terms were developed to evaluate the environmental consequences of accidents at CRBR for the CRBR FES Supplement (October 1982). The Site Suitability Source Term (SSST) methodology parallels that used for LWRs and is based on TID 14844, a reference document footnoted in 10 C.F.R. Part 100. The SSST was used to bound DBAs for the site suitability determination in combination with engineered safety features. A change in the SSST or environmental source terms in a more conservative direction is not likely to result from the efforts of the Accident Source Term Program Office. If the Accident Source Term Program Office efforts do indicate that the CRBR SSST or environmental source terms may need to be modified in a more conservative direction, it is unlikely that the Staff's conclusions would change with respect to the suitability of the Clinch River site for a nuclear power reactor of the general size and type of the CRBR or the environmental consequences of CRBR operation. However, changes in the SSST or environmental source terms may require modifications to the design of certain engineered safety features,

such as the annulus filtration system. The Staff expects that such changes could be accommodated by the CRBR design.

The source term used for CRBR to evaluate the controlled CDA has no parallel in LWRs; this source term accounts for fuel configuration, aerosol behavior and deposition within the reactor cavity and vent ducts, etc. Such considerations were not explicitly evaluated for LWRs, but are expected to be factored into the development of the LWR severe accident source term by the NRC Accident Source Term Program Office. Since such considerations have generally been evaluated for the CRBR already, a re-evaluation of them in light of a revised LWR source term would not be expected to produce larger source term estimates for the CRBR. The Staff, therefore, concludes that it is unlikely that the Accident Source Term Program Office findings will appreciably alter the controlled CDA source term or affect the Staff's conclusion that the controlled CDA doses fall within 10 C.F.R. Part 100 guidelines.

Therefore, the Accident Source Term Program Office deliberations should not result in findings that would substantially impact this construction permit proceeding. However, in the unlikely event that a design change in the Annulus Filtration System and/or the Vent/Purge Filtration System (such as by increasing filtration efficiency) would be needed as a result of the Accident Source Term Program Office findings, it is expected that such changes could be accommodated by the CRBR design. The Staff will review the

conclusions reached by the Accident Source Term Program Office as well as the conclusions reached by other interested bodies with respect to this matter, and will ensure that appropriate consideration is given to those conclusions during the OL stage of review.

PROFESSIONAL QUALIFICATIONS

CARDIS L. ALLEN

I am currently a Senior Reactor Engineer in the Clinch River Breeder Reactor Program Office of the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission.

I received a Bachelor of Science and a Master of Science degree in physics from the University of Denver.

I worked in the area of LMFBR physics and safety analysis at General Electric's Atomic Power Equipment Department in San Jose, California from 1960 to 1964. During 1965 I worked on radiation detection systems at the U. S. Naval Radiological Defense Laboratory in San Francisco. I joined the Regulatory arm of the (then) AEC in 1965. Since that time I have worked on various assignments involving safety reviews for both LWRs and LMFBRs. I am a member of the American Physical Society and the American Nuclear Society.

PROFESSIONAL QUALIFICATIONS

OF

LARRY W. BELL

I am employed as a Nuclear Engineer in the Accident Evaluation Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. My responsibilities include the reviews and analyses of designs and operations of nuclear power plant systems to determine the acceptability of the plant safety.

I graduated from the University of Maryland in 1960, with a Bachelor of Science degree in Physics. In 1967, I received a Master of Science degree in Physics from the University of Maryland. In 1960, I joined the Underwater Explosions Division of the Naval Ordinance Laboratory in White Oak, Maryland. I was assigned to the development of various computer codes to simulate the effects of underwater explosions (both nuclear and conventional). In 1969, I joined the National Aeronautics Space Administration at the Goddard Space Flight Center. I was assigned the task of developing and analyzing antenna boom deployment.

In 1974, I joined the Nuclear Regulatory Commission as a Nuclear Engineer in the Accident Analysis Branch, Division of Technical Assistance, Office of Nuclear Reactor Regulation. In January 1980, I joined the Incident Response Branch, Office of Inspection and Enforcement. I joined the Accident Evaluation Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation in June 1981. During my employment with NRC I have modeled and evaluated numerous

design basis accidents for nuclear power plants during construction permit and operating license reviews. I have been responsible for reviewing the radiation models used in equipment qualification reviews. I have been instrumental in the development of modeling techniques and the development or adaptation of computer codes used in the safety review of numerous construction permit and operating license applications.

HOWARD B. HOLZ

PROFESSIONAL QUALIFICATIONS

I am presently a Senior Reactor Engineer, Technical Review Branch, Clinch River Breeder Reactor Program Office in the Office of Nuclear Reactor Regulation. I am responsible for the mechanical and structural evaluations of safety related primary coolant system components and piping preceding and following a CDA event inside containment and the resulting challenges to containment from energetic and postulated meltdown events. I am also responsible for the review of a number of auxiliary systems in the CRBR Safety Evaluation.

I have a Bachelor of Engineering degree in Mechanical Engineering from the University of Southern California. I am a Professional Engineer registered in California. I have 27 years experience in Liquid Metal Reactors of which 16 years has been on advanced reactor systems.

I have held a number of positions bearing a range of responsibilities, including supervisor in the Fast Breeder Reactor Group at Rockwell International. I was Licensing Project Manager for FFTF, and worked on CRBR as a technical expert in the late 1970's and on LWRs following TMI-2. In this capacity I worked as a technical reviewer providing recommendations to a number of utilities (both PWRs and LWRs) on how to satisfy the TMI Task Action recommendations.

LEWIS G. HULMAN
PROFESSIONAL QUALIFICATIONS

I am presently Chief of the Accident Evaluation Branch, Division of Systems Integration, in the Office of Nuclear Reactor Regulation. I was formerly the Chief of Systems Interaction Branch and Chief of the Hydrology-Meteorology Branch, both in the Office of Nuclear Reactor Regulation.

My formal education consists of study in Engineering at the University of Iowa where I received a BS in 1958, and an MS in Engineering Mechanics and Hydraulics in 1967. In addition, I have taken post-graduate courses at the University of Nebraska, MIT, Colorado State University, and the University of California, and numerous management, technical and computer utilization courses sponsored by the government.

My employment with NRC (formerly AEC) dates from February 1971 with both the Office of Nuclear Reactor Regulation and the former Office of Reactor Standards, and for consultation on siting of materials utilization facilities. Assignments were made on both safety and environmental matters. My responsibilities in the licensing review of nuclear facilities were in the areas of site analysis, flood vulnerability, water supply, surface and groundwater acceptability of effluents, severe meteorologic events and diffusion analyses. In addition, I participated in the development of the technical bases for safety guides and standards, and research identification and analysis in these areas of interest.

From March 1980 through mid-April 1981 I was employed in private industry as a Vice President with Tetra Tech, Inc. in Pasadena, California. During this period I was responsible for business development, and for managing several contracts involving various engineering studies in water, including several contracts for government and industry. Of note were studies of a nuclear power plant in Yugoslavia for the International Atomic Energy Agency, flood protection in the Dominican Republic, a refinery intake design in Indonesia, and hurricane risk assessments in Texas, North Carolina, Florida, and New Jersey.

From 1968 to 1971, I was a Hydraulic Engineer with the Corps of Engineers' Hydrologic Engineering Center in Davis, California. I worked in special hydrologic engineering projects with most Corps' offices, participated as an instructor in training courses, and conducted research. Special projects work included water supply systems analysis for the Panama Canal, planning hydrologic engineering studies for water resource development near Fairbanks, Alaska, regional water supply and flood control studies for the northeastern U.S., hydropower and water supply studies for a dam in the northeast, and flood control studies in Mississippi.

From 1963 to 1968, I was a Supervisory Hydraulic Engineer with the Philadelphia District, Corps of Engineers. As Assistant Chief of the Hydraulics Branch, I was responsible for design aspects of multi-purpose

dams, navigation projects, coastal engineering development and special studies on modeling of dams, inlets, water supply, and shoaling, salt water intrusion, and the effects of dredging. I acted as advisor to the District Engineer, Philadelphia, on drought problems in the 1960's and represented him in technical meetings of the Delaware River Basin Commission - chaired interagency committee which evaluated the effects of the drought.

From 1958 to 1963, I was a Hydraulic Engineer with the Omaha District of the Corps of Engineers. I was responsible for the hydraulic design of flood control channels, hydraulic design of structures for large dams and several flood control projects. I also received training in hydrologic engineering, structural engineering, sedimentation, river training studies and design, and water resource project formulation.

I have published in journals of the American Society of Civil Engineers, the American Water Works Association, the Journal of Marine Geodesy, the National Society of Professional Engineers, the American Geophysical Union, and in internal technical papers and seminar proceedings of the Corps of Engineers, the AEC, and the NRC.

I am a registered Professional Engineer in the States of Nebraska and California. I am a member of the American Society of Civil Engineers, the American Meteorological Society, and the American Geophysical Union.

PROFESSIONAL QUALIFICATIONS
OF
JOHN K. LONG

My name is John K. Long. I am a reactor physicist in the Technical Review Branch, CRBR Program Office, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555.

I graduated from Columbia University with a bachelor's degree in Chemical Engineering, 1942. I was employed by the Hercules Powder Company, 1942-1945, in the manufacture of explosives. I worked at Wright-Patterson Air Force Base, Dayton, Ohio, 1945-1949, in the development of aircraft materials. I received a Ph.D. in Nuclear Physics from Ohio State University in 1953, and was employed by Argonne National Laboratory from 1955 to 1974. At Argonne I participated in and directed research on reactor critical facilities, analysis of reactor operating phenomena at EBR-2, and development of procedures for criticality control.

I have been employed by the Nuclear Regulatory Commission since 1974, and have reviewed many problems related to the licensing of sodium cooled fast breeder reactors, including problems related to the accidental generation of hydrogen and its release in containment. I was responsible for the review of aerosol and containment analyses and dose consequence analyses in the FFTF review. Since the TMI-2 accident, I have participated in accident analyses involving hydrogen problems for the TMI-1, Zion, Indian Point, Sequoyah and McGuire reactors.

With the reactivation of the CRBR licensing activities, I have been given responsibility for review of those aspects of core disruptive accidents involving thermal margins beyond the design basis.

A list of recent publications is attached.

List of Publications

A. M. Broomfield, A. L. Hess, P. I. Amundson, Q. L. Baird, E. F. Bennett, W. G. Davey, J. M. Gasidlo, W. P. Keeney, J. K. Long and R. L. McVean, ZPR-III Assemblies 48, 48A, and 48B: The Study of a Dilute Plutonium-fueled Assembly and its Variants, ANL-7759 (Dec. 1970).

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Safety Evaluation Report, Fast Flux Test Facility, NUREG-0358, August 1978, (J. K. Long responsible for accident consequence analysis)

Final Environmental statement. Clinch River Breeder Reactor Plant, NUREG-0139, February 1977 (J. K. Long responsible for chapters 8, 9, and parts of 11)

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J. K. Long, A. R. Baker, W. Gemmell, W. P. Keeney, R. L. McVean, F. W. Thalgott in "Physics of Fast and Intermediate Reactors, Vol. 1" p. 261-85. Vienna International Atomic Energy Agency (1962).

B. M. MORRIS

PROFESSIONAL QUALIFICATIONS

I am currently, Chief of the Electrical Engineering Branch, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission. During the construction permit review and safety evaluation report preparation, I was Section Leader, Technical Review Section, Clinch River Breeder Reactor Program Office, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission. In this capacity, I am responsible for direction of the technical review section's review of the fast sodium-cooled CRBRP safety review.

I received a Bachelor of Science, M.S., and Ph.D. degrees in physics from the University of Tennessee.

I spent five years teaching engineering and physics at Worcester Polytechnical Institute. I also spent five years doing research in engineering and nuclear physics at Savannah River and Oak Ridge National Laboratory. In 1977, I joined the Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation in the Reactor Safety Branch. I then worked in the Instrumentation and Control Systems Branch. I then became a Section Leader in the Reactor Systems Branch.

I have published several Journal papers in the fields of physics and nuclear engineering.

Jerry J. Swift

PROFESSIONAL QUALIFICATIONS

I am employed by the Clinch River Breeder Reactor Program Office, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. The title of my position is Reactor Engineer.

From 1980 to 1982, I worked at the U.S. Department of Energy on the environmental and safety aspects of a variety of nuclear technologies, including fusion devices, accelerators, transportation, waste management, and sewage irradiation.

From 1972 to 1980, I was employed in the Office of Radiation Programs of the U.S. Environmental Protection Agency, as a Nuclear Engineer and Environmental Protection Engineer. During this period, I was involved in evaluating the potential radiation doses and radioactive contamination of the environment that might result from severe reactor accidents, with application to the development of Protective Action Guides. I was also involved in the evaluation of normal operating releases from nuclear fuel cycle facilities in support of the development of 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operations." I assisted in development work for standards and guidance by EPA on management of high level wastes and uranium mill tailings. I aided EPA's participation in nuclear policy reviews by Presidents Ford and Carter. I managed EPA's NEPA reviews of environmental statements for two LWRs. I managed EPA's lengthy review of the Reactor Safety Study, WASH-1400.

From 1970 to 1972, I held a position as Physicist in the Institut fur Reaktorsicherheit,, (now the Gesellschaft fur Reaktorsicherheit mbH) in Cologne, Germany; in this position, I was primarily concerned with evaluating the nature and quantities of radoactive materials that might by released in postulated nuclear reactor accidents, and the resulting radiation doses that might be experienced. I was also involved inevaluating siting conditions.

While a graduate student at the Catholic University of America, from 1966 to 1970, I held the position of Assistant University Radiological Safety Official, performing health physics functions throughout much of the University.

I received the degree of Geological Engineer from the Colorado School of Mines in 1955, a Master of Science degree in Nuclear Engineering from Iowa State University in 1965, and a Ph.D. degree in Nuclear Engineering from the Catholic University of America, Washington, D.C. in 1971. I have current certification in Health Physics from the American Board of Health Physics.

In my present position, I am primarily involved in the review of those potential accident event sequences which may lead to radiological consequences. This includes both sequences within the design basis and sequences beyond the design basis. I also coordinate the efforts of other technical reviewers on these topics.

PROFESSIONAL QUALIFICATIONS

Charles R. Bell

I have been a staff member at Los Alamos since 1975. I received my Bachelor of Science degree in mechanical engineering from the University of Cincinnati in 1965 and a Ph.D. in nuclear engineering from the Massachusetts Institute of Technology in 1970. While at Atomics International from 1970 to 1975 I participated with the assessment of severe core disruptive accidents in breeder reactors. I also led an effort to design, develop, test, and apply a system analysis capability to investigate tube breaks in large sodium-water steam generators. At Los Alamos I participated in the development and application of advanced techniques for detailed assessment of severe breeder reactor accidents. I have had a major role in establishing new perspectives and in integrating these perspectives into national and international research and development programs. I am a member of the American Nuclear Society.

PUBLICATIONS

N. G. Galluzzo, C. R. Bell, and B. L. McFarland, "Design Considerations for Systems Subject to Sodium/Water Reactions," ASME Paper Number 75-PVP-68, Presented at the 2nd National Congress on Pressure Vessels and Piping Technology, San Francisco, June 1975.

C. R. Bell, "TRANSWRAP - A Code for Analyzing the System Effects of Large-Leak Sodium-Water Reactions in LMFBR Steam Generators," Presented at the American Nuclear Society Fast Reactor Safety Meeting, Los Angeles, California, April 2-4, 1974.

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R. C. Smith and C. R. Bell, "SIMMER-II Simulation of Thermite Freezing and Plugging Experiments," Trans. Am. Nuc. Soc., San Francisco, November 19-December 29, 1981.

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L. B. Luck, C. R. Bell, M. W. Asprey, and G. P. DeVault, "A Transition Phase Calculation of a Large, Heterogeneous Core LMFBR," Trans. Am. Nuc. Soc., San Francisco, November 29-December 4, 1981.

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C. R. Bell, "Breeder Reactor Safety--Modeling the Impossible," Los Alamos Science, Vol. 3, Number 1, (1981).

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L. B. Luck, C. R. Bell, and W. R. Bohl, "An Assessment of SIMMER-II Modeling Needs," Los Alamos National Laboratory report in preparation.

SOCIETIES: American Nuclear Society, Tau Beta Pi, Pi Tau Sigma

AWARDS: NSF Traineeship, AEC Fellowships

PROFESSIONAL QUALIFICATIONS

THOMAS A. BUTLER

I am a staff member in the Advanced Engineering Technology Group at the Los Alamos National Laboratory. In this position I am presently responsible for managing our group's structural mechanics effort for providing technical assistance to the Nuclear Regulatory Commission in the licensing review of the Clinch River Breeder Reactor Plant (CRBRP). Past responsibilities have included being Principal Investigator in a program to analytically predict the ultimate capacity of the Zion and Indian Point containment buildings, providing analytical support to Los Alamos's containment buckling research program, and analyzing the response of a number of structural systems subjected to thermal, seismic, impact, and blast loads.

I have a Bachelor of Science degree in Agricultural Engineering (machine design) from Colorado State University and a Master of Science degree in Aerospace Engineering (structural dynamics) from the University of Michigan. I am registered as a Professional Engineer in the state of New Mexico.

I have eleven years of professional experience in the field of mechanical engineering. I worked for Lockheed Missiles and Space Company for five years where I was responsible for structural dynamic response analysis and testing of large space vehicles and components. As a Senior Research Engineer, my responsibilities also included developing structural response computer codes, writing technical specifications, and acting as a technical monitor for subcontracts. I have been with the Los Alamos National Laboratory for the past six years where most of my work has been involved with nuclear safety.

Edmund T. Rumble, III

PROFESSIONAL QUALIFICATIONS

I am an employee and Corporate Vice President of Science Applications, Inc. (SAI), a nationwide research and consulting firm. In this capacity, I perform contract research on energy-related projects. Presently, I am a member of an SAI team providing technical assistance to the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission on safety matters related to the proposed Clinch River Breeder Reactor Plant.

I received a commission in the U.S. Navy and a Bachelor of Science degree from the U.S. Naval Academy. After graduation, I was qualified for and served as a U.S. Naval submarine officer responsible for operational and administrative aspects of a submarine nuclear power plant. Following my honorable discharge from the Navy, I received my Master of Science and Ph.D. degrees in Nuclear Engineering from UCLA. I am a Professional Engineer registed in the State of California and am listed in Who's Who in Technology Today, American Men and Women of Science, and Who's Who in California (14th ed).

I have been involved in LWR and LMFBR safety research at UCLA and SAI for the past ten years. My doctoral thesis involved modeling LMFBR core accidents. I have made technical contributions, managed, and acted as an advisor in deterministic and probabilistic safety analyses and assessments of LWRs and LMFBRs. Recently, I managed a major portion of, and technically participated in, a two-year, multi-organizational risk-oriented study of the SNR-3000, an LMFBR under construction in West Germany.

DAVID SWANSON
APPLIED SCIENCE ASSOCIATES
STATEMENT OF PROFESSIONAL QUALIFICATIONS

For the past eight years, I have assisted the Office of Nuclear Reactor Regulation in the review and evaluation of the materials interactions that can occur as a result of postulated core meltdown accidents. During this period, considerable expertise in reactor materials evaluation and analysis has been developed. Assistance has been provided on reviews for the Clinch River Breeder Reactor Plant (CRBRP), the Floating Nuclear Plant (FNP), the Fast Flux Test Facility (FFTF), the High Temperature Gas-Cooled Reactor (HTGR) and the Zion and Indian Point plants.

Technical assistance has been provided for safety issues concerned with the potential for a postulated core melt accident in the CRBRP. For this effort, the problems associated with molten core-limestone concrete and sodium-limestone concrete interactions were assessed and evaluated. Potential sacrificial materials for proposed core retention concepts were examined. Molten core debris and liquid sodium interactions with MgO were studied extensively.

In the safety review of the FFTF program, technical assistance was provided in the review and evaluation of materials issues. The interaction of liquid sodium with basalt and magnetite concrete and the commercial insulating and fire-resistant firebricks employed in the cell liners was studied. Technical assistance in this area also included an examination of the problems associated with a core disruptive accident, including the interactions between molten core debris and concrete and firebrick. The problems included the rate of penetration by sodium and molten core debris into concrete and firebrick. Other issues concerned cracking and spallation, the potential for adverse chemical reactions and the extent of gas evolution.

Technical assistance has been provided in the review and evaluation of information submitted by the FNP applicant (Offshore Power Systems) in the area of materials interactions associated with the delay of core melt penetration. This has included detailed technical assessments of molten core interactions with concrete and with sacrificial materials for use in the preparation of NRC safety and environmental review documents. In these studies, the effects of mechanical and thermal shock, brick floatation, eutectic formation, chemical interactions and slag attack have been investigated for the material (MgO) proposed for use in the core ladle.

In the area of High Temperature Gas-Cooled Reactors (HTGR), the loss of graphite strength due to oxidation induced by water vapor and the presence of contaminants has been examined. Review and assessment of certain programs at General Atomic has been provided; this has included examination of various core catcher concepts for Gas-Cooled Fast Reactors.

Other core retention proposals have been evaluated. These have included the use of borax and high-alumina cement.

I have conducted experiments to study the interactions of both concrete and MgO with molten core debris and liquid sodium. Over forty experiments with molten UO_2 and molten steel were performed, including experiments in which the UO_2 or steel was maintained in a molten state on top of either concrete or MgO. The interactions between the materials were thoroughly analyzed and have been the subject of a number of published reports.

During the course of seven years of work for the Office of Nuclear Reactor Regulation in this area, I have acquired expert knowledge in the area of nuclear reactor accident scenarios which may lead to core melting. I have also acquired a general knowledge of NRC licensing procedures, regulations, requirements and safety criteria. I hold a Ph.D. in nuclear chemistry and am a licensed professional nuclear engineer in California.

Note: - all my work has been sponsored by NRC for the last 8 years.

Unclassified Publications In The Area Of High Temperature Materials Studies Related To Advanced Reactor Systems

1. D. G. Swanson, J. N. Castle and P. D. Anderson, "Core Melt Materials Interactions Evaluations" Final Report, Applied Science Associates, Inc., Palos Verdes, CA, 1982 (to be published).

2. D. G. Swanson, I. Catton and V. K. Dhir, "A Thoria Rubble Bed For Post Accident Core Retention," Proceedings of the Fifth Post Accident Heat Removal Information Exchange Meeting (Karlsruhe, Germany), July 28-30, 1982, p.307.

3. D. G. Swanson, J. N. Castle and P. D. Anderson, "Core Melt Materials Interactions Evaluations," 2nd Annual Report, Applied Science Associates, Inc., Palos Verdes, CA, 1982.

4. D. G. Swanson, "Core Melt Materials Interactions Evaluations," 1st Annual Report, Applied Science Associates, Inc., Palos Verdes, CA, 1982.

5. D. G. Swanson, H. L. L. van Paassen and A. R. Marchese, "Molten Core Interactions with Commercial Firebricks," American Nuclear Society Transactions, 32, 525 (1979).

6. D. G. Swanson, E. H. Zehms, R. A. Meyer, J. D. McClelland, and H. L. L. van Paassen, "Evaluation of Materials for Retention of Sodium and Core Debris in Reactor Systems," USNRC, NUREG/CR-0900 (1979).

7. D. G. Swanson, E. H. Zehms, R. A. Meyer, J. D. McClelland, and H. L. L. van Paassen, "Annual Progress Report for Evaluation of Materials for Retention of Sodium and Core Debris in Reactor Systems," The Aerospace Corporation, ATR-79(7814)-1 (1978).

8. D. G. Swanson, E. H. Zehms, H. L. L van Paassen, and A. R. Marchese, "Interactions Between Molten Core Debris and Reactor Materials," American Nuclear Society Transactions, 28, 524 (1978).

9. D. G. Swanson, et al., Annual Progress Report, "Evaluation of Materials for CRBRP Core Retention," USNRC, NUREG/CR-0076 (1978).

10. D. M. Goddard, G. D. Kidwell, and D. G. Swanson, "Small-Scale Sodium Concrete Reaction Studies," The Aerospace Corporation, ATR-78(7706)-1 (1978).

11. D. G. Swanson, E. H. Zehms, D. M. Goddard, C.-Y. Ang, H. L. L. van Paassen, G. D. Kidwell, and A. R. Marchese, "Interactions Between Molten Core Debris and Core Containment Materials," American Nuclear Society Transactions, 27, 659 (1977).

12. D. G. Swanson, E. H. Zehms, C. -Y. Ang, H. L. L. van Paassen, and A. R. Marchese, "Molten Core Debris Interactions with Core Containment Materials," Proceedings of the 3rd Post-Accident Heat Removal Meeting, ANL-78-10, 408 (1977).

13. D. G. Swanson, D. M. Goddard, E. H. Zehms, J. D. McClelland, R. A. Meyer, H. L. L. van Paassen, and G. D. Kidwell, Annual Progress Report, "Evaluation of Materials for CRBRP Core Retention," The Aerospace Corporation, ATR-77(7608)-2 (1977).

14. D. G. Swanson, E. H. Zehms, C. -Y. Ang, J. D. McClelland, R. A. Meyer, and H. L. L. van Paassen, "Annual Progress Report for Ex-Vessel Core Catcher Materials Interactions," The Aerospace Corporation, ATR-77(7608)-1 (1976).

15. D. G. Swanson, D. M. Goddard, J. D. McClelland, and R. A. Meyer, "Sodium-Graphite and Sodium-Concrete Interactions Experiments," The Aerospace Corporation ATR-77(8162)-1 (1976).

16. D. G. Swanson and N. T. Porile, "Statistical-Model Calculation of the Angular Distribution of (α, n) Reaction Products," Phys. Rev. C, 1, 4 (1970).

17. D. G. Swanson and N. T. Porile, "Average Ranges, Cross-Sections and Isomer Ratios for the ^{134}Ba (α, n) ^{137m,g}Ce Reaction," Nucl. Phys. A144, 344 (1970).

18. D. G. Swanson and N. T. Porile, "Angular Distribution of ^{137m}Ce from the ^{134}Ba (α, n) Reaction," Nucl. Phys., A144, 355 (1970).

PROFESSIONAL QUALIFICATIONS

T. G. Theofanous

For over a decade now I have been heavily involved in the field of Nuclear Safety, both from the research as well as from the licensing point of view. During this period of time I have had the opportunity to make original technical contributions and to participate in the decision making process over a broad range of technical problems and licensing issues for both Light Water Reactors (PWRs, BWRs) as well as Advanced Reactor concepts (LMFBRs, HTGRs, FNPs). This rather unique mix of experience was obtained through the pursuit of two principal efforts. As a long term consultant to the ACRS I focused my efforts on LWR safety issues for accidents within the Design Basis Envelope. As a researcher at the University and in an associated activity as a consultant to the NRC (Regulatory) Staff my early work was concerned with LMFBR Core Disruptive Accidents, and more recently with LWR Class-9 Accidents. This experience was further complemented from a number of additional activities including participation in review groups, special review panels, working groups of experts, editorial activities in the scientific literature. Details are provided below:

I joined the ACRS effort during the early days of ECCS controversy and rule-making hearings. Since that time in addition to the ECCS and a number of plant-specific subcommittees I participated in the ATWS, Fluid Dynamics, TMI-II, and Pressurized Thermal Shock subcommittees. I originated the efforts (while reviewing Grand Gulf) that led to the discovery of certain "difficulties" with the MARK-I, II and III containment systems. As a member of the "Advanced Code Review Group" and the "Verification Review Group" I had significant impact in shaping the evolution of these NRC-RES efforts.

Over the period 1975-1979 I directed a University Technical Assistance effort for the NRC Staff (Division of Project Management). This effort covered the areas of Thermal Interactions, Recriticality, and Initiating Phase Phenomena (Voiding, Clad Relocation, etc.) of Core Disruptive Accidents in LMFBRs. Based on this work I provided technical assistance and worked closely with the NRC Staff during the reviews of FFTF and CRBR (Homogeneous Core Design). This work culminated with NUREG/CR-0224 which is now one of the key references in this area. Based on this background I am currently involved (as chairman of a management group) in the review and independent assessment of Core Disruptive Accident Energetics of the new CRBR licensing application (Heterogeneous Core).

Over the past three years (1980-1982) my group at Purdue has addressed problems in material interactions phenomenology arising in evaluating the consequences of LWR core melt accidents. This work was carried out under a technical contract for the NRC Staff (Division of Systems Integration). Based on this work I have provided technical assistance and worked closely with the NRC Staff during the reviews of the Zion/Indian Point and Limerick/GESSAR Probabilistic Risk Assessment Studies. In addition I provided the technical support to the NRC Staff in commenting on the NRC-RES research programs and plans in the area of Degraded Core Phenomenology.

Industrial & Consulting Experience

Consultant, U.S. Atomic Energy Commission (presently Nuclear Regulatory Commission), Advisory Committee on Reactor Safeguards (1971-present).

Consultant, Argonne, National Laboratory, Reactor Analysis and Safety Division (1971-1975).

Consultant, U.S. Atomic Energy Commission (presently Nuclear Regulatory Commission), Office of Nuclear Reactor Regulation (1974-present).

Consultant, Aerojet Nuclear Company, Idaho Falls, Idaho (1975).

Consultant, United Nuclear Industries, Richland, Washington, (1975-1976).

Vice-President, Fauske, Grolmes, Henry and Theofanous Ltd. (1979-1981).

President, Theofanous and Company, Inc. (1981-present).

Consultant, Los Alamos National Laboratory (1981-present).

Expert, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (1981-present).

Consultant, NUS (1982-present).

Special Review Panels & Assignments

Official U.S. NRC delegation to Europe (1976).
Purpose: To visit France, Germany, and the United Kingdom to learn about research safety programs and operating experience with demonstration LMFBR plants and to exchange views on licensing requirements and procedures.

Advanced Code Selection Committee, U.S. NRC Office of Nuclear Regulatory Research (1979).
Purpose: Select one of the three advanced codes being developed for the analysis of LWR Loss of Coolant Accidents.

Ad Hoc Committee for the Review of NRC's LMFBR Safety Research, U.S. NRC Office of Nuclear Regulatory Research (1981).
Purpose: Review and comment on existing programs and make recommendations for future directions.

Ad Hoc Technical Review Meeting on Steam Explosions, U.S. NRC Office of Nuclear Regulatory Research (1982).
Purpose: Review and comment on the significance (applicability) of the new steam explosion information on reactor safety.

Chairman of the Clinch River Breeder Reactor (CRBR) Core Disruptive Accident (CDA) Energetics Review Management Group, U.S. Nuclear Regulatory Commission (1982).
Purpose: Manage NRC's licensing review team (consisting of R&D groups in National Laboratories and independent consultants) to develop an independent technical position on CRBR CDA energetics.

Other Professional Activities

Member of the CSNI (OECD) group of Experts on the Science of Fuel-Coolant Interactions and of Vapor Explosions.

Member of the CSNI (OECD) Liquid Metal Boiling Working Group.

Lectured extensively in this country as well as in France, Germany, England, and Italy on topics of two-phase flow and on reactor safety.

Editor, ~~Member, Editorial Board of~~ *Nuclear Engineering and Design.*

Membership in Professional Societies

Member, Sigma Xi
Member, American Nuclear Society
Member, American Institute of Chemical Engineers

Citation in Biographical Reference Books

Who's Who in Atoms
American Men and Women of Science
Who/s Who in the Midwest

BOOK CHAPTERS

T. G. Theofanous, "Nuclear Reactor Safety: The Role of Accident Analysis," Nuclear Energy Issues and Topics, Editor K. O. Ott, (in print).

ARCHIVAL JOURNAL PAPERS

T. G. Theofanous, L. Biasi, H. S. Isbin, and H. K. Fauske, "A Theoretical Study on Bubble Growth in Constant and Time-Dependent Pressure Fields," Chemical Engineering Science, Vol. 24, No. 5, 1969, 885-897.

T. G. Theofanous, L. Biasi, H. S. Isbin, and H. K. Fauske, "Nonequilibrium Bubble Collapse--A Theoretical Study," Chemical Engineering Progress Symposium Series, Vol. 66, No. 102, 1970, 37-47.

T. G. Theofanous, H. K. Fauske, and H. S. Isbin, "On Some Aspects of Steam Bubble Collapse," Discussion J. Heat Transfer, Feb. 1970, 211-212.

T. G. Theofanous, H. S. Isbin, and H. K. Fauske, "An Integral Method for Convective Diffusion--Bubble Dissolution," AIChE Journal, Vol. 16, No. 4, 1970, 688-690.

T. G. Theofanous and H. C. Lim, "An Approximate Analytical Solution for Non-Planar Moving Boundary Problems," Chemical Engineering Science, Vol. 26, No. 8, 1971, 1297-1300.

T. G. Theofanous, "A Course in Transport Phenomena," Chemical Engineering Education, Vol. 5, No. 4, 1971, 174-177.

Henry C. Lim and Theofanis G. Theofanous, "On Unsteady State Moving Boundary Problems for Non-Plane Geometries," Letter to the Editor of J. Chemical Eng. Japan, 4, No. 1, 1971, 100-101.

V. F. Smolen, R. G. Barile, and T. G. Theofanous, "Relationship Between Dose Effect, Time and Biophasic Drug Levels," J. Pharmaceutical Sciences, Vol. 61, No. 3, 1072, 467-470.

T. G. Theofanous and V. F. Smolen, "Multiphase Dose Kinetics of Pharmacological Effects of Indirect Anticoagulants," J. Pharmaceutical Sciences, Vol. 61, No. 6, 1972, 980-982.

T. G. Theofanous and R. G. Barile, "Multiple Dose Kinetics of Oral Anticoagulants: Methods of Analysis and Optimized Dosing," J. Pharmacuetical Sciences, Vol. 62. No. 2, 1973, 261-266.

T. G. Theofanous and H. K. Fauske, "The Effect of Noncondensibles on the Rate of Sodium Vapor Condensation from a Single-Rising HCDA Bubble," Nuclear Technology, Vol. 19, No. 3, 1973, 132-139.

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ATTACHMENT 1 TO CDA TESTIMONY

POLICY ISSUE
(Information)

June 6, 1983

SECY-83-219

For: The Commissioners

From: William J. Dircks, Executive Director for Operations

Subject: STATUS REPORT ON LWR ACCIDENT SOURCE TERM REASSESSMENT

Purpose: To inform the Commission of the status of the staff reassessment of LWR accident source terms.

Category: This paper is for information purposes only.

Background: During the past 2 years, the staff has devoted significant attention to the question of the magnitude, timing, and type of radionuclide releases to the environs (source terms) under LWR accident conditions. The staff prepared a preliminary assessment of the technological basis for accident source terms, NUREG-0772, published in June 1981. Following the publication of this report, the staff refocused the research program to further develop the technical data base summarized in NUREG-0772 and to apply the data base and its computer codes to several typical LWR plants. The research program is scheduled to produce the bulk of its results at the end of 1985. However, a substantial amount of new information, including revised source term estimates on typical LWR plants, is becoming available in 1983.

To plan for the timely use of the revised source term research in policy development and reactor licensing, the staff prepared a program plan to implement the results of this research (memorandum from Dircks to Denton, Minogue, and DeYoung, dated December 17, 1982). I formed an Accident Source Term Program Office (ASTPO), by memorandum of January 20, 1983, to ensure the focused attention, on a full-time basis, of management and senior technical staff on the implementation of the source term program plan. The ASTPO staff is drawn from RES, NRR, and IE offices that have interlocking activities and responsibilities in the accident source term area. Responsibilities for implementation of regulatory actions remain in the line organizations.

CONTACT:
R. Bernero

Since the establishment of ASTPO, we have reevaluated the source term agenda and restructured the schedule presented in my December 17, 1982 memorandum. By this paper, I wish to provide you with a brief description of our revised approach for the source term work. In addition, this paper provides you with a revised schedule, current status, and set of milestones for the Source Term and Severe Accident Research Program consistent with our program review discussions on April 27, 1983.

A. Approach for Source Term Activities

The source term work falls into two general categories: (1) the technical basis for reassessment and (2) the regulatory use of revised source terms.

Technical Basis for Reassessment

The process of accurately determining fission product releases under accident conditions is a difficult one. Because of the amounts of radioactivity as well as the time and costs involved, it is not feasible to conduct full-scale experiments to determine fission product behavior during reactor core degradation, melting and containment failure. For this reason, fission product behavior during accidents must be predicted by complex computer codes which model the various phenomena which can occur to snythesize the appropriate sequence of events involved. The phenomenological models built into the computer codes rest, in turn, upon the results of a great number of small-scale experiments together with much theory that, taken together, represent our present state of knowledge of fission product behavior. To predict fission product behavior, one must ensure not only that the phenomena are adequately modeled, but that the codes are correctly employed to examine those sequences which are important in risk considerations and which represent the spectrum of current reactor designs.

We believe the above process, generally referred to as validation, to be essential and that it is best achieved by specific systematic assessment and tested by the scientific peer review process. Therefore, we have restructured the technical data part of the source term work to give greater emphasis and care to scientific validation and peer review. In general, we have divided the technical source term research work into four basic elements, which are listed in Table 1.

TABLE 1

BASIC ELEMENTS OF THE REASSESSMENT OF

THE TECHNICAL BASES FOR SOURCE TERMS

ELEMENT 1: Summary of the Data Base for Validation of Codes to Predict Releases

ELEMENT 2: Source Term Estimates for Selected Plants and Accident Sequences

ELEMENT 3: Thorough Peer Review of the Scientific Basis for Reassessment

ELEMENT 4: Appraisal of the Risk and Regulatory Significance of Reassessed Source Terms

Element 1. A summary of the technical data base and validation status of the computer codes used by Battelle Columbus (BCL) to predict plant releases will be documented by NRC contractors, principally by the Oak Ridge National Laboratory (ORNL).

Element 2. Source term estimates will be developed and documented by BCL for five LWR plants (Surry, Peach Bottom, Grand Gulf, Sequoyah, and Zion) and selected severe accident sequences.

Element 3. Each of the BCL reports summarizing the analyses and results of the plants in Element 2 will be reviewed by technical experts from the nuclear industry, universities, consultants, public interest groups, and other countries. In addition, an independent scientific organization will conduct a thorough, broad-based peer review of the scientific basis of the NRC contractor efforts listed above.

Element 4. The risk and regulatory significance of the reassessed accident source terms will be appraised by the NRC staff. This work will include staff assessments, with some technical assistance from contractors, of the state of the art of containment performance during severe accident sequences, including loads and structural response. Additional support for this element will include preparing and documenting a limited uncertainty and sensitivity analysis of the model methodology by an NRC contractor. The results of the Industry Degraded Core Group (IDCOR) and the American Nuclear Society (ANS) source term efforts will also be reviewed and appraised by NRC contractors and staff.

These four basic elements constitute the basis for a major report that will be prepared by the NRC staff: NUREG-0956, "A Reassessment of the Technical Bases for Estimating Fission Product Behavior During LWR Accidents." This report will be the sequel to NUREG-0772, "An Assessment of the Technical Bases for Estimating Fission Product Behavior During LWR Accidents," which was published in 1981. As we did with NUREG-0772, we expect to initially publish NUREG-0956 in draft form to ensure wide public review and comment since this technical document could be the basis for significant regulatory actions.

Regulatory Use of Revised Source Terms

Currently, two reactor accident source terms are used in the regulatory process. The first, the single source term stated in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," was published in March 1962. It has since been used extensively in the licensing review for calculating the consequences of design basis accidents as well as for some design standards, such as for containment building filters and equipment qualification.

The second source term widely used is the set of severe accident source terms from the Reactor Safety Study (WASH-1400), which was published in 1975. The WASH-1400 source term, as it is often called, is the basis for the analysis of the risk from the most severe accidents, which the staff provides in each reactor environmental impact statement. It is also the basis for emergency response planning since it is at the heart of the reactor risk estimates in the joint NRC/Environmental Protection Agency (EPA) document, NUREG-0396 (EPA 520/1-78-016), "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," published in December 1978. The planning basis in NUREG-0396 served as a principal foundation for the joint NRC/Federal Emergency Management Agency (FEMA) document, NUREG-0654 (FEMA-REP-1) Rev. 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," published in November 1980.

It is apparent that a significant revision in the accident source terms can have a significant impact on the regulatory process. Sensitivity analyses were published last year in NUREG/CR-2239, "Technical Guidance for Siting Criteria Development." These studies showed that a relative modest reduction in source terms could have a significant effect on the calculated radiological effects offsite from severe accidents. Source term reductions have their most dramatic impact on threshold effects like early fatalities and injuries. Because of the sensitivity of offsite accident consequences to changes in source terms, there is a need to reevaluate regulatory positions in matters such as emergency preparedness and siting.

The approach we are following in regulatory use of revised source terms is a cautious one. We are concentrating first on ensuring that we have a solid technical base with a wide review and acceptance in the technical community for any source term revisions that might lead to revised regulatory positions.

If a significant revision of the source terms appears to be justified, we intend to use the information first in emergency preparedness and later in siting and design considerations. We feel that this priority matches not only current regulatory priorities but also the degree of precision in source term estimation that is needed to deal with these issues since the bases for emergency preparedness considerations can be derived using more approximate source term estimates than those needed for considerations of specific alternative plant design features.

As we consider potential changes in emergency preparedness based on source term revisions, we should also consider possible changes based on other factors. For example, the joint NRC/EPA document (NUREG-0396) is the basis of our current Emergency Planning Zones. That document relied on the controversial accident probabilities, source terms, and health effects model from the Reactor Safety Study. In addition, the document relied on the proposed EPA Protective Action Guides (PAG). Thus, in addition to accident source terms, the accident probabilities, health effects models, and PAG levels may need to be reconsidered as well.

There is also a large body of operational experience that should be considered in evaluating possible changes in emergency preparedness planning and criteria. We have had the experience of implementing emergency plans at many sites since the planning document (NUREG-0396) was published in 1978 and the criteria document (NUREG-0654) was published in 1980. From that experience alone, we and the other concerned agencies can identify changes or refinements that are appropriate. Therefore, our approach in this source term program is to follow two parallel courses of action. While the work proceeds on the source term technical basis, we are working with EPA and FEMA to evaluate other matters and prepare for prompt consideration of source term revisions when they are available and recognized as reliable. With this parallel approach, we expect to do the technical work with continuing peer review up to a point where we can simultaneously prepare to issue (for public comment) the NRC reappraisal of source terms, the redrafts of NUREG-0396 and NUREG-0654, and proposed changes in NRC policy and rules on emergency preparedness.

3. Schedule and Status

The planned activities and schedule for the source term reassessment, including the related milestones of the Severe Accident Research Program, are depicted in Table 2 and the associated milestone chart (Figure 1). A discussion of the interrelationship of these activities and their current status follows.

Source Term Reassessment

The data base available for validating the codes used by BCL to predict plant source terms (Milestone 1) will be summarized in an ORNL report scheduled for publication in August 1983. By September 1983, the BCL draft reports on source term estimates for five LWR plants (Milestone 2) will be completed, including peer review by technical specialists. Draft reports have been prepared for the Surry plant (PWR) as well as for the Peach Bottom and Grand Gulf plants (both BWRs). The specialist peer review of the Surry plant was conducted in January 1983, and the review of the BWR plants began on May 24-25, 1983 and will continue at a meeting in July. The specialist peer review of the revised PWR reports and the data base report will be completed by September. It is possible that additional review meetings by this group may be held later in the year.

The specialist peer review meetings held so far, January 25-26 and May 24-25, have been successful. Both meetings were held in the main ACRS meeting room (H-1046) with a full transcript kept. Table 3 lists the invited reviewers and observers. Attendance and participation have been good and many of the peer comments have had a substantial impact on the work.

NRC contractor reports will provide the major informational basis for the broad-based peer review (Milestone 3) by an independent scientific organization beginning in August 1983 with completion expected in May 1984. Initial discussions with the American Physical Society have been completed and we expect to receive a study proposal from the Society in early June.

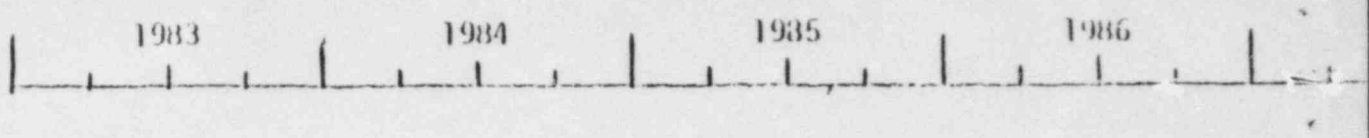
Starting in the summer 1983, the staff will initiate its appraisal (Milestone 4) of the risk and regulatory significance of reassessed source terms based on NRC contractor analyses and findings, results of the individual technical peer reviews, and review of the IDCOR work. The draft staff appraisal (NUREG-0956) will be available for public comment after receipt of the APS findings with the final document scheduled for publication in December 1984. We envision a reevaluation of the NUREG-0956 findings for confirmation and modification in May 1986 (draft) and October 1986 (final) to take advantage of the substantially augmented data base available then.

TABLE 2

MILESTONE SCHEDULE FOR SOURCE TERM REASSESSMENT
AND RELATED SEVERE ACCIDENT ACTIVITIES

1.	Summary of the Data Base for Validation of Codes to Predict Releases	August 1983
2.	Source Term Estimates for Selected Plants and Accident Sequences	September 1983
3.	Thorough Peer Review of the Scientific Basis for Reassessment	May 1984
4.	Appraisal of the Risk and Regulatory Significance of Reassessed Source Terms--Reevaluate	June 1984 (Draft) December 1984 (Final) May 1986 (Draft) October 1986 (Final)
5.	Joint Reassessment with EPA of the Planning Basis for Emergency Response	June 1984 (Draft) January 1985 (Final)
6.	Joint Reassessment with FEMA of the Criteria for Emergency Preparedness	June 1984 (Draft) January 1985 (Final)
7.	Reevaluation and Development of Proposed Changes in NRC Policy and Rules on Emergency Preparedness	June 1984 (Draft) January 1985 (Final)
8.	Staff Recommendations on the Use of KI	August 1983
9.	Develop Current Best Estimates for Accident Sequence Probabilities for Different Plant Types	October 1983 September 1985
10.	Modify Risk Codes for Current Best Estimates of Accident Consequences	February 1984 May 1986
11 and 17.	Complete Severe Accident Cost Benefit and Generic Risk Assessments--Reevaluate	June 1984 November 1986
12 and 13.	Complete Severe Fuel Damage (SFD) Experiments (Phase I) in PBF for Temperatures Less than 2400°F--Complete SFD Experiments in PBF for Temperatures Greater than 2400°K	August 1984 December 1985
14.	Complete Containment Failure Mode Studies Various Containment Types	November 1986
15.	Complete Phase I Severe Accident Research Program--Complete Phase II	August 1984 November 1986
16 and 18.	Provide Recommendations for Criteria to the Commission for Existing Plants to Prevent or Mitigate Severe Accidents--Reevaluate	September 1984 December 1986

FIGURE 1. MILESTONES FOR SOURCE TERM REASSESSMENT AND RELATED SEVERE ACCIDENT ACTIVITIES



(1) Summary of the Data Base for Validation of Codes to Predict Releases

(2) Source Term Estimates for Selected Plants and Accident Sequences

(3) Thorough Peer Review of the Scientific Basis for Reassessment

(4) Appraisal of the Risk and Regulatory Significance of Reassessed Source Terms

(5) Joint Reassessment with EPA of the Planning Basis for Emergency Response

(6) Joint Reassessment with FEMA of the Criteria for Emergency Preparedness

(7) Reevaluation and Development of Proposed Changes in NRC Policy and Rules on Emergency Preparedness

(8) Staff Recommendations on the Use of EI

(FIGURE 1, CONTINUED)

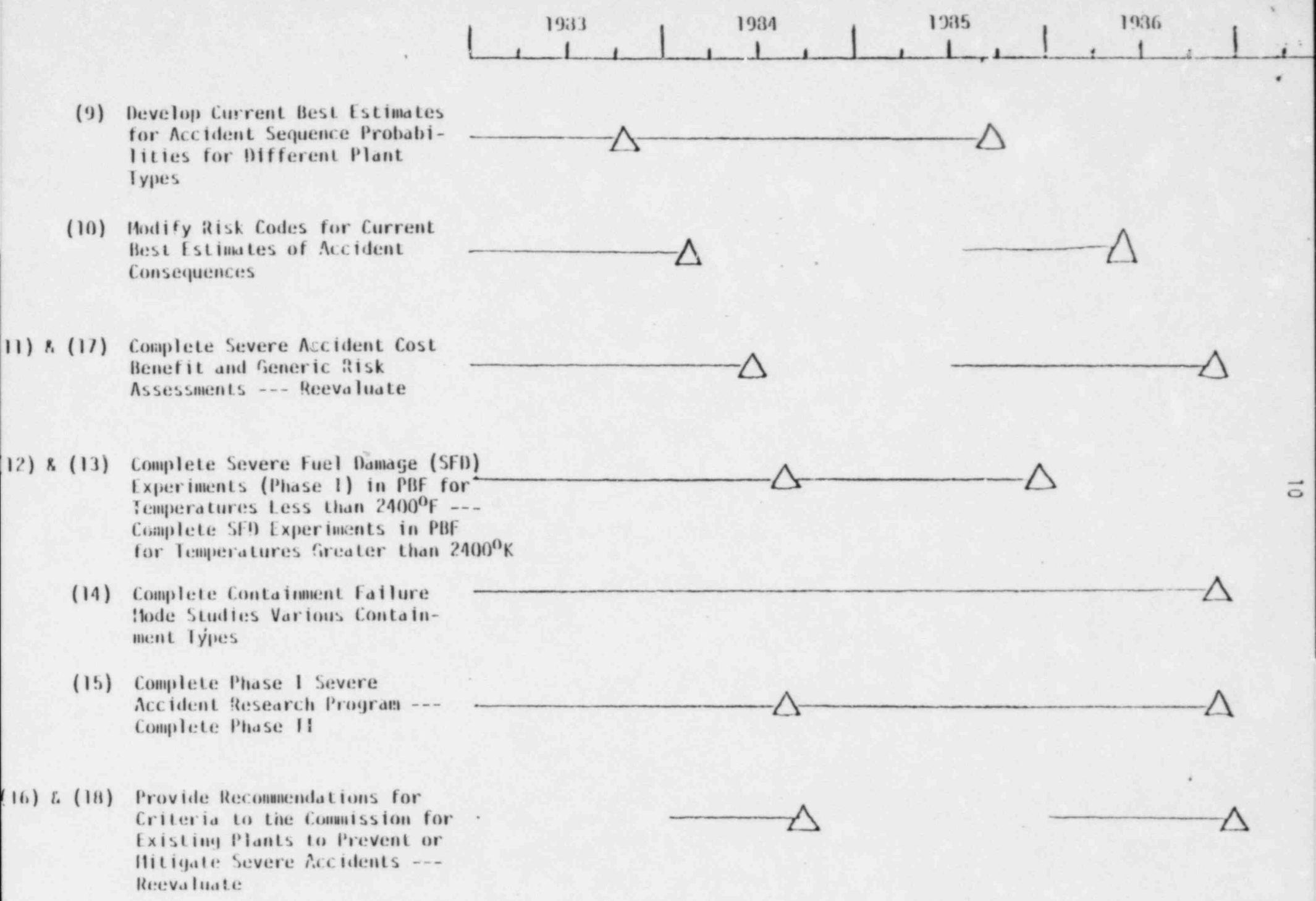


TABLE 3

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Regulatory Applications

Initial staff efforts relating to the introduction of new source term information into the regulatory process are focused on emergency planning requirements and implementation guidance.

A joint NRC/EPA reevaluation of the planning basis document (NUREG-0396) will be available in draft form for public comment in June 1984 with a final version scheduled for January 1985 (Milestone 5). The initial thrust of this reevaluation is based on a reconsideration of accident probabilities, health effects models, and PAG levels with new source term information being incorporated as it becomes available.

Several meetings have been held with EPA to arrange for conduct of the needed work by both agencies. The EPA is not prepared to staff a task force effort such as was used originally in preparing NUREG-0396. We are now working to develop a structure of specific NRC and EPA contributions which would enable NRC to prepare the revision to the document.

FEMA and NRC have established a working group which is currently reevaluating the existing guidance and criteria contained in NUREG-0654. This reevaluation will rely initially on a needs assessment developed from information collected from FEMA and NRC regional personnel and from state and local governments. Experience in using the NUREG-0654 criteria over the last 3 years will be factored into the needs assessment along with the results of the source term reassessment. The reevaluation of NUREG-0654 may indicate that a revision is warranted; and, if so, a draft of the revised document will be published for comment by June 1984 with a final document issued in January 1985 (Milestone 6). Based on the results of the above reevaluation, the staff will perform a parallel review of NRC policies and rules on emergency planning (Milestone 7). The purpose of this review is to present proposed rule changes on a schedule consistent with the availability of the draft source term reappraisal and the proposed revisions to NUREG-0396 and NUREG-0654.

Staff recommendations on the use of potassium iodide (KI) as a preplanned emergency protective measure for public use have been developed based on the earlier source term assessment (NUREG-0772, June 1981), the recent draft Battelle Columbus Laboratories report containing source term estimates for the Surry (PWR) plant, and a cost/benefit uncertainty analysis conducted by Sandia National Laboratories and the NRC staff (NUREG/CR-1433, March 1980). The staff's recommendations and analysis have received ACRS review, and we have just received