NUREG-0968
Vol. 2
Appendices

Safety Evaluation Report
related to the construction of the Clinch River Breeder Reactor Plant

Docket No. 50-537

U.S. Department of Energy
Tennessee Valley Authority
Project Management Corporation

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

March 1983

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APPENDIX A

EVALUATION OF CORE DISRUPTIVE ACCIDENTS

A.1 INTRODUCTION

The purpose of this appendix is to describe the staff's evaluation of hypothetical core disruptive accidents* which, for analytical purposes, have been postulated to occur in the CRBR. This introduction is divided into three major parts. The first (Section A.1.1) provides background information. The second (A.1.2) provides an overview of potential CDA initiating events and consequences considered for the CRBR. The third (A.1.3) describes the guidelines used in evaluating CDAs for the CRBR. Attachment A.1-I provides a schematic view of major components of the reactor systems. They will be referenced during evaluation discussions especially in Section A.3. The principal sections of this appendix deal with the staff's evaluation of the major areas associated with the assessment of CDAs.

A.1.1 Background

In other sections of this SER we have discussed our evaluation of the measures taken to assure that the risk from a range of postulated design basis events is acceptably low. Events considered in those evaluations included external events such as earthquakes, tornadoes and floods, and internal events such as fires, control system failures, coolant leaks, fuel failures, and electrical system failures. The conclusions reached, on the basis of our evaluation are; (1) that, for all credible events, the design features proposed for the CRBRP will be adequate to assure that fundamental safety functions such as reactor shutdown, decay heat removal, and containment of radioisotopes will be achieved with high reliability.

Although there is a high level of confidence that core disruptive accidents initiated by failure to achieve reactor shutdown or to remove decay heat are very unlikely and may be excluded from the traditional design requirements, these events cannot be entirely excluded as contributors to risk because of their potentially severe consequences.

In LWR safety reviews the failure to achieve a reactor shutdown has been considered in terms of the anticipated transients without scram (ATWS). In the CRBR safety review the term "unprotected transients" is used in the same manner, except that in the case of a CDA initiator, the failure of reactor shutdown is assumed to occur. This assumption is made despite the fact that an ATWS in the CRBR would require a failure of the two fast acting, independent and redundant shutdown systems. To assure the availability of the two shutdown systems, each has been given a reliability

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* For discussion purposes in this appendix we define a core disruptive accident to be a core melt accident in which sufficient fuel and clad have melted to allow relocation of enough material to substantially affect the neutronic (and hence power) behavior of the reactor. In the LMFBRs such core melt accidents have come to be known as core disruptive accidents (CDAs).

objective with an accompanying reliability program to support that objective. The staff review of this reliability program is reported in Appendix C. Therefore, the staff believes that an ATWS like incident in the CRBR is substantially less likely than for an LWR and undoubtedly belongs in the CDA category.

The applicants have considered the risk from hypothetical CDAs, and have incorporated certain design features to mitigate such events. In this appendix the staff has evaluated the residual risk from hypothesized core disruptive accidents at the CRBRP. This review and evaluation has focused on the capabilities of these mitigation features. They include; (1) the capability of the primary coolant system to withstand the mechanical stresses from a range of energetics from core disassembly (designated as structural margin beyond the design basis, SMBDB), (2) the capability of the containment and related systems and structures to withstand the long term thermal stresses from sodium-concrete reactions, sodium and hydrogen burning and effects of core debris which result from vessel melt-through (designated as thermal margin beyond the design bases, TMBDB) and (3) the radiological consequences of CDAs and specifically of intentional venting of the containment atmosphere after a CDA.

The conclusions of our evaluations in these areas are presented in the main sections of this appendix, organized as follows: Section A.2 includes our independent evaluation of the energy generated during CDAs; Section A.3 includes our evaluation of the structural capability of the primary system to contain the expected energy release without allowing sodium spray fires, large releases of vaporized fuel, or early containment failure from missiles (SMBDB); Section A.4 includes our evaluation of the capability of containment structures and mitigating systems to prevent containment failure and to reduce the release of radioisotopes to low levels (TMBDB); Section A.5 includes our evaluation of the radiological consequences of the range of CDA phenomena under the unlikely hypothesis that a CDA has occurred; Section A.6 includes the summary conclusions we have reached regarding the risk from CDAs.

A.1.2 Potential CDA Initiating Events and Consequences

Potential core disruption initiators fall in one of two general categories: an undercooling condition which results from a decrease of core cooling without an appropriate reduction in power; or an overpower condition which can result from an unprotected insertion of reactivity.* Such

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* Subassembly and fuel pin failure propagation cannot be placed conviently in either category. This potential initiating event is discussed further in Section A.2.

events are categorized as unprotected loss-of-flow (LOF) accidents or unprotected transient-over-power (TOP) accidents respectively. A third type of potential CDA initiator which has received considerable analysis is the protected loss-of-heat sink (LOHS). Although this is an undercooling transient it occurs over a much longer time scale.

The applicants have evaluated a large number of variations of the LOF and TOP events. Details of the applicants' CDA analyses are reported in References 1, 2 and 3. These analyses were revised following extensive meetings with the staff and its consultants (References 4, 5 and 6).

Based on these studies the applicants have concluded that most CDAs are non-energetic. Further, they have computed relatively small values of energy release for those events classified as energetic. The applicants thus conclude that there is little or no rapid mechanical loading imposed on the primary system and therefore, do not expect a failure of the primary system from this mode.

Although they believe that CDA energetics will be low, the applicants claim that the primary system can withstand energetic CDAs in which materials impact the vessel head with substantial kinetic energy. This evaluation was supplied to envelope cases even beyond their best estimate CDAs. They have supplied an analysis together with experimental data to support their conclusion.

Section A.2 contains a summary of an extensive independent assessment of the level of energetics associated with CDAs performed for the staff as a special task. Support for this task was provided by the Los Alamos National Laboratory (LANL) and was directed by Dr. T. G. Theofanous (Professor of Nuclear Engineering, Purdue University) and Dr. C. R. Bell (Associate Group Leader, Safety Analysis Group at LANL). The detailed study is reported in NUREG/CR-3224 (see Reference 7). As noted earlier the staff's evaluation of the capability of the primary system to accommodate energetic CDAs is described in Section A.3.

The staff has also evaluated the applicants analysis of non-energetic 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. 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, by the applicants, 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 being diluted by first venting, then purging with outside air.

Over the long term the debris generated from the CDA is expected to be retained within the cleanup system, accumulating 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 made, however, for the situation in which complete penetration is assumed to occur.

The staff's evaluation of the CDA consequences of the post vessel melt-through sequence described above is provided in Section A.4. Support in this area was provided by Mr. T. Butler (Task Leader), et. al. at LANL and by Dr. R. Gasser (Task Leader), et. al. of BNL and D. Swanson, et. al, of Applied Science Associates Incorporated. The radiological consequences of CDAs in the CRBR is presented in Section A.5. Support for major aspects of this review was provided by Dr. R. Gasser, et. al., of BNL. Our conclusions are summarized in Section A.6.

A.1.3 General Criteria for Evaluation of Core Disruptive Accidents

A.1.3.1 Staff's Criteria

The criteria used by the staff to evaluate core disruptive accidents have evolved over a period of time. On May 6, 1976, a letter was sent from R. Denise (NRC) to L. Caffey (CRBR Project) which stated the preliminary position regarding postulated core disruptive accidents (CDAs) for CRBR (Reference 8). That position included a specific criterion for containment which required that containment integrity shall be maintained for at least 24 hours before venting could be allowed. The 24-hour period was chosen to achieve approximate comparability for the progression of CRBR core melt accidents with those for LWRs. Previous analyses of LWR core melt accidents, described in WASH-1400, indicated that for representative accident scenarios, containment failure could be expected at about 24 hours although some scenarios involved failure at earlier and later times. For LWRs the release at 24 hours is assumed to be uncontrolled and unfiltered. Based on the applicants' TMBDB concept for the CRBR, the release would be controlled, scrubbed and filtered.

The staff believes that the concept of venting the CRBR containment through scrubbers and filters after a postulated CDA, as a means of preventing containment failure caused by overpressurization, is acceptable. However, we have reconsidered the specific criterion that venting may not occur prior to 24 hours after accident initiation. Instead the staff is basing its judgement of the capability of CRBR to accommodate CDAs on an integrated assessment which takes into consideration the following guidelines:

1. It must be feasible to provide adequate and reliable information for making a decision on whether to vent and when to vent containment.

2. It must be possible to implement adequate protective action plans, such as for evacuation or sheltering, during the period between (1) the indication that venting is likely to be required and (2) the time at which venting commences.

3. The combined doses as a consequence of venting and leakage should meet the CRBR version of the 10 CFR Part 100 guidelines.* and (if necessary because of poor hydrology) it must be feasible to interdict liquid pathways which could result in health effects comparable to those from atmosphere releases.

4. There must be a reasonably high level of assurance based on realistic evaluation and consideration of uncertainties, that doses significantly above 10 CFR Part 100 guidelines will not occur due to; (1) energetic core disassembly, (2) generation of heat or pressure within containment by core debris or chemical reactions or (3) failure or inadequacy of the containment venting and cooling systems.

Accident mitigation features which might be necessary to meet these generalized guidelines for accidents beyond the design basis will be evaluated on a best estimate basis. Considering the low probability of core-melt accidents we believe that use of best estimate models and values of system parameters is acceptable for use in evaluation of the performance of accommodation features involved in the BDBA. This approach is comparable to that proposed in the evaluation of core melt accidents for LWRs (Reference 10).

A number of factors have been considered in reviewing the CRBR against these guidelines. These factors include:

1 - <u>Information for Venting Decision</u>

(a) Type, location, and number of instruments.

(b) Environmental survivability of instruments.

2 - <u>Implementing Protective Measures</u>

(a) Feasibility of Protective Measures.

(b) Rate of change of conditions in containment.

3 <u>Doses</u>

(a) Containment Leakage.

(b) Filter efficiency (in degraded environment).

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* The CRBR version of 10 CFR 100 includes special considerations involving the potential effect of plutonium on critical organs such as the lung and bone. For further details see pp. III-8 and III-9 of NUREG-0786, "Site Suitability Report in the Matter of the Clinch River Breeder Reactor," June 1982, (Reference 9).

The 10 CFR Part 100 guidelines were not developed or intended for accidents beyond the design basis. They are applied here with the stipulation that these are guidelines for use with realistic assessments of CDAs rather than as limits.

(c) Transport and fallout of radioactive species in aerosol environment.

(d) Liquid Pathway Analysis.

4 - Potential for Containment Failure

(a) Sodium-Concrete reactions.

(b) Effects of core debris (including attack on base mat and impact on Sodium-Concrete reaction rates).

(c) Chemical reactions in containment including sodium burning and hydrogen burning.

(d) Reliability and capacity of annulus cooling.

(e) Capability of filters/scrubbers.

(f) Analytical methods for temperature and pressure determination.

(g) Margins to containment failure modes.

(h) Energetics beyond primary system capability.

Our evaluation of these issues is included in the following sections. The staff believes that by meeting the above criteria for CDA mitigation, the consequences from such severe accidents at CRBR will be sufficiently small that, coupled with the low probability of occurrence of such accidents, the overall risk from CRBR will be very low.

A.1.3.2 Applicants' Criteria

The applicants have followed very closely the approach recommended by the staff (Reference 10). That is, they have specified those features and functional requirements needed to accommodate events that are beyond the design basis. In this regard they have defined the accident generated loads and described the methods used to evaluate the response of the plant system when subjected to these loads.*

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* See, for example, Section 5.2 (Component Margin Requirements) 5.3 (Structural Margin Beyond the Design Basis), 5.4 (Accommodation of Component Margin Requirements) all in CRBRP-3, Volume 1 (Reference 1) and Sections 2.0 (Design Features Providing Thermal Margin Beyond the Design Basis) and 3.0 (Assessment of Thermal Margin) all in CRBR-3, Volume 2 (Reference 2).

For those aspects of CDAs associated with energetics, i.e., short term and quasi-long term mechanical effects, the applicants have specified dynamic and longer term loadings based on a non-mechanistic pressure-volume (PV) relation which they believe bounds any reasonable evaluation of energetics associated with CDAs in the CRBR. (See p. 5-1 of Reference 1 CRBRP-3, Volume 1). That PV relation is utilized in defining the loads to be considered in analyses of the major primary coolant system components. The applicants have evaluated the response of these systems and components against stress limits, strain limits and leakage criteria. The loads and the criteria are described in Chapters 5.1, 5.2 and 5.3 of Reference 2.

For those CDA consequences not associated with energetics the applicants have specified several general requirements (Section 2.1 of Reference 2) together with several feature requirements which they believe assures that the plant will accommodate the effects of a core melt accident (ibid). These features include: a barrier (seals) between the reactor cavity and the containment atmosphere, liners for the reactor cavity and pipeway cells, a reactor cavity vent system, a containment purge system, a containment vent system, a containment clean-up system, an annulus (between the steel containment vessel and concrete confinement) cooling system, a containment system leakage barrier and instrumentation specifically for monitoring parameters associated with CDAs.

The general approach followed by the applicants is satisfactory. The specific functional requirements and evaluation guidelines are discussed, as required, in each of the following sections of this appendix which cover our review of the major areas of CDA evaluations identified earlier. The discussion in each of these areas include our evaluation of how the applicants' approach satisfies the guidelines recommended by the staff.

References

1. CRBRP-3, Hypothetical Core Disruptive Accident Considerations in CRBRP, Volume 1, Energetics and Structural Margin Beyond the Design Basis, Revision 4, March 1982.

S. CRBRP-3, Hypothetical Core Disruptive Accident Considerations in CRBRP, Volume 2, Assessment of Thermal Margin Beyond the Design Base, Revision 4, June 1982.

3. CRBRP-GEFR-00523, An Assessment of HCDA Energetics in the CRBRP, Heterogeneous Reactor Core, S. K. Rhow, et. al., December 1981.

4. Letter: HQ:S:82:110 John Longenecker to Paul Check, Amendment No. 72 to the PSAR for CRBRP, dated October 29, 1982.

5. Letter: HQ:S:82 162:John Longenecker to Paul Check, An Assessment of the Unprotected LOF Accident for the CRBRP Heterogeneous Core, D. Weber, et. al., December 1982.

6. Letter: HQ:S:83 222: John Longenecker to J. Nelson Grace, Reactor Closure Head Capability to Meet Margin Requirement, dated February 14, 1983.

7. NUREG/CR-3224 A Assessment of CRBR Core Disruptive Accident Energetics, T. G. Theofanous and C. R. Bell, dated March 11, 1983.

8. NUREG-0139, Final Environmental Statement Related to Construction and Operation of Clinch River Breeder Reactor Plant, Appendix I, February 1977.

9. NUREG-0786, Site Suitability Report in the Matter of Clinch River Breeder Reactor Plant, June 1982.

10. NUREG-0850, Volume 1, Preliminary Assessment of Core Melt Accidents at Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects, November 1981.

ATTACHMENT A.1-I

Reactor Arrangement

Reactor enclosure and head access area are shown in Figure 1. The reactor head access area is below the main operating floor within containment. Figure 1 reveals that this area is very congested around the control rod drives and their seismic support structure. This congestion is simply due to the number of things that have to be in this area such as cable raceways, auxiliary shielding and cooling ducts which supply cool air to keep electrical-mechanical equipment functional.

The guard vessel and guarded elevated inlet and outlet piping is shown. In the unlikely event of sodium leakage from the reactor vessel, the purpose of the guard vessel, piping enclosure, and elevated piping is to contain sodium and maintain a minimum sodium level in the reactor vessel which is above the reactor outlet piping inside of the reactor vessel. Aerosol leak detectors will pick up very small leaks from the reactor vessel and reactor coolant system and indicate an alarm.

Figure 2 is a schematic view of the reactor locating major structural components within the reactor and their relationship such as the:

1) Core Support Plate
2) Core Support Cone
3) Core Barrel
4) Horizontal Baffel Assembly
5) Upper Internals Structure and Upper Internals Structure Support Columns (4)
6) Reactor Vessel Closure Head
7) Rotating Plug Risers (3) sets with roller bearings and ring gears
8) Sodium Inlet Nozzles (3)
9) Sodium Outlet Nozzles (3)
10) Shield and Seismic Support

Figure 3 is a cut away view of the Reactor Lower Internals Structure Assembly and shows the arrangement of the core support plate, the circumferencial lip for attachment to the core support cone, (the attachment forging is an integral part of the reactor vessel), the core barrel, lower inlet modules, upper core former segments, fuel transfer and storage assembly, and horizontal baffel. The core assemblies consisting of fuel, control elements, radial blanket and radial shield assemblies are all seated in 61 lower inlet modules, each supporting 7 removable core assemblies. The 427 core assemblies are supported at the bottom, seated in the lower inlet modules. Lateral support for all core assemblies is provided by the core support plate, lower inlet modules, lower former segment and upper core former segment. Ultimately these loads are transmitted to the reactor vessel via the core support cone, horizontal baffle and thermal liner (see Figure 1).

Figure 4 shows the upper internals structure (UIS). Its function is to provide core holddown sleeves and shroud tubes for control assembly channel positions over the core. Four support columns for the UIS also contain tube guides for thermocouple elements positioned at the exit of the core. The UIS aids in mixing the hot sodium coolant exiting from the fuel channels. The UIS is suspended by these 4 large circular columns which in turn is supported by the intermediate rotating plug. The support columns are raised simultaneously by a ball nut screw jacking mechanisms. The UIS is raised only about 2.5 inches for refueling and plug rotation.

Figure 5 shows the reactor closure head assembly which forms the upper closure of the reactor vessel. It is comprised of 3 plugs supported by 3 sets of roller bearings at the edge of each of the three plugs. The plugs are rotated by electric motors. This allows the invessel fuel handling machine, mounted on the small plug, to reach any core position or fuel storage position. The underside of each plug has several layers of steel plates separated by spacers. The layers of steel plates perform three different functions: (1) minimize gas absorbtion; (2) thermal insulation; and (3) radiological shielding.

The load support system for the 3 rotating plugs is as follows. At the top of each set of risers (see Figure 2) is a bearing, ring gear and gas seals thus the weight of the small rotating plug and attachments are transferred to the intermediate plug which in turn transfers its weight including the weight of the upper internals structure, support columns and control rod drives through the intermediate plug risers and bearings to the large rotating plug (LRP). All of these weights are transferred to the outer set of risers and bearings to the large rotating plug which in turn transfers all loads to the reactor vessel flange. The reactor vessel flange sits on a composit steel/concrete/shielding structure which transfers the loads to the support ledge in the reactor cavity.

Figure 6 shows the reactor vessel flange and ledge support. The reactor vessel flange is secured to the ledge support by 72 bolts, 3 1/2 inch diameter approximately 10 ft. long. The ledge structure sits on a support ledge in the reactor cavity which has 180 bolts, 2 1/2 inches in diameter approximately 4.7 feet long embedded inconcrete.

Supplementary Figures 7 through 10 are included to give one an additional perspective on some of the parts of the reactor not specifically mentioned.

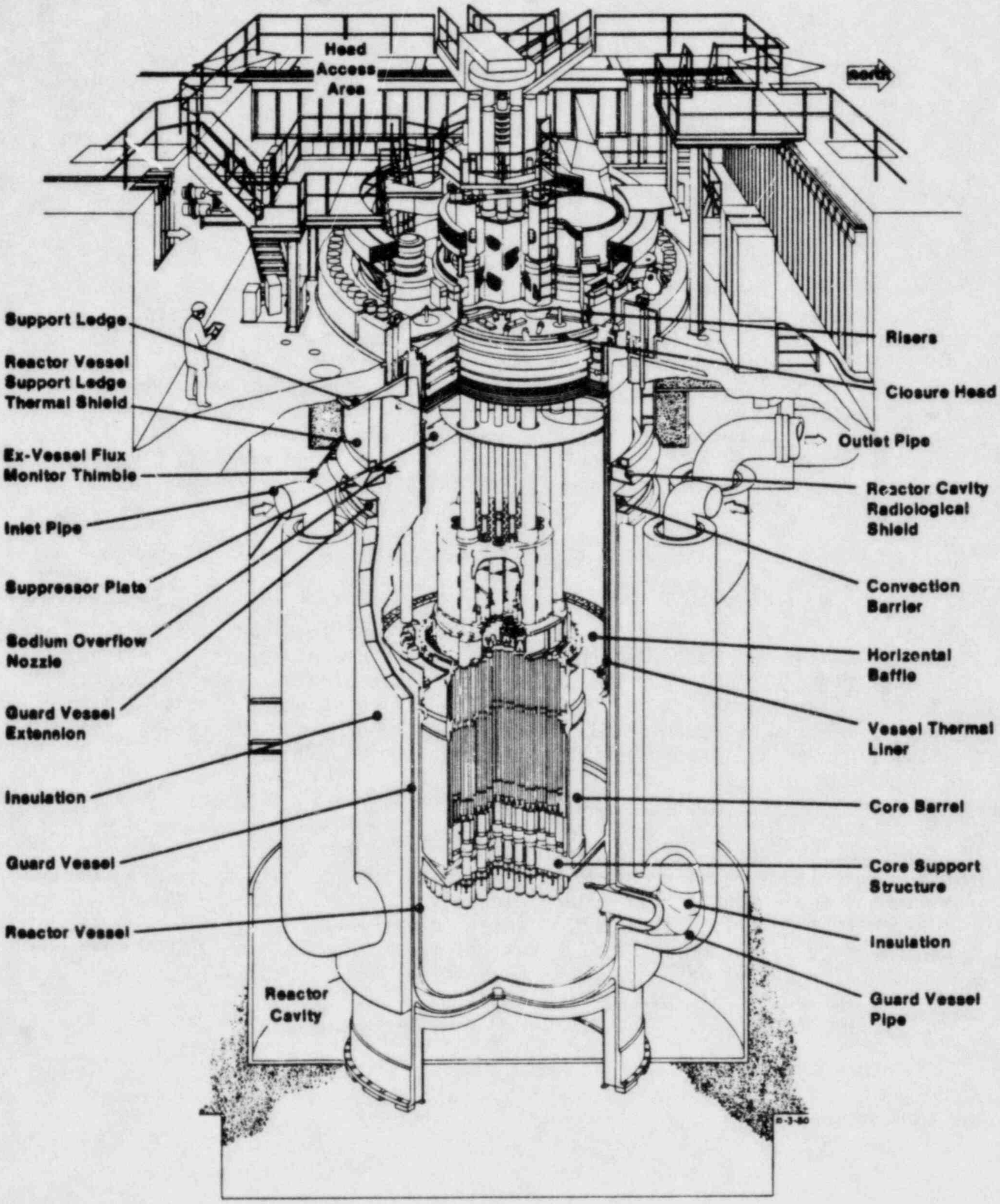


Figure 1 Reactor Enclosure System and Parts of Interfacing Systems.

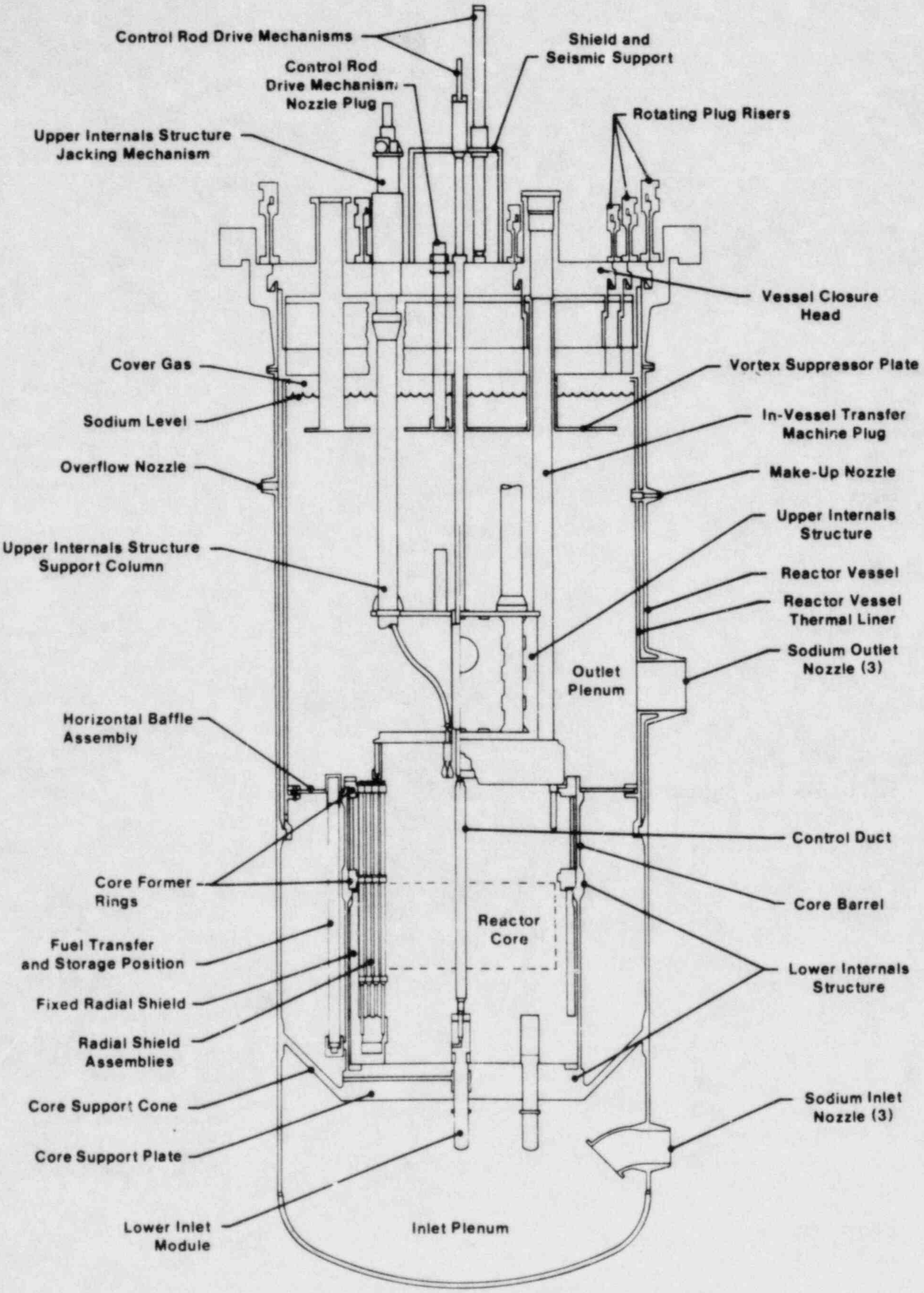


Figure 2 Schematic View of Reactor.

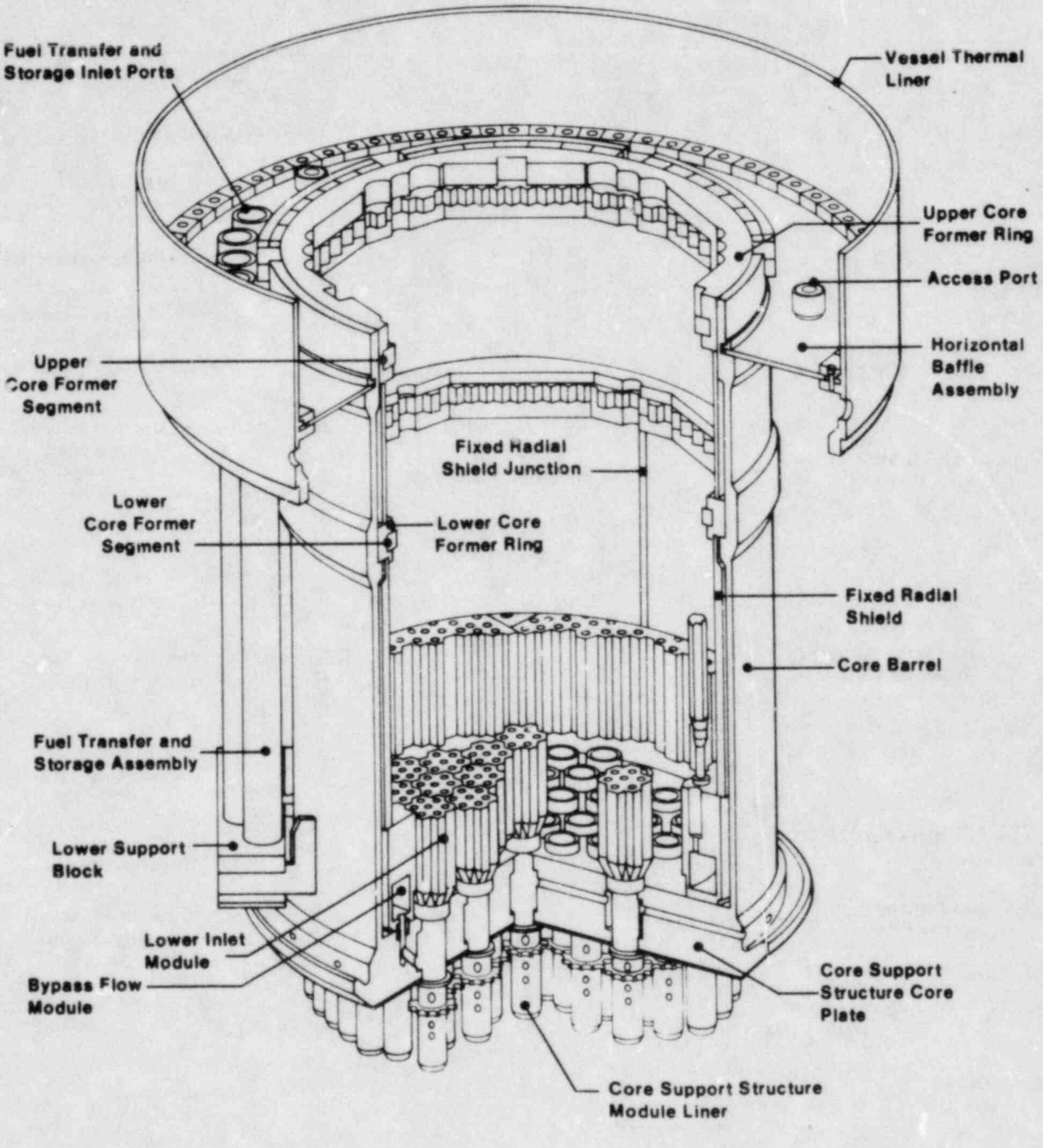


Figure 3 Lower Internals Structure Assembly.

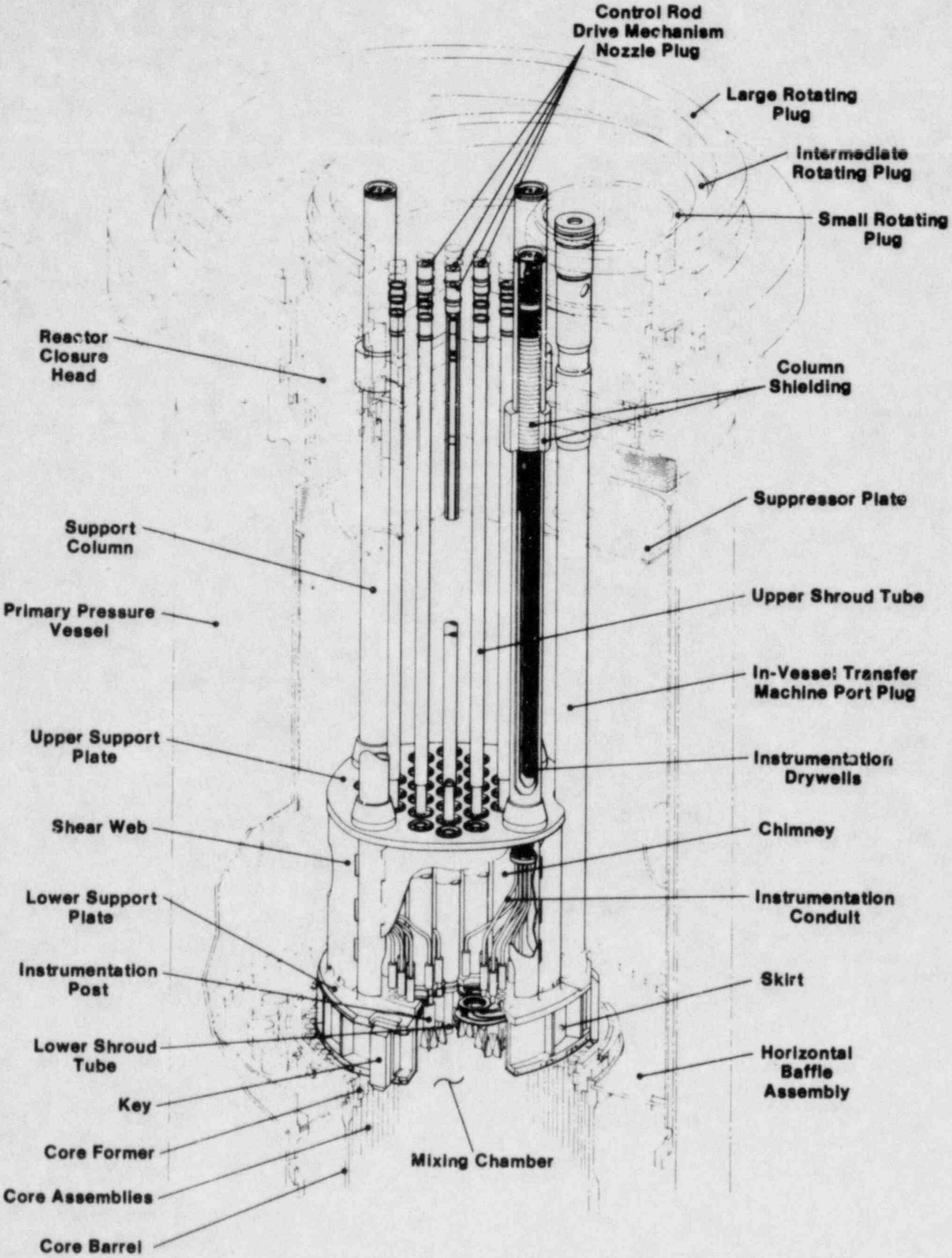


Figure 4 Upper Internals Structure

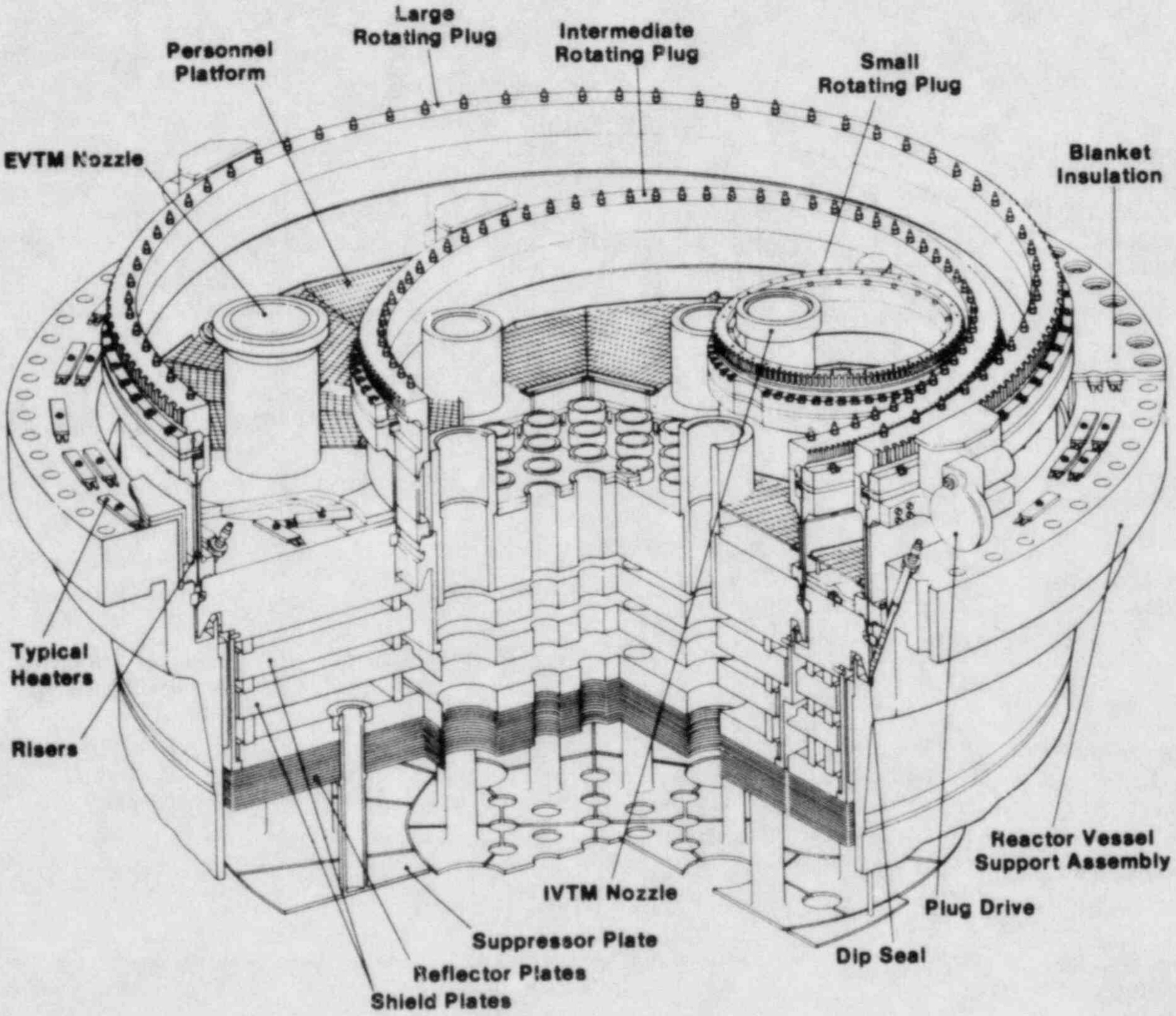
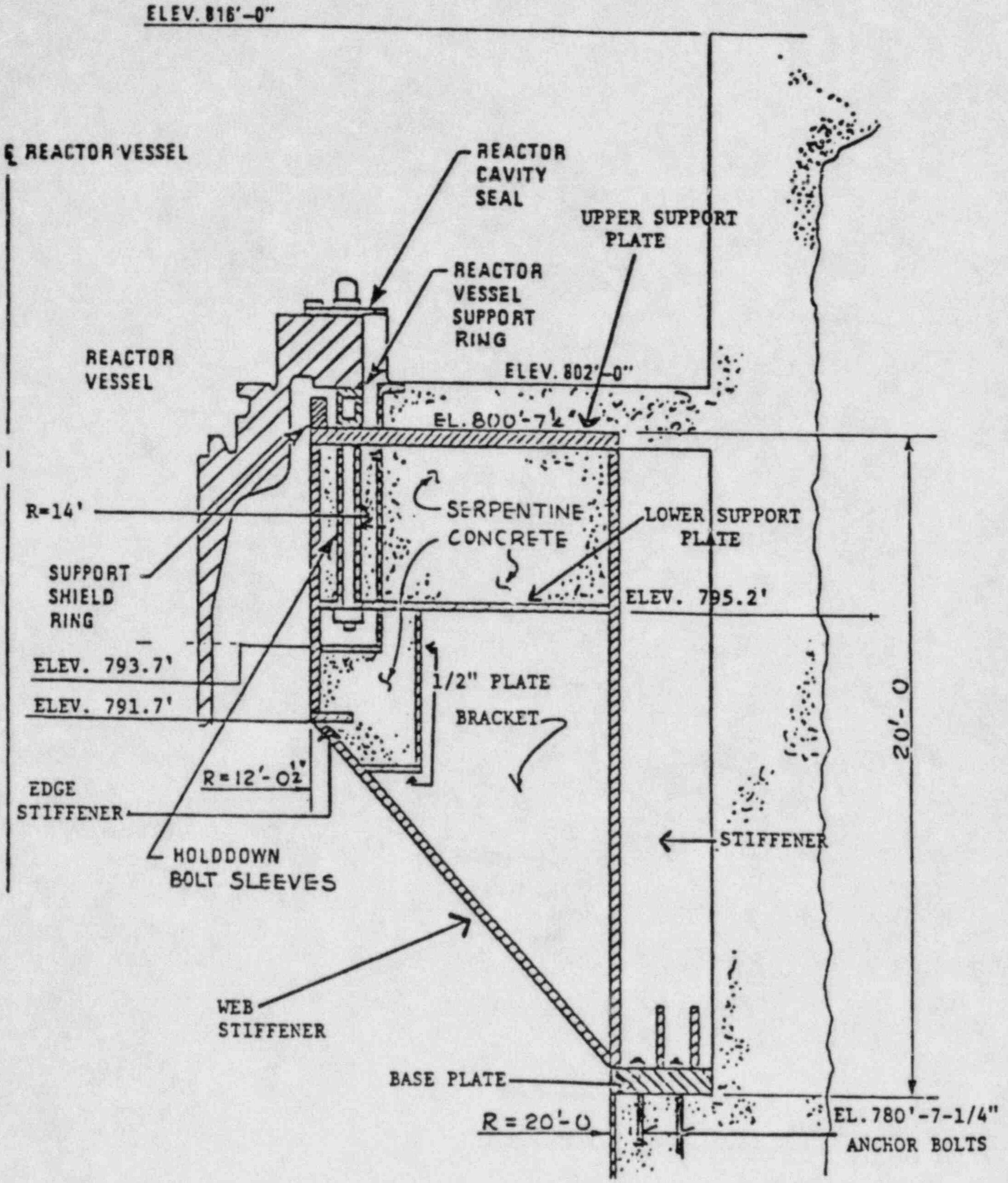


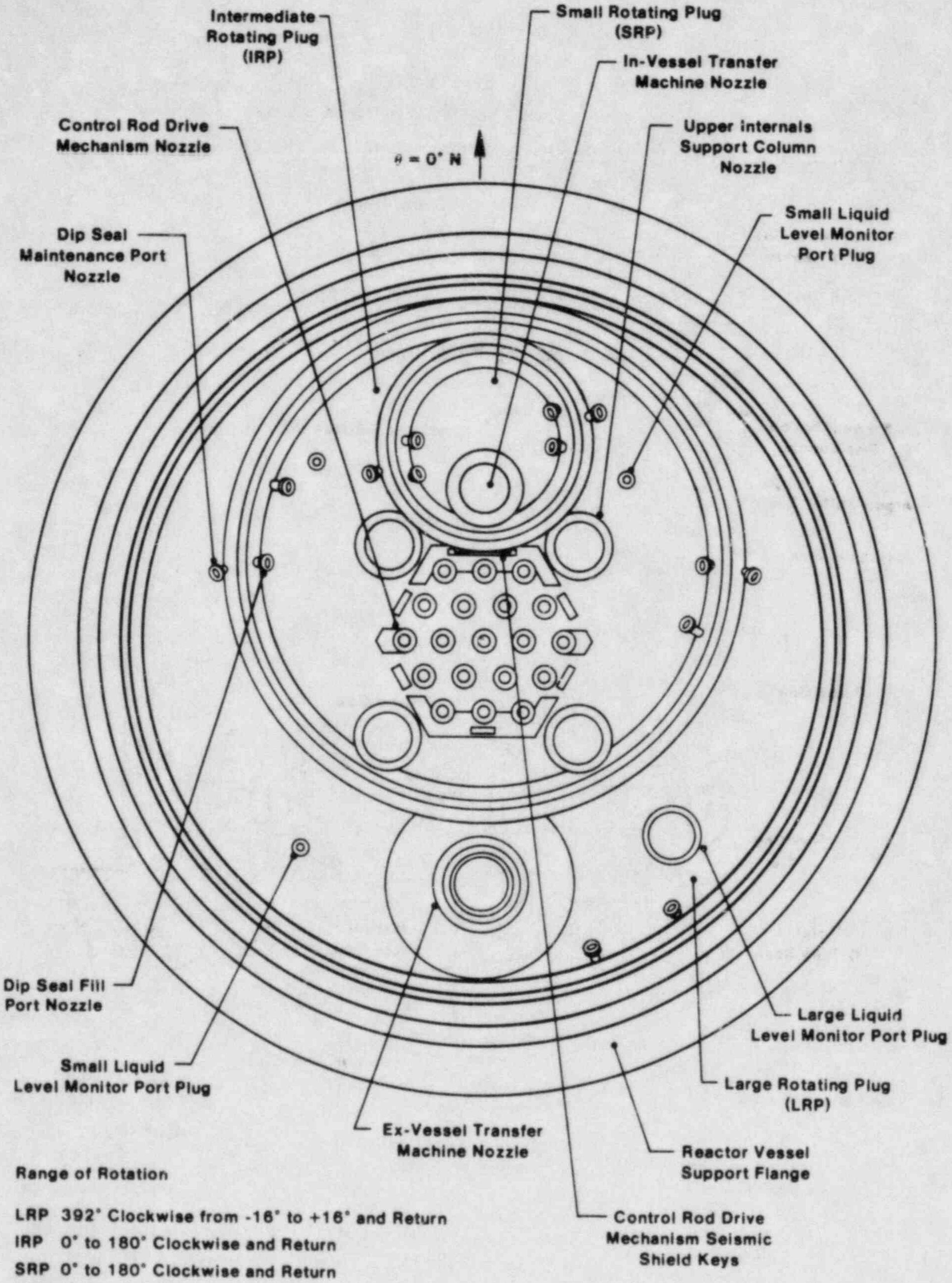
Figure 5 Reactor Closure Head Assembly



TYPICAL SECTION THROUGH REACTOR VESSEL LEDGE SUPPORT

FIGURE 6

SUPPLEMENTARY FIGURES



Range of Rotation

LRP 392° Clockwise from -16° to +16° and Return
IRP 0° to 180° Clockwise and Return
SRP 0° to 180° Clockwise and Return

Figure 7 Reactor Closure Head Assembly Plan View.

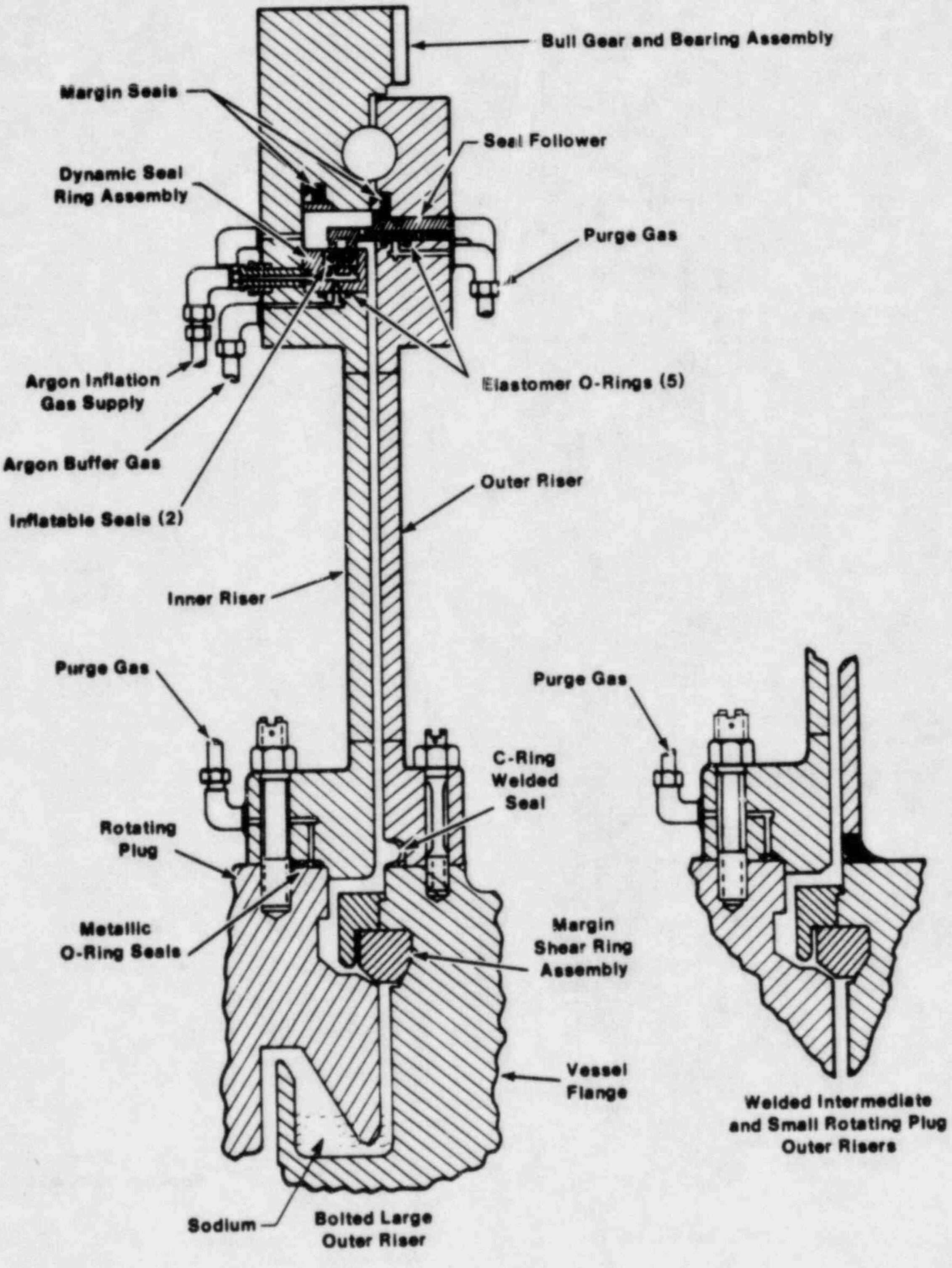


Figure 8 Riser Seals for the Reactor Closure Head Rotating Plugs.

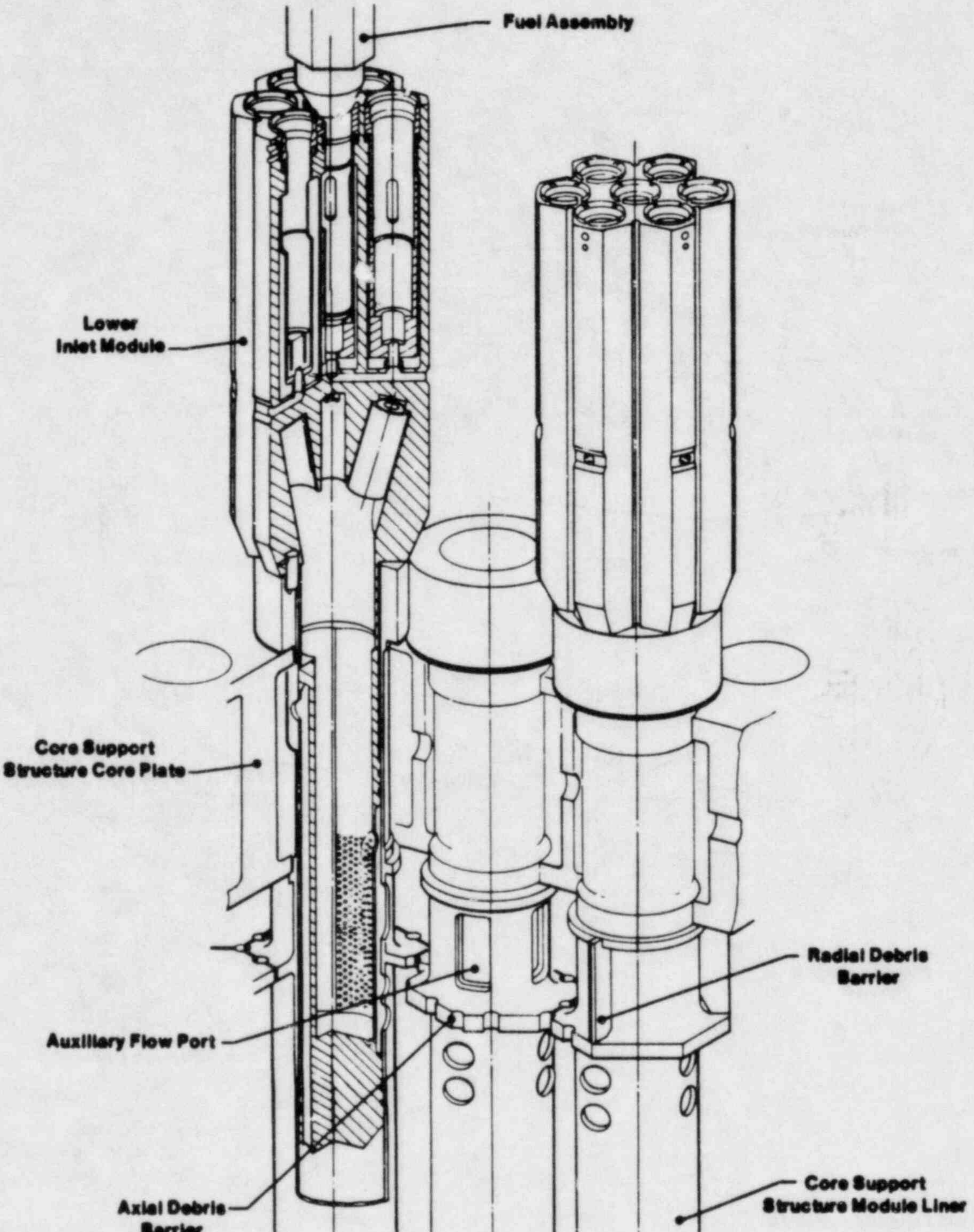


Figure 9 Arrangement of Lower Inlet Modules.

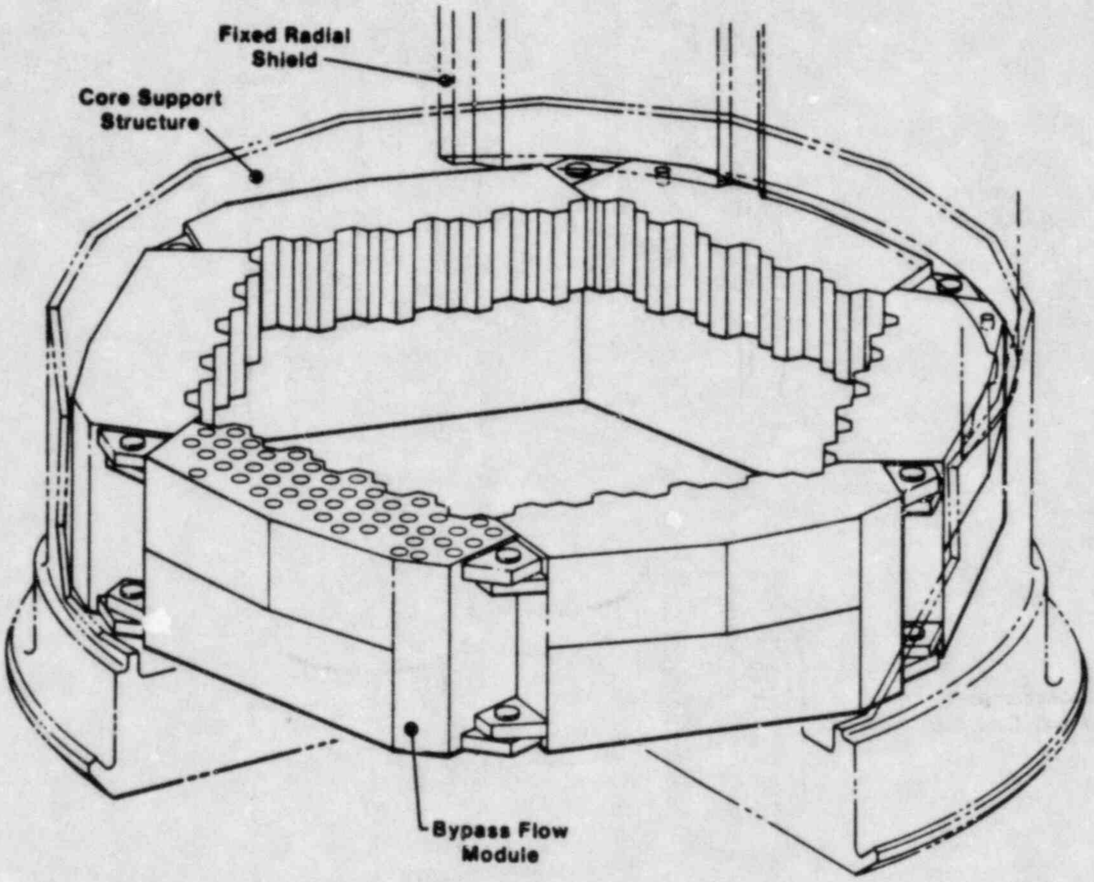


Figure 10 Bypass Flow Modules.

A.2 CORE DISRUPTIVE ACCIDENT ENERGETICS

In this section we provide a summary of an extensive independent assessment of the level of energetics associated with CDAs performed for the staff as a special task. Support for this task was provided by the Los Alamos National Laboratory (LANL) and was directed by Dr. T. G. Theofanous (Professor of Nuclear Engineering, Purdue University) and Dr. C. R. Bell (Associate Group Leader, Safety Analysis Group at LANL). The detailed study is reported in NUREG/CR-3224 (see Reference 7).

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.

The staff's review of the capability of the primary reactor coolant system to accommodate energetic CDAs is provided in section A.3. The results of the energetics study summarized here is compared by the authors to the energy absorption capability of the head established in section A.3.

On the basis of the results of their study the authors conclude that significantly energetic events are very unlikely and that, given the energy absorption capability of the head (see section A.3), the possibility of vessel head failure from CDA-induced energetics is physically unreasonable. The staff concurs in this assessment.

The following subsections contain the executive summary of the referenced report (NUREG/CR-3224).

A.2.1 Introduction

This section contains the results of our independent assessment of the energetic behavior resulting from postulated Core Disruptive Accidents in the CRBR heterogeneous core design. The object is to define in a reasonably conservative fashion, the magnitude of the mechanical energy releases against which the integrity of primary system, and of the reactor vessel head in particular, should be assessed. This effort began with a detailed review and evaluation of the applicants' positions, and their technical bases, on the subject and evolved, over a period of nearly 15 months, into a completely independent study with original elements on one or more of the following aspects: (a) accidents, phenomena, or effects taken into account, (b) analysis methods utilized, (c) experimental evidence brought to bear. The results of the review and evaluation effort have been documented in (1). The results of the independent assessment effort have been published separately (2) and are briefly summarized here. A similar structure and cross-referencing to the original document are utilized here to facilitate the search for additional details.

A.2.2 Overall Technical Approach

Depending upon whether reactor shutdown has been achieved, core disruption may initiate 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. Mechanistically a protected CDA is the result of sustained failure to remove decay heat and is commonly referred to as the Loss of Heat Sink Accident (LOHS). In the unprotected CDA case initial core disruption may occur due to either an undercooling or an overpower condition. Mechanistically, the undercooling would be the result of loss of coolant flow, known as the Loss of Flow Accident (LOFA), and the overpower due to an uncontrolled reactivity insertion, which is commonly referred to as the Transient Overpower Accident (TOP). In general terms, these three accidents exemplify the generic behavior over the whole range of the CDA spectra of circumstances, hence, they can be used to adequately characterize the spectra of energetic consequences.

Another class of CDA initiators, that of Fuel Failure Propagation (FFP) has also been identified and extensively studied in the past. The evidence is conclusive now that the attainment of whole core disruption through such a mechanism can be neglected. Finally, various combinations of functional failure events (TOP/LOF, etc.) and/or of structural failures (i.e., due to extreme external events such as earthquakes beyond the SSE, yielding core support failures, loss of piping integrity, etc.) have also occasionally been considered. Our review of these areas indicate that those few cases for which severe energetics behaviour cannot be precluded at this time are of sufficiently low probability to be neglected.

Our approach consists of realistically following each one of the three generic CDA initiators through the core disruption phases and until accident termination. These so-called mechanistic CDA analyses provide an overall framework against which the potential for energetic phenomena is assessed with due regard for the controlling physical processes. In terms of actual licensing cases, the first efforts along these lines were made during the Regulatory review of the Fast Flux Test Facility (FFTF) CDA energetics assessment. The approach further matured with the initial (homogeneous core) CRBR application and licensing review.

It would be in error, however, 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 a 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 do not attempt to associate a simple outcome to a given initiator. Rather we establish a "range of phenomenology" consistent with experience and known physical principles. Within this range we search for energetically-prone circumstances, identify the important mechanisms, and quantify the intensity of energy release in a reasonably conservative manner (i.e., avoiding excessive and clearly non-physical conservatisms). Similarly, we scrutinize for termination-favoring phenomena, identify the important mechanisms, and quantify the approach to termination by the fraction of fuel removed from the core region (at approximately 40% removal permanent subcriticality, i.e., termination, is achieved). Based on these results we complete the assessment by synthesizing sequences and respective likelihoods.

These analyses were carried out by means of the system computer codes SAS3D (and to a limited extent the most recent version SAS4A) and SIMMER-II. These codes are used as "integrators" of the technical base and their results are 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 here to provide overall guidance in this regard.

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 opinion that: (a) the LOFA phenomenology spans the range of energetically significant CDA behavior, (b) within the LOFA sequences our previous review effort identified specific and significant areas of concern, (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 the neglect of the unique aspects of these other 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.

A.2.3 The Loss of Flow Accident

- Generalities

From the initiation of core disruption (i.e., initial clad melting) the LOFA would evolve through a continuum of gradually escalating core disruption states until complete disruption (i.e., melting of all materials found within the original core confines, also known as a whole-core pool) occurs. Energetically this progression is important while a sufficient fraction of the initially present driver fuel (approximately 60% for the CRBR) 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 subcritically, 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 simply "dispersal." Our overall objective is to determine the relative likelihood of these two termination paths as a function of the degree of core disruption, and to quantify the damage potential of the energetic ones.

Energetic behavior is the consequence of rapid reactivity insertion. For the present CRBR design such reactivity increases can only result from sizable, and generally compactive, fuel motions. When such motions occur from fuel in the process of undergoing disruption we speak of "Initiating Phase Energetics." When such motions occur due to compaction from highly, but temporarily, dispersed fuel states, they are called "Recriticalities." The character of these two energetic phenomena is fundamentally different with regard to both the reactivity-yielding mechanisms as well as with regard to the resulting damage potential. The structural capability of the system provides an appropriate perspective against which the damage potential of a given energetic event must be viewed.

- Structural Capability of CRBR

The levels of energetics required to produce significant structural damage in the CRBR were evaluated (2, Section II.2) taking into account an "inner containment" formed by the Core Barrel/Upper Internal Structure/Core Support Structure envelope. In addition, the pressure transmission characteristics of the expanding core medium and other materials found within were taken into account. This mechanism has an important implication on the resulting short term loading characteristics of the immediate structures (Core Barrel). 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. Our structural analyses indicate that a level of energetics equivalent to 1130 Megajoules (MJ) (isentropic expansion yield to one atmosphere) would be required to breach this inner containment. That is, no energetic release against the boundary of the primary system can be expected for any energetics below this level. At still higher levels upward displacement of the UIS and a longer term expansion against the sodium pool would take place. For the heterogeneous CRBR core this is the only sequence leading to the opportunity for large scale fuel-coolant-interaction. Experimental evidence indicates that, under these specific contact conditions, this interaction would not yield pressure augmentation and that the energy conversion process would be controlled by two phase choking and a minimal fuel coolant heat transfer. Evaluations of the long-term expansion phenomena indicate that an energetic event of nearly twice this magnitude, approximately 2550 MJ would be required to produce a slug impact kinetic energy close to the head design capability of 75 MJ. The 1130 MJ and 2550 MJ energetic levels correspond to 100 dollars per second ($/S) and 200 $/S disassemblies respectively occurring in the two phase regime.

- Initiating Phase Energetics

A number of SAS3D analyses covering a broad range of the important parameters were carried out to characterize the range of initiating phase LOFA behavior (2, Section II.3). With the lower coolant void reactivity of the heterogeneous CRBR core, the problem area (3) for the previous homogeneous CRBR core design, is avoided. This LOF-d-TOP situation arises due to the development of high overpower conditions that lead to pin failures in unvoided subassemblies. If such failures occur in the core mid-plane, which, based on available evidence, cannot be excluded, a potentially autocatalytic behavior may result from the rapid in-pin fuel motion towards the failure location (core mid-plane).

Although these analyses revealed insufficient power augmentation to reach the LOF-d-TOP condition, even those cases calculated with the parameter choices favoring a "slow" accident exhibit a substantial neutronic activity (i.e., overpower conditions). This activity is caused by extended fuel motions (i.e., following the initial tendency to disperse due to retained fission gas pressures) and gives rise to a process we call co-disruption. Co-disruption is the result of accelerated core disruption such that there is insufficient time for the molten cladding to separate from the fuel prior to core-wide fuel disruption. Co-disruption favors dispersal since it implies higher (steel) vapor pressures, increased penetration potential into axial blanket areas, and remeltable blockages.

Plenum fission gas induced fuel compaction has been proposed as another mechanism for initiating phase energetics (4). In the presence of plenum pressure the fuel pin becomes unbalanced upon fuel disruption resulting in rapid downward ejection of the blanket and undisrupted driver fuel pellets. The applicants analyzed this mechanism in response to questioning during this review process, and concluded that there would be adequate time for the plenum fission gas to escape prior to fuel disruption. Based on the results of our own analyses we cannot agree with this conclusion (2, Section II.4). However, we have found no reasonable fuel compaction process which significantly exceed approximately 50 $/s, which as we will see below, represents an amply tolerable level of energetics. However, at the time of this energetic event, only one-half of the core has been voided and the resulting high overpower could induce an LOF-d-TOP event on the other half. Such a combination of energetic events is judged as highly undesirable. The staff requires the applicants, therefore, to address this concern in their operating license (OL) application. Resolution at that time is feasible because a design fallback has been identified which would limit the action of these pressures during the initiating phase of the LOFA (see Reference 5).

- Recriticality Energetics

The general behavior of the post-initiation period was examined (2, Section II.5) both in terms of a SIMMER-II integral system calculation as well as in terms of generic ad hoc evaluations of relevant physical processes.

The integral calculation was a continuation of one of the SAS3D analyses. The overlap portions between these two calculations were in excellent agreement. The results depict a generally active sequence, with regular power bursts corresponding to fuel reassembly motions. Some evidence of progressive coherence or "tuning" is noted, however. The power oscillations in the early portion appear substantially damped. The effect of the associated pressurization transients to force molten fuel (and steel mixture) through the axial blankets and, upon melting of subassembly duct walls that are adjacent to internal blankets, through inter-subassembly gaps, and away from the core region is evident. The modelling allows for freezing and plugging of such paths, and indeed such behavior is observed in the results. Merging of the Subassembly-Scale (S/A-scale) pools (annular pool geometry) and destruction of the internal blanket barrier (whole-core pool) occur successively within only a few seconds. Upon attainment of a two-dimensional character the power oscillations amplify because of increasing severe sloshing pool motions. However, homogenization of the internal blanket regions develops slowly, hence radially focused sloshings are inhibited, the system's total available reactivity is well below that of a homogenous pool, and the associated power bursts are non-energetic. This delay is sufficient to allow removal of the final small quantity of fuel required for termination prior to the formation of a homogeneous whole-core pool, even though the inter-assembly gap escape paths of the radial blankets were, conservatively, not modelled in this calculation.

Recognizing that this integral calculation is one of a few ever attempted, the above results must not be (and were not) taken at face value. The mild termination potential was evaluated (2, Section II.6) in terms of separate effects calculations that model in great detail the flow path, the flow constituents and thermal interactions including freezing and plugging phenomena. Prototypic experimental data were utilized to benchmark these calculations. Even under modest pressures compared to those expected in view of the continuing neutronic activity, adequate fuel removal to assure permanent termination is estimated to occur prior to the formation of a homogeneous whole-core pool.

Gravity driven recriticalities were examined from the point of view of amplification potential (2, Section II.7). For the S/A-scale and annular pool phases under natural power burst perturbations, the fuel column will separate initially into a compact lower mass and a distributed upper segment of approximately material quantity. Reassembly under conditions of reduced fuel inventory or low heat losses (minimal boilup) would produce a growing lower liquid puddle within which the peak of the axial power distribution will occur. Hence, reassembly energetics would be mitigated strongly by single phase liquid expansion feedback during the power transient. Reassembly under conditions of high inventory or high heat losses (large scale boilup) on the other hand would produce low ramp rates and therefore would be effectively controlled by two-phase dispersal during the power transient. In addition the S/A phase cannot have a core wide coherence because the time interval to S/A wall disintegration permits only a few power cycles which are insufficient to complete the "tuning" of the fluid dynamics. These reassembly ramp rates would be small. However, even if we assure complete core wide coherence, maximum ramp rates of less than 100$/s are obtained. Therefore no physically reasonable threat to the vessel head structure can be seen from these first two stages of disruption.

For the whole-core, homogeneous pool an amplification mechanism was calculated. Under perfectly symmetric conditions (geometry and power distribution) a radially focused sloshing action is observed which, under certain conditions of material configuration, may produce high reactivity insertion rates. In most such cases single phase expansions dominate and negligible energetics result. For example in one such case we considered an in-slosh with 300 $/s ramp at prompt critical yielded quick, single-phase thermal expansion shutdown and produced negligible energy release. However, there is also a narrow range of conditions over which substantial energy releases are calculated. This is particularly so when two-phase regions exist. For example in one high-inventory case considered, prompt criticality was obtained earlier in the in-slosh and while a two-phase field still dominated the central pool portion. The resulting reactivity of 125 $/s produced the energy equivalent of the 100 \$/s two-phase disassembly considered in our structural evaluation. The importance of symmetry in such evaluations is highlighted by the integral SIMMER-II calculation of core disruption. This calculation did enter the whole core pool and it did indicate radial sloshing and amplification. However, due to the system non-homogeneity in the early stages of this phase, a non-centered power distribution results, hence radial focusing is absent and a non-energetic behavior is observed. Before there was a chance to obtain homogenization of the internal blanket material, permanent termination of neutronic activity by fuel removal was indicated.

- LOFA Energetics Summary

Based on the above no physically reasonable energetic event of sufficient magnitude to violate the vessel head structure can be identified.

A.2.4 The Transient Overpower Accident

The TOP-unique behavior (2, Section III) 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 pin-internal fuel motion 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.

The applicants have provided extensive analyses for a variety of core burnup states and reactivity insertion rates. Our assessment focused, therefore, on more closely 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-3 (EOC-3) core with the replacement of the six high power driver fuel assemblies with blanket material is the most coherent. On the basis of failure modes and effects analysis of the reactor control system, we concluded that ramp rates of 10-12 cents per second (¢/s) are more than one order of magnitude less probable than those of 2 ¢/s or 5-8 ¢/s. Furthermore 15-20 ¢/s ramps are more than three orders of magnitude less likely than those of 10-12 ¢/s. As a conservative upper limit we selected therefore the 10-12 ¢/s TOP for this investigation.

The EOC-3 core CRBR TOP accident was simulated with the PLUTO2/SAS4A computer code. A failure inchoherence (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 available experimental data from the L8 TREAT TOP test (L8 is a transient test run in the Transient Reactor Test Facility). The calculated sweepout was seen to successfully mitigate pin internal fuel motion reactivity (and a small amount of sodium voiding reactivity) and to produce shutdown within the first 100 ms.

Thus, even under the most limiting conditions of core coherence and pin failure location, no energetic behavior could be found for TOPs of up to 10-12 ¢/s. For TOP cases with higher ramp rates energetic behavior cannot be precluded. However, the frequency of such events is sufficiently low that they can be excluded from consideration.

A.2.5 The Loss of Heat Sink Accident

The LOHS-unique circumstances (2, Section IV) originate from core disruption occurring at very low power and in the absence of sodium coolant. The absence of coolant is required since natural convection boiling has been shown adequate to remove heat at decay power levels. Core uncovery may occur either due to coolant boiloff or due to reactor vessel failure at the high temperature LOHS environment. The actual mechanism is not important, affecting only the disruption stage power level which in any case is very small and much more dependent upon the other aspects of the accident scenario. Characteristically, however, disruption would not occur until many hours into the accident, indicating significant margins for recovery.

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, 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 at a lower porosity. The initial porosity is approximately 65% and a porosity of approximately 50% would be required to approach criticality. This eventual reaching of 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. A whole core pool, with homogenization of all internal, axial and radial blankets would result in this case. The resulting dilution would be adequate to render the system permanently subcritical even in the absence of the steel and control rod materials which will eventually separate.

Furthermore, in the absence of the sodium pool, even the most severe recriticalities could provide no challenge to the reactor vessel head integrity. As an example we considered the 200 \$/s discussed in the LOFA as the energetic level required to substantially challenge the vessel head integrity. The expansion forces on the UIS, assuming absence of significant resistance by its 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.

A.2.6 Conclusions

- We have systematically evaluated the possible progression of all three classes of CDAs as exemplified by the LOF, TOP, and LOHS accidents. Non-negligible energetic circumstances were identified only within the LOFA sequences, and, assuming that the plenum fission gas fuel compaction mechanism becomes inoperative by design, as recommended, only as a consequence of recriticality.

- Recriticality events in the S/A-scale and annular pool phases cannot be excluded. However, thier magnitudes are limited to the order of 50 \$/s or less because of incoherence and the absence of significant amplification. 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 availability 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. The amplification is the result of radial sloshing following a centrally located and symmetrically distributed power pulse. Even so, the resulting level of energetics did 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 found within. We conclude that an 1130 MJ accident (expressed as the isentropic work potential for expansion to one atmosphere) would be required to fail this inner containing structure, and a 2550 MJ accident would be required to substantially challenge the reactor vessel head structure i.e., 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.

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

- Further, based on the projected absence of significant energetic events we conclude that the applicants' energetic source term of 661 MJ (75 MJ slug impact kinetic energy), as applied by the applicants for evaluating the structural margin beyond design basis is adequate.

A.2.7 References

1. Technical Evaluation Report, LANL 1982.

2. T. G. Theofanous and C. R. Bell, "An Assessment of CRBR Core Disruptive Accident Energetics," NUREG/CR-3224, March 11, 1983.

3. J. F. Meyer, L. Lois, J. L. Carter and T. P. Speis, "An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Accident Energetics," NUREG-0112, March 1977.

4. T. G. Theofanous, "Multiphase Transients with Coolant and Core Materials in LMFBR Core Disruptive Accident Energetics Evaluations," NUREG/CR-0224, July 1978.

5. Letter: HQ:S:83:234 John R. Longenecker to J. Nelson Grace, Fission-Gas-Driven Compaction, dated March 8, 1983.

A.3 EVALUATION OF STRUCTURAL MARGINS BEYOND THE DESIGN BASIS (SMBDB)

A.3.1 Introduction

This section addresses those CDA consequences which are postulate release sufficient energy in short enough time periods (millisecon seconds) to challenge the structural capability of the primary cool system through dynamic pressure loads. The evaluation of the poten for CDA sequences that result in such energy releases was present the preceding section. It was found that CDA progressions that re in large energy releases are very unlikely to occur.

In terms of evaluating the capability of the primary system to accommodate such events the quantity of major interest is the kinetic energy imparted to the sodium above the reactor by the vapor expansion process. The kinetic energy is specified at the time of the sodium slug impact with the underside of the vessel closure head. The studies performed to date, as reported in the preceding section, indicate that for conservative reactivity insertion rates, the sodium slug kinetic energy falls below 75 MJ at the time it impacts the head. The 75 MJ value is significant since the applicants have stated that the CRBR will be designed to accommodate the impact of slug with a kinetic energy of this magnitude (Reference 1). The evaluation of the capability of the CRBR primary system to withstand an energy release of this magnitude is the subject of the remainder of this section of the SER.

A brief description of the phenomena involved will be useful here. Reference to Attachment A.1-I Figures 1, 2 and 3 will help the reader place the following discussion in perspective. As the fuel and sodium vapor bubble expands within the core region it first loads the structures surrounding the core. The vapor also expands upward through the upper core subassembly structures (UCS). This loads the lower surface of the upper internals structure (UIS) which is located a short distance above the core subassembly outlets. As long as the UIS columns do not buckle and the UIS remains in place, the expanding fuel and sodium vapor is dispersed rapidly and is throttled by this relief path.

The ability of the expanding vapor to accelerate the mass of sodium above the core is thus limited and only a fraction of the total CDA energy is applied to the reactor head. At some energetics level the UIS columns will buckle and the UIS will be displaced reducing the throttling effect and allowing the expansion process to work directly on accelerating the sodium mass above the core. This sodium slug impacts the upper internals structure (UIS) and loads the upper vessel wall and impacts the thermal shielding suspended below the reactor head. The load path from the thermal shielding under the reactor closure head goes through the three rotating plug closure head, margin shear rings and risers and bearings and then through the reactor vessel support structure to the reactor cavity wall support ledge. As the compressed sodium loads the vessel walls and head, the cover gas compresses and sodium may enter the cover gas system. The latter event is probably of little importance in the time domain of concern in the effect of the CDA. The high pressure transient will also be transmitted into the primary heat transport system (PHTS) piping and components and into the Overflow and Makeup Systems. Rapid pressure attenuation occurs with changes in direction, area changes and gas spaces.

The energy release, for which the CRBR primary system is designed, is postulated to occur nonmechanistically by the applicants. It is characterized by a pressure-volume (PV) curve representing a hypothesized isentropic fuel vapor expansion starting from an initial, highly disrupted core state. The methodology (Reference 2) used by the applicants have been used by others (FFTF) and is basically a SAS-Venus scenario followed by REXCO-HEP hydrodynamic analysis. Detailed analysis of areas of concern are performed separately.

Only a fraction of the energy represented by the area under such a PV curve could drive a fluid slug to impact the underside of the vessel head. In the expansion process a large fraction of the energy is dissipated in other ways such as by permanent deformation of other internal structures in the vessel and in deformation of the reactor vessel and reactor closure head. Energy is also dissipated through hydrodynamic turbulence and friction. Ultimately a substantial portion of this energy is dissipated by raising the temperature of the sodium, and other materials such as reflectors and structures within the reactor vessel.

If the vessel or head and head mounted components do not fail, then all components in the reactor coolant system are subjected to a quasistatic sodium vapor of approximately 300 psi following sodium slug impact on the reactor head. Then failure of the reactor coolant boundary would most likely occur in the region of the reactor outlet nozzle piping subjected to the high pressure transient at elevated temperatures. The accident progression beyond this phase will be considered in the next section (A.4) on the thermal margins beyond the design basis. The staff and staff consultants' evaluation of the applicants' SMBDB analysis and recent SRI test results is provided in this section. As we shall see the results of these tests have led to several new considerations with regard to the current head design.

A.3.2 Criteria for Evaluation of SMBDB

A.3.2.1 Design Requirements

The design requirements for the reactor closure head and head mounted components remain as stated by the applicants in CRBRP-3, Volume 1, Revision 3, Section 5.2.2.1, Overall Requirements, paragraph C, pp. 5-15, they are restated below for convenience.

The closure head assembly, including head mounted components, shall remain intact and integral such that containment and leakage requirements are met. To meet these requirements, the following are necessary:

1. The bolting systems for head mounted components shall maintain attachment of the components to the head such that no head mounted equipment or part thereof shall become a missile above the head capable of impairing the containment barriers.

2. The nozzles of head mounted components shall accommodate the head loadings.

3. The response of head mounted components to SMBDB loadings shall not prevent head components from functioning as limited leakage barriers.

4. The staff and its consultants add the following requirement: the reactor vessel head and head mounted components shall accommodate the longer term mechanical loads from saturated vapor.

A.3.2.2 Structural Evaluation Criteria

In assessing the criteria for use in SMBDB assessments we must remember that the loads are from a BDBA and the overall requirement is for the structures not to become missiles and to maintain functional integrity in containing or controlling the release of sodium and fission products into the reactor head access area. The most appropriate criteria available in current codes or standards for evaluation of beyond design basis accidents are those presented in Appendix F of Section III of the ASME Boiler and Pressure Vessel Code for Level D Service Limits. The staff believes that SMBDB criteria should follow the spirit of Appendix F, but for the reasons given below, those criteria should be relaxed and expanded. For ASME Level D Service Limits, general deformation may occur with some consequent loss of dimensional stability and damage requiring repair. The spectrum of CDAs also includes those with little energetics or damage. That is, the structure is intended to continue in service after the component is repaired. For use of the upper bound energetics the structural concepts in CRBR SMBDB evaluations, the structure will not be in service after the accident scenario is complete and many of the internal structures may fail well before completion of the accident scenario. The limits given in Appendix F of the ASME Code relate only to stresses, strains and deformations resulting from primary loads. Secondary, or deformation controlled quantities, are not considered. For SMBDB assessments the maximum applied loads are not strictly primary and are energy controlled. Therefore, because the energy release is governed by the CDA, the applied loads decrease as plastic strains occur and energy is absorbed. Also, the strain limit restrictions for Level D Service Limits are based on stability requirements for primary loads and are therefore low; low enough to preclude ductile fracture. We can allow higher strain limits for SMBDB analyses so ductile fracture must also be considered.

Structural failure modes that would be expected during an energetic CDA result either from tensile plastic instability or local ductile rupture. The applicants present criteria in terms of membrane stresses or strain to protect against tensile plastic instability; and in terms of local stress or strain to protect against ductile rupture. If stress limits are used they are derived from elastic dynamic or equivalent static methods. If stress limits cannot be met with elastic analyses, strain limits must be satisfied for dynamic inelastic analyses.

The membrane strain limit is based on the work of Hillier (Reference 4) where the strain criterion is substantiated with experiments performed on uniformly loaded thin-walled tubes. The criterion is further substantiated by data obtained from tests performed on pressurized tubes and disks, cruciform specimens, and a small pressure vessel (Reference 6). A variety of materials were used in the tests including stainless and low-carbon steel. The criterion is comparable to that given in paragraph F-1324.4 of Appendix F to the ASME Boiler and Pressure Vessel Code, Section III, Division 1. During the staff review the applicants provided supplemental information for their structural criteria (Reference 5). The applicants will incorporate this material for additional clarity and guidance in a revision to Appendix B of CRBRP-3, Volume 1 (Reference 2) for the FSAR.

To preclude local ductile rupture the applicants use a strain criterion that is an extension of McClintock's hole growth theory for cylindrical holes in a biaxial, uniform tensile field (Reference 5). The applicants have presented data from several tests that substantiate the theory (Reference 6). Several different test configurations were used where the triaxiality factor, TF, varied from 1.0 to 4.0. Materials used in the tests included stainless- and low-carbon steels. The effect of the triaxiality factor can be seen in equation (1) where

$$\varepsilon^{p}_{max} \leq 0.7\ \varepsilon_f \left\{ \frac{\mathrm{SINH}\ [\sqrt{3}/3\ (1-n)]}{\mathrm{SINH}\ [\sqrt{3}/3\ (1-n)\mathrm{TF}]} \right\} \qquad (1)$$

ε_f is the uniaxial fracture strain and the triaxiality factor, TF, is equal to three times the first stress invariant divided by the equivalent stress. The applicants have conservatively limited TF to be equal to, or greater than, 1.0. This means that for pure shear, where TF is actually zero, it is artificially set to 1.0 giving an allowable strain equal to that for uniaxial tension. Based on equation (1), this is a conservative approach. For some materials tested in pure shear, however, the fracture strain is lower than in uniaxial tension. The staff and consultants don't believe this to be the case for materials (stainless steel and SA 508) comprising the primary boundary for SMBDB. Both the membrane and local stress criteria are derived from the strain limits described above. Because they are only used for elastic analyses, use of these criteria is very conservative.

A.3.3 Computer Codes and Models Used in the Analysis and Assessment

The REXCO-HEP computer code (Reference 3) was used to determine the dynamic response of the reactor vessel and internals to the CDA. It was also used to develop the component margin requirements (loads, displacements and energy partitioning) that are applied to the vessel head, PHTS and other components of the vessel and attached systems in order to evaluate the response of these components to transient overpressure. REXCO-HEP is a coupled hydrodynamic-structural computer code that is two- dimensional and uses axisymmetric geometry.

REXCO-HEP was developed at Argonne National Laboratory (ANL) and has been extensively benchmarked with data from the SRI scale model experiments for CRBR (Reference 6) and in the past with similar experiments for the FFTF reactor. Comparison of pre-test REXCO-HEP predictions with experimental results leads to several conclusions that are discussed in Section 3.1.1.3 of Reference 3. One of the most important of these conclusions is that, because of the general lack of energy dissipating mechanisms in REXCO-HEP, the predicted pressure loads and resulting strains in the vessel are predominantly conservative. A rigid head is assumed which also increases computed loads on the reactor vessel and internal structures. The supporting CRBR scale model tests will be discussed in a following section.

In the CRBR REXCO-HEP model the vessel and core barrel are treated as thin elastic-plastic shells along which the coolant is allowed to slide. The core support structure was represented as a very stiff elastic-plastic plate. The blankets, fission gas plena and radial shields in the model had the same geometric configuration and inertial properties as the actual materials but were considered to have no tensile strength. The vessel head was modeled as a rigid, circular plate supported at the edges and the Upper Internals Structure (UIS) were not included.

Transmission of pressure transients in the Primary Heat Transfer System (PHTS) were calculated separately with the TRANSWRAP (Reference 2) computer code.

Loads calculated at the PHTS inlet and outlet nozzle locations by REXCO-HEP were used as input to TRANSWRAP. The code is one-dimensional and uses a nonlinear method-of-characteristics formulation. The staff and its consultants concluded that because it was developed to treat only the hydrodynamic aspects of pulse propagation, fluid-structure interaction is not considered. This means that some of the structural motions, that would tend to attenuate peak pressures are neglected, and are calculated conservatively. The methods used by the applicants for generating loads on the reactor vessel, PHTS, overflow and Makeup System and Cover Gas System are generally conservative and will predict appropriate loads for evaluation of these components. All of the computer codes used in the analyses neglect several energy dissipative mechanisms and therefore predict loads that are high in absolute value. This is particularly true for the systems where TRANSWRAP and PIFITE are used because the impact conditions as defined by REXCO-HEP are conservatively high. One case where predictions may not be conservative occurs when the inclusion of dissipative mechanisms would change the frequency content of load histories to tune the loads with dynamic characteristics of the structure. Evaluation of this factor has to occur when the dynamic response of the subsystem is calculated. We do not believe the existing structures will likely fall into a category with similar frequencies because of its complex design and reinforcement from other structures.

The displacement requirements for the vessel are based on the REXCO-HEP predictions. Displacement requirements for head-mounted components were initially based on results from a model developed with the ANSYS computer code. The analytical model has been found to be overly stiff through independent analyses by the applicants and staff consultants. As a result the applicants now intend to define the requirements for head mounted components from test data to be obtained through new SRI tests (SM-9 and SM-10) to be performed by 1984. These tests are considered necessary for confirmation of the final design.

For the Overflow and Makeup System and the Reactor Cover Gas System the applicants include elbow force requirements in addition to the pressure and displacement requirements. These forces result from the acceleration of sodium through piping systems. These systems are not normally designed to resist these high inertial loads. The piping is allowed to distort in a new configuration without gross failure.

A.3.4 Evaluation of Reactor and Reactor Head Enclosure

Prior to the start of licensing review in October, 1981, the applicants used finite element analysis to demonstrate the capability of the head assembly to meet the SMBDB component margin requirements. Two basic models were used that were developed with the ANSYS finite element code.

The first modeled the three rotating plugs that make up the vessel head. Shielding plates were not included. The second model was of the local area around and including the margin shear ring between the large rotating plug (LRP) and the reactor vessel flange.

Several tests were used by the applicants to substantiate their analytical results. The first SM series consisted of five scale model tests ranging from a hydrostatic test of the head without shielding plates to dynamic tests with all major vessel and vessel internal components present. The hydrostatic test showed that failure of the vessel head is governed by disengagement of the intermediate rotating plug (IRP) from the LRP. This kinematic failure occurred in the test before there was any indication of incipient material failure.

The last two tests in the first series, SM-4 and SM-5, used scale models of the complete vessel, head, core barrel, UIS, horizontal baffle, and core support structure. The dynamic loads were provided by detonation of a low-density explosive that accelerated water above the core area toward the head in the form of a slug. The detonation was calibrated to provide the scaled equivalent of 100 MJ of work at time of slug impact. Kinetic energy in the water slug is the scaled equivalent of a 30 MJ slug in the full-scale vessel (30% of the energy at slug impact was in the form of slug kinetic energy). Results of the tests showed significant, but acceptable, plastic strain in the core barrel and vessel walls and no apparent plastic strain in the head.

Tests were performed earlier to verify the capability of the margin shear ring design. These included static, drop weight (impact), and dynamic tests of models of this portion of the structure. These tests and results of the static SM series of tests showed that the margin shear ring area can sustain the SMBDB loads without failing.

After careful review of both the head analyses and the SM series of tests the staff and consultants determined that we could not accept the applicants' conclusion that the reactor head enclosure was acceptable. First, analyses of overall head response could not be accepted until the analytical model was benchmarked with experimental data. In attempting to do this, the applicants found that the analytical model was "overstiff" and did not correctly predict the onset of material yield. Second, the method for attaching the shielding plates to the head in the test model caused it to be significantly overstiff in the dynamic experiments SM-4 and SM-5. Third, the loads delivered to the head in the SM-4 and SM-5 tests were well below 75 MJ slug impact to the head which is the design requirement.

To help resolve this issue, the applicants performed two additional hydrostatic tests during October, 1982 (SM-7 and SM-8). The first test, SM-7, used the same head configuration that was used on the SM-4 and SM-5 dynamic tests (overstiff head). The other, SM-8, was more representative of the prototypic head. Both test specimens failed in the same way as the static test of SM-1; with disengagement of the IRP from the LRP. The more prototypic head failed at a lower pressure (2010 psi) than the overstiff head (2600 psi) and absorbed significantly less energy at failure. Single Degree of Freedom (SDOF) analytical models were developed by the applicants, using stiffness data directly from the tests. The models were then analytically subjected to the loads experienced by the head during the SM-5 test. Results showed that if subjected to dynamic loads, the more prototypic head would be marginal. If submitted to the higher loads associated with the component margin requirements it would almost certainly fail by kinematic disengagement.

At this point the applicants agreed with the staff and consultants that the head design would have to be modified if it is going to meet the required loading conditions (75 MJ). To determine the most appropriate modification to the design the applicants performed a study of the primary failure mode (IRP disengagement). By drawing detailed layouts of the deformed head, as measured just prior to failure for SM-8, it was determined that disengagement was caused when the gap at the lower surface of the head between the IRP and LRP closed causing a prying action that moved the plugs relative to each other laterally. This lateral motion leads to plug disengagement. Modifications in the form of machining relief into portions of the plugs that interfere when the head deforms will eliminate the prying action and thus allow the head to deform more before failure. As it deforms it absorbs more energy. Currently the staff and its consultants believe that it can be modified to deform enough to absorb the loads introduced by a 75 MJ slug.

To evaluate where to machine the head and improve its capability, the applicants first extrapolated data from displacement histories for the SM-8 test. Layout drawings were again developed for the head in the displaced configuration. This displacement corresponds to the maximum load that the shear ring area can take prior to failure. The shear ring area is located between the LRP and vessel flange. Simple design calculations show that the failure mode will change from plug disengagement to material failure in the reactor vessel flange at the shear ring. However, failure will occur well beyond displacements required to absorb energy transferred to the head from a 75 MJ slug.

A.3.4.1 Slug Energy

Before further discussing the evaluation of the modified head, it may be helpful to discuss the physics of slug impact in the full-scale vessel, just prior to impact. The postulated CDA generated slug of sodium moves upward with a velocity of 1180 in/sec. This number assumes a slug kinetic energy of 75 MJ and a slug volume extending originally from the top of the sodium pool down to the horizontal baffle (see Figure 2 Attachment A.1-1). We can evaluate the maximum initial velocity of the head caused by the slug impact using the principle of conservation of momentum.

$$V_H = \frac{M_S V_S \ (1 + e)}{M_S + M_H} \qquad (1)$$

where

V_H = initial velocity of head,

V_S = velocity of slug impact,

M_H = mass of head,

M_S = mass of slug, and

e = coefficient of restitution (= 0 for inelastic collision).

Assuming that the collision is inelastic we calculate an initial velocity for the head of

$$V_H = \frac{953\ (1180)}{953 + 2694} = 308 \text{ in./sec.}$$

The kinetic energy that the head has to absorb becomes

$$KE_H = \tfrac{1}{2}\ (M_S + M_H)\ V_H2 = 19.5 \text{ MJ}$$

A check on whether the inelastic collision assumption is conservative can be obtained using test data from SM-2 where the scaled equivalent of a 75 MJ slug impacted the simulated head. By equating the impulse acting on the head (based on pressure measured during the test) to the change in momentum of the head we obtain an initial velocity less than that obtained above. This indicates that the momentum exchange calculation provides an upper bound on initial velocity and thus energy that the head has to absorb.

Note that only 26% of the original 75 MJ (about 20 MJ) of kinetic energy needs to be absorbed by the head. The remainder will go into strain energy in the core barrel, vessel wall, UIS, and UIS columns and into internal energy in the slug itself. Some experimental and analytical evidence of the energy distribution at and after slug impact is given in Table I.

To estimate displacement needed to absorb 20 MJ of energy we used the pressure-volume curve for the SM-8 hydrostatic test and extend the relation as a straight line (constant pressure) beyond the previous failure volume. Integrating under this curve we determined that the head must deflect enough to give a volume of 16.6 $in.^3$ (Reference 3). This is equivalent to a maximum deflection of 0.24 in. (4.7 in. for the fullscale head), which is well below the deflection that would cause disengagement with the modified head (approximately 8 in.).

Based on the applicants' study and the staff's consultants calculations performed at Los Alamos the staff believes that the vessel head, with the proposed modifications as discussed in Enclosure 1 of the applicants' letter dated February 14, 1983 titled Sodium Slug Energy Absorption Capability of a Modified CRBR Closure Head, will have the capability for withstanding an impact of a slug with 75 MJ of kinetic energy. This will be confirmed by the applicants through development of scale models and static and dynamic tests (SM-9 and SM-10). The staff requests that water released through the head on test SM-10 be measured.

TABLE I

ENERGY PARTITIONING

	SM-2	SM-5	REXCO (CRBR-3, Vol. 1)
At Slug Impact			
Slug	66%	30%	74%
Core Barrel Strain	3%	3%	17%
Vessel Wall Strain	27%	2%	5%
Fluid Internal Energy	3.5%	62%	1%
After Slug Impact			
Total Strain Energy*	47%	15%	60%

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* Does not include any strain energy for vessel head.

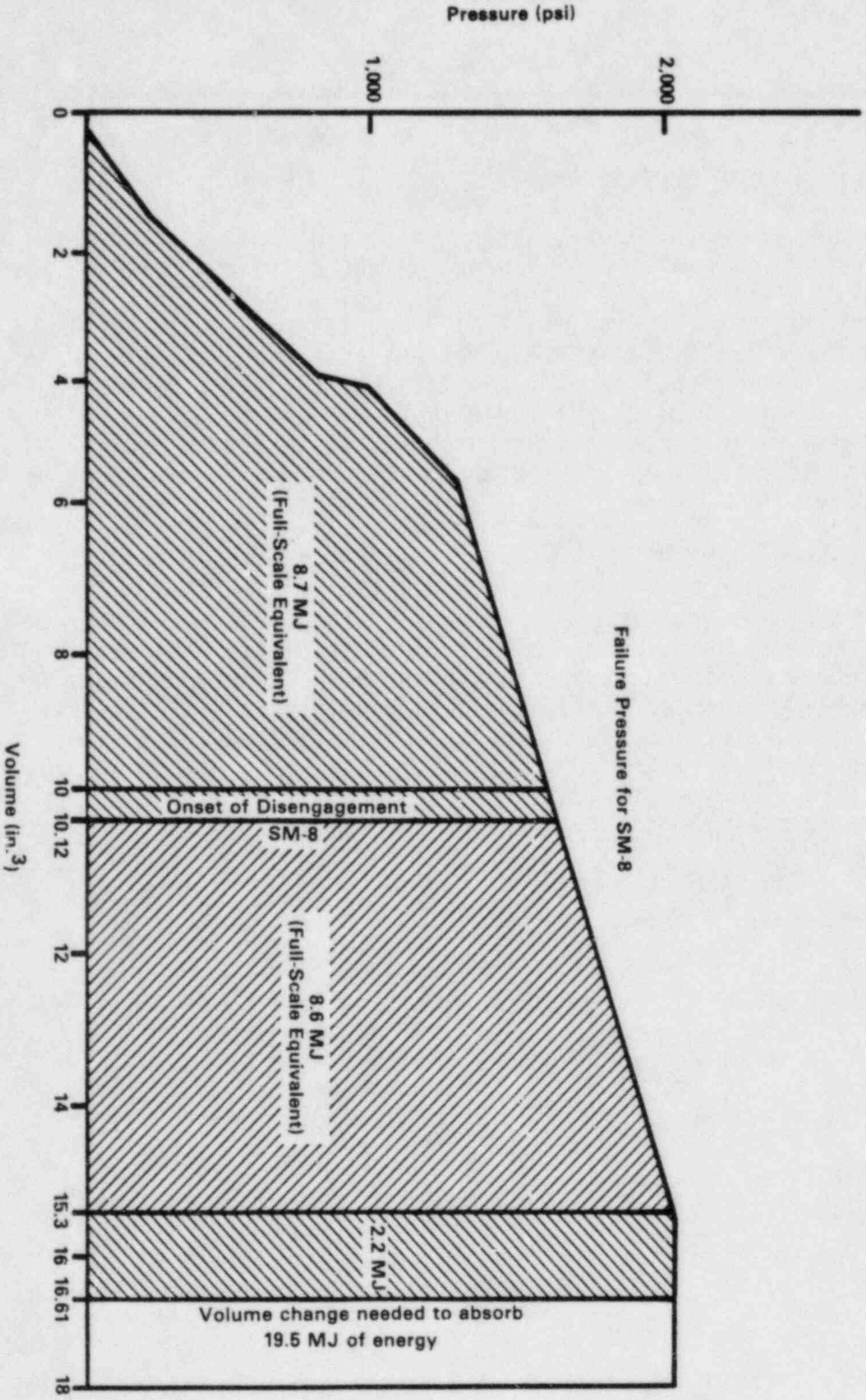


Figure A.1 Energy absorbed by head

Head Energy from SM-8 Pressure

A.3.4.2 Reactor Vessel Appurtenances

In-vessel equipment, including the horizontal baffle and the Fuel Transfer and Storage Assembly (FT&SA) were evaluated to determine whether they could become missiles that might violate containment. The analysis of the horizontal baffle was simplified (static, elastic) and showed positive margin compared to SMBDB stress criteria. The SRI tests confirmed this conclusion. The staff and its LANL consultants concur with these conclusions based on the model tests. Even if the vessel internal component did not meet the SMBDB stress criteria and broke loose during the accident they would most likely remain in the vessel by not penetrating the thick reactor vessel closure head.

Evaluation of head-mounted components was based on displacements and loads derived from a dynamic analytical model of the vessel head. The staff and its LANL consultants do not currently accept the results from the analytical model. As discussed earlier in this section the applicants are currently developing a program in which the reactor and its appurtenances will be based on data from scale model tests. These tests will be used to derive updated requirements for the head-mounted components. The staff sees no apparent reason why head-mounted components cannot be modified to meet the updated requirements as long as the strength of the head itself can be proven to be adequate by SRI tests SM-9 and SM-10.

A.3.4.3 Primary Coolant System

The staff and consultants reviewed the requirements for the primary coolant system and components. The components included the main coolant loops, the primary sodium pumps, the intermediate heat exchangers (IHX) and the reactor overflow system. The applicants have not completed their final analysis of components. The staff believes that these components will be acceptable or can be made to be acceptable.

The applicants used the pressure source derived by REXCO-HEP at the reactor inlet and outlet nozzles. These pressure-time histories were used as input source terms for the TRANSWRAP code analysis to transmit pressure pulses through the piping and equipment. The pressure-time histories for the piping, pumps and IHX range between 400 psig to 600 psig for milliseconds. Since REXCO-HEP and TRANSWRAP both conservatively over-predict the resulting pressures we believe they will be found to be acceptable in the final analyses.

A.3.5 Summary

The result of the staff and LANL consultants interaction with the applicants culminated in the applicants' letter dated February 14, 1983 from which we interpret the following commitments and develop the indicated staff positions below. Attachment A.1-I provides a discussion of the reactor and gives figures showing the reactor enclosure system parts and their arrangement.

1. The applicants' commitment to design the reactor closure head to accommodate 75 MJ of upward kinetic energy at sodium slug impact will be met. This design requirement is stated in CRBRP-3, Vol. 1, Revision 4, Table 5-2.a.

2. The applicants have committed to appropriately modify the closure head and shielding plates based on the concept in enclosure 1 of Reference 1, to prevent leverage and premature plug disengagement and thereby increase the capability of the head plug system to accommodate 75 MJ slug impact without causing margin ring disengagement.

3. In enclosure 2 of Reference 1, the applicants agree to a confirmatory test program consisting of a static test (SM-9) and a dynamic test (SM-10). Calibration testing will preceed test SM-10, to ensure that the equivalent of a 75 MJ will impact the head in SM-10. The staff proposes that the applicants consider either of the following options:

    a) Obtain a simple validated analytical model of the reactor closure head system consistent with appropriate dynamic test data including SM-10. At present the applicants have no validated simple or complex 3D model benchmarked against test data. We strongly urge the applicants to have even a simple model that has been validated by test data even if option b is selected.

    b) A second backup test model for SM-10 should be prepared as a contingency against loss of any essential data from the prototypic dynamic test (SM-10). We request that the water released through the head, during the test, be measured.

4. Enclosure 3 of Reference 1 covers large releases of sodium through the closure head. The staff will defer addressing this subject until after the testing has been completed in 1984. The introduction of spray deflectors or confinement of sodium spray is, in our opinion, a deferable item.

5. The staff recommends that CRBRP-3, Volume 1 be brought up to date concurrently with the completion of SM-10 testing. The staff recommends that for completeness the applicants should include a section on evaluation of the load path from the vessel flange through the reactor cavity wall.

The staff concludes, on the basis of the foregoing information and commitments by the applicants with respect to dynamic slug impact, supplemental scale modeling tests SM-9 and SM-10, underside plug (LRP, IRP and SRP) modification, increased energy absorption capability of the reactor closure head, and use of dynamic scale model inputs for direct evaluation of the reactor enclosure system, that the intent of 10 CFR 50.34 Part 3 iii relating to "providing reasonable assurance that the final design will conform to the design basis with adequate margin for safety" has been met for Structural Margins Beyond the Design Basis (SMBDB).

A.3.6 References

1. Letter, HQ:S:83:222, John Longenecker to J. Nelson Grace, Reactor Closure Head Capability to Meet Margin Requirements, dated February 14, 1983.

2. "Hypothetical Core Disruptive Accident Considerations in CRBRP, Vol. I: Energetics and Structural Margins Beyond the Design Base," CRBRPO report CRBRP-3, Vol. 1, Rev. 4 (March 1982).

3. Y. W. Chang and J. Gvildys, "REXCO-HEP: A Two Dimensional Computer Code for Calculating the Primary System Response in Fast Reactors," ANL-75-19, (June 1975).

4. M. J. Hillier, "Tensile Plastic Instability of Thin Tubes, Parts I and II," Int. J. Mech. Sci. 7, pp 531-549 (1965).

5. Letter, HQ:S:82:174, John Longenecker to Paul S. Check, Responses From December 9, 1982 Meeting, dated January 5, 1983.

6. "CRBR Structural Response of CRBRP Scale Models to a Simulated Hypothetical Core Disruptive Accident," WARD-D-218 (October 1978).

7. Letter, HQ:S:83:160, John Longenecker to Paul S. Check, Structural Margin Beyond Design Base Loads on the Reactor Vessel Support Ledge, dated December 21, 1982.

A.4 ANALYSIS OF THE THERMAL MARGINS BEYOND THE DESIGN BASIS (TMBDB)

As explained in the preceding sections the most likely consequence of the various CDA progressions considered in this review is the path that leads ultimately to melt-through of the reactor vessel. In this stage of the accident the course of events is largely driven by the thermal activity of the core debris and by thermal effects associated with chemical reactions inside and outside the reactor cavity. The features associated with the confinement and control of these effects are reviewed in this section.

The staff review has included several Technical Evaluation Reports (TERs) on the TMBDB capabilities of CRBR. These are reproduced as attachments to this section. The first is the BNL reveiw of the TMBDB sequence, containing the quantitative evaluation of the CACECO code and analyses. The second is a report from LANL evaluating the effects of the TMBDB events on the vital structures and cleanup equipment. The third is a review of sodium-concrete interaction data prepared by Applied Science Associates. The applicants have provided their assessment of the TMBDB scenario (Reference 1). The staff's review will address the applicants' proposed scenario point by point.

The staff's final evaluation of these items has been influenced by these reviews and also by supplementary material presented by the applicants in a series of submittals and meetings throughout 1982 (References 2 and 3).

The staff has considered, for the CRBR application, a set of specific BDBA design criteria based on similar criteria from NUREG-0850, "Preliminary Assessment of Core Melt Accidents at Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects." While these criteria are not considered the final set for LMFBR-BDBA application, the staff believes use of these criteria in the interim as general guidance provides a useful basis for evaluating the CRBR design for BDBA challenges. The proposed criteria are listed below:

(1) Group C quality standards (ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Class 3) shall be applied to the mechanical and fluid accommodation systems or their component parts.

(2) The accommodation system(s) shall be designed to remain functional, when potentially required, for the design basis environmental conditions, including the loads imposed by a safe shutdown basis earthquake (SSE). This criterion is aimed at specific accident sequences, which at another level of probability, may lead to BDBA conditions.

(3) The accommodation system(s) shall be evaluated for design-basis loads, including the safe shutdown earthquake (SSE), in order to ensure that there will be no system interactions which might affect or impair the functioning of safety-related components and systems relied upon for design-basis events.

(4) The accommodation system(s) shall be capable of performing their functions in the most realistic limiting environment estimated to occur as a result of the accidents they are designed to mitigate.

(5) The interfaces between the accommodation system(s) and nuclear safety-related components and systems shall be designed to the same safety grade (or quality group) and seismic loading requirements as the safety-related components or systems.

(6) The accommodation system(s) shall be of reliable design and construction but need not meet the single failure criterion unless the component in question is rendered totally inaccessible by the event to be accommodated; then design reliability enhancement such as redundancy, shall be used.

(7) The accommodation system(s) shall be capable of functioning independently of all offsite power systems for a period of time expected to be required for restoring AC power.

(8) The accommodation system(s) shall be capable of manual operation and control, with appropriate monitoring capability, from the control room.

(9) The incorporation of the accommodation system(s) shall not compromise design basis safety requirements.

(10) The accommodation system(s) shall be designed to allow periodic inspection and testing to verify the functional performance.

Since we are dealing with events much less likely than the DBA the considerations associated with the single failure criterion are not to be used for TMBDB.

In addition to these specific criteria, several general review guidelines have been developed to assess the adequacy of certain features, systems and actions necessary to provide adequate design margin to accommodate the consequences of a CDA to assure protection of the public. These guidelines are discussed in Section A.1.3.

A.4.1 Events Leading to the Significant Ex-Vessel Phenomena

In the early part of the accident (before vessel melt-through), the core material is retained in the primary system except for noble gases and small amounts of other material that are presumed to escape as a result of phenomena associated with the initial energetics. Some fuel may be dispersed radially and among the upper plenum and upper internal structures (Reference 1, pp. 3-2 through 3-9). Nevertheless, sufficient core debris is expected to accumulate in the bottom of the reactor vessel so that the reactor vessel becomes overheated in a localized area and fails. Failure occurs as the melt front between the fuel and steel interface heats the steel until it can no longer support the weight of fuel and structures above it. The expected nature of the failure is a rapid creep rupture. Failure of the guard vessel is postulated to occur in a similar fashion very soon thereafter (Reference 1, pp. 1-2 and 3-15 through 3-17).

Although the assumption of failure of these two vessels may appear pessimistic, the staff believes that the present state of knowledge of in-vessel coolability allows no other choice for severely degraded core geometry in the CRBR. Once failure of the primary system and guard vessel occurs, the entire inventory of primary system sodium will then rapidly flush the core debris down into the reactor cavity. It is believed that this flow of sodium would be great enough to carry with it some of the debris from the upper plenum and internal structures. The assumption is made that all the core debris is so transported, which is conservative with respect to subsequent heat loads in the reactor cavity. There are differing opinions about the degree to which this debris would form a self leveling bed on the reactor cavity floor. The staff believes considerable leveling would occur due to the high tuburlence. The applicants have presented examples of different distributions of debris in the reactor cavity (References 1, pp. 3-19, 3-70 and Appendix G.1) and expect that they will be coolable and subcritical as a result of self leveling phenomena. The staff is not convinced that either of these characteristics is guaranteed by the reactor or reactor cavity design. We agree with the applicants, however, that if criticality occurred in this configuration there is no reason to expect it to be energetic, and mild pertubations would be beneficial to self leveling. We expect that the cavity floor liner would fail as a result of the initial impact of the high temperature core debris. The applicants have included this event in their accident evaluation (Reference 1, p. 1-2).

Following failure of the reactor cavity liner the insulating concrete below the cavity liner is assumed to be penetrated rapidly, as evidenced in out-of-pile experiments, and the sodium-fuel reaction zone would then encounter the concrete structure of the lower part of the reactor cavity (Reference 1, pp. 1-2 and 3-20).

From this point onward in the accident there remains some uncertainty in the detailed course of events and in the challenges to the containment system that result. However, the overall direction is defined. The balance of this evaluation will summarize the different viewpoints concerning these events and will outline our views concerning the spectrum of consequences that may be reasonably expected to result from such events.

A.4.2 Sodium-Concrete Interactions

Following an initial loss of coolable geometry the primary system sodium is expected to be heated above the normal average sodium temperature of about 900°F by the decay heat from core debris. At the time of primary vessel penetration, the sodium drops to the cavity below at about 1000°F.

The interaction of the hot primary sodium with the structural concrete of the reactor cavity (RC) plays an important role in the sequence of events following loss of primary system integrity. The most significant considerations involve the extent of the penetration of the concrete and the associated production of aerosols and hydrogen, which could be transmitted to containment through the reactor cavity vents.

Vent paths to the containment building atmosphere are opened through rupture disks when the cavity pressure reaches about 11.5 psig. The quantity of aerosols generated and the location where they are formed has bearing on whether or not plugging of the vents from the reactor cavity is possible. Plugging of the cavity vent path is important because it would modify the heat sinks available to reduce the heat load delivered to the containment building environment.

There are differing views on the origin of aerosols in containment. HEDL has stated the view that sodium oxides, formed by air oxidation after sodium vapor enters the containment building, are the principal source of aerosols. Sandia has noted the production of aerosols even in inert atmospheres, and consequently a concern was expressed that these would be generated in the reactor cavity. The applicants maintain that any aerosols generated below the sodium pool surface, from the sodium-concrete reaction, would be retained within the pool. After review of their analysis of this matter (Reference 4 and 5), the staff agrees with this conclusion.

Aerosol production and releases, specifically the amount and size distribution in the containment building, is a key factor in the subsequent distribution of radioactivity (Reference 6). The aerosol calculations must, therefore, be done over the entire range of possible sodium-concrete reaction rates. The staff recognizes that the applicants have calculated the aerosol behavior for the cases representing various penetration rates using the HAA-3B code. The staff has compared typical HAA-3B results used by the applicants with corresponding cases using the NRC sponsored HAARM-3 code and has reached the conclusion that the applicants' aerosol calculations are conservative with respect to the predicted concentration of suspended materials.

Hydrogen is produced from the reaction of sodium with the water released during the sodium-concrete interaction (Reference 1, pp. 3-21 and 3-24). The hydrogen produced can present a combustion threat if sufficient oxygen is present. As will be seen later this becomes a significant factor in the consideration of the need for containment venting. Thus the sodium-concrete interaction rate is important for this reason as well.

The mechanisms of the thermal-chemical attack of sodium on concrete are quite complex. Significant experimental efforts over the past decade have failed to develop simple models that explain the diverse data results. The early experiments concentrated on effects that occurred within the first few hours of interaction. Size, geometry, stress conditions, amount of reinforcing, concrete type, moisture, depth of sodium, the limitations of reactants, temperature and temperature progression all seem to have influenced some of the results. As longer term test results became available, some pattern has emerged from the longer term interaction results, and it now appears possible to place reasonable upper bounds on the long term penetration rates. Although a theoretical model unifying all the data has not been developed, the staff believes that penetration rates selected by the applicants are adequately conservative.

The experimental information on penetration of sodium into concrete includes a variety of sizes, temperatures and geometry. Experiments in the size range of one to three feet in diameter did not reveal any significant scaling effects. However, since the larger of these is still smaller by a factor of ten than the linear dimensions of the reactor cavity, concern has been expressed that turbulence in the large scale could invalidate the results of the smaller scale experiments. The interpretation of the slowing down of the sodium-concrete reaction with time is based on the buildup of a protective layer of reaction products. It has been argued that, if this layer were dislodged by turbulent agitation, the reaction rate might not slow down appreciably. There are, however, experiments in which the sodium attacks the concrete on veritcal surfaces which do not collect a protective layer. These are used to establish a maximum long term penetration rate. This maximum turns out to be significantly less than one inch per hour. (About 0.2 in/hr would bound the long term effects of all the sodium-concrete reaction data.)

In view of the experimental data base, the applicants have more recently proposed a rate of 7 inches per hour for the first 20 minutes and 1 inch per hour from then to sodium boil-dry as a reasonable upper bound case, which still provides a substantial margin for additional effects due to the presence of core material, should they occur.

Since the sodium-concrete reaction rate is a key factor in the rate of production of aerosols and hydrogen, a bounding interpretation of this rate must be adopted in the calculations. For realistic upper bound calculations, the staff has used a sodium-concrete reaction rate of 7 inches per hour for the first 20 minutes and 1 inch per hour thereafter as in the applicants' analysis.

As far as the ultimate depth of penetrations is concerned, it is presumed that some limit exists based on the large quantity of reaction products ultimately produced. The staff has used a 30 inch maximum penetration depth of reactor cavity sodium for estimates of aerosol and hydrogen formation. This is consistent with preliminary calculations and modeling of the sodium-concrete reactions that are being developed at Sandia and also encompasses all the experimental data.

It is also desirable to analyze accidents based on parametric variations of the reaction rates and depths that have been observed experimentally. The two basic parameters to be varied are the rate of reaction and the depth of penetration. The applicants have previously presented fairly thorough analyses of a base case accident based on low reaction rates and modest penetration depths (one half inch per hour for four hours). More recently, a higher reaction rate case (7 inch per hour for three hours; then 1 inch per hour until boild dry) has been presented to the staff as a margin assessment case consistent with emergency planning constraints. Thus the applicants' parametric calculations cover a range from the relatively mild sodium-concrete reactions of the base case to a margin assessment case considerably beyond the realistic upper bound. The staff considers this range to be acceptable for the interpretation of the effects of sodium-concrete reactions on subsequent containment challenges.

The applicants have furnished the details of timing of aerosol formation that they have used in the margin assessment case for defining subsequent radiological dispersions. These have been used for further confirmation that the applicants' suspended aerosol concentrations are conservative.

Because some sodium will be boiled off, not all is available to react chemically with the concrete in the reactor cavity. The vaporized sodium partially refluxes back into the cavity and refluxing furnishes a mechanism for involving larger quantities of concrete in the containment as a heat sink. Some attention has been given to the possibility of plugging of cell drain lines by sodium if it should freeze. Since the sodium in the cells is about 1400°F above the freezing point this is not considered likely at this time. The staff believes that it is feasible to design these return flow channels to prevent sodium freezing, but will require confirmatory OL analysis to verify that this has been accomplished.

A.4.3 Further Considerations Regarding Concrete Penetration and Temperature Transients

Reasonable values have been established for the upper limit reaction penetration rate and the maximum depth of the sodium-concrete interaction. These are the realistic upper bound release rates used by the applicants and the 30 inch depth estimated by Sandia. Patterns of temperature and pressure in the containment and structures have been developed by the applicants consistent with these sodium-concrete reaction rates. To do this, the applicants have used models of heat distribution between the sodium pool, with and without core debris, and the concrete based on the following considerations.

Fully oxidized core debris cannot react chemically with concrete, but can provide a sustained heat source for thermal degradation. If sodium is present, the core debris can provide a heat source to maintain the sodium-concrete reaction zone at a high temperature. If sodium is not present, or is not able to receive the heat from the core debris, the debris can attack the concrete below the reactor thermally and directly. In the event that molten metals from the cladding or vessel are present with the debris, chemical reactions between molten metals and the concrete or released water vapor or CO_2 are expected. Thus the penetration rate of core debris into concrete may be expected to depend significantly on the presence or absence of an effective cooling pool of sodium. Superposed on these situations are the problems of generation of combustible gases, which require the presence of unoxidized metals, principally sodium or steel.

Consider first the case of core debris with sodium present. As a heat source, a coolable debris bed is expected to transfer its heat to the sodium so that sodium will continue to react with concrete at the rates within the realistic upper bound scenario.

The applicants also have discussed with the staff their views on the effect of uncooled core debris. Applicants have used the TRUMP heat transfer code (Reference 1, Appendix C.2 and Reference 3) to perform bounding analyses of the effect of a core debris bed that transfers the major portion of its heat in the downward direction. These analyses are applicable to debris beds which are insulated from the sodium by a barrier of reaction products, and to debris beds remaining after sodium has boiled-dry. Under reasonable assumptions of thermal properties, TRUMP analysis of the debris bed results in a penetration depth of about 60 inches in a boil-dry period of 71 hours (less than 1 inch per hour). Subsequent melt penetration was estimated to reach a depth of 15 to 25 feet into the 26-foot basemat over a period of months. These variations in the concrete penetration scenario produce loadings on the containment and the clean-up system that are within those assumed for the base case of sodium-concrete penetration. Note that the base case is limiting here since it provides the greatest time for heat soaking of massive structures. As seen from the above results, core debris/concrete reactions may or may not stop short of effectively penetrating of the basemat (26 ft.). If the core debris should penetrate the basemat the effects of transport of radionuclides in the groundwater liquid pathway are not expected to be as damaging for the CRBR as they would be for large light water reactors. This is explained in more detail in the FESS (NUREG-0139, Supplement 1, Appendix J) as follows:

> Surface water hydrologic properties at CRBRP should be similar to those used for the Liquid Pathways Generic Study (LPGS) small river site, which was based on the Clinch-Tennessee-Ohio-Mississippi Rivers system; although the river uses and populations in the LPGS (NUREG-0440) were based upon national averages and have not been directly compared to the CRBRP. The groundwater characteristics at Clinch River do not indicate any unusually adverse transport characteristics.
>
> Additionally, the CRBRP is a considerably smaller plant than the LPGS case (CRBRP is 1121 MWt vs. 3425 MWt assumed for the LPGS), and contrary to the LWR characteristics, CRBRP does not contain any large storage of water that could serve as a potential "prompt source" to the environmental liquid pathways. Therefore, only the radioactive material leached from the core debris by the local groundwater is likely to be transported to the Clinch River. This source was found in the LPGS to be considerably smaller than the "prompt source". Therefore, based on the preliminary appraisal of the liquid pathways, the staff concludes that the liquid pathways impacts of CRBRP would be probably smaller than those for the LWRs analyzed in the LPGS small river site case.

A.4.4 CACECO Modeling of Containment Transients

The CACECO computer code (Reference 7) is the principal tool used to track the progress of temperatures and pressures in the various compartments of the Reactor Containment Building (RCB). It accounts for the principle heat sources and energy and mass transfer in containment during the post melt-through period. The heat sources include decay heat, chemical reaction heat, sodium-24 decay heat and stored heat. Initially these are distributed in the reactor cavity and pipeways. Eventually the volatile reaction products are vented through pipeway cell number 2 (through one of two redundant rupture discs) to the RCB above the operating floor. Each compartment in the CACECO model can have an atmosphere, a pool of condensed material, heat sinks and sources. Materials and energy are distributed among the compartments in a series of time steps that follow the accident progression.

A number of approximations have been introduced into the CACECO code: heat transfer by radiation is generally disregarded, pool temperature is equilibrated to atmosphere temperature at each time step, the heat capacity of aerosols is incompletely treated. In addition, the backflow between cells is restricted in some options and emphasized in others, heat-transfer coefficients through liners are simplified, as are the heat transfer processes through aerosol-coated walls, and condensation heat-transfer models inadequately include the effects on non-condensible gases. It is not clear that the properties of the aerosol as a heat sink are correctly accounted for in the procedure, especially in view of the varying amount of aerosol in the containment. Thus there are a number of areas where the anlysis is quite simplified and the effect in these areas may not be totally accurate. However, we believe CACECO is adequate for making reasonable approximations to the progression of the accident on the time scale of interest.

A.4.5 The Role of the Annulus Cooling System

The applicants propose an annulus cooling system to remove containment heat and to limit the temperature of the confinement building under TMBDB conditions. Principal mechanical components of the system would be located outside the steel containment shell. The steel containment shell would be cooled by circulating outside air between the shell and the concrete confinement building and then exhausting it to the atmosphere. The capacity of the annulus cooling system is sufficient to withdraw the necessary heat loads without exceeding TMBDB allowable shell temperatures. In response to questions about the effect of accumulated aerosols on the surface heat transfer, the applicants have presented a preliminary estimate that the containment atmosphere temperature would be 80^oF higher with an aerosol layer. Shell temperature would be 30^oF lower at the particular time chosen for the comparison (36 hours).

These preliminary calculations indicate feasibility of the system. It has been suggested in Attachment 1 to this appendix, the BNL review of CACECO code, that the accumulation of aerosols on the interior of the shell would interfere with the heat transfer capability of the annulus cooling system. This accumulation late in the accident (i.e., beyond 50 hours, in the base case) might lead to temperatures in the containment atmosphere considerably higher than those calculated by the applicants. As discussed in Section A.4.11, the staff does not believe that this effect would be more significant than the applicants have estimated during the early part of the accident, and hence there is no reason to believe that the vent time would be accelerated. For the OL analysis however, it will be necessary to show that the necessary TMBDB equipment can survive whatever containment temperatures are developed.

A.4.6 Containment Structure Analysis

The containment building and its internal structures have to remain intact for certain specific times during the TMBDB scenario if the scenario, as analyzed, is to remain valid. Containment calculations are carried out to 8000 hours to assure the permanence of the containment system following an accident. Various internal structures, including cell liners, must maintain their integrity for different amounts of time during the scenario up to sodium boil-dry. Significant loads acting on the steel containment consist of pressures and temperatures beyond the design basis. If the annulus cooling system functions as required, loads after sodium boil-dry, with the exception of thermal loads in the walls below the operating floor, are enveloped by those preceding boil-dry. In the walls below the operating floor, high temperatures over a long period of time could cause reinforced structural concrete to creep resulting in enough deformation to overload the steel containment boundary. Because of the low stresses experienced by the concrete, the potential for this to be a problem is low and it is not considered an open issue. The highest loads in the steel containment shell occur at its intersection with the operating floor. Specific design features in this area (annulus cooling system and internal insulation) would keep the steel temperatures well below those which could cause material yield or local buckling. Large deformations (several inches) in the steel containment shell from thermal expansion are being factored into the design of penetrations and any equipment attached to the shell (Butler et. al., LANL TER Section L, Attachment 2).

Because the reactor cavity (RC) vent exit is not centered relative to the containment shell, nonsymmetrical heating of the shell can be expected from the sodium flame at the vent exit. Asymmetric effects are minimized because radiation heat transfer is small due to the aerosols in the atmosphere of the upper containment during the burning process. Structural loads from nonsymmetric convective effects have been evaluated by staff consultants and the applicants and are small.

The applicants have asserted that vital containment internal structures will not rapidly deteriorate to the point of collapse prior to sodium boil-dry. This is particularly important for the RC and pipeway cell structures because earlier collapse would aggravate sodium-concrete and sodium-water reactions.

Large deflections of structures in these areas could cause failure of cell liners, pieces of unreacted concrete could be injected into the sodium pool and, if failure were severe enough, sodium could flow freely into Cell 105 (Reference 1, p. 1-2, and Figure 3-5) where it would react with concrete and water. It is also important that the reactor cavity (RC) ledge not fail early causing an unrestricted flow path for sodium vapor to enter upper containment. Based on a review of the applicants' analysis the staff accepts that failure would not occur before 50 hours. The most important structures are addressed in the following paragraphs.

During the TMBDB scenario the RC wall and pipeway cell structures are loaded from dead weight, internal pressure and thermal gradients. The largest stress components come from the thermal gradients and are secondary in nature. Therefore, as localized material failure occurs in the form of concrete cracking and crushing, loads are relieved. The applicants used the ANSYS finite element computer code to perform a detailed analysis of all critical areas in these structures. Results show that the floors of the pipeway cells and the base of the RC wall are the critical areas. For high concrete penetration rates the applicants have suggested needed design enhancements in the RC floor and lower wall region to prevent failure. (Meeting of September 15, 1982, presentation by G. N. Freskakis, NRC Accession No. 8212010161.) With these changes the staff concludes that the structure will be adequate.

The staff was concerned that, late in the accident, the failure of interior structures might impose strains on piping that penetrates containment and thus might jeopardize containment seals. It is now believed that this is unlikely to be a problem, because at this stage the purge operation is in effect and the reactor is kept at a slight underpressure. Therefore, the failed seals could allow inflow only.

Cell Liners

Cell liners are evaluated as an engineered safety feature for design basis accidents in Section 6 of the PSAR.

In the TMBDB context, the cell liners play an important role in the preservation of the isolation of cells from each other. It is particularly important that sodium in the reactor cavity not be released to the air filled Cell 105 (Reference 1, p. 1-2 and Figure 3-5). Cell 105 is also used as the outlet for the water released from the concrete behind the liners, and therefore release of sodium to this area could be accompanied by unacceptable energy production. The reactor cavity floor liner is expected to fail early in the core melt sequence, but the configuration of the liners prevents lateral attack on the walls below the floor level. If the reactor cavity wall liners should fail (i.e., become penetrated), sodium attack on the walls above the floor level is possible. The applicants have estimated that if the reactor cavity wall liners remain intact for 50 hours, any subsequent RC wall liner failure would not lead to a release of sodium to Cell 105 prior to the time estimated for sodium boil-dry (Reference 1, p. 3-87). It is therefore a design objective that the reactor cavity wall liners remain intact for at least 50 hours.

The TMBDB accident consequences do not seem to be so sensitive to the early failure of other cell liners. For example, the applicants have calculated the base case scenario with failure of the pipeway cell floor liner at 30 hours, and have calculated the margin assessment case with failure at 0 hours, with no apparent impact on the release of sodium to Cell 105.

The liners are expected to be satisfactory for TMBDB strains if they first meet criteria for design basis strains. This is because of the higher ductility at the TMBDB temperatures.

A somewhat simplified interpretation of the cell liner expansion considers the liners as basically under compressive loads by reason of their being subjected to high temperatures in a confined space. The yielding that accompanies such compressive loads at elevated temperature would normally be expected to relieve the stresses without failure in this ductile material.

However, it has been considered prudent to examine in more detail the stress concentrations around penetrations, corners and stud anchors. Both experiments and analyses have been performed. Although no liner failures occurred in the experimental mockup, the results are somewhat inconclusive, since failure of the studs where they were welded to the liners relieved much of the stress.

The applicants' original analyses of the cell liner strains were audited by the NRC staff and consultants and were found to be inconclusive in several important details of non-symmetrical buckling and details of stress concentrations.

The applicants therefore proposed a program of five analytical cases examining the strain patterns in more detail using finite element analysis. The first two analyses covered the typical liner-anchor patterns and the details of a typical plate section. These are relevant to the reactor cavity wall liners, which have no penetrations near or below the sodium level. The other three analyses cover details of stress concentrations near penetrations and a composite of all effects.

From the strains computed in these supplementary analyses, it will be possible to estimate whether the liners satisfy ASME III, level D requirements, and appropriate design criteria can then be adopted.

It is the staff view that if level D strain criteria are indicated by the finite element analyses, then a supplemental experimental program might be required to corroborate and support the analyses. If the calculated strains are substantially below the level D limits this may not be necessary. The applicants have indicated agreement with this general view and will propose an appropriate experimental program, upon completion of the supplementary analyses, if necessary.

At this time, the supplementary analyses relevant to the reactor cavity wall liners have been submitted by the applicants. (Letter, HQ:S:83:211 Longenecker to Grace, 2/4/83, Acc. # 8302090326.) The strains appear to allow sufficient margin so that experimental confirmation will not be necessary for acceptance of the RC wall liner concept. These are the most critical liners for the TMBDB scenario. Some further details around the studs are still required, however.

The second most important liners are those of the pipeway cells. Here however, the problem of penetration of the floor liner has been circumvented by the placement of two feet of sacrificial concrete below the liner. This is more than enough to react with any sodium collecting in the pipeway cells and penetrating through the floor liner. Drains are provided in the pipeway cell floor so that the wall liners will not be attacked. Thus the concept of the pipeway cell liners also can meet the TMBDB design requirements.

The staff believes that the overall compressive nature of the loads and the material ductility, and the lack of liner failure in the experiments, are supporting but not conclusive evidence of the acceptability of the liners. With the additional evidence that strains are not excessive in the new finite element analyses, the staff believes that further evidence has been developed for the acceptability in the TMBDB context of the reactor cavity wall and pipeway cell liner design. The applicants have submitted a program for additional confirmatory analyses and experimental results before construction of the cell liners (A.4.10).

A.4.7 Considerations Regarding Hydrogen Production

A serious concern for any sodium cooled system is the potential for the production of hydrogen from the reaction of water and sodium. Although all unnecessary water services are excluded from proximity to sodium systems, the concrete is a substantial source of water if it should come in contact with sodium. After vessel melt-through the staff expects sodium and concrete to be in contact on a large scale. Thus substantial amounts of hydrogen will be produced. Since the reactor cavity is inerted the hydrogen does not pose a flammability hazard until it is expelled to the containment building above the operating floor. The applicants calculate that the hydrogen content of the containment atmosphere will rise to 4.4% before sufficient sodium vapor enters the containment to ensure burning. Two elements that enter into this evaluation are the holdup of a certain amount of hydrogen in the sodium pool and the delay time before sodium vapor is evolved at 6 gm/m^3 to the containment atmosphere. Neither of these factors is known with much accuracy. Elimination of the hydrogen holdup would raise the concentration from 4.4% to 5.2% at the time of ignition. Furthermore, the time for evolution of sodium vapor is strongly influenced by the recondensation of sodium in its vent path. If recondensation is more effective than expected, ignition of hydrogen could be delayed until additional quantities accumulate. It is recognized, however, that other ignition sources may be available or can be provided if necessary. Also, with regard to structural consequences, any hydrogen burn produces pressures that act quasi-statically on the structural system. Preliminary calculations have been sufficiently conservative to bracket these effects. In experiments, carbon monoxide has not been observed in the presence of sodium, but may be expected to add to the flammable content of the atmosphere after sodium boil-dry.

A.4.8 Control of Overpressure Transients (Venting and Purging)

The applicants' protection against containment overpressure depends on the ability to anticipate sudden large hydrogen burns and other sources of increased pressure and to dilute the containment atmosphere to minimize the hydrogen hazard as necessary. They assert that hydrogen concentrations can be detected in time to reduce the containment pressure to atmospheric without ever exceeding 6%, and then to dilute the atmosphere by purging before a serious hydrogen threat exists.

The applicants propose to commence the venting operation when the pressure in the containment building approaches a substantial fraction of the failure pressure, or when the hydrogen content begins to increase without ignition or in an oxygen depleted atmosphere. The staff believes that it is feasible to develop criteria of this type that will ensure the protection of the containment building from excessive hydrogen burns, within the guidelines discussed in Section A.1.3.

The time required to vent the pressure down to one atmosphere depends on the initial pressure and may be of the order of three hours. Purging is initiated after venting. The purging operation provides further air at a controlled rate to burn additional sodium and hydrogen as they are produced at a pressure at or slightly below atmospheric. The pressures, temperatures and hydrogen concentration are claimed to be manageable from then on by control of the purging operations from the control room.

This type of system appears capable of providing protection against massive uncontrolled containment failures during the TMBDB phase of the accident. The applicants' calculations have been based on their base case with modifications to take care of the higher rates of sodium-concrete interactions. The modifications have encompassed an acceptable range of sodium/concrete reaction rates to assure that margin exists for operation of the mitigation features under the anticipated extreme conditions of the accident.

The staff believes that the key to a successful vent-purge operation lies in providing the operating management with sufficient training and reliable and understandable information in the control room so that correct decisions can be made. This will require a high degree of reliability of the instrumentation and information from an extremely hostile environment. The processing of hydrogen samples is accomplished in the intermediate bay of the steam generator building (Reference 1, pp. 2-32). Entry to the intermediate bay is not required to perform the atmospheric analysis. However, entry is feasible during the early or late stages of the accident when a backup measurement at the sampling station might be desirable.

The staff believes that monitoring for oxygen and other flammable gases (e.g., carbon monoxide) in the containment atmosphere may also need to be provided. The applicants are studying this possibility and will reach a recommendation prior to the FSAR. Assurance that sampling lines will not become plugged by aerosols must be provided. The processed hydrogen, carbon monoxide, and oxygen information is then transferred to the control room, from which overall management of the vent-purge operation is accomplished. There are no questions of feasibility in the provision of additional gas analyses instrumentation, if necessary. Indicators of the overall pressure and temperature in the RCB are available for further guidance in controlling the vent/purge operations.

It is essential that the training of operators and supervisors include an appropriate level of familiarity with the TMBDB phase. The applicants' Preliminary Safety Analysis Report includes the commitment to fulfill the training requirements of the TMI Action Plan Task I.A.2 which satisfies this need.

Although the above scheme is conceptually feasible a more detailed description of the design of the system, specific venting specifications and the associated testing requirements will be required for the OL review.

A.4.9 Effectiveness of the Containment Cleanup System

A reliable and effective cleanup system is required so that vented and purged gases will not produce excessive off-site doses. The applicants have proposed a system similar in principle to that which has been built and tested at FFTF. Although the original system was designed to accommodate the applicants' base case scenario (with some margin), they state that it can accommodate the high sodium concrete interaction rate (margin assessment case) and accompanying pressure, temperature and aerosol loadings.

The earlier sodium boil-dry time accompanying this scenario must also be accommodated. The feasibility of the principles of such a system are established by the FFTF experience and the analysis and testing directly related to the CRBR system as presented to the staff.

There are three items that have recently been clarified by the applicants.

1. Concern has been expressed regarding a possible pressure spike from hydrogen burning in the cleanup system during operation. However, our present interpretation of the functioning of this system indicates that this will not be a problem.

   Typical atmospheric composition in containment (Reference 1, Figures 3-11 and 3-12) when the cleanup system is first brought into use (during the venting phase) is:

   O_2 8.4 - 3.5%

   H_2 0.0 - 3.4%

   N_2 and other inerts 91.5 - 93.0%

   This is an inert atmosphere as far as the burning of hydrogen is concerned. Moreover, this is so low in hydrogen that no flammable mixture would be produced in the cleanup system, and in fact, the system would become inerted during the venting operation.

   The staff has therefore concluded that it will be feasible to operate the cleanup system without the danger of flammable gas mixtures.

2. Nuclear criticality safety must be considered. The quantities of plutonium that could be collected in the system from the three releases of plutonium are:

   a) The initial release would contribute nothing since this would all have fallen out with sodium aerosols prior to venting.

   b) A conservatively estimated 3200 g would be released to the atmosphere during sodium boiling (Section A.5). Of this, more than 70% would precipitate with sodium aerosols during venting and purging, as shown by the HAA-3 calculations which are conservative in this respect. Thus 960 g plutonium would enter the cleanup system from this source.

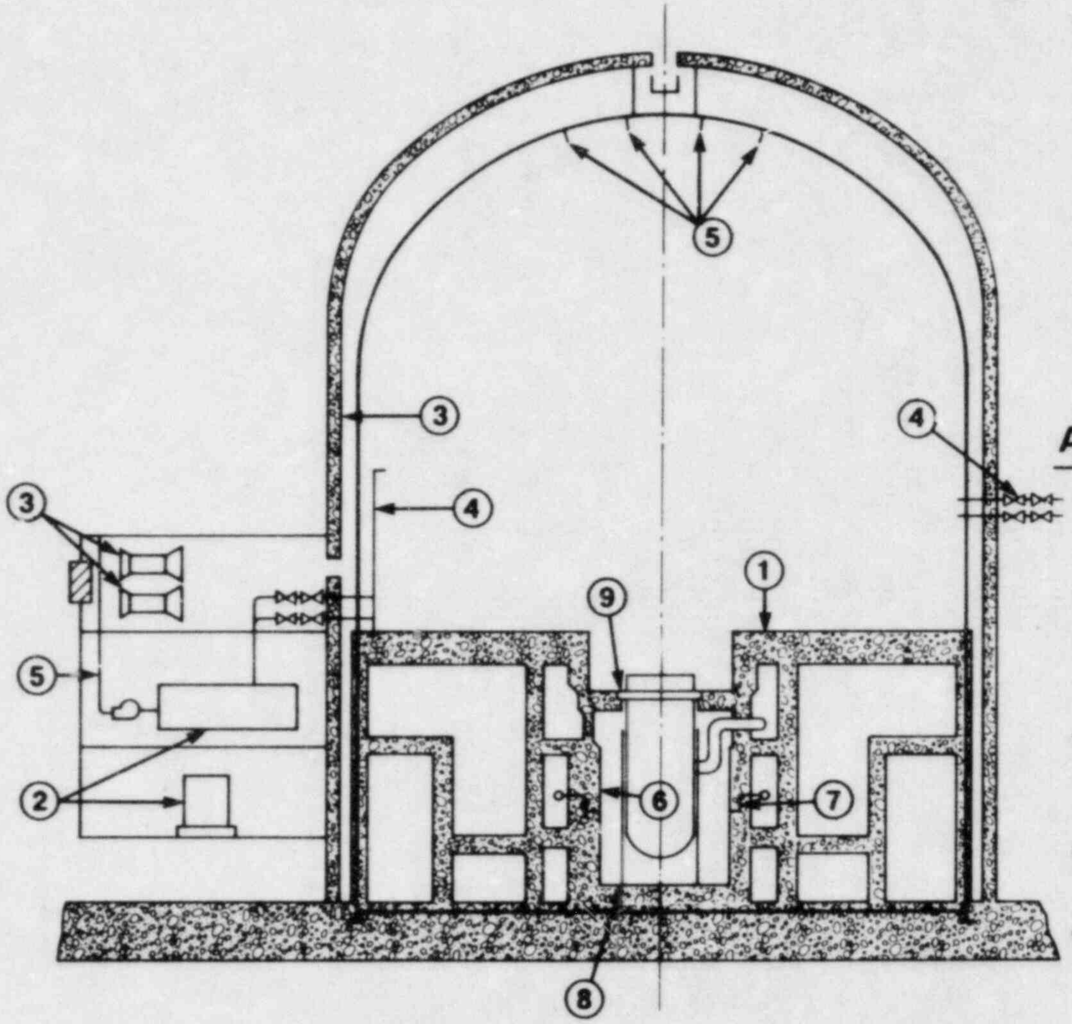
   c) Sparging of plutonium during the core debris concrete interaction could contribute another 26 g, taking no credit for fallout.

   The total, 986 g includes 838 g fissile plutonium. Some portion of this may be in the form of soluble plutonates. The minimum critical mas of a solution or fine dispersion in water of this enrichment is estimated to be about 1160 g fissile plutonium (TID-7016). Thus criticality would not be reached in the cleanup system with these quantities of plutonium. The estimates are considered to be conservative throughout, in regard to releases from the core debris and in the aerosol calculations. A further conservatism is the fact that any soluble fraction of the plutonium would tend to be dispersed throughout the cleanup system in a very dilute solution, and not subject to being configured in an optimum geometry.

The staff therefore believes that there is sufficient margin to assure nuclear subcriticality in the cleanup system.

3. Equations that are used to justify the acceptability of the design have empirical coefficients determined from model tests. The applicants have submitted a similutude analysis between the model tests and the CRBR cleanup system to show that the design equations are valid and that critical quantities that determine system features lead to a feasible design (Reference 8). The staff considers this a sufficient demonstration of the feasibility of designing such a system.

DESIGN FEATURES PROVIDING THERMAL MARGIN BEYOND THE DESIGN BASE



SPECIFIC SYSTEMS OR COMPONENTS FOR TMBDB

1. REACTOR CAVITY VENT SYSTEM
2. CONTAINMENT CLEANUP SYSTEM
3. ANNULUS COOLING SYSTEM
4. CONTAINMENT VENT AND PURGE SYSTEM
5. INSTRUMENTATION

SYSTEMS OR COMPONENTS WITH AUGMENTED CAPABILITIES FOR TMBDB

6. REACTOR CAVITY AND PIPEWAY CELL LINERS
7. LINER VENT SYSTEM
8. GUARD VESSEL SUPPORT
9. REACTOR CAVITY TO HEAD ACCESS AREA SEALS
10. REACTOR CAVITY RECIRCULATING GAS COOLING SYSTEM (NOT SHOWN)
11. EMERGENCY ELECTRICAL POWER SYSTEM (NOT SHOWN)
12. RCB STRUCTURES (ADDITION OF REINFORCING STEEL) (NOT SHOWN)

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Figure A.4.10-1

A.4.10 TMBDB Systems

The applicants have included, or upgraded, twelve features with the capability to mitigate the progress and the consequences of BDBAs. The purpose and requirements for each of these systems are briefly described and evaluated by the staff's consultants at LANL in Attachment 2 to this appendix. The features included in this group are listed below and in Figure A.4.10-1

1. Reactor Cavity Vent System.
2. Containment Venting Cleanup System.
3. Annulus Cooling System.
4. Containment Vent and Purge System.
5. TMBDB Instrumentation System.
6. Reactor Cavity and Pipeway Cell Liners.
7. Vent System for the Reactor Cavity and Piping Cell Liners.
8. Guard Vessel Support.
9. Reactor Cavity-to-Containment Barrier.
10. Reactor Cavity Penetrations and Recirculating Gas Cooling System.
11. Containment/Confinement System.
12. Reactor Containment Structures.

A brief summary of the findings in the areas of design criteria, maintenance and testing and issues which remain to be completed are given as follows.

Design Criteria

A comparison was made of the staff's criteria, Section A.4, and the applicants' "General Requirements" for TMBDB systems. The staff's consultants find that they are comparable.

Maintenance Testing

The applicants' requirements in this area for the TMBDB features have been reviewed and found to be acceptable. They are met directly by design base procedures or simple extensions of these procedures. For features specifically added for TMBDB, the systems are to be completely tested for functional capability approximately once a year with tests scheduled during a shutdown period when a CDA could not occur. The features will also be given pre-service full functional tests after construction.

Equipment Qualification

The staff shall require, for the OL review, evidence that those systems, components and instrumentation that are a part of the systems to accommodate TMBDB have been environmentally qualified. Depending on the particular item, the environmental testing shall include temperature, pressure, radiation and aerosol concentrations. Testing requirements shall include considerations of the need for repeated operation during a TMBDB scenario and the potential for degradation, warping and clogging (because of aerosol buildup).

Instrumentation Redundancy

There are six subsystems which are considered in the category TMBDB instrumentation: Containment Pressure; Containment Atmospheric Temperature; Containment Hydrogen Concentration; Containment Vessel Temperature; TMBDB Exhaust Plant Effluent Radiation Monitoring; and High-Range Containment Area Radiation. The staff believes the double redundancy provided for each measurement in the category is adequate. The staff believes the LANL recommendation for triple redundancy for these measurements is unnecessary since the plant operators should respond to the most conservative of the two measurements if there is any doubt.

Of the six instrumentation subsystems both the hydrogen concentration and pressure measurements have a backup means of obtaining data. The two continuous hydrogen analyzers are located in the Intermediate Bay of the Steam Generator Building. The applicants claim this area is accessible, prior to and after the initial venting, to obtain grab sample for analyses. This is also the station from which carbon monoxide measurements may be obtained and the information transmitted to the control room. However, since multichannel gas analyzers are available, the staff believes that the carbon monoxide measurement could also be accomplished automatically and made available to the control room, if necessary. The capability for obtaining a grab sample should remain a backup.

It is not necessary for both readings to be displayed simultaneously to the operator. However, a means to selectively obtain each reading must be provided.

Remaining Issues

The failure criteria for the cell liners under the loadings imposed by TMBDB temperatures have not been finalized. The applicant has however committed to a program of analyses and experiments to resolve this issue (Reference 9). The applicants' plan includes a confirmatory test program, to be defined jointly with the NRC, based on the complete liner - stud finite element analysis. If the test results indicate that design modifications are necessary, the applicant has several fall-back design options for consideration. The fall-back options that may be pursued include reduction in the size of the nelson stud, increasing stud flexibility, or increasing stud spacing.

The staff has examined the applicants' proposed program and schedule and has concluded that these represent a satisfactory method of confirming the acceptability of the present design or developing an acceptable fall-back design based on feasible alternatives.

A.4.11 Staff Introduction to Submitted Technical Evaluation Reports

1. An assessment of Thermal Margins Beyond Design Basis for Postulated CDAs in the Clinch River Breeder Reactor, R. D. Gasser and S. Hsieh, Brookhaven National Laboratory, dated December 1982.

   The material presented in Attachment 1 raises a number of questions about the applicants' analysis of TMBDB. The most prominent of these has to do with the accumulation of deposits of aerosol materials on the inner walls of the containment shell. If these deposits are fluffy in nature and thick enough they would create an insulating effect that could interfere with the efficient functioning of the annulus cooling system. The BNL calculations for a conservative set of parameters lead to higher temperatures in the containment atmosphere (about 2200°F) than the applicants had calculated. This would be unacceptable if confirmed. They also lead to the suggestion that venting may have to take place earlier than had previously been expected due to high containment pressure. The TER authors note that the calculations are conservative. Since these results would be unacceptable for the applicants' design, further confirmation was undertaken.

   The staff reviewed the calculations in the TER as well as supplementary material from the applicants. The thermal resistance calculated in the TER are based on deposition of aerosols calculated with the MSPEC (Reference 10) aerosol code. The principal driving force for wall deposition in this case is thermophoresis. The staff's examination of these deposition calculations indicates that the results appear to be somewhat higher than would be expected from the HEDL tests in the CSTF facility, (Reference 11) even after correction of the HEDL data for the higher heat flux of the CRBR.

   The suggestion has been made that the heat flux should be reduced as the surface deposit builds up, so that thermophoretic deposition as calculated by the MSPEC code would be reduced in the later stages of the accident. The staff does not think that a major correction is warranted due to this effect, since changes in heat flux are probably minor.

   The BNL calculation of the thermal resistance of the deposited layer (as calculated by MSPEC) is probably satisfactory. An arbitrary 50% increase in the mass of aerosol deposited has been more than compensated by a low estimate of the porosity of the deposit.

The BNL calculation is certainly conservative in assuming that the full thickness of the deposit would be present from the start of the calculation, rather than building up gradually during the boil-off period. A re-examination of this assumption has led the staff to conclude that the buildup of the deposits would not have a significant effect on heat transfer prior to venting, and hence the time for venting would not be advanced.

The staff also notes that the presence of sodium hydroxide in the wall deposit has been neglected in the BNL calculation based on the interpretation that the concrete floor is the only source of water, and this would be absorbed in floor deposits rather than on walls. The applicants have pointed out, however, that the burning of hydrogen in the containment atmosphere is a major source of water and that this water can lead to sodium hydroxide in the wall deposits. Sodium hydroxide is expected to reduce the thermal resistance of the wall deposits, especially if the deposits reach temperatures approaching the melting point of NaOH (around 600°F).

The applicants have recalculated the containment atmosphere temperature based on wall depositions that they believe are more realistic. They have modeled the gradual buildup of the deposition to a thickness that appears to be conservative with respect to the HEDL tests. This model leads to a maximum temperature in containment of about 1550 F at 110 hours. The applicants suggest that this temperature may be reduced even further as a result of more careful analysis and testing. Even if it became necessary to design the critical TMBDB equipment to a temperature around 1550°F, the applicants believe that this is feasible (Reference 12).

The vital equipment systems required here are the thermocouples, the pressure transducers, the purging isolation valve, the atmosphere monitoring systems and the cleanup system itself. Thermocouples, valves and pressure transducers are feasible at temperatures of 1550°F; the atmosphere monitoring system merely extracts a sample for analysis elsewhere and can be designed for 1550°F; and the cleanup system can be protected from excessive temperature by an increased water supply reservoir if necessary. The major structural components would be subjected to lower temperatures because of the insulation value of the aerosol deposits.

The staff therefore agrees with this assessment and will require that analysis and possible experimentation be continued to assure that a feasible limit, of about 1550°F or less, will not be exceeded in the later stages of the accident.

Another point raised in the TER is that the CACECO code is not sufficiently validated. The staff recognizes that the code has imperfections, but finds that sufficient documentation has been submitted by the applicants to support the major features of the code. Some of the criticisms raised in the TER are now recognized to be beyond the scope that can be expected in a code of this nature.

In one other area, the BNL analysis has adopted a different interpretation of the accident procedures than the applicants, leading to significantly different results. This is the criterion for venting. The time at which this criterion is met is calculated by CACECO. The TER notes that a containment pressure criterion for the initiation of venting is more appropriate in some scenarios than the hydrogen concentration criterion. The CACECO calculations reported in the TER used a pressure of 1.5 times the design pressure for the criterion (1.5 x 10 psig = 15 psig), thus leading to vent times on the order of 15 hours for a realistic upper bound concrete penetration rate. The staff has concluded that a more realistic failure pressure criterion can be used in the considerations for TMBDB. Since it has been determined that a failure pressure about four times the design pressure can be realistically expected, the staff believes that a venting criterion about 3 to 3.5 times the design pressure can be used. This will extend the vent time to the neighborhood of 24 hours, in closer agreement with the previous expectations on this subject.

The staff believes that the BNL TER has raised questions about the CACECO calculations that will require some further refinement and study. Many of these are small corrections that affect the sequence of events and their timing in minor ways. The two major points that appeared to represent potentially major problems are the accumulation of aerosols on the containment walls and the calculated reduction in vent time. The accumulation of aerosols on the walls has been recalculated based on the recent experimental data, and found to be less of a problem than originally thought, so that the feasibility of the system to meet the design criteria is not in question on this account. The earlier vent time is associated with a pressure criterion that appears to be more conservative than necessary. The staff has therefore concluded that neither of these items present major feasibility problems for the applicants' TMBDB concept.

2. Final Technical Evaluation Report of Thermal Margin Beyond Design Basis Features, by T. A. Butler et. al., Los Alamos National Laboratory, dated January 1983.

   In this TER, the authors have evaluated the criteria used to design the TMBDB features and the feasibility of the applicants' design. The authors have provided a favorable evaluation of the TMBDB design requirements.

The extreme containment atmosphere temperatures calculated in the BNL TER were furnished to LANL too late for their incorporation. It has already been pointed out that the BNL temperatures are conservative in several respects.

The staff and the authors of the LANL TER consider that feasible modifications to the applicants' design could accommodate considerably higher heat loads at the end of the sodium boil-dry period, if this should prove necessary. Nevertheless, as these temperature calculations become refined, the applicants will re-evaluate the key TMBDB features against whatever heat loads are indicated for the FSAR.

We have concluded that with successful completion of the confirmatory analysis and testing and consideration of fall back positions, an acceptable design will be developed before fabrication of cell liners.

3. Review of the Interaction of Sodium with Concrete and Other Materials, by D. G. Swanson and J. N. Castle, Applied Science Associates (prepared under contract to BNL) September 1982.

In this report the authors have collected and summarized a great deal of information from the background of experiments on reactions of sodium with various types of concrete. This review, together with the views of other consultants at Sandia and the reviews presented by the applicants, have helped the staff to reach its conclusions regarding sodium concrete reactions in the TMBDB framework. These conclusions are that the applicants have evaluated this system against a sufficient spread of conditions to enclose the principal effects from all the available experimental data on sodium concrete interactions.

The applicants have proposed the use of a dolomitic limestone aggregate in their concrete because of its availability onsite. The dolomitic limestone has a higher magnesium and lower calcium content than the calcitic limestone that has been used as aggregate in most of the limestone concrete interaction tests with sodium.

At the time of the preparation of the report by Swanson and Castle, the authors found that there was insufficient information on the comparability of dolomitic and calcitic limestone concrete to support the switch to dolomitic limestone. There has now been a very carefully controlled set of calcitic and dolomitic comparative tests between HEDL and Sandia on the 1-ft. diameter scale, plus a number of smaller independent tests. After reviewing the present state of information, our consultants at Sandia have withdrawn their previous objections to the use of dolomite in this application. They believe that although the chemical differences between the two materials produce observable changes in reaction rates, that these do not lead to higher overall penetration rates, or to higher pressure or pressure spikes in containment. Swanson and Castle now also concur with this view.

The staff has reviewed the available information on this matter and has concluded that the use of local dolomitic aggregate will not be detrimental to the TMBDB aspects of the CRBR concrete. In regard to the recommendation that concrete protective materials should be further investigated, the staff now believes that sufficient testing on ordinary concrete has been completed to demonstrate that these reactions will not exceed the capacity of the TMBDB mitigation systems.

A.4.12 Post TMI Requirements

The Post TMI requirements for light water reactors under degraded core conditions are listed in NUREG-0660. Of these, the requirements of Section II.B.8 deal with the production and possible hazards of large volumes of hydrogen, and therefore their applicability to CRBR has been reviewed. It has been found that requirements II.B.8(3), on hydrogen control, and II.B.8(4), on assurance of containment integrity in the event of hydrogen generation, are applicable, in modified form, to CRBR. The staff has reviewed these items and has determined that the design proposed by the applicants does provide for the fulfillment of these requirements and is acceptable in this regard.

A.4.13 References

1. CRBRP-3; Hypothetical Core Disruptive Accident Consideration in CRBRP, Volume 2, Assessment of Thermal Margin Beyond the Design Basis, (hereafter CRBRP, Vol. 2).
2. T. W. Ball et. al., "TMBDB Sodium-Concrete Penetration Margin Assessment for CRBRP," August 1982, Acc. # 8302010437.
3. "TMBDB Melting Scenario," Attachment to Letter HQ:S:82:127, Longenecker to Check, 11/23/83, Acc. # 8211300428.
4. Aerosol Release from Sodium-Concrete Reactions, L. D. Muhlestein and R. P. Coburn, October 1982.
5. Letter from Dana Powers, SNL to T. J. Walker, USNRC, dated January 24, 1983.
6. NUREG-0358, FFTF SER, Supplement 1.
7. R. D. Peak, Users Guide to CACECO Containment Analysis Code, HEDL-TME 79-22, June 1979.
8. Letter, J. Longenecker (DOE) to P. S. Check (NRC) December 7, 1982, Enclosure 5, NRC Accession No. 8212140374.
9. Letter, J. Longenecker (DOE) to J. N. Grace (NRC), HQ:S:83:231, "Cell Liner Design Validation Program," March 4, 1983.
10. NUREG/CR-2923(BMI-2100) MSPEC User's Manual by H. Jordan et. al., September 1982.
11. R. K. Hilliard et. al., "Submerged Gravel Scrubber Demonstration as a Passive Air Cleaner for Containment Venting," HEDL-TME-81-30, November 1981.
12. Letter, J. Longenecker (DOE) to J. N. Grace (NRC), HQ:S:83:219, "Thermal Margin Beyond the Design Base (TMBDB)", March 4, 1983.

ATTACHMENT 1

BNL-NUREG-
INFORMAL REPORT
LIMITED DISTRIBUTION

AN ASSESSMENT OF THERMAL MARGINS BEYOND DESIGN BASIS FOR POSTULATED HCDA's IN THE CLINCH RIVER BREEDER REACTOR

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December 1982

Prepared for
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555
Under Interagency Agreement DE-AC02-76CH00016

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1. INTRODUCTION AND SUMMARY

Upon resumption of the licensing procedures for the Clinch River Breeder Reactor Plant (CRBRP), Brookhaven National Laboratory was requested by the NRC to perform a review of the TMBDB (Thermal Margins Beyond the Design Basis) report.[1] A preliminary report[2] of that review was supplied to the NRC in April 1982.

In order to facilitate the resolution of particular issues related to containment analysis, Brookhaven acquired the CACECO code.[3] The CACECO code was developed at the Hanford Engineering Development Laboratories (HEDL) for the purpose of analyzing the response of the containment to hypothetical core disruptive accidents (HCDA's) in liquid metal cooled fast breeder reactors (LMFBR's). The CACECO code was used to perform "over-check" calculations on the CRBR Project's "Baseline" Case, and the "Extreme Penetration" Case. In addition, sensitivity studies were run for these cases and an attempt was made to perform the "Realistic Upper Bound" (RUB) cases which were suggested by the Project and by the NRC to be more representative of the most severe sodium-concrete reaction rates observed in the experiments. The results of the CACECO analysis are presented and discussed in Section 3.

Under subcontract to BNL, Dr. Swanson of Science Applications, Inc. performed a review[4] of the available experimental data related to the reactions of liquid sodium with concrete. His report is included here as Attachment 3 to Appendix A.

It has been perceived during the course of this review, particularly as a result of the technical exchange meetings, that a modification has occurred in the position taken by the Project staff regarding the acceptable time limit for containment vent initiation. Although a minimum vent time has never been specified in a formal manner, a vent time in excess of 24 hours has, since the inception of the CRBR licensing procedures, been used. Apparently, the Project staff have changed their position regarding a "guaranteed" minimum vent time criterion to a position which implies dependence on the TMBDB design features, namely the vent, purge, and cleanup systems. Presumably, if it can be demonstrated that the cleanup system can be relied upon under all potential accident conditions to remove a sufficient fraction of the radioactive source materials, the requirement for a minimum vent time becomes irrelevant (except with respect to the noble gas sources which cannot be filtered). This position essentially assigns most of the CRBR issues such as sodium concrete reactions, and hydrogen production to positions of secondary importance. If the Project's position has been correctly interpreted, the primary issues now relate to the reliability of the vent, purge, and cleanup systems as well as the reliability of the TMBDB-related reactor containment building instrumentation. The TMBDB instrumentation will be called upon to supply the operator with swift and accurate information regarding the conditions (temperature, pressure, H_2 concentration) within the RCB upon which subsequent operator action will be based.

In view of the above assessment of the Project's position, those issues which have been identified, both elsewhere and in this review, related to the functioning of the vent, purge, cleanup, and TMBDB instrumentation systems, are the most critical issues. These issues (see Sections 2 and 3) include; sizing of the cleanup system to account for greater thermal loading (Section 3), the potential for plugging in the cleanup system vent lines and in the instrumentation sampling and sensing equipment, the accuracy of extrapolating experimentally obtained scrubbing system efficiencies to full scale equipment, and the reliability of the TMBDB instrumentation in the RCB environment.

With respect to the TMBDB assessment of the post-accident containment environmental conditions, a number of issues remain (see Section 2). However, one issue seems to take precedence over the others. In their CACECO code analysis of the containment conditions in beyond-design-basis situations, the Project has not considered the effects of the presence of "plated-out" sodium combustion products on heat transfer processes. There exists plentiful experimental evidence[5,6] suggesting that these combustion products will be layered extensively on both vertical and horizontal surfaces within containment. The analysis performed at BNL and reported in (Section 3) seems to demonstrate that, if these combustion product layers are considered in the CACECO analysis, the post-accident containment conditions are more severe than those reported in the TMBDB assessment. The primary factor is the insulating capability of the loosely consolidated materials. The result is that heat transfer from the containment atmosphere to the passive heat absorbing structures in containment is inhibited, so that atmospheric temperatures and pressures are correspondingly elevated. These more severe conditions will impact the accident sequence in several ways. Due to higher containment pressure, venting must be initiated earlier. If a venting criterion based on 1.5 times the design pressure is used, the Project's baseline case, for example, should be vented at 22 hours compared to 36 hours for the uninsulated configuration. Similarly, the "Extreme Penetration" case would require venting at 2.5 hours as opposed to the Project's estimate of 10 hours. Clearly, the vent initiation times are significantly effected. What is also clear is that venting should be initiated not solely on the basis of hydrogen concentration, but also on the basis of containment pressure. In fact, all the insulated cases studied in this analysis required venting on the basis of overpressure rather than hydrogen concentration. Because less of the energy, both nuclear and chemical, is absorbed by the structural components of the containment building, much larger thermal loadings will be placed on the cleanup system. Thus, both the capacity and the reliability of the cleanup system may require reevaluation. An insulating layer of combustion products on the containment shell will almost certainly inhibit the efficiency of the annulus cooling system, and may, in fact, render it essentially ineffective. The largest difference seen between the Project's cases and the insulated cases occurs subsequent to the initiation of the vent/purge systems when the annulus cooling system is functioning. Finally, the containment conditions appear to be sufficiently extreme that doubt may be cast on the survivability of the currently proposed TMBDB-related instrumentation and sensing devices. For example, the peak temperature in the baseline case is increased from 917°F to 2188°F, while the average temperature subsequent to vent initiation was increased from about 750°F to approximately 1850°F. These differences are not insignificant and impact the design of the TMBDB systems. On the other hand, by virtue of the insulating effect of aerosol plateout, the thermal loading on the containment building structure is to some extent mitigated. Although the higher containment pressure requires somewhat earlier venting, as long as venting occurs soon enough to save the structure from failure by over-pressure, the threat to the containment building itself (due to high temperature in structural members) is less severe.

With regard to the baseline scenario, 22 hours has been recommended as a reasonable vent initiation time based on the analysis in Section 3. This still leaves a margin of 13 hours over the estimated 9 hour evacuation time, and, if a design pressure factor of 1.75 is used (rather than 1.5) the vent time can be extended to meet the 24 hour venting guideline.

2. DISCUSSION OF ISSUES

The purpose of this section is to review the list of issues that were identified in the preliminary TMBDB review[2] with the intention of resolving as many of these as possible, identifying those that remain unresolved, and perhaps recognizing new areas of concern that have appeared in the interim. The procedure will be to restate each issue verbatim as they appeared in the preliminary review (identified by Item) followed by a section which gives the status of the issue with regard to its degree of resolution (identified by Status).

2.1 Heat Transfer and Factors Directly Affecting RCB Pressure and Temperature

2.1.1

Item

In the absence of an active containment heat removal systems for CRBR, the Project's thermal analysis relies heavily on the collective heat capacities of the passive heat structures in the RCB in order to meet the 24 hour venting criterion. By the time that boil-dry has occurred (130 hrs), nearly 80% of the original sodium inventory has been transferred into the RCB and at least 166,000 lbs of this is oxidized (the quantity of Na needed to remove all the oxygen). The plate-out of this quantity of aerosal on the containment shell and other passive heat sinks will form an insulating layer of porous material that may significantly inhibit heat transfer to these structures. A large fraction of the containment heat capacity lies in the reactor confinement structure outside of the RCB steel shell. Effective insulation of the steel shell by aerosol plate-out would not only limit the heat storage capacity of the shell itself, but would limit the radiative and convective heat transfer to the large concrete structure behind the shell.

Such an insulating layer has not been considered in the Project's thermal analysis.

Status

The insulating effect of aerosols plated out on the containment walls and passive heat structures in the RCB is seen as one of the more important issues which has not been satisfactorily resolved. The project responded to this issue by performing a CACECO run in which the RCB structures were coated with a layer of aerosols. Unfortunately, the case was only run out to the point at which venting was initiated. Their results indicated an increase in the pressure at 36 hours (venting time) of about 3 psi and a temperature increase of about 80°F. Evidence exists[5,6] that thermophoresis plateout will be considerable on structures in the containment building and that the insulating effect may be more pronounced than that calculated by the Project.[7] A CACECO case (given in detail in Section 3) was performed at BNL in which the Project's baseline case (1/2"/hr for 4 hrs Na-concrete penetration rate) was modified to include a layer of aerosols on all RCB structures. This case showed a peak temperature at 39 hours of about 2200°F as opposed to 920°F for the baseline case. Because more of the heat generated by fission product decay and by the various chemical reactions is retained in the atmosphere and not absorbed into containment structures, the heat load on the containment cleanup system may be considerably greater than previously predicted. Also, the higher temperatures may adversely effect the structual capability of containment and reduce the usefulness of TMBDB instrumentation. These results should be considered in the final cleanup system design.

2.1.2

Item

In terms of the solution of the atmospheric energy equation; the solid sodium combustion products remain in the atmosphere in the sense that they are assumed to be in thermal equilibrium with the atmosphere. This is non-conservative on two points. First, as pointed out in 2.1.1 above, plate-out of these materials will inhibit heat transfer to the passive heat sinks. Secondly, the presence of this large mass of aerosols in thermal equilibrium with the atmosphere provides the atmosphere with a large and instantaneous heat sink which will artificially depress the containment temperature and pressure.

Status

This issue has not been resolved. The project has verified that in the CACECO code modeling the sodium combustion products are held up in the atmosphere and are an integral constituant of the atmospheric energy equation. In terms of the atmospheric temperature and pressure, this modeling is non-conservative. The Project feels that although the error is on the non-conservative side, it is not a significant one. They have, however, done no calculation to determine the extent of the error.

2.1.3

Item

In the baseline case (Chap. 3, Ref. 1), the thermal analysis of the containment shell and the confinement building is carried out in two steps outlined in Section 3.2.2.2. There is some confusion as to exactly how these calculations were performed. The statement is made that the CACECO code calculations determine the heat load to the containment shell and the TRUMP code calculates the response of the shell and the confinement building to this load. To start with, this is a coupled problem and apparently no attempt was made to iterate between the two codes to insure that energy is conserved. Secondly, it is not clear whether this uncoupled analysis commences immediately (at the beginning of the CACECO analysis) or only when the annulus cooling system is turned on. It appears from the nodalized structure (Fig. 3-27, Ref. 1) that the air space is treated by the TRUMP code as a stagnant air volume. Clearly, this is not appropriate when the annular cooling system is initiated at 36 hours.

Reference to the CACECO input parameters for the baseline case prior to 36 hours indicates that heat is transferred by natural convection from the RCB atmosphere to the shell and from the shell to the confinement wall by radiation. This is not consistent with Section 3.2.2.2 which states that heat is transferred from the shell to the confinement by radiation and from the shell to the annulus air space by convection. Subsequent to 36 hours, a constant convection heat transfer coefficient is added, but the bulk temperature to which the shell transfers heat by convection is not defined in the input. If the undefined input parameter is initialized to zero, the heat transfer rate out of the RCB may be grossly overpredicted even when the insulating effect discussed in (2.1.1) above is neglected.

More information is needed regarding the use of the TRUMP code in connection with the CACECO code for thermal analysis of the containment-confinement walls.

Status

The use of a separate code to model the annular area between the containment shell and the surrounding concrete structure was necessary due to the fact that CACECO is limited to 4 separate containment volumes, and to the fact that it cannot model the forced convection due to fan flow in the annulus cooling system. However, the criticism of the calculational approach remains valid. This is a highly coupled heat transfer problem which has been uncoupled in the solution. It is not clear whether the approach is conservative or non-conservative. The issue, therefore, at this point remains open.

2.1.4

Item

Heat transfer between the cavity atmosphere and the non-submerged cavity walls is via natural convection until "depletion of non-condensible vapors" occurs (p. 3-33, Ref. 1). Subsequently, a sodium condensation heat transfer coefficient is used. It is not specified what is meant by "depletion" in the present context. It is a well established fact that very small quantities of non-condensibles (less than 1 volume percent) will strongly inhibit the condensation heat transfer rate. On the other hand, condensation processes in the presence of quite large volume fractions of non-condensibles can produce effective heat transfer coefficients considerably in excess of those typical for natural convection. Since the temperature drop across the condensing film is negligible, no iteration is necessary and it is a relatively simple task to calculate the condensation heat and mass transfer rates as functions of the atmosphere-to-wall temperature difference and the concentration of non-condensibles.

Status

Updates to the CACECO code which were not present in BNL's version of the CACECO manual (and were, therefore, unknown to us) have supplied a condensation heat/mass transfer model to the code. That a model does exist in CACECO for condensation is no longer an issue. There are, however, some minor questions regarding the details of the model. A cursory review of the model reveals that it is strictly a diffusion model in which convective effects are ignored. In the present application, this drawback seems to be relatively insignificant, so that this point may be considered to be resolved.

2.1.5

Item

There exists a gap between each of the steel liners and the concrete walls which the liners are designed to protect. Heat transfer from the cavity atmosphere to the wall is modeled by a single heat transfer coefficient which is input as a tabular function of the temperature difference between them. In reality, several heat transfer processes are involved. Heat is transferred to the liner by convection (h_{ic}), condensation (h'_{ic}), and radiation from the pool (h_{ir}). Heat is transferred from the liner to the concrete wall by natural convection before significant out-gassing commences, by forced convection (h_{ic}) afterward, and by radiation (h_{2r}) during the entire process. Since the radiative component on the cavity side is a function of the pool temperature and not the ΔT between the atmosphere and the wall, and since the condensation heat transfer coefficient is a function of the cavity-to-wall ΔT as well as the concentration of non-condensibles, the heat tranfer coefficient cannot be accurately cast into the form described above. These individual heat transfer coefficients should be mechanistically evaluated, and the overall coefficient calculated by:

$$h_{eff} = \frac{1}{\dfrac{1}{h'_{ic}+h_{ic}+h_{ir}} + \dfrac{1}{h_{2c}+h_{2r}} + \dfrac{\delta}{k}}$$

where $\frac{k}{\delta}$ is the equivalent heat transfer coefficient through the steel liner.

Status

The primary concern in this issue is the time at which pool boiling occurs or more specifically the time at which sufficient sodium vapor pressure exist to self-ignite the hydrogen as it is vented into containment. If heat could be transferred from the pool to the surrounding cavity walls more rapidly, pool boiling could be delayed long enough to build up higher H_2 concentrations in the RCB before ignition occurs. The Project performed sensitivity study by increasing the thermal conductivity of the concrete in the cavity walls by 20% (.12 to .14 Btu/ft-hr-°F). This allows more efficient transfer of heat away from the pool. In order to ascertain the effect of even more efficient heat transfer, a similar case was run at BNL in which the thermal conductivity of the perlite concrete was increased to 0.2 Btu/ft-hr-°F (67% increase). The time required to reach pool boiling was not significantly increased. Apparently, the decay heat and the chemical energy inputs to the pool overpower the heat losses so that even pessimistically high heat transfer rates do not result in significantly greater H_2 concentration in containment.

2.1.6

Item

Some of the graphs show calculated data out past 130 hours which is the time at which the CACECO analysis was terminated. How were the pressures and temperatures obtained for these times?

Status

This question has been resolved. The graphs in question were temperature plots of heat structures. The data was obtained by the TRUMP code not the CACECO code.

2.1.7

Item

Other than the first "spike" in the temperature history which is due to an adiabatic hydrogen burn, it is not at all clear what causes the wildly fluctuating RCB atmospheric temperatures (Fig. 3-6, Ref. 1). These temperatures appear to oscillate between about 600°F to about 900°F with a period of roughly 20 hours subsequent to incipient pool boiling. In the absence of any active or passive cycling phenomena, this behavior is not understandable.

Status

This issue is resolved. The shape of the temperature curve can be explained by various events such as H_2 burns, oxygen depletion, and cell liner failures.

2.1.8

Item

It is stated (p. C.1-2, Ref. 1) that additional sodium that may be pumped into the vessel by the makeup system is ignored and that this is conservative in terms of containment loading. This is not obvious particularly in light of point (2.1.1) above.

Status

This remains a legitimate issue. In addition to the extra aerosols that would be generated, a larger sodium pool would heat up slower with a correspondingly larger H_2 concentration in the RCB. This is the same sort of effect discussed in 2.1.5 above.

2.2 Sodium-Concrete Reactions

2.2.1

Item

The issues regarding the sodium-concrete reaction rate and total penetration have not yet been resolved. It is apparent that a number of parameters such as pool depth, mass, initial sodium temperature, and scaling factors, among others, effect the sodium-concrete reaction kinetics. In some of the large scale tests performed at Sandia, penetration rates were observed considerably in excess of the most rapid rates used in the present analysis (1/2 in/hr for 4 hrs in the baseline case, and 1 in/hr for 12 hrs in the most extreme case). The present analysis, therefore, must be considered incomplete and non-conservative since the penetration rates characterized by these experiments have not been considered. Prototypical experiments should be performed to finally (if possible) resolve the differences between the HEDL and Sandia experiments.

Status

It is concluded that, although none of the present models which have been used to describe the reaction of sodium with concrete are entirely adequate, there is sufficient experimental data available to make a reasonable estimate of the sodium-concrete reaction rate and total penetration distance (see Ref. 4 for a review of the Na-concrete experimental program). The Project has analyzed a case (the "extreme penetration case") in which the penetration rate was 7 in/hr for 3 hours with 1 in/hr thereafter. This rate is much higher than anything that has been observed in the experiments. A penetration rate of 7 in/hr for 20 minutes followed by 1 in/hr has been suggested by the Project to be a "Realistic Upper Bound" (RUB) rate which adequately bounds the experimentally observed data. The NRC would prefer the RUB rate, in the interest of conservatism, to be 7 in/hr for 1 hour and 1 in/hr thereafter. Both of these cases were calculated at BNL and the results appear in Section 3.

2.2.2

Item

A final decision has not yet been taken with regard to the basemat material, although the Project has seriously considered the use of Dolomite concrete rather than the calcite based concrete, which was considered in the present analysis. If Dolomite is selected as the basemat material, this type of aggregate should be experimentally investigated with respect to its chemical behavior in the presence of sodium. There are indications that those reactions may be considerably more severe.

Status

Several experiments have been performed and several more planned in an effort to normalize dolomite concrete to calcitic concrete in terms of the sodium-concrete reaction. No significant differences have been noted and, although a statistically significant data base is not available for dolomite concrete, it is felt that, barring any unexpected results from future dolomite experiments, the data established in the sodium-calcitic concrete reaction experiments will be applicable. (See Ref. 4 for further comments).

2.2.3

Item

The TMBDB analysis utilizes a reaction heat for the sodium-concrete reaction of 331 Btu/lb (concrete). This number was obtained from a calculation made utilizing data acquired from a Differential Thermal Analyzer. It is our understanding that this equipment is not an appropriate instrument for obtaining heats of reaction. This data should ideally have been obtained using a calorimeter. If the heat released by the reaction of the water vapor and the CO_2 present in limestone concrete is added and the heat of vaporization is subtracted, the result is about 1350 Btu/lb of concrete. Obviously, some of this energy is used in breaking the bonds to release the CO_2 from the carbonate radical and the water from the hydrate bond. Nevertheless, this reaction heat is over 4 times that used in the TMBDB report. There exists a degree of uncertainty in the reaction heat, and since consideration of penetration rates consistent with the Sandia tests must ultimately be made, the energy released in the Na-concrete reaction may be comparable to the decay heat, and a reasonable estimate of the heat of reaction will be important.

Status

So far as we know, the differential thermal analyzer is not the instrumentation of choice for determining reaction heats. Criticism of the experimental methods for "measuring" the heat of reaction of the sodium-concrete reaction, therefore, remains in place.

2.2.4

Item

The CACECO code modeling allows for the addition to the pool of heat generated in the "sodium concrete reaction." This heat is specified through a "heat of reaction." In addition, water vapor and carbon dioxide, which are out-gassed from the concrete (behind the failed floor liner), are bubbled through the pool. The vapor and gas are in turn reacted with sodium, and their reaction heats are also added to the pool. This modeling appears to be inconsistent, since a large component of the sodium-concrete reaction heat comes precisely from the reactions of these constituents with sodium. It is questionable whether the concept of a "reaction-heat" can be applied to the attack of sodium on concrete. As the reaction front proceeds into the concrete, it will encounter concrete which has been progressively more "degassed" by the earlier passage of the thermal front (unless the pentration rate is greater than the thermal front propagation rate). The effective sodium concrete reaction heat must, then, be adjusted according to the quantities of reactants still present in the local reaction zone together with the quantities of reactants that are diffusing into the zone due to the thermal front propagation. It appears, therefore, that the CACECO treatment of the sodium-concrete reaction may be accounting for reaction energies redundantly.

Status

This issue has been successfully resolved. Apparently the "heat of reaction" for the Na-concrete reaction does not include the heats of reaction of water vapor and CO_2 which are constituents of the concrete. The heat of reaction includes the reaction heats for all the reactions except the last in which $H_2O(V)$ and CO_2 react with the sodium pool.

2.3 Chemical Reactions in the RCB

2.3.1

Item

The TMBDB report assumes that sodium vapor entering the atmosphere is reacted only with oxygen if the mole fraction of water vapor is less than that of oxygen. The rationale for this assumption is unclear. It is equally as reasonable to assume that sodium will react with oxygen and water vapor according to their mole fractions. If the calculation is performed with the latter assumption, two things can happen depending on the temperature of the reaction zone. If the reaction zone temperature is low enough, the hydrogen produced in the sodium-water reaction may not be burned. In this case, the CACECO modeling is non-conservative in that it does not allow for the accumulation of hydrogen from this source. If the reaction zone temperature is high enough and the hydrogen is burned, then the assumption is still non-conservative in that, although the heat of reaction is slightly larger in the event that only oxygen is consumed, the inventory of gases remaining in the atmosphere after the reaction is greater with the latter effect dominating the former.

Status

The Project basis this criterion on experiments conducted at HEDL[8] in which a jet of sodium vapor was introduced into a chamber containing varying quantities of $H_2O(V)$. In these experiments, the formation of hydrogen was observed to be negligible for $H_2O(V)$ concentration less than the O_2 concentrations. Our position is that the sodium vapor was probably combined with water vapor in proportion to its concentration but that the reaction zone was sufficiently hot so that the H_2 thus formed was itself burned in the oxygen present and therefore, not observed in the exhaust gas measurements. The sodium vapor density in the jet (20 gm/m^3) was not varied to determine if a slow accumulation of H_2 might result from a lower concentration of Na in the jet or from a slower flow rate of sodium vapor into the test cell.

2.3.2

Item

Water vapor entering the containment building is constrained to react with Na_2O to form NaOH rather than accumulating in the atmosphere (p. 3-4, Ref. 3, Item C). The argument is that water entering the RCB atmosphere from the concrete heat structures must pass through the layer of sodium combustion products plated out on these structures. This argument is inconsistent. As pointed out in (2.1.1) and (2.1.2) above, such a layer is not accounted for in terms of heat transfer to the structures, and the reaction products are assumed to remain in thermal equilibrium with the atmosphere (as if suspended, therein). A further argument maintains that even after the oxide is consumed, water will be preferentially held up in this layer due to its affinity for NaOH. However, the accumulation of water in such a layer cannot exceed that which is allowed by saturated conditions. In addition, some of the water vapor that enters the containment comes through the vent system from cell 105. The effect of this modeling is that water vapor does not accummulate in the containment building. This is non-conservative with respect to hydrogen production since water vapor is rendered artificially unavailable for reaction with the sodium vapor that is being vented into the RCB from the pipeway cells with the possible accumulation of hydrogen from this reaction.

Status

This question was transmitted to the Project staff before the first technical exchange meeting (3/17/82) and they responded during that meeting. The Project's response that this item is not an issue and that if water vapor O_2 or CO_2 exists in the atmosphere they will react with Na first before reacting with Na_2O. Basically, the Project staff answered the wrong question. As long as Na_2O exists in a cell, water vapor will not accumulate in the atmosphere. The order in which chemical reactions are calculated in the CACECO code is outlined in the User's Manual[3] (p. 3-4, Item C). The $H_2O(V)$-Na_2O reaction is calculated before the $Na(V)$-$H_2O(V)$ reaction and it goes to completion in each time step so that as long as an excess of sodium oxide exists, water vapor cannot enter the atmosphere in order to be available for reaction with sodium.

2.3.3

Item

The presence of electrical cable insulation materials, thermal insulating materials, and other material composed of hydrocarbon or halogenated hydrocarbons can decompose in the post-HCDA containment environment and/or react with sodium vapor. The range of reaction products is quite extensive. Some of these products include solid compounds which are explosively unstable. Some can react with water vapor to form acetylene which can decompose explosively even without the presence of oxygen. A careful inventory of these classes of materials should be made for the containment building in order to assess the potential danger from the more exotic reaction products.

Status

This question has not yet been addressed by the Project. In view of the results obtained in the "aerosol insulated" cases, the temperatures in containment may be higher than reported in the TMBDB report so that the potential for the formation of some of these materials may be greater. This concern, therefore, remains.

2.4 Hydrogen

2.4.1

Item

The CACECO code modeling for hydrogen flammability is outdated. There is a great deal of readily available information obtained in LWR hydrogen-related investigation. One of the criteria for venting and purging is based on a 6% hydrogen concentration. The flammability limit for hydrogen is about 4%, and more recent data indicates that a H_2 burn occurring at 6% would burn about 53% of the available hydrogen. In addition, CACECO modeling does not provide for the formation and accumulation of carbon monoxide. The presence of CO together with hydrogen could result in burns which occur at lower H_2 concentrations. This may be a problem, particularly after sodium boil-dry when the debris begins to re-melt.

Status

In general, the comment regarding the Project's overall treatment of hydrogen flammability is still appropriate. The hydrogen flammability limits used by the Project do not reflect the latest available information, and the combined effects of hydrogen together with other flammable gases such as carbon monoxide have been ignored in the TMBDB assessment.

2.4.2

Item

After the release of hydrogen from the sodium-water reaction, the CACECO modeling provides for the hydrogen to combine with sodium to form the hydride, NaH. The hydride is retained in the sodium pool until some specified temperature and/or pressure condition is obtained at which time it is decomposed, thus releasing the hydrogen to be carried into the RCB. It is probably no coincidence, however, that this criterion keeps the hydrogen confined in the pool until enough sodium vapor is available (pool temperature approaches the boiling point) such that the mixture will ignite and burn upon reaching the containment atmosphere thus avoiding any significant accumulation of hydrogen. There is no indication however, that this is a credible mechanism by which the desired end may be obtained. Reference to the inorganic chemistry texts (Appendix II) reveals several important facts regarding the formation and stability of the hydride. First, when the conditions are right for the formation of the hydride, actual formation is strongly controlled by kinetics and by the transport of hydrogen through the pool (pool depth, etc.). Secondly, the decomposition of the hydride is not kinetically controlled, but rather is specified by the equilibrium conditions. Third, the temperature range is extremely narrow over which the hydride can exist. In the neighborhood of atmospheric pressure, the hydride will not form until about 350°C (662°F) and will be almost completely dissociated at 430°C (806°F). Since the sodium enters the cavity at about 990°F, there is little, if any, potential for the formation of sodium hydride.

The Project employs the results of experiments performed by Whittingham[4] to "estimate" the equilibrium quantity of sodium hydride in the pool. (See Appendix II pp. 50 and 51, Ref. 2 for a more detailed discussion of these experiments and their application to CRBR). In general, these experiments were performed at low temperature where the sodium hydride concentration is relatively high. The temperature range was extremely narrow (610-677°K). The project has extrapolated the results of the Whittingham experiments to temperature far beyond the range of application justified by the experiments and into a temperature range to which the assumptions of the model do not apply. No direct measurement of sodium hydride concentrations have been made over the appropriate temperature range. It is our opinion that only trace quantities of this compound will exist in the post-HCDA sodium pool and that before credit can be taken for its existence under these conditions, the appropriate direct measurements must be made to establish the actual concentrations under prototypic conditions.

If the quantity of water vapor entering the sodium pool before incipient boiling (as calculated by CACECO for the baseline case) is reacted and the resulting hydrogen is released to the containment building, a 6% concentration would be predicted. This implies that by the Project's criterion (6% H_2), the containment would have to be vented and purged at or about 9 hours.

If this quantity of hydrogen were completely burned adiabatically in the RCB, the peak pressure would be about 50.3 psia (33.4 psig) and the temperature would be 1320°F. This is in excess of the estimated failure pressure (30 psig) and even if the containment boundary was not breached, it is doubtful that necessary instrumentation and equipment would survive.

Status

The Project maintains that although sodium enters the cavity at a temperature well in excess of the NaH decomposition temperature, a finite concentration of NaH will exist in the pool even at or above the boiling temperature (1618°F). The Whittingham experiments[9] were referenced as evidence for this position. In addition, the Project staff claims that even without the formation of NaH the hydrogen concentration would not exceed 5.2% and would, thus, be acceptable (below the 6% venting criterion).

This question has been discussed in detail above. Essentially, our point is that since the sodium temperature is, at the lowest point, several hundred degrees above the decomposition temperature, only trace quantities will exist under these conditions. With regard to the experiments, they were conducted at a low temperature (within the narrow temperature range in which NaH is stable), and the sodium hydride concentration was not directly measured. Without direct measurements of sodium hydride and dissolved H_2 in the appropriate temperature range, there is no basis for the quantitative calculation of equilibrium hydrogen holdup in the pool. In addition, we estimate that the H_2 concentration at the time of incipient pool boiling is about 5.8% and that other sources of hydrogen may be present during the course of the accident that local or bulk H_2 concentrations may rise to high levels.

2.5 CACECO Modeling

2.5.1

Item

The POOLR subroutine remains inactive in the present version of the code so that the atmosphere in the cell is kept at the same temperature as the pool. This modeling has several drawbacks; a main one is that the transport phenomena (heat and mass) between the pool and the atmosphere cannot be mechanistically treated. The transfer of heat from the atmosphere to the wall by convection and condensation is over-estimated because the atmosphere is held at an artificially high temperature. Prior to incipient pool boiling the concentration of sodium vapor in the cavity atmosphere should be calculated from the appropriate source and sink terms, i.e., evaporation from the pool surface, condensation at the boundaries, mass transport out of the cell.

Status

Although the criticism that the heat and mass transfer between the sodium pool and the atmosphere and heat structures above the pool is poorly modeled (due to the assumption of thermal equilibrium), is certainly an academically relevant issue, a suitable treatment of these rather complicated problems would render the code economically unfeasible to run. It is felt that the thermal equilibrium assumption probably yields a conservative result in the calculations, and that what little could be gained in accuracy by more precise modeling would be purchased at a rather high cost.

2.5.2

Item

A number of errors and modeling deficiencies were identified in the CACECO code in an earlier evaluation (Ref. 5). Most of these were called to the attention of the Project, but no feedback was subsequently received regarding corrective measures. It would be advantageous that the resolution of these problems together with those identified by other users be demonstrated.

Status

The question of code evaluation remains unresolved. The Project has issued several reports[10,11] on the subject of CACECO code evaluation, but to our knowledge no independent evaluation of CACECO has been performed other than the cursory ones[12,13] performed in earlier studies at BNL.

2.6 Mass Transfer Between Cells

2.6.1

Item

Apparently, the CRBR pipeway cell design provides for a drainage system whereby the sodium condensed on the walls of these cells after accumulating to a specified depth (equivalent to 8667 lbs of Na) will be returned to the cavity via gravity flow. This provides an effective increase in the condensing surface and heat absorbing capability before venting the sodium vapor into the RCB. Without this recirculation capability, boil-dry would occur somewhat sooner. What would be the effect on the accident scenario due to the failure of the recirculation system? What potential exists for failure of this system, i.e., plugging of piping? Does failure of the cavity floor or wall liners pose a particular threat to the integrity of the recirculation system?

Status

This question has been resolved. Apparently there is no piping involved. The cavity and pipeway cells are open for direct communication. When the sodium level in the pipeway cell reaches the overflow level, it merely overflows back into the cavity. There is, therefore, no potential for plugging of the flow pathway.

2.6.2

Item

As outlined in (2.1.4) above, the condensation of sodium vapor on the surface of the walls and structures in the cavity is not mechanistically modeled. Presumably this criticism also holds true for condensation in the pipeway cells. Since the condensation process is so critical to the sodium recirculation system described above, a mechanistic model for transport phenomena in these cells is vital if the capacity for the holdup and refluxing of sodium is to be demonstrated.

Status

As discussed under Item (2.1.4) above, the CACECO code now contains a condensation model so that this question is resolved.

2.6.3

Item

An orifice equation is utilized in CACECO to simplify the rather complex problem of determining the exchange of gases and vapors between cell atmospheres. Experience with CACECO and other containment codes has shown that this approximation leads to unstable flow patterns, which often exhibit oscillating flow directions. Particularly with systems in which transfer of chemically reactive materials between cells results in strong positive pressure feedback, the use of an orifice equation leads to diverging solutions and requires prohibitively small time steps to approach a convergent solution. A solution to this problem which has been used by the Project in previous analysis consisted of allowing flow to proceed in only one direction. This is an artificial solution (as opposed to a mechanistic solution) and obviously prevents flow in a direction which may naturally occur in the system. For the analysis of chemically reactive systems such as the CRBR containment, a more sophisticated flow model should be developed.

Status

The flow pathways between cells in CACECO are all modeled in such a way that flow can occur in only one direction. This is clearly seen, for example, in the baseline case where a partial vacuum is seen in the cooler room (cell 105) after containment blowdown. This is obviously incorrect, but one-way flow is the simplest method of suppressing the numerical instabilities which result in most codes that employ ΔP driven flow using orifice equations. Again, solutions are available for this problem, but the minimal gain in accuracy may well be over-balanced by the increase in cost.

2.6.4

Item

Table A1-1 contains a tabulation of all the concrete structures and the cells or locations to which the degassed water vapor and/or CO_2 contained in those structures are vented. In this table, if the venting destination is unchanged from one period (Column 1) to the next, that structure is not reiterated. It is noted that structures with intact liners are vented to cell 105 (except #5) and the only failed liner (the cavity floor, #10) is vented to the sodium pool. The obvious exception is heat structure #3 which is the reactor containment building floor. For this structure the water vapor and CO_2 generated therein simply disappears down the "drain" (CACECO manual's nomenclature). Evidently the mass and energy contained in these gases and vapors does not reappear in subsequent calculations. Since one of the basic laws of physics maintains that mass/energy cannot be destroyed, it is difficult to justify the existence of such a "drain" (Section B.22, p. B-19, CACECO User's Guide, Ref. 3). The net effect is that neither water vapor nor CO_2 from this structure (which is in direct contact with the RCB atmosphere) is added to the RCB cell atmosphere (nor any other atmosphere or pool). This is clearly non-conservative. This structure has a floor area of 24,179 ft^2 and a depth of 6.33 ft for a total volume of 153,125 ft^3. That quantity of concrete contains 1,475,000 lbs of water and (if it is limestone concrete) 8,040,000 of CO_2. The net result is that credit is taken for this huge structure as a heat sink, but the "penalty" is not payed in terms of the gases generated.

Also note that the structure 17 liner (a pipeway wall liner) is vented to the "drain between 30 and 30.2 hours and then re-directed to cell 105 at 30.2 hours. What is the reason for this?

Status

The Project maintains that for concrete heating rates lower than those encountered in the cavity, the dehydration and degassing rates are smaller. And since CACECO contains only a single temperature-dependent concrete degassing table that corresponds to the high heating rates characteristic of the cavity walls, that table is not directly applicable to the concrete structures within the RCB. Therefore, in the TMBDB cases analyzed by the Project, the water and CO_2 vented from these structures was directed into a "drain" and simply disappeared from the calculation. It is clear that this procedure is not conservative. Even assuming that the degassing rates are slower for lower heating rates, the water vapor and CO_2 must still be degassed and, thus, pressurize the containment, notwithstanding the heating rate. In personal communication with CRBR project personnel, it was learned that a case was subsequently run in which the heat structures were divided into degassing and non-degassing concrete structures in an effort to show the effect of partial dehydration. The result was that containment venting was required several hours earlier. A case was run at BNL (see Section 3) wherein the built-in CACECO degassing table was used for the RCB concrete floor. As expected, the level of water vapor and CO_2 in the RCB atmosphere is higher and it was confirmed that, considering only this effect, containment venting should be initiated about 2 hours earlier (34 hours).

2.6.5

Item

Presumably the reactor cavity floor liner like the other liners is vented to cell 105 which is a non-inerted cell used (in the TMBDB scenario) to contain and condense water vapor and contain the CO_2 generated behind the various liners. The assumed failure of the cavity floor liner, therefore, opens a pathway between the cavity and the cell 105 atmosphere. Relief of the cavity pressure through this pathway may result in two rather serious problems. First, the initial pressure differential may cause the injection of liquid sodium into the non-inerted cell 105 resulting in a rapid spray fire. Second, since a pathway exists for relieving the pressure between the cavity and the RCB (cavity through liner vent to cell 105 and by existing vent pathway from cell 105 to the RCB) the reduced differential pressure between the pipeway cells and the RCB may not be sufficient to activate the blowout disks and open up the pathway through the pipeway cells to the RCB. Thus, the condensing and refluxing capability of the pipeway cells may be lost. In addition, if the liner vent pathway remains open, the hydrogen problem could be much more severe due to the injection of sodium into cell 105 which is simultaneously receiving water vapor from the other liner vents.

Status

Unlike the other cell liner vent systems, the reactor cavity floor liner vent is directed into the RCB rather than into cell 105. This, of course, eliminates the sort of problems which were envisioned if this liner had been vented into the cell 105 atmosphere.

2.6.6

Item

Table A1-2 shows the histories of the venting pathways between the four cells used in the TMBDB baseline case. Inspection of the table raises a number of questions regarding the way in which the cell-to-cell flow was handled in the TMBDB analysis. The flow areas for the vent pathways between the cavity, the pipeway cells and the RCB were both increased from 0.15 ft^2 to 0.25 ft^2 (1.5 in. diameter hole to a 2 in. diameter hole) at 30 hours. What physical justification exists for changing the flow areas? What are the actual flow areas? Again at 50 hours the cavity to pipeway flow path was closed (made very small, $10^{-6} ft^2$) and an entirely new pathway opened between the cavity and the RCB of 0.25 ft^2. Figure 2-33 of the TMBDB report shows the flowpaths between the cavity, pipeway cells and RCB, and no such venting pathway is shown! How is it possible for the rather large flow path from the cavity to the pipeway cells be suddenly closed?

In the RCB-to-cell 105 vent pathway, what is the significance of a negative flow area? Negative areas are not defined in the CACECO user's manual.

Finally, it is observed that the area of the containment venting system is steadily increased starting with 0.365 ft^2 when it is originally opened at 36 hours to a final area of 1.0 ft^2 at 80 hours. Does the design for the containment venting system include a variable flow valve?

Status

The apparent anomaly associated with the changing of cell-to-cell flow orifice cross sectional areas has been resolved by personal communicating with Project personnel. Apparently due to numerical instabilities in the intercompartment flow modeling, the orifice flow areas must be selected such that pressure oscillations are avoided. Most of the vent pathways are large enough so that very little ΔP can be maintained between cells, and the primary concern is that cells which contain the driving functions (decay heat sources, sodium vapor sources, etc.) can transmit flows to receiver cells with a reasonable pressure differential and without numerical oscillations. Probably a somewhat more accurate and perhaps more analytically correct procedure would have been to decrease the time step increment to maintain numerical stability. Unfortunately, this would also have greatly increased computing cost.

In terms of the periodically increased containment vent flow area, a different problem was encountered. The purging system consists of a fan that is located on the exhaust side of the cleanup system with a check valve located on the far side of the containment where outside air is ingressed. Thus, the flow of purge air into containment is achieved essentially by holding the RCB at a slight vacuum. The CACECO code cannot model this process. The code provides for air injection so that a positive gauge pressure must be maintained to eject the appropriate amount of RCB gases out through the vent. It is necessary, therefore, to change the orifice flow area periodically in order to match the rated cleanup system flows.

2.6.7

Item

What potential exists for the failure of the containment purge system such that radioactive material from the containment escapes unfiltered out through the purging inlet?

Status

Apparently the purge system contains a check valve at the RCB inlet location. There remains some concern that the check valve mating surfaces may become encrusted with aerosol such that the seal will not be airtight.

2.7 Active Containment Systems

2.7.1

Item

No data or design information is available for the venting and aerosol scrubbing system or for the purging system. Safe operation of these systems are absolutely crucial to mitigating the effects of a core meltdown accident. The scrubbing system must operate under extraordinary temperatures (800 to 1000°F, Fig. 25 of TMBDB report) and perhaps in the presence of hydrogen. In addition, it may be called upon to remove tons of radioactive aerosols. A tentative design should be made available for these systems and a detailed study performed to assure that the designs are feasible and reliable.

Status

A detailed description of the cleanup system was given by the Project at the last technical exchange meeting (9/15/82). Several questions regarding this system remain unresolved. The most serious are related to the scaling assumptions made in extrapolating aqueous scrubbing system experimental data to the full sized system. Another concern is the potential for aerosol plugging in the pipeline which carries aerosols from the RCB to the cleanup system. As already mentioned, there is also some concern about the possible introduction of elemental sodium into the scrubbing system with the ensuing formation of hydrogen.

3. CACECO ANALYSIS

The CACECO code was obtained by BNL in order to perform "over-check" calculations for the cases presented in the TMBDB report as well as to perform sensitivity studies. The sensitivity studies were intended to determine the effects due to correcting what were thought to be weak or non-conservative assumptions in the CACECO code modeling of CRBR.

The baseline case was run as an over-check calculation and selected graphs for that case, although they are present in much greater detail in the TMBDB report, are presented here in order to facilitate comparison with the other cases. The results obtained on the BNL computing system agree exactly with the Projects results except perhaps in round-off. The results for the baseline case are shown in graphical form under the case number CRBR01 (pp. A1-30 to A1-35). Table A1-3 contains a summary of all the cases that are discussed in this section.

3.1 RCB Concrete Floor Dehydration

As discussed in Section 2 Item 2.6.4, the Project in its assessment of the baseline case failed to direct the gases and vapors which would be generated by heatup of the unlined RCB concrete structures into the containment. Although they are correct in maintaining that the rate at which this concrete would be degassed must be lower than for the cavity and pipeway concrete, it is by no means reasonable to assume that the RCB concrete does not dehydrate as a result of increased temperature. A case was run, therefore, to determine the effect of venting these gases into the RCB. The input deck for the baseline case was used except that the concrete degassing destinations (2600 & 2800 series cards) were changed so that these gases would be directed to containment. The results of this case are shown in the graphs labeled CRBR02 (pp. A1-36 to A1-41).

The Project has run a case in which they directed part of the concrete degassing products into containment and their results for the baseline case configuration indicate that containment venting must occur a couple of hours earlier than for the TMBDB baseline case. The case that was run at BNL was vented at the same time as the TMBDB baseline case (36 hours) so no comparison is possible on the relative effect of earlier vent-down time. However, a comparison of the time at which venting was initiated relative to the same H_2 concentrations in each case, would indicate that venting should have occurred for the present case at 34 hours as opposed to 36 hours in the baseline case. This seems to agree well with the information which the Project supplied in personal communication.

Comparison of the plots for the baseline case (CRBR01) and case CRBR02 reveals that the peak temperature and pressure are higher for the latter case. The peak temperature difference is more radical than is seen in the plots. Because the containment was not vented until 2 hours after hydrogen began to reaccumulate in containment, by the time purging was commenced (3 hours after vent-down), an H_2 concentration of 10.3% had accumulated in the RCB at the point of purging. Since the containment was vented, the pressure did not go very high, but the temperature during the burn went up to 1235°F. This temperature excursion could have been prevented by venting at 34 hours and purging at 37 hours. The net difference between these cases is that containment must be vented 2 hours earlier. The main point to be taken from these comparisons is that the accident scenario may be extremely sensitive to operator action. Specifically, timing of the venting and purging operations and, therefore, the instrumentation upon which these operations are based, is extremely critical.

3.2 Insulating Effect of Aerosol Plateout

In Section 2, Item 2.1.1 is discussed the fact that the TMBDB analysis did not take into consideration the plate-out of aerosols on containment building structures. A layer of this loosely consolidated material would act as an insulating layer with a tendency to prevent heat from being absorbed into the passive heat sinks during the initial phases of the accident, and would strongly inhibit the efficiency of the annulus cooling system during the period subsequent to containment venting. To assess the importance of this oversight, an analysis was made of a case (CRBR03) in which all the heat absorbing surfaces within the RCB were supplied with an intervening layer of aerosol 1.8 inches in thickness. An independent assessment[6] has estimated that, depending on the porosity, the layer thickness will be much greater than 1.0 cm. This study estimated a plate-out on the order of 2.7 gm/cm^2. The present case incorporated a 1.8 in. thick layer with a porosity of 0.6 which translates to a plate-out loading of 3.9 gm/cm^2 or about 50% higher than the recommended number. However, it is believed that the porosity may be as high as 95% which would result in greater thermal resistance and a tendency to cancel the effect of the larger plate-out loading used in the present analysis.

The Project's baseline case input was used for the present case with some minor changes. The changes consisted of adding a new material to the list of materials used in CACECO along with the appropriate thermo-physical properties and adding a layer of this material between the RCB atmosphere and the passive heat sinks in containment.

Graphical elaboration of this case appears under the case identifier, CRBR03 (pp. A1-42 to A1-47). The differences between this case and the baseline case (CRBR01) are quite apparent. At 36 hours, when containment venting is initiated, the containment temperature is 980°F and the pressure is 25.8 psig as opposed to 618°F and 13.1 psig in the baseline case. The Project ran a similar case in response to an earlier articulation of this issue. Their results, which were presented in a very abbreviated form at the first technical exchange meeting, indicated a temperature of 696°F and a peak pressure of 16.0 psig at 36 hours. The source of the apparent discrepancy is not clear. They may have used a smaller layer thickness (on the order of 1 inch), but they also used a significantly larger porosity (95%) which would tend to increase the insulating effect. In addition, the Project's case was run out only to the venting time (36 hours). Unfortunately, some of the most radical divergences are seen after venting and purging have been effected. Returning briefly, however, to the pre-venting period and assuming that the aerosol plate-out effect is, indeed, as significant as is indicated by the present case, the appropriate criterion for venting may not be hydrogen concentration, but rather, it may be determined by containment pressure. As it happens, the pressure approaches to within 5 psi of the estimated containment failure pressure (30 psig) at about 36 hours. Since the design pressure is 10 psig, venting on the basis of over-pressure should, perhaps, be initiated at a pressure not much in excess of 1.5 times the design pressure, or about 15 psig. On that basis, containment venting at 22 hours might be discrete.

Subsequent to venting and prior to purging (39 hours), partial flashing of the sodium pool quickly flushes large quantities of hydrogen from the cavity and pipeway cells into the RCB. There it is burned until the O_2 content reaches 8% at which time it accumulates and burns rapidly when oxygen re-enters the RCB during the purge phase. The effect of H_2 and sodium burning in containment just after purging, combine to yield the highest RCB temperature, 2190°F, at about 40 hours. However, after about 52 hours, the containment atmospheric temperature never falls below the pool temperature (the sodium boiling temperature, 1620°F) and probably averages about 1850°F. This is an extremely high temperature, in fact, it is well above the melting temperature of NaOH (605°F) which is predicted to be one of the major constituants of the aerosol. Another constituant, Na_2O_2, decomposes to form Na_2O at about 860°F. The final important constituant, Na_2O, sublimates at 2327°F, and would remain solid at the peak temperatures predicted here. It is difficult to predict what would occur in the RCB under these conditions. Reference to the wall temperature of the containment shell (p. A1-46) shows that the surface temperature of the aerosol layer reaches the melting temperature of NaOH sometime between 10 and 50 hours. The detailed print shows that this occurs at 30 hours. What occurs subsequently is determined by the actual constituents of the aerosol at each location. The code predicts that, for this case, NaOH is the principle constituant of the aerosol. For this case, all of the water vapor generated in the concrete structures in the RCB was directed into the RCB atmosphere rather than into a "drain" as was done in the baseline case (see

Section 2 Item 2.6.4). This water is automatically reacted first with whatever Na_2O is present in containment to form NaOH. In the absence of water vapor, Na_2O is the primary product of sodium combustion. There are two things which would, in reality, tend to limit the formation of NaOH on the containment shell. First, in the present study, a 1.8 inch layer of aerosols was plated out over all the structures in the RCB including the RCB floor which is the only concrete structure directly in contact with the RCB atmosphere. In reality, the containment floor may be expected to have a layer of aerosol at least several feet thick due to gravitational settling of aerosols This layer would tend to insulate the RCB floor much more efficiently than would be the case for vertical surfaces, and would, therefore, tend to limit the only effective source of water vapor. With the sole source of water vapor thus inhibited, the primary constituant of the RCB aerosols above the floor would be expected to be the oxide, Na_2O. Secondly, there would be a strong positional dependence in terms of the formation of the hydroxide. Since the only source of water available to form the hydroxide is the RCB concrete floor, and since it will be covered with the thickest layer of combustion products, the water vapor degassed from the concrete must pass first through this layer. Thus, the layer of oxide on the floor will absorb most of the water vapor in forming the hydroxide. The result is that the aerosol layer on the floor would contain most or all of the low melting point sodium hydroxide, while the vertical surfaces, other surfaces above the RCB floor, and the airborne aerosols would be composed primarily of the high melting point, sodium oxide. The bottom line which emerges from this long discussion is that the aerosols which plate-out on the containment shell and other surfaces above the floor will very likely remain intact (not melt) except for possible periodic scaling-off and reforming processes.

The scenario which seems to unfold for the case in which aerosol plate-out on passive heat absorbing structures within the RCB is taken into consideration may be sufficiently more severe than that considered in the design and licensing studies to warrant some re-thinking in terms of equipment qualification and other TMBDB-related design features. For example, since it appears that the passive heat sinks in containment will not be absorbing as large a share of the energy as previously thought, the thermal loading on the cleanup system may be greater, so that it may become necessary to take these results into consideration in the design parameters for that system. Similar impact may be felt with regard to instrumentation, as well as in the area of operator action.

3.3 Enhanced Reactor Cavity Heat Transfer

In Item 2.1.5 of Section 2, concern is drawn to the effects that might occur due to an underestimate of heat transfer from the sodium pool in the reactor cavity to its immediate environs. These heat transfer processes, largely due to their complexity, are handled by the CACECO code in a rather simplistic fashion. In an attempt to access the uncertainty involved in these processes, a sensitivity study was performed (Appendix F.3 of the TMBDB report) by the Project in which the thermal conductivity of the perlite concrete which insulates the structural concrete in the reactor cavity was increased by 20%. This was done to simulate enhanced heat transfer rates. The results, as expected, showed a less severe condition during the pool boiling phase and slightly more severe condition during the initial hydrogen burn. This is due to the fact that with more heat transfer out of the cavity, the sodium boiloff rate is reduced, but the time required to bring the pool to boiling increases giving more time for the accumulation of hydrogen before the initial burn. In general, the differences were not significant.

The baseline case used a value for the thermal conductivity of perlite of 0.12 Btu/ft-hr-°F. This was increased to 0.144 Btu/ft-hr-°F in the Project's sensitivity case. In a literature search[14] it was established that the thermal conductivity of perlite could be as high as 0.2 Btu/ft-hr-°F. To determine both effects due to uncertainty in the heat transfer coefficient as well as in the heat transport properties of insulating concrete, a case was run wherein the thermal conductivity of the reactor cavity insulating concrete was increased to 0.2 Btu/ft-hr-°F. Results for that case are given under the case name CRBR04 (pp. A1-48 to A1-53).

Because of the sampling rate in the plotting subroutine, the peak pressure and temperature (at about 10 hours) during the hydrogen burn are not seen in either the baseline case (CRBR01) or the present case (CRBR04). In the baseline case, the peak H_2 burn conditions were 22.4 psig and 845°F while the corresponding conditions for the present case were 23.3 psig and 867°F. The effect of enhanced heat transport rates in the cavity was to increase the time at which hydrogen burning occurred by about 13 minutes, not long enough to produce a significant difference in the H_2 burn conditions in containment. Apparently, the decay heat and exothermal chemical reaction energies completely dominate the sodium pool heatup rate, so that even large uncertainties in the reactor cavity heat transport rates have only minute effects on the overall transient.

3.4 The Extreme Penetration Scenario

Based primary on the results of a CRBR site evacuation study,[15] the Project "invented" a case which, although phenomenologically improbable, would result in a venting time consistent with the estimated evacuation time. It was found that, if the sodium concrete penetration rate was set at 7 in/hr for a duration of 3 hours followed by a rate of 1 in/hr until sodium depletion occurred, a venting time of 10 hours would be required. This compares with an estimated site evacuation period of 9 hours. The case, which, of course, does not represent a real scenario has been referred to as the "Extreme Penetration (EP) Case." This case was run with the Project's input deck and the results are reproduced here as case CRBR05 (pp. A1-54 to A1-59).

There appears to be a number of unusual characteristics for this case. An investigation of the graphs shows oscillations in the pipeway cell temperature of a rather high amplitude, and the cell pressures indicate unrealistically high pressure differences between cells which should be very nearly at the same pressure. These are probably related to the vent cross sectional flow areas which were employed in order to achieve convergence in CACECO. The hydrogen burn occurs at 1.335 hours when the H_2 concentration was only 2.5% so that the pressure "spike" was limited to about 13.9 psig. The peak RCB temperatures were slightly over 1000°F for the EP case as compared to about 900°F for the baseline case. The sodium pool was boiled dry at about 50 hours compared to 133 hours for the baseline case.

In the baseline case, the containment was vented at the point when hydrogen just begins to re-accumulating in the RCB, so that the H_2 concentration does not exceed 6% when purging commences. In order to vent at 10 hours, the rule was apparently discarded for the EP case. When venting was initiated at about 10 hours, the H_2 concentration was already at 2.5% so that an H_2 burn (from 8.7% H_2 concentration) that takes place at 14.5 hours raises the containment temperature to 2020°F. Again, this spike does not appear on the graphs due to the low sampling rate.

An additional case was run using the Project's Extreme Penetration case to determine the effect of the insulating properties of plated aerosols. The EP input deck was altered in the same manner as was the baseline case by adding a layer of plated aerosols to all the heat absorbing structures within the RCB. This case is labeled CRBR06 and the graphs for this case appear on pages A1-60 to A1-65.

As with the baseline case, inclusion of an insulating aerosol layer produce significant changes in the accident scenario. Again, the venting decision must be predicated on the basis of containment pressure rather than H_2 concentration. In fact, if an overpressure criterion of 1.5 times the design pressure is used, venting would be necessary at about 2.5 hours rather than 10 hours. Delaying vent-down until the RCB pressure reached within 5 psi of the estimated containment failure pressure (30 psig) would require venting at 7 hours, while the highly riskly choice of venting precisely at the failure pressure would still require vent initiation at 8 hours. In any case, venting apparently would be required prior to the estimated time at which evacuation would be completed.

The environmental condition for the augmented EP case are slightly more severe than for the augmented (insulated layer) baseline case. Here the containment temperature subsequent to venting and purging averages about 2200°F with a peak RCB temperature of about 2750°F.

3.5 The "Realistic Upper Bound" Scenario

A review (see Ref. 4) of all the available data both from HEDL and SANDIA on the reaction of sodium with concrete has been completed. On the basis of that review, the NRC and its consultants have arrived at a sodium-concrete penetration rate which is thought to bound the observed experimental data. The NRC has recommended a "Realistic Upper Bound" (RUB) penetration rate of 7 in/hr for a duration of 1 hour followed by a rate of 1 in/hr until sodium depletion. The Project maintains that a RUB penetration rate of 7 in/hr for 20 minutes followed by 1 in/hr is closer to an upper bound. To our knowledge, a CACECO analysis has not previously been performed to determine the containment response for either of these RUB scenarios. Four cases were run, two for each of the two scenarios. Cases CRBR07 and CRBR08 correspond to the NRC's RUB case without and with aerosol plateout, respectively. Cases CRBR09 and CRBR10 treat the Project's RUB case without plateout and with plateout. All four of these cases were run using the Project supplied EP deck with the appropriate changes in each case. It should be emphasized that no alterations in these cases were made (compared to the EP case) in terms of attempting to match the rated cleanup system forced air flowrates by adjusting the cleanup system venting area. That is to say, the same schedule of vent area changes was used in these four cases as was used in the EP (extreme penetration case). The vent areas are normally adjusted at various times to approximate the rated cleanup system fan flows. These four cases differ from the EP case only in three input parameters. The sodium concrete penetration rates were changed as discussed above. For two of the cases (CRBR08 and CRBR10) the RCB passive heat absorbing structures are insulated with a layer of sodium combustion products. And finally, the vent and purge times are adjusted using a criterion whereby venting is initiated at the point in time when hydrogen begins to accumulate in the RCB (due to oxygen depletion). Purging is commenced in each case three hours after venting is initiated.

The results of these four cases are presented in graphical form under the case numbers CRBR07 (pp. A1-66 to A1-71), CRBR08 (pp. A1-72 to A1-77), CRBR09 (pp. A1-78 to A1-83), and CRBR10 (A1-84 to A1-89). Also, see Table A1-3 for a summary of all the cases performed in this analysis.

A comparison of the "Extreme Penetration Case" (EP), case CRBR05, with the two RUB cases, CRBR07 and CRBR09, reveals some quite unexpected results. Whereas the EP case achieves a higher peak temperature (2017°F), the two RUB cases appear to be more severe in almost every respect. The peak pressures prior to venting are higher, probably because venting was delayed two to three hours beyond that in the EP case (13 hrs for CRBR07 and 12 hrs for CRBR09 compared to 10 hrs for CRBR05). The average temperature subsequence to venting is also on the order of 200°F higher than for the EP case. These results seem incongruous, since the sodium concrete penetration rate for the EP case equals or exceeds that for the two RUB cases during the first three hours. However, the subsequent penetration rates are the same. The period of rapid penetration is terminated in all three cases long before venting occurs. The delay of venting in the RUB cases results in more energy stored in the pool at the time of venting so that larger quantities of sodium are "flashed" and burned in the RCB when venting is initiated, resulting in higher average temperatures during this period. If the RUB cases had been vented at 10 hours they would likely have been similar to the EP case. Apparently, the continuous 1 in/hr penetration rate at later times dominates in the long-term over the more rapid initial rates, so that there is no significant difference between the Extreme Penetration case and the Realistic Upper Bound cases in terms of containment response. Using a venting criterion based on a RCB pressure of 1.5 times design pressure, the three cases would be vented at 6 hrs for the EP case and between 9 to 10 hours for the two RUB cases. This criterion affords a better delineation between the severity of these case scenarios.

A comparison of the "Insulated Extreme Penetration" case (IEP), (CRBR06), with the two "Insulated Realistic Upper Bound" cases (IRUB), (CRBR08, CRBR10), yield similar conclusions. Based on the H_2 concentration criterion, venting was further delayed for the IRUB cases to 15 and 14 hours, respectively. The delay further accentuates the post-venting differences in containment temperature so that the average temperature is about 300°F higher than in the IEP case. If the pressure criterion described about was used rather than the H_2 concentration criterion, the two IRUB cases would be vented at 7 to 8 hours while the IEP case would require venting at 2.5 hours. Again, as in previous cases, the effect of an insulating layer on the passive heat sinks is seen to be highly significant. The insulating layer necessitates a venting decision based not on H_2 concentration, but rather on containment pressure.

Table Al-1 Table of liner vent pathways.

Time	Structure #	Identification	Destination H_2O	Destination CO
0.0	3	RCB floor drain	Drain	Drain
	5	Head access area walls	RCB	RCB
	8	Non-submerged cavity wall	Cell 105	Cell 105
	9	Submerged cavity wall	Cell 105	Cell 105
	10	Cavity floor	Cavity pool	Cavity pool
	11	Pipeway cell wall	Cell 105	Cell 105
	12	Pipeway floor	Cell 105	Cell 105
	13	Pipeway roof	Cell 105	Cell 105
	14	Cell 105 walls	Cell 105	Cell 105
	17	Pipeway wall	Cell 105	Cell 105
	18	Pipeway wall	Cell 105	Cell 105
	19	Pipeway wall	Cell 105	Cell 105
	20	RC lower submerged wall	Cell 105	Cell 105
30 hrs.	12		Pipeway pool	Pipeway cell pool
	13		Pipeway cells	Pipeway cell
	17		Drain	Drain
30.2 hrs.	17		Cell 105	Cell 105
35 hrs.	19		Pipeway cell	Pipeway cell

Table A1-1 (Cont.)

Time	Structure #	Identification	Destination H_2O	CO
40 hrs.	18		Pipeway cell	Pipeway cell
50 hrs.	20		Cavity pool	Cavity pool
70 hrs.	9		Cavity pool	Cavity pool
80 hrs.	8		Cavity atm.	Cavity atm.
90 hrs.	11		Cavity atm.	Cavity atm.

Table A1-2 Table of venting history.

Time	Venting Pathway	Flow Area (ft^2)
0.0 hrs	Cavity - Pipeway	.15
	Pipeway - RCB	.15
	RCB - Cell 105	-.10?
	RCB - Outside	Leakage
30 hrs	Cavity - Pipeway	.25
	Pipeway - RCB	.25
36 hrs	RCB - Outside	.365
40 hrs	RCB - Outside	.600
50 hrs	RCB - Outside	.700
	Cavity - Pipeway	10^{-6}
	Cavity - RCB	.25
55 hrs	RCB - Outside	0.800
70 hrs	RCB - Outside	0.900
80 hrs	RCB - Outside	1.000

Table A1-3 Summary of CACECO analysis.

		Prior to Venting							Subsequent to Venting		
Case Number	P_{MAX} (psig)	T_{MAX} (°F)	H_{MAX} (%)	P_{VENT} (psig)	T_{VENT} (°F)	H_{VENT} (%)	t_{VENT} (hr)	t_{PURGE} (hr)	T_{MAX} (°F)	T_{AVE} (°F)	H_{MAX} (%)
CRBR01	22.4	845	4.5	13.0	618	0.0	36.0	39.0	917	750	4.0
CRBR02	22.4	845	4.5	21.0	511	1.55	36.0	39.0	1235	800	6.1
CRBR03	25.8	976	4.5	25.8	976	0.0	36.0	39.0	2188	1850	5.2
CRBR04	23.3	867	4.6	13.2	536	0.0	36.0	39.0	853	700	4.0
CRBR05	18.6	704	2.6	18.6	704	2.6	10.0	13.5	1012	950	8.8
CRBR06	34.6	1153	2.6	34.6	1153	1.0	10.0	13.5	2754	2500	13.0
CRBR07	25.8	715	2.4	25.8	716	1.3	13.0	16.0	1329	1100	13.6
CRBR08	39.3	1502	1.9	39.3	1502	0.8	15.0	18.0	3034	2800	9.4
CRBR09	25.8	934	5.1	24.8	685	0.1	12.0	15.0	1318	1100	14.1
CRBR10	40.0	1438	5.0	37.4	1361	0.0	14.0	17.0	3139	2800	8.6

P_{MAX} - RCB Maximum Pressure

T_{MAX} - RCB Maximum Temperature

T_{AVE} - RCB Average Temperature

H_{MAX} - RCB Maximum H_2 Concentration

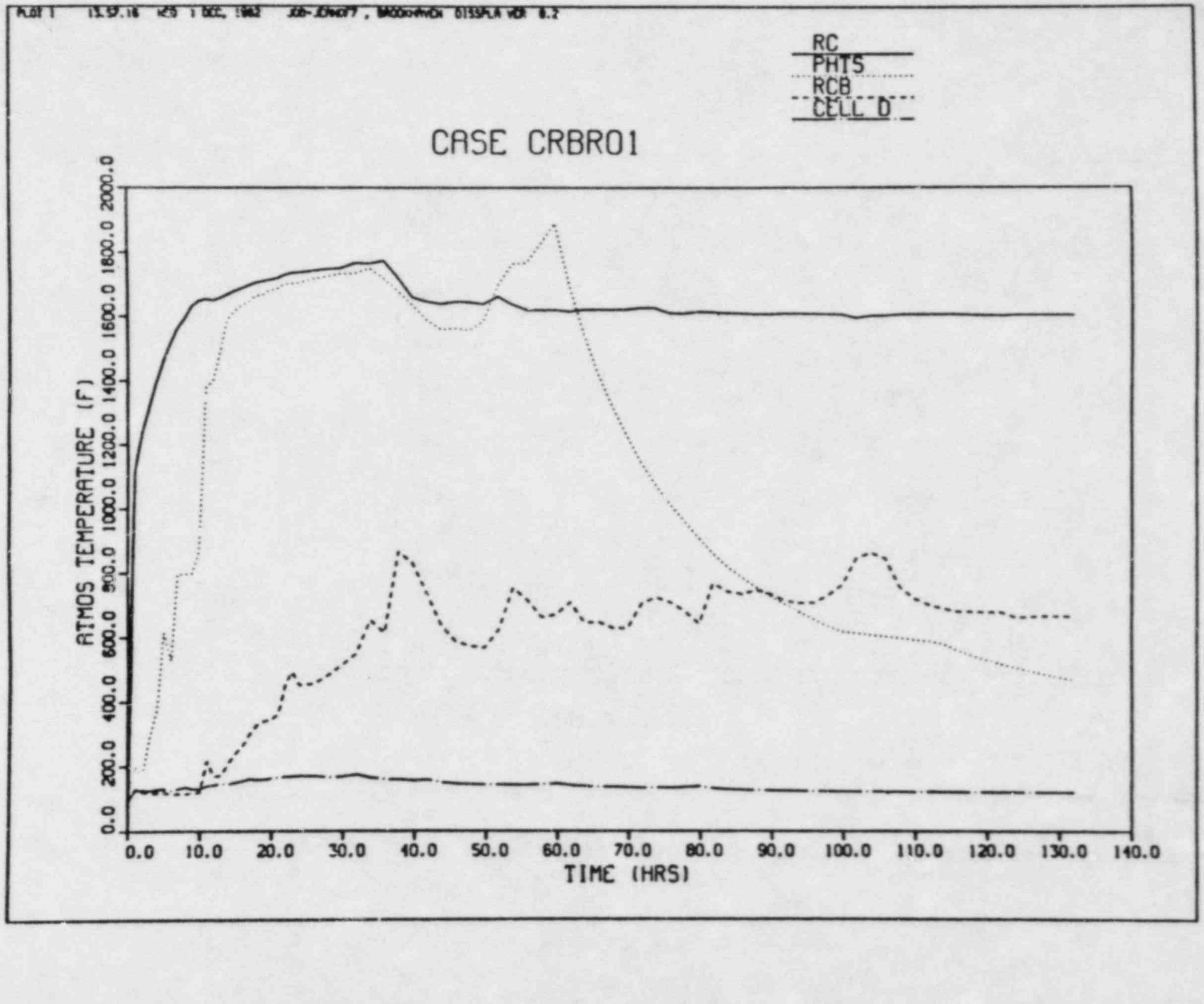
t_{VENT} - RCB Vent Time

t_{PURGE} - RCB Purge Time

P_{VENT} - RCB Pressure at Vent Time

T_{VENT} - RCB Temperature at Vent Time

H_{VENT} - RCB H_2 Concentration at Vent Time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RC
PHTS
RCB
CELL D
CASE CRBR01
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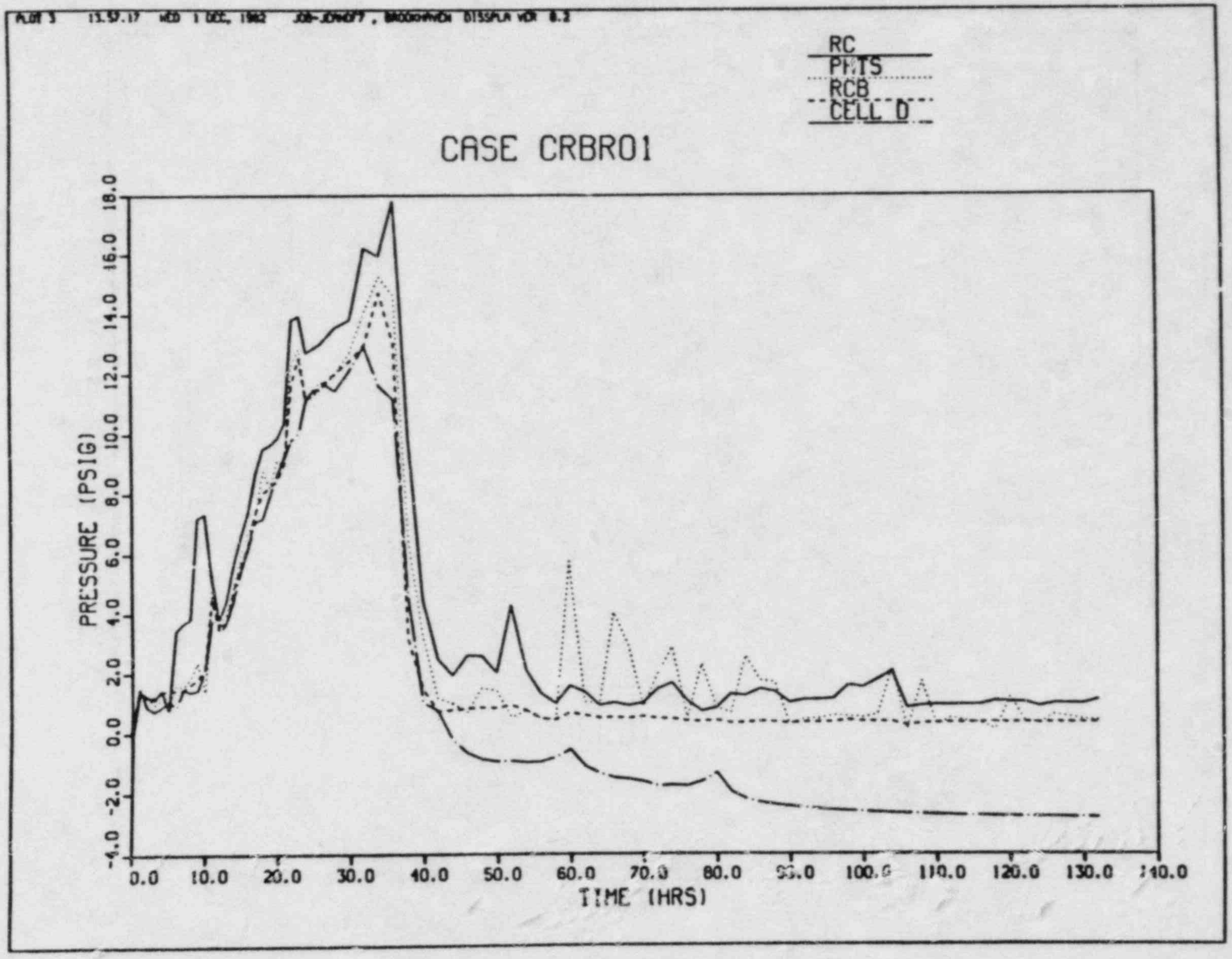
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
2000.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0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BROOKHAVEN DISSPLA VER 8.2
CASE CRBR01
RC
PHTS
RCB
CELL D
PRESSURE (PSIG)
18.0
16.0
14.0
12.0
10.0
8.0
6.0
4.0
2.0
0.0
-2.0
-4.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
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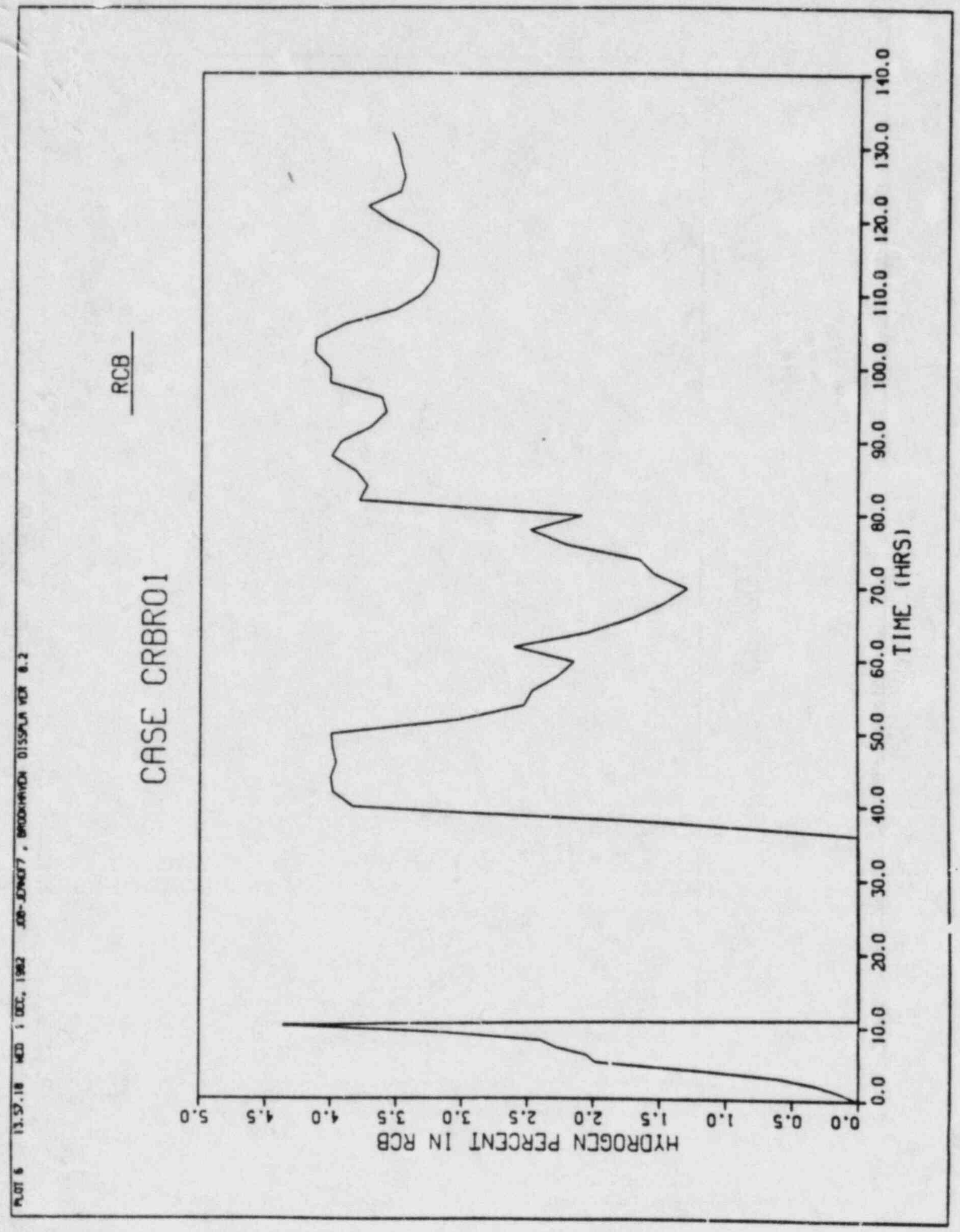
110.0
120.0
130.0
140.0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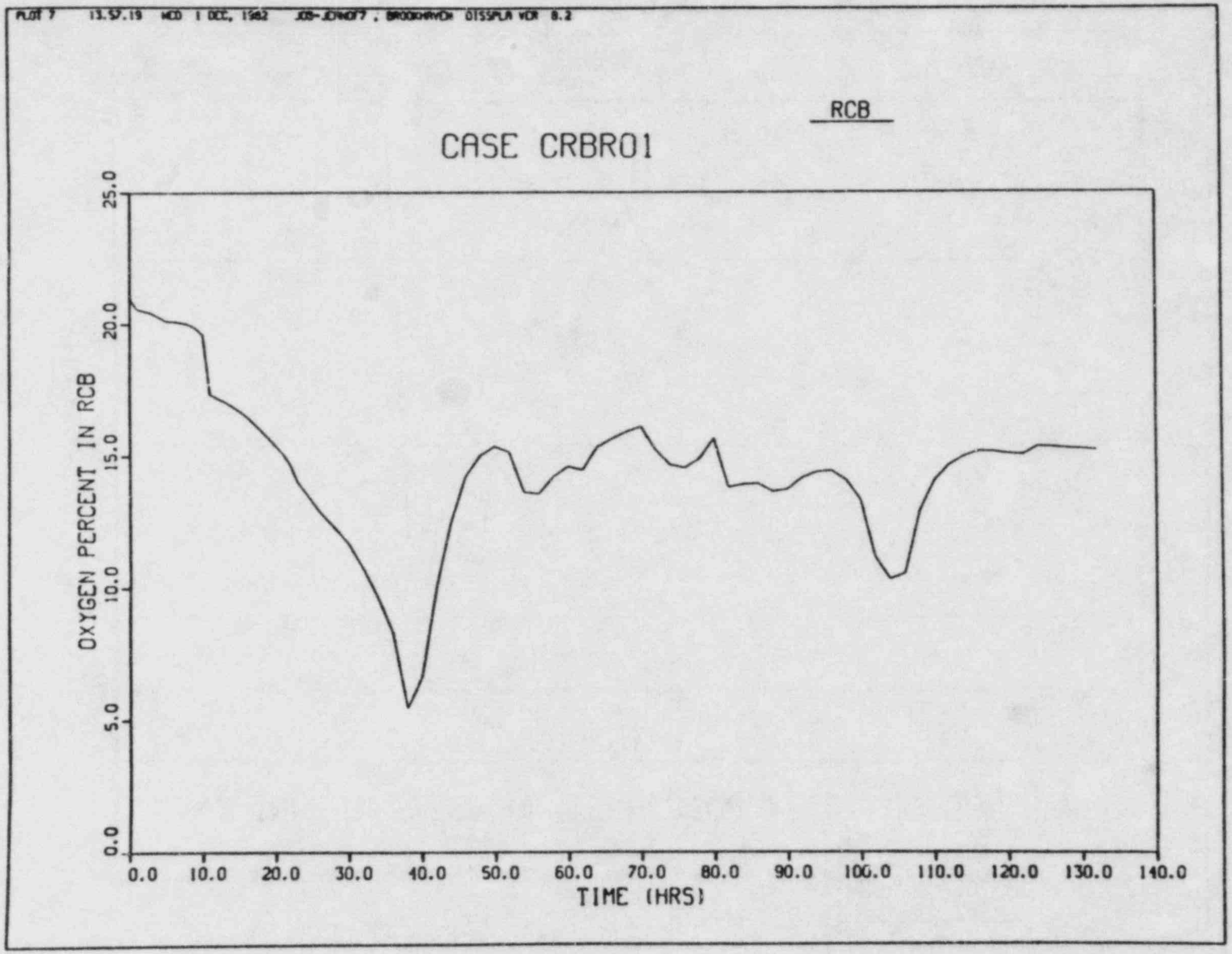
CASE CRBR01
RCB
HYDROGEN PERCENT IN RCB
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0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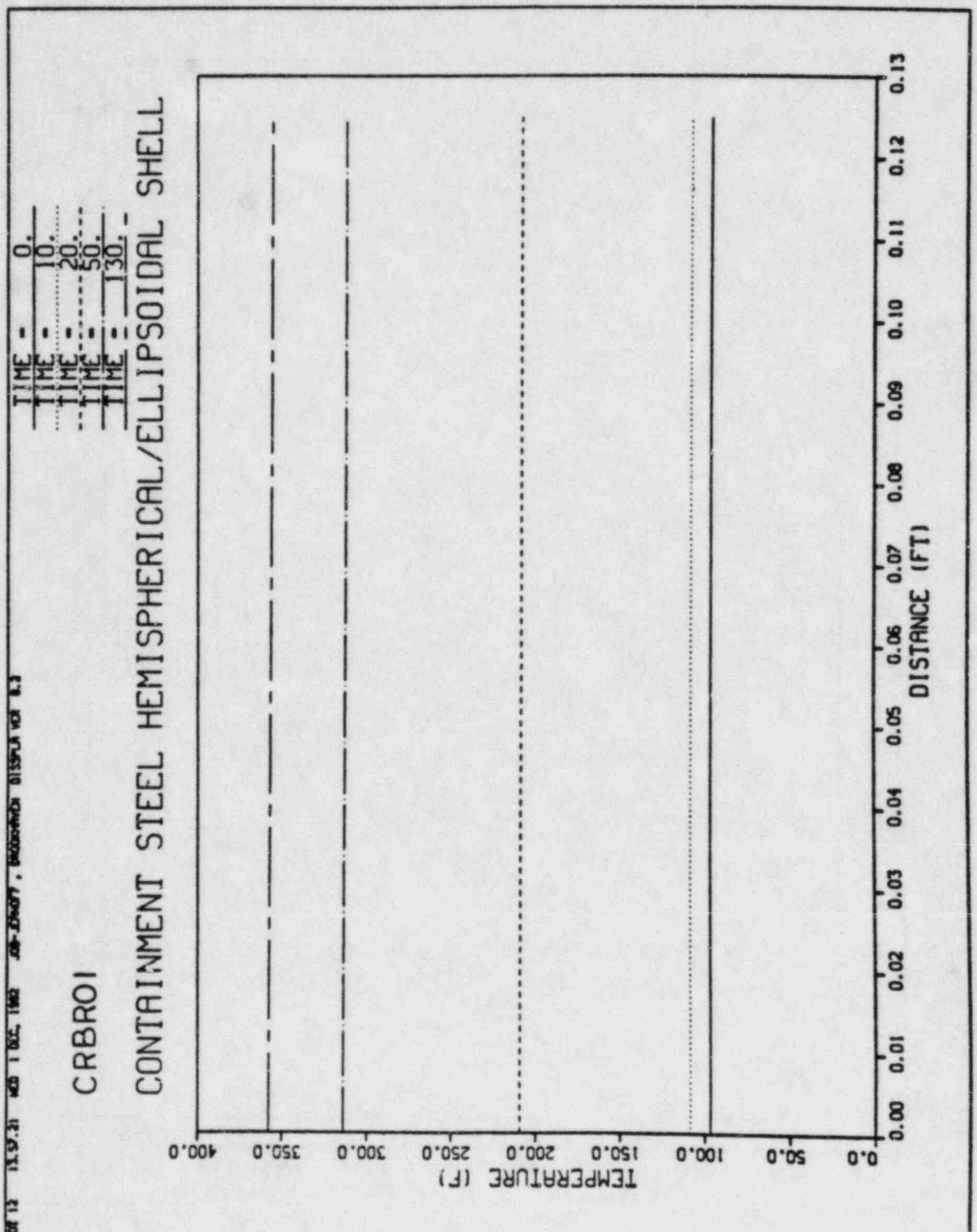
RCB
CASE CRBR01
OXYGEN PERCENT IN RCB
25.0
20.0
15.0
10.0
5.0
0.0
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0
TIME (HRS)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DISSPLA VER 8.2
CRBR01
CONTAINMENT STEEL HEMISPHERICAL/ELLIPSOIDAL SHELL
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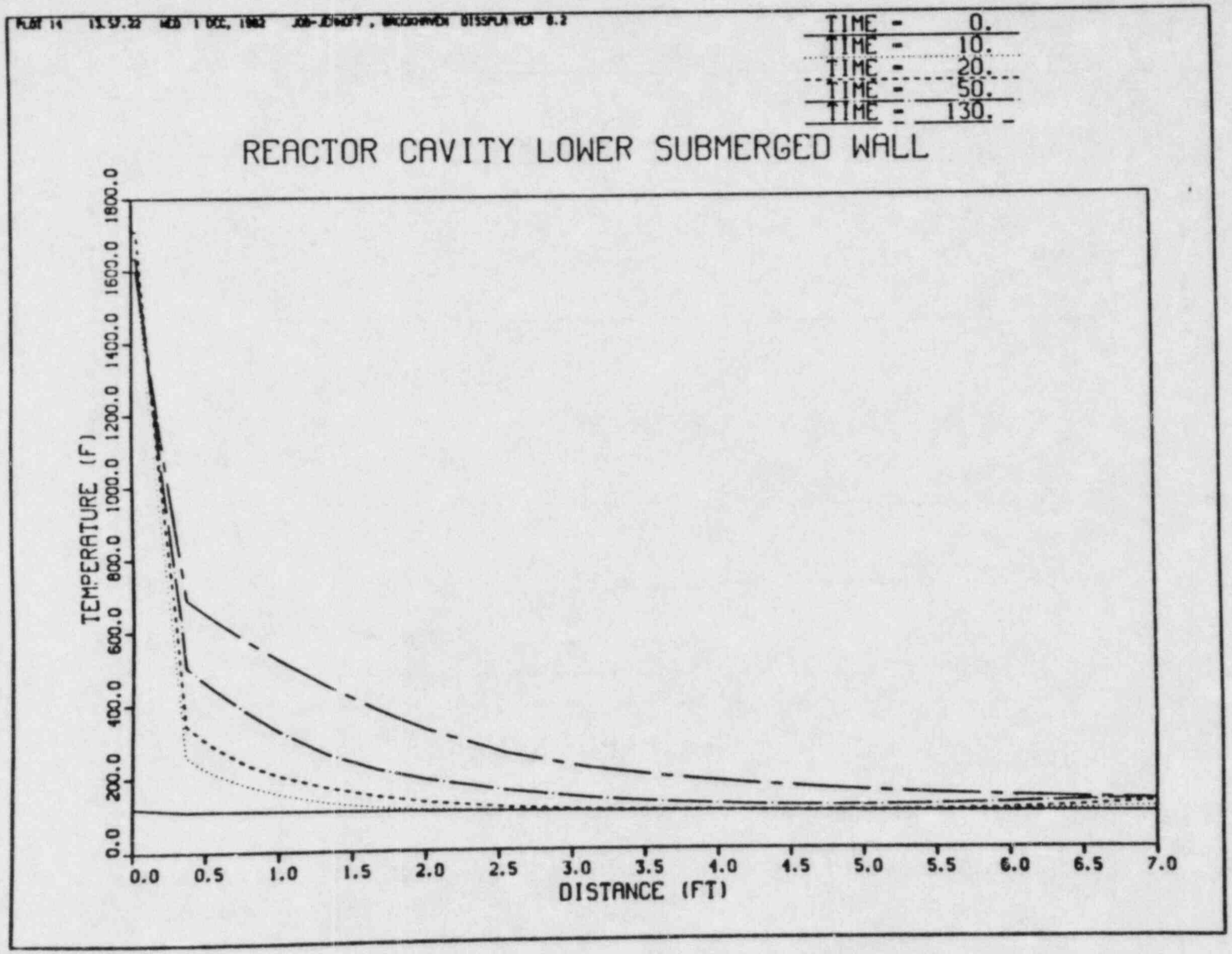
TIME = 10.
TIME = 20.
TIME = 50.
TIME = 130.
TEMPERATURE (F)
0.0
50.0
100.0
150.0
200.0
250.0
300.0
350.0
400.0
DISTANCE (FT)
0.00
0.01
0.02
0.03
0.04
0.05
0.06
0.07
0.08
0.09
0.10
0.11
0.12
0.13

PLOT 14 13.57.22 WED 1 DEC, 1982 JOB-[illegible] , BROOKHAVEN DISSPLA VER 8.2
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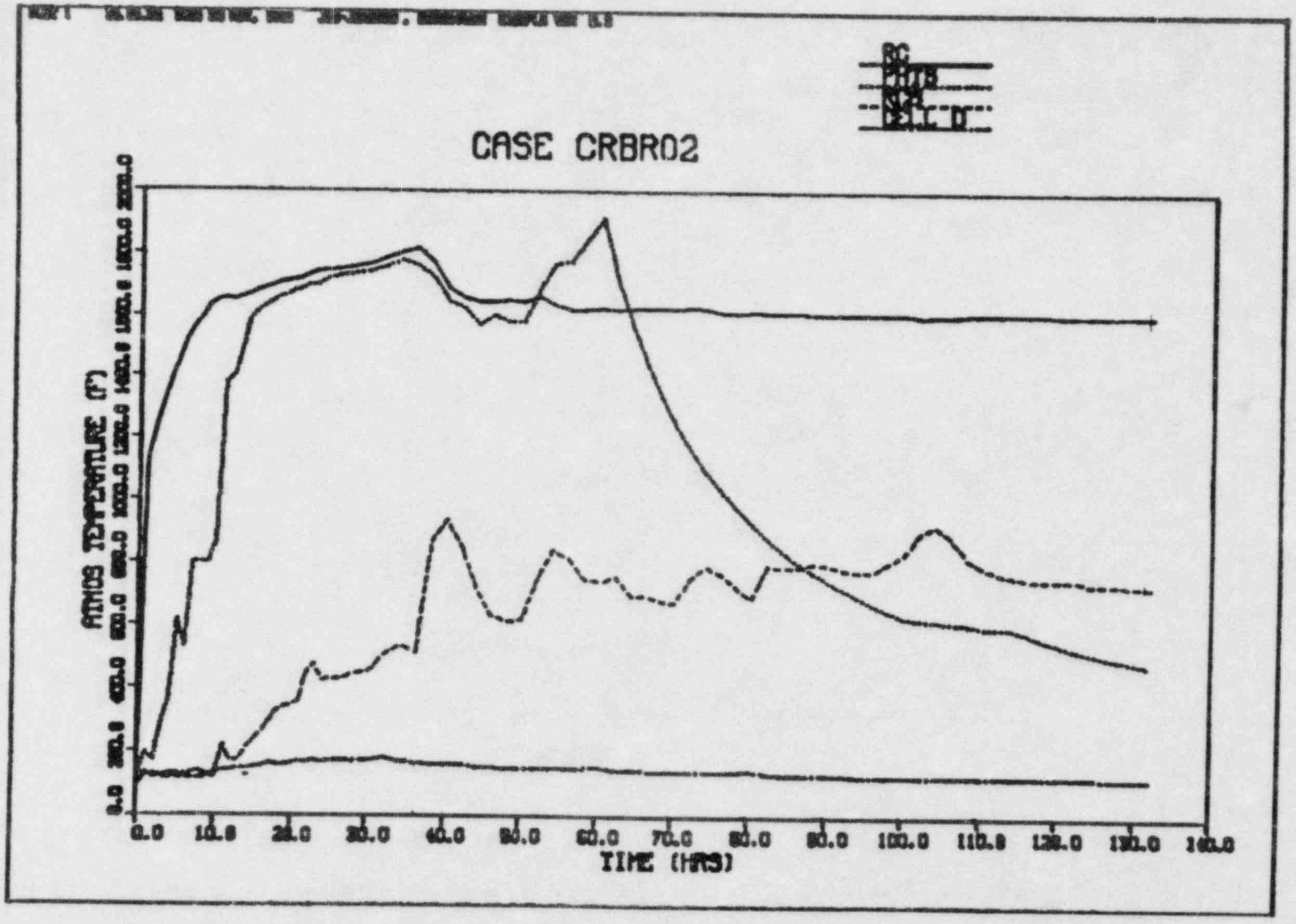
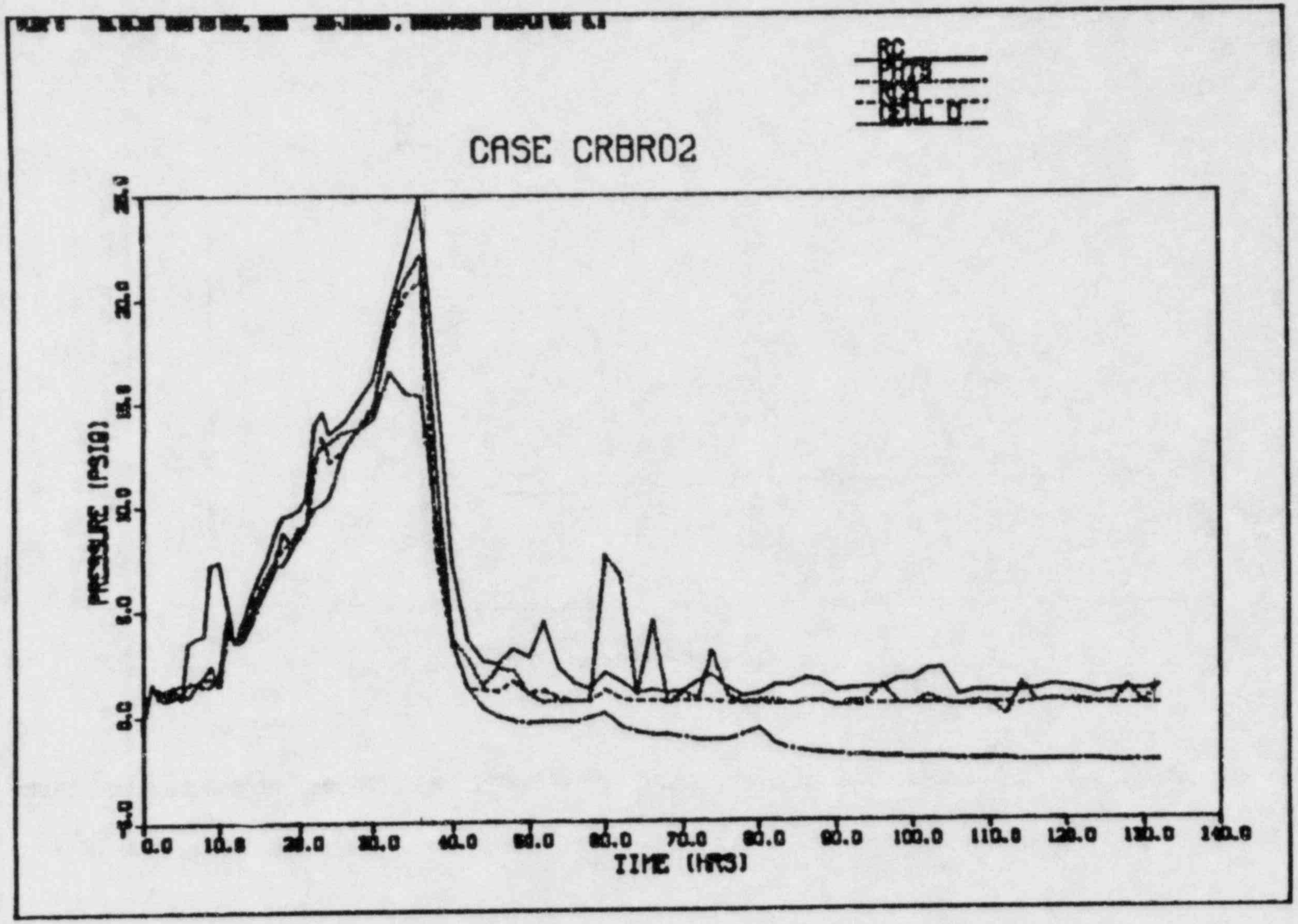
TIME - 10.
TIME - 20.
TIME - 50.
TIME - 130.
REACTOR CAVITY LOWER SUBMERGED WALL
TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
DISTANCE (FT)
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.0


[illegible]
CASE CRBR02
RC
PHTB
RCB
CELL D
ATMOS TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
2000.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0

CASE CRBR02
RC
PHTS
SGB
CELL D
PRESSURE (PSIG)
25.0
20.0
15.0
10.0
5.0
0.0
-5.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
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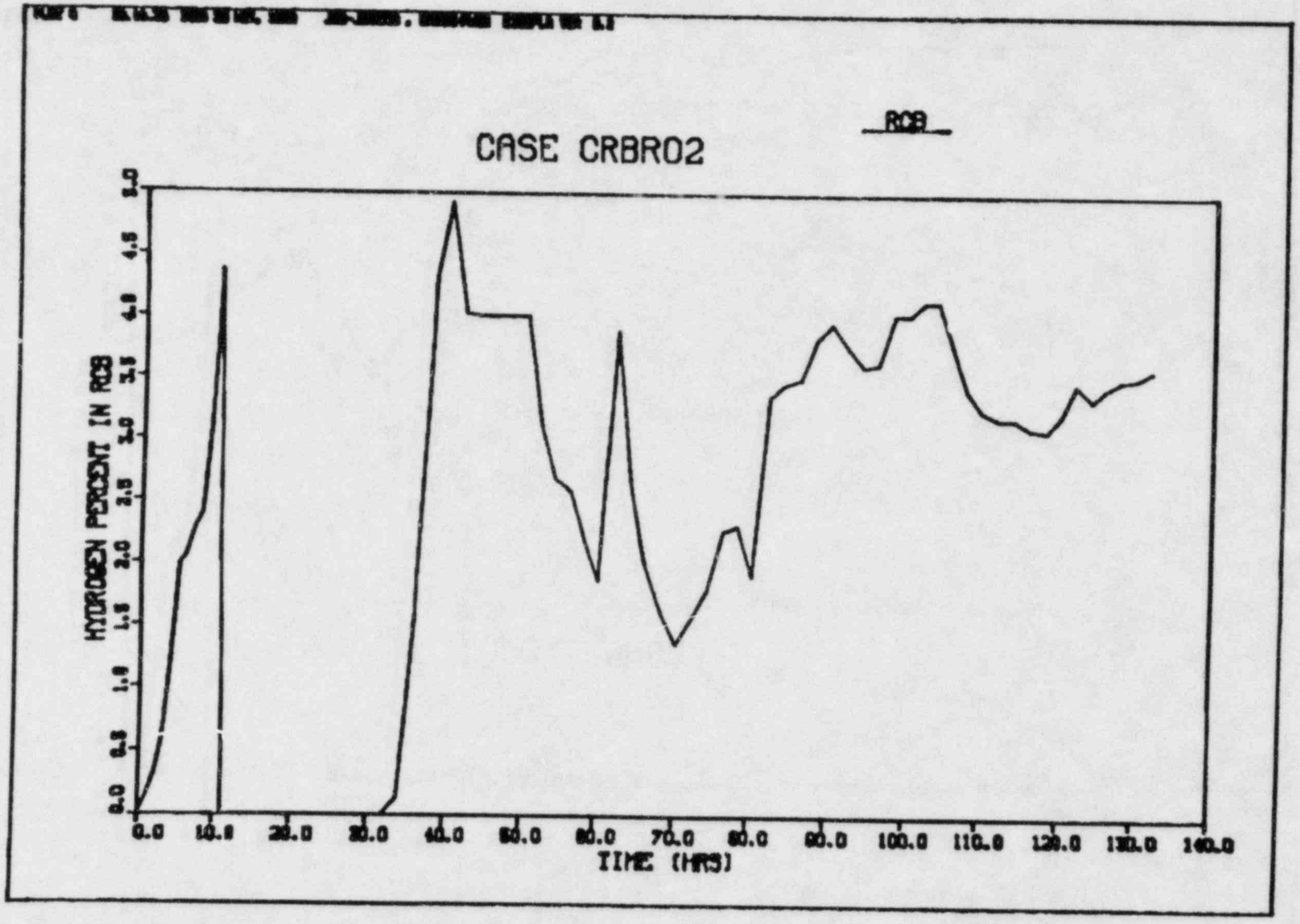
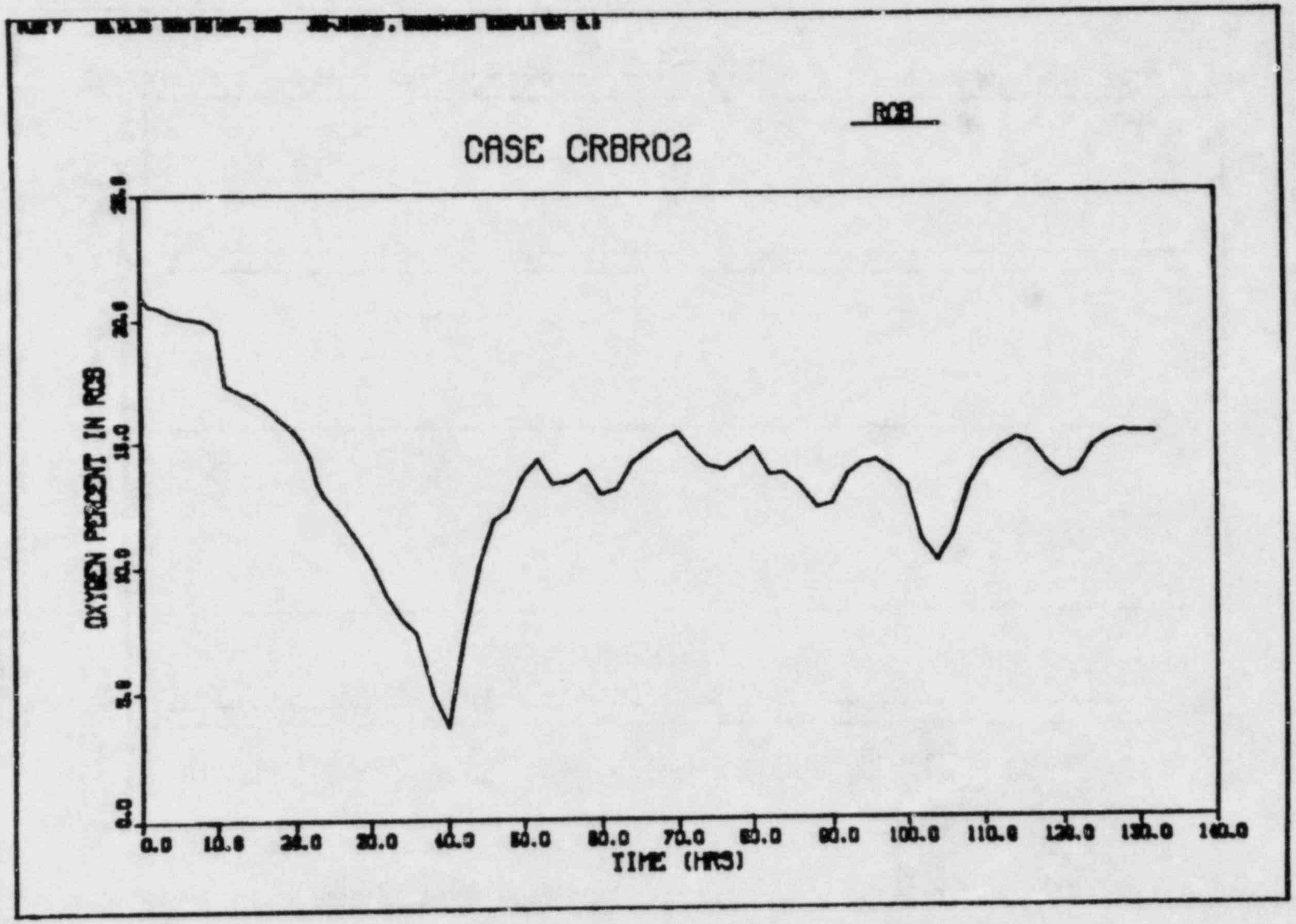
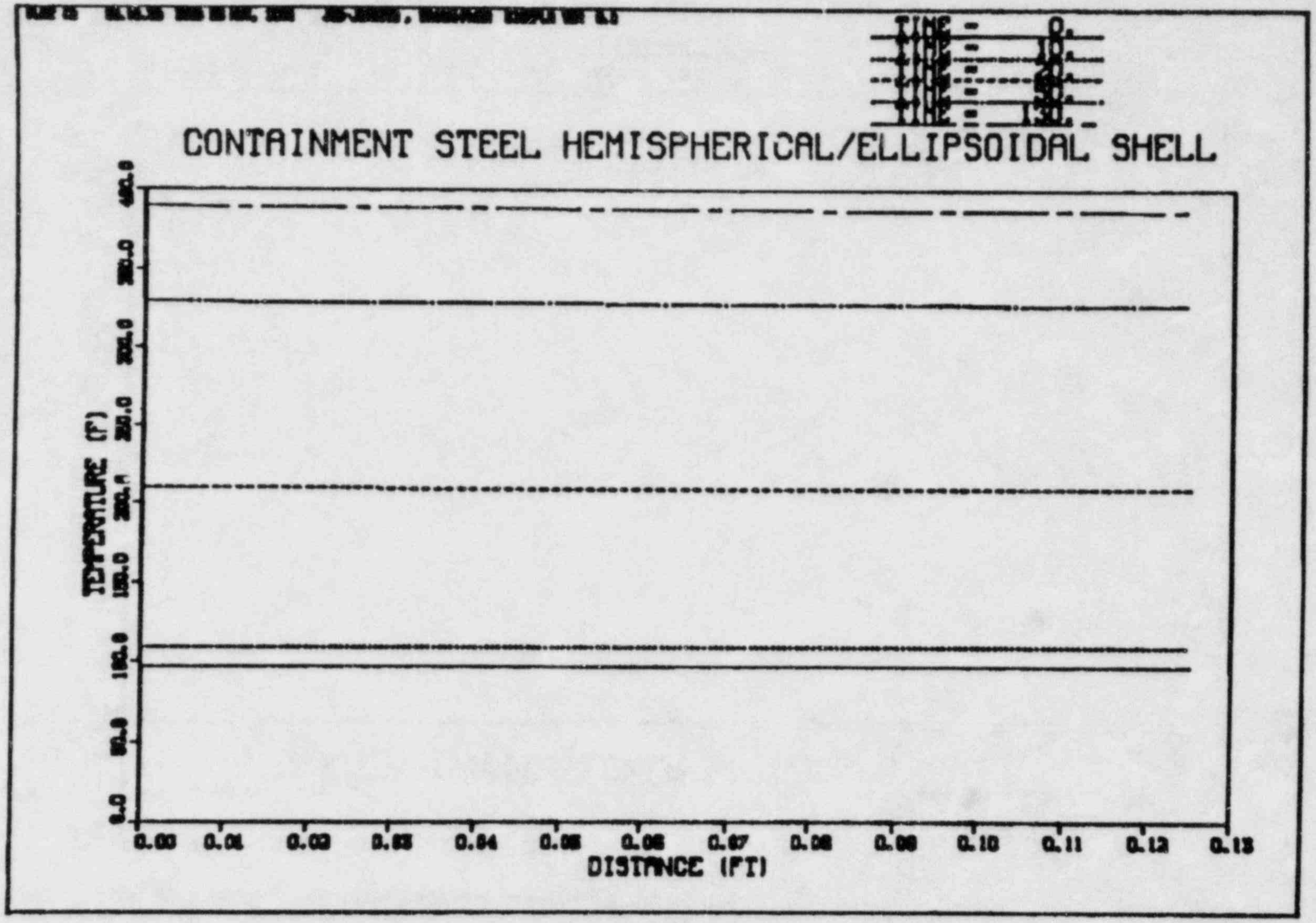
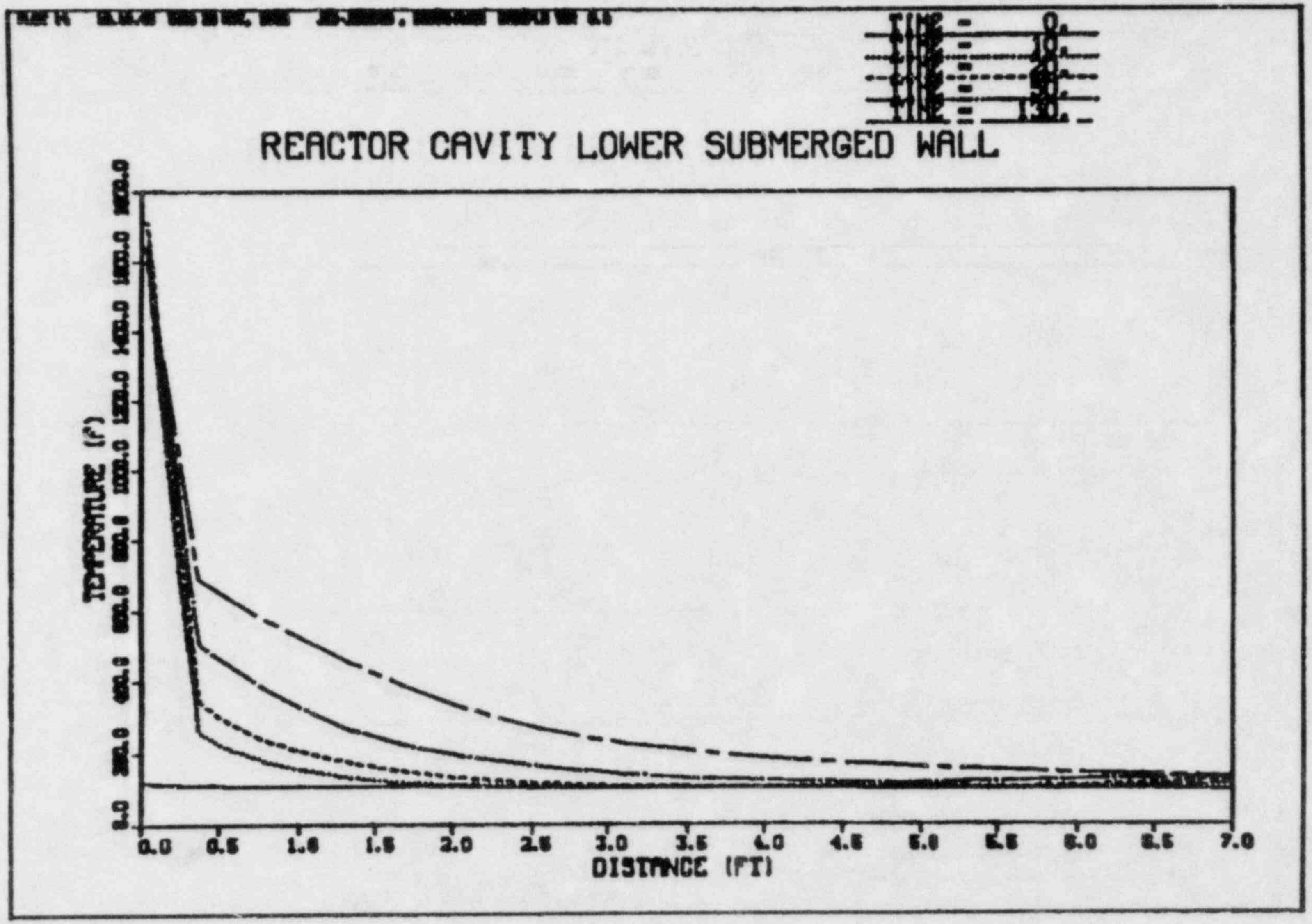
110.0
120.0
130.0
140.0

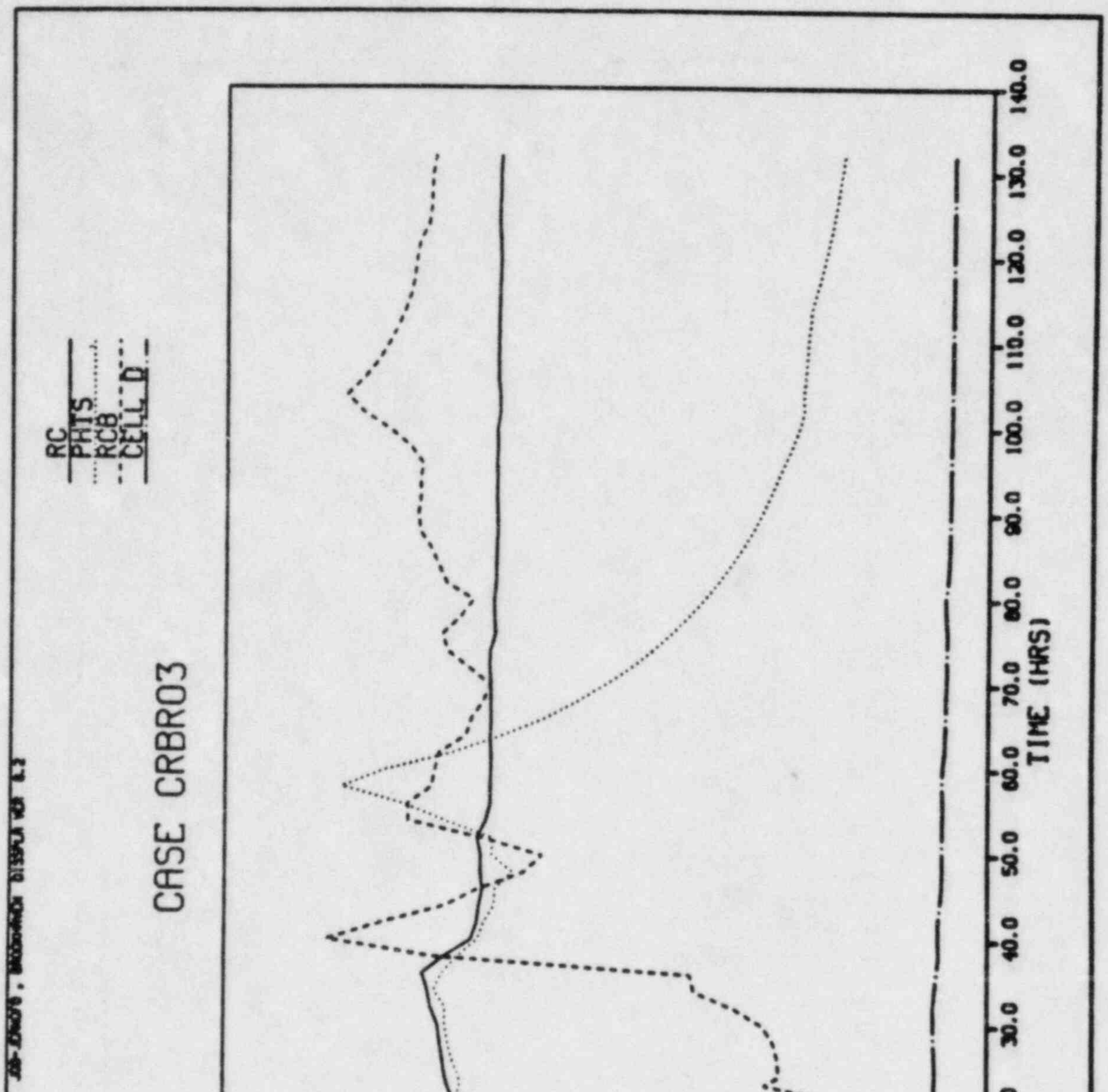

CASE CRBRO2
RCB
HYDROGEN PERCENT IN RCB
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0

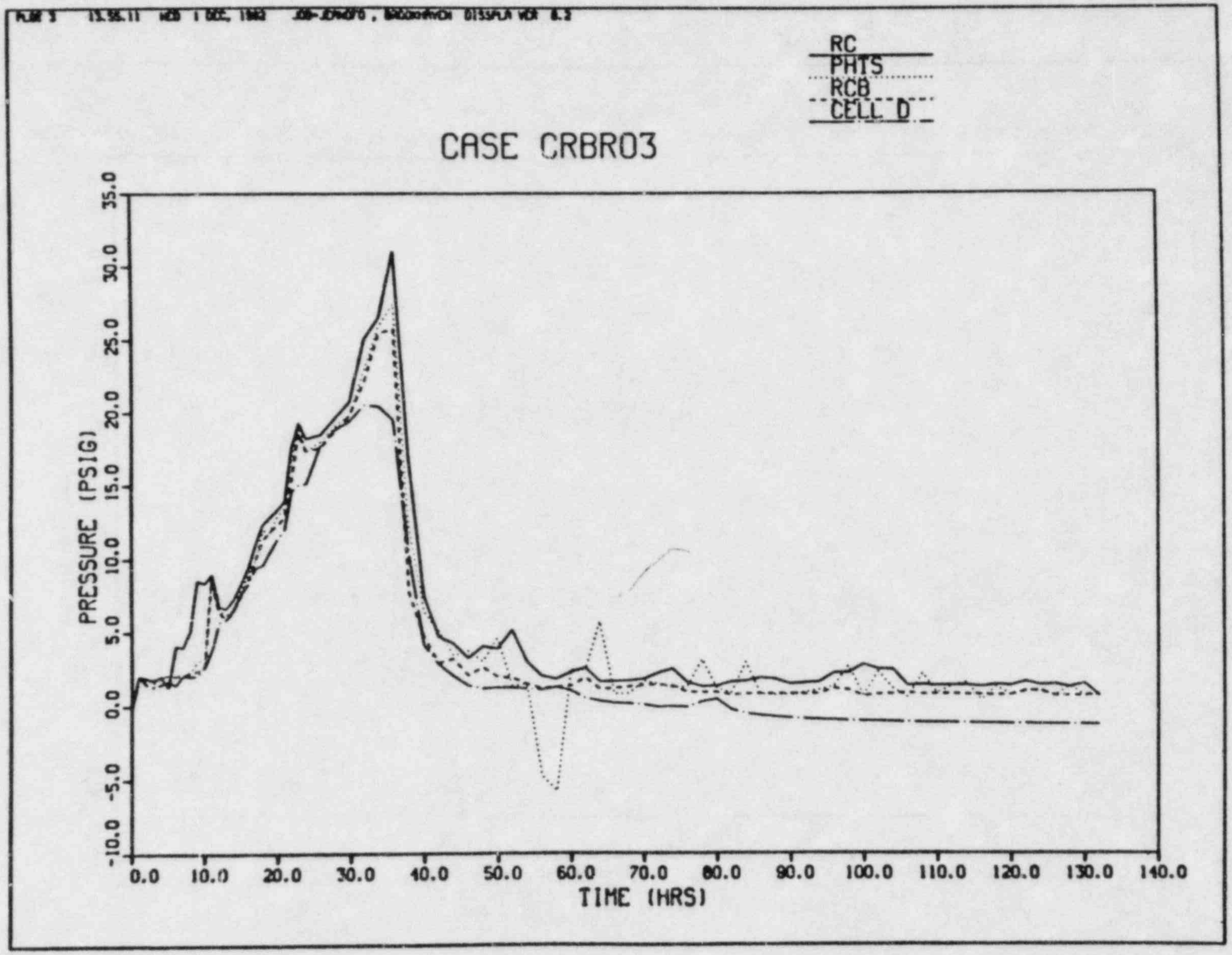

ROB
CASE CRBR02
OXYGEN PERCENT IN RCB
0.0
5.0
10.0
15.0
20.0
25.0
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0
TIME (HRS)

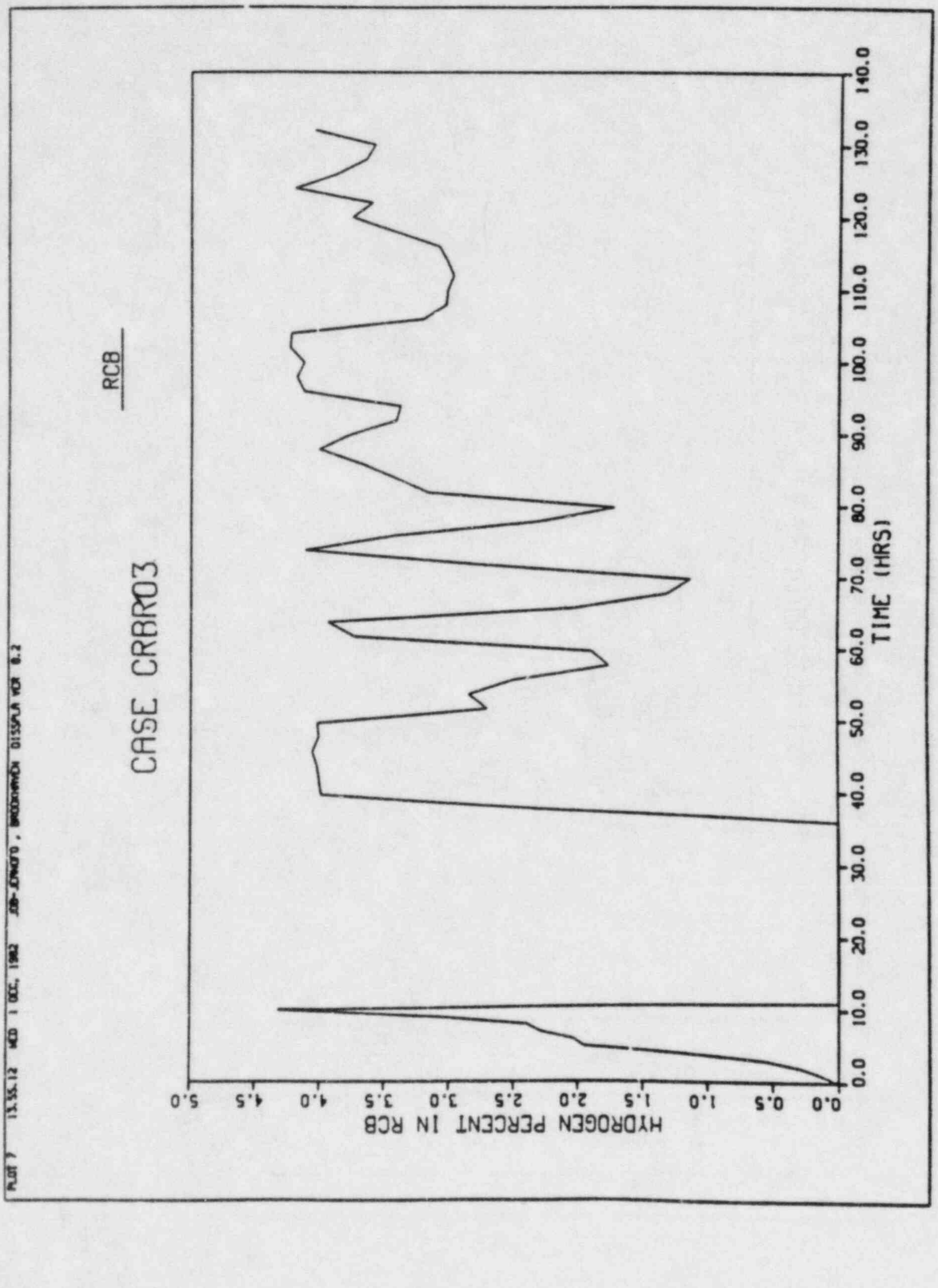


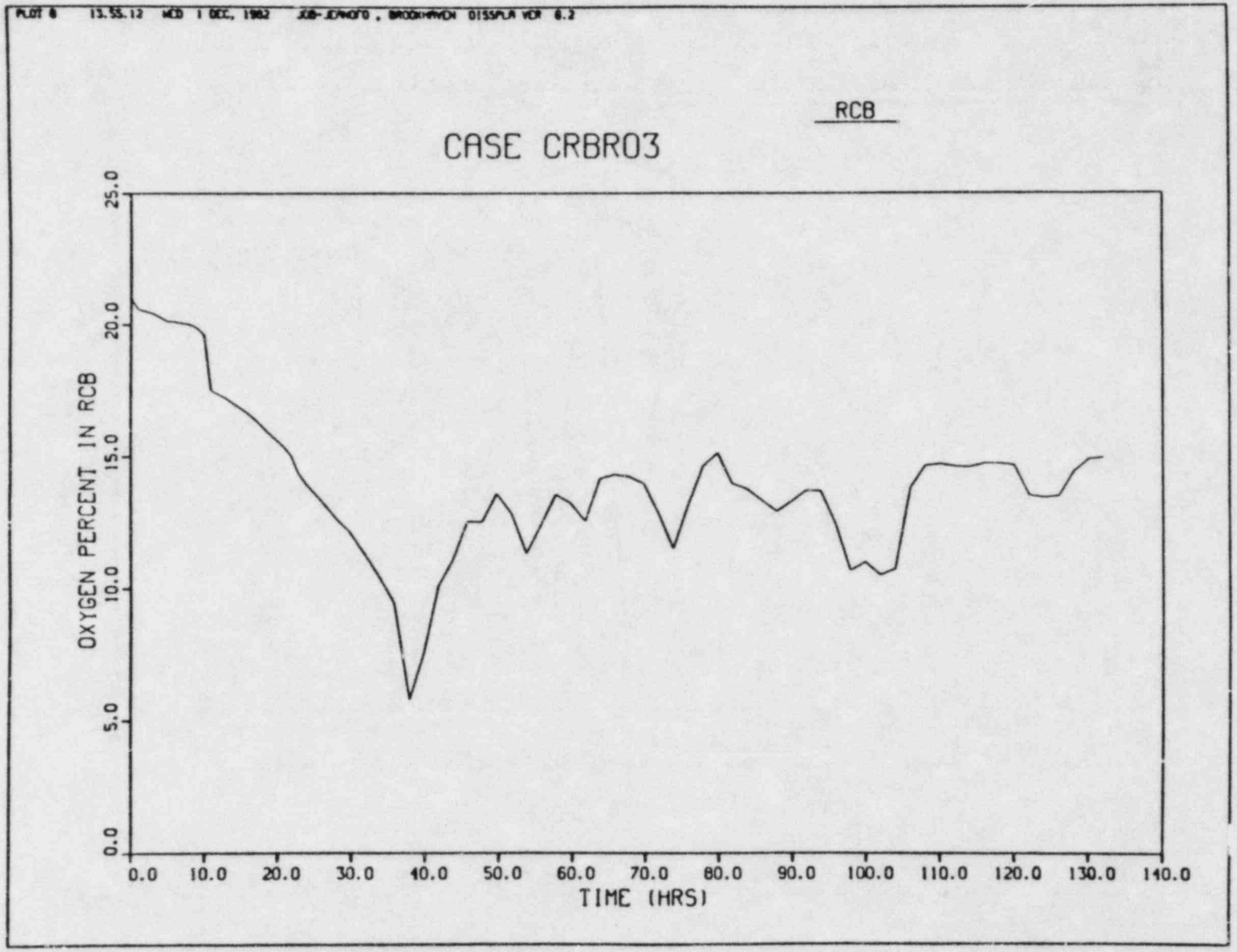
CONTAINMENT STEEL HEMISPHERICAL/ELLIPSOIDAL SHELL

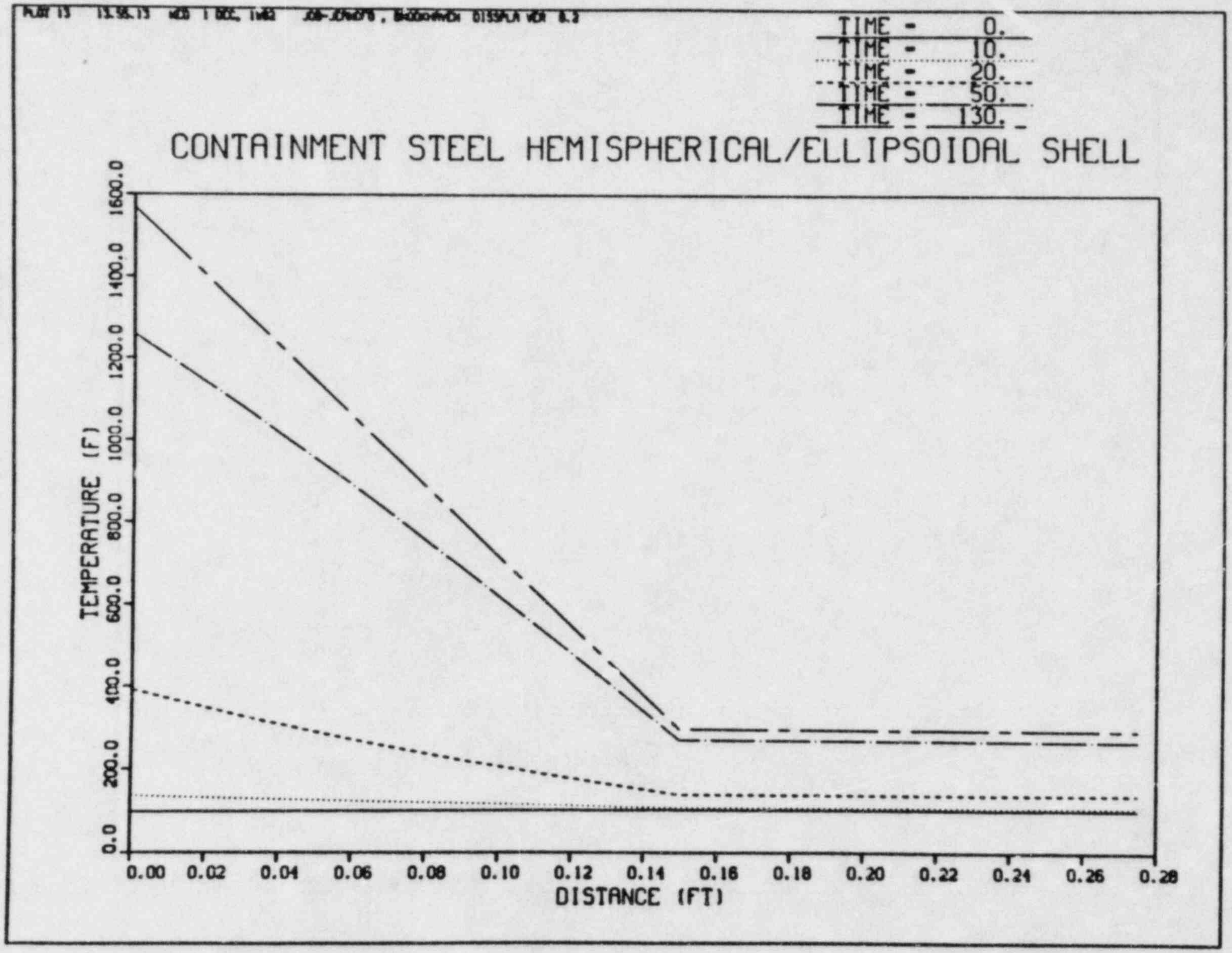

TIME = 0.
TIME = 10.
TIME = 20.
TIME = 50.
TIME = 130.
REACTOR CAVITY LOWER SUBMERGED WALL
TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
DISTANCE (FT)
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.0

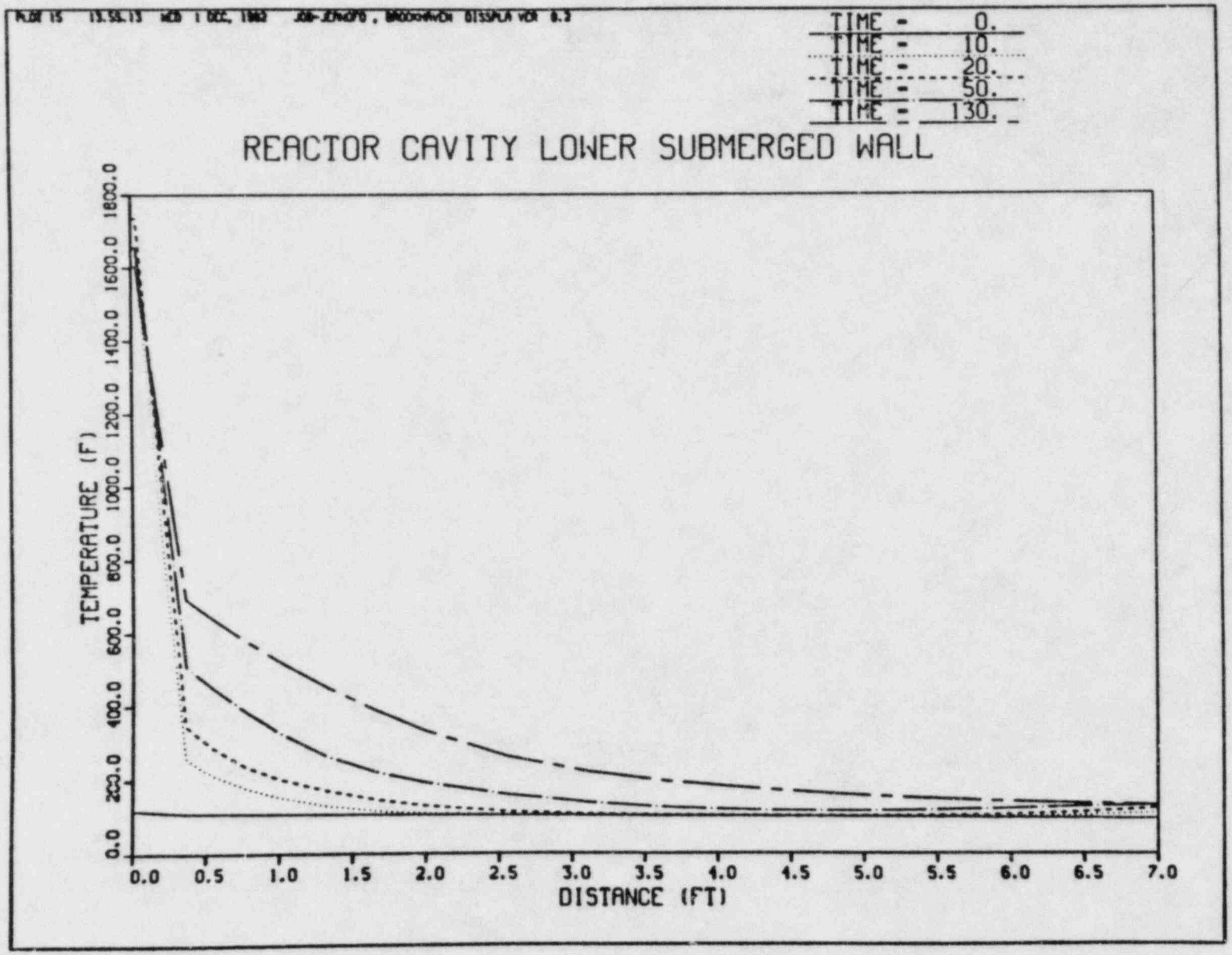

CASE CRBR03
RC
PHTS
RCB
CELL D
ATMOS TEMPERATURE (F)
0.0
500.0
1000.0
1500.0
2000.0
2500.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0


13.56.11 WED 1 DEC, 1982 DISSPLA VER 8.2
CASE CRBR03
RC
PHTS
RCB
CELL D
PRESSURE (PSIG)
-10.0
-5.0
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0


PLOT 7 13.55.12 WED 1 DEC, 1982 JOB-JONHOTO, BROOKHAVEN DISSPLA VER 8.2
CASE CRBR03
RCB
HYDROGEN PERCENT IN RCB
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0


PLOT 8 13.55.12 WED 1 DEC, 1982 JOB-JCANOTO , BROOKHAVEN DISSPLA VER 8.2
RCB
CASE CRBR03
OXYGEN PERCENT IN RCB
25.0
20.0
15.0
10.0
5.0
0.0
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0
TIME (HRS)

TIME - 0.
TIME - 10.
TIME - 20.
TIME - 50.
TIME - 130.
CONTAINMENT STEEL HEMISPHERICAL/ELLIPSOIDAL SHELL
TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
DISTANCE (FT)
0.00
0.02
0.04
0.06
0.08
0.10
0.12
0.14
0.16
0.18
0.20
0.22
0.24
0.26
0.28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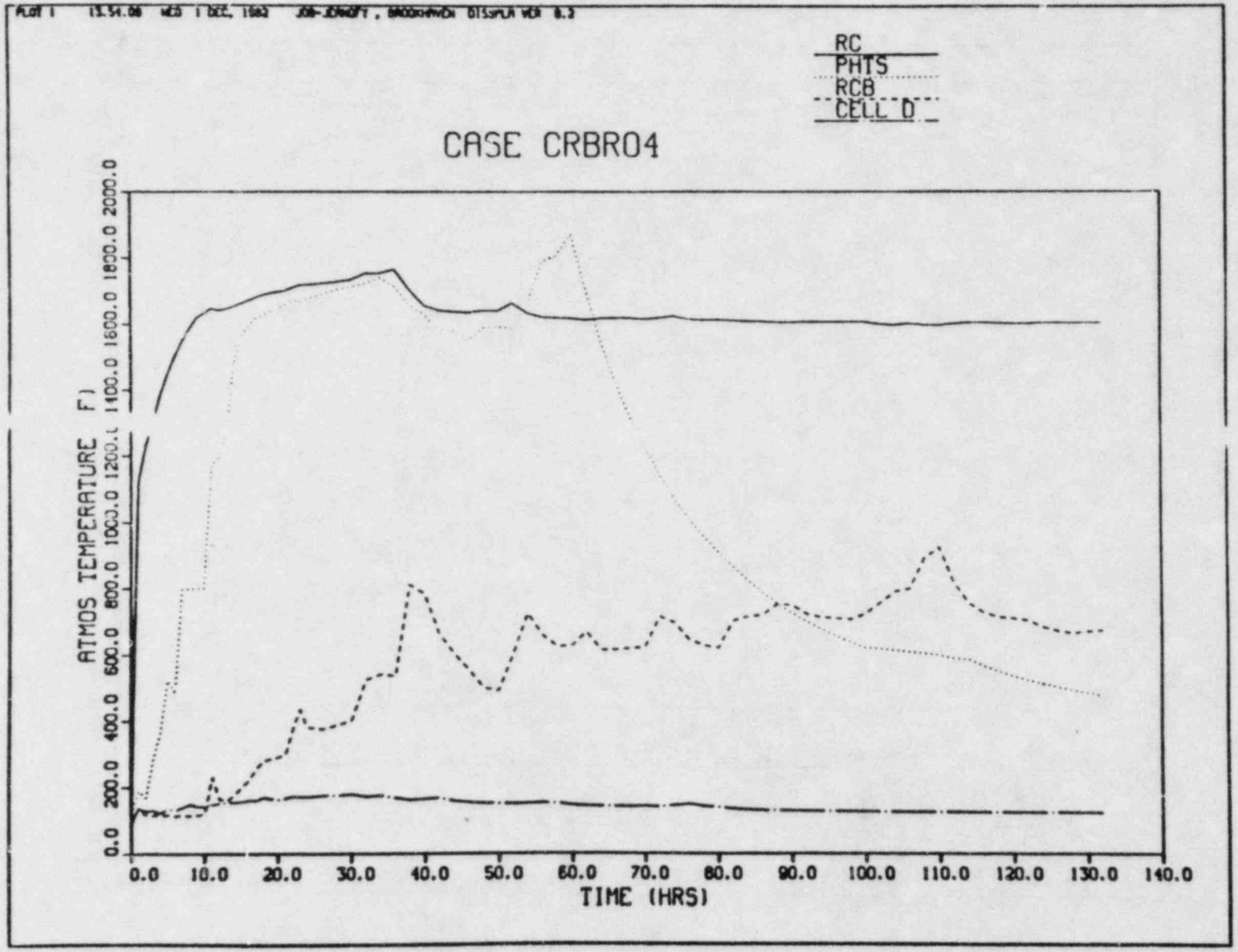
TIME - 0.
TIME - 10.
TIME - 20.
TIME - 50.
TIME - 130.
REACTOR CAVITY LOWER SUBMERGED WALL
TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
DISTANCE (FT)
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.0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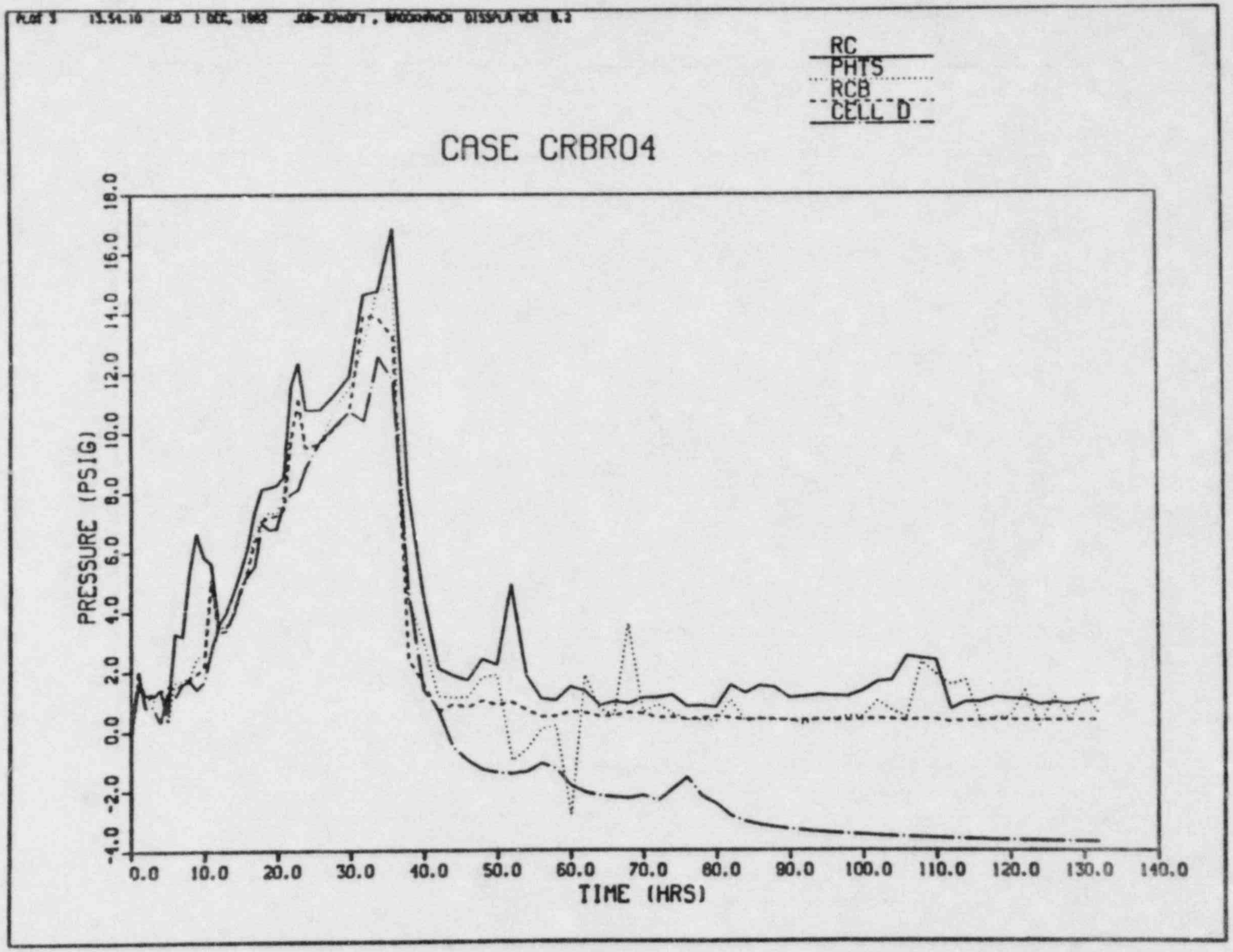
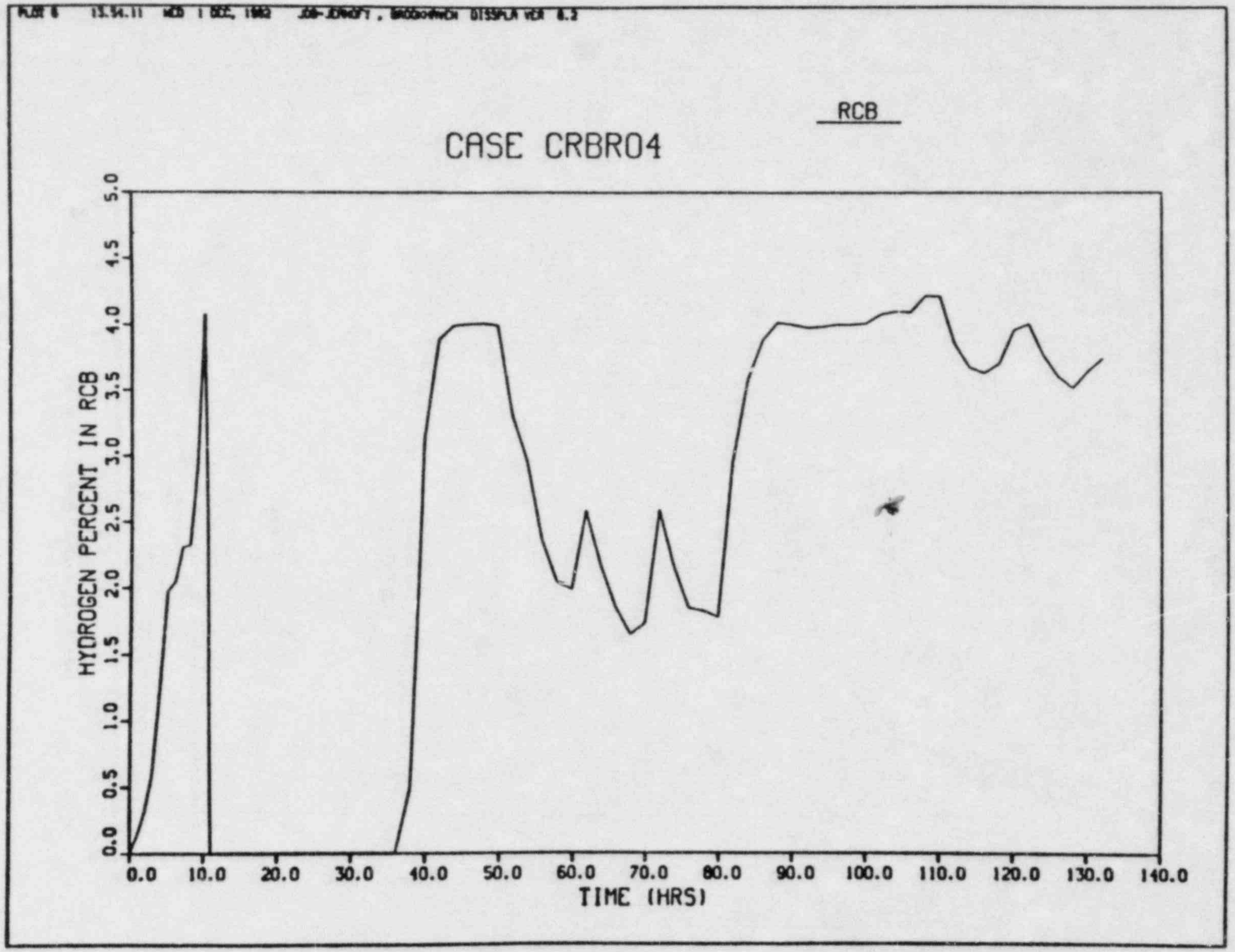
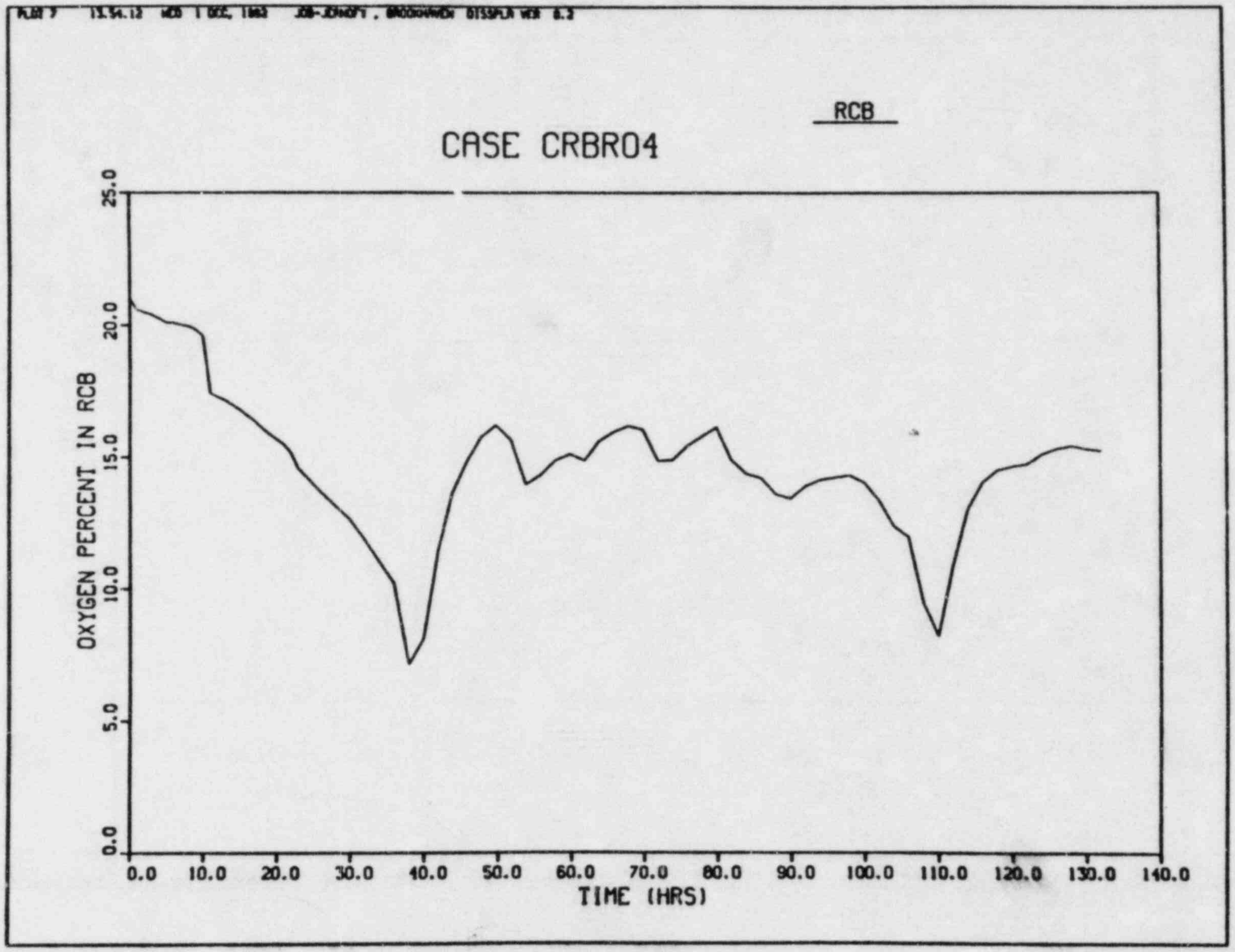
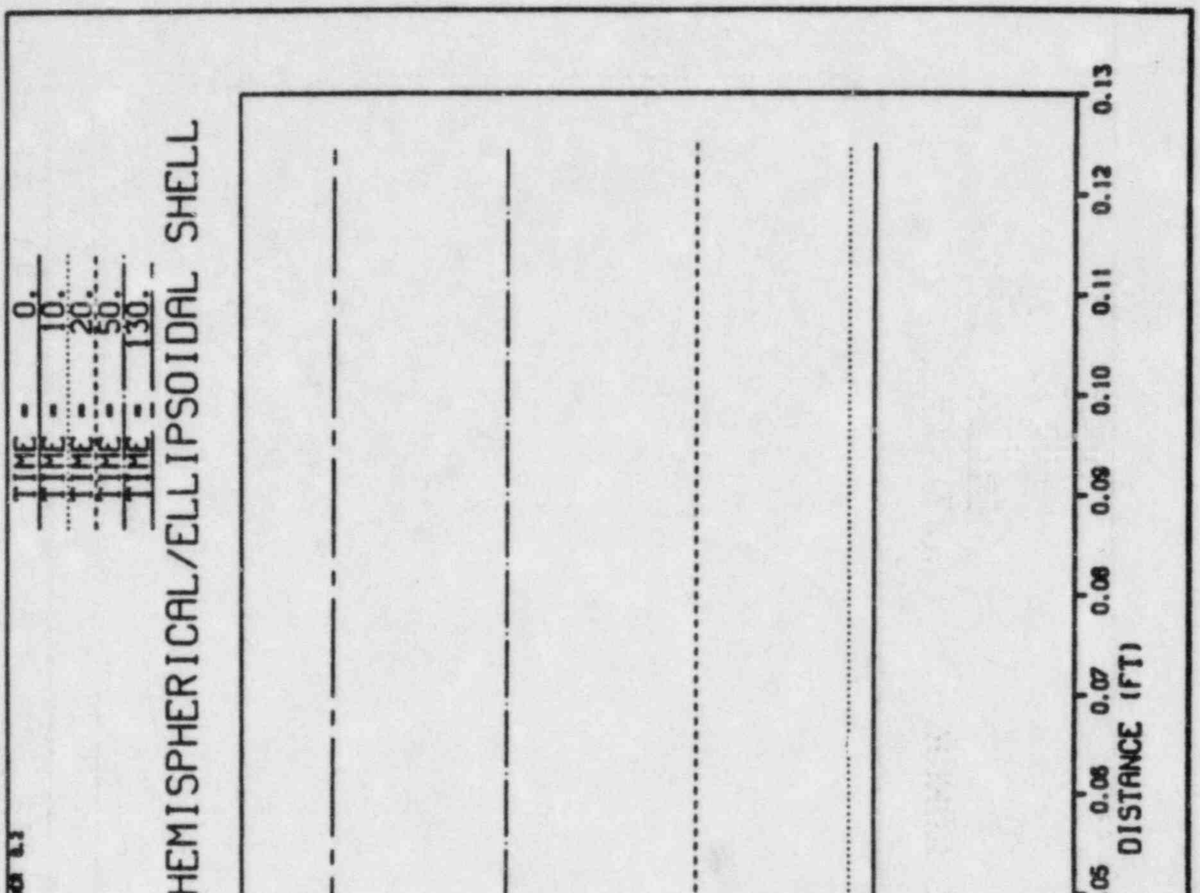
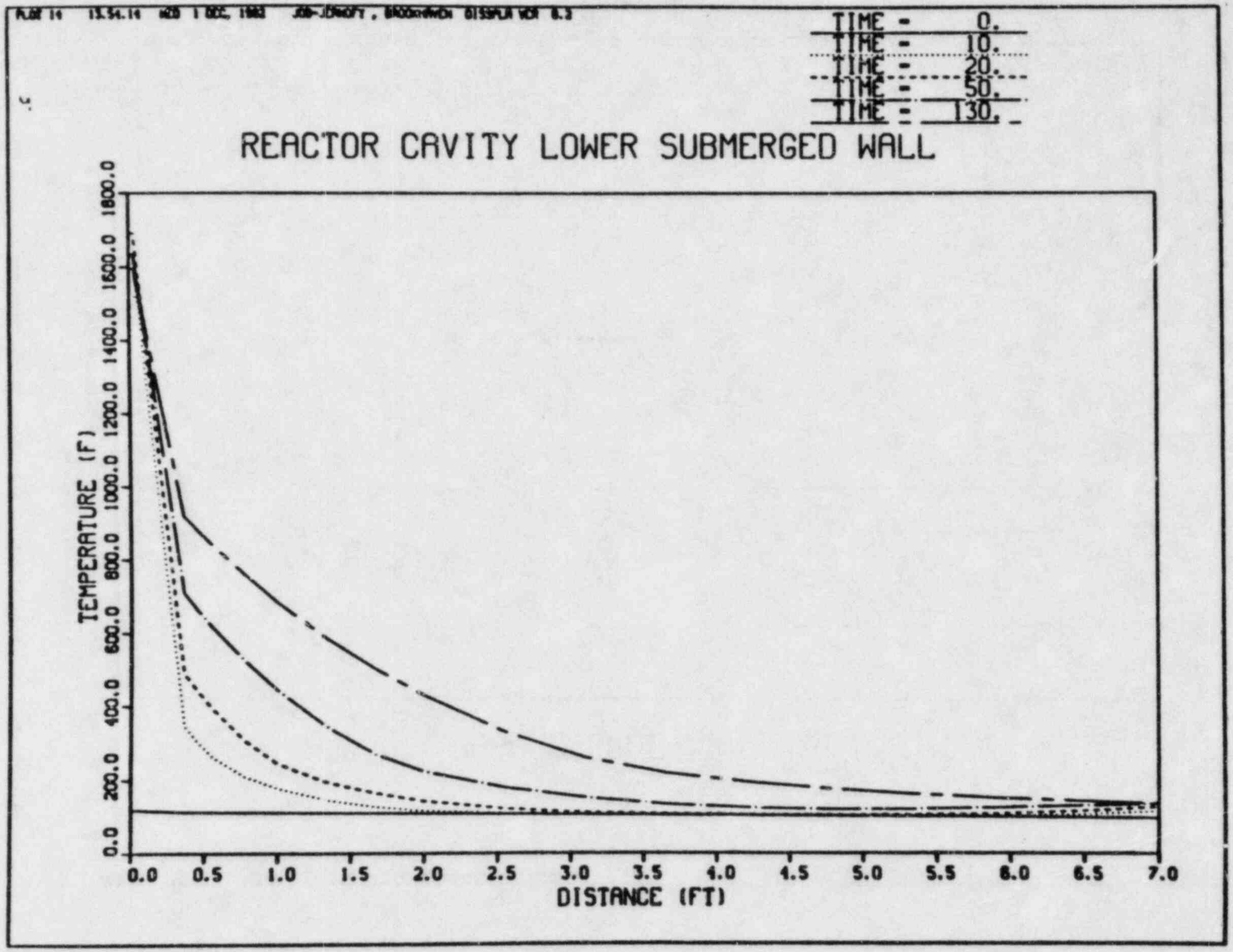
CASE CRBR04
RC
PHTS
RCB
CELL D
ATMOS TEMPERATURE F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
2000.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0

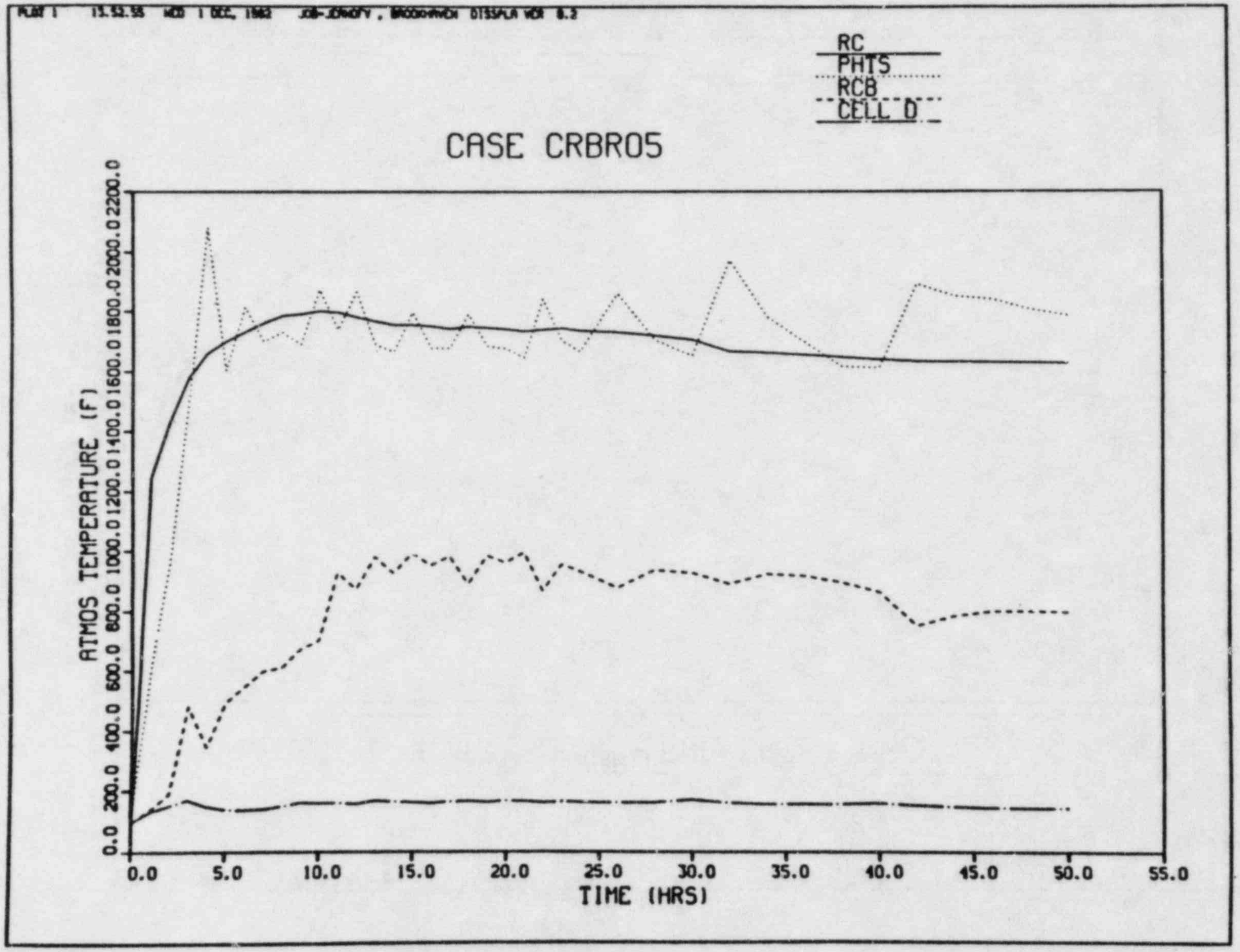

PLOT 3 13.54.10 WED 1 DEC, 1982 JOB-JCWOFT, BROOKHAVEN DISSPLA VER 9.2
CASE CRBR04
RC
PHTS
RCB
CELL D
PRESSURE (PSIG)
18.0
16.0
14.0
12.0
10.0
8.0
6.0
4.0
2.0
0.0
-2.0
-4.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0

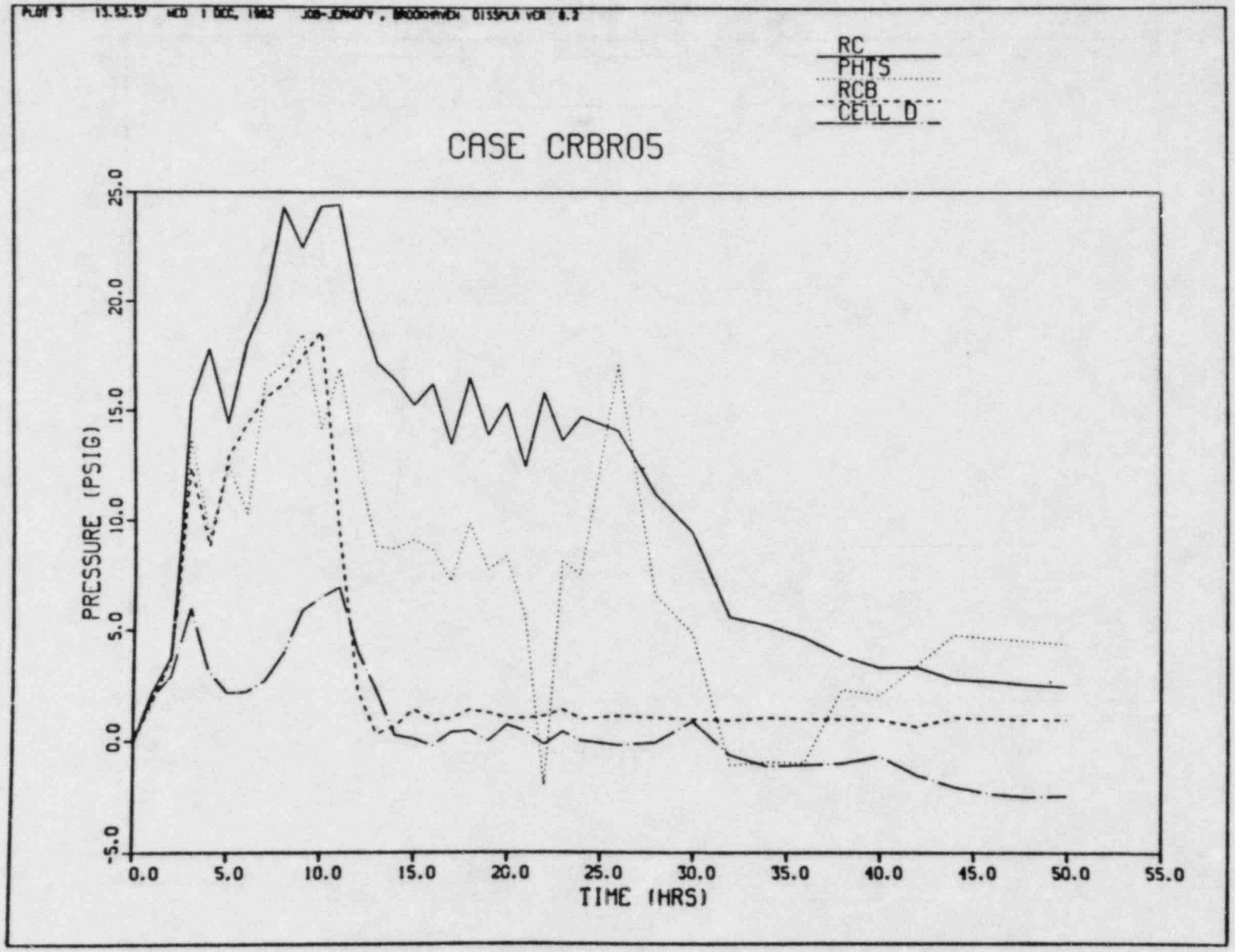

PLOT 6 13.54.11 WED 1 DEC, 1982 JOB-JONKOPT , BROOKHAVEN DISSPLA VER 8.2
RCB
CASE CRBR04
HYDROGEN PERCENT IN RCB
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0
TIME (HRS)

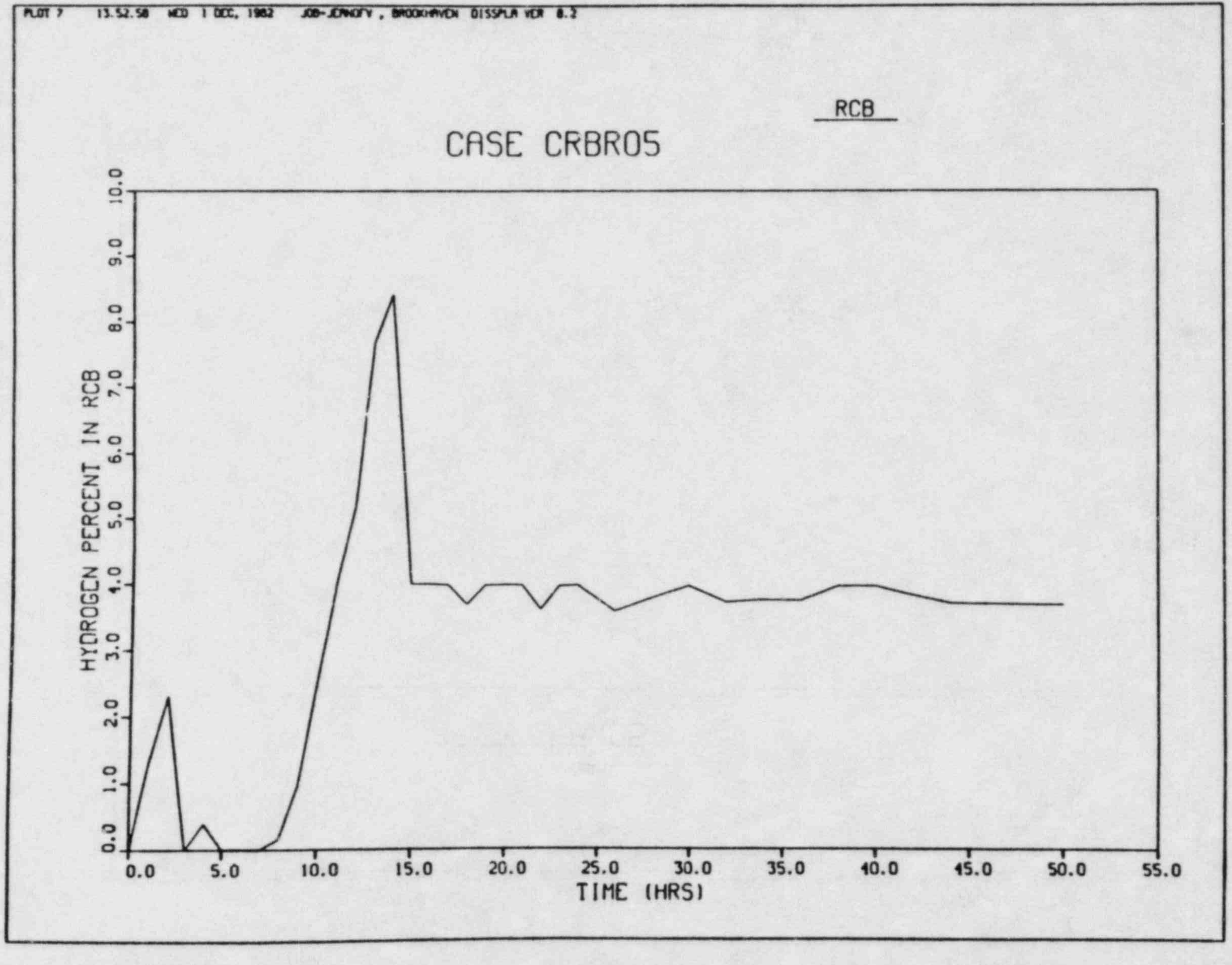

PLOT 7 13.54.12 WED 1 DEC, 1983 JOB-JSNOPY , BROOKHAVEN DISSPLA VER 8.2
RCB
CASE CRBR04
OXYGEN PERCENT IN RCB
0.0
5.0
10.0
15.0
20.0
25.0
TIME (HRS)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
130.0
140.0

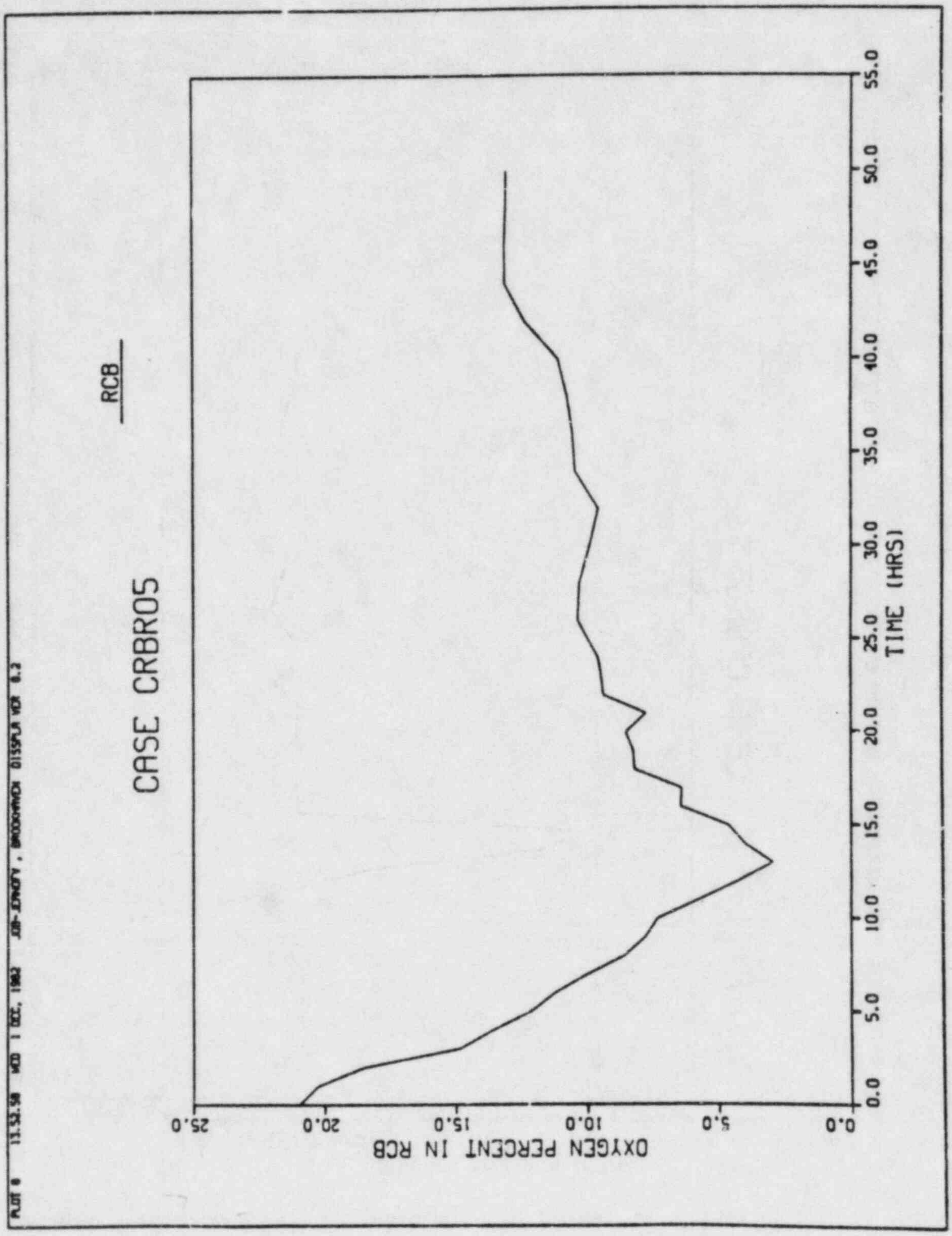

CONTAINMENT STEEL HEMISPHERICAL/ELLIPSOIDAL SHELL
TIME = 0.
TIME = 10.
TIME = 20.
TIME = 50.
TIME = 130.
TEMPERATURE (F)
0.0
50.0
100.0
150.0
200.0
250.0
300.0
350.0
400.0
DISTANCE (FT)
0.00
0.01
0.02
0.03
0.04
0.05
0.06
0.07
0.08
0.09
0.10
0.11
0.12
0.13

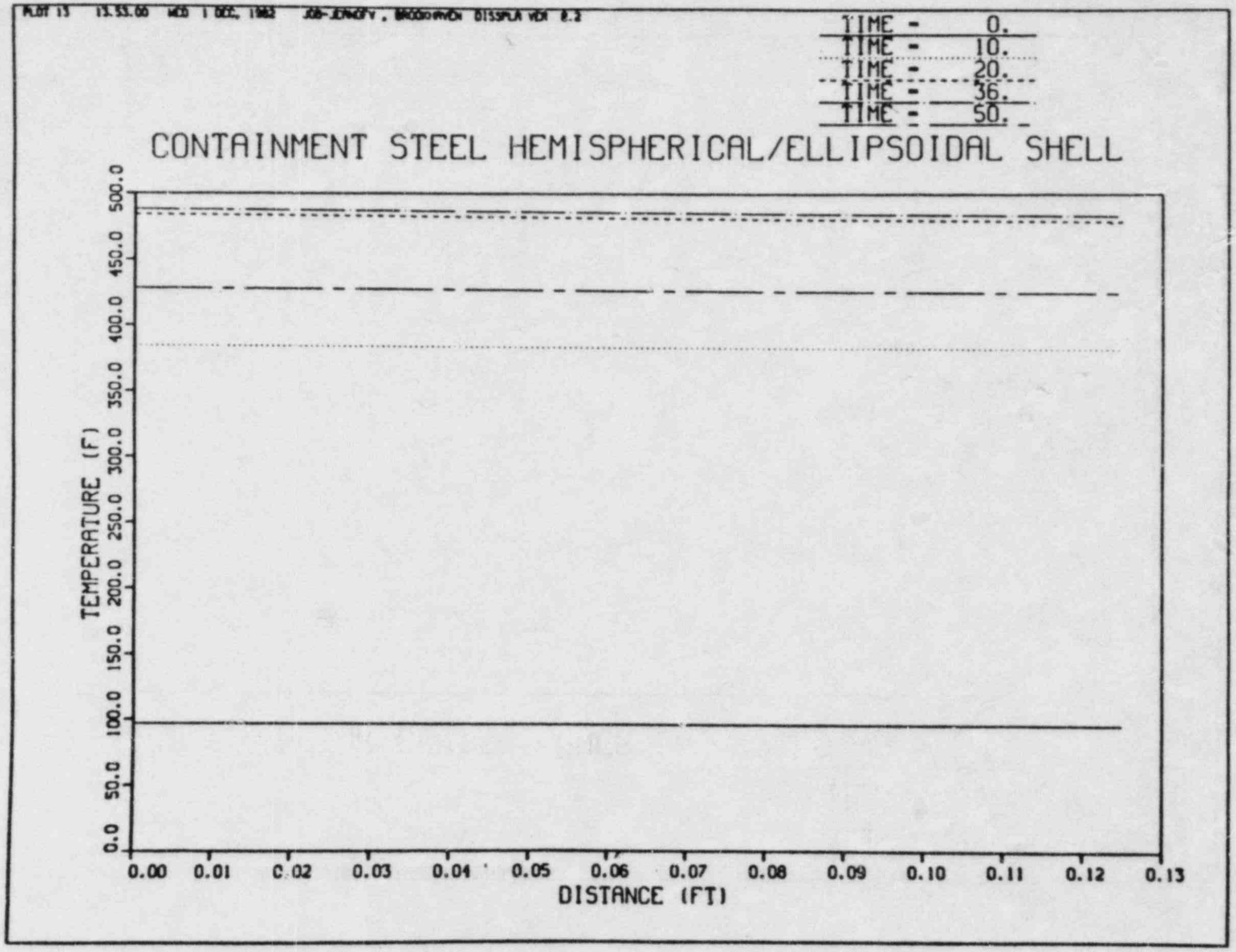

PLOT 14 13.54.14 WED 1 DEC, 1982 BROOKHAVEN DISSPLA VER 8.2
TIME = 0.
TIME = 10.
TIME = 20.
TIME = 50.
TIME = 130.
REACTOR CAVITY LOWER SUBMERGED WALL
TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
DISTANCE (FT)
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.0

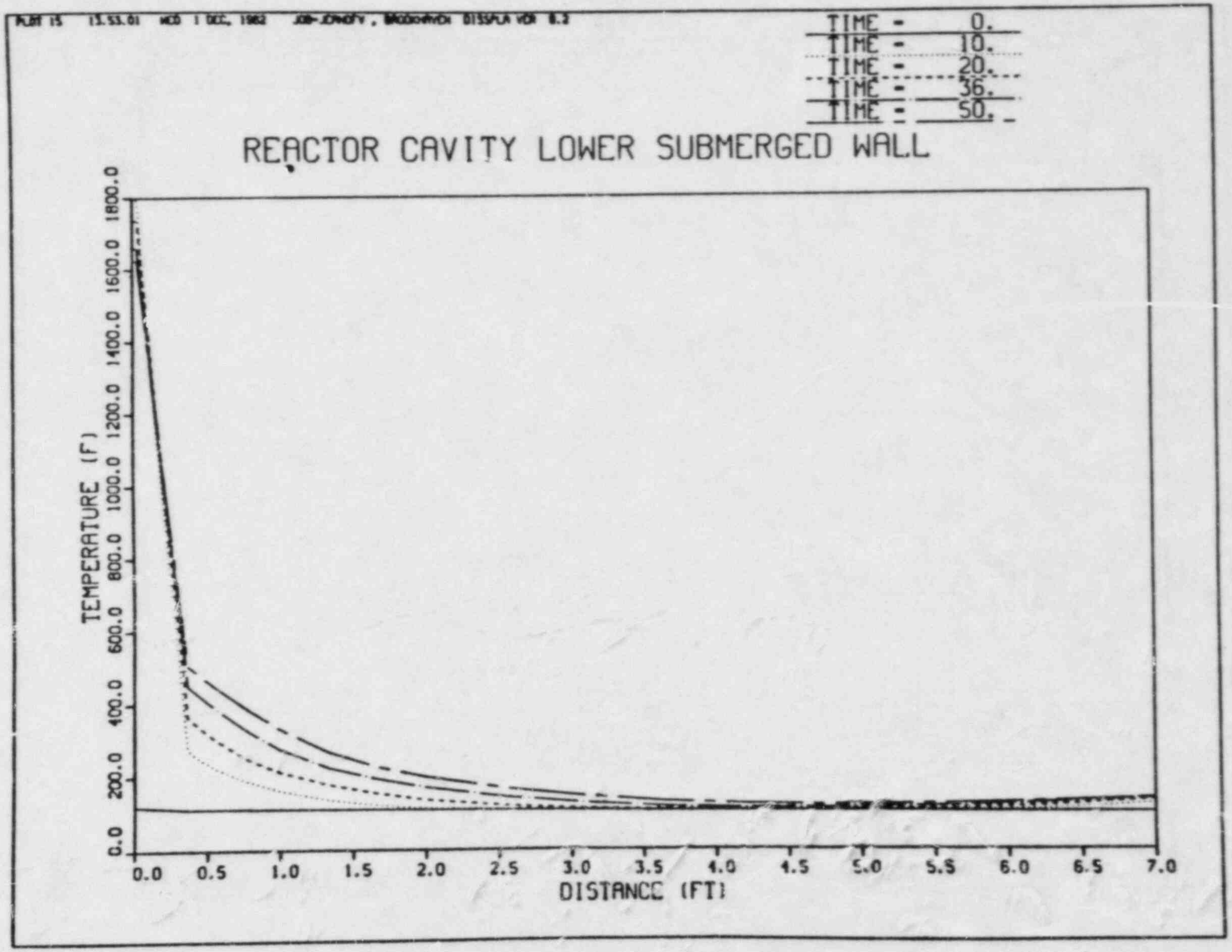

PLOT 1 13.52.55 WED 1 DEC, 1982 JOB-JCNOFY , BROOKHAVEN DISSPLA VER 8.2
CASE CRBR05
RC
PHTS
RCB
CELL D
ATMOS TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
2000.0
2200.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0

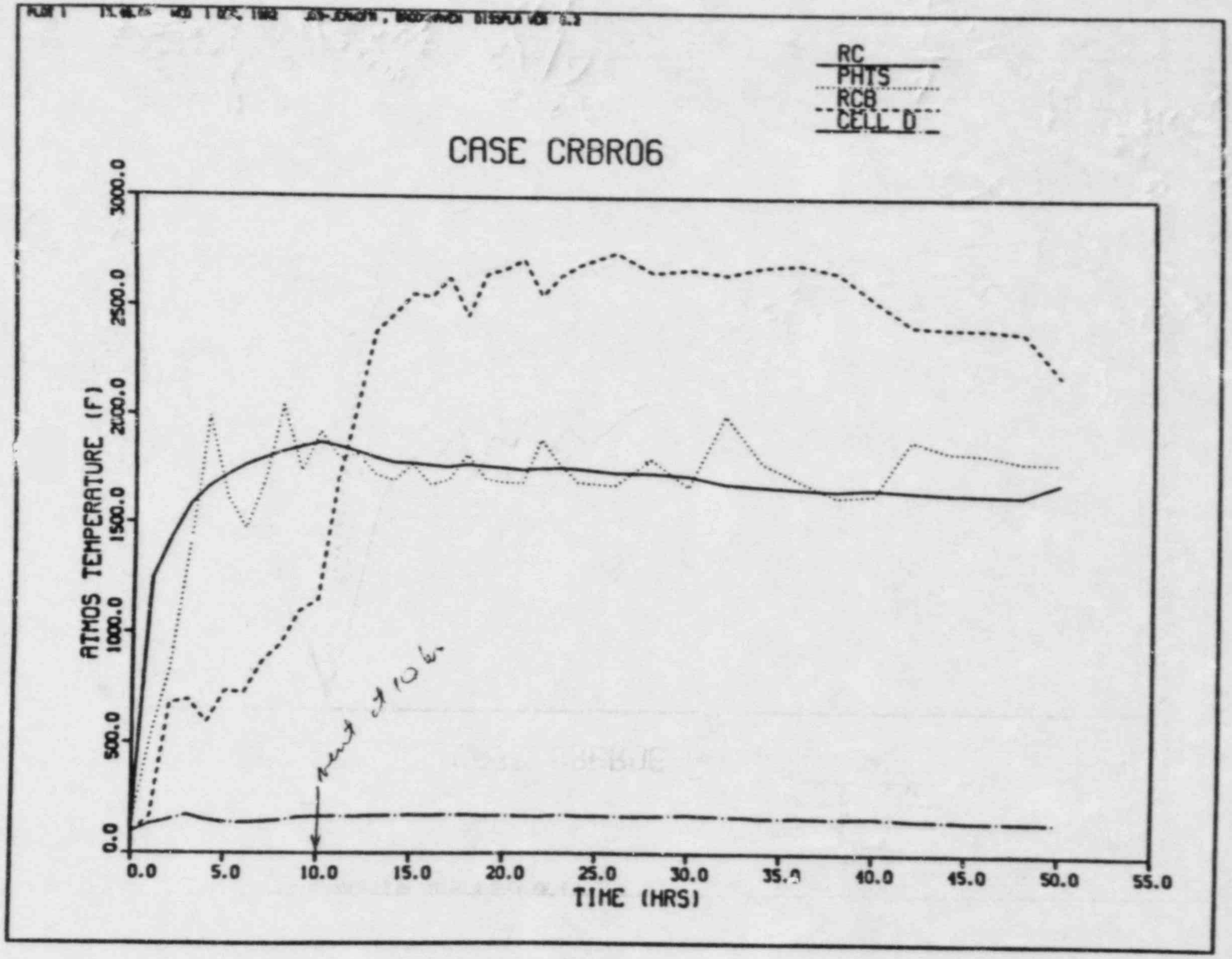

PLOT 3 13.52.57 WED 1 DEC, 1982 JOB-JCNKOFY , BROOKHAVEN DISSPLA VER 8.2
CASE CRBR05
RC
PHTS
RCB
CELL D
PRESSURE (PSIG)
-5.0
0.0
5.0
10.0
15.0
20.0
25.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0

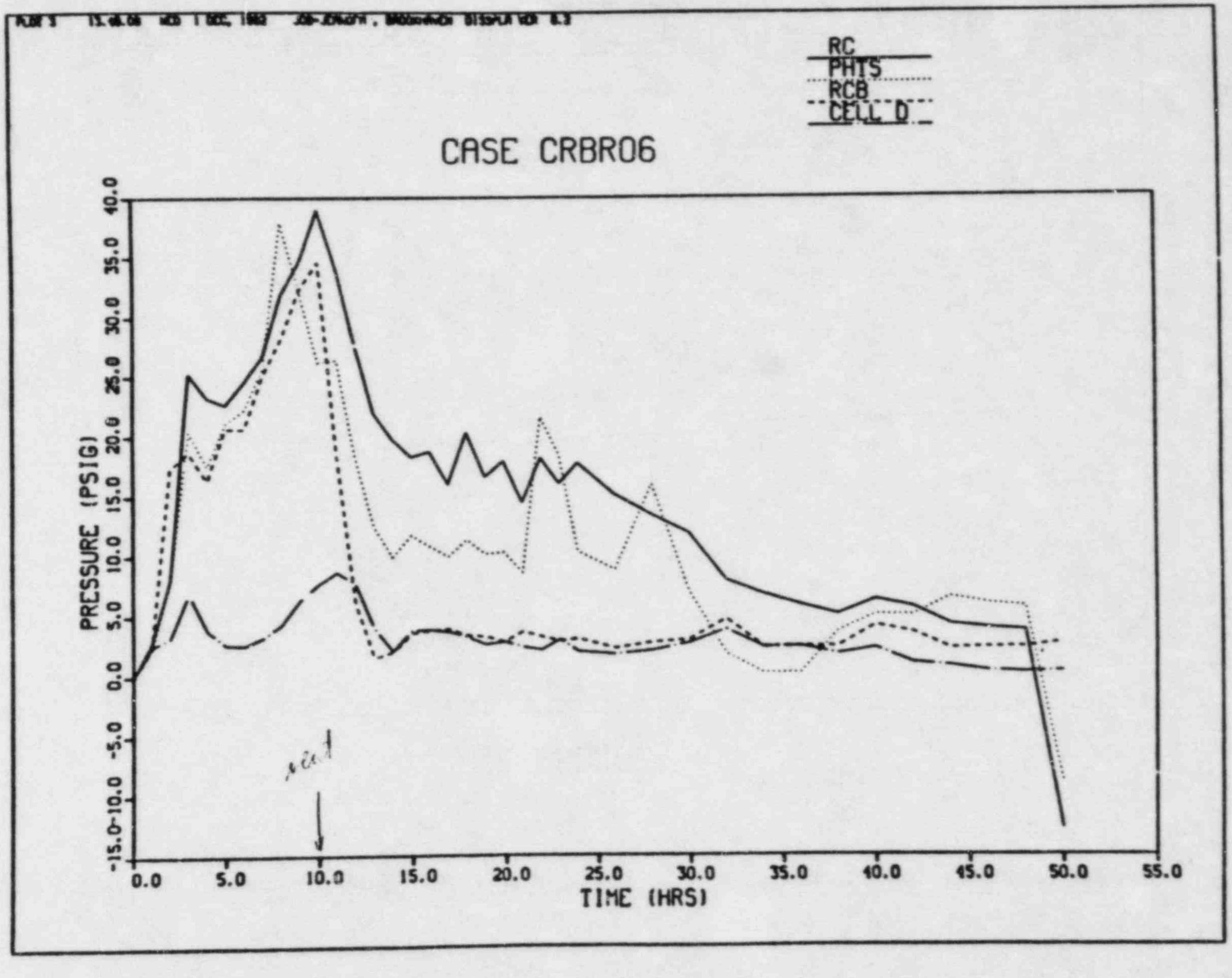

PLOT 7 13.52.58 WED 1 DEC, 1982 JOB-JCNOFV , BROOKHAVEN DISSPLA VER 8.2
RCB
CASE CRBROS
HYDROGEN PERCENT IN RCB
0.0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0
TIME (HRS)

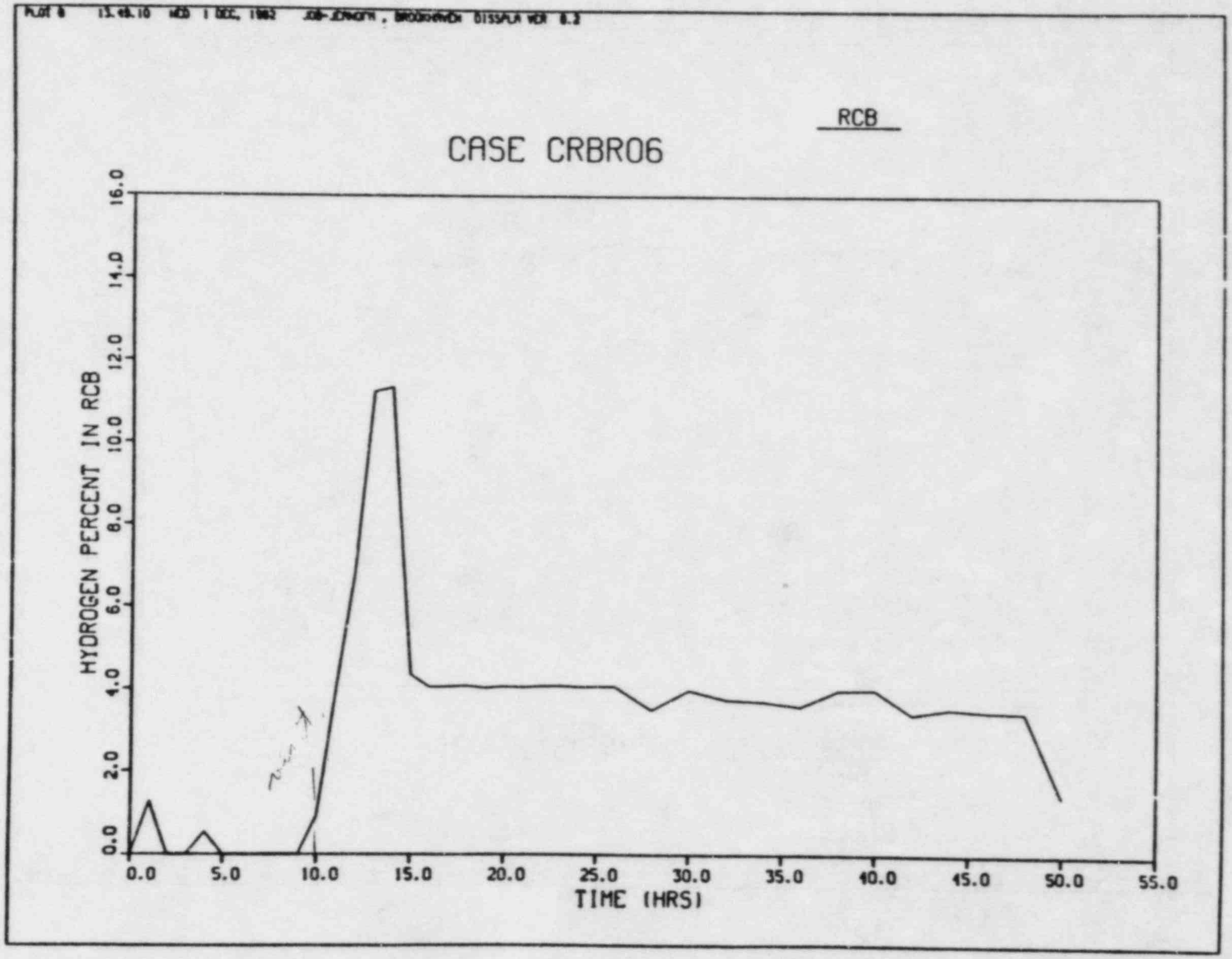

PLOT 8 13.53.58 WED 1 DEC, 1982 JOB-JCWAOFV , BROOKHAVEN DISSPLA VER 8.2
CASE CRBR05
RCB
OXYGEN PERCENT IN RCB
0.0
5.0
10.0
15.0
20.0
25.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0

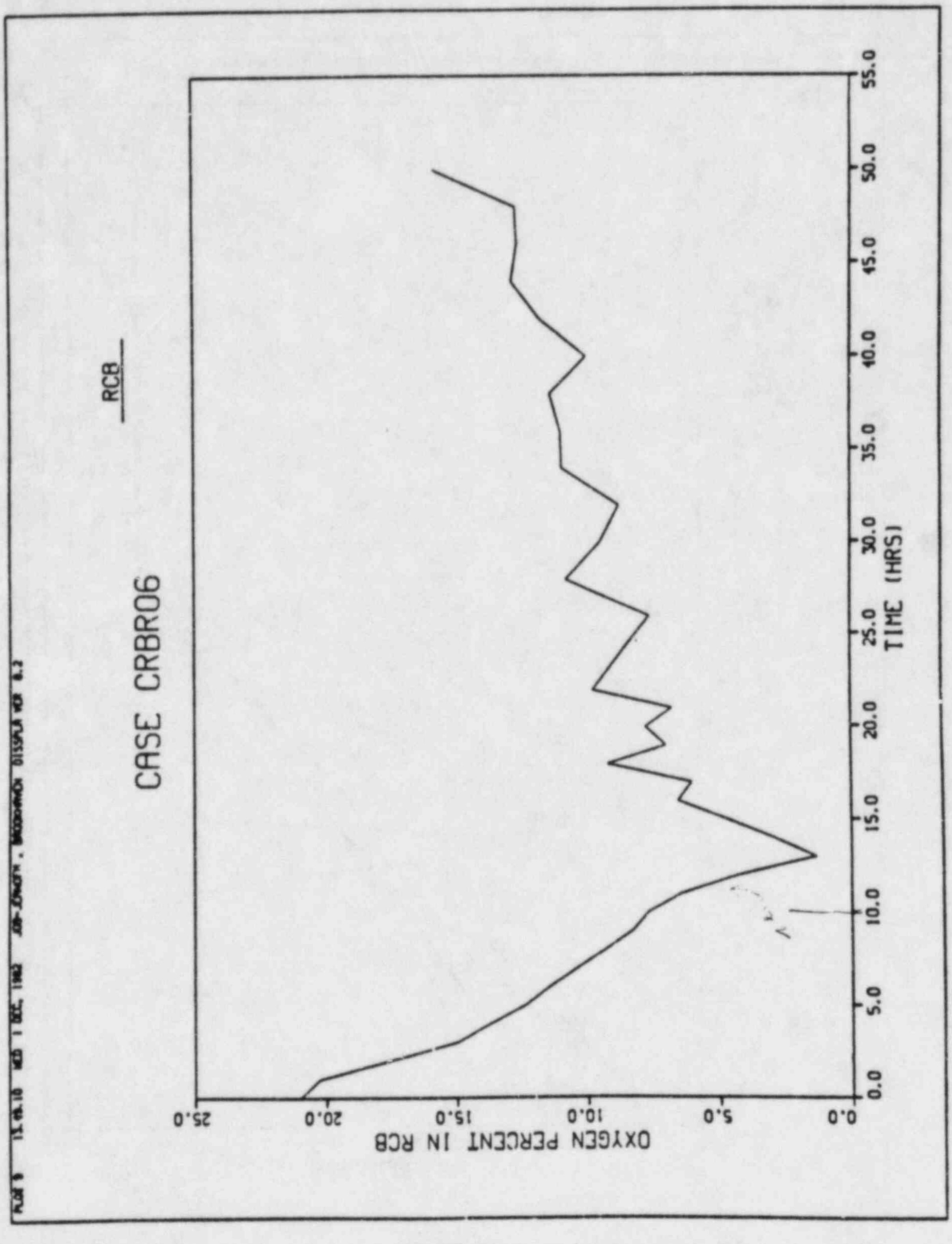

PLOT 13 13.53.00 WED 1 DEC, 1982 JOB-JENGFY , BROOKHAVEN DISSPLA VER 8.2
TIME - 0.
TIME - 10.
TIME - 20.
TIME - 36.
TIME - 50.
CONTAINMENT STEEL HEMISPHERICAL/ELLIPSOIDAL SHELL
TEMPERATURE (F)
0.0
50.0
100.0
150.0
200.0
250.0
300.0
350.0
400.0
450.0
500.0
DISTANCE (FT)
0.00
0.01
0.02
0.03
0.04
0.05
0.06
0.07
0.08
0.09
0.10
0.11
0.12
0.13

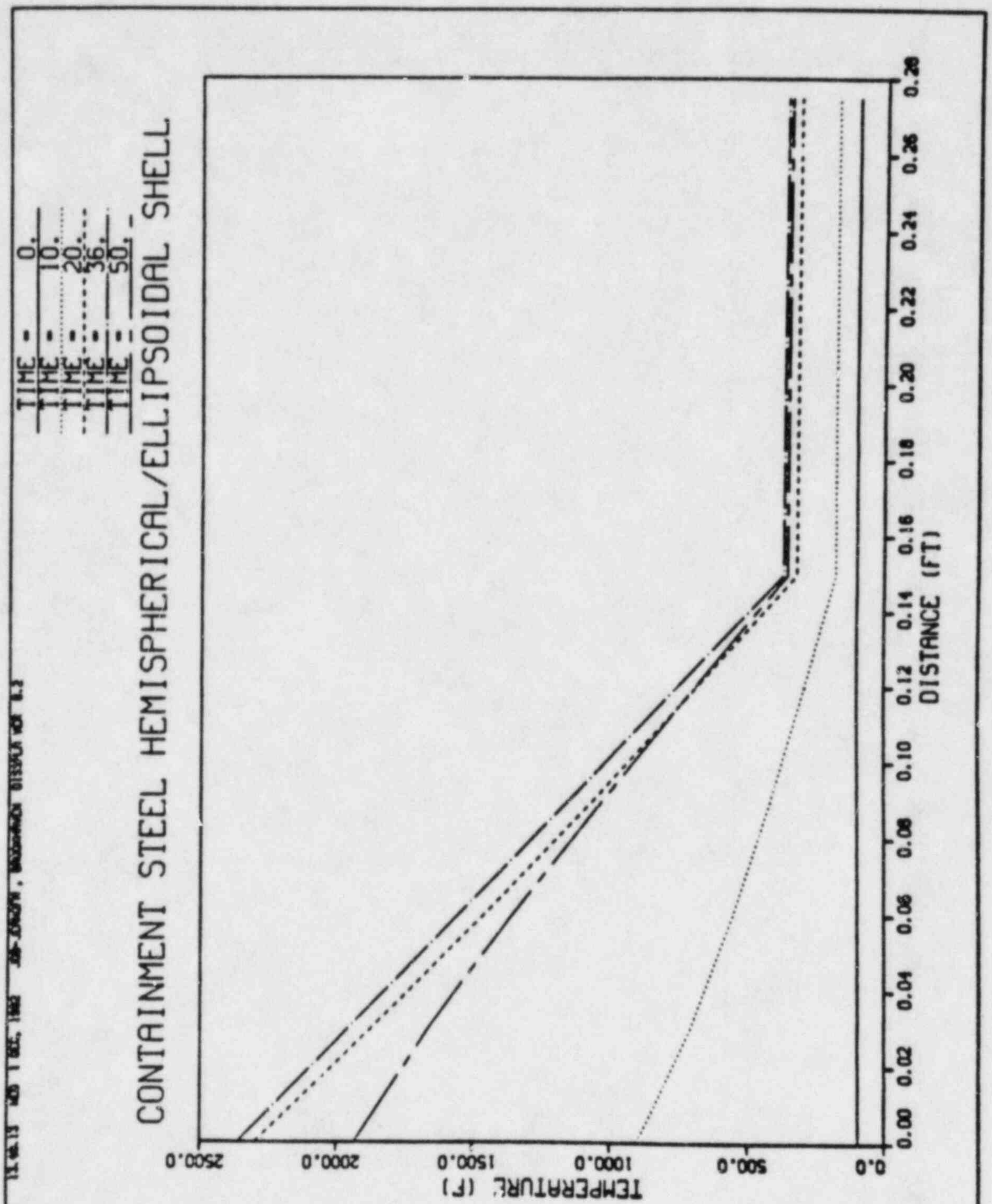

PLOT 15 13.53.01 WED 1 DEC, 1982 DISSPLA VER 8.2
TIME - 0.
TIME - 10.
TIME - 20.
TIME - 36.
TIME - 50.
REACTOR CAVITY LOWER SUBMERGED WALL
TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
DISTANCE (FT)
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.0

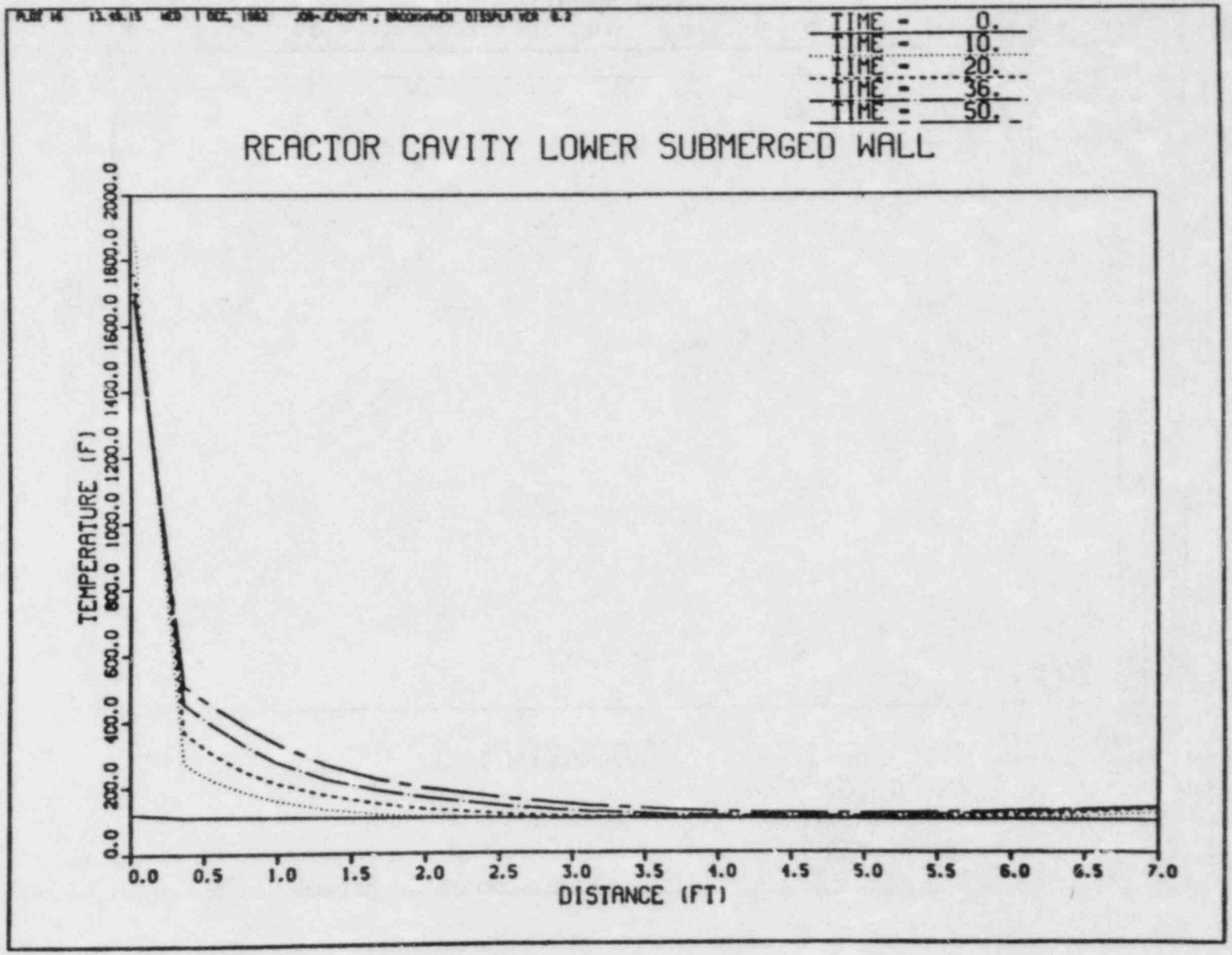
CASE CRBR06
RC
PHTS
RCB
CELL D
ATMOS TEMPERATURE (F)
0.0
500.0
1000.0
1500.0
2000.0
2500.0
3000.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0
vent at 10 h

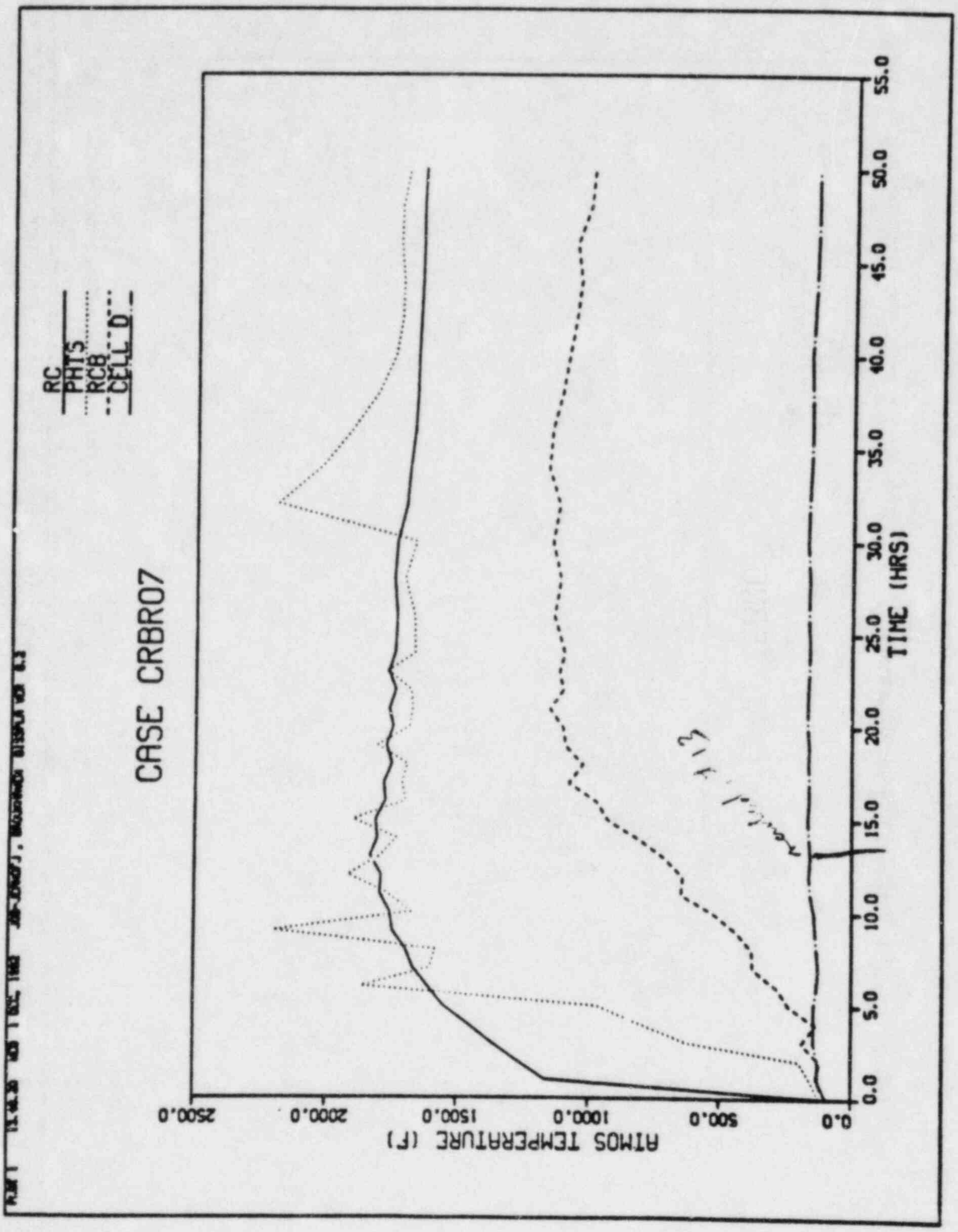

RC
PHTS
RCB
CELL D
CASE CRBR06
PRESSURE (PSIG)
40.0
35.0
30.0
25.0
20.0
15.0
10.0
5.0
0.0
-5.0
-10.0
-15.0
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0
TIME (HRS)
vent

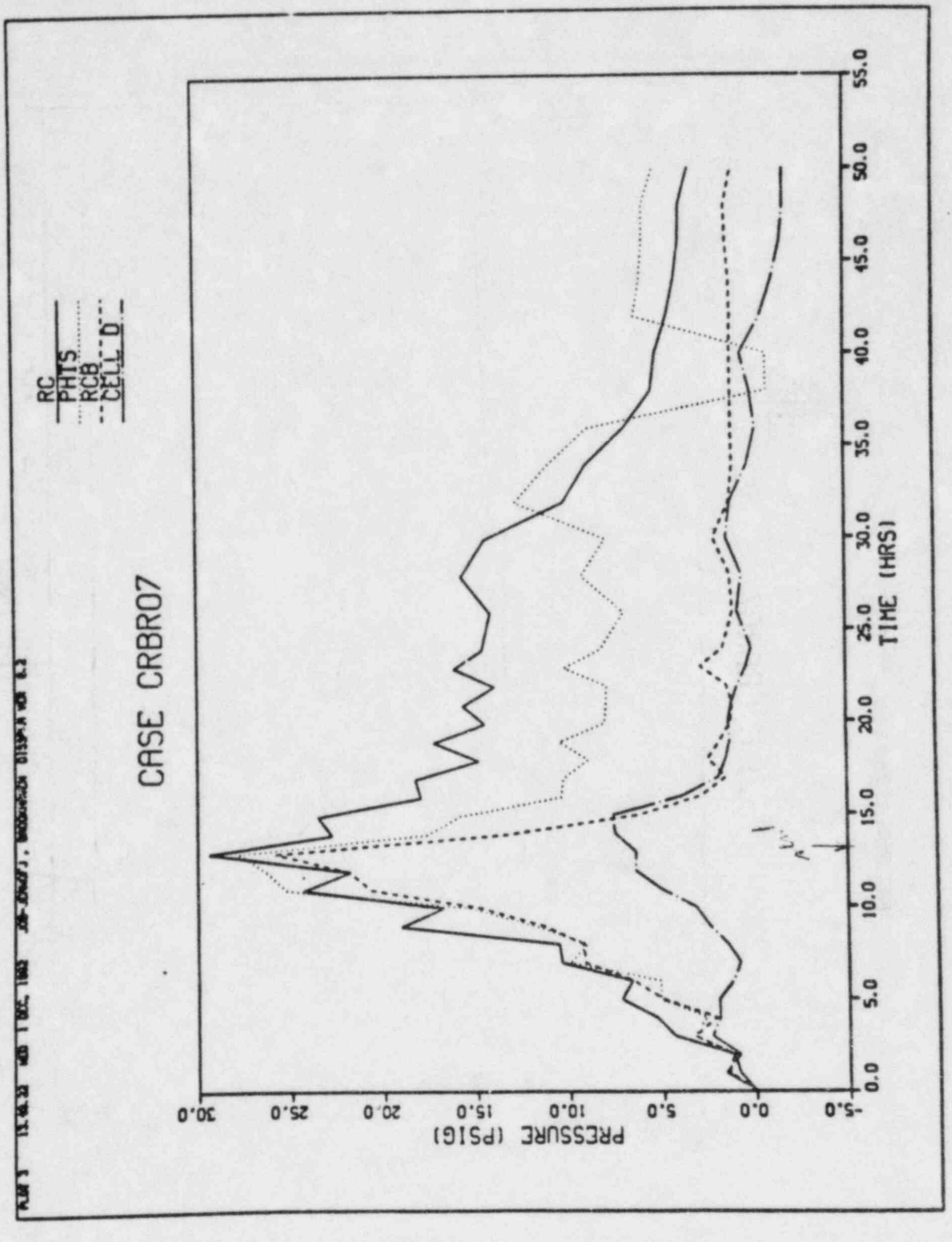

PLOT 8 13.48.10 WED 1 DEC, 1982 JOB-JENKOTH , BROOKHAVEN DISSPLA VER 8.2
RCB
CASE CRBR06
HYDROGEN PERCENT IN RCB
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0

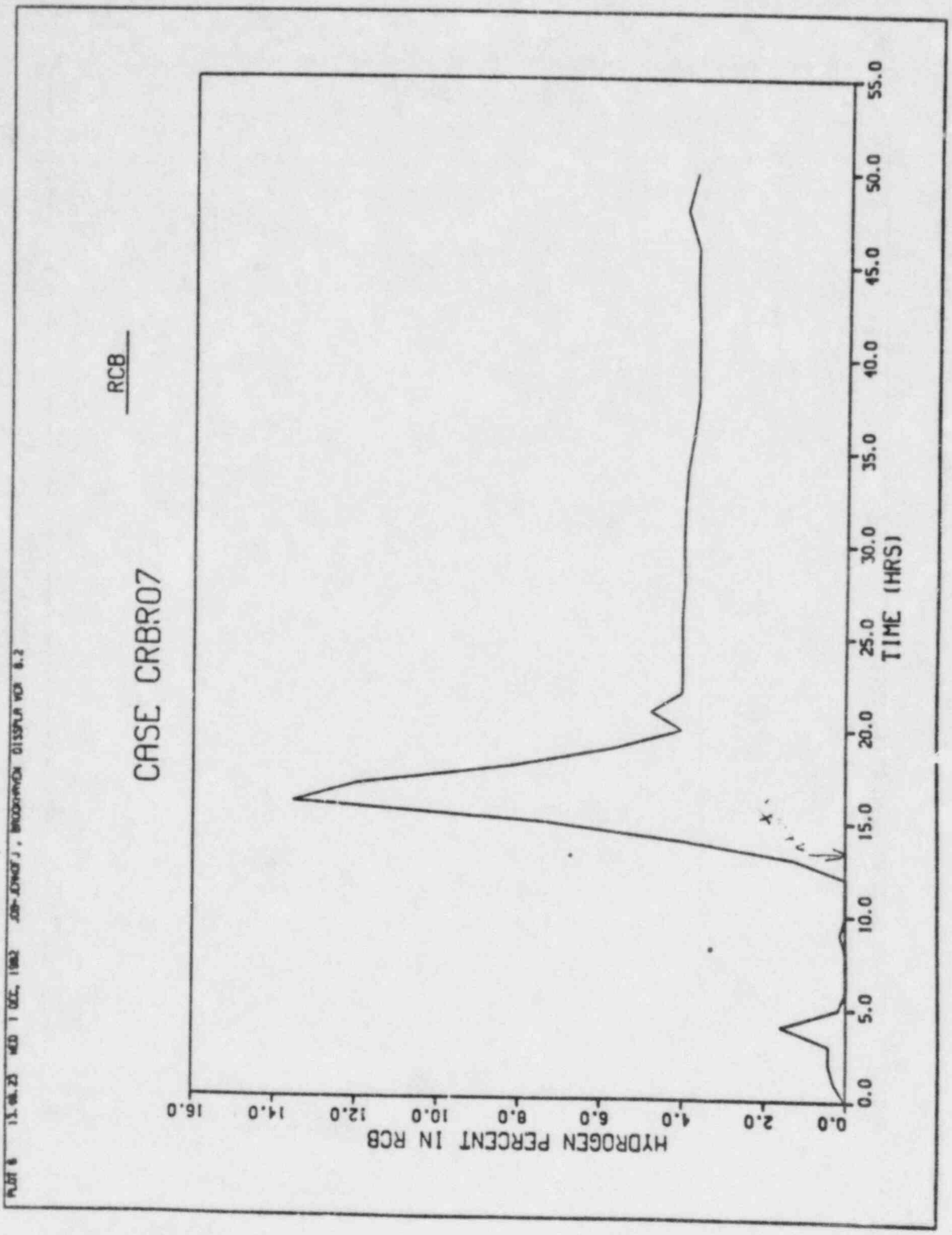

PLOT 9 13.49.10 WED 1 DEC, 1982 DISSPLA VER 8.2
CASE CRBR06
RCB
OXYGEN PERCENT IN RCB
0.0
5.0
10.0
15.0
20.0
25.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0

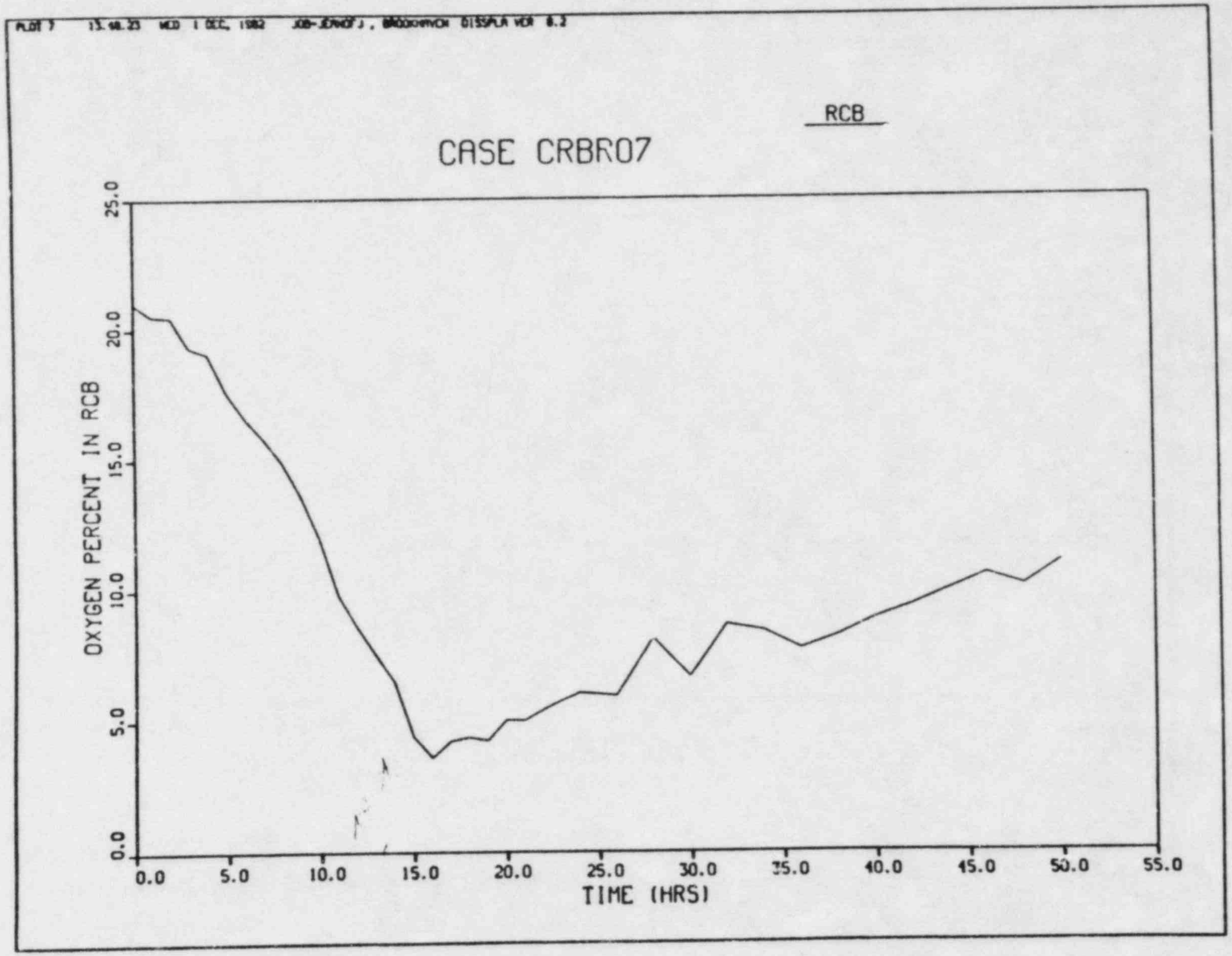

CONTAINMENT STEEL HEMISPHERICAL/ELLIPSOIDAL SHELL
TIME = 0.
TIME = 10.
TIME = 20.
TIME = 36.
TIME = 50.
TEMPERATURE (F)
0.0
500.0
1000.0
1500.0
2000.0
2500.0
DISTANCE (FT)
0.00
0.02
0.04
0.06
0.08
0.10
0.12
0.14
0.16
0.18
0.20
0.22
0.24
0.26
0.28

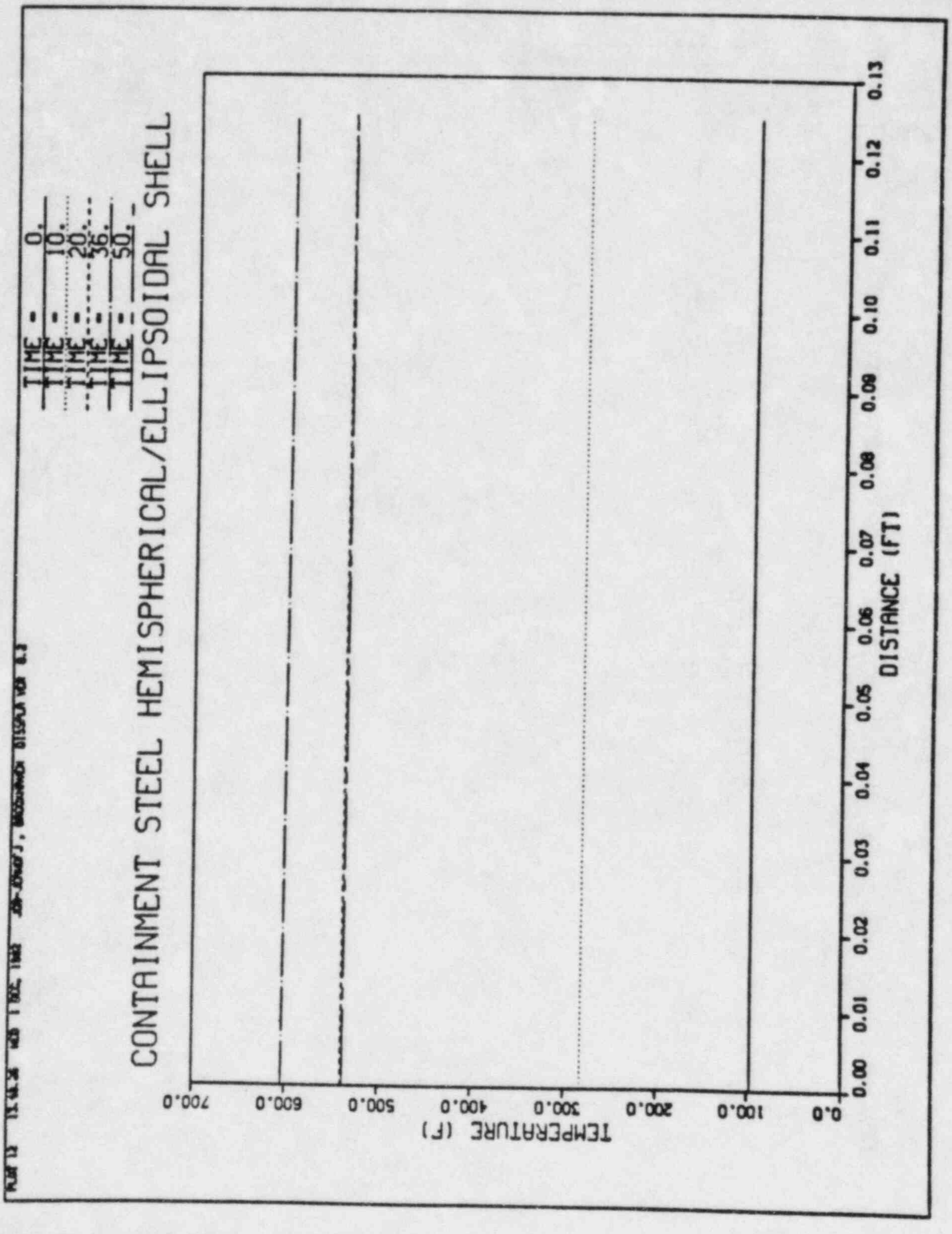

PLOT 16 13.45.15 WED 1 DEC, 1982 JOB-ZENOTH, BROOKHAVEN DISSPLA VER 8.2
TIME - 0.
TIME - 10.
TIME - 20.
TIME - 36.
TIME - 50.
REACTOR CAVITY LOWER SUBMERGED WALL
TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
2000.0
DISTANCE (FT)
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.0

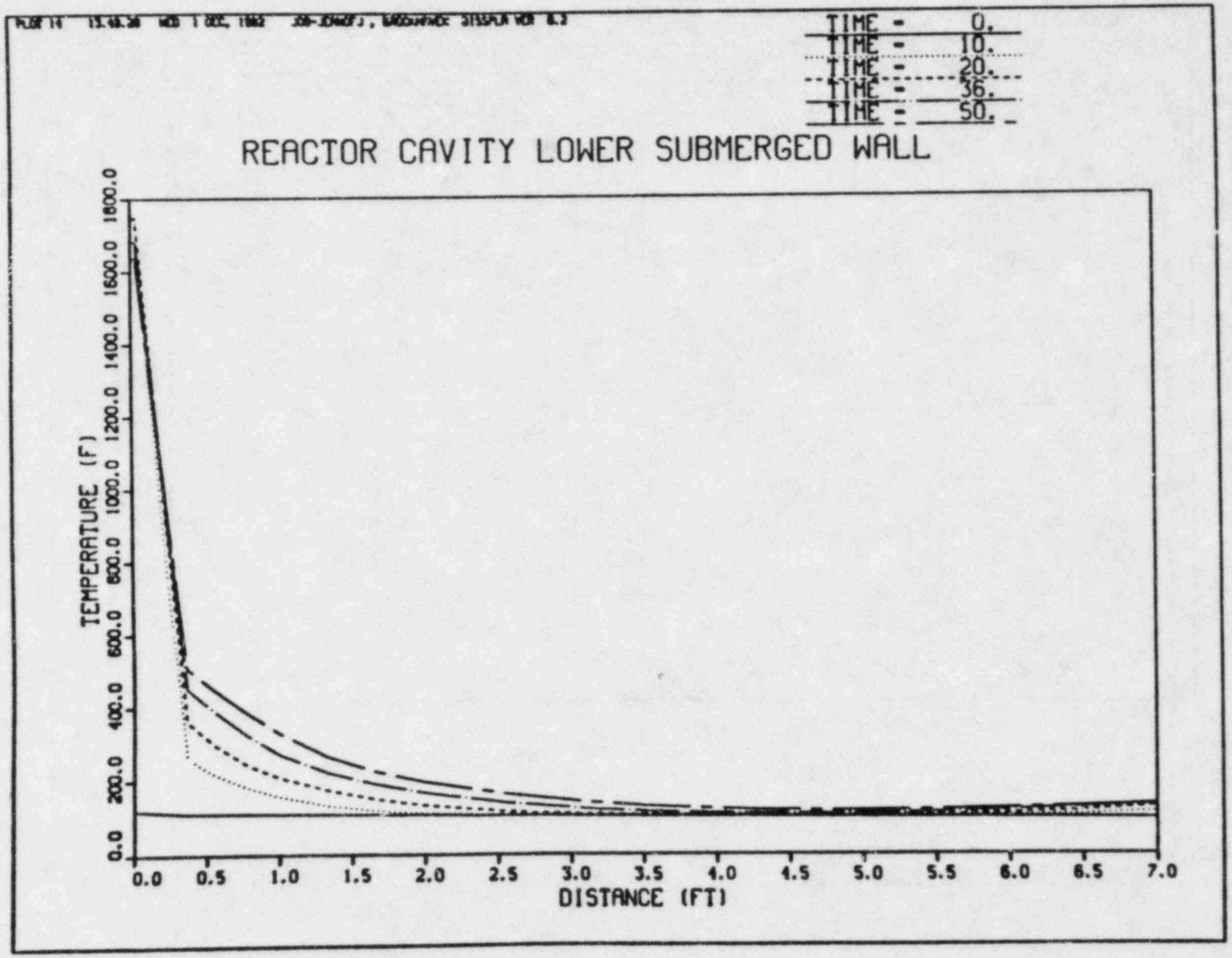

CASE CRBR07
RC
PHTS
RCB
CELL D
ATMOS TEMPERATURE (F)
0.0
500.0
1000.0
1500.0
2000.0
2500.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0

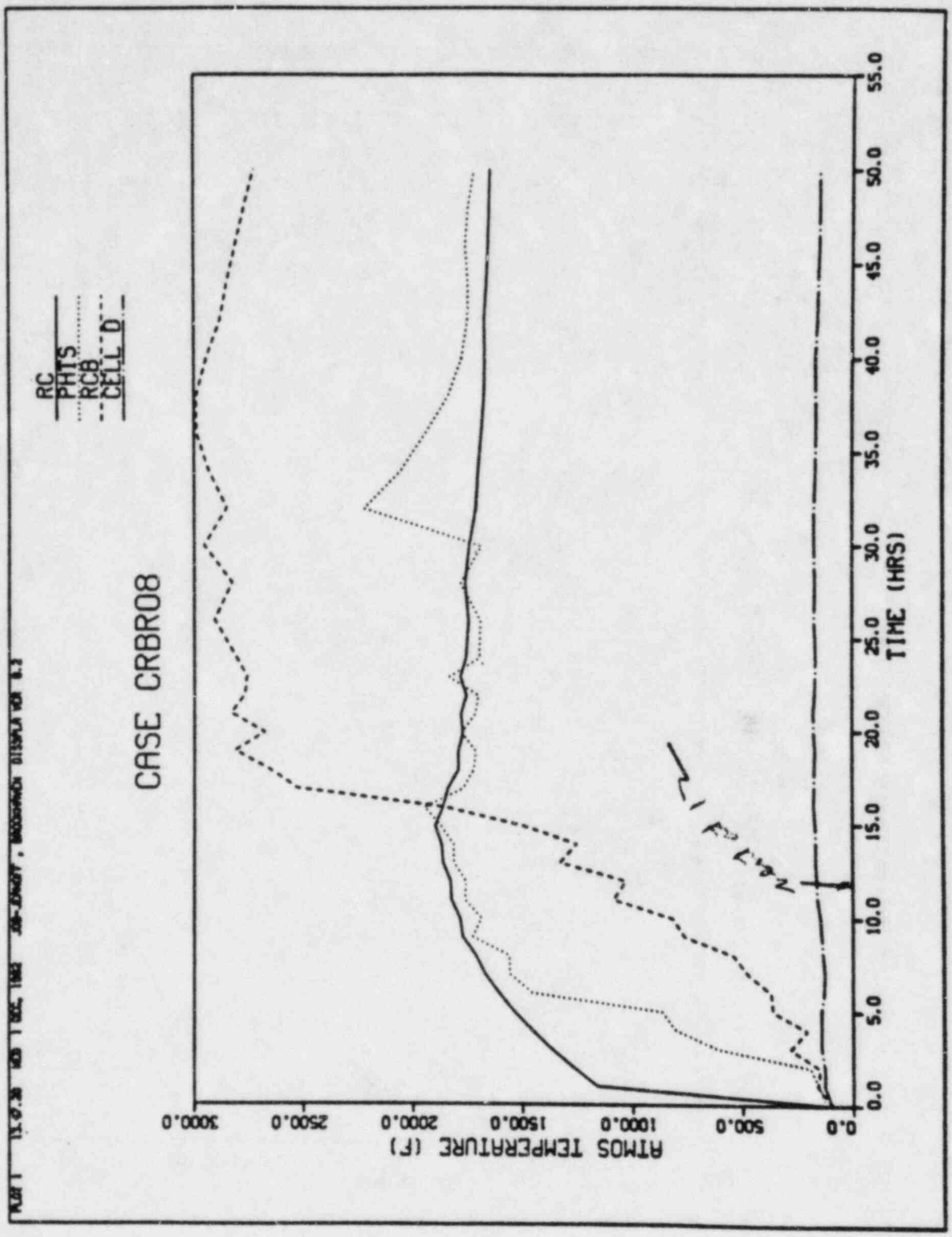

CASE CRBR07
RC
PHTS
RCB
CELL D
PRESSURE (PSIG)
30.0
25.0
20.0
15.0
10.0
5.0
0.0
-5.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0
DISSPLA VER 8.2

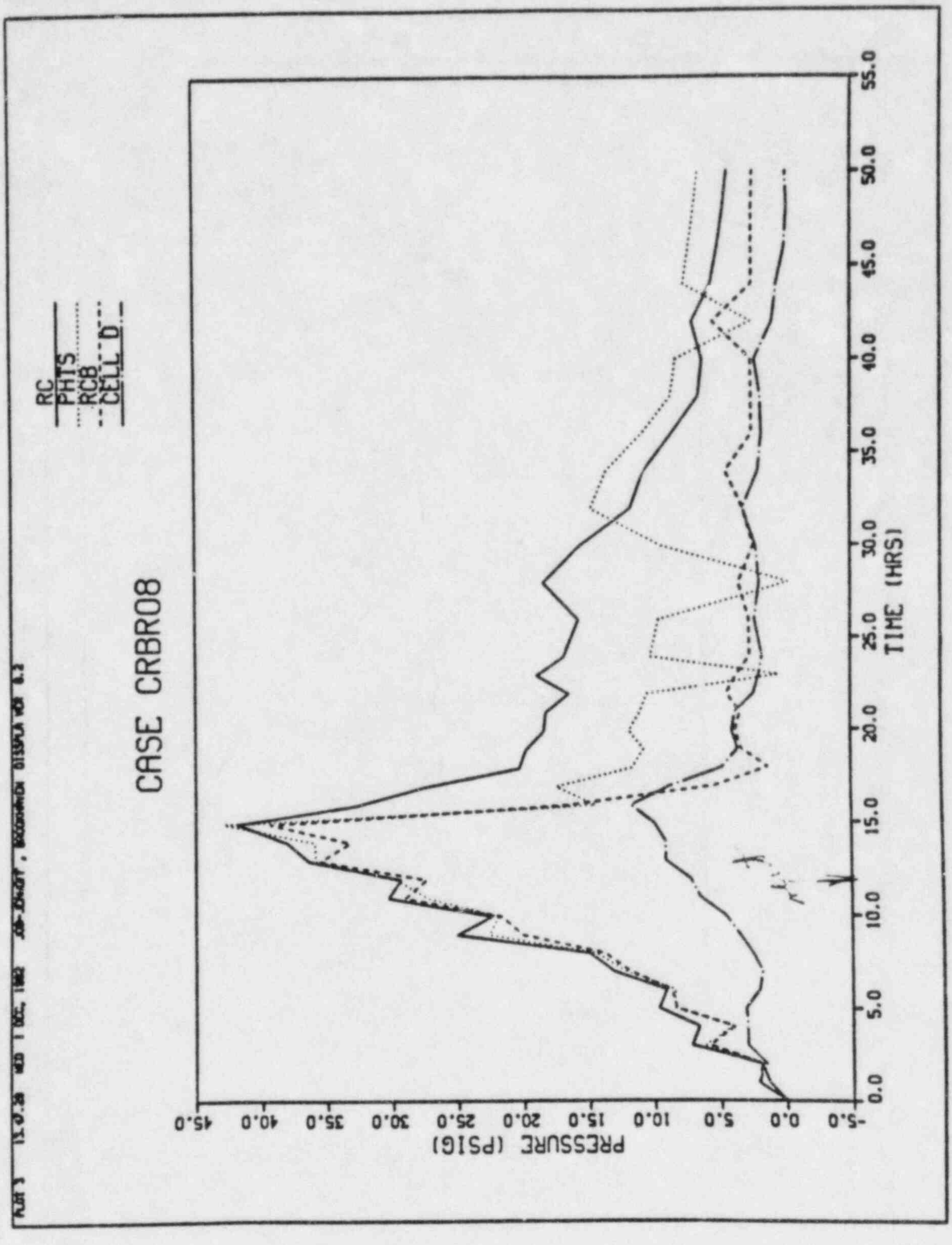

PLOT 6 13.08.23 WED 1 DEC, 1982 JOB-JONHOFJ, BROOKHAVEN DISSPLA VER 8.2
CASE CRBR07
RCB
HYDROGEN PERCENT IN RCB
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0
x


PLOT 7 13.48.23 WED 1 DEC, 1982 JOB-JENKOFJ , BROOKHAVEN DISSPLA VER 8.2
RCB
CASE CRBR07
OXYGEN PERCENT IN RCB
0.0
5.0
10.0
15.0
20.0
25.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0

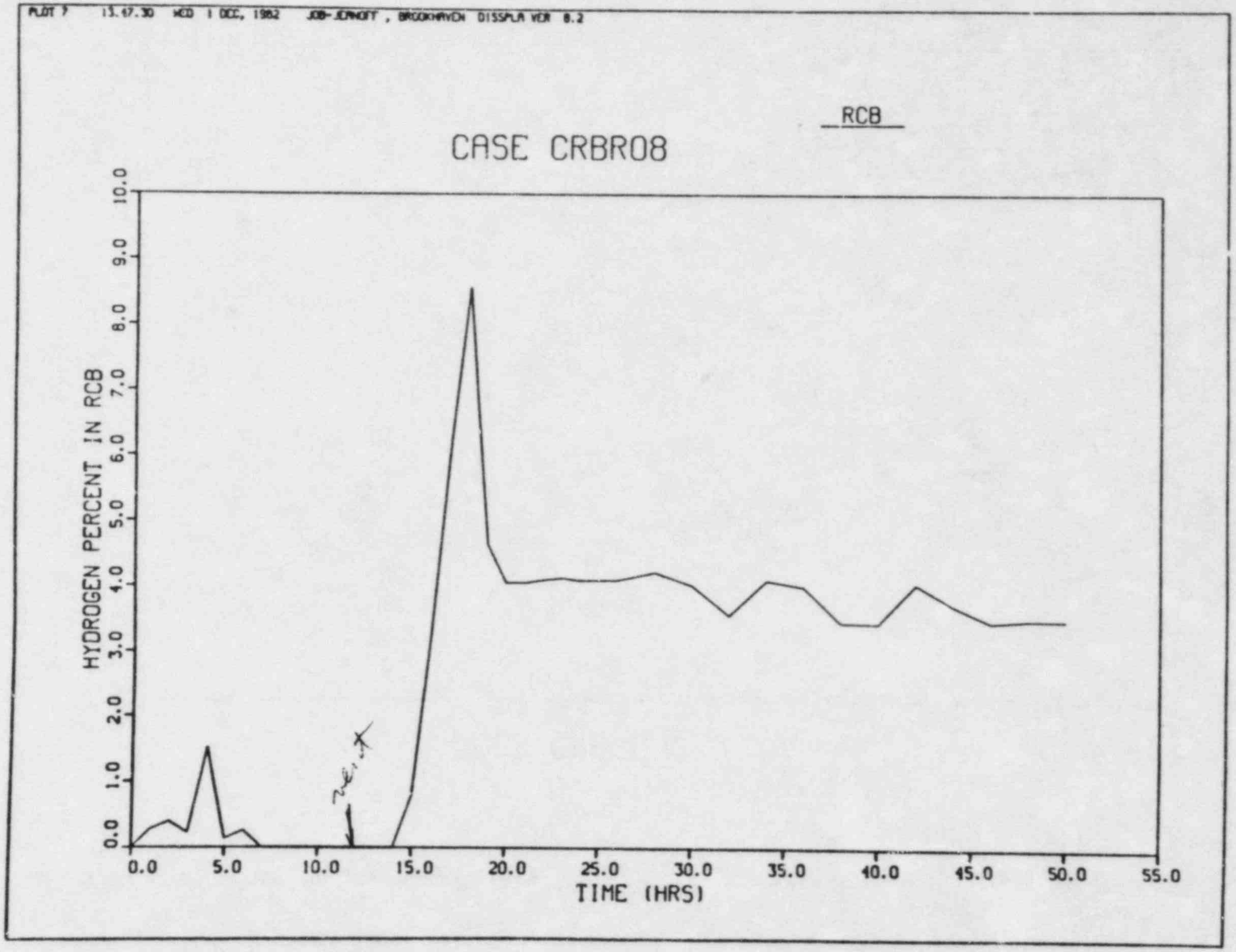
CONTAINMENT STEEL HEMISPHERICAL/ELLIPSOIDAL SHELL
TIME = 0.
TIME = 10.
TIME = 20.
TIME = 36.
TIME = 50.
TEMPERATURE (F)
0.0
100.0
200.0
300.0
400.0
500.0
600.0
700.0
DISTANCE (FT)
0.00
0.01
0.02
0.03
0.04
0.05
0.06
0.07
0.08
0.09
0.10
0.11
0.12
0.13

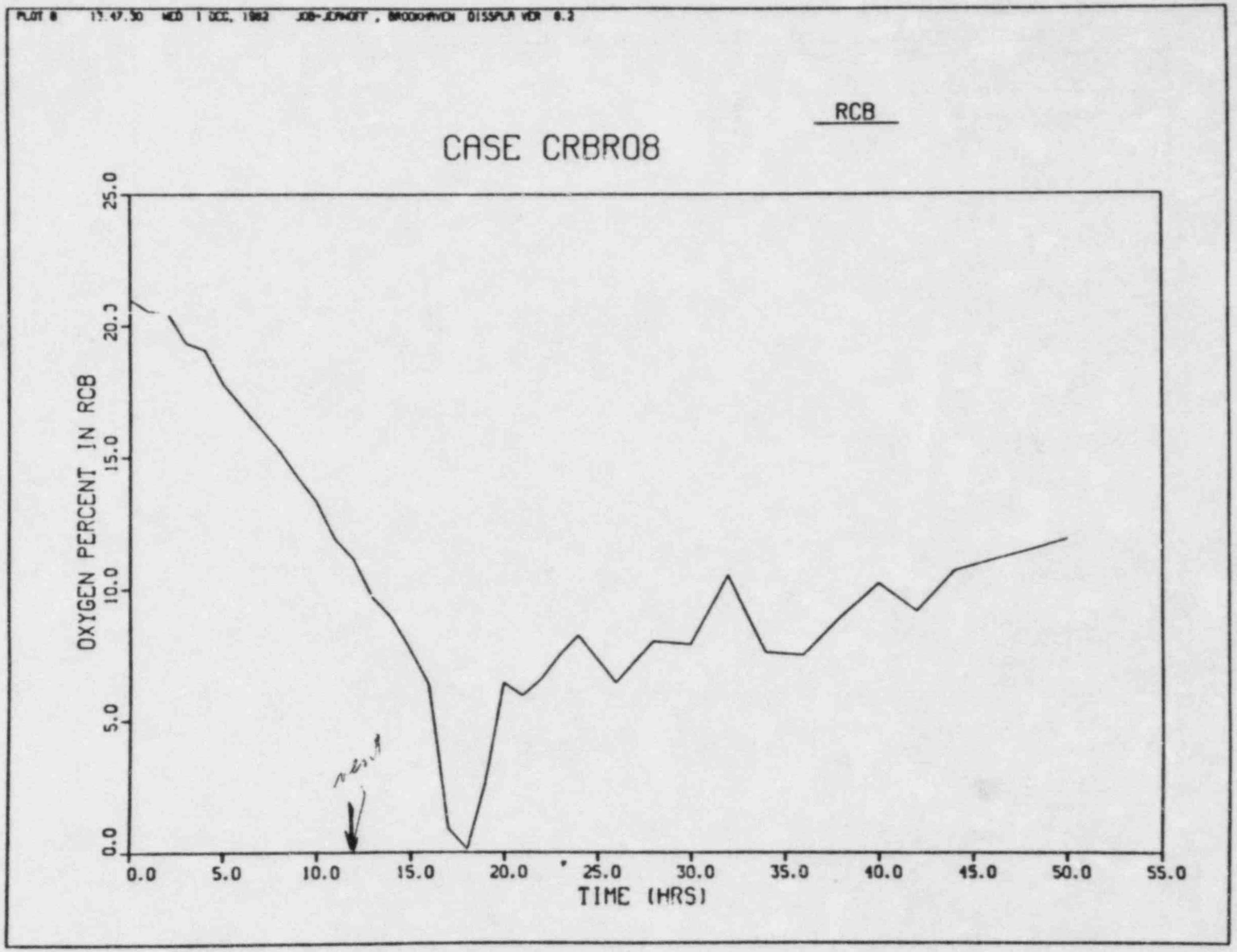

TIME - 0.
TIME - 10.
TIME - 20.
TIME - 36.
TIME - 50.
REACTOR CAVITY LOWER SUBMERGED WALL
TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
DISTANCE (FT)
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.0

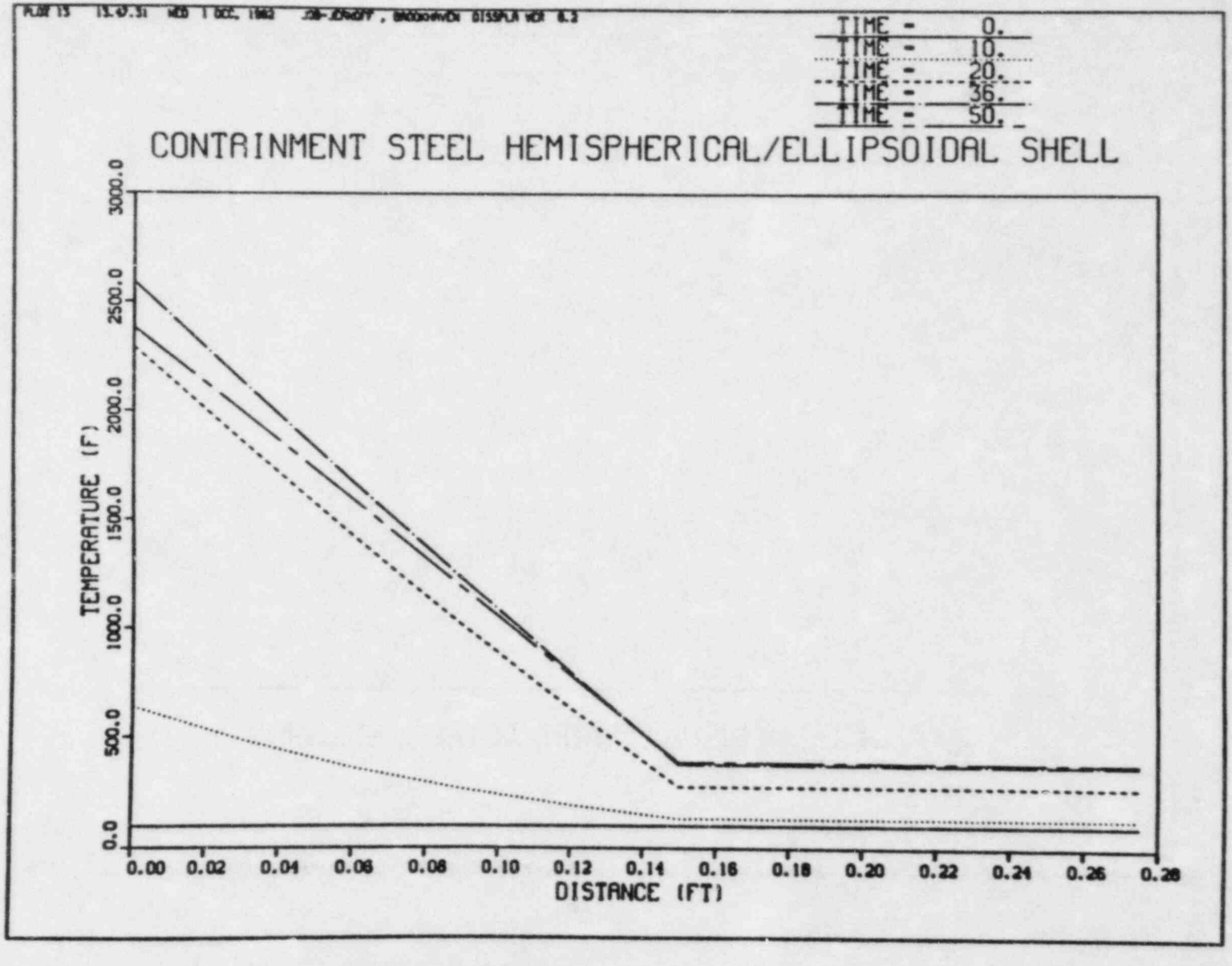

PLOT 1 15.07.28 WED 1 DEC. 1982 JOB-ZARGPP , [illegible] DISSPLA VER 8.2
CASE CRBR08
RC
PHTS
RCB
CELL D
ATMOS TEMPERATURE (F)
0.0
500.0
1000.0
1500.0
2000.0
2500.0
3000.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0
NaLFP 1 ~

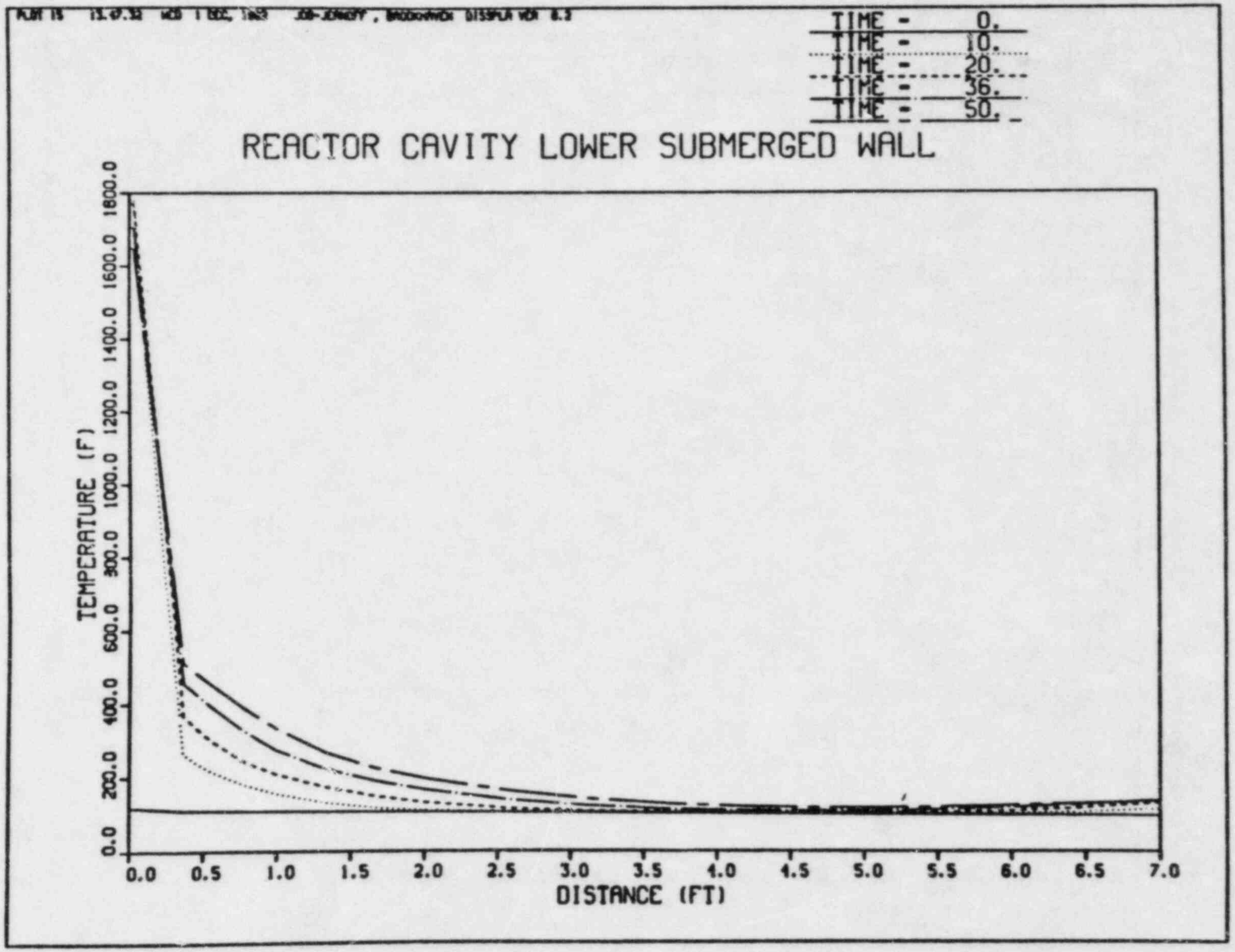


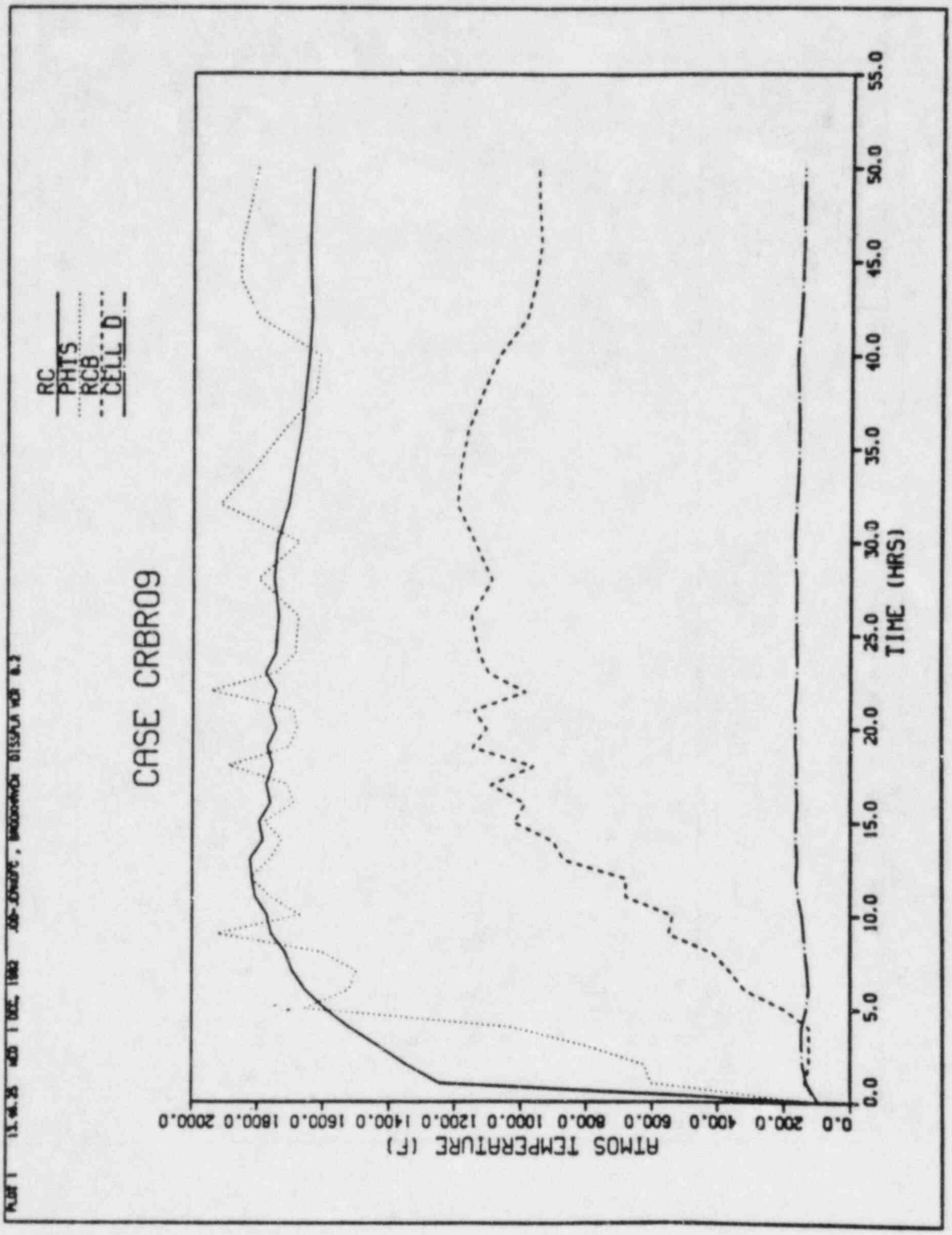
PLOT 3 WED 1 DEC, 1982 DISSPLA VER 8.2
CASE CRBR08
RC
PHTS
RCB
CELL D
PRESSURE (PSIG)
-5.0
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0

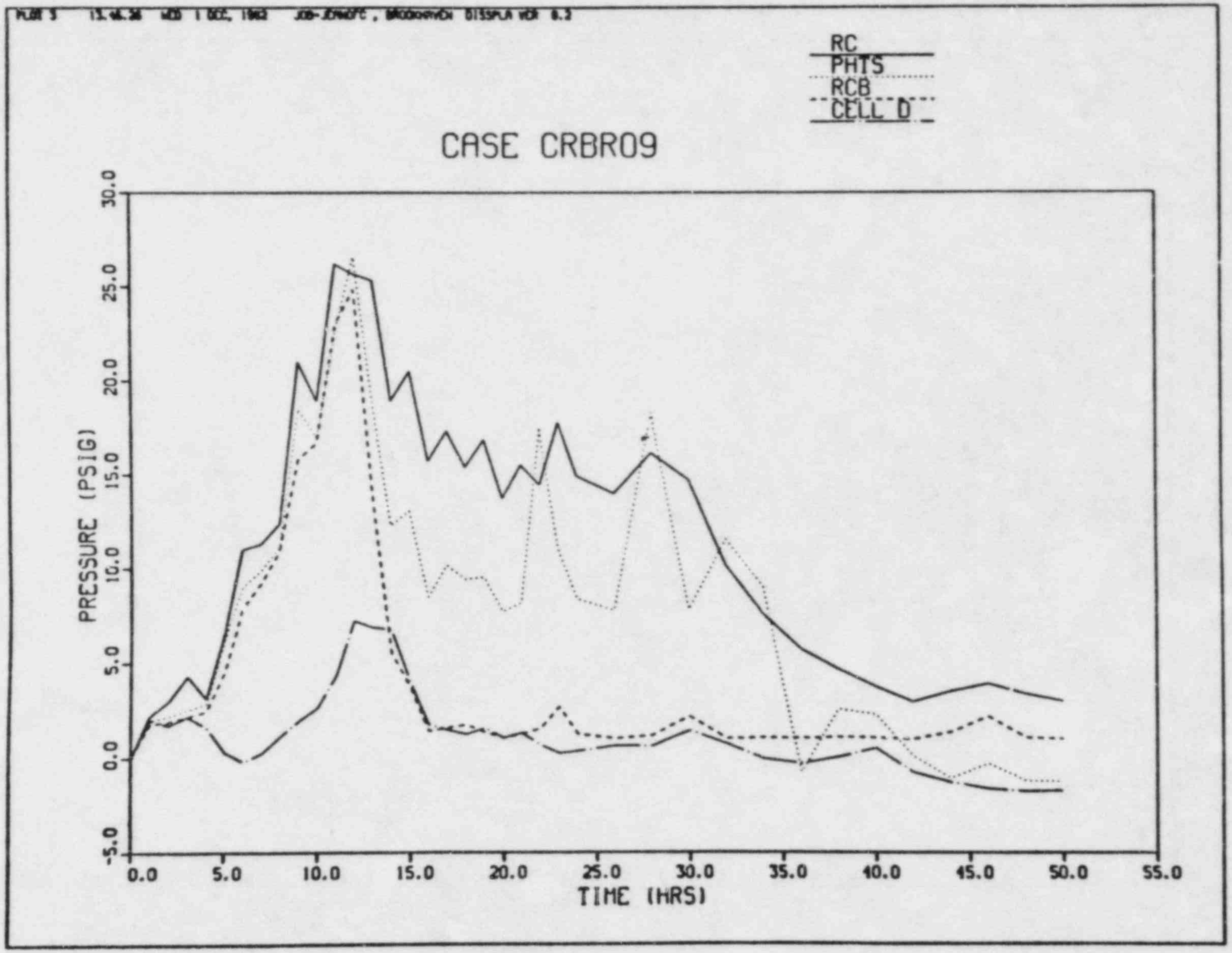

PLOT 7 13.47.30 WED 1 DEC, 1982 JOB-JCWKOFT , BROOKHAVEN DISSPLA VER 8.2
RCB
CASE CRBR08
HYDROGEN PERCENT IN RCB
0.0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0

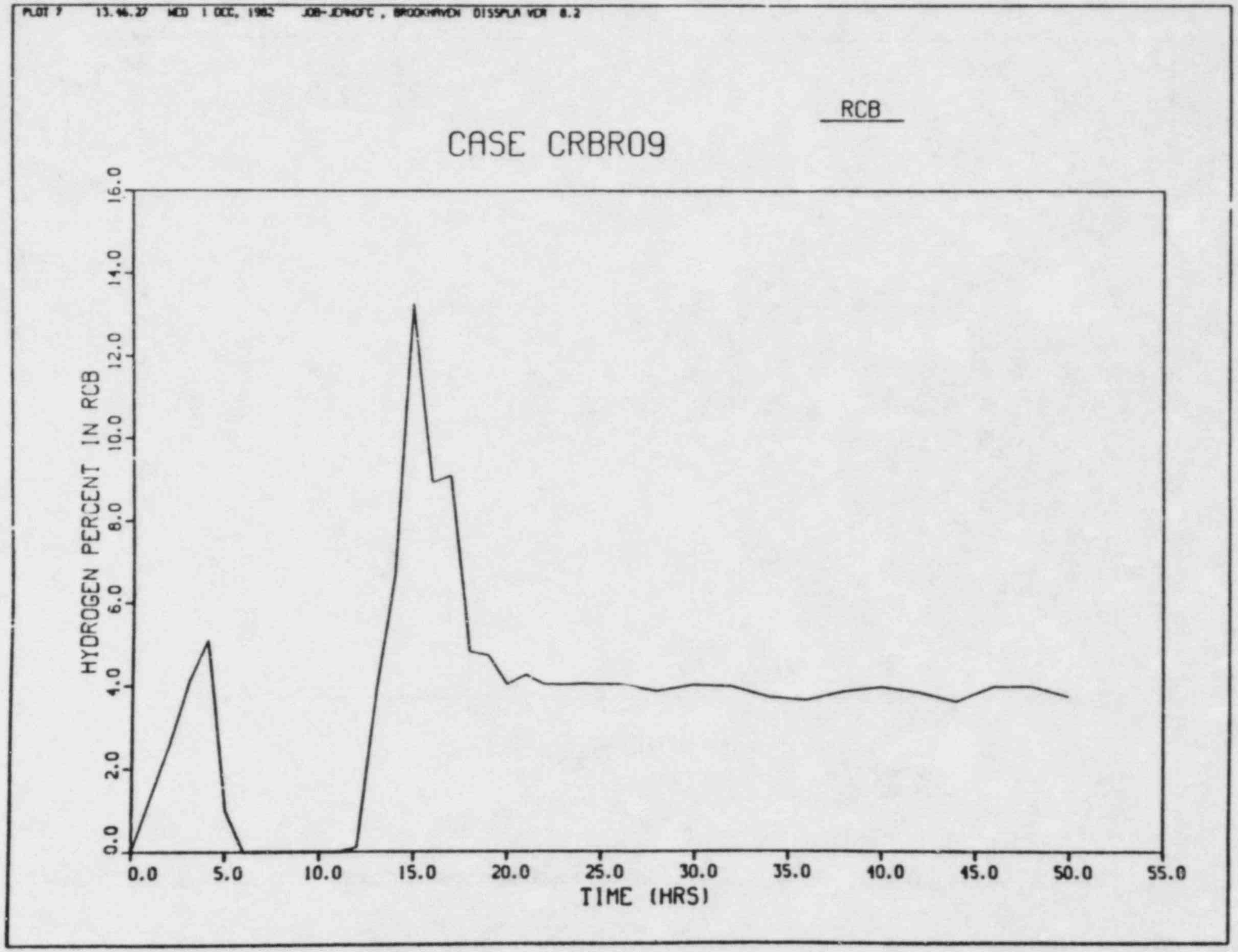
PLOT 8 13.47.30 WED 1 DEC, 1982 JOB-JCRAOFF , BROOKHAVEN DISSPLA VER 8.2
RCB
CASE CRBR08
OXYGEN PERCENT IN RCB
0.0
5.0
10.0
15.0
20.0
25.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0
vent

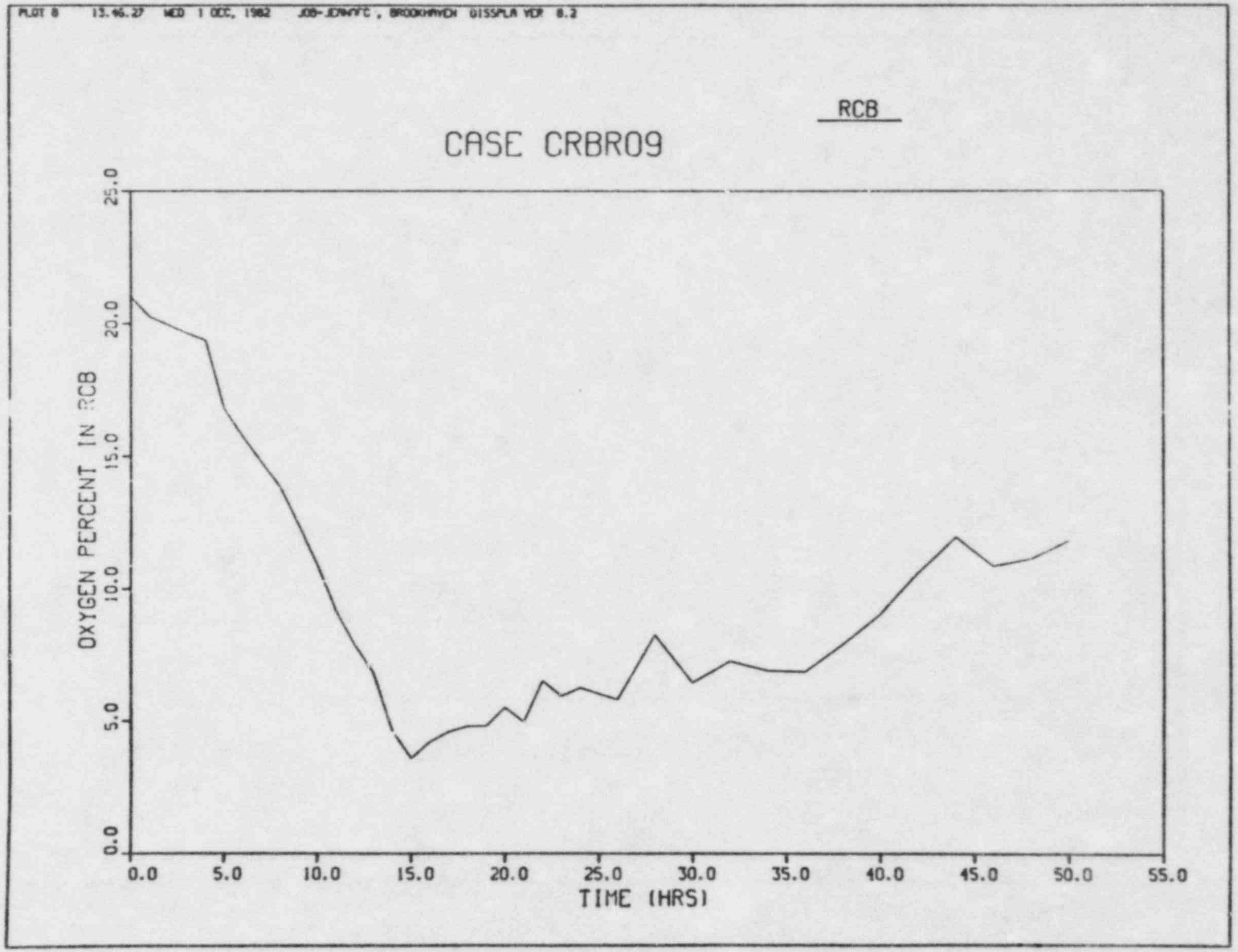

PLOT 15 13.47.31 WED 1 DEC, 1982 [illegible] , BROOKHAVEN DISSPLA VER 8.3
TIME - 0.
TIME - 10.
TIME - 20.
TIME - 36.
TIME - 50.
CONTAINMENT STEEL HEMISPHERICAL/ELLIPSOIDAL SHELL
TEMPERATURE (F)
0.0
500.0
1000.0
1500.0
2000.0
2500.0
3000.0
DISTANCE (FT)
0.00
0.02
0.04
0.06
0.08
0.10
0.12
0.14
0.16
0.18
0.20
0.22
0.24
0.26
0.28

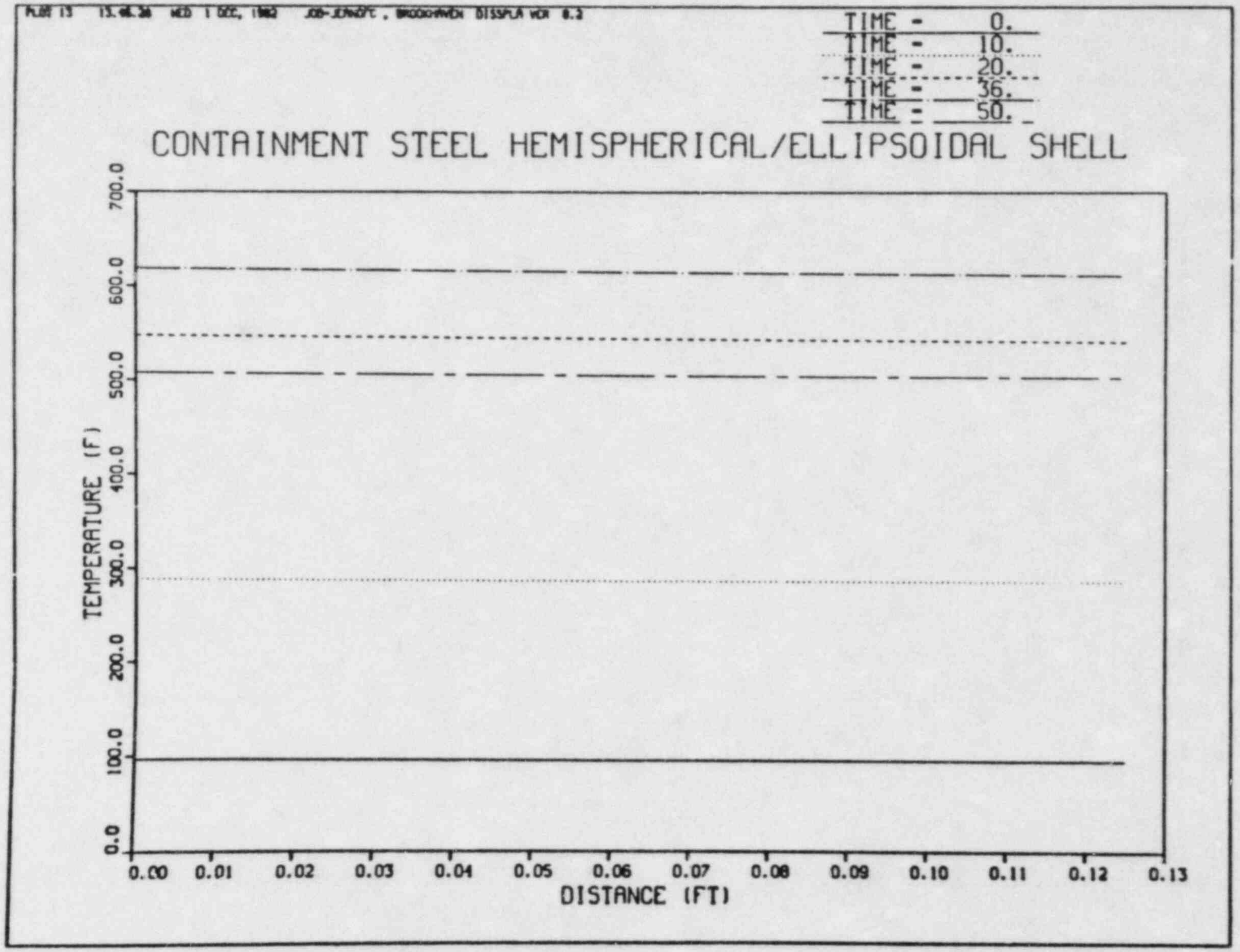

TIME - 0.
TIME - 10.
TIME - 20.
TIME - 36.
TIME - 50.
REACTOR CAVITY LOWER SUBMERGED WALL
TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
DISTANCE (FT)
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.0


CASE CRBR09
RC
PHTS
RCB
CELL D
ATMOS TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
2000.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0


PLOT 3 13.46.38 WED 1 DEC, 1982 JOB-JZWAFC , BROOKHAVEN DISSPLA VER 8.2
CASE CRBR09
RC
PHTS
RCB
CELL D
PRESSURE (PSIG)
30.0
25.0
20.0
15.0
10.0
5.0
0.0
-5.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0


PLOT 7 13.46.27 WED 1 DEC, 1982 JOB-JCMOFC , BROOKHAVEN DISSPLA VER 8.2
RCB
CASE CRBR09
HYDROGEN PERCENT IN RCB
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0
TIME (HRS)

PLOT 8 13.46.27 WED 1 DEC, 1982 JOB-JCWYFC , BROOKHAVEN DISSPLA VER 8.2
CASE CRBR09
RCB
OXYGEN PERCENT IN RCB
0.0
5.0
10.0
15.0
20.0
25.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0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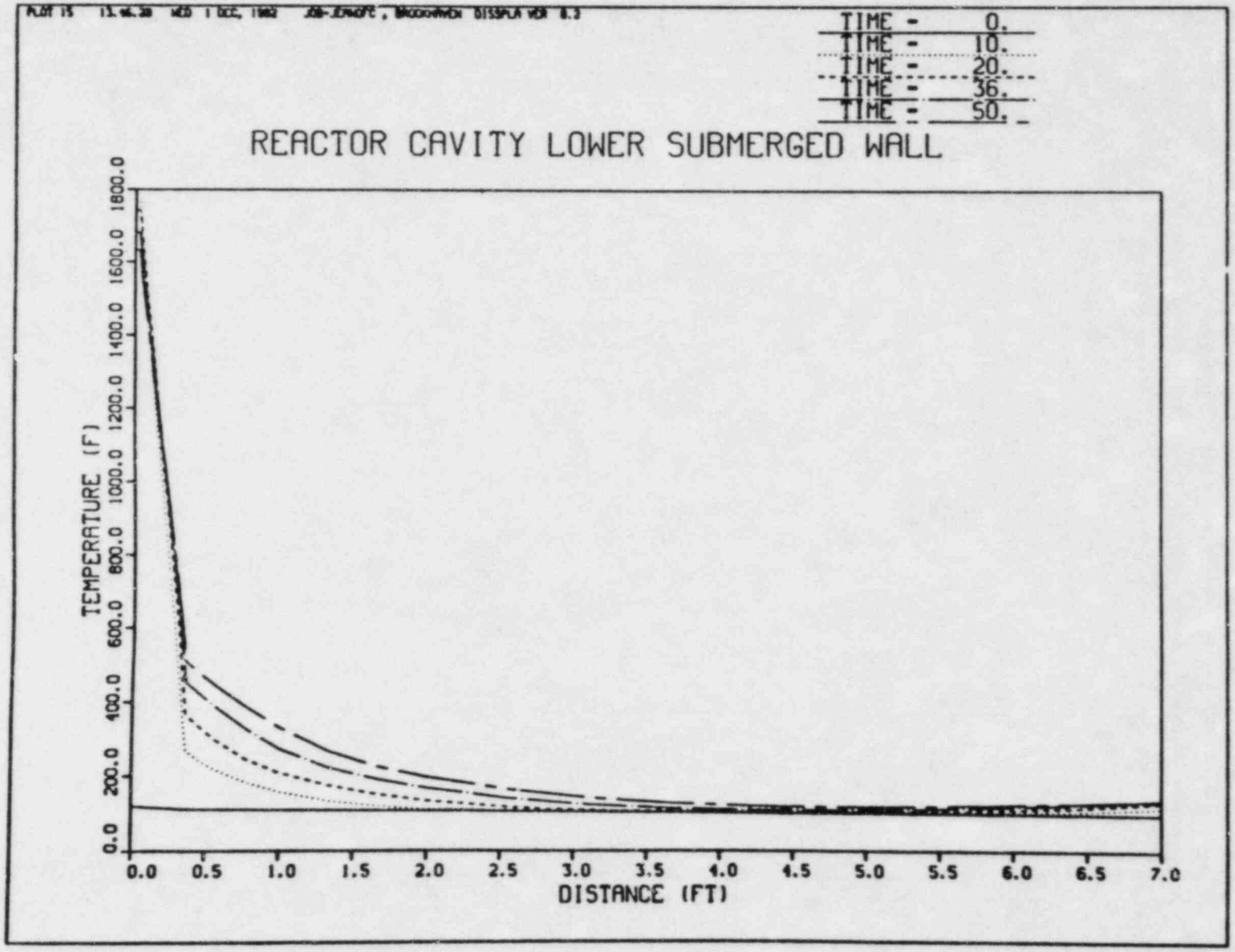
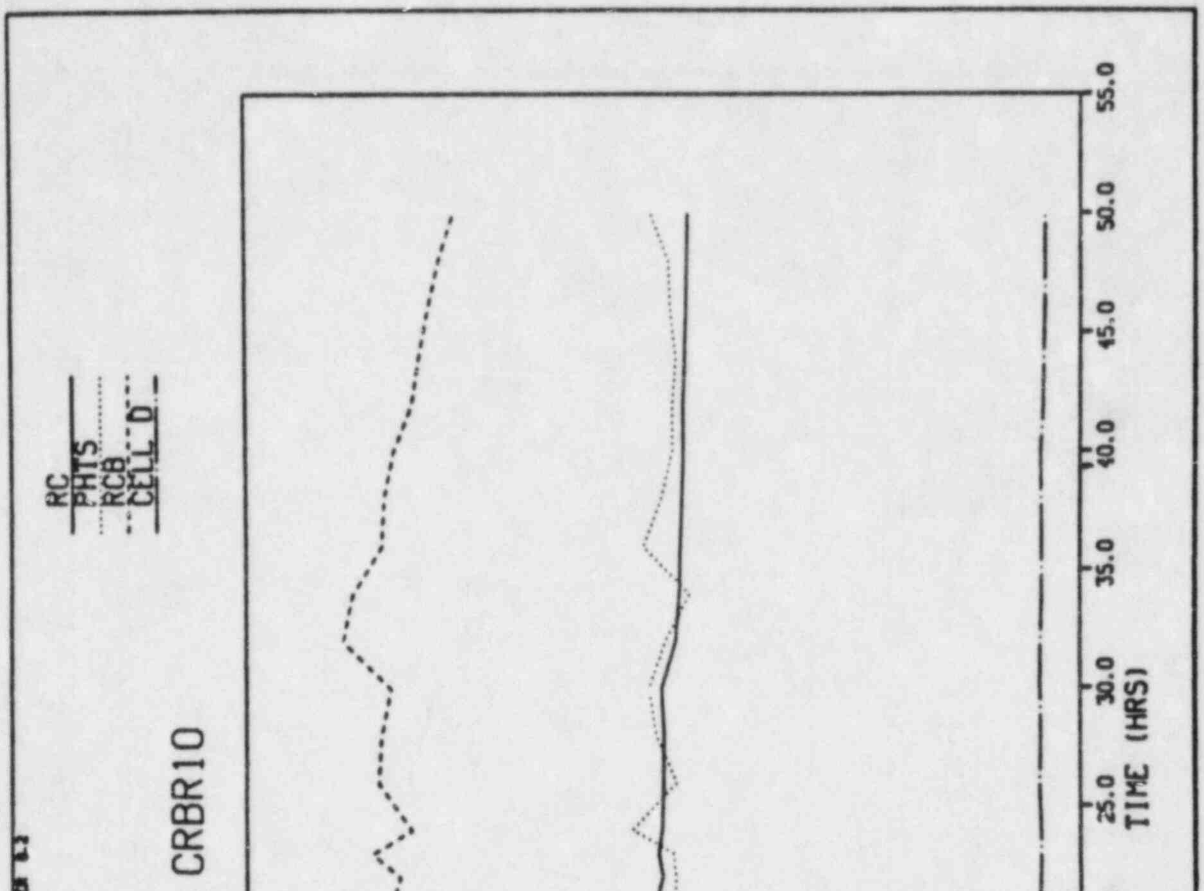
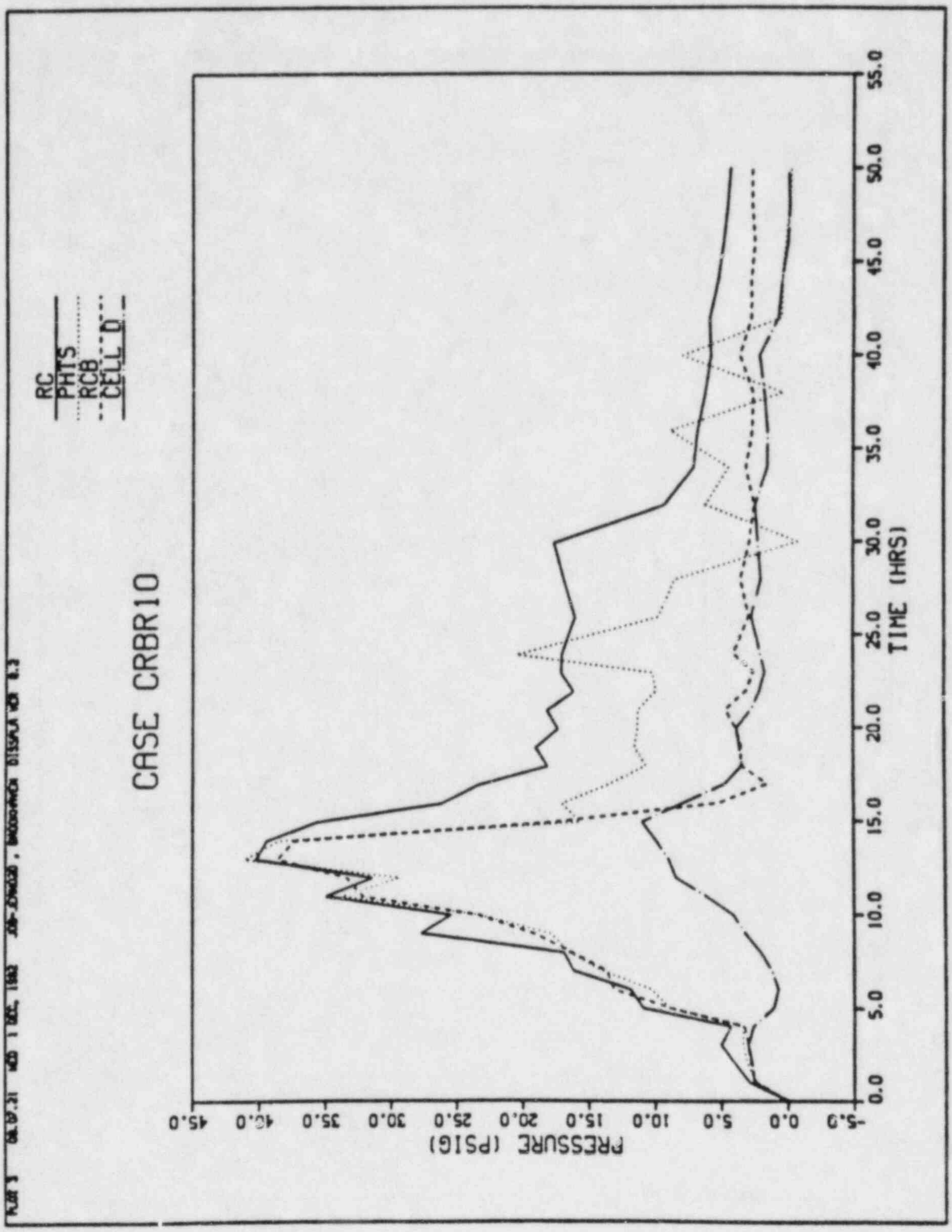
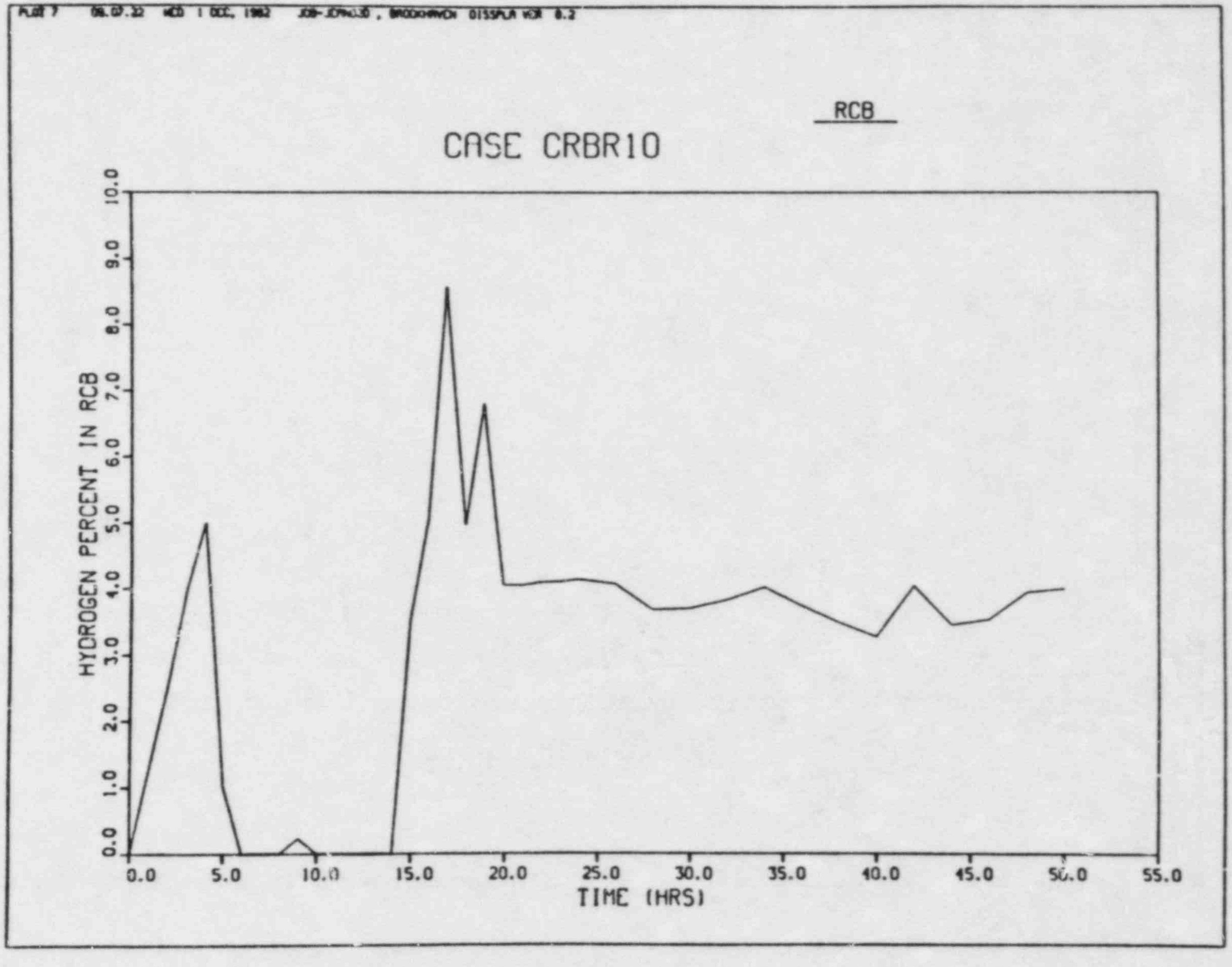
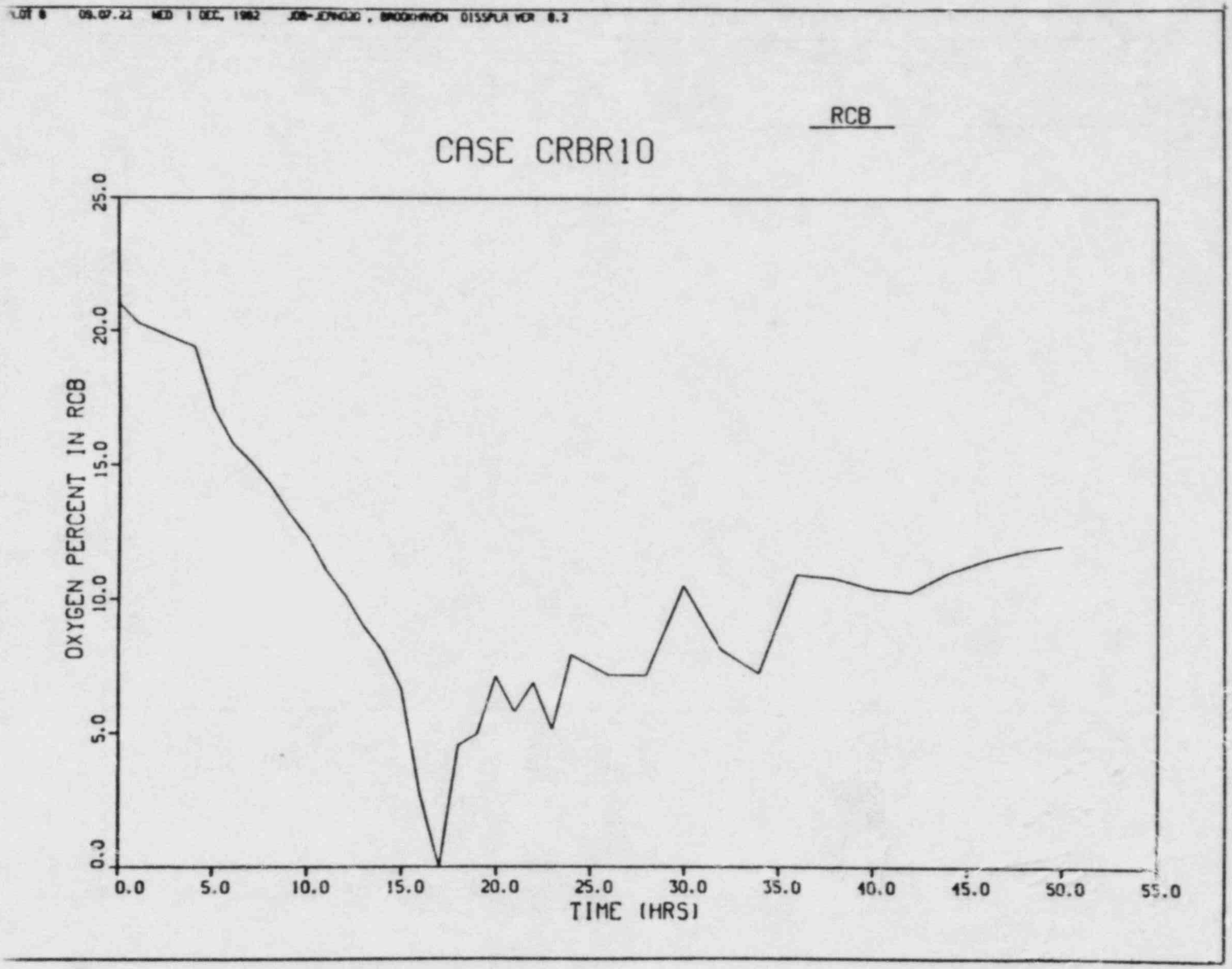
TIME - 0.
TIME - 10.
TIME - 20.
TIME - 36.
TIME - 50.
CONTAINMENT STEEL HEMISPHERICAL/ELLIPSOIDAL SHELL
TEMPERATURE (F)
0.0
100.0
200.0
300.0
400.0
500.0
600.0
700.0
DISTANCE (FT)
0.00
0.01
0.02
0.03
0.04
0.05
0.06
0.07
0.08
0.09
0.10
0.11
0.12
0.13

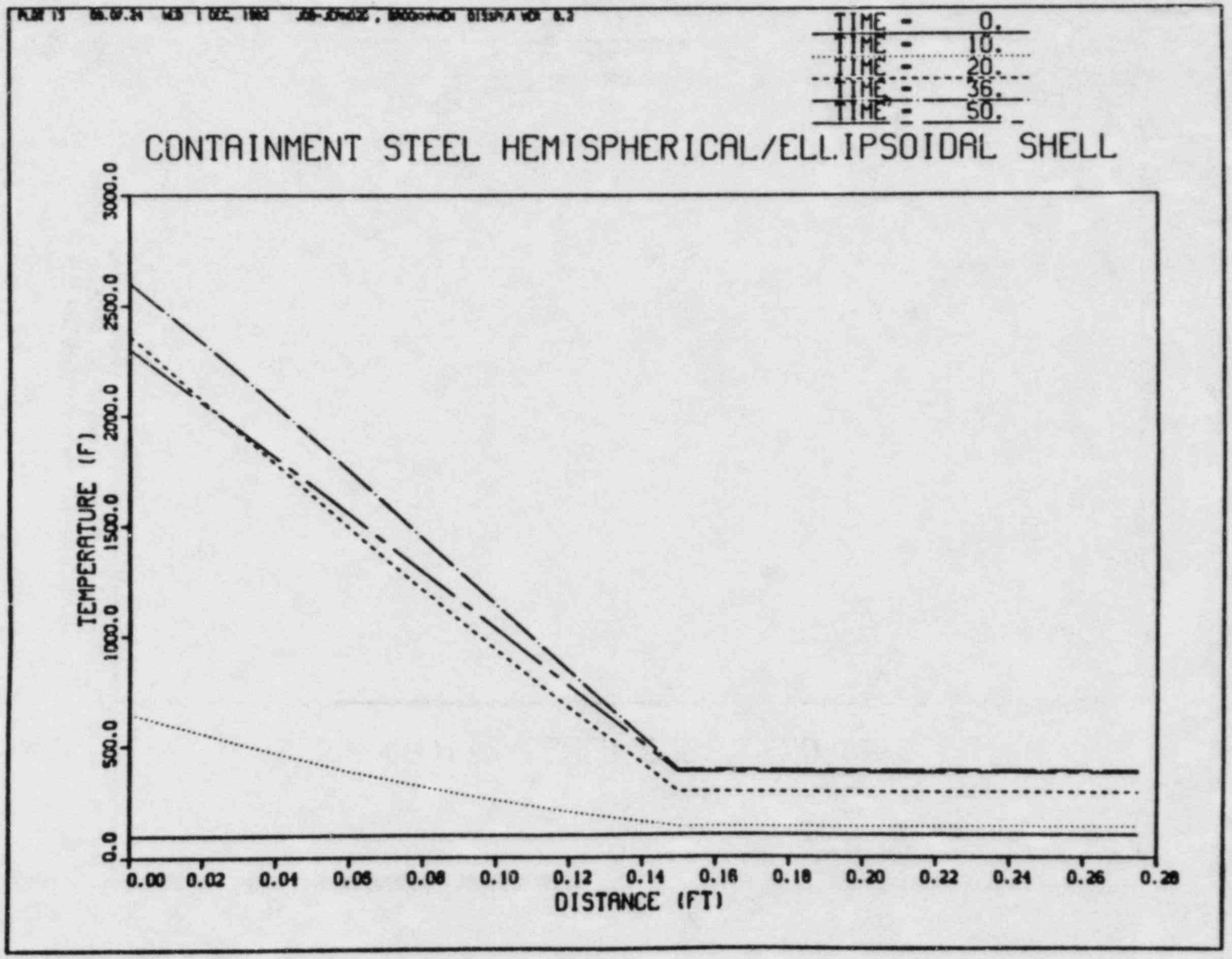

PLOT 15 13.46.20 WED 1 DEC, 1982 JOB-JCWAFE, BROOKHAVEN DISSPLA VER 8.3
TIME - 0.
TIME - 10.
TIME - 20.
TIME - 36.
TIME - 50.
REACTOR CAVITY LOWER SUBMERGED WALL
TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
DISTANCE (FT)
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.0


CASE CRBR10
RC
PHTS
RCB
CELL D
ATMOS TEMPERATURE (F)
0.0
500.0
1000.0
1500.0
2000.0
2500.0
3000.0
3500.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0


CASE CRBR10
RC
PHTS
RCB
CELL D
PRESSURE (PSIG)
TIME (HRS)
-5.0
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0


PLOT 7 08.07.22 WED 1 DEC, 1982 JOB-JCPH130 , BROOKHAVEN DISSPLA VER 8.2
RCB
CASE CRBR10
HYDROGEN PERCENT IN RCB
0.0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0

PLOT 8 09.07.22 WED 1 DEC, 1982 JOB-JEM020 , BROOKHAVEN DISSPLA VER 8.2
CASE CRBR10
RCB
OXYGEN PERCENT IN RCB
0.0
5.0
10.0
15.0
20.0
25.0
TIME (HRS)
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
50.0
55.0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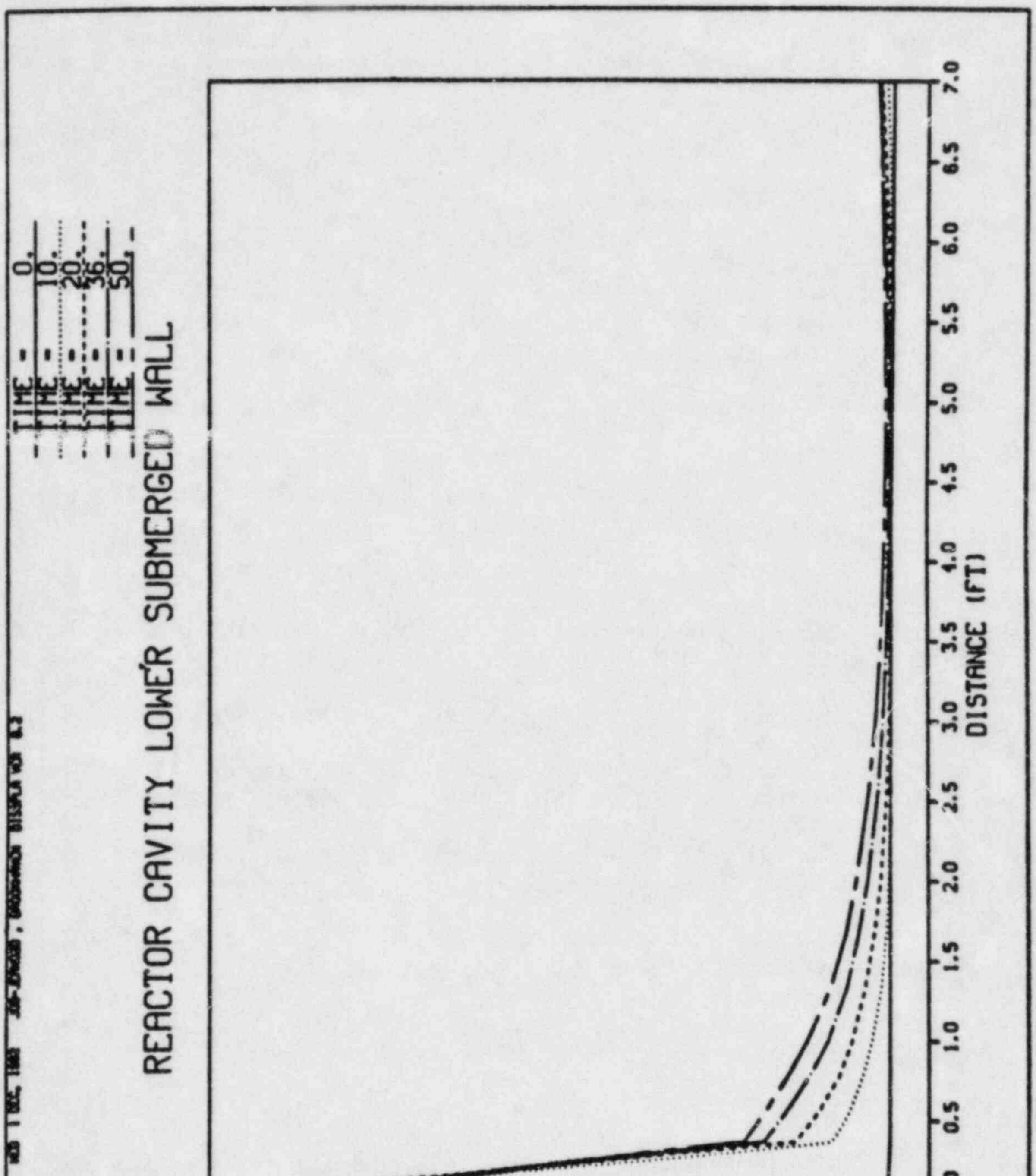
TIME - 0.
TIME - 10.
TIME - 20.
TIME - 36.
TIME - 50.
CONTAINMENT STEEL HEMISPHERICAL/ELLIPSOIDAL SHELL
TEMPERATURE (F)
0.0
500.0
1000.0
1500.0
2000.0
2500.0
3000.0
DISTANCE (FT)
0.00
0.02
0.04
0.06
0.08
0.10
0.12
0.14
0.16
0.18
0.20
0.22
0.24
0.26
0.28


REACTOR CAVITY LOWER SUBMERGED WALL
TIME = 0.
TIME = 10.
TIME = 20.
TIME = 36.
TIME = 50.
TEMPERATURE (F)
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
2000.0
DISTANCE (FT)
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.0

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13. R. D. Gasser, "Evaluation of CACECO Mass-Energy Balance and Venting Models," BNL Informal Report (not published).

14. SACRD" Database File, Table V, ORNL.

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ATTACHMENT 2

FINAL TECHNICAL EVALUATION REPORT

OF

THERMAL MARGIN BEYOND DESIGN BASIS FEATURES

by

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Frederick Ju
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Devona B. Jensen

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Los Alamos, New Mexico 87545

February, 1983

I. INTRODUCTION

In designing the Clinch River Breeder Reactor Plant (CRBRP) the applicant is considering the possibility of Core Disruptive Accidents (CDAs). The probability of CDAs occurring is very low so these accidents are excluded from the Design Basis Accident (DBA) spectrum. However, because of potential consequences the applicant has included several features in the design to prevent CDA initiation and other features to mitigate consequences of the accidents should they occur. CDAs can terminate either energetically or nonenergetically with either case leading to potential of the core melting and penetrating the reactor vessel and guard vessel. The purpose of this report is to evaluate the CRBRP features that have been added or enhanced to mitigate the consequences if such a core melt should occur. The features provide what the applicant is calling Thermal Margins Beyond the Design Basis (TMBDB). Documents reviewed for the evaluation include Refs. 1, 2, and 3 as primary sources of information.

Systems or components specifically added to CRBRP for TMBDB are the following:

1. reactor cavity (RC) vent system,
2. containment cleanup system,
3. annulus cooling system,
4. containment vent and purge system, and
5. instrumentation and radiation monitoring equipment.

Systems or components with augmented capabilities for TMBDB are:

1. dual control room air intakes,
2. RC and pipeway cell liners,
3. liner vent system,
4. guard vessel support,
5. RC to head access area seals,
6. RC penetrations and Recirculating Gas Cooling System (RGCS),
7. containment-confinement systems,
8. emergency electrical power system, and
9. reactor containment internal structures.

In this Technical Evaluation Report (TER) we evaluate all of the systems and components except for the dual control room air intakes and the emergency electrical power system. In the following sections of this TER the systems and components are described, design criteria and requirements are evaluated, the applicants methods for analysis are described and evaluated, and the potential for achieving an acceptable design is discussed.

Because of the low probability of a CDA, we do not recommend the use of the conservative criteria and analysis methods required for evaluating Engineered Safety Features (ESFs) for Design Basis Accidents. Requirements and criteria are judged on the basis of the function that the components serve and the time that they must perform those functions.

II. GENERAL REQUIREMENTS

General requirements for TMBDB features are given in detail in Ref. 1 and are used as a part of the measure of acceptability of individual system requirements and criteria. The general requirements and criteria imposed by Ref. 1 on the CRBRP are the following:

A. General Requirements

1. The design shall provide margins and features to mitigate the consequences of a hypothetical core meltdown.
2. These features shall be designed to be consistent with safety, reliability, maintainability, and availability of the total plant.
3. These features are not ESFs because they are not required to mitigate any Design Basis Event; however, these features shall be designed to the specifications and requirements associated with Safety Class 3 components and systems.
4. TMBDB components shall be designed so that appropriate testing and/or inspection can be performed after installation and periodically to provide reasonable confidence that functional capability is maintained throughout the plant life. The containment isolation valves shall be designed to be testable in accordance with CRBRP Design Criteria 43, 44, and 45.
5. The TMBDB controls and associated instrumentation shall be physically separated from other controls in the reactor control room. Inadvertent actuation of the TMBDB features shall be prevented by appropriate provisions such as administrative controls.
6. There is not a requirement to meet the allowable site boundary or low population zone doses of 10CFR100 or the control room dose of 10CFR50 under TMBDB conditions.

B. Acceptance Criteria

1. The public risk from accidents beyond the design base shall be comparable to that from light water reactors for events beyond the design base with similar probability of occurrence.
2. Containment integrity shall be maintained without venting following initiation of an accident leading to core meltdown for a period of time sufficient to allow evacuation procedures to be implemented. Per NRC guidance, the period is taken as 24 hours.

It is beyond the scope of this report to evaluate the Acceptance Criteria. However, based on the first acceptance criterion we have compared the General Requirements to those proposed for systems for mitigating the effects of core melt accidents at the Zion and Indian Point nuclear power plants.[4] One requirement not mentioned in the list of General Requirements but required in Ref. 4 is that mitigation systems shall be capable of manual operation and control from the control room. This requirement is listed as an individual system requirement for all systems or components that are specifically for TMBDB. Criterion eight is met with the requirement that Class 1E electrical power be provided to all TMBDB systems and components that require electrical power to perform their post-accident functions. Other criteria from Ref. 1 are either directly met or are met through the very General Requirement 2, listed above.

General Requirement 6 listed above is generally in line with the NRC philosophy used in evaluating Beyond Design Basis Accidents. However, the NRC staff is using the doses listed in 10CFR100 as general guidelines in evaluating requirements for TMBDB features. If the 10CFR100 doses are not met, information must be provided to show that overall safety is still similar to that for Light Water Reactors.

Maintenance and testing requirements given for TMBDB features have been reviewed and are acceptable. They are generally either met directly by design base procedures or simple extensions of these procedures. For features specifically added for TMBDB, the systems are completely tested for functional capability approximately once a year with tests always occurring during a shutdown period when a CDA could not occur. The features will also be given pre-service full functional tests after construction.

The 24 hour evacuation period given in the second acceptance criterion is not a strict NRC limit. The NRC interpretation of the required time for which containment integrity must be maintained is given in Section A.1.3 of the SER.

III. EVALUATION OF INDIVIDUAL FEATURES

A. Reactor Cavity Vent System

The RC vent system is designed to prevent overpressurization of the RC after the reactor core penetrates the reactor and guard vessels during the TMBDB scenario. The system is designed so that vented cavity gases are forced to pass through the pipeway cells to take advantage of the pipeway cell heat capacity for absorbing heat before the gases are vented to the containment above the operating floor. Flow paths through the pipeway cells are designed to maximize heat exchange by causing no less than 25% and no more than 50% of the total gas flow to pass through any one cell.

The system is actuated by a redundant set of rupture disks. The disks are set to rupture at a pressure above any pressures predicted for design basis accidents. Isolation of the rupture disks is provided by remote manually operated gas valves located between the cavity and the rupture disk assembly. In the TMBDB scenario it is assumed that only one of the rupture disks breaks.

1. Feature Requirements. The Applicant has specified the following requirement for the RC vent system.

a. To prevent RC structural and liner failure by over-pressurization, the vent system shall provide redundant flow paths between the RC and reactor containment building (RCB) when the pressure differential between the RC and containment exceeds 11.5 ± 1.5 psi. After passive initiation, the vent path shall remain open.

b. The vent system shall have a pressure drop of less than 0.1 psi with a flow rate of 4000 lb/hr of gases, a density of 0.03 lb/ft^3, and a viscosity of 0.05 lb/ft-hr. It shall remain functional if up to 450 lbs of sodium oxide aerosol enter the vent at a maximum rate of 8000 lb/hr.

c. The vent system shall be capable of performing all of its intended functions for 150 hrs in the presence of gases and vapors consisting of Ar, N_2, H_2, Na, fission products, and compounds resulting from fission product reactions.

d. To ensure that the heat capacity of the pipeway cells is employed, a minimum of 25% of the mass flow into the pipeway cells shall enter each pipeway cell.

e. To allow sodium that condenses in the pipeway cells to drain back into the RC, two drain pipes shall be provided between each pipeway cell and the RC, at the elevation of the pipeway cell floor. Each drain pipe shall be capable of a minimum flow rate of 2000 lb/hr of sodium at its boiling point with a pressure head of 0.2 ft of sodium.

f. To ensure that the flame at the vent exit does not approach the containment vessel, the pipeway cell to containment vent line diameter shall not exceed 12 in.

2. Discussion. The requirement for a disk that will rupture at 11.5 ± 1.5 psi can be met with commercially available rupture disks. For example, Ref. 5 lists a 12 in. disk designed to operate in atmospheric conditions that has a disk burst range of 10 to 13 psi.

Potential for plugging the vent system has been thoroughly evaluated by the applicant. Review of sodium-concrete reaction tests[6] shows that particulate resulting from the sodium-concrete reactions generally remains in the reaction product layer beneath the sodium pool. The only viable method for transporting particulate through the sodium pool is by mechanical entrainment, which is very inefficient. The only materials present in any appreciable quantity above the sodium pool during the tests were the inert cover gas, hydrogen, and sodium vapor. For CRBR enough oxygen could exist in the reactor to form up to 450 pounds of sodium oxide. This is not enough to cause plugging. Sodium hydride will form on some of the cooler surfaces in the RC. Because it is not stable at temperatures above approximately 420°C, it will not form enough particulate to be a plugging threat.

In Appendix G.3 of Ref. 1 several parameter studies are presented that were performed to determine the effect of maloperation of the RC vent system. The applicant concludes that the system has sufficient margin to accommodate 98% plugging of the vent line, full plugging of one line interconnecting pipeway cells, and sodium vapor flashing from the cavity sodium pool. For 98% blockage, which the applicant states to be the probable upper bound of acceptable blockages, the RC pressure is 32 psig compared to a design pressure of 35 psig, which is acceptable. When blockage occurs, the RC and pipeway cell walls will be heated to higher temperatures than for the base case (a difference of 100°F). However, this is not significant, especially since the resulting thermal stresses are secondary.

The major effect of plugging one vent line connecting pipeway cells is to increase the containment building atmosphere temperature. Full plugging of one line leads to an increase in containment temperature and pressure of 140°F and 1.7 psig, respectively. For reasons discussed above, it is unlikely that plugging would occur. If one vent line connecting pipeway cells plugged it is unlikely that the other would because of the amount of solid particulate available.

B. Containment Cleanup System

The containment cleanup system is provided to allow pressure relief for the RCB during the TMBDB scenario. By venting through this system, RCB pressures can be maintained at an acceptable level and radiological effects minimized.

The system consists of an air washer (quench tank), a venturi jet scrubber, and a high-efficiency wetted-fiber-bed scrubber and redundant blowers. Gases and aerosols from the RCB must pass through these components in sequence. The quench tank ensures that all sodium oxide is converted to sodium hydroxide and that the temperature is reduced from 1100°F to 160°F. The venturi scrubber removes large particulates and the wet-bed scrubber removes smaller particulates and provides sufficient contact to remove condensible vapor-phase species.

The wet-bed fibrous scrubber filter system requires protected water storage with recycled solutions. A heat exchanger provides cold solutions for the scrubbers.

1. Feature Requirements. The applicant has specified the following requirements for the containment cleanup system.

a. The containment cleanup system efficiency shall be a minimum of 99% for vented materials in the solid or liquid state, 97% for vapors (NaI, SeO_2, and Sb_2O_3) subject to condensation in the cleanup system, and 0% for noble gases. These efficiencies shall apply when subjected to the vent rates on Fig. 2-7 and containment atmosphere temperatures on Fig. 2-5 of Ref. 1 with a containment atmosphere density of 0.07 lb/ft^3. It shall be capable of performing all of its intended functions in the presence of Ar, N_2, H_2, H_2O, CO, CO_2, O_2, Na_2O, Na_2O_2, NaOH, Na_2CO_3, fission products, and compounds resulting from fission product reactions.

b. The containment cleanup system shall remain functional at an aerosol mass flow rate of up to 5 600 lb/hr and a total mass of 300 000 lb of aerosol entering the cleanup system. The principal constituents of the aerosol are NaOH and Na_2O, the proportions of which can vary from 0 to 100% of the aerosol, and Na_2CO_3 which can vary from 0 to 8% of the aerosol.

The aerosol particle properties are

Mass Mean Radius (microns)	$5 < r_{50} < 10$
Aerodynamic Equivalent Radius (microns)	$2.3 < AER < 4.7$
Density (g/cc)	$2.1 < \rho < 2.5$
Mass Geometric Standard Deviation	$3.0 < \sigma < 3.5$

Aerodynamic equivalent radius is based on $AER = r_{50} (\rho\alpha)^{0.5}$ where $\rho = 2.21$ and $\alpha = 0.1$.

c. The containment cleanup system shall remain functional at fission products power levels in the accumulated filter aerosol of:

Time (h)	Fission Product Power (MW)
0	0
24	3.1×10^{-5}
48	0.16*
96	0.16*
240	0.11
720	0.05

*Maximum value

d. The containment cleanup system design shall be capable of performing all its intended functions with the following chemical and physical states of the 10 most radiologically significant fission products in the containment atmosphere:

MAXIMUM PERCENTAGES OF THE FISSION PRODUCTS BY CHEMICAL AND PHYSICAL FORM

	Elemental		Oxide	
Element	Vapor	Liquid or Solid	Vapor	Liquid or Solid
Se	1%	1%	100%	100%
Rb	1	1	1	100
Sr	1	1	1	100
Zr	1	1	1	100
Sb	1	1	100	100
Te	1	1	1	100
Cs	1	1	1	100
Ba	1	1	1	100
Ce	1	1	1	100
			NaI	
I	1	1	33	100

e. The exhaust from the containment cleanup system shall have a temperature compatible with operation of the TMBDB Exhaust-Plant Effluent Radiation Monitoring System.

f The containment cleanup system operation shall be by remote manual actuation from the control room.

2. Discussion. Conditions in the RCB during cleanup system operation are the starting points for design. These conditions are the result of a complicated scenario and chain of assumptions starting with the behavior of the

fuel/debris/sodium melt. Localized effects, such as small areas of particularly severe sodium/fuel/concrete reaction, have been neglected. The debris bed is assumed to be self-leveling. The fission products and plutonium reaching containment are critical factors for meeting the dose acceptance criteria.

The offgas stream from the cleanup system is the source term for dose calculations. Feature requirements 1, 2, and 4 are designed to ensure acceptance of the limiting dose. By calculating from doses to an acceptable effluent term, a simplified criterion could be generated.

Components similar to those used in the containment cleanup system are readily available in industry. They are widely used and provide satisfactory performance when they are sized for the specific filtration demand. Because no development of new technology is required, the CRBR containment cleanup system could quite feasibly handle filtration of the containment atmosphere in the case of the CDA. Although the potential for proper filtration exists, we feel that a closer look should be taken to ensure that the acceptable effluent release levels can be reached. For instance, the HEDL CSTF tests[7,8] have been cited as supporting evidence for CRBR filtration system performance, and no similitude relationship between the components of the two systems has been established. The size distribution of the aerosol and the total aerosol loading in the CSTF tests are different from the expected size distribution and total aerosol loading in the CRBR cleanup system. We see no need for a formal comparison to be done before issuing of a Construction Permit. However, we do recommend a thorough similitude analysis at the FSAR stage.

The computer codes used in the evaluation of the scenario have given specific quantities for the expected releases of aerosol products to the cleanup system. The CACECO code calculates the amount of sodium aerosols generated from the concrete-sodium reactions as a function of time. CACECO results are used as input for HAA-3, which is an aerosol agglomeration code used both in LMFBR and LWR accident analysis. The assumptions used in the codes are reasonable. The mechanisms for aerosol agglomeration and transport are basically the same mechanisms that have been widely used in aerosol studies for light water reactor accidents.[9]

Variations in the scenario including a limiting worst case have been examined by the applicant. In two cases, most of the fission products are assumed to be instantaneously released. In two other cases, the release fraction is a function of time. When concrete-sodium reactions are extreme or severe, the codes indicate that although the aerosol generation rate is much higher, accelerated agglomeration of the particles takes place. Settling out of these agglomerated particles is a major mechanism for decreasing the amount of suspended aerosol, including fission products, that would otherwise go through the cleanup system. In the case of a less extreme concrete-sodium reaction rate, less aerosol is generated in the first place.

The applicants predict that less than 350 g of plutonium will reach upper containment during the entire TMBDB scenario. The NRC staff believes that this amount could be significantly higher (as high as 3200g). However, as described in Section A.4.9 of the SER, the amount passed on to the cleanup system will still be less than that required for accidental criticality.

Two noncondensible gases, CO_2 and H_2, are formed during sodium-concrete reactions. These gases have very little effect on the release fractions of solid and volatile fission products from the boiling sodium pool except for the gas sparging mechanisms. Carbon dioxide readily reacts with sodium, and does not leave the pool as a gas. A significant amount of hydrogen does leave the pool as a gas, but mechanical entrainment of particles is not an effective means of transport, and therefore the gas bubbles do not effectively increase the release of fission products. Some hydrogen combines with sodium vapor in the RC atmosphere above the pool to form sodium hydride. This cannot occur at temperatures above minimum RC atmospheric temperature expected during the TMBDB scenario.[6] We would therefore expect sodium hydride to form only on the initially cooler surfaces of RC and pipeway cell liners and RC vent system piping.

C. Annulus Cooling System Description

The Annulus Air Cooling System is designed to ensure that the structural integrity of the steel containment vessel and concrete confinement building is maintained based on realistic evaluation of TMBDB conditions. These conditions include an increase in the temperature of the steel containment vessel, confinement annulus air, and concrete confinement building. Before activation of the Annulus Air Cooling System during the postulated TMBDB event, the containment systems function in the same manner as for the design basis accidents occurring inside containment as described in the PSAR.

When cooling is required, outside air is introduced into the annulus area through an opening in the confinement structure. Vane axial fans located in the Reactor Service Building supply air to the annulus space. Redundant fans are provided in the system.

The confinement annulus is partitioned to provide a spiral air flow path around the containment vessel from the 816 ft elevation to above the containment spring line. The annulus partition system is designed such that an effective annular flow area of between 180 and 250 ft^2 is obtained to an elevation 926 ft-0 in. and a flow area between 450 to 600 ft^2 from elevation 926 ft-0 in. to the top of the confinement building. These flow areas ensure that velocity ranges of 2200 to 1500 FPM and 850 to 700 FPM are maintained for heat removal. Additionally, the partitions provide a platform system for periodic inspection of the containment vessel penetrations.

Leaktight, motorized dampers are provided on the entrance of the fan enclosure and on the outlet of the plenum at the top of the confinement structure. Missile hardened enclosures and intake debris screens protect the fans and the exhaust opening.

All power requirements of the Annulus Air Cooling System are supplied from Class 1E redundant power distribution systems. Backup capabilities are provided for all active components, such that failure of any one active component will not preclude 100% operation of the Annulus Cooling System.

1. Feature Requirements.

a. To ensure that containment and confinement do not fail from excessive temperatures, the annulus cooling system shall remove the expected heat load into the containment steel shell.

b. Steel containment temperatures shall be below those that cause structural failure or excessive containment leakage.

c. Concrete confinement temperatures shall be below those that cause structural failure.

d. The annulus cooling system operations shall be by remote manual actuation from the control room.

2. Discussion. The Annulus Air Cooling System is designed to augment the heat removal from the containment as part of the TMBDB. Air is drawn into the annular space between the containment steel shell and concrete confinement structure, is distributed with the aid of flow baffles, and exits at the top of the dome. The air flow rate of 400 000 cfm produces a minimum Reynolds number of 3.3×10^5 and the flow is, therefore, likely to be turbulent. For the annular configuration this results in a convection coefficient of 1.6 BTU/hr-ft^2-^{o}F. Although this value is little better than can be obtained on the exterior of the confinement structure with natural convection only, use of the annulus cooling system does remove a significant thermal barrier to the containment heat rejection, that is, the thermal resistance of the air-filled annulus and the concrete confinement structure. The effect is to substitute the external environment temperature as a boundary condition for the containment shell.

The maximum heat rejection of the annulus cooling system at the containment design temperature is approximately 5×10^7 BTU/hr. The average heat load generated from the CDA is 6×10^7 BTU/hr, which compares with the capacity of the annulus cooling system. Since most of the decay heat does not reach the containment in the initial stages because of the structure's thermal capacity, the containment design temperature and pressure conditions would not likely be violated. Based on the estimated heat generation from the CDA, the annulus air cooling system, activated at 24-36 hrs after accident initiation, is adequate to maintain the containment at a safe temperature level.

A two-dimensional thermal analysis of the containment-confinement structures was performed using the thermal analysis code TRUMP. This code is a finite difference thermal analyzer that can accommodate heat transfer by conduction, convection, and radiation. It also includes provisions for mass flow and chemical reactions. Although this code can be used successfully in modeling the thermal behavior of large structures, the adequacy of the nodal distribution and the thermal input parameters signficantly affect the results.

In evaluating the performance of the annulus cooling system, the thermal model assumes a single heat sink, the ambient atmosphere. There is some conservatism in this assumption since the foundation soil represents a second heat sink for the confinement-containment structure and was not included in the model. However, a range of air flow rates is specified within the annulus, which results in a variation in the convective heat transfer coefficient between 1.6 and 3.9 BTU/hr-ft^2-°F. Since the internal structural temperatures are dependent on the thermal coupling to the heat sink, any variation in the coupling will significantly affect the analysis results. The buildup of aerosols on interior surfaces of containment may alter the heat sink coupling. Analyses by the applicant have addressed this problem.

Another important parameter is the convection coefficient between the containment atmosphere and the containment structure. The presence of sodium vapor and non-condensible gas near the internal surface of the containment can produce a significant variation in the heat transfer coefficient. For condensing sodium vapor, a value of 200 BTU/hr-ft^2-°F is possible, while for free convection in air, a typical value is 1.2 BTU/hr-ft^2-°F. This thermal coupling is important in determining the temperature difference between the containment atmosphere and structure. In this case, large values of h result in higher containment wall temperatures and better coupling to the heat sink.

D. Description of Containment Vent and Purge Systems

The containment purge capability is provided by the containment cleanup system exhaust blowers, which draw a negative pressure in the containment building and then by opening the redundant containment purge penetrations. The two purge pipes penetrating the containment are 18 in. in diameter and are designed to the requirements of the ASME Boiler and Pressure Vessel Code, Sec. III, Division I, Class 2. Each purge line is provided with redundant normally closed isolation valves outside of the steel containment vessel.

Operation of the purge system requires the opening of the purge line isolation valves from a remote-manual station in the main control room. Flow direction sensing instrumentation is provided to automatically close the purge isolation valves if a backflow condition occurs.

The RCB vent capability is provided by the vent line connected to the Containment Cleanup System. This vent capability allows the blowdown of the RCB after some time period following a TMBDB accident to reduce the internal pressure and to subsequently reduce the hydrogen concentration through purging. The vent line is connected to the TMBDB Cleanup System through two redundant 24 in. inside diameter pipes that penetrate the RCB with isolation valves located outside the steel containment vessel. The vent line and pipes penetrating the RCB are designed to the requirements of the ASME Boiler and Pressure Vessel Code, Sec. III, Div. I, Class 2. The Vent System pipes penetrating the RCB have their valves in the normally closed position.

Depressurization will occur at the maximum rate of 24 000 cfm. To prevent inadvertent operation of the valves, no local operators will be provided and the valve actuation system will be equipped with appropriate physical restraints and warning plates. Similar safety features are provided on the purge system valves.

1. Features Requirements. Requirements for the Vent and Purge Systems proposed by the applicant are the following.

Containment Purge System.

a. To ensure that the RCB hydrogen concentration does not exceed 6% (by volume), the purge system shall be capable of injecting outside air into containment at a maximum rate of 12 000 scfm at pressures not exceeding atmospheric.

b. To ensure containment atmosphere mixing before venting, the purge air shall be injected into the containment below elevation 840 ft.

c. The purge system shall prevent backflow from the containment to the outside atmosphere.

d. The purge system, in combination with the containment vent and cleanup systems, shall maintain containment at a negative pressure after the containment pressure is reduced by the initial venting after 24 hrs.

e. The purge system operations shall be by remote manual actuation from the control room.

Containment Vent System.

a. To prevent containment failure by excessive pressure, the vent system shall have a capacity between 24 000 and 26 400 acfm with a containment pressure of 30 psia, a containment atmosphere density of 0.07 lb/ft^3 and a viscosity of 0.06 lb/ft-hr. It shall remain functional if up to 300 000 lbs of aerosol enter the system at a maximum rate of 5 600 lb/hr.

b. The vent system shall exhaust the containment atmosphere from the containment into the containment cleanup system.

c. The containment vent system shall be compatible with the following gases, vapors, and aerosols: Ar, N_2 H_2, H_2O, CO, CO_2, O_2, Na_2O, Na_2O_2, NaOH, Na_2CO_3, fission products, and compounds resulting from fission product reactions. The system must remain functional for inlet gas temperatures and pressures given on Figs. 2-5 and 2-6 of Ref. 1, and beyond 150 hours for temperatures up to 250°F.

d. The vent system operations shall be by remote manual actuation from the control room.

2. Discussion. The second criterion for the Purge System ensures that purge air will enter the containment below 840 ft to ensure adequate mixing of purge air with the containment atmosphere. In addition, the applicant has made a commitment to locate the purge and vent systems as far apart as possible in the containment. This will prevent direct passage of purge air to the vent system outlet which would not provide the opportunity for mixing purge air with the containment atmosphere.

The applicant has removed the requirement that the vent system should exhaust containment atmosphere from the top of the containment. Potential for hydrogen stratification is very low as discussed in Appendix H.3 of Ref. 1. During all time periods of the TMBDB scenario there are mixing forces acting in the containment. Early in the scenario these are from the gas stream exiting the RC vent systems. Later they are from the air flow between the purge and vent system. Additionally, hydrogen has a high molecular diffusion in this type of atmosphere, especially when introduced near the bottom of the volume. Tests performed at HEDL[13] reveal that a relatively uniform hydrogen distribution can be expected.

E. TMBDB Instrumentation System

The reactor containment instrumentation system provides measurements of reactor containment pressure, atmosphere temperature, steel shell temperature, and hydrogen levels. Each of these measurements will be redundant, designed to remain functional following a Safe Shutdown Earthquake, and qualified to ensure operability under the expected environmental conditions.

The RCB pressure is measured at two widely separated locations. The instrumentation penetrations are at 108° and 285° (0° is plant north). The design will be such that the pressure element and transmitter are located outside of the RCB and will sense pressure with an impulse or capillary line. This arrangement will allow sensing of containment pressure at temperatures up to 1100°F. Each transmitter will send a signal to the main control room. The channels will be completely independent and physically separated. Each channel will be powered from the Class 1E power system.

The reactor containment atmosphere temperature is measured near the top of the RCB. The measurement will be redundant so that any single failure will not preclude the operator from receiving temperature data. The channel will be designed to operate 500 hours at a maximum temperature of 1100°F. The signal conditioning for the temperature sensors will be located in the Steam Generator Building. Each transmitter will send a signal to the main control room. Each channel will be physically separated in accordance with IEEE 384-1974 and will be powered from the Class 1E power system.

The Reactor Containment Vessel Temperatures will be measured at selected locations on the inside of the steel shell.

The containment atmosphere hydrogen concentration measurement system consists of redundant, independent, and continuous hydrogen analyzers located in the Intermediate Bay of the Steam Generator Building. These are connected to the containment atmosphere through redundant and independent sampling lines. The inlet to the sampling lines is located at the top of the containment to prudently protect against hydrogen stratification. Each sampling station will include a hydrogen analyzer which will transmit a signal to the main control room. The channels will be physically separated and powered from the Class 1E power system.

Since the containment could be vented beyond 24 hours and therefore most of the radiological release would be through the vent and filter systems, radiation monitors are provided downstream of the filter system where the releases to the atmosphere would occur.

2. Requirements. Operator action to initiate TMBDB systems operation is required only for events beyond the design base. However, misoperation of TMBDB systems because of incorrect instrument readings in the control room could defeat ESFs required to mitigate design basis accidents. In accordance with this importance to maintain ESF capability, plant instrumentation that has been designated "TMBDB Instrumentation" shall be designed, manufactured and qualified to all standards applied to Class 1E instrumentation. Specifically the following subsystems of the Reactor Containment Instrumentation System (RCIS) and of the Radiation Monitoring System (RMS) shall be considered TMBDB instrumentation:

(1) Containment Pressure (RCIS)

(2) Containment Atmosphere Temperature (RCIS)

(3) Containment Hydrogen Concentration (RCIS)

(4) Containment Vessel Temperature (RCIS)

(5) TMBDB Exhaust-Plant Effluent Radiation Monitoring (RMS)

(6) High-Range Containment Area Radiation (RMS)

Note that subsystems (5), and (6) are not in the category of instrumentation which could be used to defeat ESFs; however, because of their importance in assessing releases from the plant during a TMBDB scenario they are included in the TMBDB instrumentation. Instrument ranges are specified in section 2.1.1.12 of Ref. 1.

3. Discussion. Operator initiation of TMBDB features requires accurate and reliable information on the state of the containment during the course of a TMBDB accident. It is unrealistic to specify ranges for TMBDB instrumentation based on analysis results which do not include margin for analysis error. Based on experience with similarly complex systems, a prudent margin for instrumentation ranges is 1.5 times the expected range of the measured variable.

In addition, the instrument accuracy when expressed as a percentage of full scale should be + 1 percent for the temperature instrumentation specified for TMBDB. This results in an absolute error of + 16°F which is not unreasonable for this application and is required for accurate assessment of incontainment thermal conditions. Because of the rapid transient response of containment atmosphere pressure and temperature under postulated CDA conditions, the instrumentation should be capable of response times of less than 30 s instead of the 5 min. specified.

Based on TMI experience, the duration of the monitoring of the containment should be approximately 5000 hrs, and the specification of all TMBDB instrumentation operational time should be consistent with this experience.

The location of the containment atmosphere temperature sensors should be such that the maximum temperature is indicated under stratified containment atmosphere conditions; therefore, measurement locations in the dome are appropriate. However, consideration must also be given to the possibility that single point instrumentation can be misleading when two-phase systems exist in the presence of a noncondensible gas. It is possible that atmosphere temperature sensors located in the containment dome would not indicate the presence of a potentially higher sodium vapor temperature at some lower elevation. This could result from the tendency of non-condensible gases to accumulate at the top of a refluxing, two-phase system.

Additional consideration has been given to HCDA containment atmosphere constituents based on the expected scenario and their effect on the TMBDB instrumentation. Aerosol loadings which affect both the accuracy and the operation of the instrumentation are important. Because of the applicant's proposal to use remote sampling of the containment atmosphere for certain measurements, the ability to meet the requirements for aerosol loading may be difficult. The applicant is conducting tests on prototype hydrogen filters to evaluate this problem.

Because of the importance of TMBDB instrumentation, adequate redundancy is an important consideration. Experience indicates that triple redundancy of each measurement is far superior to double redundancy. The major benefit is the increased probability of detecting a faulty reading among the data set. When only two data exist for a parameter, any discrepancy is irreconcilable, whereas, for a triple data set any single discrepancy can be identified. Triple measurements are preferred for each TMBDB instrumentation location. It is not necessary for all three readings to be displayed simultaneously to the operator. However, a means to selectively obtain each reading must be provided.

F. Reactor Cavity and Pipeway Cell Liners

The RC and pipeway cell liners are described in Section 3A.8.2 of the PSAR and in Section 3.2.2.5 of Ref. 1. Two features have been added to the RC cell liner to provide additional thermal margin. The space between the RC liner and concrete is divided into four zones by horizontal baffle plates which are welded to the liner. The RC liner is fabricated from carbon steel and its purpose is to protect the concrete from the sodium and to direct the steam generated behind the liner to the liner vents.

The baffle plates, shown on Fig. 3-34 of Ref. 1, are provided for zoning of the space behind the RC liner to prevent sodium, steam, or reaction products propagating from one zone to another and to positively separate the venting system into four zones (three along the vertical wall and one including the floor and corner).

The carbon steel solid baffle plates are welded to the liner and extend two feet radially into the wall. The two foot width is selected to ensure that the baffle plates extend into non-degraded concrete until well past the time that liner integrity is important. The baffles are attached to the back of the liner plates near the RC floor, 8 and 26 ft. above the floor respectively. Similarly, baffle plates are included behind the pipeway cell liners to separate the walls, floor and roof of each cell.

In addition, the anchors for the cell liner are lengthened so that they will remain anchored in non-degraded concrete until integrity is no longer important.

1. Feature Requirements. To ensure that the RCB hydrogen concentration does not exceed 6% (by volume) and to keep from exceeding the containment vent, purge and cleanup system capacities, the RC wall and pipeway cell liners shall prevent short term (less than 30 hrs) sodium-concrete reactions based on the pressure on Fig. 2-2 and the temperatures on Fig. 2-9 and Figs. 2-11 through 2-16 in Ref. 1. The results of structural analysis will be used to determine the liner failure times assumed in the TMBDB scenario.

To limit the consequences of liner failures, the liner system shall have physical barriers behind the liners between the RC floor and RC wall and at 8 ft and 26 ft above the RC floor. Likewise, the pipeway cells shall have physical barriers behind the liners to separate the vent spaces of the walls, floor, and roof of each cell. Only the spaces of adjacent walls with different liner failure times will be separated.

2. Discussion. When the sodium and core penetrate the reactor vessel and guard vessel 1000 s after the beginning of the TMBDB scenario the RC floor liner is assumed to fail immediately. The submerged portion of the RC wall liner gradually heats from approximately 1000°F soon after the sodium enters the cavity to 1800°F at sodium boildry. The nonsubmerged portion of the RC wall liner and pipeway cell liners heat up to somewhat lower temperatures depending on the location. In all cases the liners experience temperatures higher than those they are exposed to during a Design Basis Accident. For the extreme penetration and core-melt scenarios (Refs. 2 and 3), the cell liners, except for the base of the RC wall, do not experience higher thermal loads than in the base case scenario.

Failure criteria for the cell liners under TMBDB conditions are similar to those proposed for Level D design-base loads. Under biaxial tension cell liner equivalent (von Mises) strain is limited to 0.50 ε_u (ε_u is the strain corresponding to the ultimate tensile strength) for membrane and 0.67 ε_u for membrane plus bending. The criteria are relaxed considerably if there is compression in at least one direction. Then the equivalent strain is limited to 0.90 ε_u. For stud anchors axial strains are limited to 0.90 ε_u and axial plus bending strains are limited to 0.95 ε_u. As discussed in Appendix 3.8-A of the CRBRP Safety Evaluation Report (SER), the applicant has not presented evidence that the Level D criteria are acceptable. The proposed criteria don't follow the precedent of any existing codes and they have not been verified with appropriate experiments. The NRC staff's position on the criteria for Level D conditions is that they will be acceptable if a thorough analysis and test program proves them to be consistently conservative. We don't believe that the same level of conservatism is warranted for TMBDB, but the criteria have to ensure that the liner won't fail before the times specified in the TMBDB scenario. In summary, if the Level D criteria are found to be acceptable, the TMBDB criteria should be acceptable.

As stated in SER Appendix 3.8-A, we believe that an acceptable cell liner design for sustaining level D loads will be forthcoming. To verify the acceptability of the liner design for the higher temperatures experienced during TMBD the applicants will be required to perform tests similar to those required for verification of Level D criteria and liner design. Additional analyses are required to obtain more details of the three-dimensional strain field near stud-liner interfaces.

We have concluded that with successful completion of the required analysis and testing and consideration of fall back positions (Appendix 3.8-A), an acceptable design will be developed before fabrication of cell liners is started.

G. Reactor Cavity and Pipeway Cell Liners Vent System

The Reactor Cavity and Pipeway Cell Liner Vent System is designed to remove steam and gases from behind liners in all inerted cells to prevent failure of the liners from pressure buildup in the gap between the liners and concrete walls. The system is inactive and consists of embedded piping connecting the liner-concrete gap to Cell 105 for all liners except the RC floor where the system is vented directly above the operating floor.

The vent system is redundant and redundant lines are physically separated to minimize potential for line blockages from a common mode failure. Each liner vent system pipe is provided with a loop seal to prevent sodium from entering Cell 105 after liner failure. At the vent pipe inlets the cell liner is blocked out to prevent it from buckling inward and restricting flow into the vent line.

1. Feature Requirements. The applicant has specified the following requirements for the cell liner vent system.

a. To ensure that the pressure buildup, due to the gases released behind the liners, does not impair the ability of the liners to prevent sodium from reacting with concrete, all RC and pipeway cell liner vent systems shall prevent a pressure buildup behind the liners in excess of 5 psi.

b. To ensure that sodium would be prevented from reaching Cell 105 in the event of liner failure, the liner vent system for the RC floor shall vent the gases released from heated concrete to containment above the operating floor. The floor liner vent system shall have a capacity of 10 $lb/hr\text{-}ft^2$ of water vapor at a density of 0.02 lb/ft^3.

c. The liner vent system for the RC walls and pipeway cells shall vent the gases released from heated concrete to Cell 105. The liner vent system shall have a capacity of 7 $lb/hr\text{-}ft^2$ of water vapor at a density of 0.02 lb/ft^3.

d. To insure that the Cell 105 hydrogen does not exceed 6%, the sodium leakage from the RC through the liner vent system to Cell 105 shall be less than 1000 lbs. This leakage is to be based on RC pressures and temperatures and on the differential pressure between the RC and Cell 105 given in Figs. 2-2, 2-3, and 2-4 of Ref. 1.

2. Discussion. Potential failure modes that could prevent the liner vent system from meeting the specified requirements include:

a. plugging of vent system piping,

b. excessive deformation of vent system piping caused by weight of degraded concrete,

c. excessive deformation of vent system piping caused by thermal stresses, and

d. failure of the vent system piping loop seals to function as expected.

The possibility for vent system piping to become plugged with particulate from degraded concrete behind the liner has been considered. When the liner buckles from thermal loads and bears against the insulating concrete stresses in the concrete can be high enough to cause compressive failure. However, compression from the liner will tend to keep the failed concrete in place. Other particulate can come from structural concrete that is degraded as a result of high temperature experienced during TMBDB.

It is unlikely that much of this particulate can be carried through the gap behind the liner and into the vent piping because of the low gas flow rates in the gap. Flow rates become higher at the vent pipe inlet but, as mentioned previously, the liner is blocked out there so there should be no particulate formed at this location. The vent pipes are designed for maximum flow rates that occur early in the TMBDB scenario. During this time period little concrete degradation would have taken place. Later in the scenario, even if some of the redundant pipes are plugged, the remainder could handle enough gas flow to prevent excessive pressure buildup behind the liner.

Failure modes caused by excessive deformation are unlikely because of design features specifically included to prevent large loads on the vent piping system. These include providing a compressible material between the pipe and concrete to allow free thermal expansion and providing significant steel reinforcement where concrete may degrade.

The vent system piping loop seals prevent flow of sodium to Cell 105 after liner failure. One possible problem is the presence of water in the loop seal at the time the liner fails. When the sodium approaches the loop seal and reacts with the water, a high local pressure would result forcing the water out of the loop seal into Cell 105. The loop seal would then function as designed.

In evaluating the effect of failing one vent system pipe, we need to remember that the system is redundant. Also, the cell liner system is divided into several independent modules separated by baffles so that failure of one liner area does not directly affect other areas.

H. Guard Vessel Support

The guard vessel is supported by a skirt that is in turn supported on steel blocks between it and the RC floor. This provides 48 openings under the skirt that are approximately 5 in. by 6 in. and provide for dispersion of liquid sodium and fuel particulate from underneath the vessel into the entire RC.

1. Feature Requirements. To ensure that sodium and fuel particulate redistribute in the RC, a flow area of at least 10 ft^2 shall be provided under the vessel skirt bottom flange.

2. Discussion. The 48 openings of 30 $in.^2$ each give exactly 10 ft^2 of flow area under the skirt, which meets the only feature requirement proposed by the applicant. However, this should not be construed to mean that the sodium/fuel will necessarily redistribute on the RC floor as outlined in the TMBDB scenario in Sec. 3.2.1 of Ref. 1.

I. Reactor Cavity-to-Containment Barrier

This system consists of the reactor vessel closure head and associated seals between the head plugs, head-mounted components and associated seals, and the seal between the reactor vessel flange and the RC support ledge. Annuli between the head plugs have a margin seal added to meet leakage requirements associated with the Structural Margins Beyond Design Base (SMBDB) accident. These are in the head riser annuli above the sodium dip seals and inflatable seals described in PSAR Sec. 5.2.4.4. Head mounted components are also sealed to meet requirements of the SMBDB accident.

The seal between the reactor vessel flange and RC support ledge is a low alloy steel circular membrane with an L-shaped cross section. Between this membrane and the vessel head a Grafoil gasket provides the sealing and high-temperature packing provides sealing between the membrane and the support ledge. Gasket caps provide sealing over the reactor vessel hold-down bolts.

1. Feature Requirements. To ensure that the heat capacity of the pipeway cells is employed from 1000 s to 50 hrs after a CDA, the total leakage of sodium vapor through the RC to head-access area seals shall not exceed 10 000 pounds. This does not include leakage through the reactor head. During the first 1000 seconds following a CDA the vessel is not penetrated so sodium leakage to the reactor containment building occurs only through the head and head-mounted components. The requirements for this time interval are given in Ref. 10 and will not be addressed here. Leakage after 1000 s is based on the pressure differential between the RC and head access area given in Fig. 2-1 of Ref. 1 and the pressures and temperatures given in Figs. 2-2 and 2-3 in Ref. 1.

2. Discussion. Loss of functionality of the RC-to-head access area seal could result from either large deflections of the vessel or the RC support ledge near the seal or excessive temperature degrading the component materials of the seals.

Excessive deformation of the seal area before 50 hrs will probably occur during the first 100 to 150 ms following a CDA. Component Margin Requirements for the vessel and support ledge during the CDA are given in Ref. 10. Reference 1 gives allowable stresses for the RC ledge based on ACI 318-77. These stress criteria and results of an analysis of the ledge are presented in Sec. 3.2.2.5.1.2 of Ref. 1. Meeting these criteria will ensure that the ledge will not deflect excessively during the dynamic loading portion of the CDA. No displacement criteria for the vessel in the seal area are given in either Ref. 1 or Ref. 10. However, the strain criteria given in Ref. 11 will suffice.

Materials used for the RC-to-head access area seal should remain functional for the temperature predicted at the RC ledge up to 50 hrs following a CDA. In Ref. 1, Figs. 3-77 and 3-78, the applicant predicts peak temperatures in the region to be below 1400°F. The low-alloy steel used for the seal will remain intact well beyond this temperature with an eutectic temperature well above 2000°F. Reference 12 indicates that the Grafoil material will remain intact to 6600°F. High temperature packing is available that can survive temperatures much higher than 1500°F.

Although not directly related to the seal being evaluated in this section, it should be noted that the elastomer margin seals at the top of the head plug risers will begin to degrade from 500-600°F. In Appendix F.6 of Ref. 1, the applicant considers the effects of losing these seals because temperatures in this area are expected to exceed 600°F before 50 hrs has elapsed. Results of analyses that include failure of head seals show that containment will not be challenged before 24 hrs. This is principally because the risers act as efficient heat sinks.

J. Reactor Cavity Penetrations and Recirculating Gas Cooling System

The Reactor Cavity Gas Cooling System provides cooling of the atmosphere in the RC. The system is described in detail in PSAR Sec. 9.16. It has been provided with automatic gas isolation valves on the cavity cooling system inlet and outlet lines. The valves are located in Cell 105 just outside the RC wall and are actuated by the sodium leak detection system or by a high gas temperature signal. They are capable of withstanding the thermal and pressure conditions to which they will be exposed during the TMBDB scenario.

Other penetrations of the RC whose failure could permit sodium vapor flow into Cell 105 are designed to limit total leakage over the sodium boildry periods to the total specified in the feature requirements.

1. Feature Requirements. To ensure that the Cell 105 hydrogen concentration does not exceed 6%, the total leakage from the RC through RC penetrations and through the recirculating gas cooling system to non-inerted cells shall be less than 4 000 lbs of sodium from 1 000 s to 150 hrs after HCDA. These leakages shall be based on the RC pressures and temperatures on Figs. 2-2 and 2-3

of Ref. 1 and on the differential pressure between the RC and Cell 105 given on Fig. 2-4 of Ref. 1. In addition to the 4 000 lbs of leakage allocated to the Recirculating Gas Cooling System and other RC penetrations, 1 000 lbs of sodium leakage into Cell 105 is allocated to the liner vent system (Sec. 2.1.2.5 of Ref. 1).

2. Discussion. Ensuring that sodium leakage does not exceed the required values is feasible for the recirculating gas cooling system because outside the isolation valves the system is a closed circuit system so the sodium that might leak through the valves would still not be likely to enter Cell 105. The valves, being located on the Cell 105 side of the RC wall, will not be subjected directly to the severe environment encountered in the RC.

K. Containment/Confinement System

The Containment System is described in PSAR Secs. 6.2.1 and 3.8.2. The Containment Isolation System is described in PSAR Secs. 6.2.4 and 7.3.1. The design internal pressure for the containment is 10 psig, and the associated maximum allowable leakage rate is 0.1 vol%/24 hrs. A negative pressure is maintained in the Containment/Confinement annulus space and Containment/ Confinement penetrations are designed to maintain a bypass leakage value of less than 0.001 wt%/24 hrs.

Two vent and two purge lines penetrate the RCB. Each line is provided with two isolation valves external to the containment building. These are discussed further in the vent/purge section of this report.

1. Feature Requirements. The requirements proposed by the applicant are that at any given time, containment out-leakage shall not exceed the greater of:

a. the design leak rate (0.1 vol%/24 hrs);

b. the design leak rate adjusted for pressures above the containment design pressure of 10 psig. Leak rate = Design Leak rate x [Actual Pressure (psig)]$^{0.5}$/3.2.

2. Discussion. Containment isolation provisions for the vent and purge lines clearly do not meet CRBR Design Criterion 47 that normally requires one valve on each line to be inside of containment. As described in Ref. 1, placement of such valves inside containment would be unacceptable because of the severe TMBDB environment they would be exposed to before valve operation is required. Operation of valves on other lines penetrating the containment would occur before the severe TMBDB environment is established.

In Secs. 2.2.7 and 2.2.8 of Ref. 1., the applicant specifies that the purge and vent lines will be designed to requirements of the ASME Boiler and Pressure Vessel Code, Sec. III, Division I, Class 2. Acceptability of these requirements is addressed in the TER for Chapter 6 of the PSAR and will not be reevaluated here.

Excessive thermal expansion of the steel RCB during the TMBDB scenario could load the vent and purge systems considerably. These loads are to be considered under the beyond design basis load categories listed in Chapter 3 of the PSAR.

Structural Evaluation of Containment/Confinement System

In this section we review both the steel containment and the confinement building for the periods before sodium boildry and from sodium boildry to 8000 hrs.

Containment Vessel Prior to Sodium Boildry

The applicants initially determined that the ultimate capability of the containment is governed by the membrane stress intensity in the upper cylindrical portion of the containment shell. As a result of the review process, more detailed analyses were performed using criteria from recent codes and Code Cases for the ASME Boiler and Pressure Vessel Code. Results were that the governing component is the equipment hatch cover, which buckles at a pressure, as predicted by ASME Code Case N-284, lower than critical pressures for other failure mechanisms. The pressure at which this component is predicted to buckle is well above pressures that occur within containment during the base case TMBDB scenario.

Other results of the applicant's analysis are the following.

1. Up to 24 hrs the maximum containment temperature is 140°F.
2. Maximum pressure is 22 psig which will not challenge containment integrity.
3. Buckling is not a problem before 24 hrs.
4. Peak containment temperature before sodium boildry is 640°F.
5. Buckling at the operating floor level will not occur because the peak temperature there is 220°F compared to a buckling temperature of 240°F.

Based on the buckling criteria given in Chapter 3 of the PSAR, buckling will not occur at the operating floor for the predicted thermal loads. We performed independent analyses of the buckling potential and concur with the applicant. The only place where buckling is of any concern during TMBDB, other than the equipment hatch cover, is in the steel containment at the operating floor level. Elastic bifurcation buckling would occur only for temperatures much higher than those experienced in this portion of the shell. Buckling temperature is governed by a yield criterion that predicts buckling at shell temperatures of approximately 240°F. Maximum temperature that occur here are 220°F.

Additional studies were performed by both the applicant and Los Alamos to determine whether asymmetric heating of the containment might cause buckling or other failure at loads lower than those previously predicted. Results of these studies predict a maximum difference in temperature in a circumferential direction around the containment of 65°F. These temperature differences have little structural consequence.

Confinement Building Prior to Sodium Boildry

Computer code ANSYS was used to analyze the confinement building. Refer to section III.L of this TER where the RC is evaluated for our evaluation of analytical techniques and failure criteria for concrete structures. The building was modelled with an axisymmetric finite element model.

Results show the critical region to be where the confinement wall intersects the roof slabs from other buildings in the nuclear plant. An acceptable design can be achieved by adding reinforcement in this region beyond what is required for the loads combinations specified in the PSAR Sec. 3.8.

Containment Vessel After Sodium Boildry

Loads after boildry are enveloped by those prior to boildry so an acceptable design for preboildry will survive indefinitely after boildry.

Confinement Building After Sodium Boildry

Loads after boildry are enveloped by those prior to boildry so an acceptable design for preboildry will survive indefinitely after boildry.

L. Reactor Containment Structures

Several changes to the design that is based on Design Basis Accidents (DBAs) have been made so that the containment building and key internal structures can withstand the more severe TMBDB loads. These changes include additional concrete reinforcement in areas of the building where higher forces and moments from the TMBDB thermal and pressure loads occur. Also, the cell liner system was changed by increasing stud anchor size, decreasing spacing of anchors, and changing the size of supporting beams in the pipeway cell floor.

A major design change in the pipeway cell floor is also required to meet TMBDB requirements. This change, described in Sec. 3.2.2.5.1.3.2 of Ref. 1, involves changing the pipeway floor to a two layer system. The structural slab is 35 in. thick with 24 in. of sacrificial concrete above it. Further design modifications to the pipeway cell floor would be required for the extreme penetration analysis.[3] However, we agree with the applicant that sodium-concrete penetration rates determined in all tests to date do not support using the extreme penetration rates for design decisions. For both the extreme penetration analysis[3] and core melt analysis[2] the containment internal structures, except for the RC floor, pipeway cell floors, and the foundation mat, experience the same or less severe thermally induced loads than for the base case TMBDB scenario. During the core melt scenario the steel containment shell experiences a higher heat load than for the base case. This can be accommodated by turning the annulus cooling system on earlier.

1. Feature Requirements. The applicant has specified the following requirements for the containment structures.

- a. The RC and pipeway structures shall not collapse prior to sodium boildry. Structural conditions at boildry for the various scenarios are enveloped by the temperatures given in Figs. 2-9 through 2.18 of Ref. 1.
- b. The reactor containment building and confinement structure shall retain their integrity above the basemat indefinitely, based on the limiting temperatures in Figs. 2-19 through 2.31 of Ref. 1.

Additional assumptions and criteria used in evaluating structural concrete components are given in Sec. 3.2.2.5.1.2 of Ref. 1. Assumptions include the following.

- b. The stress-strain relationships for concrete and steel are temperature dependent and defined by the curves in Figs. C.3.9 and C.3-15 in Ref. 1. Other material properties are described in Appendix C.3 of Ref. 1.
- c. In the flexural analysis of concrete sections, interaction with the liner with can be neglected.

Criteria for concrete failure are:

- a. concrete at temperatures of 1200°F or more is totally degraded and incapable of carrying stress; and
- b. concrete with a stress-inducing strain exceeding the limits defined in Appendix C.3 of Ref. 1 is crushed and therefore does not develop any stress.

Additional criteria used for characterizing concrete during the TMBDB analysis are given in Appendix C, Section C.3.2 of Ref. 1. These include conservative criteria for evaluating failure in tension and shear. Acceptable criteria for failure of steel reinforcement are given in Section C.3.3.1 of the same appendix.

2. Discussion. The first requirement that the RC and pipeway structures shall not collapse prior to sodium boildry is ambiguous in that collapse is difficult to define. Even though the requirement is ambiguous, based on analyses presented in Sec. 3 of Ref. 1, the applicant has properly applied the intent of the requirement.

The assumptions used in analyzing the concrete structures (given in Sec. 3.2.2.5.1.2) are conservative and therefore, generally acceptable. One question arises in the application of temperature-dependent properties for concrete. In Appendix C.3 of Ref. 1, pC.3-2, the applicant observes that "the longer the duration of heating before testing, the larger the loss in strength. This loss in strength, however, stabilizes after a period of long isothermal exposure." Concrete properties were derived from concrete cylinders that were heated to their test temperature for 14 days. From this we can see that the properties used are conservative up to 14 days into the TMBDB scenario.

Beyond 14 days, or 336 hr, the applicant's TMBDB scenario assumes that no concrete structures other than the concrete walls sandwiching containment below the operating floor and the foundation mat have to remain intact. These concrete structures experience very small primary loads and relatively small thermal loads so the availability of data beyond 14 days for concrete exposed to high temperatures is not necessary.

The structures addressed in this section are analyzed during two separate time periods. The first is prior to sodium boildry, which occurs at 130 hrs into the TMBDB scenario. The second extends from sodium boildry to 8000 hrs. During both time periods the ANSYS computer code was used to determine response of structures to the predicted thermal gradients. The version of ANSYS that was used for these analyses did not have the capability to account for different tensile and compressive material properties. Therefore, the applicant used an interactive, iterative process to simulate concrete cracking. At each step of the iterative process tensile stresses were checked visually and any that had cracked had properties changed to simulate cracks. This procedure, even though time-consuming, should provide conservative results.

Reactor Cavity Before Sodium Boildry

The RC was analyzed using five different finite element models. These were used to evaluate the cavity midsection, base, top, and two different areas where radial restraints are significant. The models were all loaded simultaneously with thermal gradients, pressures, dead load, and hydrostatic loads, where appropriate.

The following are some results of the analysis presented by the applicant.

1. The RC ledge (vessel support ledge) will withstand the combined thermal and vessel weight loads for at least 24 hrs following TMBDB initiation.

2. The RC wall section at the base can withstand the specified loads of at least 50 hrs and collapse is not expected to occur before boildry.

3. The upper portion of the RC near the Head Access Area can withstand the loads for 80 hrs. Collapse is not expected before sodium boildry.

4. The areas of the wall where significant radial restraint exists will not fail before 70 hrs in the submerged region and before 80 hrs in the non-submerged region. Again, collapse is not expected before sodium boildry.

The assumption that collapse will not occur before sodium boildry for all regions of the RC is based on the fact that thermal stresses are secondary and self-relieve as the concrete cracks. This means that, for the assumption to be valid, stresses from other loads such as pressure and dead weight must be insignificant compared to thermal stresses. This is generally true for the RC wall because design parameters other than strength (for instance, nuclear shielding) govern the thickness of the wall.

Pipeway Cells Before Sodium Boildry

The applicant has provided structural integrity evaluations for the pipeway cell floor, two single-heated walls, and the double-heated wall between the pipeway cell and the RC. The ANSYS computer code was used with the methods described earlier in this section. Applied loads included dead load, pressure, thermal gradients, and, for the pipeway floor, relative motion between the RC wall and Cell 105 wall opposite the RC wall. The pipeway floor is critical because if it fails sodium can leak into Cell 105. It is also a sophisticated design that is difficult to analyze both thermally and structurally.

Results of the applicant's analyses include the following:

1. For the single-heated walls, capacity will not be exceeded for a minimum of 35 hrs. Beyond 60-70 hrs the walls will experience severe structural damage with leakage to be expected. Collapse is not expected before sodium boildry.

2. The double-heated wall will not fail until at least 70 hrs following TMBDB initiation. It is also not expected to collapse before sodium boildry.

3. The pipeway floor was redesigned significantly to meet TMBDB criteria. The previous floor design was changed to a two-layer design with 24 in. of sacrificial concrete above a 35 in. structural slab. The capability of the floor will not be exceeded before sodium boildry.

The same comments made for the RC concerning collapse before sodium boildry apply to the pipeway cell. Prediction that collapse will not occur before sodium boildry is based largely on intuition and engineering judgement.

We concur with the applicant's conclusions. It must be remembered that "failure" here refers to exceeding the conservative criteria specified for concrete (for example, shear failure is defined by conservative design formulas presented in the ACI code). The structures are very redundant and the major loads are secondary thermal loads. Therefore, the structures can be loaded well beyond "failure" before they collapse.

Foundation Mat Before Sodium Boildry

Temperatures do not rise significantly enough in the foundation mat before sodium boildry to present any challenge to its overall structural integrity.

Foundation Mat After Sodium Boildry

Results of a preliminary study presented by the applicant indicate that the outer part of the foundation mat that supports the RCB exterior wall and the confinement wall will remain essentially undamaged indefinitely. Where the thickness increases at a radius of 90 ft, additional reinforcement beyond that required for DBA loads has been added to ensure a conservative design.

Creep effects are not addressed in Ref. 1. At the temperatures predicted to occur in the foundation mat (up to 250°F before 8000 hr) long-term creep effects could potentially be significant. The same potential problem could occur in the RCB outer wall below the operating floor where temperatures get as high as 370°F during the 8000 hr evaluation period. However, because these concrete structures carry very little stress from primary loads, creep should not be a problem.

Reactor Containment Building Outer Wall After Sodium Boildry

The applicant has performed a preliminary assessment of the RCB outer wall below the operating floor during the period following sodium boildry. Results of this assessment show that structural integrity of the outer wall will be maintained. We agree with this preliminary assessment, which the applicants will confirm with more detailed analyses for the FSAR.

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ATTACHMENT 3

REVIEW OF THE INTERACTION OF SODIUM WITH CONCRETE AND OTHER MATERIALS

Task 1 Report

May 1982

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1. Introduction

The formal licensing review of the Clinch River Breeder Reactor (CRBR) was initiated with the submittal of the Preliminary Safety Analysis Report in 1975. This reactor differed from commercial light water reactors in several important areas, one of which is the use of sodium coolant. In the initial application permit, emphasis was placed on reliability analysis in the development of the safety-related design considerations for the plant.

One of these safety analyses considered the accident scenario for a hypothetical core disruptive accident (HCDA). One scenario postulates that the reactor vessel is breached along with the guard vessel and cavity cell liner. This postulated series of events places bare concrete in contact with hot sodium metal. An understanding of sodium-concrete interactions was necessary to assess such safety related concerns as hydrogen generation, with its attendant explosion risk, gas generation which could over-pressurize the containment building, degradation of the concrete which could lead to penetration or structural collapse, and, finally, the energy release from sodium reactions which under some circumstances would rival the decay heat.

Several organizations initiated studies of sodium-concrete interactions. The two principal investigators were Sandia National Laboratories and the Hanford Engineering Development Laboratory (HEDL). Altogether, more than 100 significant major experiments have been conducted to study sodium-concrete interactions. These tests examined concretes similar to those used at the Fast Flux Test Facility (FFTF) and proposed for use at Clinch River. The experimental results varied considerably, even under what appeared to be similar initial conditions. This variation is not completely understood and is the principal reason for concern about predicting long-term effects of a large sodium spill.

2. Review of Sodium-Concrete Experiments

Several years ago, the authors had the opportunity to review the sodium-concrete experimental data available at that time (Reference 1). Since then, a substantial number of additional experiments have been conducted but only a few of these have examined limestone concrete. In three years Sandia has conducted only four additional large scale limestone concrete tests and HEDL has conducted only six tests.

Results from the limestone concrete tests at HEDL are summarized in Table I and results from Sandia's tests are shown in Table II. Blanks in some columns for certain tests indicate that no information has been made available. Some Sandia tests show two numbers due to changes in the numbering system for these tests. The specimen thickness, area and orientation are provided. Sodium temperatures, masses and depths are also shown. Information is given on the test duration, whether or not an exothermic reaction was observed, hydrogen evolution, the extent of penetration and whether or not noises were observed suggesting a vigorous interation.

Other, smaller scale tests, conducted elsewhere are not listed although information concerning these tests can be found in References 2-4. Only limestone concrete data is listed in these tables. While many experiments have been conducted with other concretes, such as basalt concrete and magnitite concrete, it is felt that the various types of concrete are significantly different materials which must be treated separately. It is

Table I. HEDL Sodium-Limestone Concrete Interaction Tests

Test Number	Concrete Type/ Thickness, cm	Surface Area, m^2 (Orientation)	Temperature Ave, °C (Max)	Sodium Mass, kg/ (Depth, cm)	Exothermic Reaction	Length of Test (hr)	Hydrogen Evolven kg	Penetration Max. cm	Noises (Pops, Bumps, etc)
SC-4	Limestone 30	0.092 (Horizontal)	677 (802)	22.7 (25.4)	Yes	8	0.18	8.4 (6.1)	Yes
SC-5	Limestone 30	0.092 (Horizontal)	871 (871)	22.7 (25.4)	No	2	0.10	4.3 (2.5)	No
SC-6	Limestone 30	0.092 (Horizontal)	871 (871)	24.4 (27.3)	No	8	0.26	4.8 (3.8)	No
SC-8	Limestone 30	0.092 (Horizontal)	871 (871)	23.6 (26.4)	Yes	24	0.28	5.3 (4.4)	No
SC-10	Limestone 30	0.092 (Vertical)	871 (1093)	18.1 (20.3)	Yes	8	0.25	13.0 (8.9)	Yes
SC-12	Limestone 30	0.092 (Vertical)	871 (954)	19.1 (21.4)	Yes	24	0.23	14.0 (9.2)	Yes
SC-13	Limestone 30 +3.15 Kg NaOH	0.092 (Horizontal)	871 (932)	22.7 (25.4)	Yes	26	Yes	7.6 (4.4)	Yes
SC-14	Limestone 30 +3.15 Kg NaOH	0.092 (Horizontal)	871	22.7 (25.4)	-	24	Yes	7.6 (5.7)	?
SC-18	Limestone 30	0.092 (Horizontal)	871	3.15	-	8	No	5.0 (2.5)	?
SC-19	Limestone 30	0.092 (Horizontal)	677	22.7 (25.4)	-	8	Yes	6.4 (3.8)	?

Table I. HEDL Sodium-Limestone Concrete Interaction Tests (Continued)

Test Number	Concrete Type/ Thickness, cm	Surface Area, m^2 (Orientation)	Temperature Ave, °C (Max)	Sodium Mass, kg/ (Depth, cm)	Exothermic Reaction	Length of Test (hr)	Hydrogen Evolven kg	Penetration Max. cm	Noises (Pops, Bumps, etc)
LFT-4	Limestone, 30 cm MgO, 10 cm Steel, 1 cm (5 cm hole)	0.092 (Horizontal)	871	22.7 25.4	?	8	Yes	2.5/1.8	
LFT-6	Limestone, 61 cm MgO, 10 cm Steel, 1 cm (15 cm hole)	0.836 (Horizontal)	820	454/ (70)	Yes	15	Yes	7.5/5	
LSC-2	Limestone, 61 cm	0.836 (Horizontal)	475 (801)	454 (70)	Yes	100	Yes	7.5/5	
SET-12	Limestone, 61 cm 10 cm Pearlite	0.092 (Horizontal)	593 (871)	46/?	?	48	Yes	1.3 cm + 10 cm Pearlite	
LCT-1	Limestone, 61 cm (Pressure=17 ft head)	0.092 (Horizontal)	893 (871)	46 (51)	Yes	100	Yes	19 cm	
LCT-2	Limestone, 61 cm 80% dehydrated concrete (sodium limited)	0.092 (Horizontal	593 (871)	46 (51)	Yes	70	No	32.5 cm	

Table II. Sandia Sodium-Limestone Concrete Interaction Tests

Test Number	Concrete Type/ Thickness, cm	Surface Area, m^2 (Orientation)	Temperature Avg, °C (Max)	Sodium Mass, kg/ (Depth, cm)	Exothermic Reaction	Length of Test (hr)	Hydrogen Evolven kg	Penetration Max. cm	Noises (Pops, Bumps, etc)
P1, LS1	CRBRP Limestone 30.5	0.29 (Horizontal)	550 (800)	21 (8.6)	Yes	22 min	Yes	8.3	Explosion
P2, LS2	CRBRP Limestone 38.1	1.17 (Horizontal)	550 (800)	108 (11.2)	Yes	45 min	Yes	9.1 (7.6)	Yes
P3, LS3	CRBRP Limestone 38.1	1.47 (Horizontal)	550 (740)	186 (15)	Yes	3 hr	Yes	15.2	Explosion Lrg spalled chunk
P4, LS4	CRBRP Limestone 38.1	0.65 (Horizontal)	540 (450)	188 (30)	No	4 hr (8 min)	Yes	0.5	Yes (once)
LS5	CRBRP Limestone	1.17 (Horizontal)	540 (460)a	186	No	2 hr	?	Slight (0.5)	No
LS6	LS5 reused + 36 Kg NaOH		0 (700)	186	No	- 8 hr	?	Slight (< 1)	No
LS8	CRBRP Limestone	1.17 (Horizontal)	550 (450)	127 (15)	Moderate	52 min	?	1.0	-
LS9	CRBRP Limestone	0.65 (Horizontal)	600	182 (35)	Yes	3.4 hr (5 min)	Yes-H$_2$ exp T + 2 min	4.5	-
LS10	CRBRP Limestone	0.65 (Horizontal)	600	182 (38)	Yes	- 25 sec	Yes-H$_2$ exp T + 30 sec	-	Explosion terminated exp. after 25 sec
LS18	Limestone 61 cm Flawed Liner	0.65 (Horizontal)	665/575	182 (38)	No	4.2	?	1	
LS19	Limestone	0.65	695/575	68+68/29.1cm	Yes	?	?	6.4	

Table II. Sandia Sodium-Limestone Concrete Interaction Tests (Continued)

Test Number	Concrete Type/ Thickness, cm	Surface Area, m^2 (Orientation)	Temperature Avg, °C (Max)	Sodium Mass, kg/ (Depth, cm)	Exothermic Reaction	Length of Test (hr)	Hydrogen Evolven kg	Penetration Max. cm	Noises (Pops, Bumps, etc)
SET1	Limestone	0.072	600/?	4.5/0.3	No	?	?	0	
SET2	Limestone	0.072	650/?	4.5/0.3	No	?	?	0.5	
SET3	Limestone	0.072	700/?	4.5/0.3	No	?	?	0.5	
SET4	Limestone	0.072	750/?	4.5/0.3	No	?	?	0.5	
SET5	Limestone	0.072	73/?	4.5/0.3	No	?	?	?	
SET6	Limestone + perforated liner	0.072	73/?	4.5/0.3	No	?	?	?	

Notes: a=pool set point

particularly dangerous to attempt to infer the properties of one concrete from another concrete.

Some of the earliest experiments tested very small samples of concrete, typically 1 cm or 1 inch cubes. These samples were immersed in liquid sodium. Limestone, basalt and magnitite aggregate concretes were tested. It was discovered that all three types of concrete would react with sodium and totally disintegrate after several hours. In each case a certain minimum temperature had to be exceeded before a reaction would occur. It was determined that both the cement and aggregate reacted with sodium at temperatures above the 500 to 600°C. range. These reactions were exothermic with some differences noted which depended upon the kind of aggregate used and the water content of the concrete (Reference 2-4).

A larger group of tests was conducted at HEDL using 1 ft. diameter concrete surfaces which were exposed to heated sodium (Reference 5). Usually, 22.7 kg. (50 lb.) of liquid sodium were poured on these intermediate scale specimens covering them to a depth of 30.5 cm (1 ft.). Typically, the HEDL experiments were performed at higher temperatures than those at Sandia; in a majority of the HEDL experiments the tests were conducted at 871°C.

HEDL has recently conducted two large scale tests, LSC-2 and LFT-6. In these tests, 454 kg. of sodium were poured limestone concrete specimens with a surface area of 0.84m (one square yard), forming a pool with a depth in excess of two feet. Test LFT-6 included a steel liner and a 10.2 cm. thick layer of MgO aggregate. All tests were conducted with the surface under attack oriented horizontally, except for vertical tests SC-10 and SC-12.

During the same period when the tests described above were being performed, Sandia conducted a number of experiments on a significantly larger scale. Typical experiments have involved pouring approximately 180 kg of liquid sodium on specimens with areas of about 0.5-1.5 sq.m. The sodium pool depths have generally been shallower than in the HEDL tests and have ranged from 8 to 38 cm. The pouring temperature of the sodium in the experiments has varied from 450°C to 760°C. Some experiments have included steel liners.

Historically, there appeared to be a substantial difference in the results obtained by Sandia and HEDL. In the first three large-scale Sandia tests, highly energetic reactions were observed after an initial relatively quiescent phase. The energetic reaction quickly consumed all the sodium and penetrated as far as 15 cm into the limestone concrete within 3 hrs. In the first four HEDL small-scale tests (1 ft^2) on limestone concrete, penetration of less than half this depth occurred over 24 hr with an excess of sodium. All tests were made on horizontal surfaces.

The differences in results were initially attributed to scale effects, since the surface areas in the Sandia tests ranged from 3 to 15 times greater than those of HEDL. Significantly, the sodium pool depths in the initial Sandia sodium pours were approximately half those used by HEDL. Also, the initial HEDL sodium temperatures were 300°C higher than those used by Sandia;

it was thought at the time that the higher temperatures would produce more energetic reactions.

Following these initial tests, the experimental results changed. The fourth Sandia limestone concrete test employed a 0.5 m^2 surface area and twice the previous sodium depth. A penetration depth of only 0.5 cm was observed. This was followed by six more tests that produced little penetration but some experimental difficulties, such as hydrogen deflagrations or explosions. HEDL conducted two limestone concrete tests using vertical test surfaces and observed penetrations that came close to 15 cm. HEDL observed that the penetration into a vertical surface exceeded that into a horizontal surface by a factor of two. On the other hand, Sandia observed greater penetration downward than radially.

At this point, HEDL, on the basis of its tests only, proposed that concrete reaction products were responsible for the limited interactions observed in the HEDL tests. Past examinations of the HEDL specimens showed that the upper surface of the concrete was covered by a hard, strong layer of reaction products. It was HEDL's position that, during the experiment, this layer was a viscous liquid which separated the unreacted concrete from the sodium. Under generally similar conditions, experiments with vertically oriented concrete surfaces showed greater penetration into the surface, suggesting that the viscous protecting liquid slumped under the influence of gravity thereby exposing an unprotected concrete surface to the sodium. On this basis, HEDL proposed that horizontal concrete surface reactions would not progress beyond 3 or 4 inches. The problem with this interpretation was that it did not explain the results of Sandia tests P1, P2 and P3.

Next, Sandia proposed a mechanism to explain the different results obtained at Sandia and HEDL. It was suggested that NaOH rather than sodium was primarily responsible for the observed attack on concrete. The NaOH formed when sodium reacted with water driven from the concrete by heat. It was suggested that the concrete would experience little attack during the period before the sodium became saturated with NaOH. When additional NaOH was formed, the liquid NaOH formed a separate layer, which in turn attacked the concrete. It was argued that the shallow pools used in the first three Sandia limestone concrete tests quickly saturated with NaOH allowing the energetic reaction to begin promptly. The deeper sodium pools used by HEDL, and in Sandia test No. 4, required time to reach saturation with NaOH. In addition, it was suggested that reaction products would provide some shielding from further attack for the concrete. This proposed mechanism led both Sandia and HEDL to perform tests in which NaOH was added to the liquid sodium before it came into contact with limestone concrete in order to achieve immediate saturation. Sandia's test with NaOH (test No. 6) showed a lesser reaction than before, and HEDL's tests (tests SC-13, SC-14 and SC-19) showed about the same penetration. These results certainly did nothing to confirm the NaOH attack hypothesis.

Later Sandia tests (LS8, LS18, LS19) did not reproduce the high rates of erosion experienced in the early tests. As compared to the early tests, the later ones employed deeper sodium pools, specimens with smaller areas and somewhat higher temperatures. Our interpretation of these tests has been hampered by a lack of published data about them.

More recent tests at HEDL have shown a limited penetration in rather long tests. In HEDL test LSC-2, a penetration of only 7.5 cm was observed in a 100 hour test when sodium was poured on concrete at 475°C and later maintained at several different temperatures (Reference 6). However, when a similar test (LCT-1) on a smaller scale was conducted under pressure simulating a 17 foot head of sodium, the erosion increased to 19 cm. Most of the erosion occurred during the first five hours.

HEDL's last test, LCT-2, with 80% dehydrated concrete, showed an erosion of 32 cm in 70 hours. This erosion was limited by consumption of all of the sodium and presumably would have continued further if additional sodium had been present.

3. Models For Sodium Concrete-Interactions

Two models have been developed by HEDL and Sandia to explain the results observed in the liquid sodium-limestone concrete tests conducted until now. Unfortunately, the models make totally different assumptions about the role of NaOH in the reaction process. Neither model at this time can successfully explain all of the observed results.

A. HEDL's Conceptual Model of Sodium Concrete Reactions

A model for the interaction between liquid sodium and concrete has been developed at HEDL for limestone concrete [Reference 4]. The model postulates the existence of a threshold temperature of approximately 500°C which must be attained before sodium-concrete reactions occur. The existence of a threshold is well supported by experimental evidence.

The concrete erosion in this model is controlled by a reaction product layer, composed mainly of NaOH, formed by the reaction between liquid sodium and water from the concrete. Water release from the concrete begins at temperatures of about 100°C, with the following reactions occurring as a result [Reference 7].

$$2Na + H_2O \rightarrow NaOH(l) + NaH \quad (1)$$

$$4Na + H_2O \rightarrow Na_2O + 2NaH \quad (2)$$

$$2NaH \rightarrow 2Na + H_2 \quad (3)$$

A chemical analysis of the reaction products has shown that they are 37% NaOH by weight, which suggests that reaction path (1) is favored. NaOH from reaction (1) then attacks concrete:

$$2NaOH + CaCO_3 \rightarrow Na_2CO_3 + Ca(OH)_2 \quad (4)$$

Calcium hydroxide, in turn, decomposes at 580°C, thereby replenishing the water supply for further reactions with sodium:

$$Ca(OH)_2 \rightarrow CaO + H_2O \quad (5)$$

Sodium aggregate reactions will not occur until the temperature level reaches a threshold value of 500°C. The following reactions are principally responsible for the degradation of concrete integrity:

$$4Na + 3Ca\ CO_3 \rightarrow 2Na_2\ CO_3 + 3CaO + C \quad (6)$$

$$4Na + 3SiO_2 \rightarrow 2Na_2SiO_3 + Si \quad (7)$$

A provision in the model is also made for the following reaction:

$$4Na + CaCO_3 \rightarrow 2Na_2O + CaO + C \quad (8)$$

In HEDL's view, the NaOH produced by the aqueous reaction does immediately attack concrete but the rate of attack is very slow and is dependent on the dilution of NaOH with other reaction products. Sodium-concrete reactions are much more energetic but do not occur until the 500°C threshold temperature is attained; it is these reactions which are responsible for the degradation of concrete integrity. Further, at temperatures in excess of 700°C calcium carbonate in the limestone will decompose to yield CO_2. The reaction of CO_2 with sodium is highly exothermic.

Since HEDL views sodium concrete reactions as the major reactions responsible for concrete erosion, transport of sodium to fresh concrete is essential for concrete attack. However, this transport is inhibited by the presence of a layer of reaction products. Initially, the reaction product layer is small but it eventually grows to considerable thickness as the attack by sodium on concrete proceeds. The layer also may contain substantial quantities of Na_2CO_3, a relatively dense material from reaction (6). It has been suggested that this layer may be quite viscous.

The protective layer could be destabilized by a number of factors, resulting in enhanced attack on concrete. These factors include insufficient NaOH, because of low concrete water content, escape of water, or a concrete layer which is too thin and thus does not contain much water. Cracking could disturb the layer by draining it away or by fracturing it. The layer could also be disrupted by the evolution of large quantities of steam or carbon dioxide. The protective layer could also be disrupted by the formation of solid reaction products. Spalling also can disrupt a protective layer. Both spalling and cracking are probably scale dependent. HEDL has tried to overcome this problem in its experiments by rigidly constraining its test specimens in order to simulate a larger size experiment. Other factors which may serve to increase the tendency of concrete to crack or spall include the presence of rebars and the geometrical shape of the test article.

The extensive concrete decomposition in HEDL test LCT-2, which employed pre-dehydrated concrete, is cited in support of the above hypothesis. If indeed the sodium hydroxide layer serves to inhibit attack on concrete by sodium, then attack should be most extensive in concretes with little water available for sodium hydroxide formation. This has in fact been observed in test LCT-2 where the penetration, 32 cm, is the largest observed to date.

B. Comments on the HEDL Model

The HEDL model is essentially a retread of a model first proposed 3 or 4 years ago. At that time, concern was expressed about the inability of the model to adequately explain the results of Sandia tests P1, P2 and P3 in which greater than average concrete erosion was observed. In the case of Sandia test P3, this erosion reached 15 cm in a three hour test. However, in test P3 there is a possible explanation. It can be argued that the substantial erosion observed in that test was due to spallation or cracking which resulted in the separation of a large chunk of concrete, 25 cm in diameter and 7 cm thick, which was found lying above the cavity floor. The spallation could have disrupted the protective layer, thereby increasing erosion.

Another problem area is HEDL tests SC-10 and SC-12, both of which exposed vertical surfaces to a sodium pool. If reaction products are the controlling mechanism in concrete erosion, then a vertical surface should be eroded substantially since the reaction products are free to fall away under the force of gravity. In tests SC-10 and SC-12, greater erosion was observed than in horizontal tests but the erosion was not nearly so great as might be expected if reaction products are in fact the mechanism controlling concrete erosion. The observed erosion was an average of 9 cm in these tests as compared to 2.5-6.1 cm in a number of other tests. In fact, there is one horizontal test, LCT-1, with a 17 foot head of sodium, in which the erosion, 19 cm, was greater than in test SC-10 and SC-12. It should be noted that test LCT-1 was a 100 hour test whereas SC-10 and SC-12 were 8 and 24 hour tests, respectively; however, test length should not be particularly relevant in view of HEDL's assertion that nearly all concrete attack occurs within the first few hours. The point to be made is that if reaction products are a controlling mechanism in concrete erosion, then the average erosion should have been much greater in tests SC-10 and SC-12. Since the reaction products were free to fall away, erosion should have continued until the sodium supply was exhausted. This clearly did not occur.

The results obtained in HEDL tests SC-13, SC-14 and SC-19 also are of concern. In these tests, NaOH was added to the test initially. If sodium, rather than sodium hydroxide, is responsible for most of the attack on concrete, why was the attack on concrete in these tests as great as it was? Average erosions of 4.4, 5.7 and 3.8 cm have been reported, which are typical of tests where sodium hydroxide was not present initially. One might reasonably expect that the erosion would be substantially less than reported if indeed a sodium hydroxide layer is less reactive than sodium.

Finally, there is concern about the inability of sodium to wet and react with concrete directly. It has been reported in several places that sodium does not appear to wet hydrated concrete nor does it enter small cracks, presumably from a combination of surface tension effects and the counter flow of steam emerging from the cracks. The sodium hydroxide formed from the reaction of water and sodium does wet and react with the concrete surface and is responsible for the formation of the reaction product layer. It is hard to see how sodium would be able to react directly, as proposed, with a material that it does not wet.

C. Sandia's SCAM Model

A model and computer program, SCAM, has been developed at Sandia National Laboratories to describe the interactions between liquid sodium and <u>basalt</u> concrete [Reference 8]. An effort is being made to extend the SCAM model to limestone concrete at Sandia. Two "working hypothesis" have been developed; these consist of a "gas-phase model" and a "liquid-phase model". At present, this work has not been completed and is undocumented. However, it will be briefly described below.

In the SCAM model for limestone concrete, the sodium pool and concrete are divided into five regions [Reference 9]. The upper region consists of a sodium pool containing saturated sodium vapor and hydrogen bubbles. The second region consists of porous reaction products and liquid sodium. The third and fourth regions collectively constitute a "dry zone" in the concrete.

In the third region, closest to the sodium pool, sodium hydroxide and the concrete aggregate can react. Sodium vapor from the overlying pool of sodium diffuses downward passing hydrogen, CO_2 and water, which simultaneously diffuse upward through the layer.

In the upper part of the fourth region (the bottom of the dry zone), calcium carbonate decomposes due to heat, forming calcium oxide and releasing carbon dioxide. In the lower part of this region, bound water is released from the concrete.

Finally, the fifth region is separated from the others by a liquid evaporation plane and constitutes a "wet zone". In this region water migration occurs.

The gas phase reactions occurring in these regions are listed below:

$$2H_2O + 2Na \rightarrow NaOH + H_2$$

$$CO_2 + Na \rightarrow Na_2O + C$$

In the gas phase model, sodium enters the concrete pores, reacts and NaOH, Na_2O and carbon condense on the concrete aggregate. These reactions create an energy pulse in a localized zone. They are assumed to proceed at a steady rate and the model provides information on the rate of reaction but not the total extent of reaction. The model does not include a provision for either initiation or termination of concrete attack. The reactions are regarded as "secondary," following the initial set of reactions described in the liquid phase model discussed below.

In the liquid phase, the following reactions occur:

Step 1: $2Na + 2H_2O \rightarrow H_2 + (NaOH)_{Na}$ (9)

In this initial reaction, during the mild phase of the attack, sodium hydroxide is dissolved in sodium. In step 2, a sodium hydroxide layer forms when the sodium becomes saturated with NaOH:

Step 2: $(NaOH)_{Na} \rightarrow NaOH(l)$ (10)

In the Sandia hypothesis, liquid sodium hydroxide is essential to the process by which concrete is dissolved. It has been suggested that the need for a NaOH layer, which will be present only at the bottom of the sodium pool, predicts an absence of sideward attack. The next step is:

Step 3: $NaOH + CaCO_3 \rightarrow (CaCO_3)_{NaOH}$ (11)

Since the NaOH layer does not form immediately, the reaction of Step 3 is delayed. A delay time has been experimentally observed. In the next step, energetic attack occurs with the evolution of heat.

Step 4: $(CaCO_3)_{NaOH} + 6Na \rightarrow 3Na_2O + CaO + C$ (12)

The reaction accounts for the observed free carbon found during post-test analysis.

The NaOH can be regenerated so long as water is available from the concrete, as indicated.

Step 5: $Na_2O + H_2O \rightarrow 2NaOH$ (13)

In this model, the presence of NaOH is essential for concrete attack to continue. Consequently, tests can be misleading if the tests are either water or sodium limited.

If water is not available from the concrete, there will be unreacted Na_2O present which could not react in Step 5. When this occurs, the reaction below results:

Step 6: $Na_2O + NaOH \rightarrow (Na_2O)_{NaOH}$ (14)

When all of the NaOH is gone, further $CaCO_3$ dissolution ends and the reaction terminates. The sodium oxide dissolved in NaOH will eventually precipitate out as the solution cools.

SCAM employs the above information together with the continuity equation, the momentum and energy equations, the gas laws, diffusion equations and chemical kinetics to arrive at a solution. No attempt has been made to include the effects of spalling and cracking.

D. Comments on the Sandia Model

If sodium hydroxide is principally responsible for the erosion of concrete, as proposed by Sandia, then the greatest erosion should have occurred in Sandia test LS-6, and HEDL tests SC-13, SC-14 and SC-19. In these tests, sodium hydroxide was added to the sodium at the beginning of the test in order to provide immediate saturation. However, the penetration actually observed (less than 1 cm, 4.4 cm, 5.7 cm and 3.8 cm, respectively) was only about average.

Another problem area concerns HEDL tests SC-10 and SC-12. These are the only tests conducted with limestone concrete oriented vertically. If the Sandia hypothesis is correct, then the erosion in these tests should have been less than for horizontal concrete surfaces. The reasons is that the more dense sodium hydroxide layer will be at the bottom of the sodium pool in the Sandia model. Since it is primarily responsible for erosion, there should have been little erosion in a vertical test because very little sodium hydroxide would be in contact with the concrete. However, as has been mentioned earlier, the erosion observed was greater than average by a factor of approximately two. On the other hand, it must be noted that Sandia has observed relatively little attack on the crucible sidewalls in its own experiments; this observation supports its model.

Another problem area for this model is the observed interaction between liquid sodium and predehydrated concrete in HEDL test LCT-2. Since the water content of this concrete is low, little sodium hydroxide will form and, consequently, the attack on the concrete should be limited. However, the observed penetration was the greatest of any limestone concrete test. It could be that this penetration resulted from increased porosity arising from the dehydration process.

4. Discussion

In the preceeding section, the models proposed by HEDL and Sandia were both discussed. As was indicated, both models have problems in adequately describing the experimental results. In this section, the positions purported to be held by each organization on a number of issues will be discussed and further comments will be provided.

A. Erosion of Vertical Surfaces

Sandia and HEDL have proposed models which predict the opposite results for erosion of vertical walls. It should be possible, in principle, to eliminate one of the models on this basis.

Sandia's model predicts that sodium hydroxide attack on concrete is the dominant factor in concrete erosion. A NaOH layer, because of its density will be found at the bottom of a sodium pool. If it is the dominant reactant then there should be little erosion of a vertical concrete surface because the wall will be exposed over most of its area to sodium rather than NaOH. In support of their hypothesis, Sandia cites the limited erosion of the vertical surfaces of their concrete crucible tests.

On the other hand, HEDL's model predicts that sodium-concrete reactions are terminated by the accumulation on horizontal surfaces of a passivating layer containing reaction products and sodium hydroxide. Such products would be expected to fall away from a vertical surface and thus would expose fresh concrete to attack by sodium. In support of its contention, HEDL can cite its vertical surface tests, SC-10 and SC-12, in which greater than usual concrete erosion was observed.

Each group has its own test data to support its position. It should be noted that in some of the shallow pools used by Sandia, very limited sidewall erosion would be expected because they were so shallow. Other Sandia tests have employed deeper pools. It is not clear which of these experiments were the source of the data in question. Although HEDL's vertical surface tests do show greater erosion than horizontal tests, the erosion is less than might be expected if fresh concrete is continually being exposed to sodium if the latter is the dominant reactant. It is not apparent that there would be any limit to the extent of penetration of a vertical surface, other than sodium exhaustion, under the HEDL hypothesis.

The preceding discussion suggests that it should be possible to eliminate one or the other of the models by means of a test exposing both a vertical and horizontal concrete surface to the same sodium environment. If the vertical surface is eroded to a greater extent than the horizontal surface, HEDL's model is probably confirmed. If the opposite situation occurs, Sandia's model will be confirmed. While in principle, the answer could be inferred from a comparison of various HEDL tests, the experiment to experiment variability has been sufficiently great to make such a procedure dubious. Such a comparison is best done with both surfaces present in the same sodium pool so that both will be exposed to an identical initial environment.

Hanford has indicated that they believe that vertical wall erosion is irrelevant to the CRBR. The vertical wall will be protected by a layer of pearlite concrete and a steel liner. In HEDL test SET-12, the pearlite concrete reacted completely but appeared to protect the underlying limestone concrete from extensive erosion. HEDL further points out that they do not believe that there is a scenario in which the steel liner and pearlite would not be present, thereby exposing the vertical wall to direct attack. It is possible to conceive of a situation where core debris piles up against a wall and destroys the steel liner and pearlite concrete so that the limestone concrete is exposed to sodium. However, an assessment of the probability of such an event is beyond the scope of this work.

Finally, in regard to HEDL's last contention, it should be noted that there was concern some years ago about the possibility of cracks developing in the steel liner over a period of time. There was also concern that cracks might be present initially unless care was taken in fabrication. Such cracks could provide a path for sodium to enter a liner and attack a vertical wall. Whether cracks can be present initially or develop later should be examined further.

B. Role of Sodium Hydroxide

All parties agree that substantial quantities of NaOH will form due to the interaction between liquid sodium and water driven from the concrete by heat. As just discussed, the subsequent behavior of NaOH is disputed by HEDL and Sandia. HEDL has proposed that NaOH is less reactive than sodium, and thus form a passivating layer between the sodium and concrete. On the other hand, Sandia believes that NaOH is the dominant reactant species. Evidence from vertical wall erosion has already been discussed in this regard. However, there is additional information from HEDL test LCT-2, that is relevant here.

The very recently reported results of HEDL test LCT-2 describe a test made with a sodium pour on dehydrated limestone concrete. This concrete was heated at 1000°F for 24 hours to achieve 80% dehydration. When sodium was poured on the concrete the reaction was described as being very benign with virtually no hydrogen release and little energy generation. A pour of 46 kg of sodium was totally consumed in 70 hours producing an average penetration of 13 inches. Presumably, penetration would have continued if additional sodium had been present. The penetration of 13 inches is substantially greater than the 3 inches typically found in the HEDL tests with normal limestone concrete.

Since this concrete was dehydrated, there was little water available to react with sodium to form NaOH. Thus, the extensive erosion seems to support the position of HEDL that erosion by sodium is the dominant process. However, it is possible that the observed erosion was enhanced by physical and chemical changes induced in the concrete by the dehydration process. It is not certain that the additional penetration was caused by the absence of a NaOH layer.

Another issue worthy of some exploration is the water loss observed in the Sandia tests. The larger tests conducted initially by Sandia had open outside surfaces through which, it was reported, came substantial quantities of water. With water escaping through the outside surfaces, one would expect that less sodium hydroxide would be generated thereby slowing the formation of a NaOH layer. This observation could enable one to explain some of the early energetic reactions in the Sandia tests on the basis of HEDL's model. By contrast, in HEDL's well constrained tests, water driven from the concrete could exit only through the sodium pool where it would contribute to HEDL's proposed passivating NaOH layer.

C. Nature of the Layer Between the Sodium and Concrete

As discussed previously, the layer between the sodium pool and the concrete surface consists of a mixture of sodium hydroxide and reaction products. While everyone agrees that both sodium and sodium hydroxide both can attack concrete, there is disagreement concerning the relative importance of the two reactants. In HEDL's views, sodium is responsible for most of the attack on concrete and it is prevented from reaching fresh concrete by the presence of the NaOH-reaction product layer. Thus the reaction product layer serves to limit the extent of sodium-concrete reactions as it develops. On the other hand, Sandia regards NaOH as principally responsible for the reactions with concrete. If this is true, then the only limit to the extent of reaction will be exhaustion of the reactants (sodium and water driven from the concrete). It is our purpose here to examine what is known about this layer.

One of the difficulties in studying the reaction product layer is that we can only observe it after the test is over and the material has frozen. When a test specimen is sectioned, this layer appears to be homogeneous and gives the appearance of having been a liquid. The layer looks similar to sandstone. There is no aggregate or included sodium visible. At room temperature the layer appears to have a substantial mechanical strength.

When analyzed, the layer is found to contain significant quantities of sodium hydroxide. Samples of this layer will melt at the temperature of the sodium used during the reaction. The properties of the fluid are disputed and are relevant here. HEDL believes that the liquid is viscous while Sandia believes that it flows like water, based on experiments in which some of the material was remelted. The fluid does wet the concrete and forms a layer separate from the sodium pool.

The volume of the reaction product layer is quite substantial. Contributing to this volume is the substantial increase observed in the volume of concrete as it reacts with sodium hydroxide. When one observes this thick layer after a test, it is easy to believe that it would prevent contact between sodium and concrete in the HEDL model. However, its character when molten, as noted, is disputed.

It has been suggested that this layer might be displaced by convection currents in the deep sodium pool of the reactor. Other factors that could displace the layer include gas evaporation and spallation. Whether the layer is readily displaced by any of these mechanisms will depend on its viscosity, which is disputed.

Finally, it should be noted that if the view of Sandia is correct, this intermediate layer is the cause of most of the erosion of concrete. If this is true, the viscosity and displacement of the layer are not relevant in the erosion process.

D. Temperature of the Sodium

It is the view of most observers that there is a temperature threshold for sodium-concrete reactions. However, a range of values has been suggested and the threshold is probably strongly dependent on the nature of the concrete in question.

There are two difficulties in this area with HEDL's tests. First, the sodium in some of their tests has been allowed to cool to a fairly low temperature immediately after being poured on concrete. Sandia has asserted that this initial quenching process might in some manner affect the chemical process occurring and may inhibit the development of an energetic reaction. There is some merit to this point of view.

A more serious objection can be made to HEDL's conduct of nearly all experiments at a very high temperature, near the boiling point of sodium. While it is true that reaction rates usually increase rapidly with temperature, there are exceptions, which can result from competing reactions as well as in other ways. In particular, a review of Na-NaOH phase diagram data [Reference 9] suggests that the composition of the sodium pool and the development of the NaOH rich liquid layer are dependent on the pool temperature. Consequentially, the results of tests conducted at lower temperatures may be significantly different. It is suggested that some of HEDL's future tests should be conducted at lower temperatures.

E. Cracking and Spallation

The subject of cracking is highly complex in such a non-uniform material as concrete. The stress patterns in a given sample depend not only pre-existing stresses, but also on temperature distribution, the effects of chemical reactions, and external mechanical forces. With all other factors the same, the size of cracks in a sample would be expected to increase with sample size.

Cracking and spallation can provide a means for reactants to reach fresh concrete and thus can enhance erosion. HEDL has expressed the view that the effects of cracking could be serious in terms of their model. The protective layer of reaction products, which they have proposed limits the reaction, could drain away into any cracks thereby exposing fresh concrete to sodium attack.

In tests up to this time, no tendency has been observed in experiments at Sandia for sodium to enter cracks in blocks of concrete, even when there has been energetic attack. However, at some point sodium will enter a crack if it is wide enough or if the pressure is great enough.

HEDL's concern over the effects of cracking and spallation has led them to be concerned about the method of restraint of specimens in their tests. The HEDL test specimens are usually very firmly restrained in a surrounding collar of concrete which is intended to more closely simulate the reactor cavity floor. By contrast, the Sandia crucibles have been unrestrained and may consequently be more prone to cracking. This does not necessarily make the Sandia tests unrepresentative since the reactor cavity clearly does have corners. The Sandia tests should represent the corners of the reactor cavity better than HEDL's and HEDL's tests should provide a better representation of the center region of the cavity floor. The effect of rebars on cracking and spallation was examined in HEDL test LSC-2. Some local cracking was observed due to differential thermal expansion but it was concluded that rebars would not greatly increase cracking.

HEDL believes that any spallation that occurs will tend to be limited to the region close to the top surface of the concrete. While a spalled layer may spall again, they believe that spalling is a surface phenomenon which will not proceed indefinitely. When the thermal gradients in concrete level out, they feel that spallation will end.

Sandia has observed the appearance of a large chunk of concrete in one test, P-3. The chunk was approximately 25 cm in diameter and 7 cm thick. Its origin is uncertain but it probably separated from the main mass of the concrete crucible by either cracking or spallation.

Spalling also is a scale dependent phenomenon. Certainly, one would not expect to see a chunk of material of the size produced in Sandia test P-3 in a small scale test. It is by no means certain, in our view, that any tests conducted to date have been on a scale sufficiently large to demonstrate conclusively that extensive cracking and spallation will not occur. Unfortunately there is currently no way to make theoretical predictions regarding either cracking or spallation that would be meaningful.

The method of restraint probably should be a variable in some tests in order to assess its importance in promoting cracking and spallation processes. This type of information could provide some data that might be useful in assessing the extent of scaling effects.

F. Effect of Core Debris

The are at least two scenarios for the attack of core debris on concrete. In one, the core debris does not form a coolable debris bed and melts into the concrete. The formation of a crust on top of the core debris could prevent cooling once melting is initiated. The development of crusts in the presence of coolant has been observed in experiments with water and molten metals attacking concrete in experiments conducted by Dr. Peehs in Germany [Reference 10]. Data developed in experiments at Sandia [Reference 11] suggests that the erosion rate for steel on concrete at 1700°C could be as high as 25 ± 15 cm/hr. For oxide fuels at 2800°C, an erosion rate of 130 ± 50 cm/hr has been measured in tests.

Even if the core debris is initially coolable, it will tend to sink into sodium-concrete reaction products due to its greater density. It is not clear that initially coolable core debris mixed with concrete reaction products would remain coolable. Once melting begins, crust formation could prevent cooling by sodium. Substantial concrete erosion could result, either because of direct attack by core debris or exposure fresh concrete to either sodium or sodium hydroxide.

In view of the potential for substantial erosion of concrete induced by core debris, it is urged that studies should be initiated to review the feasibility of core retention devices constructed from refractory materials such as MgO. Emphasis should be placed on the selection of materials that are compatible with sodium.

C. Reactant Limited Tests

Most of the tests conducted by both HEDL and Sandia have been reactant limited. In this context, both sodium and water form the concrete reactants.

Sandia argues that HEDL's tests should have used thicker concrete specimens so that additional water would be driven from the concrete into the sodium pool where it could react and form sodium hydroxide. If Sandia's model is correct, the added NaOH would increase erosion of concrete. On the other hand, if HEDL's model is correct, the added NaOH would have little effect. This suggests that a comparison of tests with identical conditions except for concrete layer thickness could provide evidence for or against Sandia's model.

HEDL argues that Sandia's tests have been sodium limited and that erosion would have stopped by itself soon after the Sandia tests ran out of sodium. Clearly, the Sandia tests were sodium limited and more sodium should have been available because the determination of the total extent of penetration is an important issue.

Arguments have been made in the past against those Sandia experiments that used shallow pools. It is felt that pool depth should be a variable. Our understanding of the phenomena in these tests is limited and the investigation of the influence of various parameters, such as pool depth, should not be precluded.

For example, Sandia has tried to explain the delay in the onset of the energetic reactions that they observed in terms of pool depth. If the concrete is principally attacked by NaOH, rather than by sodium, a longer time will be required for attack to occur in a deeper pool. A longer time is required in order to saturate the sodium with NaOH when the pool is deep. The NaOH has only limited contact with the concrete until the pool is saturated. At that time, a separate NaOH layer forms in contact with the concrete allowing the attack to proceed expeditiously in the Sandia model. In the Sandia shallow pool test, this saturation would have occurred relatively quickly, leading to rapid penetration. In tests with deeper sodium pools, a longer time should be required for saturation.

H. Erosion Rate and Total Penetration

The greatest total penetration observed with hydrated concrete occurred in the recent HEDL test LCT-1. The test employed 46 kg of sodium covering a 60 cm (two foot) thick layer of limestone concrete with a diameter of 35 cm (13.5 in.). The test lasted for 100 hours but HEDL believes that nearly all of the erosion occurred in the first 5 hours.

A similar total penetration was observed in Sandia test P-3 several year ago. An erosion of 15.2 cm was observed in a test that lasted only 3 hours. Whether erosion would have continued is unknown.

As discussed earlier, the HEDL model predicts a self-limiting reaction in which reaction products, mixed with sodium hydroxide, prevent sodium from directly attacking concrete. The apparent termination of attack in the 100 hour tests at HEDL supports the hypothesis that reaction products can limit concrete erosion.

On the other hand, the Sandia model explains these results on the basis of the test being limited in an essential reactant, water. The Sandia model would have the NaOH continue to attack concrete until all of the NaOH is consumed. With the relatively thin concrete layer employed (compared to the quantity of concrete available in CRBR), the attack is viewed as limited by a lack of water that can be driven from concrete.

For CRBR, HEDL has indicated that they expect that the total concrete erosion would not exceed 6-9 in., based on their worst case test, LCT-1. Sandia on the basis of its model has calculated that the water present in the CRBR concrete could support an erosion of about 30 in.

The problem with accepting the total penetration that HEDL has proposed is that it is based on a small number of empirical observations from experiments that are much smaller in scale than the conditions that would exist in an actual accident. If the erosion process was better understood or if experiments could be conducted on a scale similar to reactor accident conditions, then the proposed total penetration could be more readily accepted.

In view of the uncertainties imposed by scale effects, it is our view that the most appropriate limit for sodium penetration into concrete is that imposed by exhaustion of water, or 30 in. of concrete. It must be noted, however, that even this value may not be conservative, given the data base and controversy over the basic phenomena.

There is even less reliable data to use as a basis for selection of an erosion rate. An initial rate of 2 in/hr has been suggested by HEDL based on their experimental observations. Other have suggested a 7 in/hr rate for 3 hours with a 1 in/hr rate thereafter. Given the scale effect uncertainties, a rate as low as 2 in/hr is difficult to support. On the other hand, the 7 in/hr rate may be overly conservative. There is no really good basis for making a choice, given the kind of data available and the controversy over the phenomena responsible for erosion. If it is necessary to select a rate, then the more conservative 7 in/hr rate for 3 hours, followed by a 1 in/hr rate seems a better choice.

It is also our view that the interaction between core debris, reaction products and concrete may be potentially quite serious as discussed earlier. Core debris-concrete interactions could cause very large erosion rates. For this reason the feasibility of a core retention device that is compatible with sodium should be examined.

I. Comments on Past and Future Experiments

Additional experiments should not be initiated at those laboratories that are behind in documentation of experiments until the backlog of undocumented work has been eliminated and the data have been made generally available. One difficulty encountered in the preparation of this report was a lack of documentation for experiments particularly those conducted at Sandia. There does not appear to be a consistent scheme for numbering the Sandia experiments so that confusion can be avoided. HEDL has generally documented its experiments quite well and has prepared an excellent review [Reference 4].

Experimentalists in the past have failed to obtain all the information that could be obtained from their experiments. In view of the controversy regarding the erosive processes, this is unfortunate. Information from specimens taken in the reaction product layer and in the concrete (particularly at the interface) could be helpful in developing an understanding of the interactions occurring. In addition to the conventional chemical analysis, specimens should be studied in an optical microscope and in a scanning electron microscope. Instrumental analysis, by ion microprobe mass analysis, X-ray diffraction and other techniques should be possible for concrete specimens, if not in the reaction products due to the presence of sodium. An example of a microstructural examination of concrete from sodium-concrete interface can be found in Reference 2.

We would like to be able to say that additional experiments should be performed to improve our understanding of the phenomena. However, in view of the limited progress that has been made towards resolution of the issues in this area over the past few years, it is difficult for us to be optimistic.

The issue of scale effects is especially difficult. It is not financially feasible to conduct a statistically acceptable number of experiments of a size that precludes significant scaling effects.

A few limited areas for tests are suggested. Elsewhere, a vertical concrete wall test was suggested since the models predict different results for this configuration. Future tests by both laboratories should have an adequate supply of sodium and a sufficiently thick concrete layer to properly represent water release from concrete in the CRBR.

Separate effects test to examine the effect of both sodium and sodium hydroxide on the cement, aggregate and other constituents would be useful. Dehydrated concrete probably should also be included. The purpose of these tests would be to isolate the relevant chemical reactions and establish the relevant chemical processes. It should be possible in principle to resolve issues concerning chemistry in small scale tests.

Finally, there is a need for greater cooperation and coordination of work between the two groups to avoid the current pattern of a scattered group of experiments that cannot be related to each other.

J. Selection of a Model for Sodium-Concrete Interactions

Discussion of this issue has been delayed until this point because there currently is no confirmed model that can reliably predict the extent of sodium concrete interactions.

Although a number of experiments have been conducted to study liquid sodium-limestone concrete interactions over the last 7 years, the situation really has changed little since our earlier review in Reference 1. Many of the same arguments are being made and the same points of view are being advocated by the same parties. The inconclusive nature of the experimental data lends itself to varying interpretations. As a consequence, reasonable people can disagree as to the conclusions that can be reached in interpreting the experimental results currently available.

As the discussion in the preceding sections has shown, arguments can be made for and against each of the two models. Neither model is supported by a preponderance of evidence. Consequently, it is not possible at present to select one and eliminate the other. Of the two models, the HEDL model currently seems to best fit the available data. However, there are observations and problems noted elsewhere in this report, which are not adequately explained by it. Additional experimental data may change this conclusion. Several years ago, the Sandia model seemed to fit the data better on the basis of the data available then.

In addition to the concerns raised earlier regarding the HEDL model, there are two others. First, the model remains speculative. Reaction products probably do tend to inhibit concrete erosion. Unfortunately, although this model was proposed some time ago, progress towards a resolution of outstanding issues has been limited. The proposed chemistry remains unconfirmed and the properties of the reaction product layer are not understood. Some possibilities that might be considered include studies to confirm the proposed chemical reactions and examine their chemical kinetics, determination of the viscosity of the proposed viscous protective layer and investigations of the permeability of the layer to sodium. It is recognized that it may be impractical to conduct such studies in a high temperature sodium environment; on the other hand, some information on sodium properties does exist in the scientific literature which suggests that studies are possible. Given the controversy that has surrounded this area, some confirmation of the elements of the HEDL model is needed before it will be generally accepted.

The second concern with regard to attempting predictions of sodium-concrete behavior based on HEDL's model concerns the effects of scale, especially in regard to spallation and cracking. Neither the HEDL or Sandia models account for either. HEDL believes that cracking and spallation will be limited in a reactor cavity. This may in fact be true, but there is evidence from Sandia test P-3 that at minimum raises questions about that assumption. Given the present state of knowledge, it seems possible that cracking and spallation will be limited in an accident in a reactor cavity, but we cannot be certain on the basis of the evidence currently available.

In conclusion, it is not possible at present to conclude that either of the models is significantly better than the other one. It is also unlikely that this situation will change in the near future. Since the issues probably cannot be adequately resolved in a time relevant for CRBR, strong consideration should be given to the use of alternate materials in the reactor cavity. The use of high alumina cement or magnesia might reduce the threats posed by sodium-concrete reactions. A program should be initiated to examine alternative materials as soon as possible.

K. Hydrogen Generation

As mentioned earlier, the chemistry of sodium-concrete reactions is the subject of considerable controversy. Until the chemistry is understood, there will be no way to adequately predict the extent of hydrogen production.

The problem is further complicated by the potential for a reaction between sodium and hydrogen, forming NaH, at lower temperatures. Above 500°C, the equilibrium favors dissociation back into sodium and hydrogen. In addition, NaH will react with any water present over a wide range of temperatures, thereby regenerating hydrogen.

5. Conclusions

After reviewing sodium-limestone concrete interactions, we have reached the following conclusions:

1. A total sodium-limestone concrete penetration of 30 inches is recommended for the CRBR. There is no basis for recommending an erosion rate.
2. The chemistry of sodium-limestone concrete interactions remain speculative. Hypotheses have been proposed that seem reasonable but are as yet unconfirmed. The role of sodium hydroxide has not been established.
3. Sodium pool temperature may affect the chemical reactions that occur and the reaction products that form. It is not clear that a lower temperature sodium environment is more benign than a high temperature environment.
4. Sodium pool depth may affect the time required for the occurrence of a reaction and the extent of reaction.
5. Dehydration of the concrete may affect the extent of sodium penetration.
6. Reaction products may limit the extent of concrete penetration.
7. The physical aspects (cracking, spallation) of sodium-limestone concrete interactions are poorly understood.
8. The questions of the effects of scale, geometry and mode of restraint remain unresolved.
9. The interactions between core debris, reaction products and concrete are potentially serious and require additional research. As a consequence, the possibility of using other materials to protect the concrete from sodium and core debris should be assessed in the near future.
10. In view of the limited progress that has been made in understanding sodium-concrete interactions during recent years, it is probable that the controversies will not be resolved in a time that is meaningful for CRBR.

11. Future tests should employ sufficient sodium and concrete so that they will not be either sodium or water limited.

12. A comparison of erosion on horizontal and vertical surfaces in a single test may provide a test of the two proposed models for sodium-concrete interactions.

13. Greater emphasis should be placed on post-test analysis of specimens from the concrete and reaction product layers.

14. All experiments conducted to date should be fully documented before additional work is initiated.

15. Further research is needed on both chemical and physical aspects of sodium-concrete interactions. Given the differences between the experimental conditions, it is difficult to coherently correlate the results. More detailed coordination between the laboratories in selecting experiments and experimental paramaters for study would be desirable.

6. References

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A.5 RADIOLOGICAL ASSESSMENT

This section presents the staff's evaluation of the applicants' postulated core disruptive accident (CDA) scenarios as presented in CRBRP-3 (Reference 1) with regard to the calculation of radiological consequences. The evaluation begins with the path of the radionuclides from the disrupted core to the environment and then looks at the dose calculations and their sensitivity to alternative circumstances. Our review and evaluation have been conducted within the framework of the general guidelines discussed in Section A.1. More specifically we have considered whether the realistically evaluated doses resulting from venting containment after CDAs are likely to exceed the dose guidelines of 10 CFR 100.

Whether specific radionuclides are released to the environment depends on the mode of their release to the containment atmosphere, on the conditions of the containment atmosphere, and on the containment system mode of operation during the period the radionuclides are in the containment atmosphere available for release. Figure A.5-1 shows the relative timing of these conditions for the core disruptive accident which the applicants have designated their base case in CRBRP-3 (Reference 1). The approximate timing for the staff's realistic upper bound case would have hydrogen ignition at about eight hours, venting at about 22 hours and boildry at some time greater than 70 hours.

The applicants have modeled the reactor and containment structures for calculations with the CACECO code. The results of CACECO code calculations form a significant part of the basis for their scenario of conditions and events within the Reactor Containment Building (RCB) following the time of the initial release from the CDA.

A.5.1 Modes of Containment Atmosphere Release

The release from the containment atmosphere has three design modes, all through efficient filtering systems. The first is design basis leakage, a function of pressure but established as 0.1 volume percent per day at 10 psig, with an accompanying bypass (i.e., unfiltered) leakage one-one hundredth as large. The principal (non-bypass) part of the design basis leakage, leaks into the annular space between the steel containment shell and the confinement structure. When there is a non-routine release of radioactivity to the containment atmosphere, the signal from radiation monitors initiates containment isolation and an increase of flow in the annulus ventilation from about 3000 cfm to 14000 cfm. The flow is then drawn through the annulus filtration system. About one quarter of the filtered air is exhausted to the environment from the top of the reactor containment building and the remainder is recirculated to the annulus. The air flow exhausted to the environment is that amount needed to maintain the annulus at a slight negative pressure (1/4 inch water gauge) so as to assure capture of leakage from the steel shell containment.

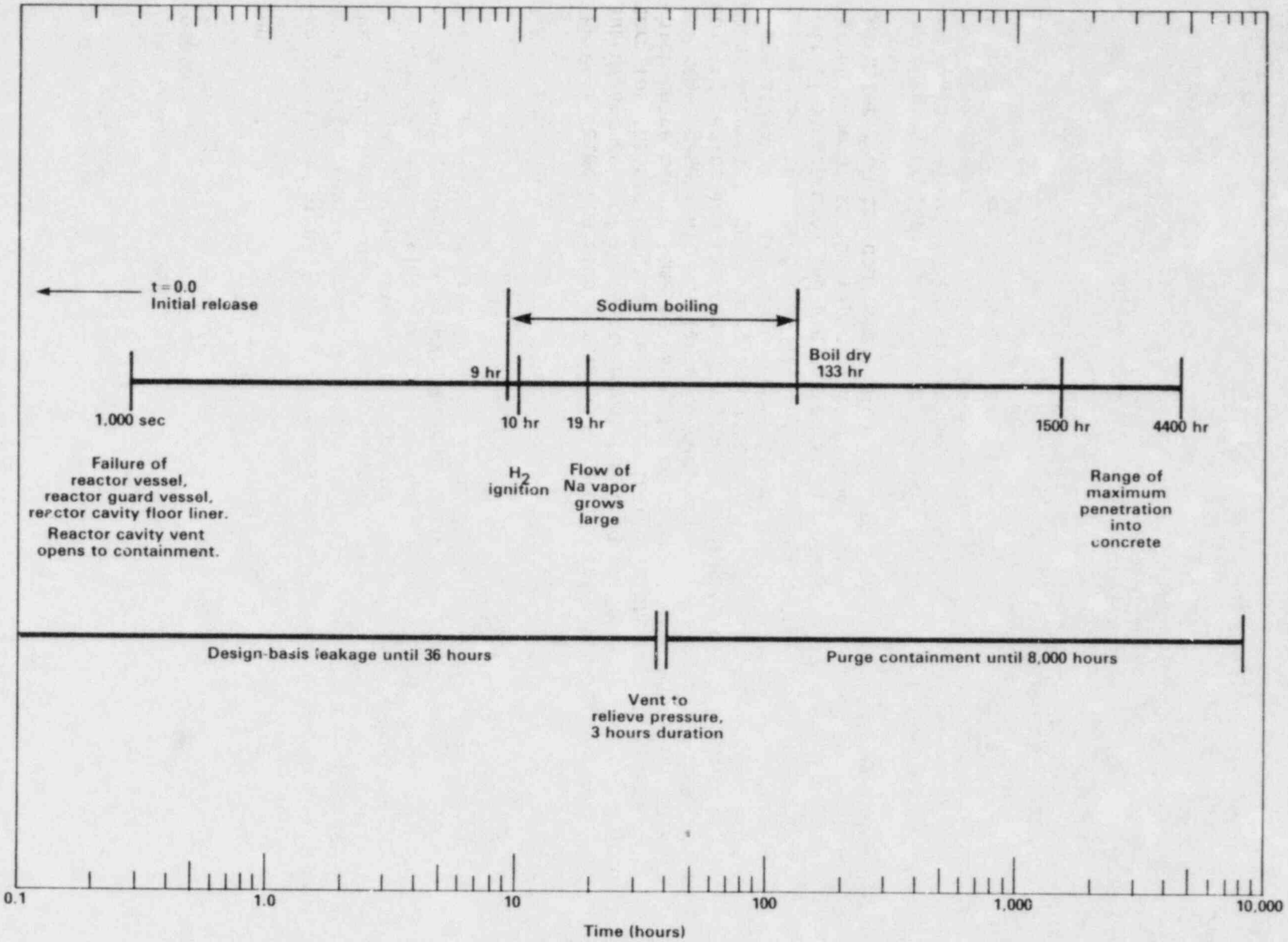

Figure A.5.1 Chronology of CDA release model applicants' base case

The second mode of release, for beyond-the-design-basis events only, is by controlled venting of the steel shell containment. When the operator decides to vent, he opens the vent system isolation valves. The decision to vent may be based on either high pressure or hydrogen buildup. Venting occurs at a controlled rate through a wet filtration system located outside the containment building. When the controlled venting is initiated, the annulus filtration system is turned off. The exhaust from the containment cleanup system is released to the environment from the top of the RCB.

The third mode of release, also for beyond-the-design-basis events only, occurs after venting drops the pressure of the atmosphere in the steel shell containment to ambient atmospheric. Then, in order to maintain low hydrogen concentration, the containment atmosphere may be purged by admission of outside air into the containment. Simultaneously, the containment atmosphere is exhausted through the containment cleanup system (the wet filtration system). This exhaust is also released to the environment from the top of the RCB.

A.5.2 Sequence of Conditions Within Containment

The postulated CDA also initiates a sequence of conditions within the subcompartments of the containment building which determine the source term for the radiological releases. Initially there is a period in which there can be some releases to containment atmosphere (especially the noble gases) without large concentrations of aerosols. When sodium boiling begins and the concentration of sodium vapor in the vented gases becomes large enough, the vented gases ignite, forming a large flame at the vent exit in the containment. Then, as sodium boilup progresses, the reactor cavity and pipeway cells are flushed of their initial atmospheres and are heated so that eventually there occurs a strong flow of sodium vapor (with hydrogen from the reaction of sodium and water from the concrete) which is vented to the containment atmosphere.

The burning of this large sodium vapor flow fills the containment atmosphere with a large concentration of aerosols, primarily of sodium oxide and other sodium compounds. This atmosphere is kept well mixed by the large flame. Within a short time the containment atmosphere reaches a quasi-equilibrium in which the rate of depletion of the aerosols by fallout and other processes is about equal to the rate of aerosol production from the sodium vapor injection. This continues until the sodium pool is boiled dry, after several days. This period, with a large flow of sodium vapor to the containment atmosphere, is perturbed by the ventdown of containment pressure. This pressure relief causes a flashing of sodium. Thus, for a few hours, a greater flow of sodium vapor occurs, followed by a few hours with a diminished flow. The applicants estimate that about three quarters of the 500 metric ton sodium pool will be boiled up to the containment. The staff concurs that this is a reasonable amount of the sodium to be involved.

At the time of sodium boildry, the remaining materials will begin to heat up due to decay heat from the core debris. The staff expects that thereafter there is a release of most of the remaining fraction of the materials that are only somewhat less volatile than sodium.

After boildry, the staff considers that the pool of core materials and sodium-concrete reaction products will heat up until it is further diluted with melted concrete and products of continuing core debris reactions with the concrete. During this period there is a greatly reduced flow of aerosols to the containment atmosphere, and the aerosol composition is changing. Eventually, the aerosols consist only of particles of the least volatile materials, transported by the steam, carbon dioxide and other gases (possibly including hydrogen and carbon monoxide) and vapors resulting from the reaction of the molten pool with the underlying concrete. Over a period of several months, the molten pool decay heating rate diminishes and the pool approaches a stable condition; aerosol release to the containment atmosphere gradually decreases to zero.

About 11 months after the CDA, the conditions within the RCB may become static, permitting purge termination.

A.5.3 Radionuclide Release Groups

At the time the core is initially disrupted, the most volatile radionuclides will be released from the disrupted core materials into the reactor vessel. The noble gases Kr and Xe will pass through the sodium to the cover gas. Other volatiles, the I, Br, Cs, Rb, Se, Te and Sb, will be released into the primary sodium. When a CDA occurs, it may involve the whole reactor (fuel, blankets and other materials) or perhaps, for example, only one third. For the purposes of radiological assessment, it is usually conservative to consider the whole reactor involved; e.g., release of 100% of the noble gases is considered to occur at once. The applicants' base case release model also includes an initial release of 1000 lb sodium containing 100 ppb plutonium, plus 0.026% of the reactor inventory of fuel radionuclides, solid fission products and halogens. This initial release is considered to pass through the head seals at the beginning of the CDA, $t = 0.0$.

Upon failure of the reactor guard vessel, the sodium enters and reacts with the reactor cavity atmosphere, causing the reactor cavity venting rupture discs to open, and the Kr and Xe may be vented into the containment atmosphere. The staff considers that, alternatively, they may be trapped in the PHTS to be released later. In either case, it is conservative to assume that they are released to the containment atmosphere at the initiation of the CDA where the Kr and Xe will be available to be leaked to the environment in accordance with the design basis leak rate until the time of venting, when a major fraction would be released to the environment. The remainder would be released with the subsequent purge flow, within the next 24 hours.

One hundred percent of the volatile halogens, iodine (I) and bromine (Br), are assumed released into the primary sodium at the beginning of the CDA. Their fractional release to the containment atmosphere is modeled by the applicants as being directly proportional to the fraction of sodium boiled up into the containment. The staff considers this a conservative approach because the halogens would tend to be retained in the sodium pool, providing time for appreciable radioactive decay of most halogen radionuclides. Further, most of the sodium is boiled up to containment after the venting initiation time, so that the mode of release of the accompanying iodine to the environment is by the TMBDB vent and purge flow, which passes through the TMBDB cleanup system before release.

The elements other than the noble gases and halogens which are considerably more volatile than sodium are cesium (Cs) and rubidium (Rb). When released from the core materials at the beginning of the CDA, the Cs and Rb would be likely to be dissolved in the sodium. The applicants have, however, modeled their release as an immediate 100% release to the containment atmosphere. This approach results in the Cs and Rb being considered available for design-basis-leakage release until they are depleted from the containment atmosphere by fallout. The following discussion shows this to be a reasonable approach. The Cs and Rb dissolved in the sodium, because of their greater volatility, will be released from the sodium pool when the sodium begins to boil. However, the factors which delay a large flow of sodium to the containment atmosphere will also delay the Cs and Rb. The staff considers that they may be released to the containment atmosphere at about the time a strong flow of sodium vapor first arrives there, at 19 hours in the applicants' base case, about 17 hours in advance of venting. The Cs and Rb enter at the vent flame and are oxidized along with the sodium. They will then be depleted with the rapid fallout due to the large sodium aerosol concentrations.

Other volatile fission products are selenium (Se), tellurium (Te) and antimony (Sb); these are considered dissolved in the primary sodium at the beginning of the CDA. Their fractional release to the containment atmosphere is modeled by the applicants as being directly proportional to the fraction of sodium boiled up into containment, 100% all together. These elements are generally less volatile than sodium and the staff considers that they would largely be retained in the sodium pool until about the time of boildry; at that time when the remaining mass of core debris and reaction products heats up, they would be among the first to be released.

The applicants have taken the approach of modeling the release of Ba, Sr, and all other non-volatile fission products, such that one percent of the inventory of each is released to the containment atmosphere along with the boiled up sodium. The elements Ba and Sr are only slightly more volatile than the uranium and plutonium oxides and the remaining fission products. Because of their low volatility, little will be released to the containment atmosphere. That part not released into the sodium will eventually be available for sparging from the pool of core debris, molten concrete and reaction products. The staff considers that appreciably less than one percent of the Ba and Sr would be sparged to the containment atmosphere, and therefore the applicants' approach may be regarded as reasonable.

The largest group of radionuclides are those which can be classified as least volatile. Two modes of release to the containment atmosphere are considered of possible significance for these solids. At the time of core disruption, when molten or vaporized core materials are quenched in the primary sodium, they fragment into particles ranging in size downward from about one millimeter diameter. An appreciable fraction, estimated at 15%, may form particles so small that they remain in suspension in the sodium. A small part of this material, about one part in one thousand, is carried over with boiled up sodium to the containment atmosphere. The other mode of release is by gas sparging of the post-boildry pool of core debris, molten concrete and reaction products. In this mode, gases released from the underlying concrete reaction zone, bubble through the molten pool and entrain a small fraction of the solids, which then is carried to the containment atmosphere. The staff considers the applicants' modeling approach, which releases 1% of these solids to the containment with the boiled up sodium, to be conservative.

Plutonium is among the most radiologically significant of the radioactive materials in the reactor, present as plutonium oxide in the core fuel and in the blankets. Plutonium oxide and uranium oxide have low volatility and their release fractions will be among the smallest. Their releases to containment are modeled by the applicants as 0.015% carried over with the sodium vapor during sodium boiling and another small fraction, much less than one percent, sparged from the post-boildry pool of materials and carried up with the gases released from the concrete. For plutonium, the applicants estimate that these amount to about 320 grams and 26 grams, respectively, that are released to the containment atmosphere.

The staff considers that the applicants' estimates of plutonium releases to containment involve the principal uncertainties in the radiological consequences. The estimates consist of only the above 320 grams and 26 grams, plus about 546 grams in the 0.026% initial release of fuel and about 0.045 grams in the initial release of 1000 lb sodium. These are each small fractions of the core inventory of roughly 2.1 million grams; an error in the retention factors could impact estimated consequences significantly. The staff believes that the most significant area of uncertainty is the amount of plutonium carried up with vapor from the boiling sodium. The experimental results of Jordan and Ozawa

(Reference 2) are in general directly applicable and show partitioning at the boiling surface of more than a factor of 1000. However, the staff considers that the sodium boiling rate (Reference 3), and potential chemical differences between experiment and CDA circumstances, which may influence the formation and release of sodium plutonates (Reference 4), introduce substantial uncertainty into the estimate of the 320 grams of plutonium boiled up with the sodium. Based on review of References 3 and 4, the staff believes this uncertainty to be not more than a factor of ten. Therefore, the staff has used 0.16% of the inventory of fuel radionuclides and non-volatile fission products as the fraction boiled up to containment with the sodium. This fraction is considered large enough to also account for the minor amounts of the same radionuclides that might be sparged up after boildry.

A.5.4 Calculation of Radiological Consequences

The applicants have presented doses calculated with computer codes HAA-3 and COMRADEX (Reference 1). The release of radioactivity to the environment depends on the concentration of radioactive materials in the containment atmosphere and upon the rates at which they are added and removed. The applicants have calculated the time dependent suspended aerosol concentration in the containment and the rate of aerosol depletion with the computer code HAA-3, taking into consideration the source generation rate, the aerosol deposition rate, and the rate of removal by leakage, venting or purge flow. The output of the HAA-3 code serves as input to the COMRADEX code. The staff has found that the HAA-3 code tends to overestimate the suspended aerosol concentration and therefore provides conservative estimates of the amounts leaked.

The applicants have used the COMRADEX computer code to calculate radiation doses. COMRADEX includes radioactive decay within containment in the calculation of the rates of release of radioactivity to the environment. Using 50% frequency atmospheric dispersion parameters, the applicants used COMRADEX to calculate radiation doses at the exclusion area boundary and at the low population zone boundary. The calculation includes doses from direct gamma shine, inhalation of radioactive material and submersion in the radioactive cloud. Table A.5-1 lists some of the data and assumptions used in the dose calculations. Table A.5-2 presents the calculated doses; as shown the applicants calculated CDA doses, based on use of best estimate rather than conservative assumptions, are smaller than the 10 CFR Part 100 dose guidelines for design basis accidents.

A.5.5 Comparison of Dose Calculations

In Table A.5-2 the doses from the CRBR Site Suitability Source Term calculation are shown with the applicants' TMBDB base case doses.

The Site Suitability Source Term (SSST) results are from Reference 8. The SSST consists of 100% of the core inventory of noble gases, 50% of the iodines, and 1% of the solid fission products and plutonium; all are released instantaneously into the containment atmosphere, in a non-mechanistic manner. Releases to the environment and doses were calculated conservatively, e.g., using 5% meteorology, and the calculation considered only the design basis leakage release mode, because the calculation was made to satisfy the requirements of 10 CFR Part 100.

Table A.5-1 PARAMETERS IN DOSE CALCULATIONS [a]

Power Level, MWt - 975		
Core Inventory - End of Equilibrium Cycle		
Initial Plutonium Composition - FFTF Grade		
Containment Volume, ft^3	3.6×10^6	
Containment Leak Rate, %/day at 10 psig	0.1	
Bypass Fraction	0.001	
Filtration Efficiencies, %		
Particulates	99	
Chemically Reactive Vapors	97	
Flow Rates		
Annulus Filtration System, scfm	14,000	
Recirculated, scfm	11,000	
Exhausted, scfm	3,000	
Containment Release Cleanup System		
Venting, cfm	24,000	
Purging, cfm	17,000	
Atmospheric Dilution Factors, 50% X/Q (sec/m^3)		
Exclusion Area Boundary, 0.42 miles	Applicants' Values	Staff's Values
0-2 Hours	1.01×10^{-3}	1.3×10^{-4}
Low Population Zone, 2.5 miles		
0-2 Hours	1.59×10^{-4}	(1.1×10^{-5})
2-8 Hours	2.30×10^{-5}	1.1×10^{-5}
8-24 Hours	3.58×10^{-6}	1.0×10^{-5}
1-4 Days	2.29×10^{-6}	8.0×10^{-6}
4-30 Days	2.60×10^{-6}	5.7×10^{-6}

a. From References 1 and 5.

Table A.5-2 Radiological Consequences

		Doses in rem[a]	
	10 CFR 100 Dose Guideline[b]	Staff's SSST Doses[e]	Applicants Base Case[c]
2 Hour Exclusion Area Boundary			
Bone Surface	300	31[d]	0.19
Red Marrow	—	24[d]	0.040
Bone (total)	—	10	—
Lung	75	0.4	0.032
Liver	—	1	0.060
Thyroid	300	12	0.020
Whole Body	25	0.6	0.82
30 Day Low Population Zone Boundary			
Bone Surface	300	27[d]	0.95
Red Marrow	—	2[d]	0.19
Bone (total)	—	9	—
Lung	75	0.4	1.6
Liver	—	1	0.36
Thyroid	300	7	85
Whole Body	25	0.3	2.1

Notes:

a. Bone surface and marrow doses calculated with dose conversion factors from NUREG/CR-0150 (Reference 6), all others from NUREG-0172 (Reference 7).

b. For comparison purposes. These are the 10 CFR Part 100 dose guidelines as supplemented for CRBRP. As specified in "Site Suitability Report in the Matter of Clinch River Breeder Reactor Plant," NUREG-0786, June 1982 (Reference 8), there is an additional guideline of 34 rem whole body mortality risk equivalent. The requirements of 10 CFR Part 100 dealing with these dose guidelines apply only to accidents within the design basis and not to TMBDB cases.

c. Reference 5.

d. Reference 9.

e. Reference 8.

A.5.6 Sensitivity to Time Venting Initiated

Due to the uncertainties associated with the potential rate of sodium-concrete reaction, the applicants have assessed several cases involving different rates. Bounding these is the Margin Assessment Case (Reference 10), a case based on the venting initiation time 10 hours after the beginning of the CDA and used sodium-concrete reaction rates beyond any experimentally observed rates. The time for venting initiation of 10 hours is predicated on the basis of allowing 10 hours for the decision to recommend protective actions offsite and for implementation of those protective actions.

For the Margin Assessment Case, in order to achieve projected conditions within the containment which could call for venting initiation at 10 hours, the applicants assumed a very conservative sodium-concrete reaction rate of 7 inches of concrete per hour for 3 hours, followed by 1 inch per hour until sodium boildry.

Table A.5-3 presents the applicants' calculated results, comparing the Margin Assessment Case to their base case; the applicants' base case assumes one-half inch per hour for 4 hours. The Table shows that the containment conditions do not differ greatly from the applicants' base case. The applicants assert feasible modifications can increase the design margins so that the plant design will accommodate the Margin Assessment Case. Table A.5-4 presents calculated radiological consequences of these two cases. The staff's evaluation of these results and those for intermediate reaction rates and venting times is that these results are reasonable and that the radiological consequences are relatively insensitive to venting initiation times between 10 and 36 hours.

A.5.7 Sensitivity to Initial Release

The applicants' base case includes an initial release to containment at the beginning of the CDA of 1000 lb sodium containing 100 ppb plutonium, plus 0.026% of the reactor inventory of fuel radionuclides, solid fission products, and halogens. The applicants have also calculated radiological consequences for a selection of such initial releases, ranging from none up to 50% of fuel and fission products, and including sodium releases ranging up to 7000 lb. In each case, the containment system is considered to perform as designed. Table A.5-5 presents some of the bone surface doses calculated by the applicant (Reference 5); the bone surface dose is the only organ dose that exceeds the 10 CFR Part 100 dose guidelines, and then only in the 50% release case, a physically unrealistic case. The narrow range of the bone surface doses is the result of the small leak rate of the containment combined with the rapidity with which the source, suspended in the containment atmosphere, is depleted by fallout and plateout. The initial release to containment is almost totally depleted before the time of venting, due to the rapid depletion rate that results when large masses of aerosols are injected into the containment atmosphere.

TABLE A.5-3

SUMMARY OF APPLICANTS' MARGIN ASSESSMENT CASE RESULTS

	Base Case	Margin Assessment
Initial Hydrogen Ignition		
Time (hrs.)	10.0	1.4
RCB Atmosphere Temperature (^{o}F) (before/after)	120/845	145/570
RCB Pressure (psig) (before/after)	2.2/22	2.4/14
Hydrogen Concentration (Vol. %) (before/after)	4.5/0.0	2.5/0.0
Initiation of RCB Venting		
Time (hrs.)	36	10
RCB Atmosphere Temperature (^{o}F)	617	710
RCB Steel Shell Temperature (^{o}F)	400	390
RCB Pressure (psig)	13	19
RCB Hydrogen Concentration (%)	0.0	2.6
RCB Oxygen Concentration (%)	8.4	7.4
Maximum Conditions During Venting		
Maximum Venting Rate (CFM)	24,000	28,000
Purge Rate Assumed (SCFM)	8000	8000
Peak Hydrogen Concentration (Vol. %)/Time (hr.)	4.0/40	8.7/14
RCB Atmosphere Temperature (^{o}F)/Time (hr.)	915/40	1020/15
Aerosol Comparisons		
Maximum Rate to the RCB Cleanup System (lb/hr)	4400	5100
Total Aerosols to the RCB Cleanup System to Boildry (lb)	260,000	170,000

TABLE A.5-4

COMPARISON OF APPLICANTS' RADIOLOGICAL CONSEQUENCES, MARGIN ASSESSMENT CASE

	2 Hour Exclusion Area Boundary Doses (rem)[a]	
Organ	36 Hour Vent Base Case	10 Hour Vent Margin Assessment
Bone[c]	0.028	0.44
Lung	0.0055	0.082
Thyroid	0.0096	0.023
Whole Body	0.16	1.9

	30 Day Low Population Zone Doses (rem)	
Organ	36 Hour Vent Base Case	10 Hour Vent Margin Assessment
Bone[c]	55[b]	55[b]
Lung	4.0	3.9
Thyroid	99	95
Whole Body	3.5	13

a. Doses calculated with dose conversion factors from NUREG-0172 (Reference 7), and the radionuclide inventory of the homogeneous core design.

b. Results include earlier, extremely conservative estimate of 13 Kg Pu (Reference 11) sparged to containment atmosphere, vs 26 g (Reference 5).

c. Bone surface doses would be about 3X as large, if calculated with dose conversion factors from NUREG/CR-0150 (Reference 6).

TABLE A.5-5

BONE SURFACE DOSE CHANGES WITH INITIAL RELEASE SIZE

	Bone Surface Doses (rem)	
Initial Release Size[a]	Exclusion Area Boundary	Low Population Zone
Zero	0.027	0.92
Base Case	0.19	0.95
1% Fuel	6.5	2.5
5% Fuel	32	8.2
10% Fuel	64	15
50% Fuel	320	70

a. Includes 100% noble gases and Cs, Rb in all cases, 1000 lb sodium in all but the zero case, and 100% halogens and volatile fission products in the 1%, 5%, 10% and 50% cases.

The thyroid doses, due primarily to iodine releases, are also affected by the aerosol depletion in containment. The thyroid doses of the applicants' base case, 0.02 rem at the exclusion area boundary and 85 rem at the low population zone boundary, change to about 23 rem and 8 rem respectively when the release mode for halogens is changed from 100% boiled up with the sodium (as in the base case) to 100% in the initial release.

It is the staff's assessment that, given a CDA, initial releases will be either zero or, as in applicants' base case, small. Further, the radiological consequences will not be greatly changed by initial releases within a reasonably expected range.

A.5.8 Sensitivity to Alternative Scenarios

Several alternative scenarios have been considered. In the principal alternative, the applicants have presented information indicating that after the core debris penetrates the reactor vessel and reactor guard vessel, and has dropped into the reactor cavity with the sodium, it will form a uniformly-distributed bed on the reactor cavity floor liner (Reference 1). The uniform bed of particles of fuel, blanket and structural materials would be stable for the interim because the sodium would remove sufficient heat to keep the bed from changing character or penetrating the steel floor liner. The staff considers that in this scenario, initial releases, if any, would be the same as in the base case scenario, and releases during the sodium boilup phase would also be substantially the same. Exceptions are that the time to boiling and the time to boildry would be lengthened due to the absence of heat generation by sodium-concrete reactions, and the time to venting would be lengthened, hydrogen production would be less. Also, there would be no sodium-concrete reaction products and little NaOH and Na_2O with the debris bed at boildry. Once failure of the liner occurs and interactions with the concrete begin, the circumstances would be similar to the base case, with the exception of the absence of sodium-concrete reaction products. If the liner failed at some intermediate time, e.g., after 50 hours, the ensuing sequence would also be similar to the base case.

In another scenario, the sodium could be drained into the reactor cavity in advance of core disruption, leaving the core to overheat and enter the sodium pool later. Halogens and other volatiles released in the reactor vessel could either be dissolved in the sodium or be trapped in the PHTS. Once the core debris enters the sodium, this scenario would be similar to the base case.

Other alternative sequences are possible, but it is the staff's assessment that it is highly unlikely that there would occur circumstances such that the radiological consequences of the base case would be greatly exceeded.

A.5.9 Uncertainties

Although a great amount of experimental data relevant to the TMBDB scenario has been accumulated over the years, the staff believes there still remain substantial uncertainties in the estimation of the radiological consequences. The estimate of plutonium releases is judged to contribute the most to the uncertainty attributable to release quantities, possibly as much as a factor of 20. However, in Reference 1, the applicants have evaluated CRBRP's beyond-the-design-basis margins for a number of variations of their basic TMBDB scenario and have shown that the calculated range of radiation doses are limited. It is the staff's assessment that the applicants have adequately shown that the increase in dose, due to the range variations of circumstances, is small.

In a number of instances, the modeling for calculation of consequences includes conservative assumptions and use of values conservatively selected from the maximum of experimental observations; the result is that, even though these are called "realistic" calculations, there is a basis for belief that they tend to overestimate the consequences. An example of such conservatisms is considering immediate 100% release to containment of noble gases and Cs and Rb rather than considering them trapped in the PHTS above the sodium pool, and rather than considering that the Cs and Rb would be retained in the sodium and in the vent system for some hours before being vented into containment.

Of course, the dose estimates also include all the other uncertainties normally found in such estimates, e.g., in meteorological dispersion, in filter efficiencies, etc.

A.5.10 Conclusions

The staff has done scoping analysis for radiological consequences and while the staff conclusions are not identical to the applicant's, but show that the dose guidelines of 10 CFR 100 as augmented for CRBR, may be exceeded, the staff has determined that sufficient improvements in the containment cleanup system filtration efficiency are easily achievable and therefore this is acceptable for the CP stage. However, conclusions presented here are based on projected performance of the proposed design of the TMBDB systems. As stated in section A.4.10, satisfactory equipment qualification and demonstration of performance of the TMBDB systems will be required at the OL stage. The staff advises that in the meantime, before installation of the TMBDB system are undertaken, that the applicants should provide for review of the parameters, and ranges of values, on which testing for operational qualification of the TMBDB systems will be performed.

References

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7. G. R. Hoenes and J. K. Soldat, "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake," NUREG-0172, November 1977.

8. "Site Suitability Report in the Matter of Clinch River Breeder Reactor Plant," NUREG-0786, June 1982.

9. "NRC Staff's Supplemental Answers to Natural Resources Defense Council, Inc. and the Sierra Club Twenty-Sixth Set of Interrogatories to Staff," August 5, 1982.

10. Letter: HQ:S:82:112 John Longenecker to Paul Check, *Thermal Margin Beyond Design Base (TMBDB) Margins Assessment Document*, dated October 20, 1982.

11. CRBRP-3, *Hypothetical Core Disruptive Accident Considerations in CRBRP, Volume 2, Assessment of Thermal Margin Beyond the Design Base*, Revision 0, March 1980.

A.6 Summary Conclusions

The conclusions summarized here are contingent on:

(1) Satisfactory completion of the required fuel pin design modification studies and testing directed at mitigation of the effects of plenum fission gas during CDA progressions (p. A.2-6). The applicants have agreed to provide this modification.

(2) Satisfactory completion of the commitments associated with modifications of the rotating plugs in the reactor closure head and related efforts including evaluation of the SRI tests and updating of appropriate documentation (p. A.3.5 and p. A.3-14). The applicants have agreed to modify the plugs.

(3) Satisfactory resolution of the cell liner criteria based on forthcoming analyses and testing or the adoption of satisfactory fall-back positions as discussed in Section A.4.10. The applicants have agreed to this resolution.

(4) A scoping equipment qualification program has been developed by the applicants and reviewed by the staff. The staff finds this acceptable for the CP. However, the staff requires confirmation of the specific values of the parameters (temperature, pressure, etc.) during the OL review.

The staff's conclusions are developed as a result of the evaluation of the CDAs in terms of the general criteria as discussed in Section A.1.3. Based on the independent evaluation of core disruptive accident energetics described in Section A.2 and on the mechanical capability of the reactor vessel discussed in Section A.3 we conclude that, assuming a CDA occurs, containment failure from spray fires or missiles is not of concern, and further that no significant leakage of vaporized fuel will occur from the RCB. This means that the radiological consequences of CDAs are principally determined by the degree to which TMBDB features prevent containment failure from thermal phenomena such as aerosol generation, sodium fires and hydrogen burning. As discussed in Section A.4, containment failure from such phenomena is unlikely because the containment vent-purge system can relieve internal pressures and effectively control hydrogen in the RCB. The doses from venting, to which an individual would be exposed, if he remained 30 days at the plant's low population zone boundary, can be brought below the CRBR version of the 10 CFR 100 dose guidelines (realistically calculated). This, taken in conjunction with the low probability of such events, leads us to conclude that the risks from such events at CRBR will be very small, and not significantly different from the risks from typical LWRs.

APPENDIX B

UNRESOLVED SAFETY ISSUES

NUREG-0606, "Unresolved Safety Issues Summary," lists several safety issues which are undergoing NRC study before the staff can make judgments as to whether existing requirements should be modified. These issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. The staff has screened the unresolved safety issues relative to their applicability to CRBR and asked the applicants to respond as to how they plan to treat each applicable issue during the licensing activity. The applicants' response and the staff's assessment of each issue are detailed in this section.

B.1 WATERHAMMER (Unresolved Safety Issue (USI) A-1)

Applicants' Assessment of Applicability to CRBRP

Waterhammer and its equivalent, sodium hammer, are applicable to the CRBRP. Waterhammer events introduce a range of hydraulic loads, or pressure pulses, into a fluid system and are the result of rapid condensation of steam pockets, steam-driven slugs of water, pump startup into voided lines, and improper (or sudden) valve closures. Where waterhammer has occurred in water lines, the principal damage in most instances has been to pipe hangers and snubbers. Occasionally pipe welds have experienced small cracks. In none of the reported LWR waterhammer incidents has there been a release of radioactive material or a disabling of safety systems.

Applicants' Suggested Resolution for CRBRP

This issue has been technically resolved for the CRBRP. The water and steam systems of the CRBRP (i.e., the steam generator auxiliary heat removal system (SGAHRS) are described in PSAR Sections 5.5 and 5.6.1, respectively. Design resolution of waterhammer will be accomplished by including fill and vent holes in the auxiliary feedwater sparger in the steam drum to preclude waterhammer effects resulting from steam-driven slugs of SGAHRS water, and by including hydraulic dampers in the actuators of the water and steam isolation valves to preclude waterhammer effects resulting from the overly rapid closing of a valve. The vent holes are described in revised PSAR Section 5.5.2.3, and the hydraulic dampers are discussed in Section 5.5.3.1.5.2.

Protection against the effects of pipe breaks and waterhammer loads are incorporated in ASME design codes that require consideration of impact loads and dynamic loads in the structural design. The ASME codes are applied to the sodium systems of CRBRP, that is, the primary heat transport system, the intermediate heat transport system (including the steam generator), and the sodium-water reaction pressure relief system, as well as to the water-steam systems.

The occurrence of sonic pulses, similar to those produced in waterhammer incidents, has been considered in the design of the intermediate heat transport system (IHTS), described in PSAR Section 5.4. Sonic pulses may occur as a result of a large sodium-water reaction caused by a postulated steam generator tube rupture. In addition, the effects of accelerated sodium slug flows in the component and piping design has been considered in the design of the sodium-water reaction pressure relief subsystem, described in PSAR Sections 5.5, 7.5.6, and 15.3.3.3.

The absence of sodium isolation valves in the IHTS precludes high decelerations of sodium that could cause waterhammer effects in sodium. The high normal boiling point and high heat of vaporization of sodium make vapor-driven sonic pulses extremely unlikely.

NRC's Position

The staff concurs that the applicants are addressing the waterhammer/sodium-hammer phenomena analytically and in the proper manner relative to the CRBRP application. However, the applicants must verify that unacceptable feedwater hammer will not occur by performing acceptability tests, approved by NRC, as described in BTP ASB 10-2. (Standard Review Plan, Section 10.4.7-7)

B.2 STEAM GENERATOR TUBE INTEGRITY (USIs A-3, A-4, and A-5)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. The design designates steam generators in each of the three heat transport system loops for the transfer of heat from the secondary sodium loop to the water systems. The issue concerns the capability of steam generator tubes to maintain their integrity under normal operation and accident conditions, should mechanisms exist that could result in tube degradation.

Applicants' Suggested Resolution for CRBRP

This issue has been technically resolved for the CRBRP.

The CRBRP steam generator design has minimized the potential for corrosion/erosion degradation common to steam generators in pressurized-water reactors (PWRs). The tubes in the CRBRP steam generator will be exposed to the water environment only on their inside surface. The water side will consist of smooth wall tubes terminated in spherical plena. This will greatly reduce the potential for tube degradation by corrosion-induced wastage, cracking, and denting. Preferential corrosion product formation or deposition will be minimized because there will be no restrictions, crevices, water levels, or structure-related concentration sites. Water-side chemistry will be maintained by state-of-the-art, all-volatile chemistry control, which has been modified from PWR practice and which will incorporate fossil plant experience with 2¼% Cr-1% Mo tube material. Full flow demineralizers, a 2:1 full-power recirculation ratio (for each two parts of water flowing into the steam generator, one part will be recirculated and one part will be fresh feed), and 10% blowdown will contribute to minimizing the potential for water-side corrosion-related problems.

Steam generator tube integrity has been properly addressed in the CRBRP design by specifying that a total of 29% of the 0.109-in. tube wall thickness (Section 5.5.2.3.4 of the PSAR) will be allocated for corrosion, cleaning, and wear allowances. The reduced thickness will be used for all stress and strain calculations, and the full thickness will be used for weight and seismic calculations. In addition, allowances will be provided to compensate for material strength degradation by postweld heat treatment, thermal aging, and decarburization. In spite of these reductions in thickness and material conservatively based on the end-of-life condition, the tube will have a 38% margin over the ASME Code, Class 1, criteria for pressure retention.

Erosion of tubes as a result of tube vibration is being addressed in three ways, as discussed in PSAR Section 5.5.

(1) The design and material selection of the shell (sodium-containing) side of the steam generator (SG) will provide for acceptable accommodation of tube vibrations; all known flow-induced vibration mechanisms have been evaluated. Tube-to-spacer plate gaps will be consistent with guidelines used throughout the heat exchanger industry. Tube-spacer plate material (Inconel 718) has been chosen, since it has a low coefficient of friction when coupled with the tube material (2¼% Cr-1% Mo).

(2) To confirm that all flow-induced vibration mechanisms are considered, a flow-induced vibration program has been implemented using both a full-scale model closely representing the prototype unit and a 0.42 scale model. The scale model flow-induced vibration tests will ensure that mechanisms of unexpected origin in the plant unit design do not exist.

(3) The applicants have developed an ultrasonic tube inspection technique that can detect the tube wear well before the tube wall is thinned beyond that spcified for the design. This technique is discussed in PSAR Appendix G.

NRC's Position

The staff agrees that the actions proposed by the applicants will minimize the probability of steam generator tube degradation resulting from wastage and flow-induced vibration. The inservice inspection technique (volumetric eddy current and continuous monitoring) and the intervals for volumetric examination of the tubing suggested in Section 5.11 are adequate to ensure that no major undetected steam generator tube degradation will take place during the life of the plant.

The scale model flow test results will be reviewed during the operating license review process to ensure that the described design has been successful in meeting flow-induced vibration requirements.

All-volatile chemistry control techniques proposed for the CRBR water treatment system have proven to be an effective method for reducing wastage and stress-corrosion cracking in LWRs.

Finally, the staff recognizes the less severe safety role a failure of a CRBR steam generator tube imposes relative to that of an LWR steam generator tube failure. However, the same ASME Code design rules are being imposed on the CRBR steam generators. Large and small steam generator tube failures have been addressed in the Chapter 15 review, and the proposed accommodation criteria, including immediate reactor shutdown, have been found acceptable.

B.3 ANTICIPATED TRANSIENTS WITHOUT SCRAM (USI A-9)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. The issue is the potential for a common mode failure to reduce the reliability of protection systems in such a way that the reactor might not shut down as required when an anticipated transient occurs.

Applicants' Suggested Resolution for CRBRP

The applicants' view is that this issue is resolved for CRBRP because CRBRP incorporates into its design two independent shutdown systems, either of which will have the capability, of itself, to terminate reactor power transients and to effect rapid shutdown of the reactor automatically, and further, because strict attention to the diversity and independence of the two shutdown systems will reduce the likelihood of their simultaneous failure to such a low level that additional design features to improve reliability of shutdown will not be necessary.

NRC's Position

The Commission has initiated a rulemaking on this anticipated transient without scram (ATWS) issue. The ATWS issue is discussed for light-water reactors (LWRs) in "Anticipated Transients Without Scram for Light Water Reactors," NUREG-0460, Volume 4, March 1980, where specific design features and analyses are prescribed for LWRs. These prescriptions are, however, tailored to each type of LWR and thus generally not appropriate for CRBRP.

The staff's conclusions on this issue for CRBRP are based on its review of the redundancy, independence, and diversity embodied in the proposed designs for CRBRP's two shutdown systems (as discussed in Sections 4 and 7 of this SER), on the acceptability of the applicants' Reliability Assurance Program (as discussed in Appendix C of this SER), and on the assessment that even if an ATWS event should occur at CRBRP and lead to core disruption, the risks would be acceptably low (Appendix A of this SER). The staff concludes that the ATWS issue will be resolved for CRBR upon implementation of the design as modified by the findings of this SER. However, additional insight in this area may be gained from continued evaluation of operating experience at LWRs and other nuclear reactors. Therefore, the staff will expect the applicants to address in the FSAR those measures taken in response to lessons learned from reactor operating experience during the period from the issuance of the construction permit to the issuance of the FSAR, and specifically the implications of the ATWS at the Salem reactor in February 1983.*

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*It should be noted that at the time this SER was issued, the staff had determined that the ATWS event at Salem was caused by failure of the scram breakers to open when required. The scram breakers to be used on CRBR are of different design than those used at Salem and will be identical to those used on FFTF and in the Naval Reactors Program. These breakers have undergone extensive testing and have operated successfully. Therefore, they are not expected to lead to an ATWS event at CRBR. Nevertheless, they will be included in the applicant's Reliability Assurance Program.

B.4 FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (USI A-12)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. It concerns the low fracture toughness and potential lamellar tearing in materials used for heat transport system component supports.

Applicants' Suggested Resolution for CRBRP

The design of that portion of the CRBRP steam generator supports that is in accordance with the ASME Code requires that impact testing (Charpy V-notch) of all materials of construction be performed according to Paragraph NR-2311 of ASME Code, Section III. The acceptance standards of Paragraph NR-2330 must be met at 50°F, maximum. Since the lowest operating temperature for the steam generator support will be 125°F, there will be adequate margin for protection against nonductile failures. In addition to the materials fracture toughness requirements, postulated defects will be evaluated using the procedure in Appendix G of ASME Code, Section III, for all applicable conditions plus shipping, lifting, and installation. Therefore, the concern relating to fracture toughness of steam generator supports has been properly and adequately addressed in the CRBRP design.

The building structural steel that supports steam generators will be designed in accordance with the requirements of the American Institute of Steel Construction Code using American Society for Testing Materials (ASTM) A-36 steel and SA-540 bolting material. Sandia Laboratories report SAND78-2348 (Appendix C to NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports - Resolution of Generic Technical Activity A-12 for Comment") classifies USI A-36 as falling within material group II, that is, intermediate susceptibility to brittle fracture, and identifies that group II materials have been judged adequate. SAND78-2348 classifies SA-540 bolting material as falling within material group III, which has also been judged adequate.

The supports for reactor coolant pumps and intermediate heat exchangers will be type 304 stainless steel, connected to ASTM A-36 embedded plate with SA-540 bolting material.

CRBRP design criteria applied to the reactor vessel and steam generator supports will preclude conditions leading to lamellar tearing (e.g., material selection, welded joint orientation, and fabrication sequence).

NRC's Position

The staff agrees. The heat exchanger supports, steam generator supports, and the primary and secondary pump supports will be in cells or local regions where temperatures will not fall below 125°F before reactor operations. This temperature limit could conceivably be a technical specification even though the CRBR primary coolant contains relatively low stored energy.

B.5 SYSTEMS INTERACTION IN NUCLEAR POWER PLANTS (USI A-17)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. It concerns the sufficiency of integration of divided responsibilities for design, analysis, and installation of systems among teams of engineers with functional specialities, such as civil, electrical, mechanical, and nuclear, to ensure that adverse operational interactions between plant systems will be minimized.

Applicants' Suggested Resolution for CRBRP

The applicants have implemented a combination of programs and activities directed toward ensuring an integrated design that has considered the potential for and will provide protection against adverse operational interactions between plant systems.

These include the CRBRP quality assurance program, a comprehensive design control program, specialized design reviews, and reliability and probabilistic risk assessment programs.

The plant has been designed to requirements that support a defense-in-depth philosophy. These requirements ensure physical separation and independence of redundant safety systems, diversity of safety features, and protection against hazards such as sodium leaks, sodium-water reactions, line ruptures, missiles, tornadoes, floods, seismic events, fires, human errors, and acts of sabotage. These requirements are described in PSAR Section 1.1.2 and Chapter 3.

To ensure that these requirements will be properly implemented, the CRBR Quality Assurance Program addresses the design process. This program requires that during the design process emphasis be placed on the control of interfaces between systems. This interfacing is described in PSAR Section 17A.3.1. Independent design reviews, with interdisciplinary memberships and objectives, are required at various stages of the design process. Requirements for these independent design reviews are described in PSAR Chapter 17, Appendix G.

Extensive key systems reviews (KSRs) cutting across system boundaries have been conducted. Multidisciplined groups of individuals conducted these reviews with objectives that included assessments of plant and operator responses during offnormal and accident events. Interactions between systems were explicitly considered as part of these reviews. Evaluations of the results of these reviews addressed the potential for adverse systems interactions, including consideration of human, spatial, and functional coupling effects. A summary report of these KSRs was provided in response to an NRC question.

The CRBRP safety-related reliability program is described in PSAR Appendix C. The results obtained in this program provide additional confidence that the requirements for the systems designs will minimize the potential for adverse operational interactions.

In response to an NRC question, the applicants developed the CRBRP Probabilistic Risk Assessment (PRA) Program Plan, which will include tasks to demonstrate that the risks at CRBRP will be acceptably low. The planned methodology will use event trees and fault trees to identify the component failures combinations that could result in a loss of safety function. The PRA activities will specifically evaluate potential adverse interactions between plant systems.

NRC's Position

The staff's systems interaction program was initiated in May 1978 with the definition of USI A-17 ("Systems Interaction in Nuclear Power Plants") and was intensified by Task Action Plan (NUREG-0660) Item II.C.3 ("Systems Interaction"). The concern arises because the design, analysis, and installation of systems are frequently the responsibility of teams of engineers with functional specialties such as civil, electrical, mechanical, or nuclear. Experience at operating plants has led to questions of whether the work of these functional specialties is sufficiently integrated to enable them to minimize adverse interactions among systems. Some adverse events that occurred in the past might have been prevented if the teams had ensured the necessary independence of safety systems under all conditions of operation.

The staff's current procedures assign primary responsibility for review of various technical areas to specific organizational units and secondary responsibility to other units where there is a functional interface. Designers follow somewhat similar procedures and provide the analyses of systems and interface reviews. Under Task A-17, methods are being developed that could identify adverse systems interactions that were not considered by current review procedures. The first phase of this study began in May 1978 and was completed in February 1980 by Sandia Laboratories under contract to the staff.

The Phase I investigation was structured to identify areas where interactions are possible between systems and the potential for negating or seriously degrading the performance of safety functions exists. The study concentrated on commonly caused failures among systems that would violate a safety function. The investigation was to then identify where NRC review procedures may not have properly accounted for these interactions.

The Sandia Laboratories used fault-tree analysis on the selected LWR plant design to identify component failure combinations (cut-sets) that could result in loss of a safety function. The cut-sets were further reduced by incorporating six linking systems' failures into the analysis. The results of the Sandia effort indicated a few potentially adverse systems interactions within the limited scope of the study. The staff reviewed the interactions for safety significance and generic implications.

NUREG-0660, Section II.C.3, provides for a systems interaction follow-on study. Since April 1980, the staff has intensified the effort both by broadening the study of methods to identify potential systems interactions and by preparing guidance for audit reviews of selected plants for systems interactions. The staff's recent experience provides a basis from which it is developing a more efficient review process for potential systems interactions. The process will provide for a resolution of USI A17, assimilate operating reactor experience, and rank identified systems interactions by their relative importance to safety.

It is expected that the development of systematic ways to identify, rank, and evaluate systems interactions will go further to reduce the likelihood of intersystem failures resulting in the loss of plant safety functions. A comprehensive program is expected to employ analytical methods, visual inspections, experience feedback, and experiments for dependencies. The light-water-reactor industry's current experience with systems interaction reviews is fragmented. Experience like that gained by the Phase I study is an essential ingredient to the staff's considerations of a comprehensive systems interaction program.

CRBRP has been evaluated against current licensing requirements that are founded on the principle of defense in depth. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems and protection against hazards such as high-energy-line ruptures, missiles, high winds, flooding, seismic events, fires, human factors, and sabotage. These design provisions are subject to review against the Standard Review Plan (NUREG-0800) which requires interdisciplinary review of safety-grade equipment and addresses different types of potential systems interactions. Also, the quality assurance program that is followed during the design, construction, and operational phases for each plant contributes to the prevention of introducing the potential for systems interactions by error. Thus, the current licensing requirements and procedures provide an adequate degrees of plant safety.

In addition the applicants have described a program, Key Systems Reviews, that separately evaluates all structures, systems, and components important to safety for the three categories of adverse systems interactions, that is, spatially coupled, functionally coupled, and humanly coupled. Also, the applicants have committed to a reliability program that will include failure modes and effects analyses, system level fault trees or their equivalent, accident event sequences, and systematic searches (such as plant walk-throughs) for adverse systems interactions. This program is discussed in more detail in Appendix C of this SER.

Although the staff at this time cannot be sure that the applicants' programs will be equivalent to the methodology eventually arrived at for resolution of USI A-17 for LWRs, the staff believes it is acceptable for the CP stage of licensing. The staff expects the applicant to monitor the ongoing generic effort on USI A-17 during the intervening period between CP issuance and OL licensing review and to develop an equivalent program to that adopted for LWRs.

B.6 ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT (USI-A-24)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. CRBRP design will include Class 1E equipment that must be qualified for the environmental conditions in which it may be required to perform.

Applicants' Suggested Resolution for CRBRP

The technical resolution of this issue for LWRs and NRC's position are contained in NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." The issue is resolved for CRBRP through a program for environmentally qualifying safety-related electrical equipment that is consistent with the objectives and requirements contained in NUREG-0588, Revision 1, as applied to CRBRP technology. This program is outlined in response to NRC Question CS270.1 and in PSAR Section 3.11.

NRC's Position

The staff has reviewed PSAR Section 3.11 and supplemental letters from the applicants relative to the proposed CRBRP environmental qualification program and finds it acceptable as detailed in Section 3.11 of this SER. Major highlights of the review are:

(1) The applicants' list of systems and components that are required to perform safety-related functions, as presented in WARD-D-165, Revision 6, was found acceptable.

(2) The temperature, pressure, and humidity conditions, for both inside and outside containment, were properly specified by the applicants.

(3) The applicants' approach to qualifying the equipment for a sodium aerosol environment was found acceptable.

(4) The applicants have committed to follow the recommendations in RG 1.33, Revision 2, to identify and prevent significant age-related degradation of electrical equipment.

(5) The applicants have defined the design methodology used to calculate the radioactive environment based on three different source terms (site suitability source term, sodium storage tank failure source term, and cover gas release source term). The staff has reviewed the proposed methodology and finds it acceptable for use in the qualification of electrical equipment.

(6) The applicants have committed to meet the documentation requirements identified in IEEE Std. 323, 1974. The staff finds this plan for documentation acceptable and in accordance with 10 CFR 50.49.

B.7 RESIDUAL HEAT REMOVAL REQUIREMENTS (USI A-31)

Applicants' Assessment of Applicability to CRBRP

This issue is not applicable to CRBRP. It concerns the capability of PWRs to go from hot to cold shutdown without the availability of offsite power.

A safe shutdown condition equivalent to a PWR cold shutdown condition will be achieved in CRBRP when the plant will be brought down from operating temperature to 600°F using the plant shutdown heat removal systems. At the 600°F temperature the plant will be in a safe and stable state, and long-term cooling will be in effect. There is no subsequent requirement to proceed to another mode or state to effect long-term shutdown.

The normal decay heat removal path will be through the use of the main condenser and feedwater train. However, since the main condenser and feedwater train will not be available on loss of offsite power, the steam generator auxiliary heat removal system, which is a safety-related system, will be provided for shutdown heat removal and long-term decay heat removal, and will not depend on the availability of offsite power. The initial heat load will be dissipated through the use of power relief valves in the steam generator loops.

NRC's Position

The NRC concurs with the applicants' assessment.

B.8 CONROL OF HEAVY LOADS NEAR SPENT FUEL (USI A-36)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. Although the design of CRBRP does not designate spent fuel pools, this concern is applicable to the control of heavy loads over the ex-vessel storage tank closure head and striker plate, and over the fuel-handling cell.

Applicants' Suggested Resolution for CRBRP

The technical resolution of this issue and NRC's position are contained in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

The issue is resolved for CRBRP by the application of a single-failure-proof crane (in accordance with NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants") in both the reactor service building and reactor containment building for all critical lifts. The project application of NUREG-0612 is presented in response to NRC Question CS410.3.

NRC's Position

The applicants have applied the two proper criteria to resolve this issue for CRBRP, namely, NUREG-0554 and NUREG-0612. These two criteria will be the basis during the OL licensing review.

B.9 SEISMIC DESIGN CRITERIA (USI A-40)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. It concerns the conservatism of certain aspects of the overall seismic design criteria.

Applicants' Suggested Resolution for CRBRP

This issue has been technially resolved for CRBRP. The seismic design bases and the seismic design of CRBRP conform to the current NRC criteria. CRBRP seismic design criteria are described in PSAR Section 3.7. NRC has not established any other bases that would render conformance to the current criteria inadequate.

NRC's Position

Pending the formal supportive documentation relating to the adequacy of assumptions used in the rock structure interaction model, the NRC concurs that the seismic design criteria and procedures used by the applicants for CRBRP are adequate, namely:

(1) assigning two levels of earthquake (SSE at 0.25 g and OBE at 0.125 g) which reflects appropriate consideration for the most severe earthquake recorded for the site with an appropriate margin

(2) appropriate combinations of the effects on normal and accident conditions with the effect of the natural phenomena

(3) appropriate consideration of the safety functions to be performed--the use of a suitable dynamic analysis or a suitable qualification test to demonstrate that structures, systems, and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate consideration.

B.10 STATION BLACKOUT (USI A-44)

Applicants' Assessment of Applicability to CRBRP

A loss of offsite ac power involves a loss of both the preferred and backup sources of offsite power.

If offsite ac power is lost, three diesel generators and their associated distribution systems will be designed to deliver emergency power to safety-related equipment.

If both offsite and onsite ac power are lost, CRBRP will be designed to remove reactor-generated decay heat on natural circulation with the heat sink provided by the steam generator auxiliary heat removal system. This capability will ensure that adequate cooling can be maintained for at least 2 hours, which will allow time for restoration of ac power from either offsite or onsite sources.

The decay heat generated in the spent fuel in the ex-vessel storage tank (EVST) will be capable of being removed by natural circulation. This will be provided by the third EVST cooling loop which will be designed to remove all decay heat produced in the EVST during natural circulation.

The ex-vessel transfer machine will be designed to ensure that cladding temperature will be maintained within limits by a natural convection cooling system. This will ensure cooling of a fuel assembly in transit between the reactor and EVST during a station blackout.

Two-hour station blackout during the handling of a bare fuel assembly during normal fuel-handling cell (FHC) operations would result in release of fission products to the environment. The potential radiation doses at the site boundary resulting from such a release have been calculated to be below established limits.

NRC's Position

The NRC concurs with this assessment provided the applicants demonstrate that adequate natural circulation capabilities exist in the main heat transport systems and the ex-vessel storage tank natural circulation heat removal loop.

Also, the adequacy of the circuit breaker realignment capability during a station blackout must be demonstrated during the prestartup test program.

On the basis of the staff's confidence that these provisions will be met, the staff concludes that the CRBRP will have the capability to withstand a station blackout comparable to that of a PWR. The final generic resolution of this issue for PWR will probably be determined after the CP license is issued. On the basis of the similarity of the CRBR electric power system and auxiliary feedwater system to those of PWRs, the staff anticipates that the generic PWR station blackout resolution will be generally applicable to CRBR. The applicant should adopt that generic resolution for CRBR or develop an equivalent resolution in time for the OL review.

B.11 SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (USI A-45)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. It concerns the sufficiency of plant capability to remove decay heat. CRBRP must have a highly reliable capability to remove decay heat from the reactor.

Applicants' Suggested Resolution for CRBRP

This issue has been resolved for CRBRP by incorporating into the design, multiple, independent, and highly reliable heat transport paths, any one of which will have sufficient capacity to remove the reactor decay heat by itself. The various heat removal paths and their operating modes embody substantial diversity.

The CRBRP heat transport system (HTS) will use three independent loops, each of which will provide a separate path from the reactor vessel to the ultimate heat sinks. The normal heat removal path includes the main condenser and feedwater train, which is used for normal operation and some shutdown heat removal conditions. However, for each path an alternative safety-related path will be provided through the SGAHRS, which will provide its own heat sinks. Thus, it will not be necessary to rely on the main condenser and feedwater train, since SGAHRS will be available for all anticipated plant events.

The SGAHRS will include the auxiliary feedwater subsystem (AFWS) and protected air-cooled condensers (PACCs), which will serve as alternative heat sinks.

Also, to ensure that the operation of safety system equipment will not be impaired, the single-failure criterion has been applied in the plant design. PSAR Section 7.2.2 discusses plant protection system (PPS)-control system interaction. The CRBRP PPS will be composed of two independent subsystems, either of which will be capable of bringing the plant to a safe shutdown condition.

Further, these two subsystems will employ diverse trip functions for PPS activation. Therefore, for any design-basis transient, there will always be more than one trip function provided by these two totally independent subsystems to activate the PPS and terminate the ensuing transient. Details of this design are described in PSAR Section 7.2 and Table 7.2-2.

A wide range of bounding transients and accidents currently is being analyzed to ensure that the postulated events would be adequately mitigated by the safety systems. In addition, systematic reviews of safety systems have been performed with the goal of ensuring that the control system failures will not defeat safety system action. The worst conditions for each given type of transient are assumed in the accident analyses. This information is provided in PSAR Chapter 15.

The AFWS provides water makeup to the closed loops between the steam generators and the PACCs. The AFWS includes two motor-driven pumps and one steam-turbine-driven pump.

The sodium in the primary and intermediate systems of the HTS loops will always be at temperatures well below the flash point. Thus, in the unlikely event of a sodium pipe leak in any loop, there will not be a loss of heat removal capability resulting from loss of coolant inventory through flashing. Also, degradation of one loop will not affect heat removal capability in either of the other two loops.

Thus, the plant configuration will provide multiple independent paths through the heat transport system, which will contribute to the high reliability of the plant systems for removing reactor decay heat. These capabilities are discussed in PSAR Sections 5.6 and 5.6.1.

In CRBRP there is an additional path for decay heat removal, the direct heat removal service. This system provides a diverse heat removal path to yet another redundant and diverse set of air-cooled heat exchangers. This is described in PSAR Section 5.6.2.

NRC's Position

The acceptance criteria for CRBR shutdown decay heat requirements are more stringent than those for LWRs, namely, the probability of loss of all ultimate heat sink must be sufficiently low so as to allow treatment of the consequences of the event beyond the design basis. These consequences are discussed in Appendix A of the SER. Principal design criterion (PDC) 35 on residual heat removal has been developed for CRBR and contains requirements more conservative than those for LWRs. In addition, the plant will be designed to remove decay heat via natural circulation in the main HTS loops so that even a total loss of offsite and onsite ac power will not prevent decay heat removal. The applicants have also committed to perform a realiability risk assessment of the AFWS during the operating license review.

Furthermore, the direct heat removal service goes a long way toward resolving USI A-45 for CRBRP. However, this issue has not yet been generically resolved for LWRs. The staff will, therefore, reconsider this issue in the OL review. The applicant should consider the applicability of the LWR resolution to USI A-45 to the CRBR and provide justification in the FSAR that a comparable level of safety has been achieved. As noted in Appendix D, the PRA to be performed by the applicants will include consideration of enhancements in the heat removal capability.

B.12 SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (USI A-46)

Applicants' Assessment of Applicability to CRBR

This issue is not applicable to CRBRP. The issue is whether operating plants must be reassessed to ensure the adequacy of their seismic qualification of equipment. Construction of the project has not yet commenced and thus, it is not an operating plant. CRBRP resolution of USI A-40 ensures the adequacy of seismic design criteria applied to it.

NRC's Position

The NRC concurs with the applicants' assessment.

B.13 SAFETY IMPLICATIONS OF CONTROL SYSTEMS (USI A-47)

Applicants' Assessment of Applicability to CRBR

This issue is applicable to CRBRP. CRBRP will depend on the proper functioning of control systems to maintain the plant in a safe condition for all normal operations and accidents. This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration.

Applicants' Suggested Resolution for CRBR

This issue has been technically resolved for CRBRP. Design features ensure that control system failures will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or to maintain the plant in a safe shutdown condition following any anticipated operational occurrence or accident. This will be accomplished by providing independence and physical separation between safety system trains and between safety and nonsafety systems. For the latter, as a minimum, isolation devices will be provided. These devices will preclude the propagation of nonsafety equipment faults to the protection systems.

NRC's Position

A number of concerns have been expressed regarding the adequacy of safety systems in the mitigation of the kinds of control system failures that could actually occur at nuclear plants, as opposed to those analyzed in PSAR Chapter 15 safety analyses. Although the Chapter 15 analyses are based on conservative assumptions regarding failures of single control systems, systematic reviews have not been reported to demonstrate that multiple control system failures beyond the Chapter 15 analyses could not occur because of single events. Among the types of events that could initiate such multiple failures, the most significant are, in the staff's judgment, those resulting from a failure or malfunction of power supplies or sensors common to two or more control systems. To provide assurance that the design-basis event analyses adequately bound multiple control system failures, the applicants were asked to provide the following information:

(1) Identify those control systems whose failure of malfunction could seriously impact plant safety.

(2) Indicate which, if any, of the control systems identified under Item (1) receive power from common power sources. The power sources considered should include all power sources whose failure of malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses resulting from the failure of higher level distribution panels and load centers.

(3) Indicate which, if any, of the control systems identified under Item (1) receive input signals from common sensors, common hydraulic headers, or common impulse lines.

Section 7 of this SER verifies that the design criteria for the control systems will be such that simultaneous malfunctions of control systems that could result from failure of a power source, sensor, or sensor impulse line supplying power or signals to more than one control system will be bounded by the analysis of anticipated operational occurrences in Chapter 15 of the Final Safety Analysis Report.

B.14 HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT (USI A-48)

Applicants' Assessment of Applicability to CRBRP

This issue is not applicable to CRBRP. Design-basis accidents within the CRBRP containment will not lead to the generation of hydrogen. Accordingly, there will be no effect of hydrogen burns that could impact the capability of safety-related equipment to perform its intended safety function. However, accidents beyond the design basis involving hypothetical core disruptive accidents may produce hydrogen as a result of sodium-concrete interactions. The control and burning of the hydrogen from a hypothetical core disruptive accident is addressed in the CRBRP Thermal Margin Beyond Design Basis (TMBDB) (Westinghouse, CRBRP-3, Vol. 2). In the TMBDB scenario, the hydrogen is ignited in the containment atmosphere by sodium burning with the oxygen in containment. CRBRP-3, Volume 2, also demonstrates how containment integrity will be maintained.

NRC's Position

The staff agrees with this position. The control and burning of hydrogen from a core disruptive accident is further discussed in Appendix A of this SER.

13.5 REFERENCES

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American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Appendix G.

Institute of Electrical and Electronics Engineers (IEEE), Std. 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

Project Management Corporation, Clinch River Breeder Reactor Plant, "Preliminary Safety Analyses Report," Docket No. 50-537, through Rev. 68, May 1982.

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---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR Edition, July 1981.

---, RG 1.33, "Quality Assurance Program Requirements (Operation)," Rev. 2, Mar. 1978.

Westinghouse Electric Corporation, "Hypothetical Core Disruptive Accident Considerations in CRBRP," Vol. 2, "Assessment of Thermal Margin Beyond the Design Base," Pittsburgh, PA.

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APPENDIX C

RELIABILITY ASSURANCE PROGRAM

C.1 INTRODUCTION

The applicants have identified several activities which are under way or are to be performed as part of final design to enhance and assess the reliability of certain CRBRP systems considered important to safety and to estimate the risk associated with CRBRP operation.

The objective of these reliability assurance activities, as stated by the applicants in a January 11, 1983 letter (J. R. Longenecker to P. S. Check, #HQ:S:83:184), is to "provide additional assurance that the inherent reliability in the CRBRP design concept is achieved and that the likelihood of exceeding the offsite radiological dose guidelines of 10 CFR 100 is acceptably low. The overall aiming point of these activities is to ensure that the risk to the public from CRBRP is comparable to that of a current LWR." This represents an effort beyond that which is required for the licensing of an LWR and in the staff's judgment is a positive step toward enhancement of CRBRP reliability.

Traditionally the reliability of nuclear power plant safety systems has been enhanced by the application in the design of the principles of:

(1) single-failure criterion
(2) redundancy
(3) diversity
(4) independence

For CRBRP these principles and requirements are specified in 10 CFR 50 and the principal design criteria.

However, application and implementation of 10 CFR 50 and the design criteria has been and continues to be guided by engineering judgment. Additionally, reliance on operator action to terminate or mitigate accident conditions has been minimized in CRBRP to limit, to the extent practical, the potential for operator error.

More recently greater emphasis has been placed on more qualitative and quantitative attention to reliability and risk assessment as an additional tool with which to improve designs and to assess the risk of plant operation. Major developments in this area have been:

(1) Following the accident at Three Mile Island, Unit 2, a requirement for each applicant to perform a probabilistic risk assessment and factor the results of this assessment into the design was issued in NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permit and Manufacturing License," August 1981.

(2) The current Standard Review Plan for the auxiliary feedwater system (Section 10.4.9 of NUREG-0800) has included a requirement to demonstrate that the reliability of the system meets certain quantitative goals.

(3) In January 1983 the NRC issued a policy statement on safety goals which embody the principle of acceptable risk.

In the case of CRBRP, however, limited experience with the design and operation of similar facilities (relative to experience with LWRs) has been accumulated. Thus while there are no regulatory requirements (other than discussed above) to conduct a reliability program within the licensing process, it is the staff's judgment that additional reliability assurance activities should be applied to CRBRP to compensate for the lack of an experience base comparable to that available for an LWR.

The staff has reviewed the various activities outlined by the applicants in the January 11, 1982 letter as contributing to the CRBRP Reliability Assurance Program. As part of this review it was considered necessary to develop criteria addressing what constitutes an acceptable Reliability Assurance Program for CRBRP and against which the applicants' activities could be judged. The criteria developed are considered as requirements for CRBRP and are discussed in detail in Section 2.

The staff believes that the overall objective of the CRBRP Reliability Assurance Program (henceforth called the Program) should be to evaluate and enhance the potential safety-related reliability inherent in the application of 10 CFR 50 and the principal design criteria. This evaluation and enhancement should provide further assurance that the CRBR design will be capable of providing for accident prevention, termination, and mitigation so that the likelihood of a core disruptive accident or of exceeding 10 CFR 100 guidelines is extremely low. In general terms, the activities under the Program should be performed to ensure that the risk to the public from CRBRP is at least no greater than that from a current LWR. It is envisioned that such a Program be conducted to provide reliability feedback information comprehensively during design, fabrication, construction, and operation (including maintenance and surveillance testing) of CRBRP. This feedback should also cause the plant design and operating, surveillance, and maintenance procedures to be changed where considered appropriate.

As stated previously, the staff has developed criteria for the program specifying the appropriate breadth and depth of activities to be performed. The applicants' Program should be structured to meet these criteria. Further, the staff intends to review the applicants' Program through audits and reviews of the process and results to ensure that the overall objective is being met. The staff may perform independent reliability-oriented studies to gain additional confirmatory understanding of the Program. This latter activity is, however, of secondary importance to the staff audits and reviews of the applicants' Program based upon traceable and auditable documentation.

C.2 EVALUATION CRITERIA

In this section, the staff's criteria regarding what constitutes an acceptable program are presented. These criteria define the nature and extent of the Program in broad terms.

The staff in conjunction with its consultant, Science Applications, Inc., has developed an outline of a comprehensive Program with the potential for assessing and impacting CRBRP reliability. The elements of this Program are described below as criteria and are considered by the staff as requirements for a reliability program for CRBRP.

Considering CRBRP characteristics, a unique set of evaluation criteria has been defined for CRBRP. This set contains three elements:

(1) reliability information gathering
(2) feedback to design, operation, surveillance, and maintenance
(3) traceability and auditability

The following activities will generate reliability information:

(1) component level evaluations
(2) system level evaluations
(3) accident sequence level evaluations
(4) common cause failure analyses
(5) system interaction analyses
(6) equipment testing
(7) equipment qualification
(8) failure evaluation

It is the staff's opinion that the existence of these activities in appropriate depth (as discussed later) ensures completeness of a reliability program. The first three activities above--evaluation of the component, system, and overall accident sequence levels--ensures that potential malfunctions at all levels are examined. The common cause failure analyses help provide assurance that built-in design redundancies and mitigative functions are not defeated by common environmental factors, common support systems, or common initiating malfunctions. System interaction analyses are needed to identify system malfunctions, which may be acceptable by themselves, but which could propagate to other systems with unacceptable consequences. Equipment testing is used in a developmental program to verify design. In some cases component failure mode or failure rate data can also be generated. Equipment qualification is a standard requirement for the nuclear industry to ensure performance under required environmental conditions. Failure evaluation is a necessary ingredient to ensure appropriate design feedback and corrective action.

Given these activities of a program, the next essential step is to apply them to the correct components, systems, features, and operational aspects of the plant. In this regard the staff concluded that the Program should be applied to those systems and features whose functions are necessary to prevent core disruptive accidents and to ensure that the likelihood of exceeding 10 CFR 100 dose guidelines is acceptably low. It was judged that if these functions are performed in a reliable manner, then the risk to public health and safety from CRBR operation would be acceptably low and comparable to that from an LWR.

The extent of the reliability activities performed for each system depends upon (1) whether or not the system has active components or features, (2) the accumulated base of directly applicable experience in LWRs or other LMFBRs, (3) whether the system is designed for prevention or mitigation, and (4) the judged importance to protection of public health and safety. In ranking the CRBR systems the reactor shutdown and shutdown heat removal functions are considered of primary importance, and thus those systems utilized in fulfilling these two functions should receive emphasis in the Program. Furthermore, it was concluded that both the front line and support systems necessary to perform each function and feature should be included in the Program. The functions and features judged to fall in this category are:

(1) reactor shutdown
(2) shutdown heat removal
(3) coolant system boundary integrity
(4) features to prevent core flow blockage
(5) features to prevent failed fuel propagation
(6) containment
(7) spent fuel cooling
(8) active features to mitigate core disruptive accidents

In addition to the information-gathering activities mentioned above, the Program includes two additional elements. Feedback to design and operation provides a means of improving the design or operating, surveillance, and maintenance procedures should this be judged appropriate. Traceability and auditability enables determination of the status and appropriateness of the Program. Each of the three elements are discussed in the following sections.

C.2.1 Content of Safety-Related Reliability Information-Gathering Activities

The set of activities for gathering information within the Program are described in more detail below.

(1) Component Level Evaluations

Failure modes and effects analysis (FMEAs) are the basic tools of reliability evaluations applied at the component (pumps, valves, sensors, and so forth) level which form the foundation upon which higher level evaluations are built. Emphasis regarding FMEAs should be placed on components unique (or unique in application) to CRBRP or those components for which a statistically significant reliability data base has not been established. Documented reliability data from previous experience can be used in connection with or instead of FMEAs. Components of this nature which are incorporated in the reactor shutdown and shutdown heat removal functions are of primary importance. Component failures critical to operational success should be systematically identified and evaluated as to both severity and likelihood of occurrence. The principal output of the component-level FMEA is

(a) a comprehensive list of failure modes
(b) a list of potential causes
(c) the effect of the failure and its importance
(d) qualitative or quantitative estimate of the likelihood of occurrence

This output provides the initial assessment of design strengths and weaknesses and can also be used in defining a test program. Component-level FMEAs are also a source of data for determining which system components are susceptible to failure from a common cause.

(2) System Level Evaluations

System level evaluations should be performed to relate failure information to its impact upon system performance. In this context examples of systems are primary shutdown system, secondary shutdown system, auxiliary feedwater system, direct heat removal service (DHRS), and so forth. Fault trees or a combination of system-level FMEAs and logic block diagrams are acceptable methods to produce a display of system components and failure modes. From this evaluation, combinations of component failure states that lead to system failure can be derived. The system reliability can be assessed by qualitatively or quantitatively evaluating the complete (or significantly complete) set of combinations of component failure states (cut sets) that lead to system failure. As part of the process of evaluating systems, known dependencies resulting from common supporting components or subsystems, tests, operations, and maintenance or human interactions should be considered. The output should be an evaluation of all system failure modes and their likelihood of occurrence.

(3) Accident Sequence Level Evaluations

A main interest with regard to accident prevention is the identification of combinations of systems, components, or features which when failed in combination lead to core disruptive accidents or radiation releases in excess of 10 CFR 100 guidelines. For given classes of accident initiation, event trees should be used to display the combinations of systems and features that effect the likelihood and nature of core damage. System-level logic displays of accident sequences, similar to system-level evaluations only on a larger scale, that could lead to core disruption or radiation releases in excess of 10 CFR 100 should then be identified and quantified for each significant event tree branch.

As part of the process of evaluating accident sequences known intersystem dependencies (from supporting and interfacing systems) should be modeled as well as potential adverse effects of other failed systems and operator errors. The output of these evaluations should be a comprehensive list of all accident sequences that could lead to core disruption or radiation releases in excess of 10 CFR 100 guidelines and their likelihood of occurrence.

(4) Common Cause Failure Analyses (CCFAs)

The CCFA is an organized method of analyzing the extent and significance of common failures that may occur which degrade inherent redundancy designed into systems and functions. Common failures can be caused by internal events such as temperature, sodium aerosol concentration and pressure extremes, common locations, design, manufacture, proximity to degrading influences, fires, human errors, and external events such as seismic events, tornados, floods, lightning, chemical effects, radiation, explosions, and aircraft or missile impacts. Typically the analysis proceeds by developing a list of potential causative factors which relate to CRBRP operating conditions. For each causative factor, the system and accident sequence cut sets are checked to determine two factors:

(a) whether or not each component is susceptible to the causative factor

(b) whether or not the causative factor has the opportunity to affect each component

Based on this two-step process, the cut sets can be screened to determine the extent and significance of common cause events.

For specific common cause events, especially external events, a two-phase process is useful in which the first phase bounds the problem and checks its significance. Should this preliminary study indicate that one or more accident sequences may contribute to the risk in a significant manner, a more detailed analysis to ascertain its risk significance more realistically may be warranted. The output from CCFAs should be a system by system comprehensive list of common cause failures and should feed into the system and accident sequence evaluations. Although CCFA is inherently part of component, system, and accident level evaluation, it is highlighted as a separate criterion to emphasize its inclusion in the overall program.

(5) Systems Interaction Analyses

One or more independent components or components of a redundant grouping of components may fail or become more unreliable because of the interaction with other adjacent or nearby system failures. For example, a high-energy-feed-line or steamline break could cause rotating machinery in proximity to fail, or a nonseismically qualified structure adjacent to one train of a seismically qualified system might collapse on the train degrading the system's redundancy.

Although consideration of these dependency conditions is made during design and construction, an organized approach to reviewing the facility for potential systems interaction is warranted. The methods employed may use appropriate lists of component cut sets found in the system and accident sequence level evaluations and the common cause failure derived in the CCFA. In-plant walk-throughs on a compartment basis are needed to check the potential of systems interaction causative factors such as seismic and high-energy-line breaks. A program to accomplish the above should be developed with the output being a a comprehensive list of potential interactions. This information may also be used as input to the system and accident sequence evaluations. A generic investigation of systems interaction is being pursued by the staff for LWRs, and a discussion of the relation of this program to CRBR is provided in Appendix B of this SER.

(6) Equipment Testing

Testing should be performed at the component and subsystem level to explore failure modes, equipment performance, and extended limits of operation in a qualitative reliability sense. Accelerated life testing can be employed to provide early feedback concerning potential failures. A test program should be developed and documented which provides data to demonstrate performance and support reliability assumptions. Emphasis regarding equipment testing should follow the guidelines as described under component-level evaluations. Namely, emphasis should be placed on equipment unique (or unique in application) to CRBRP. Well-documented reliability data can be used in conjunction with or in lieu of equipment testing. Equipment associated with the reactor shutdown system or shutdown heat removal system are of primary importance. In addition, it is expected that the natural circulation and direct heat removal service (DHRS) testing described in Section 4.4 of this SER will also contribute to the overall plant reliability assessment.

(7) Equipment Qualification

Qualification should be conducted to ensure that components and systems can perform their intended safety functions under the anticipated service conditions in which they are required to perform. Section 3 of the SER provides the staff's evaluation of the applicants' Equipment Qualification Program.

(8) Failure Evaluation

Procedures should be established to provide assurance that the cause and mode of each failure during development and operation of CRBRP are identified, that the potential safety and availability implications are evaluated, and that corrective action is taken.

Although quality assurance is not a reliability-gathering activity as defined in this Program, it is an integral part of reliability assurance and should be considered in the applicants' evaluations. Section 17 of the SER discusses the staff's evaluation of this program.

Considering the above activities and the various safety functions and features in CRBRP, a matrix showing specific elements, which in the staff's opinion are required for each function, is shown in Figure C.1.

C.2.2 Feedback to Design, Operation, Surveillance, and Maintenance

The above activities will provide a large and varied amount of reliability-oriented information regarding the safety functions of CRBRP.

The second element of the Program is that this information must be fed back into the design, operation, surveillance, and maintenance documentation in time to support final design as well as remain in place during the lifetime of the facility as a tool with which future changes and the impact of operating experience can be assessed. As a result of this requirement, there will be a number of decisions to be made by the applicants regarding whether changes should be implemented. Thus, there is a need to ensure that the process by which the information is fed back into the design and the criteria or rationale used to control this process are documented and auditable. The key criteria to be used by the applicants to determine whether or not design changes will be implemented need to be documented. Generic criteria applicable project wide are preferred with additional considerations or criteria on a case-by-case basis. For example, specific reliability information may be compared against the principal design criteria, against comparable performance in modern LWRs, or against NRC's safety goals. Further, the reliability information may be compared internally to identify specific large contributors to risk. The probabilistic risk assessment may also be an acceptable tool to help guide judgments regarding design and operational improvements. In any event, the final decisions will be based on engineering judgment using some of the above or other considerations as appropriate. Regardless of the specific considerations utilized, it is important that the applicants provide clear documentation to assist the staff in understanding these considerations and how they are applied in the feedback on the design, operation, surveillance, and maintenance of CRBRP.

C.2.3 Traceability and Auditability

The third element of the Program, traceability and auditability, allows determination of the reliability function performed and verification of the appropriateness of its performance. This element requires clear documentation of all elements of the Program. The staff desires documentation of the Program plan before the operating license review so that the program can be audited before completion of final design.

An example of a required traceable, auditable function would be performance tests of reactor components. Documents must be available indicating the tests performed, the test conditions, and the test results.

C.2.4 Schedule Requirements for Program

The basic design features of the plant which contribute to reliability are those of redundancy, diversity, and independence of safety equipment. These are established by the principal design criteria and construction permit review. The intent of this Program is to enhance and evaluate the reliable performance of the plant safety functions. It is the staff's judgment that implementation of the Program should be on a time scale which allows impact on design, operation, maintenance, and surveillance, if the results indicate change is warranted.

It is also the staff's judgment that this Program should not end at the completion of final design, but rather, should continue throughout the life of the plant as a tool for assessing the impact and acceptability of plant design and procedure changes and the impact of plant operating experience on overall plant risk.

The applicants' schedule for implementing the Program consistent with the above is required.

C.3 APPLICANTS' RELIABILITY ASSURANCE PROGRAM

The applicants' overall Reliability Assurance Program was outlined in a letter dated January 11, 1983 (J. R. Longenecker to P. S. Check, #HQ:S:83:184). This letter describes all of the efforts under way by the applicants to ensure reliable plant operation. The applicants' Program is composed of the following elements:

(1) Design approaches used in ensuring reliability--this includes design reviews, development, and environmental qualification testing, quality assurance, and safety analysis.

(2) The Safety-Related Reliability Program for the reactor shutdown system (RSS) and the reactor residual heat removal system (RRHRS) as described in Appendix C of the PSAR.

(3) The probabilistic risk assessment (PRA) for the entire plant as described in Appendix J of the PSAR.

(4) Key system reviews--review of the interfacing and safety aspects of all systems required for reactor residual heat removal. The review considers failure modes and effects, operation, maintenance, and testing. These

reviews are documented in a letter from J. R. Longenecker to P. S. Check, dated February 19, 1982 (#HQ:S:82-005).

(5) Systems interaction analysis--review of plant systems associated with maintaining high plant availability.

(6) Equipment testing--includes development testing on first-of-a-kind components to verify their performance. Is not intended to develop a statistical data base but may identify failure modes.

(7) Equipment qualification--a test program designed to qualify safety-related equipment to the environment and conditions under which it has to perform. This program is documented in Westinghouse report WARD-D-0165, "CRBR Requirements for Environmental Qualification of Class 1E Equipment."

(8) Failure evaluation--a program for the evaluation of failures resulting from the Equipment Testing Program.

(9) Quality assurance program--the applicants have described an approriate quality assurance program including the following programmatic practices:

    (a) program management
    (b) design control
    (c) procurement control
    (d) manufactoring and construction control
    (e) operation control

    and the following work-oriented practices:

    (a) inspection
    (b) examination
    (c) testing

Details of the quality assurance program are included in Chapter 17 of the PSAR.

For a more detailed description of the applicants' program, the reader is referred to the documents referenced in this section.

C.4 ASSESSMENT OF APPLICANTS' PROGRAM

The applicant's Program, as outlined in the January 11, 1983 letter, contains many of the activities described in the staff's criteria in Section 2. Additionally, the applicants have reviewed the staff's criteria and have committed to revise their Program to comply with all of the staff's criteria (see letter J. R. Longenecker to J. N. Grace, "CRBRP Reliability Assurance Program," dated March 2, 1982, HQ:S:83:229). Based upon this commitment, the staff concludes that the applicants' Reliability Assurance Program is acceptable for a construction permit.

As part of the assessment of the reliability of those systems and features that prevent CDAs, it is essential that the effect of human error be considered. In this regard it is suggested that, as part of this program, the benefit of maintaining diversity in the operation, surveillance, and maintenance of those diverse plant systems associated with prevention of CDAs be explored. If maintaining diversity in this area would contribute significantly to maintaining the reliability of the functions, then it is suggested that this diversity be adopted in the plant operating philosophy or justification be provided as to why this is not desirable.

Because of the unique nature of this Program it is the staff's goal to work with the applicants to ensure development of a meaningful well-documented Program which can be used by both the applicants and the staff as a tool in assessing CRBRP reliability and risk.

It is the staff's plan that as design proceeds and the applicants' activities are further defined and implemented, the staff will periodically audit the Reliability Assurance Program to determine if it is accomplishing the intent of the above criteria and is being implemented in a fashion which contributes to the reliability of CRBRP.

CRBR SAFETY FUNCTIONS (1) / PROGRAM ACTIVITIES	REACTOR SHUTDOWN	SHUTDOWN HEAT REMOVAL	COOLANT SYSTEM BOUNDARY INTEGRITY (2)	PREVENTION OF CORE FLOW BLOCKAGE	PREVENTION OF FAILED FUEL PROPAGATION (3)	CONTAINMENT	EVST HEAT REMOVAL	BEYOND BASIS FEATURES
COMPONENT LEVEL EVALUATIONS	X	X	X	X	X	X	X	X
SYSTEM LEVEL EVALUATIONS	X	X	X	(6)	X	X	X	(6)
ACCIDENT SEQUENCE LEVEL EVALUATIONS	X	X	X	(5)	X	X	X	(6)
COMMON CAUSE FAILURE ANALYSIS	X	X	X	X	X	X	X	(6)
SYSTEMS INTERACTION ANALYSIS	X	X	X	(5)	X	X	X	(6)
EQUIPMENT TESTING (4)	X	X	X	X	X	X	X	X
EQUIPMENT QUALIFICATION	X	X	X	X	X	X	X	X
FAILURE EVALUATION	X	X	X	X	X	X	X	X

NOTES:
(1) The applicable front line and support systems for each function heading are those that are necessary to fullfill the specific safety function.
(2) Leak detection system is an active part of this function.
(3) Delayed neutron detection system is an active part of this function.
(4) Reliability testing of passive features is not required.
(5) Not required because these features are not a system.
(6) Not required because reliability emphasis should be on systems which prevent core disruptive accidents.

Figure C.1 Reliability assurance program activities required for each safety function

APPENDIX D

PROBABILISTIC RISK ASSESSMENT -- CLINCH RIVER BREEDER REACTOR PLANT

D.1 INTRODUCTION

The CRBRP Probabilistic Risk Assessment is one of the principal components of the applicants' Reliability Assurance Program. The PRA provides a mechanism for integrating the deterministic analyses (e.g., failure mode effects analysis, common cause failure analysis) into a complete model of the plant that can be used to obtain an understanding of the relative importance of individual systems and components to overall plant reliability and risks.

Since the Reactor Safety Study (published as WASH-1400, now NUREG-75/014) was performed in the early 1970s, probabilistic risk assessment (PRA) has increasingly been accepted as a means of assessing relative risks in nuclear power plant operations. One of the earliest such safety studies, after WASH-1400, was published as "CRBRP Safety Study, An Assessment of Accident Risks in the CRBRP," CRBRP-1, March 1977 (a Westinghouse document now out of date and not a docketed item). Acceptance of PRA has since reached the level where, in "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," NUREG-0718, Revision 1, June 1981, Requirement II.B.8(1) states:

> Applicants shall: (1) commit to performing a site/plant-specific probabilistic risk assessment and incorporating the results of the assessment into the design of the facility. The commitment must include a program plan, acceptable to the staff, that demonstrates how the risk assessment program will be scheduled so as to influence system designs as they are being developed. The assessment shall be completed and submitted to NRC within two years of issuance of the construction permit. The outcome of this study and the NRC review of it will be a determination of specific preventive and mitigative actions to be implemented to reduce these risks. A prevention feature that must be considered is an additional decay heat removal system whose functional requirements and criteria would be derived from the PRA study.
>
> It is the aim of the Commission through these assessments to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. Applicants are encouraged to take steps that are in harmony with this aim.

D.2 PRA PROGRAM PLAN

The applicants have prepared a probabilistic risk assessment program plan which was submitted in June 1982 and is incorporated into the PSAR as Appendix J. The program plan has been reviewed and found to be a responsive plan to meet the NUREG-0718, II.B.8 requirement. The plan includes what is, in the terminology of the PRA Procedures Guide (NUREG/CR-2300), a Level III PRA, that is, an in-depth PRA.

Major tasks under this part of the plan include initiator development, plant model development and quantification, core and containment accident modeling, and analysis of offsite consequences. Plant model development and quantification include system functional event tree development, fault tree development, analyses of plant response, accident sequence quantification, uncertainty analysis, and common cause failure analysis. Under core and containment accident modeling are the development of phenomenological event trees and the evaluation of source terms.

Other tasks in the program plan, which support application of the PRA, are the development of operator action event trees, an assessment of the effectiveness of design variations including consequence mitigation features, adaptation of the study to a continuing risk management program, providing input to site emergency procedures, and studies which aid in understanding the plant, such as evaluating sensitivities to testing and maintenance intervals.

D.3 PROGRAM PLAN IMPLEMENTATION

This PRA effort was begun about June 1981; the schedule calls for a final report in December 1984.

The PRA effort has been subdivided into two phases. Products from Phase I are a list of initiating events, a set of system event trees, a set of phenomenological event trees with heat transport states and success criteria, a package of fault trees with the data base for quantification, quantification of dominant accident sequences, a dependency analysis, and a sensitivity analysis. The products of Phase I were delivered in early February 1983.

Phase II Part A of the PRA effort includes review and validation of Phase I, plus the tasks remaining to satisfy Level III PRA requirements of the PRA Procedures Guide, NUREG/CR-2300. This includes the radionuclide release, health consequence and risk analyses, the uncertainty analysis, and the common cause failure analysis. Phase II Part B consists of PRA application tasks including adaptation of the PRA to the continuing risk management program, in which application of the PRA can continue through the operating life of the plant.

The applicants had under contract for Phase I, for the accident sequence definition and quantification, EG&G of Idaho, assisted by Wood-Leaver & Associates, Inc. The firm of Fauske & Associates, Inc. was under contract for the accident process analysis.

The Technology for Energy Corp. (TEC) of Knoxville, Tennessee, was awarded the contract for Phase II of the PRA. The results of Phase I have been transferred to TEC.

D.4 NRC REVIEW

The staff is conducting a review of the applicants' PRA effort in which the staff maintains cognizance of applicants' ongoing efforts and provides review and comment for product documents at various stages of their development. The review effort is being conducted with contracted assistance from Science Applications, Inc.

Activities of the review effort include continued monitoring of the ongoing PRA effort by review of the PRA products and by participating in interaction meetings with the applicants, detailed review of specific major elements of the study, and integrated review of the overall PRA. The applicants have committed to interactive meetings to convey early information on methodology and interim results to facilitate the staff review.

Other efforts by the staff related to the review of the PRA are the performance of selected independent assessments, that is, a risk reduction feasibility study of selected modifications to CRBRP safety systems, and a preliminary estimate of release frequencies for CRBRP potential core disruptive accidents.

D.5 FUNCTIONS OF THE PRA

D.5.1 Principal Functions

In addition to its primary function in the CRBRP Reliability Assurance Program as the integrated plant model used to determine the relative importance of individual systems and components to plant reliability and safety, the principal functions of the PRA are (1) to identify specific preventive and mitigative actions to reduce risks, (2) to feed back to the facility design process information which can permit any identified cost-effective risk reduction to be incorporated in the design, (3) to feed back to the reliability program any information needs that the reliability program can provide toward improved risk management. In addition, the PRA establishes the foundation and framework for a continuing risk management program as an aid to plant operations.

D.5.2 Safety Objective and Safety Goals

In the "Final Environmental Statement Related to Construction and Operation of Clinch River Breeder Reactor Plant," NUREG-0139, February 1977, Appendix I, is a letter of May 6, 1976, in which the following, concerning a safety objective, was stated:

> We use the further safety objective that there be no greater than one chance in one million per year for potential consequences greater than the 10 CFR 100 dose guidelines for an individual plant, for example, CRBR; this is a design objective rather than a fixed number which must be demonstrated for a given plant.

This safety objective has been used as an "aiming point" in the safety review of CRBRP.

However, the Commission will issue a Policy Statement on Safety Goals for the Operation of Nuclear Power Plants in the Federal Register. In this Policy Statement the Commission will set forth:

(1) Two qualitative safety goals:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

(2) A cost-benefit guideline:

- The benefit of an incremental reduction of societal mortality risks should be compared with the associated costs on the basis of $1,000 per person-rem averted.

(3) Three quantitative design objectives:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes.
- The likelihood of a nuclear reactor accident that results in a large-scale core melt should normally be less than one in 10,000 per year of reactor operation.

These three quantitative design objectives have been taken by the staff as a candidate to replace the earlier safety objective. Conceptually, PRA can provide results to compare with the quantitative design objectives. The Commission recognized that "because of the sizable uncertainties still present in the methods and the gaps in the data base...[for PRA]...the design objectives should be viewed as aiming points or numerical benchmarks which are subjected to revision." The CRBRP PRA can, however, be of value in indicating whether these "aiming points" are being adequately approached.

The qualitative safety goals supported by the quantitative design objectives have been adopted by the Commission for use during a 2-year evaluation period. They "will not be used in the licensing process or be interpreted as requiring the performance of probabilistic risk assessments during the evaluation period. The goals and objectives are also not to be litigated in the Commission's hearings." If following the 2-year evaluation period, the Commission should elect to extend implementation of the qualitative safety goals and quantitative design objectives to specific cases, for example, CRBRP, the CRBRP PRA will facilitate such further implementation.

D.5.3 Statement of Interim Policy

The Commission's statement of interim policy regarding nuclear power plant accident considerations under the National Environmental Policy Act of 1969 (45 FR 40101, June 13, 1980) requires environmental impact statements to "include a reasoned consideration of the environmental risks (impacts) attributable to accidents at the particular facility" in which "approximately equal attention shall be given to the probability of occurrence of releases and to the probability of occurrence of the environmental consequences of those releases." The statement of interim policy is applicable to environmental impact statements rather than to the safety review, and its requirements are met by the scoping analysis of the risks of accidents at CRBRP which the staff provided in Appendix J of the "Supplement to Final Environmental Statement Related to Construction and Operation of Clinch River Breeder Reactor Plant," NUREG-0139, Supplement No. 1, Vol. 2, October 1982. The Appendix J analysis showed the risk to be similar to that from LWR plants and acceptably low. The Appendix J analysis is independent of the PRA being performed by the applicants; however, the PRA is expected to confirm the results and conclusions of the Appendix J analysis.

D.6 REFERENCES

U.S. Nuclear Regulatory Commission, NUREG-75/014 (formerly WASH-1400), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants." The Rasmussen Report, Oct. 1975.

---, NUREG-0139 "Final Environmental Statement Related to Construction and Operation of Clinch River Breeder Reactor Plant," Feb. 1977; Supplement No. 1, Vol. 2, Oct. 1982.

---, NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," Rev. 1, June 1981.

---, NUREG/CR-2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," Jan. 1983.

---, "Statement of Interim Policy Regarding Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969" (48 FR 40101, June 13, 1980).

APPENDIX E

CHRONOLOGY

October 11, 1974	Project Management Corporation (PMC) and Tennessee Valley Authority (TVA) tender application, Chapter 2 of the PSAR (Vols. 1 and 2) and Environmental Report (ER) (Vols. 1-3), for license to construct and operate the Clinch River Breeder Reactor Plant (CRBRP).
November 14, 1974	Summary of meeting with PMC to discuss the design to be presented in the PSAR.
November 19, 1974	Letter to PMC rejecting the ER for lack of sufficient information and requesting additional information.
November 27, 1974	Letter to PMC requesting additional information on site hydrology.
December 19, 1974	Site visit by staff, PMC and their consultants, and State of Tennessee.
December 27, 1974	Summary of meeting with PMC, TVA, GE, and EPA on December 12, 1974 to discuss scram reliability.
February 14, 1975	Summary of meeting with PMC, Westinghouse, GE, and ERDA on January 23, 1975 to discuss core disruptive accident analysis.
March 31 & April 1, 1975	Site visit by staff, PMC, and TVA.
April 10, 1975	Letter to PMC advising that additional material submitted to satisfy major deficiencies in ER and Chapter 2 of PSAR are acceptable for staff review.
April 11, 1975	Application docketed.
April 11, 1975	PMC submits PSAR (Vols. 3-10) for acceptance review.
April 15, 1975	Summary of meeting with PMC, GE, Westinghouse, and TVA on March 18, 1975 to discuss the reliability of systems designed to remove decay heat from CRBR.
April 16, 1975	Summary of meeting with PMC, ERDA, and Westinghouse on March 20, 1975 to discuss general design criteria.

May 5, 1975	Summary of meeting with PMC, Westinghouse, and ERDA on April 16, 1975 to discuss PMC's progress in the area of radiological source terms for routine releases.
June 5, 1975	Letter to PMC accepting PSAR and requesting additional information.
June 11, 1975	Summary of meeting with PMC, GE, and Westinghouse on May 30, 1975 to discuss information needed for environmental and site suitability reviews.
June 11, 1975	Letter to PMC requesting additional information on site suitability evaluation.
June 12, 1975	Notice of Hearing issued (40 FR 25708, June 18, 1975).
July 17, 1975	ACRS Subcommittee meeting to develop information for consideration of its review of the application.
July 28, 1975	PMC submits Amendment 1 to the PSAR, consisting of responses to requests for additional information.
July 31, 1975	Summary of meeting with PMC, Westinghouse, GE, and ERDA on July 11, 1975 to discuss the status of source terms for site suitability accidents.
August 8, 1975	Summary of meeting with Natural Resources Defense Council on July 14, 1975 to discuss and clarify the scope and status of the radiological and environmental reviews.
August 22, 1975	Summary of meeting with representatives of the Oak Ridge Gaseous Diffusion Plant on July 16, 1975 to discuss postulated releases of toxic chemicals which could adversely affect operation of the plant.
August 25, 1975	PMC submits Amendment 2 to the PSAR, consisting of responses to requests for additional information.
August 27, 1975	Letter to PMC clarifying staff's presentation at July 17, 1975 ACRS Subcommittee meeting relative to preliminary radiological dose assessments on the context of suitability of the proposed site for other reactor types as well as for the CRBRP.
August 29, 1975	PMC submits Amendment 3 to the PSAR consisting of additional responses to requests for information and site suitability source term for the parallel design.
September 2, 1975	Summary of meeting with ERDA and Westinghouse on August 14, 1975 to clarify staff comments made at previous meetings in the matter of requirements for an LWA, with emphasis on radiological site suitability.

September 5, 1975	Summary of meeting with ERDA and consultants on August 15, 1975 to discuss the safe shutdown earthquake and the intensity-acceleration relationship for the plant.
September 12, 1975	PMC submits Amendment 4 to the PSAR, consisting of the site suitability source term for the reference design.
October 6, 1975	Letter to PMC requesting additional information on codes and references cited in the PSAR.
October 6, 1975	PMC submits Amendment 5 to the PSAR consisting of updated appendices for the primary pipe rupture fallback system and the core disruptive accident accommodation.
October 8, 1975	Summary of meeting with ERDA and Westinghouse on August 19, 1975 to discuss current views on apparent critical areas under discussion between PMC and staff, with specific attention to the needs and requirements associated with staff decisions on an LWA.
October 17, 1975	Summary of meeting with ERDA on September 12, 1975 to discuss the seismic design analysis.
October 23, 1975	PMC submits Amendment 6 to the PSAR consisting of responses to requests for additional information and additional design information.
October 24, 1975	Letter to PMC concerning the establishment of a review schedule.
October 29, 1975	Letter to PMC requesting additional information on the reference design.
November 4, 1975	Meeting with ERDA to discuss piping integrity and associated fracture mechanics studies.
November 7, 1975	PMC submits Amendment 7 to the PSAR consisting of responses to requests for additional information and an updated seismic model.
November 13, 1975	Summary of meeting with PMC, ERDA, Westinghouse, and Burns and Roe on October 21, 1975 to discuss the quality assurance program.
November 18, 1975	Letter to PMC requesting additional information.
November 20, 1975	PMC submits an updated shutdown system reliability assessment.
November 26, 1975	Summary of meeting with PMC, ERDA, Westinghouse, and PNL consultants on November 6, 1975 to discuss fuel design and fuel design limits.

December 3, 1975	Summary of meeting with ERDA and PMC on November 5, 1975 to discuss safety system classifications and to clarify PMC's interpretation of seismograph traces recorded at ORNL during injection well operations.
December 4, 1975	Letter to PMC providing additional clarification of the requests for additional information of October 29 and November 18, 1975.
December 5, 1975	Summary of meeting with State and local officials on September 17, 1975 to discuss their specific concerns with the CRBRP.
December 5, 1975	Letter to PMC requesting additional information.
December 11, 1975	Summary of meeting with ERDA and Westinghouse on November 14, 1975 to discuss ERDA-sponsored efforts to quantitatively assess the containment thermal margins in the reference design.
December 11, 1975	PMC submits Amendment 8 to the PSAR, consisting of responses to requests for additional information, a new Appendix 2-C to Chapter 2 incorporating test grouting program report, and revisions to the quality assurance program.
December 17, 1975	PMC submits Amendment 9 to the PSAR, consisting of responses to requests for additional information.
December 17, 1975	PMC submits progress report, "Summary of CRBRP Inherent Retention Analysis," providing scoping analysis of Class 9 events for the reference design.
December 18, 1975	PMC submits a topical report on piping integrity in the primary heat transport system.
December 30, 1975	Letter to PMC requesting additional information on parallel design features.
December 30, 1975	PMC submits Amendment 10 to PSAR, consisting of responses to requests for additional information.
January 9, 1976	Letter to PMC advising of the design criteria which will be used by NRR staff in review of the application.
January 9, 1976	PMC submits a reliability plan for activities which ensure that core disruptive accidents are of sufficiently low probability to be excluded from the design basis.
January 15, 1976	PMC sumbits Amendment 11 to the PSAR, consisting of responses to requests for additional information and WARD quality assurance plans.

January 15, 1976	PMC letter furnishing responses to questions on the industrial security plan (proprietary).
January 21, 1976	Summary of meeting with PMC, ERDA, and Westinghouse on November 13, 1975 to discuss the scope and content of the reliability assessment of the reactor shutdown system.
January 23, 1976	Summary of meeting with PMC, ERDA, and Westinghouse on January 13, 1976 to discuss lack of adequate PSAR documentation for R&D in support of CRBRP.
January 27, 1976	PMC submits report, "Update of the Preliminary Reliability Prediction for CRBRP Shutdown Heat Removal System."
January 28, 1976	Letter to PMC requesting GROWS, SPRAY, FXVARI, ANSYS, TRANSWRAP, PLAP, AND SOFIRE computer codes.
January 28, 1976	Letter to PMC requesting additional information on the possible use of land near the site not presently identified in the PSAR.
January 28, 1976	Letter to PMC requesting additional information concerning injection well activities at ORNL.
January 29, 1976	PMC letter advising that NRC recommended revision to GDC is acceptable for the application and submitting recommended clarification for GDC 15, 27, 29 and 35.
January 29, 1976	Letter to PMC advising of staff position on defining PSAR terminology important to the review.
January 30, 1976	PMC submits "Interim Status Report on Inherent Retention Capabilities of the CRBRP," dated January 1976.
February 2, 1976	Summary of meeting with PMC and ERDA on January 22, 1976 to discuss the site suitability source term.
February 6, 1976	PMC submits Amendment 12 to the PSAR, consisting of responses to requests for additional information.
February 9, 1976	Summary of meeting with PMC, ERDA and Westinghouse on January 16 to discuss NRR staff questions on the parallel design.
February 18-19, 1976	ACRS Subcommittee meeting.
February 20, 1976	PMC submits Amendment 13 to the PSAR, consisting of responses to requests for additional information.
February 20, 1976	PMC submits onsite meteorology and χ/Q calculation.

February 26, 1976	PMC submits response to staff question concerning the safe shutdown earthquake.
March 3, 1976	Summary of meeting with ERDA on February 5, 1976 to discuss system safety classification, design criteria, and piping failure outside containment.
March 5, 1976	Summary of meeting with ERDA on February 3, 1976 to present current understanding of accident energetics and its basis and current and/or future R&D aimed at improving this understanding.
March 5, 1976	Letter to PMC providing the results of staff assessment of their proposed revisions to safety classification and design criteria discussed in the February 2, 1976 meeting.
March 8, 1976	PMC submits Amendment 14 to the PSAR, consisting of responses to requests for additional information and new general arrangement drawings.
March 9, 1976	Letter from PMC transmitting WARD-D-0033, "Preliminary Thermal and Hydraulic Evaluations in the Development of the CRBRP Primary Control System Design."
March 12, 1976	PMC submits response to NRC position on site suitability source term.
March 21, 1976	Meeting with ERDA to discuss injection well activities.
March 22, 1976	Letter to PMC requesting referenced report used as basis for analyses regarding turbine failure and appropriate turbine missile protection.
April 1, 1976	Letter to PMC setting forth areas of disagreement relating to the core disruptive accident and energetics discussed at February 3, 1976 meeting.
April 1, 1976	PMC submits Amendment 15 to the PSAR, consisting of responses to requests for additional information; Appendix F to Chapter 17; and description of S&W quality assurance program.
April 9, 1976	PMC submits report on turbine missile data in response to NRC request dated March 22, 1976.
April 14, 1976	PMC submits Amendment 16 to the PSAR, consisting of responses to requests for additional information.
April 19, 1976	Summary of meeting with ERDA on March 5, 1976 to discuss sodium fire codes SPRAY, CACECO, and SOFIRE.

April 19, 1976	Summary of meeting with PMC on March 11, 1976 to discuss the intensity rating of the maximum historical earthquake and selection of the attendant design ground acceleration.
April 19, 1976	Summary of meeting with PMC on March 19, 1976 to discuss meteorology.
April 22, 1976	PMC submits report, "Third Level Thermal Margins in the CRBRP."
April 23, 1976	Letter to PMC expressing concern regarding their intent and/or capability to document information so as to expedite the resolution of technical issues.
April 30, 1976	PMC submits Amendment 17 to the PSAR, consisting of responses to requests for additional information, and updates to Chapter 17.
April 30, 1976	PMC submits Amendment 18 to PSAR, consisting of additional features to provide additional margin in the reference design.
May 6, 1976	Letter to ERDA (Denise to Caffey) providing comments and guidance on the overall approaches being evaluated and requesting their response.
May 6, 1976	ERDA, PMC and TVA submit Amendment 1 to the Clinch River application to reflect the realignment of responsibilities of the several participants in the project. (ERDA becomes the lead participant.)
May 13, 1976	ERDA submits Amendment 19 to the PSAR, consisting of responses to requests for additional information.
May 14, 1976	Summary of meeting with ERDA and PMC on April 7, 1976 to discuss round two questions on quality assurance.
May 20, 1976	Summary of meeting with ERDA on April 6, 1976 to discuss the reliability program.
May 24, 1976	Letter from ERDA, responding to NRC guidance on CRBRP licensing approach dated May 6, 1976.
May 25, 1976	Letter from ERDA providing additional R&D to support the core disruptive accident analysis.
May 27, 1976	Letter to ERDA providing staff position concerning the safe shutdown earthquake.
May 27, 1976	Letter from ERDA transmitting meteorological data for the period February 11 - March 31, 1976.

May 27, 1976	ERDA submits Amendment 20 to the PSAR, consisting of responses to requests for additional information.
May 27, 1976	ERDA submits additional information on fuel penetration models and experiments.
June 1, 1976	Summary of meeting with ERDA on March 10, 1976 to discuss decay heat removal system redundancy and diversity.
June 2, 1976	Letter from ERDA transmitting "Summary of CRBRP Transient Testing Portion of the Plan for the National LMFBR Mixed Oxide Fuel Transient Performance Program."
June 3, 1976	ERDA submits Amendment 21 to the PSAR, consisting of an assessment of the additional plant margin available under various postulated HCDA mechanical loading conditions.
June 9, 1976	ERDA submits WARD report, "The Development and Application of a Cumulative Mechanical Damage Function for Fuel Pin Failure Analysis in LMFBR Systems."
June 11, 1976	Letter from ERDA advising that a determination by NRC of an appropriate factor for wind meander and agreement is needed in order to calculate the χ/Q.
June 17, 1976	ERDA submits Amendment 22 to the PSAR, consisting of responses to requests for additional information.
June 21, 1976	ERDA submits plan for verification of natural circulation.
June 23, 1976	Letter to ERDA requesting additional information concerning the industrial security and emergency plans.
June 23-24, 1976	ACRS Subcommittee meeting.
June 27, 1976	Summary of meeting with ERDA on June 17, 1976 to discuss the TLTM report, their responses to NRC May 6, 1976 position letter, and schedule considerations.
June 30, 1976	ERDA submits Amendment 23 to the PSAR, consisting of responses to requests for additional information.
June 30, 1976	Summary of meeting with PMC on March 31 to discuss injection well activities.
July 1, 1976	ERDA submits information on the reactor vessel head margin shear ring.
July 2, 1976	Letter to ERDA transmitting staff position on safe shutdown earthquake.

July 8, 1976	Letter from ERDA transmitting correction pages to the "Summary of CRBR Transient Testing Portion of the Plan for the National LMFBR Mixed Oxide Fuel Transient Performance Program."
July 9, 1976	ACRS meeting.
July 14, 1976	ERDA letter transmitting their position on HCDA and siting problems.
July 15, 1976	ERDA letter advising of plans for core drilling and test to determine potential onsite source for concrete aggregate and Class A fill.
July 16, 1976	ERDA letter requesting NRC agreement with their position on appropriate factor for wind meander.
July 22, 1976	ERDA submits Amendment 24 to the PSAR, consisting of responses to requests for information.
July 28, 1976	ERDA letter submitting additional information supporting assessment of plant margin in HCDA mechanical loading conditions.
July 30, 1976	ERDA letter submitting additional information on feature to accommodate site suitability source term.
August 5, 1976	ERDA letter transmitting leak detection information requested at June 18, 1976 meeting.
August 12, 1976	Letter from ERDA Project Office submitting summary of the June 17, 1976 meeting on third level thermal margin report.
August 13, 1976	ERDA submits Amendment 25 to the PSAR, consisting of responses to requests for additional information.
August 17, 1976	Letter to ERDA requesting additional information.
August 20, 1976	ACRS report on hypothetical core disruptive accident for liquid metal fast breeder reactors.
August 27, 1976	ERDA submits Amendment 26 to the PSAR, consisting of responses to requests for information and withdrawal of Appendix E, "Primary Pipe Rupture Accommodation."
August 27, 1976	ERDA letter advising of investigation of previously unidentified linears in vicinity of site.
August 27, 1976	ERDA letter enclosing a plan and schedule for an alternate fuel management scheme.

August 31, 1976	Summary of meeting with ERDA on August 3, 1976 to discuss analysis of structural and mechanical response to CDA.
September 1, 1976	ERDA letter requesting clarification of NRC guidance provided in May 6, 1976 letter concerning plutonium dose guidelines.
September 3, 1976	ERDA letter advising that their position regarding appropriate SSE ground acceleration continues to be 0.189.
September 8, 1976	Meeting with ERDA to discuss containment cell liners and design of basic pipe leaks.
September 8, 1976	ERDA letter providing summary of materials properties of reactor vessel head and surrounding structures.
September 9, 1976	Meeting with ERDA to discuss structural design aspects of the plant.
September 9, 1976	Letter to ERDA documenting NRC staff evaluation of short-term atmospheric dispersion.
September 15, 1976	Letter to ERDA requesting the AYER computer code.
September 16, 1976	Summary of meeting with ERDA on June 18, 1976 to discuss leakage detection for sodium piping.
September 17, 1976	Meeting with ERDA on CACECO code.
September 17, 1976	ERDA letter transmitting additional information on sodium leak detectors.
September 17, 1976	ERDA letter appealing the NRC staff requirement for site suitability source term stated in May 6, 1976 letter.
September 20, 1976	ERDA letter stating environmental qualification of safety-related instrumentation.
September 20, 1976	ERDA letter transmitting additional information on subassembly faults.
September 22, 1976	Meeting with ERDA to discuss NRC staff's CDA analysis.
September 23, 1976	ERDA letter advising that their evaluation of events beyond the design basis are scheduled to be available in early 1977.
September 24, 1976	Letter from ERDA Project Office transmitting additional information on sodium leak detectors.
September 28, 1976	ACRS Subcommittee meeting at Oak Ridge, Tennessee.

October 1, 1976	ERDA submits Amendment 27 to PSAR, consisting of responses to requests for additional information.
October 5, 1976	Summary of meeting with ERDA on August 26, 1976 to discuss their response to EICSB acceptance review and first round questions.
October 5, 1976	ERDA submits report, "Exposure Dependent Cladding Deformation," WARD-D-0146, July 1976 in response to staff request.
October 5, 1976	Meeting with ERDA to discuss atmospheric dispersion of effluents.
October 6, 1976	Letter to ERDA requesting cell design information.
October 7, 1976	Letter to ERDA emphasizing staff position on PHTS piping integrity and requesting additional information.
October 7, 1976	Letter to ERDA advising that staff is not able to confirm that OBE ground acceleration stated in ERDA's September 3, 1976 letter is appropriate and requesting schedule for providing justification to modify the selection criteria of OBE.
October 8, 1976	Letter to ERDA requesting information on materials compatibility between core debris and refractory materials.
October 8, 1976	ERDA letter transmitting reports of experimental information on halogen attenuation and fission gas bubble breakup.
October 14, 1976	ERDA submits Amendment 28 to PSAR, consisting of responses to requests for additional information.
October 15, 1976	ERDA letter addressing NRC staff positions regarding adequacy of decay heat removal system and outlining approach for resolution of concern.
October 15, 1976	ERDA letter providing information on cell liner design.
October 19, 1976	Letter to ERDA highlighting discrepancy in CDA analysis presented in PSAR and ANL/RAS 75-29 "An Analysis of Unprotected Transients Under Cooling and Transient Overpower Accidents in CRBR."
October 20, 1976	Summary of meeting with ERDA on September 27, 1976 to discuss status of their information regarding emergency planning provisions.
October 21, 1976	ERDA sumits the AYER computer code.

October 28, 1976	ERDA sumits Amendment 29 to PSAR, consisting of responses to request for additional information.
October 29, 1976	ERDA letter transmitting additional information on SRI scale model tests.
November 2, 1976	ERDA letter concerning resolution of the site suitability of source term.
November 5, 1976	ERDA letter transmitting an updated analysis of third level margins for the first 24 hours.
November 5, 1976	ERDA letter submitting report, "Fuel Rod Bowing," WARD-D-0150, August 1976, in response to NRC request.
November 11, 1976	ERDA submits Amendment 30 to PSAR, consisting of responses to requests for additional information.
November 23, 1976	Summary of meeting with ERDA on October 21, 1976 to discuss resolution of the site suitability source term.
November 24, 1976	ERDA letter transmitting report, "FORE-2M: A Modified Version of the FIRE-II Computer Program for the Analysis of LMFBR Transients" in response to staff request for information on this subject.
November 30, 1976	ERDA letter transmitting additional information on cell liners.
November 30, 1976	ERDA submits Amendment 31 to PSAR, consisting of responses to requests for additional information.
November 30, 1976	ERDA submits information on industrial security plan.
December 1, 1976	Letter to ERDA requesting additional information.
December 1, 1976	ERDA letter transmitting information on materials compatibility between core debris and refractory materials.
December 6, 1976	Letter to ERDA concerning implementation of CRBRP 1200 MJ "appeal" decision.
December 7, 1976	ERDA letter submitting information on CDA analyses.
December 17, 1976	ERDA submits "An Analysis of Reactivity Effects of Bubble Collapse in a Boiled-up Molten Pool in CRBRP" in response to staff request for information.
December 22, 1976	ERDA submits Amendment 32 to PSAR, consisting of responses to requests for additional information.

December 23, 1976	Letter from ERDA Project Office submitting schedule for round 0, 1, and 2 questions.
December 27, 1976	ERDA submits "TRANSWRAP--A Code for Analyzing the System Effects of Large Leak Sodium-Water Reactions in LMFBR Steam Generation" in response to staff request.
January 3, 1977	Summary of meeting held on October 13, 1976 with Directors of DPM and DSE and their staffs to discuss the staff position regarding CRBRP site suitability source term.
January 3, 1977	ERDA transmits 9 of 12 references to report on third level thermal margins.
January 13, 1977	ERDA submits Amendment 33 to PSAR, consisting of responses to requests for additional information.
January 14, 1977	Summary of meeting held on October 13-14, 1976 with CRBRP representatives to discuss fuel design limits; bases and criteria; and R&D commitments related to fuel design.
January 14, 1977	Letter to ERDA advising that as a result of unscheduled receipt of all necessary information, staff unable to conduct complete review of pipe integrity in limited time suggested by them; pending satisfactory review and resolution of piping integrity issue, cannot agree that PHTS pipe breaks should not be considered a design-basis event.
January 18, 1977	Meeting to discuss structural adequacy of reactor head design.
January 18, 1977	Letter to ERDA transmitting a copy of the December 15, 1976 meeting summary and advising that timely and satisfactory course of action to resolve staff's concerns is not evident.
February 3, 1977	ERDA submits Amendment 34 to PSAR consisting of responses to requests for additional information.
February 4, 1977	Letter to ERDA providing supplementary comments regarding their summary of the December 15, 1976 meeting.
February 11, 1977	ERDA submits report "Simulation Model, DAHRS," CRP002, Revision 3, and "Flow Induced Vibration of Fuel Rods in CRBRP," WARD-0166, December 1976, in response to staff's request for additional information.
February 16, 1977	ERDA submits "Radial Blanket Power to Melt Analysis" in response to request for additional information.

February 17, 1977	ERDA submits "A Recent Evaluation of Foreign Wastage Data from Sodium-Water Reaction Investigation" and "Summary of Design and Development Status of the Liquid Metal to Gas Leak Detection System for the CRBRP" in response to a staff request for information.
February 18, 1977	ERDA submits Amendment 35 to PSAR, consisting of responses to requests for additional information.
March 7, 1977	ERDA submits additional information pertaining to the leak detection system.
March 11, 1977	ERDA submits Amendment 36 to PSAR, consisting of responses to requests for additional information.
March 14, 1977	ERDA transmits "CRBRP Risk Assessment Report."
March 14, 1977	ERDA submits revised information concerning the industrial security plan.
March 15, 1977	Letter to ERDA transmitting Sandia report regarding strength characteristics of concrete at Sandia and requesting review to determine whether this concrete can be expected to be representative of that anticipated for CRBRP.
March 17, 1977	ERDA submits report, "Seismic Evaluation Methods and Criteria for CRBR Fuel; Assembly Duct Structure," WARD-D-0158, October 1976, in response to request for additional information.
March 22, 1977	Letter to ERDA requesting additional information on SRI test programs.
March 23, 1977	Letter from ERDA Project Office informing NRC of their reevaluation of the component fabrication delays.
March 24, 1977	Followup letter from ERDA Project Office on inservice inspection, leak detection, safeguards, and load combinations.
March 25, 1977	ERDA submits Amendment 37 to PSAR, consisting of responses to requests for additional information.
March 25, 1977	ERDA submits, at request of ACRS, a document providing an overview of CRBR design.
March 30, 1977	Letter to ERDA requesting additional information on third level thermal margin report, protection against core meltdown.
April 1, 1977	ERDA submits revised description and schedule for site preparation activities.

April 5, 1977	ERDA submits letter to L. W. Coffee from R. J. Hart, OROO, concerning analysis of potential impact of CRBR operation on ORNL and Oak Ridge Gaseous Diffusion Plant.
April 7, 1977	ERDA submits drawings of models to be used in SRI tests.
April 21, 1977	ERDA submits seismic margin report.
April 22, 1977	ERDA submits Amendment 38 to PSAR, consisting of responses to requests for additional information and revisions to Chapter 14 providing test abstracts that define summary test objectives for first-of-a-kind principal design feature; also response to staff concerns about fuel design.
April 27, 1977	Summary of meeting held on March 9, 1977 with CRBRP representatives and their contractors to discuss the seismic analysis and design margins in the CRBR design.
April 28, 1977	ERDA submits "Plan for the National LMFBR Mixed Oxide Fuel Transient Performance Program."
May 5, 1977	ERDA letter requesting that staff shift review emphasis from environmental hearing preparation to resolution of so-called CP issues.
May 9, 1977	ERDA letter concerning status of agreement between ERDA and NRC relative to treatment of postulated core descriptive events.
May 11, 1977	Summary of meeting held on February 15, 1977 with CRBRP representatives at Westinghouse (WARD) to review the plant protection systems.
May 19, 1977	Letter to ERDA concerning the analysis of margin shear ring and transmitting the March 16, 1977 meeting summary on the subject.
May 20, 1977	Summary of meeting held on March 3, 1977 with CRBRP representatives and ANL to discuss additional calculations and analyses performed by ANL of the LOF accident using the SAS3D computer code.
May 27, 1977	ERDA submits Amendment 39 to PSAR, consisting of responses to requests for additional information.
May 27, 1977	Letter to ERDA (Denise to Caffey) concerning the confusion and misunderstanding which continues to exist by Project of staff's intentions and responsibilities in its technical review of CRBR.

June 14, 1977	ERDA submits, in response to request for information, report, "Impact of Fuel Densification on CRBRP Fuel Performance," WARD-D-0168. March 1977.
July 12, 1977	ERDA letter providing tentative design mix and aggregate specifications for use in test programs.
July 15, 1977	ERDA submits Amendment 40 to PSAR, consisting of responses to requests for additional information.
July 15, 1977	ERDA submits topical report, "CRBRP Closure Head Capability for Third Level Structural Margin Loading," WARD-D-0178, June 1977.
July 18, 1977	Summary of meeting with ERDA on January 26-27, 1977 at ANL, Argonne, Illinois, to discuss LOF CDA energetics.
July 22, 1977	ERDA letter requesting clarification of NRC schedules for review of CRBR application and issuance of SER.
August 5, 1977	ERDA letter advising that information requested on failed fuel on October 6, 1975 was included in analysis of fuel failure propagation furnished on September 20, 1976.
August 8, 1977	ERDA submits report, "Internal/External Cladding Degradation," WARD-D-0147, February 1977.
August 9, 1977	ERDA submits description of LIFE III code.
August 29, 1977	ERDA submits report, "Geological Investigations," S:L: 1531, August 1976.
September 30, 1977	ERDA submits CRBRP piping integrity report.
October 2, 1977	ERDA submits revised GE turbine missile report.
October 14, 1977	DOE submits Amendment 41 to PSAR, consisting of responses to requests for additional information.
November 4, 1977	DOE submits Amendment 42 to PSAR, consisting of revisions to reactivity feedback component of overall power coefficient.
January 27, 1978	DOE submits Amendment 43 to PSAR, consisting of responses to requests for additional information.
March 20, 1978	DOE submits report entitled "Active Pump and Valve Operability Verification Plan," WARD-D-0174.
April 21, 1978	DOE submits Amendment 44 to PSAR, consisting of updates to sections on reactor refueling system, emergency and normal chilled water systems, and other updates and revisions.

May 24, 1978	DOE letter requesting the status on the current staff review of their application.
July 28, 1978	DOE submits Amendment 45 to PSAR, consisting of updates to chapter on quality assurance, to Appendix A, "Computer Codes," to impurity monitoring and analysis system, as well as responses to requests for additional information contained in NRC letter dated August 17, 1976.
August 17, 1978	DOE letter advising of plan to test for determination of potential onsite source for concrete aggregate.
August 25, 1978	DOE submits Amendment 46 to PSAR, consisting of revisions to geology and seismology, seismic design, auxiliary liquid metal system, and general plant description.
September 1, 1978	DOE letter transmitting responses to seismic design questions.
October 6, 1978	DOE submits Topical Report WARD-D-0165, Revision 1, "Requirements for Environmental Qualification of Class 1E Equipment."
November 9, 1978	Letter from W. P. Gammill, NRC, to L.W. Caffey, Director, CRBRP Project, Subject: NRC Discontinuing the safety review of the CRBRP and the staff's status report on major outstanding issues.
November 14, 1978	DOE letter advising of potential industrial development adjacent to CRBRP site.
November 30, 1978	DOE submits Amendment 47 to PSAR, consisting of revisions to industrial security, communication system, compressed gas system, buckling stress criteria and other updates and revisions.
December 13, 1978	DOE submits reports, "Structural Response of CRBRP Scale Models to a Simulated Hypothetical Core Disruptive Accident" (WARD-D-0218), and "Closure Head Capability for Structural Margin Beyond Design Base Loading" (WARD-D-0178).
January 3, 1979	DOE letter concerning CRBRP licensing status.
February 16, 1979	DOE submits topical reports on loss of heat sink, WARD-D-0169 and WARD-D-0170.
February 16, 1979	DOE submits topical report on HCDA's CRBRP-GEFR-00103.
February 23, 1979	DOE submits Amendment 48 to PSAR, consisting of revisions to inert gas receiving and processing system, conventional fire protection system, and other revisions.

March 5, 1979	DOE letter evaluating the NRC staff review of CRBRP.
April 20, 1979	DOE submits Amendment 49 to PSAR consisting of revisions to heating, ventilating, and air conditioning system; radioactive waste management; radiation protection; and other revisions.
June 1, 1979	DOE letter transmitting updated information on industrial security.
June 29, 1979	DOE submitted Amendment 50 to the PSAR.
September 14, 1979	DOE submitted Amendment 51 to the PSAR.
October 19, 1979	DOE submitted Amendment 52 to the PSAR.
December 14, 1979	DOE forwards WARD-D-0050, Revision 3, "Facility Core Assembly Hot Channel Factors Preliminary Analysis."
January 31, 1980	DOE submitted Amendment 53 to the PSAR.
March 11, 1980	DOE forwards WARD-D-0210, "Predicted Steady State Thermal Hydraulic Performance of Fuel and Blanket Assemblies in Plant Heterogeneous Core, Rev. L."
March 25, 1980	DOE submitted CRBRP-3, Vol. 2, "Hypothetical Core Disruptive Accident Considerations: Assessment of Thermal Margin Beyond Design Base."
April 4, 1980	DOE submitted final report on base materials tests for cell liner steels.
April 11, 1980	DOE submitted Revision 1 to WARD-D-0218, "Structural Response of Scale Model to Simulated Hypothetical Core Disruptive Accident."
June 5, 1980	DOE submitted Amendment 54 to the PSAR.
June 27, 1980	DOE submitted Amendment 55 to the PSAR.
June 27, 1980	DOE submitted physical security plan.
June 27, 1980	DOE submitted revised responses to questions 421.3 and 421.10 regarding physical security plan.
August 22, 1980	DOE submitted CRBRP-ARD-0204, "CRBRP Fuel Assembly Structural Analysis in Support of the Final Design Review."
August 29, 1980	DOE submitted Amendment 56 to the PSAR.
November 7, 1980	DOE submitted Amendment 57 to the PSAR.

November 26, 1980	DOE submitted Amendment 58 to the PSAR.
November 28, 1980	DOE provided information concerning pre-test prediction of FFTF natural circulation.
December 30, 1980	DOE submitted Amendment 59 to the PSAR.
February 13, 1981	DOE submitted Amendment 60 to the PSAR.
August 13, 1981	Request from applicants for NRC to resume review of the CRBRP project.
September 18, 1981	Applicants submitted Amendment 61 to the PSAR which includes: updates to Section 1.4, "Identification of Project Participants"; Chapter 3, "Design Criteria--Structures, Components, Equipment and Systems"; Chapter 13, "Conduct of Operations"; Chapter 14, "Initial Tests and Operation"; Section 15.1.2, "Requirements and Criteria for Assessment of Fuel and Blanket Rod Transient Performance"; and Section 16.6, "Administrative Controls."
September 24, 1981	Letter to applicants apprising them of the steps NRC has taken in resumption of the review of CRBRP.
September 29, 1981	Summary of the general LMFBR design considerations and the specific CRB design features presented to the NRC by applicants on September 23, 1981.
October 6, 1981	Meeting notice for October 14 and 15, 1981 with applicants to discuss containment accommodation of core disruptive accidents.
October 19, 1981	Summary of the October 14 and 15, 1981 meeting with applicants.
October 23, 1981	Meeting notice for November 2 and 3, 1981 to discuss electric power systems, heat removal systems and probabilistic risk and reliability analysis,
November 9, 1981	Summary of the November 2 and 3, 1981 meeting with the applicants.
November 10, 1981	Notice of meeting with applicants for November 17, 1981 to discuss systems similar to LWR systems, unique systems, and Chapter 10 systems.
November 13, 1981	Applicants submitted WARD-D-0165, "CRBRP Requirements for Environmental Qualification of Class 1E Equipment," Revision 5.
November 13, 1981	Applicants submitted Revisions 1 and 2 of CRBRP-3, Volume 2, "Hypothetical Core Disruptive Accident Considerations in CRBRP: Assessment of Thermal Margin Beyond the Design Base."

November 13, 1981	Applicants submitted Amendment 62 to the PSAR which includes: updates from previous responses to requests for additional information; revisions to Section 1.4, "Identification of Project Participants"; Section 5.3, "Primary Heat Transport Systems"; Section 5.5, "Steam Generation System"; and an annual update to Chapter 17, "Quality Assurance."
November 13, 1981	Notice of meeting with applicants for November 24, 1981 to discuss CRBR equipment qualification program and compliance with NUREG-0588.
November 13, 1981	Notice of meeting with applicants for December 1, 1981 to discuss CRBRP physical security plan.
November 16, 1981	Notice of meeting with applicants for December 3, 1981 to discuss CRBR control room design.
November 16, 1981	Notice of meeting with applicants for December 10, 1981 to discuss CRBR emergency plans.
November 18, 1981	Letter to applicants requesting they address the informational, environmental, and programmatic changes that have occurred, and the regulatory guidance and requirements that have been promulgated since NRC's review was suspended.
November 19, 1981	Applicants submitted revised responses and revised PSAR figures to the CRBRP physical security plan.
November 20, 1981	Notice of meeting with applicants for December 8 to discuss applicability and compliance with regulatory guides.
November 20, 1981	Notice of meeting with applicants for December 9, 1981 to discuss TMI-related licensing requirements as defined in NUREG-0718, Revision 1.
November 20, 1981	Letter to applicants requesting submission of magnetic tape of onsite meteorological data for evaluation of the radiological consequences of normal and accidental releases to the atmosphere.
November 24, 1981	Notice of meeting with applicants for December 15, 1981 to discuss CRBR QA organization and QA plan.
November 30, 1981	Summary of the November 17, 1981 meeting with applicants.
November 30, 1981	Summary of the November 24, 1981 meeting with applicants.
November 30, 1981	Notice of meeting with applicants for December 14, 1981 to discuss CRBR instrumentation and control systems.
November 30, 1981	Applicants request authorization of the NRC, under 10 CFR 50.12, to conduct site preparation activities for the CRBRP project.

December 3, 1981	Notice of meeting with applicants for December 18, 1981 at Waltz Mill, Madison, Pennsylvania, for discussion and tour of Clinch River test facilities.
December 4, 1981	Summary of the December 1, 1981 meeting with applicants.
December 7, 1981	Summary of the December 3, 1981 meeting with applicants.
December 15, 1981	Summary of the December 10, 1981 meeting with applicants.
December 15, 1981	Summary of the December 14, 1981 meeting with applicants.
December 18, 1981	Applicants submitted Amendment 63 to the PSAR which includes: revisions to Section 1.4, "Identification of Project Participants"; Chapter 8, "Electric Power"; and Chapter 17, Appendix D, and Appendix E, "A Description of the Lead Reactor Manufacturer and Architect-Engineer Quality Assurance Programs."
December 28, 1981	Notice of meetings with applicants for January 11 and 12, 1982 to discuss CRBR electrical drawings and tour of electrical cabinets and prototype panels at Waltz Mill, Pennsylvania.
December 29, 1981	Summary of December 8 and 9, 1981 meetings with applicants.
December 30, 1981	Summary of December 15, 1981 meeting with applicants.
December 30, 1981	Letter to applicants requesting additional information in the geotechnical engineering area.
December 31, 1981	Applicants file with the NRC currently available documentation supporting the factual representations in the November 30, 1981, 10 CFR 50.12 exemption request.
January 6, 1982	Dircks to Commissioners: Staff Responses to Commission Requests--December 9, 1981 Briefing on CRBR Activities.
January 7, 1982	Notice of meeting with applicants for January 15, 1982 to discuss sodium-concrete interactions.
January 8, 1982	Applicants submitted topical report, "An Assessment of HCDA Energetics in the CRBR Heterogeneous Reactor Core, CRBRP-GERF-00523."
January 8, 1982	Notice of meeting with applicants for January 25, 1982 to discuss seismic and dynamic qualifications of mechanical and electrical equipment.
January 8, 1982	Notice of meeting with applicants for January 26, 1982 to discuss the natural circulation test results.

January 8, 1982	Notice of meeting with applicants for January 27, 1982 to discuss structural margin beyond design basis (SMBDB) phenomonology test programs.
January 13, 1982	Letter to applicants requesting additional information in the radiation protection area.
January 15, 1982	Notice of meeting with applicants for January 22, 1982 to discuss the impact of a possible request for an LWA-2 on the safety review schedule.
January 15, 1982	Notice of meeting with applicants for January 28, 1982 to discuss the Stanford Research HCDA scale model test.
January 22, 1982	Notice of meeting with applicants for February 11, 1982 to discuss Appendix R requirements and to discuss sodium fire protection.
January 22, 1982	Letter to applicants requesting additional information in the core energetics area.
January 25, 1982	Summary of the January 15, 1982 meeting with applicants.
January 26, 1982	Notice of meetings with applicants for February 9 and 10, 1982 to discuss the structural design within the design bases.
January 27, 1982	Notice of meeting with applicants--rescheduled from January 22, 1982 to February 8, 1982.
January 28, 1982	Notice of meeting with applicants for February 10, 1982 to discuss the ongoing sodium concrete interaction test programs at HEDL and Sandia laboratories.
January 29, 1982	Applicants submitted Amendment 64 to the PSAR which includes: new Section 6.4, "Cell Liner System"; and revisions to Chapter 4, "Reactor"; Section 6.2, "Containment Systems"; Section 9.2, "Maintenance"; Section 9.13.2, "Sodium Fire Protection System"; Section 11.23, "Gaseous Waste System"; and Section 15.6, "Sodium Spills."
February 2, 1982	Notice of meeting with applicants for February 16, 1982 to discuss CRBR structural design. (Rescheduled for February 17, 1982.)
February 5, 1982	Notice of meeting with applicants for February 12, 1982 to discuss the auxiliary liquid metal systems.
February 5, 1982	Notice of meeting with applicants for February 18, 1982 to discuss the qualification of the applicants as required by NUREG-0718.

February 8, 1982	Notice of meeting with applicants for February 24, 1982 to discuss the scope of loose parts monitoring for CRBR.
February 9, 1982	CRBR Program Office to ACRS--Providing copies of the CRBRP principal design criteria.
February 11, 1982	Notice of meeting with applicants for February 18, 1982 to discuss the structural margin beyond the design basis (SMBDB).
February 17, 1982	Notice of meeting with applicants for February 25 and 26, 1982 to discuss the CRBR accident analyses.
February 17, 1982	Letter from Los Alamos National Laboratory to NRC submitting a set of questions for PSAR Section 4.2, 15.1, and 15.2 to be responded to by applicants.
February 19, 1982	Letter to applicants requesting additional information on inservice inspection.
February 19, 1982	Applicants submitted requested information on core energetics.
February 19, 1982	Applicants submitted a "Summary Report on the Conduct of the Clinch River Breeder Reactor Plant (CRBRP) Key Systems Reviews," which provides a description and overview of system reviews conducted on the integrated performance of selected CRBRP systems.
February 24, 1982	Notice of meeting with applicants for March 4, 1982 to discuss the qualification of the applicants as required by NUREG-0718, Revision 2.
February 26, 1982	Summary of the February 18, 1982 meeting with applicants.
February 26, 1982	Letter to applicants requesting additional information on materials engineering.
February 26, 1982	Applicants submitted Amendment 65 to the PSAR which includes: revisions to Section 2.3, "Meteorology"; Section 9.3, "Auxiliary Liquid Metal System"; Chapter 11 "Radioactive Waste Management"; Chapter 12, "Radioactive Protection"; Section 13.3, "Emergency Planning"; and Appendix G, CRBRP Plan for Inservice and Preservice Inspections."
February 26, 1982	Letter to applicants requesting additional information on structural engineering.
February 26, 1982	Letter to applicants requesting additional information on mechanical engineering.

March 1, 1982	Memorandum to ACRS providing a list of special CRBR review matters that the CRBR Subcommittee is particularly interested in dealing with early.
March 3, 1982	Notice of meeting with applicants for March 10, 1982 at GE ARSD, Sunnyvale, California, to discuss structural margins beyond the design basis.
March 3, 1982	Applicants submitted a report entitled "Summary Report on the Current Assessment of the Natural Circulation Capability with the Heterogeneous Core," CRBRP-ARD-0308, which presents a description of the natural circulation event, the analysis methods, input data, and results of the current assessment of the CRBRP natural circulation capability with the heterogeneous core.
March 4, 1982	Letter to applicants requesting additional information on effluent treatment systems.
March 9, 1982	Letter to applicants requesting additional information on equipment qualification.
March 9, 1982	Summary of the February 24, 1982 meeting with applicants.
March 9, 1982	Summary of the February 25 and 26, 1982 meetings with applicants.
March 11, 1982	Summary of the February 17, 1982 meeting with applicants.
March 11, 1982	Summary of the March 2, 1982 meeting with applicants.
March 11, 1982	Letter to applicants requesting additional information on equipment qualification.
March 11, 1982	Letter to applicants requesting additional information on pipe rupture design criteria and mechanical component design.
March 12, 1982	Summary of January 25, 1982 meeting with applicants.
March 12, 1982	Summary of February 12, 1982 meeting with applicants.
March 12, 1982	Notice of meeting with applicants for March 23 and 24, 1982 to discuss the mechanical, neutronic, and thermal-hydraulic design of the reactor core; design criteria, acceptance criteria; analysis tools; and their verification. (Postponed by applicants.)
March 12, 1982	Notice of meeting with applicants for March 25, 1982 to discuss the structural margin beyond the design basis.
March 15, 1982	Letter to applicants requesting additional information on auxiliary systems.

March 16, 1982	Letter to applicants requesting additional information on power systems.
March 17, 1982	Summary of January 26, 1982 meeting with applicants.
March 17, 1982	Summary of February 11, 1982 meeting with applicants.
March 17, 1982	Summary of March 4, 1982 meeting with applicants.
March 17, 1982	Notice of meeting with applicants for March 29, 1982 to discuss leak detection system.
March 17, 1982	Notice of meeting with applicants for April 1, 1982 to discuss containment systems.
March 17, 1982	Notice of meeting with applicants for April 6 and 7, 1982 to discuss CRBR materials and mechanical engineering.
March 17, 1982	Applicants submitted requested information in the radiation protection area.
March 19, 1982	Summary of February 9 and 10, 1982 meetings with applicants.
March 22, 1982	Applicants submitted Revision 3 of CRBRP-3, Volume 2, "Hypothetical Core Disruptive Accident Consideration in CRBRP; Assessment of Thermal Margin Beyond the Design Base (TMBDB)."
March 23, 1982	Summary of January 15, 1982 and February 10, 1982 meetings with applicants.
March 23, 1982	Summary of January 27, 1982, February 18, 1982, and March 10, 1982 meetings with applicants.
March 23, 1982	Letter to applicants requesting additional information on core performance.
March 23, 1982	Letter to applicants requesting additional information on chemical engineering.
March 24, 1982	Notice of meetings with applicants for April 5, 1982 to discuss Chapter 15, "Accident Analyses."
March 24, 1982	Notice of meetings with applicants for April 13 and 14, 1982 to discuss seismic and structural engineering. (Postponed.)
March 25, 1982	Letter to applicants requesting additional information on core performance.
March 25, 1982	Letter to applicants requesting they address the applicable safeguards regulations.

March 29, 1982	Notice of meeting with applicants for April 16, 1982 to discuss thermal margin beyond the design basis (TMBDB).
March 29, 1982	Applicants submitted requested information on core energetics.
March 31, 1982	Applicants submitted Amendment 67 to the PSAR which includes responses to NRC requests for additional information contained in a letter dated January 13, 1982; and revisions to Section 5.6, "Residual Heat Removal Systems"; Section 7.2, "Reactor Shutdown System"; and Section 7.9, "Operating Control Stations."
April 2, 1982	Summary of January 28, 1982 meeting with applicants.
April 7, 1982	Applicants submitted WARD-D-0165, Revision 6, "CRBRP requirements for Environmental Qualification of Class 1E Equipment."
April 8, 1982	Notice of meeting with applicants for April 26, 1982 at Westinghouse, Waltz Mill site, to discuss materials compatibility test facilities.
April 9, 1982	Letter to applicants requesting additional information on geology and seismology.
April 9, 1982	Letter to applicants requesting additional information on instrumentation and control systems.
April 13, 1982	Notice of meeting with applicants for May 6, 1982 to discuss CRBR management review. (Postponed.)
April 13, 1982	Notice of meeting with applicants for April 21, 1982 to discuss probabilistic risk assessment.
April 14, 1982	Applicants submitted requested information on CRBRP security systems.
April 16, 1982	Notice of meeting with applicants for April 27 at Argonne National Laboratory, Argonne, Illinois, to discuss structural margin beyond the design basis.
April 16, 1982	Notice of meetings with applicants for May 11 and 12, 1982 to discuss Chapter 14, "Reactor Design."
April 16, 1982	Notice of meetings with applicants for May 13 and 14, 1982 to discuss seismic and structural engineering. (Rescheduled from April 13 and 14, 1982.)
April 19, 1982	Applicants submitted requested information on the CRBRP inservice inspection program.

April 20, 1982	Applicants submitted a revision to CRBR-3, Volume 1, "Hypothetical Core Disruptive Accident Consideration in CRBRP: Energetics and Structural Margin Beyond the Design Base."
April 21, 1982	Applicants submitted requested information on chemical technology.
April 26, 1982	Applicants submitted a correction page to their inservice inspection response of April 19, 1982.
April 28, 1982	Summary of meeting with applicants on April 1, 1982 to discuss containment systems.
April 29, 1982	Applicants submitted requested information on chemical and mechanical engineering.
April 30, 1982	Letter to applicants requesting additional information on Chapter 15, "Accident Analyses."
May 7, 1982	Applicants submitted requested information on equipment qualification.
May 11, 1982	Summary of April 6 and 7, 1982 meeting with applicants.
May 11, 1982	Summary of March 29, 1982 meeting with applicants.
May 14, 1982	Letter to applicants requesting additional information on core disruptive accident analyses, the fuel-handling system, and sodium fire protection.
May 14, 1982	Letter to applicants requesting additional information on emergency planning.
May 14, 1982	Applicants submitted requested information on auxiliary systems.
May 14, 1982	Applicants submitted a list of topics and reports to be submitted in the near future in support of the CRBRP's assessment of thermal margin beyond the design base.
May 17, 1982	Applicants submitted requested information on equipment qualification.
May 17, 1982	Applicants submitted requested information on mechanical engineering.
May 17, 1982	Notice of meeting with applicants for June 3, 1982 to discuss licensee qualification. (Rescheduled meeting.)
May 18, 1982	Applicants submitted requested information on effluent treatment.

May 18, 1982	Summary of April 5, 1982 meeting with applicants.
May 26, 1982	Applicants submitted Amendment 2 to their "Statement of General Information."
May 28, 1982	Applicants submitted Amendment 68 to the PSAR which includes responses to CRBR Program Office's requests for additional information contained in letters dated February 26, 1982 and March 28, 1982 and revisions to Section 13.7, "Radiological Security."
June 1, 1982	Applicants submitted requested information on structural engineering.
June 1, 1982	Applicants submitted requested information on core performance.
June 1, 1982	Applicants submitted requested information on the reactor system, heat transport piping system, and Class 1E equipment qualification.
June 1, 1982	Applicants submitted requested information on core performance.
June 1, 1982	Applicants submitted requested information on power systems.
June 2, 1982	Notice of meeting with applicants for July 27, 1982 at Waltz Mill site, Madison, Pennslyvania, to discuss steam generator system.
June 2, 1982	Applicants submitted requested information on seismic qualification of mechanical components, materials engineering, reactor physics, and seismic structures.
June 2, 1982	Summary of March 25, 1982 and April 27, 1982 meetings with applicants.
June 3, 1982	Summary of May 11 and 12, 1982 meetings with applicants.
June 7, 1982	Notice of meeting with applicants for June 18 to discuss sodium fire protection.
June 8, 1982	Notice of meeting with applicants for June 22, 1982 to discuss mechanical, nuclear, and thermal hydraulic design of the CRBRP core.
June 8, 1982	Notice of meeting with applicants for June 23, 1982 to discuss fuel failure monitoring system.
June 8, 1982	Applicants submitted requested information on structural engineering.

June 8, 1982	Applicants submitted requested information on instrumentation and controls and design criteria.
June 8, 1982	Applicants submitted requested information on geology and seismology.
June 8, 1982	Applicants submitted a description of the CRBRP steam generator test program and a detailed analysis of the May 25, 1982, General Accounting Office report.
June 9, 1982	Letter to applicants requesting additional experiments to confirm the structural capability of the CRBRP vessel head to accommodate core disruptive accidents and to benchmark the analytical models used to analyze the vessel head response and failure modes.
June 9, 1982	Letter to applicants requesting additional information on nuclear design.
June 9, 1982	Letter to applicants requesting additional design layout drawings.
June 9, 1982	Notice of meeting with applicants for June 16-17, 1982 to discuss structural margin beyond the design basis. (Cancelled.)
June 9, 1982	Notice of meeting with applicants for June 22-24, 1982 at Burns and Roe, Oradell, New Jersey, to perform a CRBR seismic and structural engineering audit of calculations.
June 10, 1982	Applicants submitted a copy of the report entitled "Verification of Natural Circulation in the Clinch River Breeder Reactor Plant--An Update."
June 11, 1982	Memorandum from CRBR Program Office to ACRS transmitting NUREG-0786 "CRBRP Site Suitability Report."
June 14, 1982	Applicants submitted requested information on piping design, auxiliary systems, and instrumentation and control systems.
June 16, 1982	Notice of meeting with applicants for June 30, 1982 and July 1, 1982 to discuss structural margin beyond the design basis.
June 17, 1982	Applicants submitted the following report "ES-LPD-82-007, 008, 009, 011" and requested information on materials engineering.
June 17, 1982	Applicants submitted requested information on mechanical and structural engineering.

June 17, 1982	Applicants submitted requested information on the dynamic and static analysis used to determine the structural and functional integrity of selected seismic Category I components.
June 18, 1982	Applicants submitted requested information on power systems and core performance.
June 21, 1982	CRBR Program Office requested additional information on the core disruptive accident energetics analyses presented in GEFR-0523.
June 21, 1982	Applicants submitted requested information on the probabilistic risk assessment program plan.
June 21, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
June 25, 1982	Applicants submitted a revised PSAR figure for the CRBRP physical security plan.
June 25, 1982	Applicants submitted requested isometric drawings on the piping fabrication for the direct heat removal system.
June 25, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
June 25, 1982	Applicants submitted an update to PSAR Section 13.5 on plant procedures.
June 25, 1982	Applicants submitted the requested drawings P&ID BM502, "Main Steam System," and Instrument Loop Diagram BE4107, "Main Steam System."
June 29, 1982	Applicants submitted requested information on sodium fire protection.
June 29, 1982	Applicants submitted information requested by the Technical Review Section.
June 30, 1982	Notice of meeting with applicants for July 8, 1982 to discuss CRBR hydrology review.
June 30, 1982	Applicants submitted a revised response on instrumentation and control systems.
June 30, 1982	Applicants submitted information requested by CRBR Program Office Technical Review Section.
July 2, 1982	Applicants submitted requested information on core performance.
July 2, 1982	Applicants submitted the requested CACECO computer code.

July 6, 1982	Applicants submitted requested information on ASME Publication PVP-63, "A Procedure to Evaluate Structural Adequacy of a Piping System in Creep Range."
July 7, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
July 13, 1982	The USGS submitted input to NRR/GSB on the suitability of the CRBRP.
July 13, 1982	Letter from ACRS Chairman Shewmon to NRC Chairman Palladino with a report on the suitability of the CRBRP site.
July 13, 1982	Summary of April 16, 1982 meeting with applicants.
July 13, 1982	Summary of June 18, 1982 meeting with applicants.
July 13, 1982	Summary of June 22, 1982 meeting with applicants.
July 13, 1982	Summary of June 23, 1982 meeting with applicants.
July 14, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
July 15, 1982	Applicants submitted information on the post-test analyses of the FFTF natural circulation tests--reports CRBRP-ARD-0310, "Verification of the CRBRP Natural Circulation Core Analyses Methodology with Data from FFTF Natural Circulation Tests--June 1982" and WARD-NC-94000-6 "DEMO Post Test Analysis of the FFTF Transient Natural Circulation Tests--June 1982."
July 15, 1982	Applicants submitted requested information on thermal and hydraulic design.
July 15, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
July 16, 1982	CRBR Program Office asked applicants to assess the applicability of identified unresolved (some resolved) generic safety issues to CRBRP.
July 16, 1982	Notice of meeting with applicants for July 23, 1982 to discuss probabilistic risk assessment.
July 22, 1982	Applicants submitted Revision 4 of CRBRP-3, Volume 2, "Hypothetical Core Disruptive Accident Considerations in CRBRP; Assessment of Thermal Margin Beyond the Design Base."
July 26, 1982	CRBR Program Office to applicants transmitting a copy of the ACRS report on the site suitability of CRBRP to Chairman Palladino.

July 26, 1982	Notice of meeting with applicants to discuss auxiliary liquid metal system.
July 28, 1982	Applicants submitted requested information on emergency planning.
July 28, 1982	Applicants submitted a drawing as further response to emergency planning questions.
July 29, 1982	Applicants forwarded updated pages for reference 106 of PSAR Section 1.6, CRBRP-3, Volume 2, "Hypothetical Core Disruptive Accident Considerations in CRBR; Assessment of Thermal Margin Beyond the Design Base."
July 30, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
July 30, 1982	Applicants submitted Amendment 69 to the PSAR which includes responses to CRBR Program Office's requests for additional information contained in letters dated February 26, 1982; March 11, 1982; March 15, 1982; March 23, 1982; March 25, 1982; and April 9, 1982; Revisions to Section 3.7, "Seismic Design"; Section 3.8, "Design of Category I Structures"; and Chapter 4, "Reactor."
July 30, 1982	Applicants submitted corrected page replacement guide to Amendment 69.
August 6, 1982	Applicants submitted design layout drawings for the containment penetrations, the containment ring stiffeners and overhead crane support, the structures within the containment-confinement annulus, cell, and cell liners, and the reactor vessel support ledge requested by CRBR Program Office.
August 6, 1982	Applicants submitted a request for authorization to proceed with LWA-2 activities.
August 10, 1982	Notice of meeting with applicants for August 17 to discuss thermal margins beyond the design base.
August 13, 1982	Summary of July 23 meeting with applicants. (Draft report on "Analysis of Nominal Heat Removal Capacity of the CRBRP in the Natural Circulation Mode.")
August 20, 1982	Applicants submitted Amendment 70 to the PSAR which includes responses to CRBR Program Office requests for additional information contained in letters dated February 26, 1982; revisions to Chapter 13, "Conduct of Operations"; Sections 17.0, 17.1, 17A, 17C, and 17F, "A Description of the Owner Assurance Program"; and Appendix C, "Safety Related Reliability Program."

August 20, 1982	Applicants submitted design layout drawings for all components and structures within, comprising, or attached to the reactor enclosure requested by CRBR Program Office.
August 23, 1982	Notice of meeting with applicants for September 15, 1982 to discuss thermal margin beyond the design base.
August 24, 1982	Notice of meeting with applicants September 8 and 9, 1982 to discuss mechanical engineering.
August 24, 1982	Summary of July 8, 1982 meeting with applicants.
August 24, 1982	Applicants submitted requested information on instrumentation and control systems and information requested by the CRBR Program Office Technical Review Section.
August 26, 1982	Applicants submitted requested information on sodium dump system, argon cover gas monitoring, reactor delayed neutron monitoring subsystem, and the effects of high temperatures in reference legs of steam drum water level measuring instruments.
August 31, 1982	Applicants informed CRBR Program Office of the initiation of site preparation activities.
September 1, 1982	Applicants submitted an action plan to resolve questions relating to monitoring component degradation in the nuclear steam supply systems.
September 7, 1982	CRBR Program Office provided ACRS with results of staff's review of potential effects of a CRBRP-type plant on the Oak Ridge Gaseous Diffusion Plant (K-25).
September 7, 1982	Summary of August 17, 1982 meeting with applicants.
September 8, 1982	Summary of August 5, 1982 meeting with applicants.
September 8, 1982	Notice of meeting with applicants for September 21 and 22, 1982 to discuss instrumentation and control systems.
September 8, 1982	Applicants submitted requested information on instrumentation and control systems not required for safety.
September 10, 1982	Notice of meeting with applicants for September 16 and 17, 1982 to discuss structural engineering.
September 13, 1982	Notice of meeting with applicants for September 15, 1982 to discuss LWA-2 TMBDB and SMBDB issues. (Revised from August 23, 1982 notice.)

September 13, 1982	Notice of meeting with applicants for September 21, 1982 at Argonne National Laboratory, Argonne, Illinois, to discuss hypothetical core-disruptive accidents and structural margins beyond the design basis.
September 14, 1982	Applicants submitted requested information on the soduium fire protection system, the fuel failure monitoring system, the design of the CRBR purge system, and the applicability of the RDT standards to safety-related instrumentation and control systems.
September 20, 1982	Summary of September 15, 1982 meeting with applicants.
September 21, 1982	Applicants submitted a summary of the September 8 and 9, 1982 meetings.
September 22, 1982	Notice of meeting with applicants for September 28, 1982 to discuss leak detection.
September 23, 1982	CRBR Program Office provided comments to DOE on the CRBRP Probabilistic Risk Assessment Program Plan.
September 24, 1982	Applicants submitted a summary of the September 21 and 22, 1982 meetings.
September 24, 1982	Notice of meeting with applicants for September 29, 1982 to discuss CRBR principal design criteria.
September 27, 1982	Notice of meeting with applicants for October 6, 7, and 8, 1982 to discuss direct heat removal system.
September 28, 1982	Notice of meeting with applicants for October 5, 1982 to discuss structural and seismic review.
September 28, 1982	Applicants submitted requested microfiche containing CACECO input/output for extreme penetration cases.
September 29, 1982	Applicants submitted information for review of CRBRP-3, Volume 2, "Letter Report, TMBDB Instrumentation Development."
September 30, 1982	Applicants submitted Amendment 71 to the PSAR which includes responses to CRBR Program Office requests for additional information contained in letters dated April 19 and 30, May 14, June 9 and 21, and July 16, 1982; revisions to Chapter 7, "Instrumentation and Controls"; and Sections 17E and 17I, "A Description of the A-E and GE-ARSD-RM Quality Assurance Programs."
October 4, 1982	Applicants submitted the additional information on instrumentation and control systems requested at the September 21 and 22, 1982 working meeting.

October 4, 1982	Applicants submitted requested information on instrumentation and control systems.
October 7, 1982	Notice of meeting with applicants for October 14, 1982 to discuss reactivity control.
October 7, 1982	Applicants submitted a report on "Preliminary Analysis of Heat Generating Blockages in CRBRP Fuel and Radial Blanket Assemblies To Determine Detection Requirements, CRBRPO-ARD-0119," in response to a request from the CRBR Program Office Technical Review Section.
October 7, 1982	Applicants submitted a summary of the September 28, 1982 meeting.
October 12, 1982	Notice of meeting with applicants for October 20 and 21, 1982 to discuss power systems.
October 12, 1982	CRBR Program Office requested additional information on thermal stress.
October 12, 1982	Notice of meeting with applicants for October 18, 1982 to discuss containment systems.
October 13, 1982	Notice of meeting with applicants for October 19 and 20, 1982 to discuss power systems. (Revised from October 12, 1982.)
October 14, 1982	Notice of meeting with applicants for October 19, 1982 to discuss reactor control room design.
October 14, 1982	Notice of meeting with applicants for October 20, 1982 to discuss CRBR thermal hydraulics.
October 15, 1982	Applicants submitted a summary of the September 21, 1982 meeting on HCDA energetics.
October 15, 1982	Applicants submitted a summary of the September meeting on ASME Code comparison.
October 20, 1982	Applicants submitted additional information requested at the September 8 and 9, 1982 meeting with the Mechanical Engineering Branch.
October 20, 1982	Applicants submitted a document entitled, "Thermal Margin Beyond the Design Base Sodium-Concrete Penetration Margins Assessment for the CRBRP."
October 21, 1982	Applicants submitted a summary of the October 19, 1982 meeting on control room design philosophy and approach.
October 21, 1982	Applicants submitted a summary of the October 20, 1982 meeting on the decay heat removal and thermal hydraulics.

October 22, 1982	Notice of meeting with applicants for October 28 and 29, 1982 to discuss CRBR materials and mechanical issues, including leak before break and leak detection.
October 26, 1982	Applicants submitted requested information on instrumentation and control systems.
October 26, 1982	Applicants submitted additional information requested at the decay heat removal meeting of October 20, 1982.
October 29, 1982	Applicants submitted Amendment 72 to the PSAR, which includes responses to CRBR Program Office's requests for additional information contained in letters dated April 19 and 30, May 14, June 9 and 21, and July 16, 1982; revisions to Section 11.4, "Process and Effluent Radiological Monitoring System"; and Chapter 12, "Radiation Protection."
November 1, 1982	Notice of meeting with applicants for November 8, 1982 to discuss loose parts monitoring.
November 2, 1982	Applicants submitted additional information requested at the electrical power meeting of October 19, 1982.
November 3, 1982	Applicants submitted additional information requested at the instrumentation and control systems meeting of September 21 and 22, 1982.
November 3, 1982	Applicants submitted a summary of the October 28 and 29, 1982 meeting on piping integrity.
November 9, 1982	Applicants submitted a summary of the November 8, 1982 meeting on component degradation monitoring.
November 10, 1982	Notice of meeting with applicants for November 15, 1982 to discuss control rod logic design and function.
November 12, 1982	Notice of meeting with applicants for November 16 and 17, 1982 to discuss instrumentation and control.
November 12, 1982	Notice of meeting with applicants for November 17, 1982 to discuss structural engineering.
November 12, 1982	Notice of meeting with applicants for November 18 and 29, 1982 at the Project Office, Oak Ridge, Tennessee, to discuss CRBR control room design.
November 12, 1982	Applicants submitted a summary of the September 15, 1982 meeting on thermal margin beyond the design base.
November 12, 1982	Applicants submitted a report entitled, "Supplementary Manual for the FØRE-2M Computer Program, CRBRP-ARD-0257."

November 16, 1982	Notice of meeting with applicants for November 22 and 23, 1982 at Waltz Mill, Madison, Pennsylvania, to discuss mechanical engineering calculations.
November 19, 1982	Notice of meeting with applicants for November 22 and 23, to discuss reactor design.
November 23, 1982	Applicants submitted the additional information requested at the September 8 and 9, 1982 meeting with the Mechanical Engineering Branch.
November 23, 1982	Applicants submitted a document entitled, "TMBDB Melting Scenario."
November 30, 1982	Applicants submitted Draft B of the report entitled, "Fire Hazard Analysis (FHAR)."
November 30, 1982	Applicants submitted Amendment 73 to the PSAR which includes revisions to Section 2.4, "Hydrologic Engineering"; Section 7.2, "Reactor Shutdown System"; and Section 17J, "A Description of the ESG-RM Quality Assurance Program."
December 1, 1982	Applicants submitted a response to item 6 of the action items from the October 18, 1982 meeting on containment systems.
December 1, 1982	Applicants submitted a summary of the November 23, 1982 meeting on equipment qualification.
December 2, 1982	Notice of meeting with applicants for December 8, 1982 to discuss inservice inspection review items.
December 2, 1982	Notice of meeting with applicants for December 8 and 9, 1982 to discuss structural engineering review items.
December 2, 1982	Notice of meeting with applicants for December 16, 1982 to discuss shutdown heat removal systems.
December 6, 1982	Applicants submitted a summary of the November 25 and 26, 1982 meeting on reactor design.
December 6, 1982	Applicants submitted requested information on electric power and mechanical systems.
December 6, 1982	Applicants submitted requested information on instrumentation and control systems.
December 6, 1982	Applicants submitted additional information requested at the December 2 and 3, 1982 meetings on control room design.
December 7, 1982	Applicants submitted requested information on thermal margins beyond the design base.

December 13, 1982	Summary of the September 15, 1982 meeting with applicants on thermal margins beyond the design base.
December 13, 1982	Applicants submitted a revision to "CRBRP-3 Volume 1, Structural Margin Beyond the Design Base."
December 14, 1982	Applicants submitted requested information on instrumentation and control systems.
December 14, 1982	Applicants submitted additional information requested by Mechanical Engineering Branch.
December 14, 1982	Applicants submitted a summary of the December 9, 1982 meeting on structural margin beyond the design base.
December 17, 1982	Applicants submitted additional clarification of CRBRP training program.
December 20, 1982	Applicants submitted a summary of the December 16, 1982 meeting on shutdown heat removal.
December 20, 1982	Applicants submitted additional information on instrumentation and control systems.
December 20, 1982	Applicants submitted additional information requested at the December 15, 1982 meeting on plant auxiliary systems.
December 21, 1982	Applicants submitted additional information requested at the December 8, 1982 meeting on thermal margin beyond the design base.
December 21, 1982	Applicants submitted a summary of the December 20, 1982 meeting on the reliability program.
December 21, 1982	Applicants submitted requested information on the secondary control rod system.
December 21, 1982	Applicants submitted additional information requested at the December 9, 1982 meeting on structural margin beyond the design base loads on the reactor support ledge.
December 21, 1932	Applicants submitted a summary of the December 20, 1982 meeting on environmental qualification of equipment.
December 22, 1982	Applicants submitted additional information regarding emergency planning.
December 22, 1982	Applicants submitted requested information on the intermediate heat transport system tee.
December 22, 1982	Applicants submitted additional information requested at the November 22-24, 1982 meeting with the Mechanical Engineering Branch.

December 23, 1982	Applicants submitted requested information on instrumentation and control systems.
December 23, 1982	Applicants submitted requested information on containment systems.
December 23, 1982	Applicants submitted requested information on energetics analysis.
December 28, 1982	Applicants submitted requested information on margin in the plant protection system setpoints.
December 28, 1982	Applicants submitted requested information regarding the plant procedures.
December 28, 1982	Applicants submitted additional information on seismic qualification.
December 28, 1982	Applicants submitted requested information concerning project technical resources, training, and utilization of industry experience.
December 29, 1982	Applicants submitted a summary of the SER open-item meeting held on December 21, 1982.
December 29, 1982	Applicants submitted updated information on the environmental design of mechanical and electrical equipment.
December 29, 1982	Applicants submitted a summary of the December 8, 1982 meeting on containment vessel/code case(s) analysis.
December 30, 1982	Applicants submitted Amendment 74 to the PSAR which includes revised responses to NRC question CS430.1 through 104; revisions to Section 3.2, "Classifications of Structures, Systems and Components"; Section 6.2, "Containment Systems"; Chapter 7, "Instrumentation and Controls"; Chapter 8, "Electric Power"; and Section 9.14, "Diesel Generator Auxiliary Systems."
January 5, 1983	Applicants submitted a revision to Section 17D, "A Description of the Nuclear Steam Supply System (NSSS) Supplier Quality Assurance Program."
January 5, 1983	Applicants submitted the DEMO code output assumptions used for the pipe break analysis.
January 5, 1983	Applicants submitted requested information on seismic margin beyond the design base criteria and a writeup on the benchmarking analyses against the SM-1 test.
January 6, 1983	Applicants submitted requested information on the direct heat removal service.

January 7, 1983	Applicants submitted a report on the "Methodology for CRBRP's Application of Radiological Source Terms in Containment."
January 7, 1983	Applicants submitted additional information on the reactor vessel and ex-vessel storage tank nondestructive examination.
January 7, 1983	Applicants submitted responses to questions concerning auxiliary liquid metal systems and plant fire protection system.
January 10, 1983	EG&G report to CRBR Program Office entitled "Comparison of Clinch River Breeder Reactor Design Basis Accidents With Those for Light Water Reactors and Liquid-Metal-Cooled Fast Reactors," EGG-NTAP-6152.
January 11, 1983	Applicants submitted additional information on steam generator nondestructive examination and reactor vessel core support cone structural integrity.
January 11, 1983	Applicants submitted additional information on mechanical engineering.
January 11, 1983	Applicants submitted additional information on material surveillance.
January 11, 1983	Applicants submitted additional information on core instrumentation.
January 11, 1983	Applicants submitted personnel résumés of key positions for CRBRP management organization.
January 11, 1983	Applicants submitted clarifying information on the selection of the groundwater level for use in seismic design of Category I structures.
January 11, 1983	Applicants submitted the "CRBRP Reliability Assurance Activities" program.
January 12, 1983	Applicants submitted a summary of the November 17, 1982 meeting of seismic/structure/cell liner analysis and responses to questions brought up at the meeting.
January 12, 1983	Applicants submitted a modification of PSAR Section 14 clarifying the application of operational and test experience from similar operating reactors to the CRBRP test program
January 12, 1983	Applicants submitted additional and revised information on the plant auxiliary systems.

January 20, 1983	Applicants submitted additional information on inerted cells in the reactor service building.
January 20, 1983	Applicants submitted additional information on the composition of NaK and its solidus temperature.
January 21, 1983	Applicants submitted additional information on CRBRP engineered safety features and maintenance system.
January 25, 1983	Applicants submitted additional and revised information on CRBRP auxiliary systems.
January 26, 1983	Applicants submitted additional information on the electric power system.
January 26, 1983	Applicants submitted additional information on the instrumentation and control systems.
January 26, 1983	Applicants submitted additional information on the primary heat transport system hot-leg piping code evaluation.
January 27, 1983	Applicants submitted additional information on sodium spill volumes for inerted cells.
January 27, 1983	Applicants submitted additional information on reactor material surveillance.
January 27, 1983	Applicants submitted additional information on heat removal service temperature limits.
January 27, 1983	Applicants submitted additional information on qualification of mechanical equipment.
January 27, 1983	Applicants transmitted CRBRP-ARD-0315, "Clinch River Breeder Reactor Plant Verification of FØRE-2M Computer Code."
January 27, 1983	Applicants submitted additional information on stainless steel and insulation properties of engineered safety features and on welding qualification in areas of limited accessibility.
January 28, 1983	Notice of meeting with applicants for February 9, 1983 on Phase II of the Probabilistic Risk Assessment effort.
January 29, 1983	Applicants submitted additional information on mechanical engineering.
February 2, 1983	Applicants' response to the recently issued NRC CRBRP principal design criteria.

February 2, 1983	Applicants submitted two pages that were inadvertently left out of response dated December 14, 1982 on instrumentation and control systems.
February 2, 1983	Applicants submitted additional information on ex-vessel storage tank cooling.
February 3, 1983	Notice of meeting with applicants for February 9, 1983 with the Mechanical Engineering Branch.
February 4, 1983	Applicants submitted additional information resulting from open-items meeting of December 21, 1982.
February 4, 1983	Applicants submitted additional information on sodium spills.
February 4, 1983	Applicants submitted additional information on mitigation of waterhammer in the steam generator system.
February 4, 1983	Applicants submitted information on precautions that preclude assembly blockages.
February 8, 1983	Applicants submitted Amendment 75 to the PSAR which includes: Revisions to Section 1.4, "Identification of Project Participants"; Section 5.0, "Heat Transport and Connected Systems"; Chapter 7, "Instrumentation and Controls"; Chapter 9, "Auxiliary Systems"; and Section 17D, "A Description of the Westinghouse Quality Assurance Program."
February 10, 1983	Applicants forward correction pages to "PSAR Amendment 75 page replacement guide.
February 10, 1983	Applicants submitted information on confirmatory high temperature design programs.
February 14, 1983	Applicants submitted additional information on potential highway accidents with resulting toxic plumes that could impact CRBRP.
February 14, 1983	Applicants submitted additional information on nitrogen gas services system.
February 14, 1983	Letter from applicants on reactor closure head capability to meet margin requirements.
February 15, 1983	Applicants submitted additional information on instrumentation and control.
February 15, 1983	Applicants submitted additional information requested by the Mechanical Engineering Branch at the February 9, 1983 meeting.

February 15, 1983	Applicants submitted the revised Section 3.1 of the PSAR that incorporates the final principal design criteria.
February 23, 1983	Applicants submitted additional information on the circulating water system.
February 23, 1983	Applicants submitted additional information on the secondary control rod system.
February 24, 1983	Applicants submitted supplemental information to the Mechanical Engineering Branch.
February 25, 1983	CRBR Program Office transmits criteria requirements that the CRBRP Reliability Assurance Program must meet.
February 25, 1983	Applicants submitted additional informaton on plant emergency planning.
February 28, 1983	Applicants submitted additional information on nondestructive examination procedure.
February 28, 1983	Applicants submitted additional information on primary sodium gas entrainment and assembly flow blockage criteria.
March 2, 1983	Applicants provide further responses on the CRBRP Reliability Assurance Program.

APPENDIX F

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This Safety Evaluation Report is a product of the NRC staff and consultants. The NRC staff members listed below were principal contributors to this report. A list of consultants follows the list of staff members.

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APPENDIX G

REFERENCES

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APPENDIX H

U.S. GEOLOGICAL SURVEY FINAL SEISMOLOGY AND GEOLOGY REVIEW OF THE CLINCH RIVER BREEDER REACTOR SITE

United States Department of the Interior

GEOLOGICAL SURVEY
RESTON, VA 22092

January 19, 1983

Mr. Robert E. Jackson
Chief, Geosciences Branch
Division of Engineering
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Dear Bob,

Enclosed is the Geological Survey final Seismology and Geology Review of the Clinch River Breeder Reactor Site.

Sincerely,

Robert H. Morris for

James F. Devine
Assistant Director
for Engineering Geology

Enclosure

Final Review
Geology
D. D. Dickey
R. C. McDowell
Seismology
D. M. Perkins
S. T. Algermissen
January 19, 1983

U.S. Department of Energy
Clinch River Breeder Reactor Plant
Oak Ridge, Tennessee
NRC Docket No. 50-537

Introduction

The U. S. Geological Survey has reviewed the geological and seismological data and analysis in the Preliminary Safety Analysis Report (PSAR) for the Clinch River Breeder Reactor Plant site located about 25 miles west of Knoxville, Tennessee

Geology

Introduction

The U.S. Geological Survey (USGS) reviewed the geologic data and analysis in the Preliminary Safety Analysis Report (PSAR) for the Clinch River Breeder Reactor plant site and has compared it to the geologic literature of the area including the Phipps Bend PSAR, Docket Nos. 50-553, 50-554, and the Watts Bar SER Docket Nos. 50-390 and 50-391. A field inspection of the site and surrounding area was made June 2 and 3, 1982.

The Clinch River site, in the southwest corner of the U.S. Department of Energy Oak Ridge Reservation, Roane County, Tennessee, is inside a meander loop of the Clinch River at the upper end of Watts Bar Lake.

The site is in the Valley and Ridge Physiographic Province, which extends about 500 miles northeastward from Alabama to Virginia and is about 25 to 50 miles wide. The northeast-trending valleys are underlain by easily erodible shale and mudstone and soluble limestone, whereas the ridges are supported by more resistant sandstone, siltstone, and siliceous limestone and dolomite.

The topography in the vicinity of the site is characterized by such northeast-trending ridges with intervening valleys. Normal lake level in the valley is about 740 ft above mean sea level and Chestnut Ridge to the northwest of the site stands at about 900 ft.

Stratigraphy

The Rome Formation and the Conasauga, Knox, and Chickamauga Groups constitute most of the bedrock of the Valley and Ridge Province in Tennessee. The Rome Formation, of Middle Cambrian age, consists mainly of red, green, and yellow shale, siltstone, sandstone, and minor gray dolomite, with a maximum exposed thickness of 1,200 ft. The Conasauga Group, of middle and Late Cambrian age, consists of about 2,000 ft mainly of alternating gray shale and limestone. The amount of limestone decreases northwestward where, at the province boundary, the Conasauga is nearly all shale. The 2,500-to 3,000-ft-thick Knox Group of Late Cambrian and Early Ordovician age is predominantly chert-bearing dolomite and lesser amounts of limestone. The Chickamauga Group of Middle and Late Ordovician age consists of alternating layers of gray and maroon limestone, calcareous siltstone, and shale. The thickness ranges from 8,000 ft in the southeastern part of the province to 2,000 ft in the northwest (PSAR, p. 2.5-4).

Calcareous mudstone and limestone of the Knox and Chickamauga Groups underlie the plant site. Typical strike and dip of the beds are N. 52° E. and 37° S.E. The bedrock is covered by a veneer of residual soil, through which scattered outcrops protrude in the central part of the site. The southern part of the site, and terrain near the river, are covered with alluvial soil. Weathered rock and soil attain a maximum thickness of 78 ft in the northeastern part of the site (PSAR, p. 2.5-15a). The applicant prepared a contour map of the top of "continuous rock," (unweathered rock which shows no significant discontinuities) based on 129 borings and seismic refraction work (PSAR, fig. 2.5-16).

Structure

During Paleozoic time, northwest-southeast compressional forces thrust rocks from the southeast over rock to the northwest. A succession of such thrust faults in the site area characteristically dip southeastward near the ground surface and flatten with depth (Harris and Milici, 1977, fig. 1, plate 5, 6). Swingle (1973, fig. 1) postulated a flat sole fault, which the thrust faults join, at a depth of about 9,000 ft. Harris and Bayer (1979, fig. 3) put the depth of the decollement at closer to 15,000 ft. This later work benefited from the COCORP seismic profiling (Cook and others, 1979). Rodgers (1970, p. 64) believes that the deformation and major structural features in the southern Appalachians were completed well before Late Triassic time.

The CRBRP site is located between two of these thrust faults--the Copper Creek and Whiteoak Mountain Faults. The Copper Creek fault at its closest point to the site is about 3,000 ft to the south. The strike and dip are N. 52° E. and 25° S.E. The site is near the midpoint of the 100-mile mapped length of the fault. The Rome Formation was thrust over rocks of the Chickamauga Group for a horizontal distance estimated in miles and a stratigraphic displacement of about 7,200 ft (PSAR, p. 2.5-21).

The Whiteoak Mountain Fault system consists of a main thrust fault with several subsidiary branch faults, the nearest trace being 1.7 miles northwest of the site. This northeasterly-trending fault is tens of miles long and is estimated to dip 45 to 50 degrees S.E. near the site (PSAR, p. 2.5-22).

Discussion

The general concept of the geology, presented by the applicant, is based upon a survey of the literature supplemented by drill core, radiometric dates, and geophysical work by them and their contractors. Although it is a simplistic presentation, we are in general agreement with the conclusions.

The items of major concern arising in our review of the PSAR were (1) the possibility of a limestone cavern underlying some portion of the site, because caverns are known to be present nearby, and (2) identification of active faulting, because seismicity is present in the province, although at a relatively low level.

Examination of the drill-core and the geologic cross-sections of the site drawn by the applicant, limitation of known caverns to the Knox Group (PSAR, p. 2.5-7), and the concept of "continuous rock" based on core-hole data and seismic refraction work, makes reasonable the applicant's contention that the presence of a major undetected cavity beneath a site structure is unlikely (PSAR, p. 2.5-15a).

Seismic events occur infrequently in the site area. The applicant states (PSAR, p. 2.5-25) without supporting data, that the "normal" focal depth for seismicity is 50,000 to 65,000 ft, well below the decollement and, therefore, unrelated to the shallow structure. Although data from the literature indicate that this is a reasonable hypothesis (for example, Bollinger and others, 1973), complete independence of seismicity and shallow structures has not been demonstrated, and the focal-depth range cited appears to be much too limited.

Recent thrust faulting in the Appalachians was the subject of a study by Schäfer (1979) (PSAR, p. 2F-3). His evidence for recent thrusting was offsets along subhorizontal fractures and bedding planes, of holes drilled during construction of roads. He noted such offsets, 12 years after roadway construction, at several locations, the closest to the Clinch River site being on Interstate Highway 40 between Harriman and Rockwood. Further study by Hatcher and Webb (1981) allowed them to conclude that offset is not a result of recent tectonism. Evidence of two kinds led them to this conclusion: (1) in multiple offsets amount of offset increases upward, and (2) offsets are not consistent in direction and favor a displacement direction toward the center of the highway. Stress relief, a factor noted in other studies in the Appalachians (Wyrick and Borchers, 1981) may be called upon as an explanation even for those offsets in directions parallel to directions of past thrust faulting.

Further, the applicant supports assignment of an ancient age for movement on the Copper Creek Fault with a radiometric age of mylonite from the fault zone of 285 million years (PSAR, p. 2.5-22). Although this is a reasonable date for major movement of the thrust faulting in the Valley and Ridge Province, such dating techniques do not preclude subsequent movement on the fault after erosion reduced the confining cover so that mylonite would not be formed. Evidence such as that from mapping and/or trenching of alluvial terraces across critical faults was not obtained. Such evidence could have demonstrated conclusively that the Copper Creek fault and Whiteoak Mountain fault are not capable.

Conclusion

In conclusion, although there has not been as definitive a demonstration as possible of noncapability for faulting in the area, the analysis of site geology by the applicant results in reasonable conclusions based upon current theories of Appalachian tectonics and upon the data available. It may be appropriate to note that to date no active faults have been recognized throughout the Appalachian region.

Seismology

Introduction

The U. S. Geological Survey has reviewed the seismological analysis in the Preliminary Safety Analysis Report (PSAR) for the Clinch River Breeder Reactor Plant site and compared it with the seismological literature for the region and with some results of on-going research in the Survey.

Applicant's safe-shutdown earthquake (SSE)

On seismicity maps prepared without specialized relocation techniques the Clinch River Breeder Reactor Plant (CRBRP) lies in the midst of a diffuse band of earthquake epicenters running roughly from Alabama to West Virginia. Because this band is spatially associated with the Appalachian mountains, it is natural, in the absence of more specific knowledge about the seismotectonics of the region, to consider that this seismicity is a feature of a so-called Southern Valley and Ridge seismotectonic province. The applicant has taken the largest historical earthquake in this province, the Giles County earthquake of 1897, and hypothesized a similiar eavent in the vicinity of the site. The applicant accepts an assessment of the maximum epicentral intensity of this earthquake as being a modified Mercalli intensity VII or VII+ (PSAR, p. 2.5-25). The safe-shutdown earthquake ground motion (SSE) is taken to be .25 g, corresponding to epicentral intensity VIII on a correlation of intensity with near-field strong motion acceleration (PSAR, p. 2.5-26). This intensity and the corresponding acceleration value are reasonable results of the application of Appendix A procedures to the regional seismic history. We point out that the recent analysis of Bollinger (1981) of the last 20 years of epicentral data in the Giles County area suggests the possibility of the existence of a structure capable of a maximum magnitude between $M_s = 6.0$ and $M_s = 7.0$ that might have an intensity greater than VIII.

The conservatism of the applicant's SSE

Appendix A to 10 CFR Part 100 defines a deterministic procedure the purpose of which is to arrive at an assessment of maximum ground motion at a site. In assessing the conservatism of this SSE, we have looked at the exceedance probability of .25 g, when considered in the light of the assumptions we have made in producing probabilistic ground motion maps for the eastern United States (Algermissen and others, 1982). Included in the assumptions for these maps were that the seismicity was diffuse and uniform over an Appalachian province and that the earthquakes were crustal earthquakes that could be modelled as point sources near the surface. For these assumptions, .25 g has an annual exceedance probability of 2×10^{-4} if statistical variability in the attenuation function is not taken into account, or 4×10^{-4} if the standard deviation of the attenuation variability is taken to be 0.6.

Maximum magnitude is important in the above results. If the maximum magnitude in the model is assumed to be as low as $M_s = 5.8$, the above exceedance probabilities are expected to decrease by a factor of 3.

A possible local seismic source

The above results depend upon a model of diffuse uniform seismicity in a broad Appalachian source zone. Hadley and Devine (1974) show the Appalachian seismicity to have a "hot spot" in eastern Tennessee. Much more recently, Dewey (personal communication) and Gordon (personal communication) have relocated a large number of instrumentally recorded eastern U.S. earthquakes. (A list of these earthquakes and their relocated coordinates have been sent to the NRC and the applicant's consultants). Nine of these relocated earthquakes can be seen to make up a zone 15 km wide and 180 km long, extending from about 34°57'N lat., 84°36'W long., to 36°25'N lat., 83°40'W long. A line connecting these points runs through Knoxville and forms an azimuth of nearly 20 degrees more northerly than the surface trend of the Appalachians. This may represent a concentration of seismicity in eastern Tennessee. Although there is insufficient evidence of a specific structure, it is possible that this alignment represents a basement seismic source zone or fault analogous to the proposed structure for the Giles County earthquake (Bollinger, 1981).

The conservatism of the SSE, assuming a local source

It might be asserted that the consequences of the existence of the hypothetical local source is already anticipated by the movement to the CRBRP site of a hypothetical earthquake of epicentral intensity VIII. The applicant acknowledges (CRBRP PSAR, amendment 71, page 2F-5) that Bollinger suggests that magnitudes up to $M_s = 7.0$ are possible on the hypothetical Giles County structure. The hypothetical structure considered for the vicinity of Knoxville is significantly longer than that proposed by Bollinger for the Giles County structure. Accordingly, there is some possibility of an earthquake on this hypothetical structure having an epicentral intensity greater than VIII. Because this structure is not proven, under Appendix A it would be inappropriate to bring an intensity greater than VIII to the site on the basis of this structure. However an assessment of the significance of such a structure is addressable through a probabilistic ground motion analysis.

As before, assessment of the conservative nature of the SSE depends upon a calculation of the exceedance probability of .25 g, given this hypothetical structure. Accordingly we assumed a line source, 140 km long, on which earthquakes were modelled as ruptures, with the rupture lengths depending on magnitude. (The 140 km length for the line source was chosen because it is the average of the different lengths obtained by removing zero, one, or two earthquake events from either end of the alignment). The maximum magnitude assumed was $M_s = 7.0$ and the b-value was 0.9. This source was assumed to lie at a radial distance of 15 km from the CRBRP. A major uncertainty in modelling this source is the determination of a suitable annual rate of seismicity attributable to this source. More precisely, the rate may be represented by the annual rate of earthquakes of magnitude greater than 4 (epicentral intensity equal to or greater than V) in the source. Two estimates for this rate were made. For the first estimate, one-quarter of the seismicity of the Appalachian seismic source zone (zone 100 from the national model of Algermissen and others, 1982) was considered attributable to the hypothetical line source. This fraction was an approximation resulting from inspection of seismicity maps, considering the contiguity of the seismicity to the hypothetical structure. For the second estimate, a list appearing in Bollinger and others (1976) for historical earthquakes occurring in the vicinity of the Maryville, Tennessee, earthquake of 1973 was used. All of these events were attributed to the hypothetical structure, and those with magnitude greater than 4, or intensity V or greater, were counted. The annual rates derived from the two procedures agreed within 15 percent of one another. The use of their average in the model yielded an annual exceedance probability for .25 g of 17×10^{-4} for no attenuation variability or 21×10^{-4} for a standard deviation of attenuation variability equal to 0.5. These results do not change significantly if the maximum magnitude is reduced to $M_s = 6.4$. Exceedance probabilities of these sizes may be legitimate cause for concern. However, if the maximum magnitude on this source is 5.8, the exceedance probability is 0 for no attenuation variability and 7×10^{-4} with attenuation variability.

In a more formal and complete probabilistic assessment of the exceedance probability of the CRBRF SSE, the following items would be found to be most important and would have to be treated probabilistically: (1) the seismic rate assigned to the fault, (2) the likelihood of the existence of this fault, and (3) the distance of the fault to the plant. The exceedance probability is directly proportional to factors (1) and (2) above and, over a limited distance range, inversely proportional to some power of (3). (If the distance is taken to be 20 km instead of 15 km, the exceedance probability decreases by about a factor of 1.6). It is, of course, factor (2) which is most in dispute. For the remainder of the review we address those facts relating to the credibility of the structure.

Evidence for and against the hypothetical structure

The seismological evidence which best addresses the hypothetical structure, other than the relocated epicenters themselves, is that information generated by investigations into the Maryville, Tennessee, earthquake of 1973. Maryville is on the apparent alignment of the relocated epicenters, and it is reasonable to expect that this earthquake should give evidence of a structure associated with the alignment, if indeed the alignment exists on a real structure. Most of the information about this earthquake appears in Bollinger and others (1976). The authors, however, believe that their evidence is not definitive enough to support any particular interpretation.

The map of epicenters of aftershocks of the Maryville earthquake shows a NNE trend. The main shock P-wave first motion focal-mechanism solution shows a NE-striking nodal plane, as does Herrmann's (1979) combined P-wave and surface-wave solution. However, the former solution is consistent with normal faulting down to the southeast, and the latter solution is consistent with reverse faulting on a northwest dipping plane. Both solutions have strike-slip motion component of the same sense. However, Bollinger and others (1976), on the basis of in situ stress measurements and well-hole data, prefer an alternative focal mechanism solution for the same P-wave arrivals. Their alternate main shock solution yields reverse faulting on a NW-trending fault plane. The composite mechanisms for the aftershocks give solutions inconsistent with the main shock mechanisms proposed by Bollinger and others (1976).

The major geophysical feature in the vicinity of the alignment is the "New York-Alabama lineament", which runs NE through Knoxville. This lineament has been interpreted (King and Zietz, 1978) as a fault juxtaposing different basement rock types along which strike-slip movement may have taken place. King and Zietz authors reject the interpretation of normal faulting along this line.

The alignment of epicenters does not coincide with the "New York-Alabama lineament" but rather has a more northerly strike. If the epicenters of the alignment are plotted on the Bouguer anomaly map of figure 2 of Keller and others (1982), the epicenters are found to lie not on the regions of strong gravity gradient, but rather on the tops or flanks of small gravity highs of length 40 to 60 km. The alignment, if it is as much as 180 km long must span three of these local highs. This may argue against a single structure and may limit the potential maximum magnitude.

Summary

The selection by the applicant of the Giles County earthquake in the Southern Valley and Ridge Province as the controlling earthquake at the site is reasonable. We also concur with the assessments of the maximum intensity and SSE, and the anchoring of a Regulatory Guide 1.60 response spectrum to this 0.25 g SSE. Furthermore, the CRBRP SSE has a conservative exceedance probability if one can confidently adopt a diffuse seismicity model to an Appalachian province. However there is evidence of a more concentrated local source in the vicinity of the

CRBRP. This source appears to have sufficient linear extent to generate large magnitude events. Furthermore the seismicity in the vicinity of this source, if attributable to a fault, is sufficient to imply that the CRBRP SSE has an exceedance probability notably higher than 1×10^{-4}. At the present time, the data are insufficient to establish the situation one way or the other. Accordingly, we believe that although the CRBRP SSE is reasonable on the basis of present data, a definitive seismological investigation would be required to address the problem of a possible concentrated seismic source in eastern Tennessee. This probably would require a local network, velocity models, and source mechanism determinations.

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NRC FORM 335
(7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
NUREG-0968
Volume 2

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
Safety Evaluation R[illegible] Related to the Construction of the Clinch River Breeder Reactor Plant
Volume 2-Appendices A-H Docket No. 50-537

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
Clinch River Breeder Reactor Program Office

5. DATE REPORT COMPLETED
MONTH March YEAR 1983

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Clinch River Breeder Reactor Program Office
Office of Nuclear Reactor Regulation
Washington, DC 20555

DATE REPORT ISSUED
MONTH March YEAR 1983

6. *(Leave blank)*

8. *(Leave blank)*

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.

13. TYPE OF REPORT
Safety Evaluation Report for a CP

PERIOD COVERED *(Inclusive dates)*
April 11, 1975 through March 1983

15. SUPPLEMENTARY NOTES

14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

The Safety Evaluation Report for the application by the United States Department of Energy, Tennessee Valley Authority, and the Project Management Corporation, as applicants and owners, for a license to construct the Clinch River Breeder Reactor Plant (Docket No. 50-537) has been prepared by the Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. The facility will be located on the Clinch River approximately 12 miles southwest of downtown Oak Ridge and 25 miles west of Knoxville, Tennessee. Subject to resolution of the items discussed in this report, the staff concludes that the construction permit requested by the applicants should be issued.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

Clinch River Breeder Reactor Plant
(CRBRP)
Safety Evaluation Report

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS *(This report)*
Unclassified

20. SECURITY CLASS *(This page)*

21. NO. OF PAGES

22. PRICE
$

NRC FORM 335 (7-77)

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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